Sixth ERDA Workshop on Personnel Neutron Dosimetry

July 11-12, 1977

Oak Ridge, Tennessee

Prepared for the
U.S. Department of Energy
under Contract EY-76-C-06-1830

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PACIFIC NORTHWEST LABORATORY
operated by
BATTELLE
for the
UNITED STATES DEPARTMENT OF ENERGY
Under Contract EY-76-C-06-1830

Printed in the United States of America
Available from
National Technical Information Service
United States Department of Commerce
5285 Port Royal Road
Springfield, Virginia 22151

Price: Printed Copy \$____*; Microfiche \$3.00

*Pages	NTIS Selling Price
001-025	\$4.50
026-050	\$5.00
051-075	\$5.50
076-100	\$6.00
101-125	\$6.50
126-150	\$7.00
151-175	\$7.75
176-200	\$8.50
201-225	\$8.75
226-250	\$9.00
251-275	\$10.00
276-300	\$10.25

SIXTH ERDA WORKSHOP ON PERSONNEL
NEUTRON DOSIMETRY

Meeting Coordinator:
Edward J. Vallario
U.S. Energy Research and Development Administration*

Meeting Program Co-Chairmen:
Dale E. Hankins
Lawrence Livermore Laboratory
and
Jack Hoy
Savannah River Laboratory

Scientific Secretary:
Philip E. Bramson
Battelle Pacific Northwest Laboratories
Richland, Washington

July 11-12, 1977
Oak Ridge, Tennessee

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* U.S. Department of Energy after October 1, 1977

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Participants

Foreign

Dr. G. Burger
Gesellschaft für Strahlen- und Umwelt-
forschung mbH.
Ingolstädter Landstr. 1
8042 Neuherberg bei München
Germany

Dr. J. David
Gesellschaft für Strahlen- und Umwelt-
forschung mbH.
Ingolstädter Landstr. 1
8042 Neuherberg bei München
Germany

Robert Facey
Central Health Physics Services
Ontario Hydro
Pickering G. S.
Box 160
Pickering, Ontario L1V 2R5
Canada

Franz N. Flakus
A-1011 Vienna
P.O. Box 645
Austria

Dr. John Harvey
Central Electricity Generating Board
Berkeley Nuclear Laboratories
Berkeley
Gloucestershire
England

Dr. Harry Ing
Atomic Energy of Canada, Ltd.
Chalk River, Ontario
Canada

Dr. Roger Medioni
Centre d'Etudes Nucleaires
B. P. No. 6
92260 Fontenay aux Roses
France

Dr. David W. O. Rogers
X-Rays and Nuclear Radiations
Division of Physics
National Research Council
Ottawa, Canada K1A 0R6

Meady Sohrabi
Atomic Energy Organization
of Iran
Radiation Protection Dept.
P.O. Box 41/2663
Tehran, Iran

Dr. L. Tommasino
Comitato Nazionale per
l'Energia Nucleare
Centro di Studi Nucleari della
Casaccia
Casella Postale 2400 - 00100
Roma A.D.
Italy

U.S.

Robert Alexander
Chief, Occupational Health
Standards Branch
Office of Standards Development
Nuclear Regulatory Commission
Washington, DC 20555

Edward Anderson
Mound Laboratory
Miamisburg, OH 45342

Dr. Bill Bartlett
Battelle Northwest
P.O. Box 999
Richland, WA 99352

Sy Block
Nuclear Regulatory Commission
Washington, DC 20545

Phil Bramson
Battelle Northwest
P.O. Box 999
Richland, WA 99352

Joseph Cortez
University of California
Los Alamos Scientific Laboratory
P.O. Box 1663
Los Alamos, NM 87545

Sue Crain
Mound Laboratory
Miamisburg, OH 45342

Jan Cusimano
Health Physics Laboratory
U.S. Energy Research and
Development Administration
P.O. Box 2108
Idaho Falls, ID 83401

Howard Dickson
Oak Ridge National Laboratory
P.O. Box X
Oak Ridge, TN 37830

Carl Distenfeld
Associated Universities, Inc.
Brookhaven National Laboratory
Upton, NY 11973

Dr. Elwyn Dolecek
Argonne National Laboratory
9700 S. Cass Ave.
Argonne, IL 60439

Bill Endres
Battelle Northwest
P.O. Box 999
Richland, WA 99352

Dick Gammage
Oak Ridge National Laboratory
P.O. Box X
Oak Ridge, TN 37830

Eric Geiger
Eberline Instrument Corporation
P.O. Box 2108
Sante Fe, NM 87501

Dick Griffith
Hazards Control Department
L-520
Lawrence Livermore Laboratory
Livermore, CA 94550

Ed Gupton
Oak Ridge National Laboratory
P.O. Box X
Oak Ridge, TN 37830

Ferenc Hajnal
Energy Research and Development
Administration
Health and Safety
Laboratory
376 Hudson Ave.
New York, NY 10014

Dale Hankins
L-383
Lawrence Livermore Laboratory
Livermore, CA 94550

Jack Hoy
E.I. DuPont de Nemours and Co.
Savannah River Plant
Aiken, SC 29801

John L. Lobdell, Supervisor
Radiation Surveillance and
Service Section
Tennessee Valley Authority
Muscle Shoals, AL 35660

Bill O'Neil
Division 3312
Sandia Laboratories
Albuquerque, NM 87115

Leigh F. Phillips
Brookhaven National Laboratory
Building 535, S&EP Division
Upton, NY 11973

William M. Quam
EG&G Inc.
130 Robin Hill Road
Goleta, CA 93017

Dr. Harald H. Rossi
Department of Radiology
Columbia University
Broadway and W. 116th
New York, NY 10027

Dr. Robert Schwartz
National Bureau of Standards
Washington, DC 20234

John Shubeak
National Bureau of Standards
Washington, DC 20234

Dr. Ellery Storm
University of California
Los Alamos Scientific Laboratory
P.O. Box 1663
Los Alamos, NM 87545

Dan Thompson
Division 3312
Sandia Laboratories
Albuquerque, NM 87115

E. J. Vallario
U.S. Energy Research and
Development Administration
Washington, DC 20545

Robert V. Wheeler
R. S. Landauer, Jr. and Co.
Glenwood Science Park
Glenwood, IL 60425

SIXTH ERDA WORKSHOP ON PERSONNEL NEUTRON DOSIMETRY

SUMMARY

This workshop was the sixth of a series and was held on July 11 and 12, 1977, at the Oak Ridge National Laboratory in Oak Ridge, Tennessee.

The first AEC Workshop on Personnel Neutron Dosimetry was held September 23-24, 1969, in Bethesda, Maryland and reported as BNWL-1340. The second Workshop was held July 8-9, 1971, in New York City and reported as BNWL-1616. The third Workshop was held March 16-17, 1972, in Savannah River Plant, Georgia, but no documented summary of the third Workshop was prepared. The fourth Workshop was held June 14-15, 1973, in Miami Beach, Florida and was reported as BNWL-1717. The fifth Workshop was held May 6-7, 1975, at the National Bureau of Standards, Washington, DC and reported as BNWL-1934.

Those presenting papers at the Sixth Workshop prepared summary reports of their recent work for inclusion in this document. The reports are reproduced here as submitted by the participants, with only minor editing.

This year's Workshop took a decidedly international flavor, with participants from seven countries in addition to the United States. The significance of this group's contributions has raised the possibility that the next Neutron Dosimetry Workshop may be held in Europe.

Of particular interest at the Workshop was the keynote address by Dr. Harald Rossi. He commented that there is evidence that 1) accepted values of RBE for low absorbed doses of neutrons may be low by an order of magnitude or more and 2) the risk of leukemia is significant at 0.5 rad to the bone marrow. A reduction of the limit for permissible neutron exposure, which could result from consideration of this information, would necessitate major improvements in our "middle ages" neutron dosimetry.

A number of participants reported conversions to thermoluminescent dosimeter (TLD) systems. This move has not been unanimous, however, as there were several reports of apparently satisfactory fission fragment, activation foil, and NTA film dosimeters. While the mention of NTA film resulted in the usual discussion of energy cutoff and humidity effects, it seems the use of NTA in accelerator environments still has some merit. Discussion of fission fragment dosimeters centered around track etching techniques, which have shown some improvement. Of particular interest was Tommasino's report on the use of polycarbonate centrifuge tubes as the sensitive element.

Thermally stimulated exoelectron emission (TSEE), never very popular for personnel dosimetry, has lost additional ground with the report that the neutron/gamma response ratio is much less than earlier work had indicated.

Initial work with an experimental Harshaw CaF:thulium TLD shows some promise. This TLD exhibits a double-humped glow peak (150°C and 240°C), which may permit separation of neutron and gamma components from a single TLD chip.

There was a plea for uniformity in neutron-to-dose conversion factors. A typical example of 120% differences between commonly used factors was cited. Perhaps it is time for a new look at the problem by an international standards-setting group.

Measurement of neutron spectra around facilities would be easier with a good, portable multichannel analyzer. Dr. Harry Ing may have one. It has 64 channels in a 6-in. x 8-in. x 2 1/2-in. package (including the display) and is battery powered.

A new intermediate energy neutron source is available at Berkeley Nuclear Laboratories (UK). An SbBe source in a water-filled boron ball was reported to produce a neutron spectrum centered around 1/2 keV. Additional efforts of this type coupled with the "monoenergetic" sources available at the National Bureau of Standards will permit significant improvement in neutron dosimeter calibrations.

KEYNOTE ADDRESS

A PROPOSED REVISION OF THE QUALITY FACTOR AND ITS
IMPLICATION TO PERSONNEL NEUTRON DOSIMETRY

Harald H. Rossi
Radiological Research Laboratory
Columbia University*
New York, NY 10027

When the relation between Q and LET_{∞} was formulated more than twenty years ago information on the relative biological effectiveness (RBE) of high LET radiation was limited. In the intervening period and especially in the last few years it has become apparent that the values of Q which were originally considered to contain factors of safety are in fact considerably less than the maximum RBE values for neutrons having energies of the order of 1 MeV. Values of RBE (relative to x or γ rays) in excess of 100 have been repeatedly observed. These occur only at small absorbed doses which are however the domain of concern to radiation protection.

Although these experimental findings as well as theoretical considerations strongly indicate the need for a new quality factor, q , that ranges over more than two decades, the adoption of q would not necessarily reduce the maximum absorbed dose of neutrons since the maximum dose equivalent might be raised as well.

However an analysis of data from Hiroshima and Nagasaki (1) indicates an appreciable hazard of absorbed neutron doses that are comparable to current permissible limits. Thus individuals receiving a mean absorbed dose of 0.5 rad to the bone marrow (the maximum permissible annual occupational absorbed dose) exhibited an excess incidence of leukemia that is roughly equal to the natural incidence.

A reduction in permissible levels of neutron radiation will add to the already difficult problems of personnel neutron dosimetry. The most promising improvement would seem to be the development of miniature pulse ionization chambers proposed by Colvett (2). Modern microelectronics should make

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*This investigation was supported by Contract EY-76-C-02-3243, from the Energy Research and Development Administration.

such devices quite feasible.

References

1. ROSSI, H.H. The Effects of Small Doses of Ionizing Radiation: Fundamental Biophysical Characteristics, Radiat. Res. 71, 1, (1977).
2. COLVETT, R.D. (1977) To be published.

I.

NEW DOSIMETERS

RECENT DEVELOPMENTS IN THE UNITED KINGDOM

J. R. Harvey
Central Electricity Generating Board
Berkeley Nuclear Laboratories
Berkeley, Gloucestershire, UK.

1. Personal Neutron Dosimetry

Within the Central Electricity Generating Board techniques of personal neutron dosimetry are essentially the same as those described at the last workshop by I.M.G. Thompson (1975). The boron-loaded albedo dosimeter (Harvey, Hudd and Townsend 1973) is utilized in the nuclear power stations and the readings corrected on the basis of instrumental survey of the neutron fields. This dosimeter was shown in an independent investigation to have favourable characteristics when compared with a number of alternative albedo systems (Douglas and Marshall), and will be used by the United Kingdom Atomic Energy Authority as part of a personal neutron dosimetry system in a plutonium-handling facility (Gibson 1977).

During the last year there has been extensive development of this latter system within the UKAEA. It comprises a fission track dosimeter based on neptunium 237, a low-pressure proportional counter, two CEGB albedo dosimeters and four criticality dosimeters. The devices are fixed to a harness which will be taken off when the worker leaves the radiation area.

The neptunium-based dosimeter described by Harrison (1977) is of particular interest. The fission foil contains 4 mg ^{237}Np adjacent to a 10 μm makrofol foil. The fission fragment tracks in the makrofol are etched by a 60 min immersion in 7.1N KOH at 60°C and the foils read out on an automatic spark counter based on the original idea of Cross and Tommasino (1970).

Dosimeters based on ^{237}Np are the closest to the ideal personal neutron dosimeter that can presently be achieved, since the fast fission cross section has a fairly low threshold (say 500 keV), and there is a finite fission cross section at intermediate and thermal energies. Although the response (in terms of dose equivalent) is very low between 500 eV and 500 keV, and "spiky" at intermediate energies, the albedo neutrons from the body produce

an energy response which is roughly proportional to dose equivalent when the dosimeter, on the body, is irradiated with most neutron spectra met with in practice.

The response of the neptunium dosimeter was measured with various neutron sources both in free air and on a phantom. The significant effect of the reflected neutrons can be seen from the results tabulated below.

Neutron Energy	Free air response /counts rem ⁻¹	On phantom response /counts rem ⁻¹	Ratio <u>On phantom</u> free air
Thermal	156 ± 13	(281)	(1.8)
~0.5 keV	313 ± 20	410 ± 100	1.31
23 keV	39 ± 5	183 ± 21	4.69
100 keV	(16)	50 ± 10	3.1
0.24MeV	19 ± 2	30 ± 3	1.58
0.55MeV	152 ± 6	193 ± 7	1.27
1.12MeV	223 ± 5	260 ± 6	1.17
1.72MeV	241 ± 5	279 ± 12	1.16
Cf 252	260 ± 6	266 ± 5	1.02
Pu-Be	293 ± 11	314 ± 11	1.07

The problems with this form of dosimeter are: (i) radiotoxicity - the dosimeter must be subject to careful control; (ii) cost - these dosimeters are much more expensive than other types; (iii) gamma dose from the neptunium to parts of the body near the dosimeter. This is reduced by lead shielding in the AERE dosimeter to 8 mrem/40h (skindose).

Another recent development in this field has taken place at British Nuclear Fuels Ltd., Windscale, where R. T. Brunskill (1977) has developed a two-component albedo dosimeter based on the two types of thermoluminescent lithium borate; one with normal abundance isotopes of lithium and boron and the other with ⁷Li and ¹¹B. The thermoluminescent materials are cemented onto graphite discs which are heated by an RF heater.

2. Novel Sources of Intermediate Energy Neutrons

At Berkeley Nuclear Laboratories some novel sources of intermediate energy neutrons have been identified and used. A source has been developed in which an Sb-Be source is located inside an 11 cm diameter water bath

surrounded by a 1 cm thick boron-loaded plastic shell (Harvey and Bending, 1976). This source, which emits a broad spectrum of neutrons centered on 1/2 keV, has been used to calibrate a range of instruments in a hitherto unexplored energy region (Harvey, Lavender and Thompson 1976, Harrison, Harvey and Boot, 1977).

Another study has been made of a possible facility for producing intermediate energy neutrons, based on a high flux reactor. A number of neutron filters have been identified which could be located in the beam tube of the reactor to give a range of roughly mono-energetic neutron beams within the range 60 eV-160 keV. The lower neutron energies cannot be produced by other means. A number of applications in the fields of health physics, radiobiology, radiotherapy and nuclear physics have been identified (Harvey and Mill 1977).

References

1. Brunskill, R.T., 1977. Health Physics, 32, 455.
2. Cross, W.G. and Tommasino, L., 1970, Radiation Effects, 5, 85.
3. Douglas, J.A. and Marshall, M., to be published.
4. Gibson, J.A.B., 1977, IRPA Symposium, Paris, April 1977. Paper 306.
5. Harrison, K.G., 1977. Private Communication.
6. Harrison, K.G., Harvey, J.R., Boot, S.J., 1977. Submitted to Nuclear Instruments and Methods.
7. Harvey, J.R., Bending, R.C., 1976. Phys. Med. Biol., 21, 1, 85-97.
8. Harvey, J.R., Hudd, W.H.R., Townsend, S., 1973. "Neutron Monitoring for Radiation Protection Purposes." Symposium, Vienna 1972, p.199-218. STI/PUB/318.
9. Harvey, J.R., Lavender, A., Thompson, I.M.G., 1976, Health Physics, 31, 363-369.
10. Harvey, J.R., Mill, A.J., 1977. "A proposed intermediate energy neutron facility and possible applications." Proceedings of the third symposium on neutron dosimetry in biology and medicine. 23-27 May 1977, Neuherberg, Munich, Federal Republic of Germany. To be published by the Commission of the European Communities.

11. Thompson, I.M.G. Fifth ERDA workshop on personnel neutron dosimetry, page 68. BNWL-1934-UC-48. 1975.

Acknowledgment

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NEUTRON DOSIMETRY SYSTEM

W. M. Quam
EG&G, Inc.
P. O. Box 98
Goleta, CA 93017

The neutron dosimetry system described in this paper consists of four activation foils, Au, Au + Cd, As and S. The activity data from the exposed foils are used as input to a spectrum unfolding code, CRYSTAL BALL, which calculates a differential neutron spectrum. The spectral data are calculated at 621 points from 10^{-9} to 2×10^1 MeV. This 621 point spectrum is then converted to Element 57 dose, point by point, using conversion factors interpolated from Auxier, Snyder and Jones*. The REM exposure is calculated from the rads-tissue matrix by using an analytical version** of the 1973 ICRP recommendations.

The dose data derived from this system agree with that provided by the DOSAR group for both lucite shielded and bare reactor exposures within 3 to 11%.

The spectral data for a free in air exposure and for a front of phantom location are given. In addition to these data, the tissue-rad exposures, the REM dose, the spectrum averaged quality factor, the average neutron energy and integral fluence data above various thresholds are given for several HPRR exposure conditions.

The dosimetry system is non-radioactive, low cost (~\$10 per foil set) and relatively accurate. The 3σ limit of detectability is ~15 tissue-rads at 8 days after exposure with a 2x2 NaI counter.

The signal to noise ratio (or $S/\sqrt{B}$) at eight days after exposure to 48 tissue-rads is ~10.7. In practice a value of 3 to 5 would suffice.

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* In "Radiation Dosimetry," Vol. 1, p. 275 ff, Attix, Roesch, and Tochlin, (1968) Academic Press.

** A. Rindi, Health Phys., 27, 322 (1974).

The foils can be evaluated with NBS traceable equipment which does not need a neutron exposure for its calibration. Counting and complete evaluation of 4 sets of foils can be completed in about 20 hours. Computer costs at the highest priority are about \$20 per set of foils.

CHARACTERISTICS OF SANDIA'S PERSONNEL NEUTRON DOSIMETER

Daniel J. Thompson
Sandia Laboratories
P. O. Box 5800
Albuquerque, New Mexico 87115

SUMMARY

Sandia's personnel neutron dosimeter is an albedo type dosimeter that must be worn on the body to work properly. The dosimeter is evaluated on a Harshaw automated 2271 TLD system. Each dosimeter contains two TLD cards. Each card consists of one LiF TLD-600 and one LiF TLD-700. With these 4 TLD chips we perform non-penetrating, gamma and neutron dosimetry.

The dosimeter holder consists of an open window, aluminum filters, and a borated polyethylene filter. The polyethylene filter is shaped like an open ended box half the height of the TLD card. The TLD cards are arranged such that one TLD-600 is bare and one TLD-600 is in the borated jacket.

The use of the TLD-600 "Bare/Boron" ratio allows one to identify neutrons whose energies are below 1 eV. As the neutron energy decreases below 1 eV the "Bare/Boron" response ratio increases from approximately 5 to 40. The dosimeter response in nanocoulombs/rem, to neutrons between 1 eV and 10 keV, is relatively energy independent - decreasing by only a factor of 2 over this range. For neutrons over 10 keV the neutron sensitivity drops rapidly. For dosimeter evaluation it is necessary to know the source of the neutrons and if possible to calibrate to that source.

The change in the "Bare/Boron" ratio can also be useful to detect changes in expected neutron spectrums. For example, the "Bare/Boron" ratio ranges between 4 and 5 for neutron energies above 1 eV; if there is an increase in this ratio it could mean a change in how the source is being used.

This dosimeter has a greater sensitivity to neutrons below 500 keV than to gamma radiation. For neutrons over 500 keV the detectability is reduced further in mixed gamma and neutron exposures. For example, it is possible to estimate a .05 rem Cf-252 neutron dose within 50% when 1.00 rem of Co-60

gamma is added; however, to achieve the same accuracy for a .05 rem PuBe neutron dose, only .05 rem of Co-60 can be added.

In routine use an energy code, corresponding to the source of neutrons or previous calibration, is assigned to each dosimeter user. A computer code has been written that allows the computer to evaluate each dosimeter and assign the neutron dose if (1) certain conditions are met, and (2) the dose is less than 100 mrem. If the dose is greater than 100 mrem or if the conditions are not met, the dosimeter data is returned on a report for special attention. Using this method the computer completes about 95% of the neutron calculations.

Sandia's Beta, Gamma and Neutron TLD Badge

Personnel Dosimeter Response to ^{252}Cf vs Added Lucite Moderator

ADDED MODERATOR	REM DOSE	NEUTRON RESPONSE (NANOCOULOMBS)		SPECIFIC RESPONSE (NANOCOULOMBS/REM)		RATIO BARE / ^{10}B	NORMALIZED RESPONSE	
		BARE	^{10}B	BARE	^{10}B		BARE	^{10}B
0	9.70	840	103	86.6	10.6	8.2	1.00	1.00
1/4"	9.38	1138	121	121.3	12.9	9.4	1.40	1.22
1/2"	8.91	1219	126	136.8	14.1	9.7	1.58	1.33
1"	8.01	1450	145	181.0	18.1	10.0	2.09	1.71
2"	5.96	1845	152	309.6	25.5	12.1	3.58	2.41

A COMBINATION TL-FILM PERSONNEL NEUTRON DOSIMETER

M. E. Anderson and S. L. Crain
Mound Laboratory*
Miamisburg, Ohio 45342

Introduction

Personnel neutron exposures at Mound Laboratory are principally from $^{238}\text{PuO}_2$ with additional exposures from isotopic sources such as $^{239}\text{Pu-Be}(\alpha, n)$ and ^{252}Cf . In a few areas the sources of neutrons are essentially unshielded, while in others the shielding ranges from light to heavy. Some personnel may regularly be exposed to neutrons of generally the same energy spectrum, while others may be exposed to a variety of spectra. Until recently, NTA film was used for all neutron monitoring with a single calibration factor serving the entire laboratory. Because of the wide variety of spectra, a single calibration factor was not altogether satisfactory. In order to provide better estimates of neutron exposures, a program was initiated to upgrade the neutron dosimetry system including, probably, a different personnel dosimeter. Because of the time lag until such a system would become operational, it was decided to modify the present system. There seemed to be two choices: 1) to continue to use just film and assign calibration factors on the basis of administrative control, or 2) to modify the dosimeter in such a manner as to determine the calibration factors from the dosimeter readings themselves. The latter method was chosen and was accomplished by the incorporation of a $^6\text{LiF(TL)}$ chip in the film badge holder.

Description and Calibration of the Dosimeter

Incorporation of TL chips into the film badge holder was fairly easy, for the holder already had a suitable cavity with 0.005" (0.013 cm) cadmium and 0.017" (0.043 cm) tantalum on each side. In addition to a ^6LiF chip, a ^7LiF chip was added at the same location in order to correct the ^6LiF reading

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*Mound Laboratory is operated by Monsanto Research Corporation for the U. S. Energy Research and Development Administration under Contract No. EY-76-C-04-0053.

for photon response. (This ^7LiF chip is in addition to other chips already in the holder for photon dosimetry.)

Calibration was accomplished by exposing dosimeters mounted on a 16" X 16" x 4" (41 cm x 41 cm x 10 cm) polyethylene phantom to various sources of neutrons. Six dosimeters were used for most of the exposures with three being flush with the phantom and three positioned 3 cm before it. The data obtained indicated no significant differences between these positions, therefore, the results were averaged.

Three exposures were made with unmoderated isotopic sources in a low scattering area. Sources used were: $^{238}\text{Pu}-^{18}\text{O}(\alpha, n)$, average neutron energy ($\bar{E}_n$) = 2.1 MeV [1]; $^{239}\text{Pu}-\text{Be}(\alpha, n)$, $\bar{E}_n$ = 3.9 MeV [2]; and $^{238}\text{Pu}-\text{F}(\alpha, n)$, $\bar{E}_n$ = 1.3 MeV [1]. Five exposures were made in glovebox operating areas where room scattering was much greater. Four additional exposures were made in the low scattering area with the Pu-0 source placed successively at the centers of 5" (12.7 cm), 8" (20.3 cm), 10" (25.4 cm), and 12" (30.5 cm) diameter polyethylene spheres. Dose equivalents for all exposures were obtained by using a 10" Bonner sphere with an LiF scintillator as the detector. This instrument was calibrated with the Pu-0 source using a fluence-to-dose equivalent conversion factor of 3.4×10^{-5} mrem/n-cm $^{-2}$. Exposure times varied from 16 to 88 hours. TL readings were made at least 24 hours following the completion of the exposure to allow time for some low temperature peaks to decay. A Harshaw model 2000-A reader with a programmed heating cycle was used for reading the TL chips. The NTA films, used with no special packaging, were developed within six hours of the completion of each exposure. Small corrections ($\leq 6\%$ in each case) were made to account for fading which may have occurred during the exposures.

A ratio was obtained using counting rates from the 10" sphere and from a cadmium-covered 3" sphere for each exposure. These ratios give indications of the average energies of the neutron spectra for these exposures, with higher energies, in general, giving higher 10/3 ratios. Figure 1A is a plot with the TL responses per mrem (TL/H) as ordinates and the 10/3 ratios as abscissas. Figure 1B is the same for film, with the film response (F/H)

given as tracks/ $1.8 \text{ mm}^2\text{-mrem}$. The figures show the decrease in TL response and the increase in film response as neutron energy increases. The ratio between the high and low TL responses is 9, while that for the film responses is 14 for these data. A comparison of these two graphs reveals a bigger spread for the film responses than for the TL responses for the exposures in operating areas. The relative uncertainty in the ordinate for each datum is $\leq 10\%$. For TL this is based on the variance of the responses, and for film it is based on the total number of tracks counted. To determine absolute errors, one would need to take into account the uncertainty introduced by using the 10" sphere for the dose equivalent measurements. Some of the spread in the data for the operating area exposures, where room scattering is significant, may have been introduced by the phantom. The phantom will partially shield the dosimeters from neutrons entering from the back, but these neutrons will still appear in the dose equivalent determinations since no phantom was used with the 10" sphere measurements.

For unmoderated Pu-0 and Pu-Be spectra, at least, the 10/3 ratio is a poor indicator of average energy. The Pu-0 spectrum has an average energy about half that of the Pu-Be spectrum and yet the two have almost the same 10/3 ratios. To try to compensate for this bit of untidiness, the TL and film responses have been plotted in Figure 1C against average energies for the unmoderated isotopic sources.

In Figure 2 the TL and film responses have been plotted against each other. One additional datum that has been added is the response for the unshielded Pu-0 source after room scattering has been removed. Each datum in Figure 2 was obtained with a single neutron source or at a single location. If, for example, a dosimeter were exposed to two different neutron spectra, the resultant datum would lie on a straight line connecting the points representing those spectra. Therefore, a straight line was chosen for a calibration curve. Because most of the exposures being monitored by this dosimeter were for neutrons from PuO_2 , it was decided not to use the Pu-Be and Pu-F data for the calibration curve. Also, the data for the Pu-0 source in moderating spheres were not used because these results did not represent actual operating conditions. Using, then, only the data from

operating areas and for the unshielded Pu-0 source, an unweighted least squares fit was made assuming both abscissas and ordinates to be in error. The line in Figure 2 represents this fit and its equation is

$$I = 0.39 (TL/H) + 4.4 (F/H) \quad (1)$$

where (TL/H) and (F/H) are as previously defined. A dose equivalent determination from an actual exposure with responses TL and F would be obtained from

$$H = 0.39 (TL) + 4.4 (F) \quad (2)$$

To indicate the potential error due to the use of this calibration curve, an exposure corresponding to the 10" and 12" spheres would be overestimated by 50 and 60%, respectively. An exposure corresponding to the Be datum would be overestimated by 40%, while one corresponding to the Pu-F datum would be underestimated by 30%. All of the other points are within $\pm 20\%$ of the curve.

For these exposures there was one additional ^6LiF chip in each film badge holder. It was located in an open window with a ^7LiF chip used for photon dosimetry. The response of this open window ^6LiF chip was from 2.5 to 3 times that for the chip located between cadmium and tantalum. However, the spread in responses for the open window chip for exposures made in operating areas was about 20% greater than for the cadmium-filtered chip. Because of the smaller spread in the responses for the cadmium-filtered chip and because the extra sensitivity did not appear to be needed, the cadmium-filtered chip was used for the TL portion of this dosimeter.

Summary

This combination TL-Film dosimeter seems to be functioning satisfactorily. Differences in neutron spectra are being accounted for to some extent by the combination of the TL and film responses. The dosimeter was not investigated as thoroughly as it might have been, had it been designated for long-term use. In particular, it would have been instructive to move the spheres containing

the Pu-0 source into areas where room scattering is greater. Also, no special packaging was used with the film to reduce fading. Some fading studies done this spring at Mound Laboratory indicated that periods for 50% fading vary from 16 to 28 days. Dosimeters are generally worn for 2 or 4 weeks; fading, therefore, presents a potential error comparable to that due to the calibration curve. Use of improved film packaging techniques could probably improve this factor [3]. Also, if this dosimeter were to be used extensively, additional work should be done on fading as a function of neutron energy spectrum. Although we used a cadmium-filtered ^6LiF chip for these data, the results would have been only slightly less satisfactory had we used the open window chip.

References

1. R. A. Neff, M. E. Anderson, A. R. Campbell, and F. X. Haas, "Some Neutron and Gamma Radiation Characteristics of Plutonium Cermet Fuel for Isotopic Power Sources," NASA TM. X-2440 (1972).
2. M. E. Anderson and W. H. Bond, Jr., "Neutron Spectrum of a Plutonium-Beryllium Source," Nucl. Phys. 43, 330 (1963).
3. L. F. Philips, "NTA Proton Recoil Track Fading in Personnel Neutron Dosimetry," BNWL-1934 (1975).

FIGURE 1a. TL Response Versus 10/3 Counting Ratio

FIGURE 1b. Film Response Versus 10/3 Counting Ratio

FIGURE 1c. Film and TL Response Versus Average Neutron Energy

FIGURE 2. TL Response Versus Film Response

A TLD SYSTEM FOR PERSONNEL MONITORING

John L. Lobdell
Radiation Surveillance and Service Section
Tennessee Valley Authority
Muscle Shoals, AL 35660

In January, 1977, the Tennessee Valley Authority converted from film to TLD's for personnel monitoring. The Harshaw 2271 automated system was purchased.

Figure 1 shows the arrangement of filters and cards for detecting exposure to beta, gamma, and neutron radiation. The badge is seen from the top. The ^7Li chips on the beta-gamma card, numbered 1 and 2 for differentiation, are in the bottom row. The second ^7Li chip is shielded with a cadmium filter, where the first ^7Li chip is not shielded. Both ^6Li chips are shielded with metal filters; the first ^6Li chip is positioned behind a tin filter, and the second is positioned behind the same cadmium filter that shields the second ^7Li chip. This means the three shielded chips will detect no beta or X radiation and are essentially reporting equivalent doses of gamma radiation. The thicknesses of the two metal filters are chosen to present the same amount of matter, in milligrams per square centimeter, to incident gamma radiation. The cadmium filter over the second ^6Li chip will stop all incident thermal neutrons, but the tin filter over the first ^6Li chip will stop no incident thermals at all; thus, the first ^6Li chip will detect incident thermals and the second will not.

The neutron card will detect neutrons of all energies if the badge is placed in front of a large thermalizing medium such as the human body or a water jug. A hydrogenous substance is needed to thermalize the intermediate and fast neutrons and then reflect them from the medium to be detected by the ^6Li chips which are sensitive to thermal neutrons. The body will reflect part of the incident neutrons--fast, intermediate, and thermal--as thermal. It is imperative that individuals exposed to neutrons wear the badge as close to the body as possible to maximize reflection to the badge.

The grossly simplified equations at the bottom of the figure detail the process we use to measure dose from all radiations. First, the second ^7Li

chip is insensitive to or shielded from all but gamma radiations; therefore, this chip measures gamma exposure. The first ^7Li chip is insensitive to neutrons and measures exposures due to beta, X, and gamma radiations. The difference between the responses of the two ^7Li chips gives the amount of exposure to beta or X radiations. If we do not know which of these two radiations the person was exposed to, we apply the more conservative beta calibration.

The second ^7Li chip and both ^6Li chips are gamma equivalent. Therefore, when the response of the second ^7Li chip is subtracted from that of the ^6Li chips, any remainder is due to neutrons. When the response of the second ^6Li chip is subtracted from the response of the first ^6Li chip, any residual is due to thermal neutrons. Because thermal neutrons can be reflected off the body to expose both ^6Li chips, a certain fraction of the thermal response determined previously must be subtracted from the response of both ^6Li chips. Any remainder is due to intermediate or fast neutrons.

If the badge arrangement is calibrated for neutron spectra similar in energy distribution to the neutron spectra to which personnel will be exposed, a dose can be determined fairly accurately. However, if the neutron spectra are unknown, significant error can result. The dose determinations for beta, X and gamma radiations should be accurate.

Worker's body

$$\begin{aligned} \text{gamma} &= {}^7_2\text{Li} \\ \text{beta} &= {}^7_1\text{Li} - {}^7_2\text{Li} \\ {}^1_0n_{\text{th}} &= {}^6_1\text{Li} - {}^7_2\text{Li} \\ {}^1_0n_{\text{intermediate}} &= {}^6_2\text{Li} - {}^7_2\text{Li} - f {}^1_0n_{\text{th}} \\ &\text{and fast} \end{aligned}$$

FIGURE 1.

A PERSONNEL ALBEDO MONITOR FOR NEUTRONS

G. Burger and A. Morhart
Gesellschaft für Strahlen-
und Umweltforschung mbH München
Institut für Strahlenschutz
8042 Neuherberg,
Ingolstädter Landstr.1
Germany

Abstract

This work deals with the study, design and test of a personnel neutron monitor based on the detection of the neutron albedo fluence emerging from the human body when it is exposed to an incident fast neutron fluence. The proposed system provides sufficient sensitivity to incident neutrons of energies below 100 keV. It consists of a pair of TLD-ribbons (^6LiF - ^7LiF) mounted inside a polyethylene body, in order to increase the overall sensitivity, and it is shielded by a borated resin layer to decrease the high sensitivity below 100 eV. The response of the dosimeter to mono-energetic neutrons from a 3 MeV Van de Graaff and its directional sensitivity were investigated. Phantom neutron penetration calculations were performed by means of the SN-code DOT-2 to study also the theoretical response of an albedo-system. This system, together with a ^{237}Np fission track detector and a sulphur pellet, provides adequate information to assess the maximum body dose equivalent originating from neutrons in the energy range $E < 15$ MeV, for routine and accidental monitoring provided that spectrum dependent correction factors are applied.

The same holds in principle, however with less accuracy, for the use of the albedo dosimeter alone.

1. Introduction

A dosimetric system incorporating an albedo dosimeter, a fission track detector (neptunium) and a threshold activation detector (sulphur) was designed by the GSF Munich-Neuherberg in collaboration with the IRD, Rio de Janeiro (1). The goal was to have a simple system which can be used partly or as a whole for routine neutron personnel monitoring as well as for operational or accidental dosimetry. The combination of the albedo and the

neptunium dosimeter allows one to achieve a certain approximation to the recommended dose equivalent-to-fluence response function, at least below 100 keV and above 500 keV. Besides this, the adding of the $^{32}\text{S}(n,p)$ ^{32}P -reaction gives a better evaluation of the Maxwellian fast spectrum, and consequently a more accurate determination of the absorbed dose, in case of accidents. This paper limits itself only to the discussion of the albedo detector.

2. Description of the System

For personnel monitoring purposes TLD detectors are used to measure the albedo neutron flux at the surface of the body, by means of the reaction $^6\text{Li}(n,\alpha)^3\text{H}$. The use of pairs of ^6LiF and ^7LiF allows the subtraction of the γ -contribution in a mixed n/γ -field.

Shape, size and composition of the dosimetric system may cause substantial modifications in the original albedo response (2). For this reason, the influence of some of these parameters was investigated. A simple version of the albedo dosimeter, consisting of only a pair of ^6LiF - ^7LiF placed at a phantom surface, would be extremely sensitive to thermal neutrons, which is not desirable for monitoring purposes. In order to improve this situation, an adequate shield layer must be added (3). Some materials and dimensions were checked. Cadmium, due to its cut-off, causes a deformation in the response curve by suppressing the sensitivity of the system to incident thermal neutrons. A material with a $1/v$ -cross section, like boron, is more appropriate.

As additional shielding attenuates the response of the system, a polyethylene moderating body was added in order to improve the sensitivity, following essentially the design of Hoy (4) and the results of Hankins (2). A prototype of the above described dosimeter is shown in Figure 1. It consists of a polyethylene hemisphere of 32 mm diameter with a cylindrical extension of 10 mm, with the external surface being covered by a 1 mm layer of a polymer resin containing 0.75 g/cm^3 of natural boron. The TLD detectors used are Harshaw ribbons TLD-600 and TLD-700.

FIGURE 1. Prototype of the Albedo Dosimeter

3. Experimental Procedure

To determine the response curve of this prototype and to evaluate the influence of some geometrical parameters, irradiations were performed with monoenergetic neutrons from a 3 MeV Van de Graaff accelerator, at the low scattering facility of the "Institut für Strahlenschutz" of GSF (München).

The gamma calibration of the TL detectors was performed by means of a ^{60}Co source. The phantoms used were laying cylinders (40 cm in diameter and 20 cm thickness) made from PVC and polyethylene, filled with a solution similar to soft tissue in density and neutron cross-section equivalence (65.6% water, 26.8% glycerin, 7.6% urea). The relative neutron fluence was determined by using a precision De Pangher Long Counter (5); the absolute source strength at 15 MeV was determined by the accompanied particle technique.

The dosimeters were fixed at the phantom surface. Four dosimeters were used in frontal, side-on and back position.

4. Results and Discussions

The experiment performed with the PVC showed a somewhat lower sensitivity than with the polyethylene phantom. This may be due to the high neutron absorption cross section of chlorine in PVC.

The energy response was experimentally determined between 100 keV and 15 MeV. The theoretical response curve calculated by Alsmiller (6) for a similar dosimeter designed by Hoy (4) was normalized to our experimental data at an energy of 1.2 MeV. Our calculations have been performed by use of the computer code DOT-2. The uncollided flux and the first collision source distribution inside the phantom were calculated by use of an extended version of the code UNCL (7). As DOT is only applicable to geometries which can be described by two spatial coordinates, the phantom has to be chosen as a laying cylinder, the axis of which is parallel to the beam axis.

The source for the cross section was DLC-9 (7). The condensation from 122 groups to 37 groups was done with the code ANISN (7). In a first step the existence of a true dosimeter setup was ignored and the calculations were performed for the pure phantom (8). Table 1 shows the results for the thermal neutron fluence normalized to the axial incoming neutron fluence for 15 MeV neutrons.

Figure 2 shows the results of Alsmiller, some first results of our own calculations and the experimental points. A good agreement between the calculations of Alsmiller and the experiment was obtained.

The directional response is shown in Figure 3. The average response was obtained from these three measurements, simulating the real situation of a monitored individual moving around randomly in a radiation field.

Finally, in Figure 4, the results of an intercomparison of albedo dosimeters performed in Harwell (9) show the neutron sensitivities of different prototypes, demonstrating that the energy response of each albedo dosimeter above 100 keV is identical, independent of the detailed construction and evaluation principles of the device. The use of a polyethylene body, however, increases considerably the overall neutron sensitivity; this may be crucial for the accuracy and precision of neutron dose determination

TABLE 1. Thermal Neutron Fluence Per Unit Incoming Neutron Fluence
In and Outside a Water Phantom for 15 MeV Neutrons

R	Z 1	Z 2	Z 3	Z 4	Z 5	Z 6	Z 7
1	2.48338E-02	2.75223E-02	3.28778E-02	3.31047E-02	1.92781E-01	3.97377E-01	4.71256E-01
2	2.45886E-02	2.73809E-02	3.23290E-02	3.24783E-02	1.88933E-01	3.89501E-01	4.62325E-01
3	2.32189E-02	2.73214E-02	3.21560E-02	3.25502E-02	1.88627E-01	3.88760E-01	4.60799E-01
4	1.85488E-02	2.31228E-02	2.75840E-02	2.90808E-02	1.74989E-01	3.59773E-01	4.23989E-01
5	1.10493E-02	1.29343E-02	1.44204E-02	1.46241E-02	8.74403E-02	1.78081E-01	2.09011E-01
6	6.78273E-03	6.40152E-03	5.12071E-03	3.88469E-03	9.53340E-03	1.83885E-02	2.19538E-02
7	6.06898E-03	5.62488E-03	4.47329E-03	4.21921E-03	9.20925E-03	1.66782E-02	2.02649E-02
8	5.02288E-03	5.02068E-03	4.54140E-03	4.71641E-03	8.16060E-03	1.31659E-02	1.61201E-02
9	3.99594E-03	4.74652E-03	4.82721E-03	5.28995E-03	7.28667E-03	1.01912E-02	1.20634E-02
R	Z 8	Z 9	Z 10	Z 11	Z 12	Z 13	Z 14
1	5.09196E-01	5.27533E-01	5.35383E-01	5.17235E-01	4.51984E-01	2.34101E-01	3.80303E-02
2	5.00141E-01	5.18594E-01	5.26585E-01	5.08737E-01	4.44596E-01	2.30513E-01	3.74325E-02
3	4.97665E-01	5.15856E-01	5.24553E-01	5.08013E-01	4.44763E-01	2.30647E-01	3.75937E-02
4	4.55304E-01	4.71057E-01	4.80478E-01	4.68358E-01	4.12771E-01	2.15329E-01	3.37014E-02
5	2.23917E-01	2.31978E-01	2.38018E-01	2.34410E-01	2.09822E-01	1.15923E-01	1.70896E-02
6	2.35273E-02	2.44508E-02	2.46967E-02	2.39836E-02	2.06519E-02	1.09638E-02	4.64224E-03
7	2.16038E-02	2.22831E-02	2.25726E-02	2.20246E-02	1.86155E-02	1.05251E-02	4.97609E-03
8	1.76562E-02	1.81667E-02	1.83814E-02	1.73589E-02	1.45512E-02	9.20166E-03	5.43967E-03
9	1.34992E-02	1.40573E-02	1.36516E-02	1.27218E-02	1.09611E-02	7.99644E-03	5.93050E-03
R	Z 15	Z 16	Z 17	Z 18	Z 19	Z 20	Z 21
1	3.78161E-02	3.19758E-02	2.91390E-02	1.48788E-02	5.40721E-03	2.41621E-03	1.09577E-01
2	3.72730E-02	3.18564E-02	2.88668E-02	1.52390E-02	5.65286E-03	2.84869E-03	1.59813E-02
3	3.71674E-02	3.18359E-02	2.72950E-02	1.45423E-02	5.41560E-03	2.72454E-03	5.87352E-03
4	3.20225E-02	2.70144E-02	2.18658E-02	1.26514E-02	4.88405E-03	2.17906E-03	2.80453E-03
5	1.68461E-02	1.51835E-02	1.31691E-02	8.81125E-03	4.05642E-03	1.84332E-03	1.89167E-03
6	6.03506E-03	7.55788E-03	8.13285E-03	6.59477E-03	3.70377E-03	1.74120E-03	1.66358E-03
7	5.24137E-03	6.62635E-03	7.28335E-03	6.11612E-03	3.62336E-03	1.72842E-03	1.61652E-03
8	5.23963E-03	5.83070E-03	5.98097E-03	5.31062E-03	3.32965E-03	1.74867E-03	1.56386E-03
9	5.45387E-03	5.39482E-03	4.68909E-03	4.18284E-03	2.83299E-03	1.64829E-03	1.53913E-03

Z	R (cm)	Z (cm)
1	1,5	25
2	4,0	22,5
3	7,0	21,0
4	12,0	20,4
5	17,5	18,5
6	20,4	16,0
7	21,0	14,0
8	22,5	12,0
9	25,0	10,8
10		8,0
11		6,0
12		4,0
13		1,5
14		- 0,4
15		- 1,0
16		- 2,5
17		- 5,0
18		-13,1
19		-35,0
20		-68,1
21		-100,0
22		

Radial and axial depth
zones and their midpoint
distances in cm.

Irradiation geometry and mesh
pattern for the calculations
(1=air, 2=water)

FIGURE 2. Response Curve of Albedo Dosimeter Per Unit Fluence

FIGURE 3. Directional Response of the Phantom-Dosimeter System

FIGURE 4. Dose Equivalent Response of Different Albedo Dosimeters (Results of an Intercomparison at Harwell)

in a mixed field, since that determination is based on the difference in the neutron- and gamma-sensitivity of the ^6Li - ^7Li detector pair.

Acknowledgments

The measurements at GSF were performed by Wilson R. D. Carvalho, the neutron fields monitored by Dr. H. Schraube. We gratefully acknowledge their help and that of the GSF accelerator team.

References

1. Carvalho, W. B. D., G. Burger and D. C. C. Reis - "Personnel Monitor for Neutrons Based on the Albedo Technique" - Proceeding of the Vth Int. Conf. Lum. Dosimetry, 14.-17.2.77, Sao Paulo.
2. Hankins, D. E. - "Factors Affecting the Design of Albedo Neutron Dosimeters Containing Lithium Fluoride Thermoluminescent Dosimeters" - Report LA-4832. Los Alamos Scientific Laboratory (1972).
3. Brown, J. B. C., F. M. Gayton, J. A. Hall, J. R. Harvey, G. A. M. Webb - "Recommendations Concerning the Use of the Personnel Neutron Albedo Dosimeter" - Berkeley Report RD/B/R-828 (1967).

4. Hoy, J. E. - "Personnel Albedo Neutron Dosimeter With Thermoluminescent ^6LiF and ^7LiF " - Report DP-1277. E. I. du Pont de Nemours, Savannah River Plant (1972).
5. Schraube, H., F. Grünauer, G. Burger - "Calibration Problems With Neutron Moderator Detectors" - Neutron Monitoring for Radiation Protection Purposes - Vol. II - (Proc. Symp. Vienna, 1972) IAEA, 1973.
6. Alsmiller, R. G., J. Barish - "The Calculated Response of ^6LiF Albedo Dosimeters to Neutrons With Energies <400 MeV" - Health Physics, 26 (1974) 13-28.
7. Data and Code Library Collection of the Radiation Shielding Information Center - RSIC-Oak Ridge.
8. Morhart, A. and G. Burger - "Neutron Deep Penetration Calculations for Uncollimated Phantom Irradiations" - GSF-Report 1977 (in preparation).
9. Gibson, J. B. - Private Communication.

CURRENT WORK IN NEUTRON DOSIMETRY AT CRNL

H. Ing, W. G. Cross and B. J. Tymons
Atomic Energy of Canada Limited
Chalk River Nuclear Laboratories
Chalk River, Ontario, K0J 1J0

Knowledge of the neutron spectrum is considered essential for neutron dosimetry at CRNL. This knowledge is required for selecting a neutron dosimeter most appropriate for a particular radiation environment and compensating for the energy dependence of most dosimeters.

We would like to be able to get spectral information either by calculation or by measurement - whichever is most convenient for a particular environment. To date, we have concentrated on calculations. This approach is particularly useful when the spectrum from the source is known and the environment can be simply described mathematically. An example of this is a neutron source, such as ^{252}Cf or 14-MeV neutrons, shielded by a known thickness of material. For such problems, we have adapted the 05 R Monte Carlo code⁽¹⁾ to the CRNL computer and have had considerable experience with the code for a variety of calculations. Recently we have performed extensive calculations on the attenuation of 14-MeV neutrons through concrete⁽²⁾. Since much of our calculational work has been published or compiled⁽³⁾, I will not elaborate on this work.

Today, I wish to report on our program for measuring neutron spectra, which is very active at the moment. We use a liquid scintillator and several spherical proportional counters in these measurements. Such detectors have been used in several laboratories.

The liquid scintillator is a cylindrical (5 cm x 5 cm) NE-213 detector shown in the first figure. It is mounted on an RCA-8575 photomultiplier and the signals are processed by standard ORTEC modules. Most of the electronics are associated with the n/γ discriminator using "leading-edge" and "cross-over" timing techniques. This spectrometer was intended to cover the energy range 0.5 to 20 MeV. However, with the present n/γ discriminator, there is imperfect gamma rejection and incomplete registration

of neutron signals below about 1.5 MeV. Thus the measured spectra are only reliable above this energy.

I will show some of the preliminary data obtained with this spectrometer, primarily to test its performance. Figure 2 shows the spectrum about 1 m in front of the D-T generator. This spectrum was obtained by unfolding the pulse-height distribution with the FERDOR code⁽⁴⁾ but using the recent response matrix of Ingersoll et al⁽⁵⁾ instead of the FERDOR matrix. We found the response functions of Ingersoll et al to be much closer to those of our detector. The two curves represent the 67% confidence limits defining the spectrum. The peak at 14.6 MeV is quite symmetric, having a resolution of about 10%. The structure in the lower energy region may be due to scattered neutrons. However, we shall be doing some more measurements to ensure that this structure is not produced by an imperfectly matched response matrix.

Figure 3 shows the spectrum of 2.6 MeV neutrons produced by the D-D reaction using the same accelerator. The peak is also well defined. It is interesting that there are a few 14-MeV neutrons as well. These result from tritium adsorbed on the surface of the target holder or in the target itself.

Figure 4 shows the measured spectrum of a small (10^5 n/s) ^{239}Pu -Be source. All the peaks observed are known⁽⁶⁾ from theoretical calculations based on the $^9\text{Be}(\alpha, n)$ cross section data. This spectrum is in good agreement with that measured by Kluge et al⁽⁷⁾ (using a recoil proton telescope) when the areas above 1 MeV are used as normalization.

For measurement of lower energy neutrons, we use four spherical proportional counters, shown in Figure 5. The three smaller counters (2 cm in radius) are filled with 1, 4 and 10 atmospheres of H_2 while the large counter (7.6 cm radius) is filled with about 5 atmospheres of methane-argon. The detectors are sealed in plastic bags containing a drying agent to keep the couplings between the counters and preamplifiers free from moisture. These counters cover the energy range 50 keV to 4.5 MeV with generous overlap between the counters and with the NE-213 scintillator. Because we anticipated thermal neutrons in most measurements associated with radiation protection, these counters were not filled with small amounts of "thermal-neutron

sensitive" gas such as ^3He , normally used for energy calibration purposes.

Since we do not have a Van de Graaff accelerator in our building, the energy scales of these counters have been calibrated using resonances in absorber materials by placing the counter with the absorber into a cavity of a graphite assembly containing a strong (10^7 n/s) Pu-Be source. The absorbers used are teflon (containing fluorine) and SiO_2 (having high oxygen content).

Figure 6 shows the effects of the fluorine resonance at 100 keV and of a broad maximum in the cross section around 300 keV detected by the three small detectors. These data were obtained by unfolding the pulse height distributions with the SPEC 4 code⁽⁸⁾. We have really over-stretched the range of our counters in producing this plot. Normally, we use only the sharp 100 keV resonance for calibrating the 1-atmosphere counter. This resonance is below the operating range of the other two counters and in the region of significant gamma interference. (Normally, the 440 keV resonance in oxygen is used for calibrating the 4-atmosphere counter and the 1 MeV resonance in oxygen is used for calibrating the 10-atmosphere counter.) However, it is interesting that the resonance structure can still be observed outside the range of these counters, where spectral data would not be valid.

In spite of the apparent difficulty in locating the resonance energy accurately, we have found this calibration method acceptable and have checked it using known monoenergetic neutrons produced by the $\text{Li}^7(p,n)^7\text{Be}$ reaction. Figure 7 shows measurements of 118 keV and 397 keV neutrons using the resonance calibration method. The energies are in good agreement.

To facilitate the operation of the four proportional counters, we have constructed an electronic unit which would enable us to use the counters simultaneously. This unit is shown in Figure 8. It provides proper high voltage for the four counters, amplifies and shapes the signals from them and routes the signals into four quadrants of the analyzer memory. The SPEC-4 program is being modified so that the data can be processed in a single run.

Figure 9 shows preliminary results for the measurement of 14-MeV neutrons transmitted through about 5 cm of ^{238}U . The data from the small 10-atmosphere counter, the large (7.6 cm radius) counter and the NE-213 scintillator are presented. Although these data require extensive examination before I consider them valid, it is encouraging that this spectrum is in reasonable agreement with that obtained by our Monte Carlo calculations⁽⁹⁾. Thus, I am optimistic that we shall be able to measure neutron spectra reliably soon.

Acknowledgment

We wish to thank D. Rogers and L. Van der Zwan of NRC (Ottawa) for providing monoenergetic neutrons with their Van de Graaff accelerator.

References

1. Coveyou, R. R., Sullivan, J. G., Carter, H. P., Irving, D. C., Freestone Jr., R. M. and Kam, F. B. K., Oak Ridge National Laboratory Report ORNL-3622 (1965).
2. Cross, W. G. and Ing, H., "Spectra and Dosimetry of Neutrons Interacting with Concrete Shielding" presented at the IVth Congress of the International Radiation Protection Association, Paris, (April 1977).
3. Ing, H. and Makra, S., "Compendium of Neutron Spectra for Criticality Accident Dosimetry" (IAEA, 1977) in press.
4. Burrus, W. R. and Verbinski, V. V., Nucl. Instr. Meth. 67, (1969) 181.
5. Ingersol, D. T., Wehring, B. W. and Johnson, R. H., Oak Ridge National Laboratory Report ORNL/RSIC-40 (1977) 47.
6. Geiger, K. W. and Van der Zwan, L., Nucl. Instr. Meth. 131, (1975) 315.
7. Kluge, H., Weise, K. and Zill, H. W., IAEA-SM-167/7 (1972) 13.
8. Benjamin, P. W., Kemshall, C. D. and Brickstock, A., Atomic Weapons Research Establishment (U. K.) Report AWRE 09/68 (1968).
9. Ing, H. and Cross, W. G. "Calculated Leakage Spectra from ^{238}U Sphere Containing Sources of 14-MeV Neutrons for Radiobiological Applications" presented at the Third Symposium on Neutron Dosimetry in Biology and Medicine, Munich, (May 1977).

FIGURE 1.

FIGURE 2.

FIGURE 3.

FIGURE 4.

FIGURE 5.

FIGURE 6.

FIGURE 7.

FIGURE 8.

FIGURE 9.

DEVELOPMENTAL SUMMARY OF A COMPOSITE GAMMA AND ALBEDO
NEUTRON PERSONNEL DOSIMETER USING
TEFLON MATRIX ELEMENTS

C. Distenfeld*
Associated Universities, Inc.
Brookhaven National Laboratory
Upton, NY 11973

Thermo-luminescent materials are either directly sensitive to albedo neutrons or can be made responsive. In both cases the sensitive ingredient has a high thermal neutron cross section. Incorporation of this material as part of the TL phosphor can provide the response. Alternately, mixtures of neutron insensitive phosphors with non-active materials possessing high thermal neutron cross sections can serve. We have been developing a mixture of $\text{CaSO}_4:\text{DY}$ and non-active ^6LiF in a teflon matrix.

Mixtures and internally sensitive albedo phosphors are both responsive to gamma radiation through the detection of charged particles. Thus, both albedo systems must be able to discriminate against gamma exposure to provide a clear neutron signal. Development of the $\text{CaSO}_4:\text{DY}-^6\text{LiF}$ -teflon neutron detector was preceeded by an investigation of the gamma energy response, Figure 1. Calculations were made for different CaSO_4 cavity sizes in a teflon matrix. Measurements of the CaSO_4 particle size distribution were used to predict the mass weighted particle size response of the composite phosphor. Results are shown as open circles in Figure 1. The phosphor response should have significantly more energy dependence than that measured by Eggermont. This is reasonable in light of the 50 micron mean particle diameter compared to the 5 microns cited by Eggermont.

A correction filter was calculated to suppress the enhanced response. Predictions for a slotted Cu filter, 1% holes, with an aluminum inner cover afforded a usable flat response, Figure 2.

Measurements were made to confirm the predictions, Figure 3. Two filter configurations were tested. Both contained the same slotted outer

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*Consultant for Teledyne Isotopes, Westwood, N.J.

filter. The first design had a 0.038" Cu and a 0.032" Al rear filter sandwiched between the TL teflon dosimeters and a boron back shield. The second rear filter design was a composite of three materials. A 0.032" Al filter was adjacent to the TL dosimeter. This was followed by a 0.005" Cu and a 0.032" Cd. Both filter schemes provide an albedo neutron absorber between the wearer's thorax and the TL dosimeters. Results show, Figure 3, a virtually flat response to less than 30 keV for either badge design.

Interpretation of a neutron exposure depends on the difference between 1) a neutron-insensitive and gamma-responding dosimetry element and 2) the remaining neutron- and gamma-responding element. The two adjacent teflon matrix dosimeters must have the same energy dependence for universal use. Ratios of the outer CaSO_4 :DY teflon dosimeter to the inner dosimeter were determined, Figure 4. Apparently the outer dosimeter attenuated the photons below 40 keV by 10-20%. Thus the difference method should be useful for mixed neutron and gamma fields of any gamma energy above about 30 keV.

The angular response of the slotted filter scheme was investigated, Figure 5. The slotted Cu filter comprised 10% of one readout area. The slot was horizontally arranged to afford the best angular response in the usual exposure orientation. Measurements show a flat response for $\pm 30^\circ$ and a usable response to $\pm 45^\circ$.

Several CaSO_4 - ^6LiF -teflon mixtures were tested for photon and neutron response, Figures 6 and 7. As expected, the gamma response was a direct function of the amount of CaSO_4 present. On the other hand the neutron response increased with decreasing CaSO_4 per unit weight of sample. Maximum was reached between a ^6LiF mole fraction of 0.6 to 0.8. Addition of teflon powder does not appear to significantly alter the previous measurements. Maximum sensitivity does not necessarily afford the best mixture. Figure 8 shows the total to gamma response for three systems. A plain CaSO_4 ^6LiF powder system appears to increase exponentially with increasing ^6LiF content per unit weight of sample. Addition of teflon causes competition for some of the alpha particles liberated by the ^6Li n-alpha reaction. This powder system was loaded to 16% by weight of ^6LiF . Measurements were made with the standard ^6Li and ^7LiF teflon dosimeters. The three powder

system is about 1/2 as sensitive as an LiF teflon dosimeter and about 1/3 as sensitive as the plain powder system. Work is underway to produce extruded CaSO_4 -LiF-teflon dosimeters loaded to about 30%. This heavier loading should provide sufficient enhancement for full testing of an albedo neutron dosimeter.

FIGURE 1.

FIGURE 2.

FIGURE 3.

FIGURE 4.

FIGURE 5.

FIGURE 6. ^{60}Co Response of Several CaSO_4 -LiF Mixtures in Teflon

FIGURE 7.

FIGURE 8.

SIXTH WORKSHOP ON PERSONNEL NEUTRON DOSIMETRY

Eric Geiger
Eberline Instrument Corporation
Santa Fe, NM 87501

SUMMARY

The method described by Eric Geiger and Al Doles (Proceedings of the Ninth Midyear Symposium of the Health Physics Society, Pages 845-849, February 1976) has been used by Eberline for about 18 months. This technique is used to: (1) identify standard LiF dosimeters that have been irradiated with neutrons, (2) calculate neutron and gamma components of a mixed irradiation, and (3) estimate the deep dose equivalent index from neutrons in situations where the spectrum of neutrons remains relatively constant. This method is based on the fact that natural lithium fluoride responds differently to neutrons than to gamma, beta, or X-rays, with more of the stored energy attributable to neutrons being released at a higher temperature (325°C). The neutron response is dependent on the ratio of thermal neutrons to intermediate or fast neutrons. In a facility where this ratio is relatively constant, the technique may be used to estimate neutron dose. In a facility where the ratio is highly variable, the technique may be used to identify persons who should routinely use one of the other methods of neutron dosimetry. The neutron component is reported in counts, with the customer converting counts to mrem based on calibration under typical conditions of neutron moderations at that facility. This calibration is done with a neutron rem-counter and Eberline TLD badges mounted on a one gallon jug of water. The customer applies the mrem/count conversion factor that was determined for his facility. This neutron dose equivalent is then added to the dose record maintained by Eberline for that individual. The dose equivalent (deep and shallow) from X-ray, gamma and beta also are reported by Eberline for each badge.

This technique has worked well over the past 18 months with a few exceptions. Independent performance testing generally provides the kind of data shown in the attached table. Occasionally, a portion of gamma readings

is assigned to neutrons when there has been no neutron irradiation. When this happens, the total deep dose equivalent is overestimated. I think this can be attributed to incomplete readout of the peak at 250°C and can be corrected by increasing the read temperature for the first cycle.

Use of this technique has detected routine neutron irradiations that were completely missed with fast neutron film. These include use and storage of moisture gauges containing ^{241}Am -Be sources, neutrons from a neutron generator penetrating a biological shield, and neutrons at nuclear power plants.

TABLE 1. Independent Performance Test of Eberline Personnel
Dosimetry TLD Service(a)

Badge No.	mR	mrem	mrem Reported			Total Ratio Reported/Delivered
	¹³⁷ Cs	²⁵² Cf	Gamma	Neutron	Total	
26	30		35	0	35	1.17
27	31		38	0	38	1.23
28	39		47	0	47	1.21
29	49		57	0	57	1.15
30	52		61	0	61	1.17
31	72		79	0	79	1.10
32		145	33	97	130	0.90
33		161	30	163	193	1.20
34		266	27	250	275	1.03
35		297	46	243	289	0.97
36		304	43	223	266	0.88
37	119		130	0	130	1.09
38	127		142	0	142	1.12
39	131		154	0	154	1.18
40	240		265	0	265	1.10
41	250		271	0	271	1.08
42	275		323	0	323	1.17
43	62	95	98	33	131	0.83
44	122	128	168	53	221	0.88
45	122	131	178	73	251	0.99
46	98	162	139	127	266	1.02
47	174	274	244	247	491	1.10
48	403	787	506	830	1336	1.12
49	877	1125	1065	1343	2408	1.20
50		100	34	33	67	0.67
51		675	20	817	837	1.24
52	Control					
					$\bar{X} \pm \sigma$	1.07 ± 0.14

(a) Badges were shipped to the testing laboratory for receipt prior to February 1, 1977. They were received by the testing laboratory on January 26, and exposed on February 11 and 12. They were shipped back to Eberline on February 14, received by Eberline on February 18, read and reported by Eberline on February 22, 1977. All badges were exposed on a phantom.

ANL PERSONNEL NEUTRON MONITORING STUDY
EMPLOYING ^6LiF ALBEDO DOSIMETERS

Elwyn H. Dolecek
Argonne National Laboratory
Argonne, Illinois 60439

The potential for neutron exposures exists at several areas at Argonne National Laboratory (ANL). Specific areas include the Zero Gradient Synchrotron (ZGS), Dynamitrons, Van de Graaf Accelerators, Californium Sources and the Zero Power Reactors (ZPR). Until rather recently the method employed for personnel neutron monitoring was to use the traditional film-filter system. Although it is a known fact that the film response is unsatisfactory to a band of neutron energies, exclusion control has been utilized to hold assessed neutron exposures to a small fraction of the radiation worker's maximum permissible dose (MPD) limit.

However, the ZPR facility, which is used to study the behavior of various fuel configurations, presents a special case of concern since a great deal of manual handling with significant quantities of plutonium is performed by radiation workers, thus giving rise to the possibility of a person receiving a large fraction of their MPD via neutron exposure. Although consideration of automating many of the manual operations is underway, the time involved in design and implementation has accelerated the need for developing an effective personnel monitoring program for neutron exposures. Since the neutron spectrum present around the ZPR was known to have a large component of intermediate energy neutrons, a dosimetry system more responsive to these neutrons was necessary. Solid state track recorders (SSTR's) were tested and then utilized for neutron monitoring since they would provide the needed sensitivity to the intermediate energy neutrons.¹ However, this dosimeter does present several disadvantages when employed as a personnel monitor:

- 1) The dosimeter contains approximately one-half gram of ^{235}U thereby leading to an unnecessary exposure to the person wearing the badge;

- 2) The presence of the fissile ^{235}U presents an accountability problem;
- 3) The ^{235}U trips the alarm systems employed by the safeguards and security group.

Therefore, it seemed advisable to seek a different type of neutron monitor which would replace the SSTR's. Since the objective was to obtain quickly a more suitable monitoring system this precluded a great deal of time and effort for development and testing. Therefore, a survey of the literature was conducted for the possible types of neutron dosimeters available.^{2,3} After studying the papers dealing with the development and testing of albedo-neutron dosimeters employing ^6LiF it was decided to investigate the possibility of utilizing the albedo-neutron dosimeter designed by Hankins.

In order to adequately employ the Hankins albedo-neutron dosimeter several aspects concerning the dosimeters and the Argonne facilities where neutron exposures could be expected were investigated.

- 1) Determination of ^6LiF output response to homogeneous neutron energy exposures.
- 2) Supralinearity of ^6LiF crystals to neutrons.
- 3) Expected neutron spectrums that the dosimeters would receive and the response of the ^6LiF dosimeter to the heterogeneous neutron spectrums while employed in the dosimeter package.
- 4) Applicability of ^6LiF -Hankins dosimeter as a personnel neutron monitor.

The study indicated that the dosimeter design demonstrated an inverse dosimeter response to neutron energy relationship as shown in Figure 1. As the neutron energy was increased from 50 keV to 2.5 MeV, the thermoluminescent output decreased by a factor of approximately 30 for the same mrem dose equivalent received by the dosimeter.⁴

FIGURE 2. Dose Linearity of Neutron Exposed ^6LiF .
Dose response of ^6LiF exposed to thermal neutrons.

FIGURE 3. Dosimeter Response to ZPR Neutron Spectrum.
Composite graph of mRem/ ^6LiF count versus neutron energy for a Rando phantom exposed at several locations to the neutron spectrum from the Argonne ZPR

II.

TESTS OF EXISTING SYSTEMS

NEUTRON DOSIMETRY STUDIES AT THE
LAWRENCE LIVERMORE LABORATORY*

Dale E. Hankins
Lawrence Livermore Laboratory
Livermore, CA 94550

This report summarizes work performed since the last Workshop meeting. The studies have been divided into the following sections:

1. Response of albedo neutron dosimeters to low energy neutrons.
2. Discussion of dose-equivalent conversion factors.
3. Modification of the A-B remmeter.
4. The effect of distance from the body on the response of albedo neutron dosimeters.
5. Comparison of albedo neutron dosimeter techniques.
6. Modified NAD badge for additional beta shielding and albedo neutron dosimetry.

Response of Albedo Neutron Dosimeters to Low Energy Neutrons

A study of the response of albedo neutron dosimeters to intermediate-energy neutrons was made to confirm the accuracy of the response curves calculated by Alsmiller and Barish.¹ The intermediate-energy neutrons were from the NBS reactor and have predominant energies at 2.0, 25 and 144 keV. The beams are obtained by using a combination of resonant scatterers and filters. They are of high intensity, have low backgrounds of other energy neutrons and gamma rays, and are well calibrated and stable. The 2-keV beam is of particular interest since it provides a calibration point which is about one decade lower in energy than has been available previously. We also made measurements at neutron energies between 30 keV and 5.67 MeV, obtained from the Lawrence Livermore Laboratory cyclograph accelerator. The energy dependence of two neutron remmeters was also included in this

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*Work performed under the auspices of the U. S. Energy Research and Development Administration, under Contract No. W-7405-Eng-48.

study. The remmeters were the 9-in. sphere Portable Neutron Rem Counter Model PNR-4 manufactured by Eberline Instrument Corporation, Santa Fe, NM, and the Andersson-Braun (A-B) type Remmeter built to LLL specifications. The results obtained with these instruments are available elsewhere.²

The albedo neutron dosimeters were the Hankins-type³ which have cadmium completely surrounding the TLDs. Additional results were obtained by placing TLDs on the bottom of the albedo dosimeter (between the dosimeter and the phantom) to simulate the simpler type albedo dosimeter.

The thermal and intermediate-energy neutron beams from the NBS reactor at ~2 m from the reactor shield are small (~4.3- to 8.9-cm diam) requiring that the phantoms be placed on a scanning table. This table moves at a constant speed horizontally and in steps vertically. The table can be programmed to scan the area desired. The cyclograph results were obtained by supporting the phantoms on tripods in a low-scatter target room.

The albedo neutron dosimeters were placed on a chest manikin filled with water. Fluences for the cyclograph exposures were determined using distances measured from the center of the source to the face of the phantom. The results obtained in this study are given in Table 1.

The relative response of the albedo neutron dosimeters as a function of neutron energy is shown in Figure 1. A curve is shown for the response of the Hankins-type albedo neutron dosimeters and for TLDs taped to the bottom of this dosimeter. The two curves are similar, indicating that changing the TLD location does not cause a change in the energy dependence of the dosimeter. The calculated values are from Alsmiller and Barish,¹ and were made for the Hankins dosimeter. Very good agreement is found between experimental and observed results.

The ratio of 9- to 3-in. spheres, given in Table 1, is shown in Figure 2. This ratio is used to determine the calibration factor for albedo neutron dosimeters.^{3,4} This ratio can also be used to determine the average neutron energy, but care should be used in applying average energies since they can be misleading. We used the ratio in this study to assure us that the desired neutron energy was, in fact, being produced.

TABLE 1. Results Obtained with Albedo Neutron Dosimeters and the Ratio of the 9- and 3-in. Sphere When Exposed at Various Neutron Energies

		Albedo Neutron Dosimeters		
		Relative Response		Spheres
		Hankins Type	Under Cd	Ratio 9/3
Cyclograph				
30	keV	10.25	31.8	0.20
49.7	keV			0.23
70	keV	5.57	18.8	0.24
100	keV	1.99	6.0	0.27
146	keV			0.34
200	keV	1.16	3.6	0.42
300	keV	0.885		0.52
350	keV	0.701		0.63
400	keV			0.61
500	keV	0.251	0.949	0.97
580	keV			0.90
700	keV			1.08
800	keV			1.23
1	MeV	0.135	0.383	1.70
1.2	MeV			1.85
1.476	MeV	0.0900		2.32
2.0	MeV	0.104		2.48
3.053	MeV	0.788		3.27
5.666	MeV			4.59
4.52	MeV			4.21
N.B.S.				
Thermal		0.281	37.0	1.45
2	keV	17.7	48.7	0.13
25	keV	11.5	35.7	0.21
144	keV	1.48		0.38

The NBS and cyclograph exposures were made with greatly differing experimental procedures, but the results plotted in Figures 1 and 2 at overlapping energies show good agreement. This gives us confidence in the accuracy of both experimental procedures.

FIGURE 1. Curves Showing the Energy Dependence of the Hankins-Type Albedo Neutron Dosimeter and of TLDs Taped to the Bottom of the Dosimeters. (Also Shown is Calculated Reponse by Alsmiller and Barish.)

FIGURE 2. Ratio of Response of the 9-in.-Sphere Remmeter to the Response of the 3-in. 10-mil Cd-Covered Sphere

Discussion of Dose-Equivalent Conversion Factors

We now have available for use at the NBS, neutron beams of 2, 25 and 144 keV. The NBS reports the intensity of these beams in units of fluence and not dose.

The conversion factors from fluence to dose equivalent are given in NCRP Report No. 38 for specific neutron energies.⁵ For energies other than those given, it is recommended in the report that "linear interpolation between neighboring energies" given in the report be used. Frequently, values at other energies were taken from curves drawn on log or semilog graph paper between the specified energies. The linear interpolation gives a peculiar shape to the dose-equivalent curve when plotted on log or semilog paper and at 25 keV the values obtained by various extrapolations will vary by as much as a factor of 2. Jack Hoy of the Savannah River Plant requested additional calculations which were made by Poston and Jones,⁶ using the same computer program that was used to calculate the values given in NCRP-38. They indicate that for neutron energies of 2, 25 and 144 keV, the linear interpolation procedures give values very close to the calculated conversion factors. For additional information and discussion see also Reference 7.

It is therefore recommended that: 1) the linear interpolation recommended in NCRP Report 38 be followed; and 2) the conversion value used by the author be specifically stated. A table of the dose equivalent values using linear interpolation from the NCRP report is given in Table 2 below. Also shown are the values applicable at 25 and 144 keV for evaluation of the NBS beams.

Modification of the A-B Remmeter

The Andersson-Braun remmeter⁸ consists of a cylindrical moderator, and consequently has a directional dependence caused by this cylindrical shape. It also has a directional dependence to thermal neutron beams which varies greatly; being low by a factor of ~ 3 when exposed to the side, when exposed to the end it is low by about a factor of 9 but is high by a factor of 17 when exposed at the instrument packet end.

TABLE 2. Linear Interpolation of Dose-Equivalent Conversion Factors from Table II. of NCRP Report No. 38

Energy	Dose Rate*	Energy	Dose Rate*	Energy	Dose Rate*
Thermal to					
0.1 eV	272**	90 eV	231.1	90 keV	72
0.1 eV	272**	100 eV	232**	100 keV	46**
0.2 eV	267	200 eV	236	200 keV	37.2
0.3 eV	261	300 eV	241	300 keV	28.4
0.4 eV	256	400 eV	245	400 keV	19.6
0.5 eV	251	500 eV	250	500 keV	10.8**
0.6 eV	245	600 eV	254	600 keV	10.2
0.7 eV	240	700 eV	259	700 keV	9.5
0.8 eV	235	800 eV	263	800 keV	8.9
0.9 eV	229	900 eV	268	900 keV	8.2
1.0 eV	224**	1 keV	272**	1 MeV	7.6**
2.0 eV	224	2 keV	272.9	2 MeV	7.87
3.0 eV	224	3 keV	273.8	2.5 MeV	8.0**
4.0 eV	224	4 keV	274.7	3 MeV	7.68
5.0 eV	224	5 keV	275.6	4 MeV	7.04
6.0 eV	224	6 keV	276.4	5 MeV	6.4**
7.0 eV	224	7 keV	277.3	6 MeV	6.6
8.0 eV	224	8 keV	278.2	7 MeV	6.8**
9.0 eV	224	9 keV	279.1	8 MeV	6.8
10.0 eV	224**	10 keV	280**	9 MeV	6.8
20 eV	224.9	20 keV	254	10 MeV	6.8**
30 eV	225.8	30 keV	228	11 MeV	6.3
40 eV	226.7	40 keV	202	12 MeV	5.8
50 eV	227.6	50 keV	176	13 MeV	5.3
60 eV	228.4	60 keV	150	14 MeV	4.8**
70 eV	229.3	70 keV	124		
80 eV	230.2	80 keV	98		

* mrem/h per n/cm²/sec

** Values from Table II 25 keV 241
144 keV 49.9

We have modified the remmeter by making some rather simple and inexpensive (~\$50) changes to the moderator. The modifications consist of partially rounding the end of the instrument, moving the BF₃ tube forward by 1/2 in., increasing the size of the holes in the boron-loaded polyethylene sleeve, replacing one of the boron-loaded end plugs with a polyethylene plug, changing the locations of the other end plug, and adding a small disk of cadmium over the hole where the BF₃ tube enters the moderator.

These modifications to the instrument increase the thermal neutron sensitivity by a factor of ~ 3 when exposed at the side and a factor of ~ 9 at the end. The high thermal sensitivity at the instrument packet end (a factor of ~ 17) is eliminated, and the directional dependence to both thermal and fast neutrons is reduced to $\pm 10\%$ (except at the packet end). The sensitivity of the instrument to fast-energy neutrons is increased by $\sim 15\%$.

The sensitivity of the modified remmeter to intermediate energy neutrons is increased, and is similar to that of the 9-in. sphere. We compared the readings of the modified instrument, the Andersson-Braun remmeter, and the 9-in. sphere remmeter to numerous spectra encountered in field application. This study proved to be more interesting than expected because the neutron fields varied and at different places could be defined as point, plane and isotropic sources. The comparison of the remmeter readings and the modification we made to the instrument is described in Reference 9.

The Effect of Distance from the Body on the Response of Albedo Neutron Dosimetry

During one of my trips to the NBS facilities we exposed several albedo neutron dosimeters to the thermal column beam of the NBS reactor. From previous work^{10,11} we knew that the thermal response of an albedo neutron dosimeter increases rapidly as the distance from the body increases. At that time we did not thoroughly study this effect. In the current study we used 30-mil-thick cadmium disks of 3/4-, 1-, 1.5- and 2-in. diameters; a boron disk of 1-1/4-in. diameter; and TLDs mounted under the Hankins-type albedo neutron dosimeter. Exposures were made with the dosimeters at contact with the phantom and in 1-cm increments to a distance of 4 cm away from the phantom.

The readings of the TLDs at various distances from the phantom are given in Table 3. In Figure 3 the results are shown normalized to the contact reading. As the distance from the phantom is increased, the TLD readings first increase rapidly, and then gradually decrease. The magnitude of the increase depends on the size of the dosimeter, being higher for the larger dosimeters.

TABLE 3. TLD Readings Obtained at Various Distances from a Phantom and with Several Types of Albedo Neutron Dosimeters. (All exposures were to thermal neutrons from the NBS reactor.)

<u>Albedo type</u>	<u>TLD reading in light units</u>				
	<u>Contact</u>	<u>1 cm</u>	<u>2 cm</u>	<u>3 cm</u>	<u>4 cm</u>
Bare TLDs	~50,000				
3/4 in. diam Cd	9 686	18 255	18 389	17 528	15 493
1 in. diam Cd	6 615	15 113	16 935	16 403	13 973
1.5 in. diam Cd	4 140	10 231	13 439	13 375	12 174
2.0 in. diam Cd	2 670	6 233	9 426	10 029	9 647
1.5 in. diam B	6 294	11 723	14 675	14 184	13 356
Under Hankins- type dosimeter	7 417	15 545	19 609	18 660	17 106

The readings given in Table 3 for the TLDs located behind the 2-in.-diam. disk at contact with the phantom are a factor of 4 lower than the readings behind the 3/4-in. disk. When the phantom-to-dosimeter distance is 4 cm, this drops to a factor of 1.5.

The results with the boron disks at contact with the phantom (see Table 3) indicate that there is some leakage of thermal neutrons through the 30-mil Cd disks. This thickness of Cd was known to allow some thermal neutron leakage and this leakage was used in the design of the Hankins-type albedo neutron dosimeter to give it a response to thermal neutrons equal to its response to albedo neutrons at ~1.0 MeV.³

FIGURE 3. The Relative Readings of Several Types of Albedo Neutron Dosimeters at Various Phantom-to-Dosimeter Distances. The exposures were made to thermal neutrons from the NBS reactor. All readings were normalized to the contact reading (0 distance from the phantom).

The large increase in the response of albedo neutron dosimeters to thermal neutrons described above must also have an effect on the response of dosimeters when exposed to fast neutrons. A drop in the fast neutron response of albedo neutron dosimeters as the distance from the phantom is increased has been reported by several investigators, but the magnitude of this decrease varied. It appeared to us that at least part of the discrepancies could be caused by a small component of thermal neutrons which was different for the various studies. We investigated this by making a series of exposures in our "low scatter" neutron facility at LLL. The thermal neutron dose rate is approximately constant throughout the facility, but the fast neutron dose rate varies rapidly ($\sim$ inverse square) as the distance from the source changes. Therefore, the ratio of the dose rates from thermal to fast neutrons can be changed by making exposures at various distances from the source.

We repeated the type of study we made with thermal neutrons by exposing several types of albedo neutron dosimeters to a ^{252}Cf neutron source in our neutron facility at several source-to-phantom distances. The results we obtained have been plotted in Figure 4 and show that as the thermal component increases (distance from the source increases), the decrease in the albedo neutron reading at distances from the phantom becomes less and at 3 m we actually observe higher readings at 1 and 2 cm from the phantom.

We have drawn only two curves in Figure 2 at each source-to-phantom distance, since six curves proved to be confusing. The bottom curve at each distance is drawn through the data points for the boron disks and the top curve is an "eyeball" average through the other points.

We made an additional exposure in the same facility using a PuBe neutron source and the 1-in.-diam Cd and 1-1/4-in. boron disks. The results are plotted in Figure 5 and show some significant differences from the ^{252}Cf results of Figure 4. At 50 cm and 1 m the curves in Figure 5 overlap and at 2 and 3 m the boron curve is higher than the curve for the cadmium. There is also less drop in the response of the dosimeter at the 4-cm dosimeter-to-phantom distance with PuBe at 50 cm than for ^{252}Cf at the same location.

FIGURE 4. The Relative Response of Several Types of Albedo Neutron Dosimeters at Various Phantom-to-Dosimeter Distances. The exposures were made to a ^{252}Cf source in a "low-scatter" neutron facility at several source-to-phantom distances.

FIGURE 5. The Relative Response of Two Types of Albedo Neutron Dosimeters at Various Phantom-to-Dosimeter Distances. The exposures were made to a PuBe source in a "low-scatter" neutron facility at several source-to-phantom distances.

The boron disk results at 3 m for PuBe are higher than the ^{252}Cf results by ~40%. This and the overlapping of the curves at 50 cm and 1 m are caused by the slightly different energy dependence of cadmium and boron albedo neutron dosimeters. This difference was investigated at LLL by V. Boggs,^{12,13} but was found to be too small for practical application in field use.

The thermal neutron dose rate in this study varied from about 0.03% of the total dose rate at 50 cm to ~0.6% at 3 m. The thermal component of the dose rate was about 20% larger for the ^{252}Cf source than for the PuBe source at all distances.

This study indicates that the change in reading of an albedo neutron dosimeter as a function of distance from the phantom is very large for thermal neutrons, and for fast neutrons 1) depends on the type of source used, and 2) is affected by the scattering conditions (thermal neutron fluence) at the facility. The magnitude of the TLD readings is also affected by the dosimeter material (boron vs. Cd).

The above study indicates to us that for fast neutrons in field conditions there is probably an increase in the reading of a dosimeter if it is worn away from the body. This assumption is based on measurements made in field conditions where the thermal component of the neutron dose rate normally is between 3 and 6% and seldom is as low as the 0.6% which exists in our "low scatter" facility. This means that the importance of holding an albedo neutron dosimeter tightly to the body needs more study under field conditions. We intend to make such a study when time permits.

Comparison of Albedo Neutron Dosimetry Techniques

T. R. Crites of the LLL has been working with an albedo neutron dosimeter which is evaluated by using a ratio of the TLD readings from above and below a boron disk.¹⁴ A correction for the energy dependence of the albedo dosimeter can be made by applying the appropriate calibration factor. This factor is obtained from a curve of the ratio vs. sum of the two readings. A curve is drawn for a particular source and moderator type and in Ref. 14 Crites gives four curves obtained in our neutron facility for PuBe and ^{252}Cf sources and with moderators of hydrogen and D_2O . One can make the assumption that in field application most results should follow the ^{252}Cf hydrogen-moderated curve.

To check the accuracy of this assumption, results were obtained with instruments at numerous field locations and the percent of the total neutron dose rate delivered by thermal neutrons was calculated. This "percent thermal" was plotted as a function of the albedo neutron dosimeter calibration factor in Figure 6 for numerous survey locations at LLL. The calibration factors were determined by using the 9- and 3-in. sphere technique. The spread of data points indicates that there is no strong relationship between the percent thermal and the calibration factor and therefore,

FIGURE 6. The Calibration Factors for Albedo Neutron Dosimeters as a Function of the Percent of Dose Rate Delivered by Thermal Neutrons

the calibration factor is not closely related to the incident thermal neutron dose. For example, at an albedo calibration factor of ~ 0.35 the percent thermal in Figure 6 varies from ~ 0.2 to 5.0, and at a calibration factor of ~ 1 the percent thermal varies from 0.9 to 9.

We continued this study by comparing, at 18 selected field locations, the results obtained using the Crites technique and those obtained from the Hankins-type albedo neutron dosimeter. The Hankins dosimeter results were interpreted by 1) determining the ratio of 9- and 3-in. sphere at the exposure location and from this ratio obtaining the calibration factor for that point, and 2) applying the average calibration factor determined for the building or facility. The results using procedure 1 were understandably better: within $\pm 23\%$ with two exceptions (-40 and $+46\%$) compared to $\pm 33\%$ with three exceptions (-40 , -61 , $+40\%$). The Crites technique gave results similar to the above with the exception of the reactor results, which had a large error. With the exception of the reactor results, neither technique was decidedly more accurate.

The decision was made to use the Hankins-type dosimeter at LLL. This was based on several considerations: 1) neither dosimeter proved significantly superior in the above test; 2) the Hankins dosimeter does not have to be oriented properly or held tightly against the body; 3) information on the exposure location or type is not required with the Crites technique but at LLL, personnel who are exposed to neutrons do not work at various locations and normally are not exposed to different types of neutron spectra; 4) the Hankins dosimeter could be mounted on the existing Nuclear Accident Dosimeter (NAD) badge giving a one-piece dosimetry package; 5) the Crites dosimeter has greater sensitivity but the Hankins dosimeter is considered to have adequate sensitivity for the type of spectra our personnel usually encounter; and 6) the results of the Hankins dosimeter can be used with the TLD 100 results of the personnel badge to make an analysis similar to that of Crites.

In the future, when the Hankins dosimeter is in extensive use, we hope to make an analysis of the TLD 100 and albedo dosimeter results. This study may be helpful in evaluating exposures where the appropriate calibration

factor may not be known. We are also planning to replace part of the polyethylene in the albedo neutron dosimeter with track etch foils using recoil particles and the electrotechnical etch technique. A report on this work is given by Griffith and is included elsewhere in this report.

Modified NAD Badge for Additional Beta Shielding and Albedo Neutron Dosimetry

The LLL Nuclear Accident Dosimeter (NAD) is issued to personnel who work in an area where a criticality accident could occur. The dosimeter (see Figure 7) snaps around the personnel TLD dosimeter (upper left). We have modified the NAD to include additional beta shielding for the gamma-sensitive TLD. The beta shield for this TLD is presently limited to the security badge which allows high energy betas to be recorded as penetrating gamma-rays.¹⁵ Personnel who do not have a NAD badge will wear a lexan disk for additional beta shielding. The modified NAD also includes at the bottom of the badge a holder for the albedo neutron dosimeter. The albedo neutron dosimeter will be pressed into the holder and held in place by tape.

The entire dosimeter packet will be exchanged at monthly intervals since the albedo TLDs must be handled by trained personnel and the TLD disks can be inserted backwards.

In most cases, the personnel at LLL who receive neutron exposures are already wearing an NAD badge. We also plan to replace part of the polyethylene in the albedo neutron dosimeter with a recoil particle track etch dosimeter. See the section by Griffith in this report for details.

FIGURE 7. The NAD Badge Modified To Include Additional Beta Shielding and Provide a Holder for the Albedo Neutron Dosimeter. At the top right is the old badge, in the center the modified badge, and at the bottom is the assembled and disassembled albedo neutron dosimeter.

References

1. Alsmiller, R. G., Jr., and Barish, J., Oak Ridge National Laboratory, Report ORNL-TM-3984 (1972).
2. Hankins, D. E., Proceedings, International Radiation Protection Association IVth International Congress, Paris, France, pp 553-556 (1977).
3. Hankins, D. E., Proceedings, 9th Midyear Topical Symposium, Health Physics Society, Denver Colorado, pp 861-868 (1975).
4. National Council on Radiation Protection and Measurement (NCRP) Report 38, Washington, D.C. (1971).
5. Poston, J. W. and Jones, T. D., personal communication (1976).
6. Rogers, D. W. O., On the Definition of Dose Equivalent Received from a Neutron Flux, National Research Council, Canada, Report PXR-2432.
7. Andersson, I. O. and Braun, J., Nukleonik 6 237 (1964).
8. Hankins, D. E., paper to be presented at the 1977 Health Physics Society meeting, Atlanta, GA, Preprint UCRL-78928 (1977).
9. Hankins, D. E., Los Alamos Scientific Laboratory, Report LA-4832 (1972).
10. Hankins, D. E., IAEA Symposium on Neutron Monitoring for Radiation Protection Purposes, Vienna (1972).
11. Unruh, C. N., Ed., Battelle Northwest Laboratory Report BNWL-1616 (1971).
12. Griffith, R. V., IAEA Symposium on Neutron Monitoring for Radiation Protection Purposes, Vienna (1972).
13. Crites, T. R., Health Phys 31 154-157 (1976).
14. Hankins, D. E., and Madden, T. W., Lawrence Livermore Laboratory, Hazards Control Progress Report 50, UCRL-5007-71-1, pp. 3-5 (1974).

NEUTRON SPECTRA FROM REACTORS BY THE DELTA SPECTROMETER

Robert A. Facey
Central Health Physics Services
Ontario Hydro
Pickering, Ontario L1V 2R5
Canada

Fission neutrons and photo-neutrons can contribute significantly to the radiation hazard in some areas around power reactors. The energy degradation of the neutrons is generally unknown. With the albedo dosimetry system⁽¹⁾ presently being introduced at Ontario Hydro generating stations, there is considerable neutron energy dependence.⁽²⁾ A knowledge of the neutron energy spectrum in each area is necessary to assign a corrected value of the dose equivalent.

The "Delta" spectrometer, developed by the CEGB at Berkeley Nuclear Laboratories, England^(3,4) has been adopted by Central Health Physics Services^(5,6) for use by Ontario Hydro. Briefly, the instrument consists of a 30-cm-long moderating cylinder of polyethylene (PE) shielded against neutrons on all surfaces except the front end. In a removable core, ⁶LiF TLDs are used to measure the linear distribution of thermal neutrons along the cylinder axis. Gamma subtract is obtained with ⁷LiF. The thermal neutron distribution along the axis is determined by the spectrum of neutrons incident on the front end. A computer program uses the response from previous monoenergetic neutron calibrations to unfold the measured response to obtain the unknown neutron spectrum.

Typical spectra obtained are shown in the Figures. Fifteen energy groups were used, of which the groups from 1 eV to 10 MeV are shown in the plots. The display is fluence per unit lethargy and all graphs have been normalized to a constant number of neutrons (i.e., constant area of plot).

As a test of the method, the spectrometer was exposed to an Am-Be source, bare in air and also inside spheres of water of radii 4.6, 9.1 and 23.3 cm, respectively. The measured spectra were compared with calculated spectra of Ing⁽⁷⁾ and the agreement appeared reasonably good. Figures 1 and 2 show

the measured and calculated spectra for the 9.1-cm-radius sphere. A quantity of low-intermediate or epithermal neutrons is seen in Figure 1, probably due to scattered neutrons from the environment, which naturally are not present in the calculation of Figure 2.

Figure 3 shows the spectrum inside the containment vault looking at the reactor face of a CANDU reactor. It was made at low reactor power during the commissioning of a 750-MWe reactor at Bruce Generating Station. The spectrum is broad, peaking at 100 keV to 1 MeV. A minor peak at 10-30 keV may possibly be related to the 25-keV window in Fe in the steel end shield.

A rather similar spectrum is shown in Figure 4. Here a contributing source is photo-neutrons from boilers and heat transport pumps in one of the Pickering boiler rooms. At another location in the same room the spectrum is narrowed and peaks at 30-100 keV (Figure 5).

In the Fueling Machine Service Room, where there is much scattering of the neutrons, the spectrum shows a double peak (Figure 6).

Recent modifications to the delta spectrometer include the use of a polycarbonate (PC) moderator. Being of lower hydrogen content than PE, the PC instrument gives greater spatial distribution of the thermal neutrons and should give improved resolution. Another method being developed is the use of activation foils (Dy) to measure the thermal neutrons. This has the advantage of better precision than TLD and total gamma immunity.

References

1. J. E. Hoy, Health Phys., 23,385 (1972). Also "Personnel Albedo Neutron Dosimeter with Thermoluminescent ^6LiF and ^7LiF ", Savannah River Laboratory Report DP-1277 (1972).
2. R. G. Alsmiller and J. Barish, Health Phys., 26, 13 (1974).
3. E. M. Gayton, J. A. Hall and G. A. M. Webb, "The Use of Passive Detectors for the Evaluation of Neutron Shielding", Radiation Measurements in Nuclear Power, paper 3.4 Institute of Physics and the Physical Society (1966).
4. J. P. Longworth, "A Neutron Flux Spectrometer with Nearly Constant Sensitivity over the Range Thermal to 14 MeV", C.E.G.B. Report RD/B/N1416 (1970).

5. M. L. Walsh and B. H. Orr, "Neutron Spectra in Ontario Hydro CANDU Reactors", Health Physics Society Annual Meeting, San Francisco (1976).
6. M. L. Walsh and B. H. Orr, "Measurement of Neutron Spectra in Ontario Hydro CANDU Reactors", Fifth International Conference on Luminescence Dosimetry, Sao Paulo, Brazil, (1977).
7. H. Ing, personal communication to M. L. Walsh (Feb. 1975).

FIGURE 1. Measured Am-Be + Medium Sphere

FIGURE 2. Am-Be + Medium Sphere (Calc.)

FIGURE 3. Bruce-A Unit-2 Reactor Vault

FIGURE 4. Pickering B. Rm. Location No. 1

FIGURE 5. Pickering B. Rm. Location No. 4

FIGURE 6. Pickering FM Service Rm.

THREE COMMON PROBLEMS

D. W. O. Rogers
Physics Division, National Research Council
Ottawa, Canada K1A 0R6

At N.R.C. we have been involved in calibrating an albedo neutron dosimeter to be used in Ontario Hydro's CANDU power reactors. The results of our calibrations of the Hoy dosimeter from 20 keV to 14 MeV will be published in Health Physics shortly¹⁾. The next step in implementing the dosimeter is to measure the neutron spectra found around the reactors. Bob Facey from Ontario Hydro is reporting on these results at the workshop. The final step is to numerically fold the known dosimeter response with the spectra to obtain calibration factors for the various areas where neutron exposure occurs. Rather than discuss these results, I would like to raise three well known problems which have arisen in this work.

The first concerns the factor used to convert from neutrons per cm^2 to rem. There are at least three widely used sets of factors based on the calculations of W. S. Snyder (NCRP #38, ICRP #21, the Defence Nuclear Agency cross-section library). Figure 1 shows the startling variations in these three recommended sets of conversion factors. The differences are mostly due to different interpolation procedures, but they also arise because of the difficulties in determining the maximum dose equivalent from calculations which just give the integral of the dose equivalent over 2-cm-thick volumes. In principle these variations are not too upsetting because they only reflect our true ignorance. In practice however, they are extremely confusing since they make research results hard to compare. Results should be quoted in terms of n/cm^2 where this is what is measured, and in the longer term, the ICRP and NCRP should adopt a uniform recommendation after more detailed calculations are done (especially in the 10-keV to 1-MeV range).

The second problem arises from the following considerations: i) the neutron flux around a reactor may be highly directional or isotropic; ii) an albedo dosimeter has at most a 3 or 4% response to most reactor neutrons incident from the "far" side¹⁾; iii) an albedo dosimeter has a 40% response

to neutrons incident uniformly from all sides compared to a frontal flux; and iv) due to the definition of dose equivalent as a maximum value (ICRU #25), even an ideal isotropic rem-meter calibrated in a mono-directional beam will over respond by a factor of 2-1/2 to 3 in an isotropic flux of intermediate energy neutrons. These various factors mean that a single albedo dosimeter which is properly calibrated for a given spectrum in a mono-directional flux will give a dose equivalent reading which is: i) about right in an isotropic field (the decrease in response matches the drop in rem per unit flux); ii) low by a factor of ≈ 3 for a flux from the side or below; and iii) low by a factor of 20 for a flux from the rear. Addition of a second dosimeter at the rear which is added to the first badge will improve things but still gives a 30% low reading for a flux from the side or bottom and an over response by a factor of ≈ 2 in an isotropic field. Another aspect of this problem is that we cannot test our dosimeters at a reactor site by merely comparing them to a calibrated rem-meter unless we have a good knowledge of the directional characteristics of the field as well as the spectrum since the rem-meter and dosimeter have different directional responses. In short, these considerations appear to dominate the uncertainties involved in using an albedo dosimeter, even if the difficult question of calibration for a given neutron spectrum can be solved.

The last concern I would like to raise is the response of Andersson-Braun-like rem-meters (SNOOPY) in the 10- to 75-keV region as shown in Figure 2. The 20- to 50-keV points were obtained using the $^{45}\text{Sc}(p,n)$ reaction²⁾ and thus avoided the problems inherent in using the $^7\text{Li}(p,n)$ reaction in this energy region. The over response is even higher than previous reports indicate^{3,4)} mainly due to the NCRP #38 conversion factors used. In the worst case we have found around a CANDU reactor, this over response would lead to a 140% overestimate of the total dose. More typically the reading would be 30% higher and for harder spectra there is no effect. Nonetheless, this response curve emphasizes the danger of using a rem-meter as a calibration instrument in an unknown field and the danger inherent in using the difference between a SNOOPY reading and a fast neutron dosimeter to estimate the dose due to intermediate energy neutrons.

References

1. D. W. O. Rogers, M. L. Walsh, B. H. Orr, N. Teekman, Albedo-Dosimeter Response to Monoenergetic Neutrons, in press, Health Physics, 1977.
2. D. W. O. Rogers, The $^{45}\text{Sc}(p,n)$ Reaction as a Source of Monoenergetic 10 to 50 keV Neutrons, Nucl. Instr. and Methods **142** (1977) 475-478.
3. J. R. Harvey, A. Lavender and I. M. G. Thompson, The Calibration of Neutron Sensitive Instruments, Health Physics **31** (1976) 363-369.
4. D. Nachtigall and G. Burger, Dose Equivalent Determinations in Neutron Fields by Means of Moderator Techniques, Ch 7 in Topics in Radiation Dosimetry, ed F. H. Attix, (Academic Press, New York, 1972).

FIGURE 1. The Percentage Difference Between NCRP #38 Recommendations (Using a Linear Interpolation Between Values of $(n/\text{cm}^2)/\text{rem}$) and Those of ICRP #21 and the DNA Cross Section Library.

FIGURE 2. The Response of a SNOOPY to a Beam of Monoenergetic Neutrons Falling on the Side of the Cylinder. (The dashed line is the quoted calibration and the black square on the right is for an Am-Be source. The neutrons were provided by the $^{45}\text{Sc}(p,n)$ (20, 50 keV), $^7\text{Li}(p,n)$ and $^{11}\text{B}(p,n)$ (1 MeV) reactions.)

NEUTRON DOSIMETRY AT FONTENAY-AUX-ROSES, FRANCE

Roger Medioni
C.E.A. DPr--Step
Center d'Etudes Nucleaires
B.P. No. 6
92260 Fontenay-Aux-Roses, France

Neutron dosimetry is one of the major concerns of our department at present.

Fast neutron dosimetry is always carried out with the aid of Kodak NTA film. For low energy neutrons, we are at the moment using the ionographic detector (CA 80-15), but have high hopes of the possibilities of exoelectron emission, a technique which is being studied in our laboratories.

1. Ionography (1)

We use the commercial film CA 80-15 (Kodak-Pathe). It comprises a thin layer of natural lithium tetraborate deposited on cellulose nitrate (100 μm). The low energy neutrons produce (n, α) reactions in the boron 10 and lithium 6. The α particles are registered in the cellulose nitrate.

The treatment is very straightforward:

- washing of the dosimeter in de-ionized water to eliminate lithium tetraborate
- chemical treatment (3 hours at 40°C in a solution of 2.5N NaOH)
- rinsing and washing
- counting with a light microscope the tracks produced.

The dosimeter is compact (45 x 15 x 4 mm) and one part is shielded with cadmium (thickness 0.8 mm, diameter 8 mm).

For thermal neutrons ($E_n < 0.4 \text{ eV}$) the yield is 1100 tracks/cm² for 1 mrem, and for epithermals ($0.4 \text{ eV} < E_n < 1 \text{ keV}$), 70 tracks/cm² for 1 mrem. These figures are determined by standardization in a thermal pile and in a k/E spectrum.

The minimum detection limit for thermal neutrons is 2 mrem and for epithermal neutrons 30 mrem. The maximum doses we can measure are limited by the superimposition of the tracks (1 rem and 15 rem, respectively).

This detector is completely insensitive to γ -rays. It is practically insensitive to neutrons of an energy superior to 10 keV; therefore, its utilization in optimum conditions necessitates a knowledge of the neutron spectrum, which permits determination of the standardization coefficient applicable in the measurement zone under consideration.

At present a study involving about 60 persons is being carried out at Fontenay-aux-Roses. The ionographic dosimeters exhibit thermal neutron doses (10 to 40 mrem in one month) but very rarely show epithermal doses; classical dosimeters give no dose reading for the same personnel. This study will be continued until October, when we will draw the final conclusions.

11. Exoelectron Emission (2)

These studies are carried out in our laboratories by Mr. M. Petel. We have developed readers adapted to two forms of stimulation (thermal and optical).

Following the difficulties met by those who have used non-conductive BeO ceramics, we have concentrated our efforts on α -aluminium oxide mixed with an electrical conductor.

The first results obtained in the field of neutron dosimetry are promising. Using a PuBe source, the relative sensitivity to ^{60}Co γ -rays is:

0.40 when irradiated under 2 mm of plexiglass

0.08 when irradiated under 2 mm of graphite.

The most common electrical conductor used in exoelectron studies is graphite. This however does not lend itself easily to incorporation in ceramic preparations. As a substitute we have chosen β -aluminium oxide which is not exoemissive but a good ionic conductor. Solid compact stable dosimeters may be made with the mixture of these two aluminium oxides (α and β) and we are in the process of studying, using polyethylene convertors, their response to neutrons.

111. Other Studies

A systematic study of the response of thermoluminescent detectors to neutrons has been undertaken. Results obtained have recently been published at the IRPA congress (3).

A study concerning dosimetry by the Albedo system was started in 1977. In the first part we are examining the possibilities of the detector proposed by Harvey (4).

References

1. Chapuis A. M., A.T.S.R. IXeme Congrs sur le Contrle des Rayonnements Ionisants. 15-18 April 1975 Grenoble, France.
2. Petel M., Holzapfel G., Exoemissive properties of a mixture of aluminium oxides in α and β phases: Dosimetric applications. IVeme Congrs International de l'IRPA Paris 1977.
3. Medioni R. et al., IVeme Congrs International de l'IRPA, Paris 24-30 April 1977. 4 No 165.
4. Harvey J. R. et al., CEGB Report RD/B/N1547 Berkeley Nuclear Laboratories, U.K.

RESPONSE OF THE SAVANNAH RIVER ALBEDO NEUTRON
BADGE TO MONO-ENERGETIC SOURCES AT NBS

Jack Hoy
E. I. DuPont de Nemours and Co.
Savannah River Plant
Aiken, SC 29801

In the Figure below, the response is compared to Asmiller's calculated response. The three open points represent the new data. The 25 keV point reported last year was apparently low since it increased to follow the general shape of the theoretical response curve.

FIGURE 1.

CURRENT SITUATION FOR EXOELECTRON DOSIMETERS
OF BeO CERAMIC IN NEUTRON DOSIMETRY*

R. B. Gammage
Health and Safety Research Division
Oak Ridge National Laboratory
Oak Ridge, Tennessee 37830

Much of the early enthusiasm for using exoelectron dosimeters (ceramic BeO Thermalox 995) in neutron dosimetry was predicated on the belief that the response to fast neutrons, relative to gamma rays, was 0.18 to 0.28 on an R_{γ} equiv/tissue rad n_f basis for neutron energies between 0.1 and 16 MeV.¹ Pairs of BeO disks had to be used, one covered with a polyethylene radiator for producing recoil protons, and the other covered with Teflon. More recent studies indicated a considerably lower ratio of 0.11 for Health Physics Reactor Research fission neutrons.² In the earlier work the BeO was coated with gold to enhance the surface conductivity during reading of the thermally stimulated exoelectron emission (TSEE). No metallic coating is now deemed to be necessary. Perhaps thermal neutron contamination of the fast neutron beams due to some thermalization within the hydrogenous radiator was sufficient to cause the high apparent fast neutron sensitivity via n, γ reactions. Whatever the cause, however, the lower value of 0.11 has caused a marked subsidence of enthusiasm in this technique of fast neutron monitoring.

The greatest drawback arises when attempting to measure neutron doses in mixed gamma, neutron fields in which the ratio of doses is biased in favor of the gamma component. A TSEE reading is accurate to no better than 5%. Since the neutron dose is estimated from the differential of the readings of two exoelectron dosimeters, the relative error will be about 65%. Since in a 1:1 neutron to gamma field, the dosimeter with the proton radiator reads only 11% higher than the Teflon covered dosimeter, the difference of the TSEE readings is two-thirds as large as the desired fast neutron TSEE response. Such an undesirable situation is depicted in Figure 1 with

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*Research sponsored by the Energy Research and Development Administration under contract with Union Carbide Corporation.

large error bars assigned to the fast neutron doses at positions under the boron paraffin shield. Inside the central cavity, where the neutron-to-gamma ratio of doses is 7:1, the relative error in measuring the neutron dose is lowered to 13%. In this experiment the fast-neutron dose was being profiled spatially. A TSEE reading takes only a few minutes so that the profiling operation was completed within 30 minutes.

The fast neutron response rapidly decreases to zero when the neutron beam becomes nearly parallel to the face of the BeO disk (recoil protons no longer enter the BeO). The directional response to fast neutrons and gamma radiation is shown in Figure 2. A cube for holding BeO disks with polyethylene plugs (Figure 3), which permits the incident direction of the neutrons to be rapidly ascertained,² has been tested as a nuclear accident monitor.

It is worth mentioning that a dual TL-TSEE reading of a single polyethylene covered BeO disk now provides the same data as the previously used pair of BeO dosimeters.³ The BeO as a TLD has only an extremely weak response to the recoil protons. The gamma dose can, therefore, easily be measured from the TL reading alone. This technique allows the number of BeO disks for mixed fast-neutron, gamma-radiation field measurements to be reduced by 50%.

Successful operation is, however, still dependent on the presence of a neutron rich field. This same requirement forces us to conclude that this dosimetry system is unsuitable for generalized fast neutron personnel monitoring.

References

1. K. Becker and K. W. Crase, Nucl. Instrum. Methods 82, 297 (1970).
2. R. B. Gammage, F. F. Haywood, M. H. Lee and J. S. Cheka, "Fast Neutron Monitoring with Exoelectron Emitting BeO Dosimeter", ORNL-TM-4624 (1974).
3. R. B. Gammage and A. K. Garrison, Proc. 4th Int. Symp. on Exoelectron Emission and Dosimetry, 1973, Ed. A. Bohun, Czech. Acad. of Sci. and A.E.C., 93 (1973).

FIGURE 1. Fast Neutron (closed points) and Gamma (open points) Dose Profile Below a Shield Designed for the Partial-Body Irradiation of Mice

FIGURE 2. Directional Response of Ceramic BeO Disk TSEE Dosimeters to 14-MeV Neutrons and Gamma Radiation, Using Thick Polyethylene and Teflon Radiator Covers

FIGURE 3. Teflon Cube, 2.5 cm in Diameter, with Six Recesses and Plugs of Polyethylene or Teflon to Hold Six BeO Ceramic Disks 1.25 cm in Diameter. The device is used for locating the direction of a neutron beam as well as for measuring the fast neutron and gamma doses.

ELECTROCHEMICAL ETCHING FOR PERSONNEL NEUTRON DOSIMETRY

L. Tommasino

Laboratorio di Dosimetria e Biofisica delle Radiazioni
Centro di Studi Nucleari della Casaccia
Casaccia, CNEN, Roma, Italy

The detection of damage tracks of fast-neutron-induced recoils by electrochemical etching seems to provide promising applications in personnel neutron dosimetry (1).

The electrochemical etching of damage tracks is a very complex but poorly understood phenomenon (2). Better understanding of the mechanisms of this type of etching might be useful to explain troublesome background phenomena, the peculiar effects of forbidden overlapping of tracks, detector directional response, etc. (3-5).

By definition, electrochemical etching is obtained whenever damage track detectors are stressed by an ac electric field during chemical etching (2,6). In principle, a large variety of combinations of electric fields and frequencies (up to GHz) could be used. The most often used electrochemical etching for fast-neutron-induced recoil tracks is obtained by electrical fields from 10 kV/cm up to 50 kV/cm with frequencies in the range of 1 kHz to 10 kHz (1-6). In the following discussion, we will try to analyse the mechanisms of electrochemical etching under these particular electrical conditions.

The first stage of electrochemical etching concerns the formation of pits in the neutron-induced recoil tracks on the detector surface. When compared with etching made only chemically, the track pit formation is different since the high electrical field applied to the insulating detector enhances the penetration of the etching electrolyte and its ions both into the tracks and into the bulk of the polymer. It has been recently shown that the "absorption" of the electrolyte and its ions into various organic polymers under alternating stress occurs as effectively as under direct electric stress (7,8). This is contrary to the expectation that there should not be any net absorption under alternating stress.

Once the track pits are formed the electrical fields at the tips of the tracks can be much larger than the average electrical field applied to the insulator. To evaluate the electrical field enhancement at the tip of a metallic needle, Mason derived an equation (9), which may be assumed valid also for the needle-shaped tracks filled with conductive etching. In this equation (reported in Fig. 1), the ratio between the maximum electric field at the tip of the needle, $E_{\max}$, and the average electric field, E_{avg} , is given versus the radius of the needle point, r , and the distance, d , between this point and the opposite surface. If we use Mason's equation for a track tip with a radius of 0.1 micron and with an insulator thickness of 200 microns, the electric field enhancement at the tip of the track is about 450. For an electric field applied, E_{avg} , of about 20 kV/cm, the field of strength, $E_{\max}$, at the tip of a track can be as high as 9 MV/cm.

This high stress concentration initiates localized breakdowns at the track tips. These partial breakdowns are known as treeing phenomena (10), since the type of damage they produce looks like a tree when observed through the dielectric section as shown in the lower part of Fig. 1. Trees at damage tracks were first reported by R.V. Griffith, through observations made by the electron scanning microscope (11).

A comprehensive analysis of electrochemical etching phenomena is very difficult (2). The simplified model described here is useful for the understanding of some important characteristics of electrochemically etched track detectors. For example, two tracks can be so near one another that no voltage gradient exists between the tips of the tracks. The propagation of treeing is then forbidden in the region between the two tracks. This explains why track overlapping is forbidden with electrochemical etching, a phenomenon which makes possible measurements of neutron doses up to thousands of rads (1,3,5).

The strong directional response for the detection of fast-neutron-induced recoil tracks can be explained considering that tree growth initiates preferentially at the tips of tracks parallel to the field, where the highest local field gradient is present. Since the reaction of fast-neutron-induced heavy recoils is not isotropic (12), then the detector response must be dependent on neutron directions.

Treeing phenomena have been known for 40 years (13), but various laboratories throughout the world became interested in these phenomena only recently (14), because of the importance they may have in the failure of high voltage cables. The investigations already carried out on treeing in insulators might be useful for a better understanding of electrochemical etching. For example it has been shown that trees initiate and propagate from channels originated in the amorphous regions of partially crystalline materials (15). Polymers with the highest percentage of crystallinity exhibit the greatest resistance to the initiation of trees. To register electrochemically etched tracks with low background, detector materials should have high crystallinity.

Since thermal treatments alter the physical structure of polymers (16), it might be possible that the number of background tracks increases after thermal annealing (17). If the annealing is carried out in such a way that the crystallinity or the structure compactness of the polymers increases, then the number of background tracks should decrease.

Experiments carried out in our laboratory have indicated that the background can be lowered by a factor of 5 when the polycarbonate detectors are properly annealed (18).

FIGURE L. Basic Mechanisms of Electrochemical Etching

References

1. Sohrabi M. (1974) Health Physics 27, 598-600
2. Tommasino L. (1970), CNEN Report RT/PROT (71), 1
3. Sohrabi M. and Morgan K.Z. (1975), Prog. Rept. OR0-4814-1
4. Somogyi G. (1977), Nuclear Track Detection 1, 3-18
5. Hassib G.M., Tuyn J.W.N. and Dutrannois J. (1976) 9th Int. Conf. on Solid State Nuclear Track Detectors-Munich
6. Tommasino L. (1970) Italian Pat. N° 51929A/70
7. Auckland D.W. and Cooper R. (1975) Proc. IEE, 122, 860-864
8. Tobazeon R. and Gartner E., 1974 Annual Report on Conference on Electrical Insulation and Dielectric Phenomena, (Printing and Publishing Office-National Academy of Science, 1975), p. 404-413
9. Mason J.H., in "Progress in Dielectrics", Vol. 1, (Birks J.S., Schulman J.H.; Heywood & Co. Ltd, London, 1959)
10. Tommasino L. and Armellini C. (1974) Journal of Applied Physics 45, 949-950.
11. Griffith R.V. (1976), Report UCRL-50007-76-2
12. Angular Distribution in Neutron-Induced Reactions (1970) (Garber D.I., Stromberg L.G., Goldberg M.D., Cullen D.E. and May V.M. editors) BNWL 400, Vol. 1
13. Robinson D.M., Dielectric Phenomena in High Voltage Cables (Chapman & Hall Ltd, London, 1936).
14. McMahon E.J., 147th Meeting of the Electrochemical Society, Toronto, Canada, May 16 (1975).
15. Wagner H., 1974 Annual Report on Conference, Electrical Insulation and Dielectric Phenomena (National Academy of Sci., Washington D.C., 1975) p. 62
16. Marchetti M., Tommasino L. and Casnati E. (1974), Radiation Effects, 21, 19-24
17. Cotter S.J. and Gammage R.B., Twenty-second Annual Meeting of the Health Physics Society, July 3-8, 1977
18. Zapparoli G. and L. Tommasino, unpublished.

LLL DEVELOPMENT OF A COMBINED ETCH TRACK - ALBEDO DOSIMETER*

R. V. Griffith, J. C. Fisher and C. A. Harder
Hazards Control Department
Lawrence Livermore Laboratory
Livermore, CA 94550

The use of thermoluminescent albedo dosimeters is becoming increasingly popular for personnel neutron dosimetry. High sensitivity and ease of automation make these dosimeters an attractive alternative to NTA film or fission track detectors for routine personnel dosimetry. The major limitation of albedo detectors is that the dose equivalent response is severely neutron energy or spectrum dependent (1). They are, primarily, low energy detectors (Figure 1). In fact, the relative dose equivalent response can vary as much as a factor of 30 between exposure to an unmoderated fission source and one heavily moderated by D₂O (Figure 2). Although, in practice, the spectral variation found in the workplace should not cause such severe dosimeter response variations, considerable uncertainty about proper calibration factors may still exist.

One attractive approach used by others (2,3) has been the addition of a fast neutron detector such as threshold fission foils to the albedo neutron dosimeter so that the fast neutron dose component can be estimated more accurately without losing the advantages of the TL detector. At LLL, we have found the use of electro-chemically etched polycarbonate (ECEP) to be an attractive alternative to fission foils or NTA film for this purpose (4). Laboratory employees retain their security badges with built in dosimeter when they go home. As a result, we are not able to use radioactive material such as ²³²Th or ²³⁷Np for personnel monitoring. ECEP, on the other hand, is nontoxic, inexpensive, can be included directly in an albedo badge (unlike NTA film), and has an energy response similar to ²³²Th (n, fission) (Figure 3). One additional advantage is that the ratio of the track density on the neutron incident side of an ECEP detector foil to the exit side track density is significantly greater than 1.0 for neutrons near the energy

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*Work performed under the auspices of the U.S. Energy Research and Development Administration under contract No. W-7405-Eng-48.

threshold (Figure 4). This provides one with the potential of being able to estimate the orientation of the person being monitored at time of exposure. Knowledge of body orientation would be very helpful in attempting to evaluate the effect of body shielding or scattering on dosimeter response.

Conceptually, an albedo-ECEP dosimeter is evaluated as follows. The TL detector is read routinely for each monitored employee. If, and only if, the TLD indicates a significant neutron exposure ($>50 - 100$ mrem), we will process the ECEP component. This consists of removing the protective polyethylene film from each of four 0.25-mm thick pieces of polycarbonate located in the bottom of the dosimeter cup (Figure 5). The samples are then exposed to a UV lamp for one hour to complete a process of photo-oxidation that appears to be necessary to render many of the tracks etchable (4). The etching process involves placing the samples in a holder between two volumes of 28% KOH, and applying a 2-kHz, 800-V ac potential across the sample for 5 hours. The resulting tracks are actually tree-shaped, sub-surface voids (Figure 6). Viewed from the surface of the sample, they have a circular cross section with an average diameter of $70\text{ }\mu\text{m}$, so that they can be counted easily with a microfiche reader (Figure 7).

In evaluating dosimeter response we begin by determining the ratio of track densities on each side of the ECEP foils. An air/body track density ratio greater than about 1.5 indicates that the neutrons were incident on the air side; i.e., a frontal exposure. A ratio of less than 0.5 indicates the wearer had his back to the neutrons. The most likely situation is a ratio of about 1.0, indicating that the neutrons were not unidirectional; i.e., that the source was spatially distributed and/or that the wearer was moving around in the field. The result of the orientation evaluation will govern the dosimeter calibration factors to be used.

For dose equivalent evaluation, the combined dosimeter is considered to have a two component spectral response; ECEP for neutrons with energies greater than 1.5 MeV and albedo for neutron energies less than 1.5 MeV (the value of 1.5 MeV is somewhat arbitrary). The effective dose equivalent response of ECEP to neutrons with energies above 1.5 MeV is relatively

constant for fission or moderated fission spectrum neutrons. Using a ^{252}Cf source, and spectra calculated with the ANISN discrete ordinate code, we obtain a value of $80 \text{ tracks} \cdot \text{cm}^{-2} \cdot \text{Rem}^{-1}$ ($E_n > 1.5 \text{ MeV}$) for a frontal exposure and $6.4 \text{ tracks} \cdot \text{cm}^{-2} \cdot \text{Rem}^{-1}$ ($E_n > 1.5 \text{ MeV}$) for a dosimeter on the back of a phantom.

The effective dose equivalent response of the albedo detectors for neutrons with energies below 1.5 MeV varies significantly with neutron spectrum. Rather than using a single effective calibration factor, we will use the ratio of ECEP response to albedo response to determine the spectral quality. Knowing something about the spectral quality, we will have a more accurate estimate of proper effective dose equivalent response for the albedo detector. To understand more about the relationship between ECEP-albedo ratio and the effective conversion factor for albedo response, we used Monte Carlo calculated spectra for many moderated fission neutron configurations (5), to calculate the effective response of albedo detectors and compare them with the equivalent calculated ECEP-albedo ratio (Figure 8).

It is clear that the relationship between relative albedo dosimeter response and ECEP-albedo ratio depends heavily on the moderator material. However, quality of the relationship can be roughly characterized as falling in two groups - low Z moderation (H_2O , D_2O , concrete, graphite, beryllium) and "metal moderation" (iron, copper, lead, uranium). It, therefore, would be valuable and probably not impractical to know something about the moderator materials used in the working environment.

A relationship of the form $C = e^{(a-b\ln R)}$ can be determined, which approximates each class of moderator response (Figure 8), where C is the effective albedo detector calibration factor, a and b are constants dependent on the spectral sensitivities of the ECEP and albedo detectors, and R is the ECEP-albedo ratio. Using experimental data normalized to the responses shown in Figure 8, we obtain the following equation for evaluation of dose equivalent from the ECEP-albedo dosimeter components:

$$\text{DOSE EQUIVALENT, D.E.} = Ae^{(a-b\ln R)} + cT = a'Ae^{-b\ln R} + cT$$

where   A = ALBEDO RESPONSE - Gamma Rad
           T = ECE RESPONSE - Track/mm²
           R = ECE RESPONSE/ALBEDO RESPONSE
           a' and b depend on sensitivity and readout of specific dosimetry systems
           c = 1.25 Rem/Track/mm² (E_n > 1.5 MeV, Front Exposure)
           c = 16 Rem/Track/mm² (E_n > 1.5 MeV, Rear Exposure)
           Typical Background - 0.01 Tracks/mm²
           Typical Area of Count - 10³ mm²

We have made preliminary tests of a prototype combined dosimeter at the recent ORNL Personnel Dosimetry Intercomparison and at a location in our plutonium storage vault. Reference dose equivalent data for the Intercomparison were provided by the ORNL staff. An Anderson-Braun rem-meter was used as a reference for the vault measurements. The results are presented in Table 1.

TABLE 1. Albedo - ECEP Test Exposures

<u>SOURCE</u>	<u>REFERENCE</u>	<u>DOSE EQUIVALENT - REM PHANTOM FRONT</u>	<u>PHANTOM REAR</u>
HPRR - BARE	0.635	0.63	0.55
HPRR - LUCITE	0.443	0.31	0.30
HPRR - STEEL	0.719	0.60	0.80
332 - VAULT	0.425	0.33	0.48

Conclusion

The addition of polycarbonate sheet to albedo detectors for electrochemical etching provides a simple, inexpensive way to reduce the spectral sensitivity of the personnel dosimeter without losing the albedo features of sensitivity and ease of automation. The ECEP technique also provides the dosimetrist with the potential for identifying conditions of body orientation that might otherwise lead to significant error in dosimeter evaluation.

Acknowledgments

We wish to thank Dale Hankins for guidance on the use of albedo detectors and Leroy Davis for processing the samples.

References

1. R. G. Alsmiller, Jr. and J. Barish, "The Calculated Response of Albedo Neutron Dosimeters to Neutrons with Energies ≤ 400 Mev." Health Physics 26, 13 (1974).
2. B. J. Tymons, J. W. N. Tuzn and J. Barli, "System for Personnel Dosimetry in Mixed Radiation Fields", Proc. Symp. Neutron Monitoring for Radiation Protection Purposes", Vol. II, Vienna (1973), p. 63.
3. C. Distenfeld, "Calibration of the Experimental BND Personnel Neutron Dosimeter for General Use", Fifth ERDA Workshop on Personnel Neutron Dosimetry, p. 20, 1975 (BNWL-1934).
4. R. V. Griffith, J. C. Fisher and C. A. Harder, "Light Particle Track Registration for Fast Neutron Personnel Dosimetry", Hazards Control Progress Report No. 52, p. 40, (1976) UCRL-50007-76-1.
5. H. Ing and S. Makra, Compendium of Neutron Spectra for Criticality Accident Dosimetry, to be published, IAEA.

FIGURE 1. Calculated Neutron Response of "Hankins Type" Albedo Dosimeter* Compared with ICRP-21 Dose Equivalent Conversion Factors
(*After Alsmiller and Barish, Ref. 1)

FIGURE 2. Relative Response of "Hankins Type" Albedo Dosimeter to Fission Spectrum Neutrons in Spherical Moderators (After Ing and Makra, Ref. 5)

FIGURE 3. Measured Neutron Response of Electro-Chemically Etched Polycarbonate (ECEP)

FIGURE 4. Measured Ratio R. of Track Densities on Incident to Exit Sides of ECEP Vs. Neutron Energy

FIGURE 5. Drawing of Combined ECEP, Albedo Dosimeter

FIGURE 6. Cross Section of ECEP Sample Showing Profile of a Track

FIGURE 7. Photograph of a Commercial Microfiche Reader Showing ECEP Sample

FIGURE 8. Calculated Relative Sensitivity of "Hankins Type" Albedo Dosimeter to Neutrons with Energies Less than 1.5 MeV for

- a. "Low Z" Moderated Fission Spectra
- b. "Metal" Modulated Fission Spectra

COMMERCIAL IMPLEMENTATION OF TRACK ETCH NEUTRON DOSIMETRY

Robert V. Wheeler
R. S. Landauer, Jr. & Co.
Science Road
Glenwood, Ill. 60425

In keeping with the concept of providing dosimetry services of highest quality it was determined that the use of track etch techniques for determination of the neutron portion of personnel exposure would result in an overall improvement in this state-of-the-art.

The development program pursued included not only the determination of relevant parameters of dosimeter response, but also an evaluation of whether fission fragment radiators or damage sites from inherent recoil particles was the appropriate course to follow. Also needed was an analysis of conventional etching procedures and electrochemical etching techniques.

It was determined that polycarbonate plastic foils would be used as the dosimeter material without any external radiator material. This decision was made in part due to the preferred method of foil etching, specifically electrochemical etching.

We do not intend to imply that track etch dosimetry represents any optimum or ideal system, only, in our opinion, the best technique available at this time.

When polycarbonate foils are etched at 1000 volts rms, 2000 Hz, 22°C, using 28% KOH, the following parameters are recorded:

Foil Background	25 mRem (equivalent), 4-5 tracks/cm ²
Gamma Response	None noted
Fading	Less than 5% (22°C) in 4 months
Directional Response	40% at 90° with phantom
Energy Response	1.0 MeV threshold with increasing sensitivity to 3.0 MeV and Rem Response above 3.0 MeV
Sensitivity	40 mRem ± 30%, 300 mRem ± 10%, 1600 mRem ± 7% for an AmBe energy spectrum

Effort is being directed toward identifying a realistic model of the process being measured, with the objective of improving the overall system performance.

EVALUATION OF DIFFERENT POLYMERS FOR FAST NEUTRON PERSONNEL
DOSIMETRY USING ELECTROCHEMICAL ETCHING*

R. B. Gammage and S. J. Cotter
Health and Safety Research Division
Oak Ridge National Laboratory
Oak Ridge, Tennessee 37830

There is considerable optimism for the enhancement by electrochemical etching of fast neutron-induced recoil tracks in polycarbonate for the purpose of personnel dosimetry. The threshold energy, however, is rather high. A desirable improvement would be to lower this energy below 1 MeV. With this objective in mind, we have commenced an investigation of cellulose acetate, triacetate, and acetobutyrate in addition to polycarbonate. These cellulose derivatives are chemically more reactive and physically weaker than polycarbonate. It might, therefore, be possible to initiate the electrochemical amplification at the sites of shorter recoil atom damage tracks than is possible with polycarbonate. Katz and Kobetich¹ published threshold energies for etchable tracks produced by carbon ions in cellulose nitrate and polycarbonate that correspond to threshold neutron energies of 0.4 and 1.3 MeV, respectively. Threshold neutron energies significantly lower than 1.3 MeV are anticipated for cellulose derivatives and especially for the nitrate.

Some characteristics important for electrochemically etching in aqueous electrolytes are listed in Table 1. Here chemical etching is combined with treeing,² an electrical breakdown process that starts when the dielectric strength is exceeded. These mechanical and electrical properties pertain to the dry plastics. The absorption of water molecules and electrolyte ions will cause these values to be reduced.

Each of the fast neutron irradiated cellulose derivatives was etched (25°C) in 30% by weight KOH solution at the arbitrarily chosen frequency and

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* Research sponsored by the Energy Research and Development Administration under contract with Union Carbide Corporation.

TABLE 1. Properties of Commercial Plastics

Polymer	Average Tensile Strength (psi)	Dielectric Strength (volts/25 μ m)	Volume Resistivity (ohm-cm)	Water Absorptivity (wt.% in 24 hrs)
Cellulose Acetate	~6550	~500	10^{13}	2-7
Cellulose Triacetate	~7000	~300	$\sim 10^{14}$	2-4
Cellulose Acetobutyrate	~5900	~400	10^{12}	1-2
Polycarbonate	7500	1500	10^{16}	0.15-0.35

voltage of 1000 Hz and 1000 volts (RMS), respectively. Lower values of 500 Hz and 500 volts were also used.

Cellulose Acetate

This plastic absorbed so much water that the surface developed a sponge-like texture making it difficult to distinguish the etched tracks from the background (Figure 1). Cellulose acetate has the highest water absorptivity of the four plastics under consideration. Unless the electrochemical etching can be conducted in a suitable nonaqueous electrolyte, this material is unsuitable for use as a fast neutron personnel dosimeter.

Cellulose Triacetate

Water absorption in triacetate is high enough to cause a general clouding effect, but the problem is less severe than for the acetate. Lowering the frequency and voltage (500 Hz, 500 V) increased the size and improved the shape so that the etched tracks became sufficiently contrasted against the surrounding milky background to permit them to be counted. Optical and scanning electron micrographs are reproduced in Figures 2 and 3.

Cellulose Acetobutyrate

At the higher frequency and voltage, the electromechanical stresses fractured the foil causing the two halves of the electrochemical etching chamber to short. Even at 500 Hz and 500 V, cracks again developed but this time they were nonpenetrating. Microscopic examination revealed the cracks and their positional relationship to the etched recoil tracks (Figure 4). The poor electrochemical strength of this plastic is to be expected from the characteristics listed in Table 1. A parallel study of the electrochemical etching of alpha induced damage tracks in acetobutyrate by Ashraf³ identified 400 Hz (at 1000 volts) as being the optimum frequency for clarity of tracks in visual counting without problems of cracking.

Unlike the previous two cellulose derivatives, however, water absorption was insufficiently high to cause clouding and the attendant viewing problems for track counting. Cellulose acetobutyrate is an attractive candidate for further studies of fast neutron detection and particularly with respect to an anticipated lower neutron energy threshold limit.

Polycarbonate

In keeping with the findings of other investigators such as Sohrabi⁴ and Griffith,⁵ the etched recoil atom tracks proved to be very uniform in shape (round). The subsurface and surface appearances are shown in Figures 5, 6 and 7. No clouding or cracking problems have been experienced at frequencies up to 2300 Hz. In these respects, polycarbonate is the material of choice for fast neutron dosimetry except that the threshold energy is somewhat above 1 MeV,⁽⁶⁾ which is higher than desirable.

Structure of Electrochemically Etched Tracks

Each of the four plastics studied reveals a common structure for the electrochemically etched, recoil atom track. A small entrance hole at the surface penetrates to a much larger tree-structured cavity in the bulk. This cavity is partially collapsed and covered by an inwardly sagging skin that has a "dished" appearance. These effects are depicted particularly well in the micrographs of Figures 2 and 3. After etching of the triacetate for four hours, the entrance holes at the surface have widened to only a few

microns in diameter, while the partially collapsed areas have diameters of about 150 microns.

The ideas advanced at this workshop by Luigi Tommasino for the treeing phenomenon and the cross-sectional micrographs shown by Dick Griffith provide a likely explanation for the overall shape and structure of the electrochemically etched tracks. As Tommasino points out, in stage 1 a pointed pit is first etched through the surface until it is large enough to accommodate conducting fluid. One then has the equivalent of a metallic needle pressing into the insulator. Application of the electric field gradient at the tip starts stage 2 and the treeing process, which is electrical breakdown combined with chemical etching [$E_{\text{maximum}}/E_{\text{average}} = 2d/(r \ln(1 + 4d/r))$], where d is the foil thickness and r is the tip radius in the direction of the applied electric field]. The tree tends to grow into the bulk of the plastic; the "branches" do not grow back toward the surface because the effective radius of the tip becomes very large as a branch approaches an inclination normal to the direction of the applied field. The skin overlying the cavity, therefore, remains immune to the electrochemical etching process, and presumably retains its integrity until destroyed by normal etching processes.

Conclusions and Directions for Future Research and Development

Cellulose acetate is unsuitable for electrochemical etching because of severe fogging and swelling caused by water absorption. The same problem exists for cellulose triacetate but to a lesser degree, and the problem can be reduced by lowering the frequency and voltage. Cellulose acetobutyrate does not fog but it can be fractured macroscopically by electromechanical stresses unless the frequency is lowered to only a few hundred Hz. Successful electrochemical etching of these plastics depends on the use of lower frequencies and voltages than are optimum for polycarbonate and/or perhaps the use of nonaqueous electrolytes. These behaviors are in keeping with the known physical and chemical characteristics of these plastics.

There is good reason to hope, therefore, that with optimized electrochemical etching, a cellulose derivative like acetobutyrate (or perhaps nitrate) will perform well as a fast neutron dosimeter and with a significantly lower energy threshold limit than that of polycarbonate.

The pitting and treeing mechanism of electrochemical etching proposed by Tommasino provides an adequate explanation for the general structure of the etched artifacts in cellulose derivatives and polycarbonate, viz., a sagging skin with a small entrance channel leading to a much larger diameter, tree-structured cavity.

References

1. R. Katz and E. J. Kobetich, Phys. Rev. 170, 401 (1967).
2. L. Tommasino and C. Armellini, J. Appl. Phys. 45, 949 (1974).
3. J. Ashraf, "Optimization of Electrochemical Etching Parameters for Alpha Particle Registration in Polymers and Its Application to Radon Monitoring," M. S. Thesis, University of Tennessee, December 1976.
4. M. Sohrabi, Health Phys. 27, 498 (1974).
5. R. V. Griffith, Hazards Control Progress Report No. 51, University of California, Lawrence Livermore Laboratory Report UCRL-500007-75-2, 1 (1976).
6. K. Jesezewicz, Nucl. Instrum. Methods 93, 369 (1971).

FIGURE 1. Optical Micrograph of Recoil Particle Tracks in 762- μ m-Thick Cellulose Acetate Etched for Eight Hours in 30% KOH at 1000 V and 1000 Hz. (500X)

FIGURE 2. Optical Micrograph of Recoil Particle Tracks in 250- μ m-Thick Cellulose Triacetate Etched for Four Hours in 30% KOH at 500 V and 500 Hz. (500X)

50 MICRONS 200
0.002 0.004 300X 0.008 0.010
INCHES

10 20 MICRONS 60 70
0.001 1000X 0.003
INCHES

FIGURE 3. Scanning Electron Micrographs of Recoil Particle Tracks in 250- μ m-Thick Cellulose Triacetate Etched for Four Hours in 30% KOH at 500 V and 500 HZ [(a) 300X; (b) 1000X].

FIGURE 4. Optical Micrograph of Recoil Particle Tracks in 250- μ m-Thick Cellulose Acetobutyrate Etched for Four Hours in 30% KOH at 500 V and 500 Hz. (500X)

FIGURE 5. Optical Micrograph of a Recoil Particle Track in 250- μ m-Thick Polycarbonate Etched for Four Hours in 30% KOH at 1000 V and 1000 Hz. (500X)

FIGURE 6. Scanning Electron Micrograph of a Recoil Particle Track in 250- μ m-Thick Polycarbonate Etched for Four Hours in 30% KOH at 1000 V and 1000 Hz. (300X)

FIGURE 7. Scanning Electron Micrograph of a Recoil Particle Track in 250- μ m-Thick Polycarbonate Etched for Four Hours in 30% KOH at 1000 V and 1000 Hz. (1000X)

SOME EXPERIENCES ON THE ELECTROCHEMICAL
ETCHING AMPLIFICATION OF CHARGED
PARTICLE TRACKS IN POLYMERS

M. Sohrabi
Radiation Protection Department
Atomic Energy Organization of Iran
P.O. Box 41/2663
Tehran, Iran

It is the purpose of this summary to quote the existing data available to the author on the electrochemical etching (ECE) of charged particle tracks in polymers and their applications in many areas of science and technology. The "electrochemical etching" method should be distinguished from the widely used method known as "conventional etching."

The conventional etching method is based on immersing the irradiated foils into a chemical solution under optimum etching conditions. It was originated by Young¹ and was followed by extensive developments and findings by Fleischer, Price, and Walker. Over 1500 papers and reports have been published as quoted in the literature.²⁻⁴

The development of the elegant electrochemical etching method proposed by Tommasino in 1970 brought a new dimension into the charged particle track registration technology.⁵ It is based on local energy loss along the tracks in an alternating field of sufficient strength of a certain frequency in a chemical solution of known composition, concentration and temperature.

Due to a high demand for the development of neutron personnel dosimeters that avoid the shortcomings of the existing techniques, an attempt was made by the author to apply the ECE method for the amplification of fast-neutron-induced recoil particle tracks, and alpha particle and fission fragment tracks in some polymers for personnel dosimetry. The results of that investigation were very interesting but did not seem to be very practical.⁶

Surprisingly, no further studies were done until 1973 when an intensive investigation was initiated at the Georgia Institute of Technology. The results of this investigation were very encouraging and practical enough to

lead to the proposal of a fast neutron dosimeter with many interesting and convenient features.⁷⁻⁹ Attempts were also made to develop instrumentation for large-scale dosimetry.

Electrochemical etching parameters were studied¹⁰⁻¹³ over a wide range of applied voltage, frequency, polymer thickness, etchant composition, concentration and temperature, and distance between the electrodes. In the course of these investigations, many interesting observations were made. Figures 1-6 show the micrographs of fast-neutron-induced recoil particle tracks in polycarbonate foils under different electrochemical etching conditions. Figure 7 shows a direct photograph of recoil particle tracks in polycarbonate foils which shows the simplicity of counting the tracks. Figure 8 shows a fission fragment track developed by a combination of conventional and electrochemical etching methods. Figure 9 shows conventionally etched recoil particle tracks.

As stated above, the ECE method depends on many parameters which can be varied to obtain different amplification results.¹⁰⁻¹³ These features will make the method very interesting for a number of applications as well as for basic research. The neutron dosimetry based on this method was applied to neutron depth dose studies inside heterogeneous phantoms, specifically at bone-tissue interfaces, to neutron radiography, and to neutron dosimetry inside high energy x-ray beams.^{10,14,15} The etching systems to etch one foil,⁶ six foils,⁷⁻⁹ and 34 foils were also developed to demonstrate the method's feasibility for large-scale neutron dosimetry applications.¹⁴

Due to many advantageous characteristics of such dosimetry methods and supporting data and comments from some investigators, it seems logical to state that this dosimetry method might lead to a widely used neutron dosimeter of the near future, at least for some applications, and become a powerful tool for basic research.

Many laboratories now seem to be engaged in research and development on the basic mechanisms and applications of ECE methods for charged particle track registrations. The Georgia Institute of Technology is continuing further development and application under the direction of Professor Karl Z. Morgan.^{16,17} There are also other investigations directed by R. B. Gammage

FIGURE 1.

FIGURE 2.

FIGURE 3.

FIGURE 4.

FIGURE 5.

FIGURE 6.

FIGURE 7.

FIGURE 8.

FIGURE 9.

at Oak Ridge National Laboratory,^{18,19} by Tommasino at CNEN, Italy,²⁰ by R. V. Griffith at Lawrence Livermore Laboratory,²¹ by G. Somogi at the Institute of Nuclear Research in Hungary,²² and by the author at the Atomic Energy Organization of Iran, and probably many other investigators intend to establish or have already established such a neutron track etching program.

It was not the purpose of this summary to give any data on the subject. For any technical information and details on the technique and the figures, see reference 10.

References

1. Young, D. A., "Etching of Radiation Damage in Lithium Fluoride," *Nature*, 182, 375 (1958).
2. Becker, K., *Solid State Dosimetry*, CRC Press, Cleveland, Ohio (1973).
3. Griffith, R. V., "Particle Registration in Dielectric Materials, A Bibliography," USAEC, Cont. No. W-7405-Eng-48, Lawrence Livermore Lab. (1974).
4. Fleischer, R. L. Price, P. B., and Walker, R. M., *Nuclear Tracks in Solids*, University of California Press (1975).
5. Tommasino, L., "Electrochemical Etching of Damage Track Detectors by H.V. Pulse and Sinusoidal Waveforms," *Proc. 7th Int. Coll. Corpuscular Photog. Solid Detectors*, Barcellona (1970); See also Tommasino, L., and Armellini, C., *Rad. Effects*, 20, 253 (1973).
6. Sohrabi, M. and Becker, K., "Some Studies on the Application of Track Etching in Fast Neutron Personnel Dosimetry," ORNL-TM-3605, ORNL (1971).
7. Sohrabi, M., "The Amplification of Recoil Particle Tracks in Polymers and Its Application in Fast Neutron Personnel Dosimetry," *Health Physics*, 27, 598 (1974).
8. Sohrabi, M., and Morgan, K. Z., "Recent Developments in Fast Neutron Personnel Dosimetry Using Track Etch Methods," *Proc. 3rd Eur. Cong. of IRPA*, Paper 14, Amsterdam, Netherlands (1975).
9. Sohrabi, M., and Morgan, K. Z., "Some New Characteristics of a Polycarbonate Fast Neutron Personnel Dosimeter," Paper 186, *American Industrial Hygiene Conference*, Atlanta, Georgia (1976).

10. Sohrabi, M., "Electrochemical Etching Amplification of Low-LET Recoil Particle Tracks in Polymers for Fast Neutron Dosimetry," Ph.D. Thesis, Georgia Institute of Technology (1975).
11. Morgan, K. Z., and Sohrabi, M., Progress Report, ORO-4814-1, USERDA Cont. No. AT-(40-1)-4818 (1975).
12. Sohrabi, M., and Morgan, K. Z., "The Effect of Etchant Parameters on the Electrochemical Etching of Recoil Particle Tracks in Polycarbonate," Paper IK-1, 21st Annual Meeting, Health Physics Society, San Francisco (1976).
13. Sohrabi, M., and Morgan, K. Z., "Effect of Polycarbonate Foil Thickness on Electrochemical Etching Amplification of Recoil Particle Tracks for Fast Neutron Dosimetry," Proc. 9th Midyear Symposium, Health Phys. Society, Denver, Colorado (1976).
14. Sohrabi, M., and Morgan, K. Z., A Progress Report, ORO-4814-9, USERDA Cont. No. At-(40-1)-4818.
15. Sohrabi, M., and Morgan, K. Z., "Neutron Dosimetry in High Energy X-Ray Beams Produced in Medical Accelerators," Paper 152, 22d Annual Meeting, Health Physics Society, Atlanta, Ga. (1977).
16. Su, S. J., Stillwagon, G. B., and Morgan, K. Z., "Neutron Dosimetry Using Electrochemical Etching," paper 21, 22d Annual Meeting, Health Physics Society, Atlanta, Ga. (1977).
17. Stillwagon, G. B., Su, S. J., and Morgan, K. Z., "Adapting the Polycarbonate Detector and Electrochemical Etching to the Microdosimetry of Pu-239 in Bone," paper 22, 22d Annual Meeting, Health Physics Society, Atlanta, Ga. (1977).
18. Ashraf, J., "Optimization of Electrochemical Etching Parameters for Alpha Particle Registration in Polymers and Application to Radon Monitoring," M. Sc. Thesis, University of Tennessee (1976).
19. Johnson, S. A., "Evaluation of Polymers for Fast Neutron Personnel Dosimetry By Electrochemical Etching," M.Sc. Thesis, Purdue University (1977).
20. Tommasino, L., Personal Communication (1977).
21. Griffith, R. V., and Fisher, J. C., "Light Particle Registration for Fast Neutron Personnel Dosimetry," Paper IK-2, 21st Annual Meeting, Health Physics Society, San Francisco (1976).
22. Somogi, G., "Processing of Plastic Track Detectors," Nuclear Track Detection, 1, 3 (1977).

III.

STANDARDS AND CALIBRATIONS

PERSONNEL NEUTRON DOSIMETER CALIBRATION FACILITIES
AT THE NATIONAL BUREAU OF STANDARDS

Robert B. Schwartz
U.S. National Bureau of Standards
Washington, DC 20234

Several neutron fields suitable for personnel dosimeter calibration are now available at the National Bureau of Standards. Two of the fields are continuous neutron spectra: the spontaneous fission neutron distribution from a high-intensity ^{252}Cf source, and a thermal Maxwellian beam with a very high cadmium ratio. There are also monoenergetic reactor beams with energies of 2, 24, and 144 keV, produced by filter techniques. All of these facilities are discussed in some detail in the User's Manual¹ and the monoenergetic reactor beams were further discussed at a recent conference.² Since copies of this Manual are available for distribution, these facilities will not be further discussed, but the filtered beam characteristics are listed in the Table. It should be pointed out that while the data are believed accurate at this writing, both the quality of the neutron fields and the accuracy of our measurements are being improved. Hence, some of the numbers may change somewhat with time, but we do not anticipate any large changes.

More recently, monoenergetic neutron fields in the range 100 keV to 1.5 MeV have become available at the NBS Van de Graaff Laboratory.³ Typical neutron fluxes now available are $\sim 10^3 \text{ n}/(\text{cm}^2\text{-sec})$ (corresponding to a dose equivalent of the order of 100 millirem/hr.), in a 20 cm diam. beam. While these fluxes are from one to two orders of magnitude lower than those from the reactor filtered beams, the beam area at the Van de Graaff is larger, so that the products of flux times area are comparable for the two cases. The measured gamma background at the Van de Graaff is $\sim 1.5 \text{ mR/hr}$, and the low energy ($E < 1 \text{ keV}$) neutron background flux is $< 1 \%$ of the main flux.

A 14 MeV neutron beam is planned for later this year. The expected flux is $\sim 6 \times 10^3 \text{ n}/(\text{cm}^2\text{-sec})$, corresponding to a dose equivalent of 1.4 rem/hr.

TABLE 1. Characteristics of NBS Filtered Beams

Major Energy	<u>2 keV</u>	<u>25 keV</u>	<u>144 keV</u>
Filter Material	Scandium	Iron- Aluminum	Silicon
Beam Diameter, FWHM			
at <u>225</u> cm from Reactor, cm	4.25	6	4.5
Average Beam Intensity, at 225 cm			
from Reactor, n/cm ² -sec	3.9×10^4	3.6×10^4	1.4×10^5
Beam Dose Rate in Soft Tissue,			
mrad/hr.	68	45	410
Average Dose Equivalent Rate, mrem/hr*	140	150	3,200
Gamma Background, mR/hr	4	3	4
[†] Neutron "Background," mrem/hr*	6	8	<30

*Dose equivalent interpretation is based on NCRP Report #38, p. 16.

[†]These numbers refer to the dose due to contaminant neutrons in the beam; i.e., due to neutrons of energies other than the major energy in each beam. In addition, there is a "room" background of 2-3 mrem/hr, but this number may vary as different beam ports are opened or closed.

References

- 1) User's Manual - Standard Neutron Fields at NBS, J. E. Hoy and R. B. Schwartz, Dec. 8, 1976.
- 2) R. B. Schwartz, "Calibration and Use of Filtered Beams," Proc. Int., Specialists Symp. on Neutron Standards and Applications. NBS, Gaithersburg, MD, March 28-31, 1977 (to be published).
- 3) O. A. Wasson, "NBS Facilities for Standardization of Neutron Dosimetry from 0.001 to 14 MeV," loc. cit.

DETECTOR RESPONSE DETERMINATIONS WITH
NON-MONOENERGETIC NEUTRON SOURCES

Ferenc Hajnal
U.S. Energy Research and Development Administration
Health and Safety Laboratory*
376 Hudson Street
New York, NY 10014

The calibrations of the albedo neutron dosimeters and the apparent discrepancies between some of the experimentally measured and calculated energy response values¹ suggest that more study might be necessary to determine the true response functions. The problem is complicated by the lack of large monoenergetic neutron beams for the energy response function determinations needed for the albedo dosimeter calibrations. For example, the small sizes of neutron beams require one to scan a comparatively small area of the phantom.

The suggestion is made that the inverse of the spectral unfolding is applicable to detectors, provided that neutron sources with known energy spectral distribution and intensities are available to perform the necessary measurements. Accelerator-produced neutrons from proton and deuteron bombardment of thin lithium and beryllium targets, fission neutrons, CH₂ moderated fission neutrons, and nearly monoenergetic, filtered neutron beams can provide a wide variety of known neutron energy spectra with fairly accurately known intensities.

The detector response, $R(E)$, can be determined from the known energy spectra, $\phi_i(E)$, and the measured quantity, M_i , by solving the homogeneous Fredholm equations,

$$M_i = \int_0^{\infty} R(E) \phi_i(E) dE \quad (1)$$

for $R(E)$ from $i = 1, \dots, N$ measurements.

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* U.S. Department of Energy, Environmental Measurements Laboratory
(after October 1, 1977).

The energy response unfolding of the detectors is identical to the problem of spectral unfolding, and the same procedures and codes may be used. The calculated response of albedo neutron dosimeters² can be used as a trial function for solving Equation 1 via iterative techniques.

The success of determining a response function $R(E)$ would depend on:

- a. the availability of a sufficient number of different spectra $\phi_1(E), \dots, \phi_N(E)$,
- b. the extent by which the N spectra differ, and
- c. the accurate knowledge of each energy spectrum over the entire energy range.

The above method would be very useful to determine the response function of complex detectors whose response can be calculated only approximately or with great simplifications, such as the precision long counter and the Andersson-Braun counter.

The author wishes to thank James E. McLaughlin and Keran O'Brien for helpful discussions and their encouragement.

References

1. T. E. Hoy and R. M. Hall, Response of Savannah River Thermoluminescent Neutron Dosimeters to Monoenergetic Neutrons. Fifth ERDA Workshop on Personnel Neutron Dosimetry, May 6 - 7, 1975, Washington, D. C., BNWL-1934 (1975).
2. R. G. Alsmiller Jr. and T. Barish, The Calculated Response of Albedo - Neutron Dosimeters to Neutrons with Energies ≤ 400 MeV. USAEC Report, ORNL-TM-3984, Oak Ridge, Tenn. (1972).

NRC PROGRAM FOR PERFORMANCE TESTING
OF PERSONNEL DOSIMETERS

Robert E. Alexander
Office of Standards Development
Nuclear Regulatory Commission
Washington, DC 20555

Summary

Perhaps the best way to begin this talk is to say that Uncle Sam is unhappy with the performance of some personnel dosimetry processors. Performance testing conducted by Battelle Northwest Laboratories in 1967 and again about two years ago revealed that while some processors are reasonably accurate in their work the performance of others is very poor.

The NRC staff, working with ERDA, BRH and the NBS, has developed a program plan intended to correct this undesirable situation. The staff plans to propose to the Commission that 10 CFR Part 20 be amended to state that the only personnel dosimetry results which would be acceptable to the NRC would be those provided by processors who have successfully passed certain performance criteria. Development of the criteria has been one of the major problems in the implementation of this plan. Criteria have been published in the past by the National Sanitation Foundation and by ANSI. However, these criteria have not proven satisfactory. Therefore, the Health Physics Society Standards Committee (HPSSC) was requested to form a working group to develop new criteria which would be practical and yet provide an adequate test for accuracy. This working group has developed a draft, which is now available in the NRC Public Document Room, the content of which I will discuss presently.

The program plan called for the conduct of a public meeting on this subject; the meeting was held late in 1976. At the public meeting several individuals stated their belief that prior to the issuance of the new regulation in Part 20 a pilot study should be conducted which would (1) give processors an opportunity to discover and correct any problems, and (2) allow for a thorough test of the new performance criteria. The NRC staff was receptive to this suggestion and is currently in the process of selecting a laboratory

for a two-year pilot study. A third purpose added by the staff for this study is to develop a detailed procedures manual for use by subsequent testing laboratories. This manual would also be distributed to all processors so that they could see for themselves exactly how the irradiations are to be conducted. We expect to publish the proposed rule in the Federal Register for comment during the pilot test phase. If everything goes well, we would then expect the rule to become effective about one year after the pilot study.

With regard to the new HPSSC criteria, let me first provide the necessary definitions:

P_i is the performance index for the i th dosimeter.

H_I is the assigned dose equivalent index.

H'_i is the reported dose equivalent index.

B is the bias.

$\bar{P}$ is the average fractional difference.

$\bar{H}_I$ is the average assigned dose equivalent index.

P_i is defined as follows:

$$P_i \equiv \left[\left(H'_I \right)_i - \left(H_I \right)_i \right] / \left(H_I \right)_i$$

B is defined as:

$$B \equiv \bar{P} \equiv \frac{1}{n} \sum_{i=1}^n P_i$$

The standard deviation, S , is given by:

$$S = \left[\sum_{i=1}^n \frac{(P_i - \bar{P})^2}{n - 1} \right]^{1/2}$$

With these definitions, the criterion may be stated as:

$$|B| + 2S \leq L$$

Thus the magnitude of the bias plus two standard deviations must not exceed the limit L . The standard provides one value of L for photons (Category I)

and another value for all other types of radiation considered (Categories II through VIII). For Category I, photons with energy between 300 keV and 3 MeV, the value of L is 0.3, or $6/\sqrt{\bar{H}_I}$, whichever is larger. Note that if $\bar{H}_I$ is 400 mrem, $6/\sqrt{\bar{H}_I}$ is equal to 0.3. Thus where $\bar{H}_I$ is less than 400 mrem the criterion is less restrictive. For Categories II through VIII, the value for L is 0.5 or $15/\sqrt{\bar{H}_I}$, whichever is larger. Note that if $\bar{H}_I$ is 900 mrem the value of $15/\sqrt{\bar{H}_I}$ is 0.5. Thus for doses of less than 900 mrem the criterion is less restrictive for these categories. Categories II through VIII are defined in the standard as follows:

- II Photons, $30 \text{ keV} < \bar{E} \leq 300 \text{ keV}$
- III Photons, $15 \text{ keV} < \bar{E} \leq 30 \text{ keV}$
- IV Beta Particles
- V Fast Neutrons
- VI Photon Mixtures
(any combination of Categories I and II)
- VII Mixtures, Photons and Beta Particles
(combination of Category I or II, and Category IV)
- VIII Mixtures, Photons and Fast Neutrons
(combination of Categories I and V)

A processor's results would be acceptable for any category which had been passed and unacceptable for any category which had been failed. The number of tests per year which would have to be passed is a problem still under consideration by the NRC staff.

IV.

OTHER TOPICS

REPORT ON NEUTRON DOSIMETRY WORK BEING
PERFORMED IN EUROPE

Dale E. Hankins
Lawrence Livermore Laboratory
Livermore, CA 94550

I recently attended the International Radiation Protection Association meeting in Paris, France, April 24-29, 1977; prior to and after this meeting I included side trips to determine the status of personnel neutron dosimetry work being performed in Europe. This report includes comments on the trips to each of the laboratories visited which does not have a representative at this workshop. The status of neutron dosimetry programs in use or being considered at these laboratories is given and the development work on neutron instruments and personnel neutron dosimeters being performed or planned at each laboratory is described.

A large effort in neutron dosimetry studies was found in France and England. Most development work in personnel neutron dosimeters is on albedo neutron dosimeters but some work is being done on ^{237}Np and thorium fission track etch, pocket alarm meters, and recoil proton detectors.

Visits were made to the following laboratories:

1. CEGB Laboratory at Berkeley, England on April 21; contact J. R. Harvey.
2. Harwell Laboratories in England, April 22; contacts T. O. Marshall and R. T. Brunskill from Windscale and Calder Works in England.
3. Fontenay aux Roses Laboratory near Paris, France, April 26; contact G. Portal.
4. Centre d'Etudes Nucleaires de Cadarache, France, May 2; contact M. Bricka.
5. Kernforschungszentrum, Karlsruhe, Germany, May 4; contact E. Piesch.
6. Physikolisch Technesche, Braunschweig, Germany, May 6; contact S. Wagoner.
7. Studsvik AB Atomenergi, Sweden, May 11; contact Carl O. Wide11.

At the Central Electricity Generating Board, Berkeley Nuclear Laboratories at Berkeley, England, my contacts were J. R. Harvey and I. M. G. Thompson.

Thompson is presently working on establishing standards for the calibration of neutron instruments and dosimeters, and he is also working on evaluation of all types of health physics survey instruments. This is an extensive program involving evaluation of the instrument's performance, including radiological electrical and environmental, as well as an examination of its design features. In one of the reports they have completed they recommend the Studsvik Remmeter for neutron monitoring.

John Harvey is attending this workshop and his work is reported elsewhere in these proceedings.

At the Harwell Laboratories I visited with T. O. Marshall, Kevin Harrison, John A. Douglas and visiting from the Windscale and Calder Works were R. T. Brunskill and F. S. W. Hwang.

They are developing a ^{237}Np personnel neutron dosimeter at Harwell. It contains considerable Np compared to other Neptunium dosimeters. The beta dose rate from the front of the dosimeter is ~ 100 mR/h with the cover removed. They have lead around the Np foil to reduce the dose rate to the wearer to 0.5 mR/h and at the top surface to 2.5 mR/h. They will use standard etching and spark counting (1000 volts prespark and 700 volts for readout). They intend to use only one readout spark and will detect non-counting foils by their lack of background, which will be a few tracks each month. The sensitivity is higher than other systems because they use larger quantities of Np which will give them a sensitivity of "100's of counts" per rem. The Np is obtained from ORNL and is plated on a disk with a 2 mg protective coating.

They have developed for use at the Dunray reprocessing facility, a neutron dosimetry system consisting of two of the Harvey type albedo neutron dosimeters (one on the front and the other on the back), the Np track etch detector discussed above and a neutron alarm meter. All these items are worn on a harness along with the regular TLD or film badge.

The neutron alarm meter was quite large, about 8" long, 3" wide and 1-1/4" thick. It was also surprisingly heavy (590 g). The unit consisted of a small recoil proton proportional counter and is sensitive to neutrons above 10 keV in energy. It has 3% nitrogen to give a response to thermal and intermediate energy neutrons. The alarm is set for 20 and 100 mrem. When these are reached,

an audible alarm would sound. As yet, they have only one prototype of the meter. The dose accumulated in the meter can be determined on a reader and recorded on the individual's exposure records.

Brunskill reported on his albedo neutron dosimeter, which consists of two 1/4" thick, 3/4" wide and 3" long slabs of borated polyethylene obtained from a commercial vendor in the U.S. Mounted on these disks are teflon washers. Inside these washers are graphite on which is located a TLD of lithium borate. The TLDs are read by heating the graphite with R.F. The teflon insulates the graphite from the polyethylene. Heating the graphite also heats the TLD and releases the stored energy. Brunskill uses TLDs below and above boron and his interpretation of the results is similar to the technique developed independently by Tom Crites of LLL.

I received a copy of an unpublished report from Marshall, which compares the responses of five types of albedo neutron dosimeters. As expected, they all have the same energy dependence except at thermal energies. The Burger dosimeter had the highest sensitivity. The lowest sensitivity was from the Preston dosimeter which is designed with Cd on both sides to avoid orientation problems of the dosimeter on a person. This dosimeter is similar to the albedo dosimeter designed by me and in use at LLL.

To determine the calibration factor for albedo neutron dosimeters they use the results from an A-B remmeter and a smaller (Basson) neutron detector. This is similar to the procedure we use with the 3-and 9-in. spheres.

They are investigating the possible use of a single TLD for thermal neutron and gamma measurements. This is done by reading the TLD twice or possibly reading it slowly and recording the light emission peaks. The thermal neutron response of the TLD occurs in the highest energy trap and can be separated from the gamma dose.

They have made measurements of the response of Bonner spheres, albedo dosimeters, ^{237}Np fission foil dosimeters and remmeters to the low energy neutron source developed at Berkeley. They have prepared a report of this work. Generally, the results are about as expected, except that the ^{237}Np fission fragment dosimeters have a response in the intermediate region very close to

the dose response at higher neutron energies. Usually the response of ^{237}Np is used to measure dose in Rads. The response to low energy neutrons has been considered to be negligible but when the neutron dose is measured in Rem the response at intermediate energies becomes significant and the dosimeters respond to all energy neutrons with approximately the same sensitivity.

England still uses film badges as the official dosimeter. There are some TLD systems in use at various installations in England, but these are used only to supplement the film where it is felt that the TLD may give more accurate results. NTA film is still used for neutron dosimetry but there is considerable interest in developing albedo and other types of neutron dosimeters. They are working very hard on these programs.

My prime contact at Cadarache, France, was Dr. M. Bricka. I had discussions with Messrs. Mourgues, Buxerolle and Benezech also.

At Cadarache, I found considerable interest in neutron monitoring instrumentation, most of which was based on Bonner sphere type instruments. They have designed a small (~ 5 in. diameter) sphere with ^3He detector for portable survey work at their power reactors. They have found that although such an instrument is very energy dependent, at reactor facilities the leakage spectrum is fairly constant and an instrument of this type can be used very accurately.

For the Bonner sphere work, they initially changed from LiI crystals to BF_3 tubes but now are again changing to very small ^3He tubes with a small, sensitive volume (1 cm diam x 1 cm long). The ^3He tubes have 8 bars of pressure and are made by L.M.T. of Paris. The ^3He counter increases the sensitivity of the instruments from 16% for LiI up to 36% for the ^3He . They place 2 mm of lead around the ^3He tube to decrease the gamma sensitivity.

For Bonner sphere work they have 9 spheres (2, 2.5, 3, 4.2, 5, 6, 8, 10, 12-in. diam) but for routine work usually use only 3 spheres (2.5, 4.2 and 10-in. diam). The 2.5 sphere is used with and without Cd as is the bare probe. Bricka has a computer program for an H.p.-65 calculator which gives the spectrum. This appears to be a useful and reasonably accurate method of interpreting the sphere results.

They are also studying the use of a modified 3-in. Bonner sphere. The sphere is covered with ^{10}B , which changes its energy dependence. The modified sphere's response may be useful in determining more accurately the neutron spectrum from Bonner spheres, especially around the MeV region. This work has just begun and they do not have results as yet.

They have developed a neutron detector system consisting of 4 spheres and dysprosium foils as passive detectors. It consists of a 10" sphere with legs and a 4.2" sphere mounted on a rod above the 10" sphere; supported above this on each end of a rod are two 2.5 in. spheres, one Cd-covered. They place this combination of spheres in a location at a reactor. It is recovered later and the foils counted. The sensitivity is reasonably high; in a 100 mrem/h field the dosimeter can be left for ~15 minutes and reasonable counts from the dysprosium foils are obtained.

The group at Cadarache appears to be working quite actively in neutron dosimetry and instrumentation.

My contact at Karlsruhe, Germany, was Dr. E. Piesch. Also present for most of the discussions was B. Burkhardt. Dr. Piesch has been working with albedo neutron dosimeters for several years. He designed his own dosimeter recently which consisted of borated plastic somewhat similar to the English Harvey dosimeter. The Piesch dosimeter, however, has three sets of TLDs for measuring incidental thermal neutrons, albedo neutrons of thermal energies, and albedo neutrons of intermediate energies. In practice, the dosimeter would not contain six TLDs (3 ^6Li and 3 ^7Li , but would only have one ^7Li TLD since the ^7Li TLD readings normally are the same. The dosimeters are used in pairs, one on the front and one on the back, and are mounted on a lightweight plastic belt with holes cut to assure proper positioning of the dosimeter on the belt.

They have designed an instrument to simulate the response of the three TLD positions in their albedo neutron dosimeters. It consists of three BF tubes (similar to the one in the 9-in. sphere) and they have placed various moderators on the probes. The unit has just been completed and they will begin to study the responses of the detectors soon.

Piesch is interested in doing studies to establish the practical application of electrochemical etch procedures for recoil particle tracks. They are also considering using the thorium and ^{235}U fission track etch dosimeters developed in Switzerland and now available commercially. They have some interest in making a neutron remmeter with higher response in the >7 MeV region. They have considered large spheres with holes drilled in them but have made no studies to date.

The PTB facility in Braunschweig, Germany, has low energy neutron beams similar to the beams at the NBS. My contact was Dr. S. Wagoner and I also visited with Dr. R. Jahr and Dr. W. Alberts. The low energy neutron beam facilities at the PTB do not have the background problem that exists at the NBS and they have erected massive shielding around the exposure areas. The PTB reactor facility is not as crowded as the NBS reactor. They obviously have fewer studies at the reactor and can devote more space for these facilities. They also are required by a new law to provide neutron standards for instrumentation and dosimeters. While I was there they had several study groups from various countries in Europe making exposures during the "workshop."

In comparing my results with some they obtained with similar instruments, we found some disagreement. They found that the instrument results at 2 keV gave higher readings than at 25 keV, which does not agree with my findings. I feel that we need to have an intercomparison of the NBS and PTB beam intensities and the response of an instrument in each of the beams. Both are supposed to be standards but right now there may be some error in one or both of the beams. The intercomparison should be done soon since PTB is being used by numerous laboratories throughout Europe and the NBS facility is also ready for use.

They usually do not scan the dosimeters through the beams and use a number of fixed positions which he calculated overlap properly.

They also have a fast neutron facility which is a large Van de Graaff housed in a very large building to prevent room return. The building size is easily three times as large as the NBS or LLL facilities and should give a very low scattered neutron component at the detector location.

My primary contact at Studsvik, Sweden, was Carl Olof Widell of the Health and Safety section. I also met and talked with Dr. S. Hagsard and L. Svansson. I discussed with Carl Widell the personnel dosimetry program in Sweden. They use a TLD system containing four TLDs (3 of Li borate and 1 of ^7Li fluoride). Two of the Li borate TLDs are used for beta and thermal neutron measurements. The LiF TLD is used to measure the gamma dose. The fourth TLD is normally not read and is kept as an integrating dosimeter for long term exposures. The TLD readers use hot nitrogen gas to heat the TLDs. Neutron dosimetry for fast neutrons is performed at present by NTA film. Carl indicated they have a serious fading problem with the film and are buying fission track dosimeters developed in Switzerland and available from a commercial supplier in Germany called Geha Heist. They contain ^{235}U in an aluminum disk and thorium as the fissile materials. The ^{235}U is for thermal neutron detection and the thorium for fast neutrons. The detection threshold for thorium is slightly above that for film but Carl indicated the advantage of these dosimeters is that they do not fade. There is some exposure of the wearer from the badge materials; to keep this small, lead is placed on both sides of the thorium, with thicker lead on the wearer's side. The dosimeters must be properly oriented to keep the exposure low and are worn on a belt or with a pocket clip.

In place of their nuclear accident dosimeters, which presently contain bare and Cd-covered gold and phosphorous, they are going to substitute a silicon diode which is sensitive to neutrons. These dosimeters have a fairly strong energy dependence, dropping rapidly below 1.5 MeV, but for accident dosimeters they feel the dosimeters are satisfactory.

Carl indicated that the neutron exposures at Studsvik are low but at some of the accelerators and PWR reactors there are significant neutron exposures. They presently use the TLD badge to measure thermal neutron exposures at the reactors and apply a calibration factor for correction to total neutron dose. This procedure is similar to that used at LLL. Carl said for a given reactor the calibration factor is reasonably constant.

I was shown by Svansson a criticality dosimeter which they had developed at Studsvik. It consisted of 10 of the neutron diodes wired in series. The electronics are designed to detect rapid changes in the resistance of the

diodes. Slow changes are not detected; therefore, it could be used in a neutron field and still only detect a criticality accident. The diodes are almost completely insensitive to gamma rays. They used the detector for two years with no false alarms. Unfortunately, the need for this type of detector at Studsvik no longer exists and they are offering the detector on loan to anyone who may have an interest.

I was shown the BF_3 tubes which they are using in the new Studsvik remmeter. This BF_3 tube has been modified by placing the wire and insulator on a small projection on the front of the tube. This gives a BF_3 tube with the same sensitive volume as the old tube but allows them to place more polyethylene between the sensitive volume and the front of the instrument. The polyethylene moderator is made shorter and still retains the same directional dependence. This reduces the weight of the instrument.

They are considering making BF_3 tubes with a built-in source and shutter to permit them to check the operation of the BF_3 tube. This is done on some proportional counters but to my knowledge has not been done with BF_3 tubes before.

Other items of interest which should be reported are as follows.

Dr. Max Heinzelmann at Julich, Germany, is working on an active albedo neutron dosimeter using a ^3He detector and 2-in. polyethylene moderator hemisphere. The detector is worn on the body, and by carefully adjusting the high voltage, a relatively energy-independent response is obtained. Unfortunately, this requires very stable electronics. A copy of this paper is included in the proceedings of the IRPA meeting, pp. 143-146.

Dr. M. Hofert from CERN indicated they were doing some development work on electrochemical etching of recoil particles and some investigations with thorium fission fragment foils and albedo neutron dosimeters (Piesch type).

Dr. C. Burger of Munich, Germany, has developed an albedo neutron dosimeter containing a small hemisphere of polyethylene. It is about half the size of the Hoy dosimeter and has a sensitivity between that of the simple dosimeter and the Hoy dosimeter.

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GERMANY

J. David
Gesellschaft für Strahlen- und
Umweltforschung mbH
Ingolstädter Landstr. 1
8042 Neuherberg bei München
GERMANY

10 Meadi Sohrabi
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