NOTICE

This report was prepared as an account of work sponsored by the United States Government. Neither the United States nor the United States Atomic Energy Commission, nor any of their employees, nor any of their contractors, subcontractors, or their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness or usefulness of any information, apparatus, product or process disclosed, or represents that its use would not infringe privately owned rights.

DEVELOPMENTS IN NEUTRON DOSIMETRY FOR RADIATION PROTECTION***J. A. Auxier**

**Health Physics Division
Oak Ridge National Laboratory
Oak Ridge, Tennessee 37830**

Abstract

The information which is presented here is representative of that developed in the ORNL Health Physics Division's Radiation Dosimetry program, though it includes information from several laboratories. It is the primary objective of the ORNL effort to develop detector systems and techniques for characterizing radiation fields in which man has been, or may be, exposed. More specifically, a majority of the individual programs are directed toward detecting neutrons and describing the many ways in which neutrons interact and deposit energy in living matter. The spectrum of activities includes detector development (especially in the area of solid-state devices), the establishment of monitoring systems in accordance with criteria from international commissions and regulatory bodies, development of generalized concepts of radiation protection, an extensive program of calculating and measuring dose distributions in phantoms, and the formulation of national and international studies for the intercomparison of dosimetry systems.

* Research sponsored by the U.S. Atomic Energy Commission under contract with the Union Carbide Corporation.

MASTER

Reg

INTRODUCTION

Although there has been steady improvements in neutron dosimeters and detectors since the introduction of the nuclear reactor, the need for better instruments and techniques has increased faster than the developments. Most of the instruments and techniques for neutron dosimetry have been developed to meet specific needs. Although this has also been the case at the Oak Ridge National Laboratory (ORNL), and will continue to be the case to a limited extent, the chief effort at ORNL is now aimed at greater generalization of dosimetry. Of central importance to this philosophy is the recognition that it is seldom, if ever, possible to measure all of the various parameters necessary to characterize the radiation field adequately, especially within the human body. Therefore, the research is concentrated in two areas: (1) Development of detector systems for measuring key parameters of the field, e.g., the energy spectrum, and the application of these systems; and (2) a comprehensive calculational program for producing a complete description of the radiation field, including depth-dose and LET distributions, which can be checked experimentally at only one or a few steps. For example, including in the calculational description of depth dose a computer printout of an average value proton recoil dose permits an indirect experimental "check" of the dose distribution by means of a simple system such as a proton recoil proportional counter. This system requires rather extensive use of large digital computers.

In addition to the research in the areas of instruments and techniques for dosimetry, there is a major effort in the development of theoretical principles for radiation protection and in providing an internationally available intercomparison facility for dosimetry. The continuing program of intercomparisons has been important in the improvement in neutron dosimeters, especially for nuclear accidents, largely due to the spirit of cooperation and involvement by the many participants from many countries.

STATIONARY AND PORTABLE ELECTRONIC DEVICES

One of the most recent developments in neutron dosimetry involves an "active" dosimeter, based on electronic signal processing and a new scintillator. The development of an efficient organic scintillator has made possible the design of an LET measuring device using a matrix of small spheres of the scintillator imbedded in a transparent, tissue-equivalent cylinder [1]. The scintillator gives excellent light output as a function of energy deposited by recoil particles, a requirement for measuring sufficiently short path lengths and the resulting low energy losses.

Hydrogen-filled proportional counters have gained considerable acceptance for the measurement of neutron spectra in the energy range of a few keV to one or two MeV. Complicated computer techniques are required to reduce the pulse height spectra to neutron spectra [2,3]. By using

the techniques described by Hurst and Ritchie in their paper on the generalized concept of dosimetry, such complicated data reduction can be bypassed if dose information is all that is required [4]. Because pulse shape discrimination techniques can be used to differentiate between pulses produced in the proportional counter by neutrons and gamma rays, this technique shows considerable promise for measuring neutron dose around reactor shields which is a particularly troublesome combination of neutron spectrum and gamma-ray background. A wider energy range could be covered by a similar detector that used recoils from a heavier gas than hydrogen.

Since knowledge of the neutron spectrum is an excellent basis for determining neutron dose, efforts are being made in several laboratories to develop spectrometry systems that are particularly suited to radiation protection purposes [5]. There are two basic requirements for such applications, sensitivity and ease of interpreting the accumulated data. The need for sensitivity is obvious; the second requirement is a result of the need in radioprotection to provide rapid evaluations of a radiation hazard. In other words, a dosimetrist seldom has the time available to perform the lengthy data analysis that workers in nuclear physics can enjoy. Unfortunately, these two requirements appear to be mutually exclusive. The most sensitive neutron spectrometers are those that measure the entire recoil proton spectrum produced in an organic scintillator by the incident neutrons. These data are complex and require considerable time and sophisticated techniques to reduce [6]. As with the proportional counter, this step might be bypassed and the data converted directly to dose with an operational analyzer. Generally, neutrons below a few hundred keV cannot be measured with these detectors.

A recoil proton telescope will exhibit similar low energy limits and considerably less sensitivity; however, the data recorded will have a better correlation with the neutron spectrum than can be obtained from a single organic scintillator [7]. It is possible to design a telescope for which the recorded pulse height spectrum is within a factor of two of the neutron spectrum over an energy range of 500 keV to 15 MeV.

Thus it appears that at present the most promising applications of electronics to neutron dosimetry are those which utilize spectrometry in its most general sense. As dosimetry concepts become more sophisticated, additional efforts will have to be applied to neutron spectrometry.

ACCURACY CONSIDERATIONS IN FAST NEUTRON PERSONNEL DOSIMETRY

Even a cursory examination of the accuracy requirements for dosimetry for human exposures indicates that some guidelines are needed. The statements which can be found in more or less official recommendations (ICRP 1969, IAEA 1971) are rather vague, suggesting tolerable uncertainties in the ± 25 to 50% range. ICRP Report No. 12 states that "the uncertainty in assessing the annual dose...should not exceed 50% of the recorded dose, or 1 rem, whichever is larger".

MASTER

The relation between the dose reading, which is usually derived from a single dosimeter worn on the front of the trunk and what one would actually like to measure, namely the risk of somatic or genetic radiation damage in an individual is, of course, quite complicated (for a detailed discussion, see Ref. 8). Considering the pronounced drop of neutron dose with increasing depth in the body (Fig. 1), it can be assumed only in extremely rare cases, such as multidirectional incidence of neutrons above 15 to 20 MeV in energy, that the dose indication of a detector at the body surface agrees within $\pm 50\%$ with the gonad, bone marrow, or average (midline) dose. In the vast majority of situations in personnel fast neutron monitoring, the situation is considerably more complicated, and a single dosimeter will, depending on the direction of neutron incidence, seriously overestimate or underestimate the neutron dose. This can be avoided to some extent by either a careful reconstruction and analysis of each case of suspected serious exposure, or by the use of more than one dosimeter, for example, in one of the dosimetry belts which are currently used in many nuclear establishments for certain high-risk employees.

Fortunately, total errors in dose estimates by a factor of 2 or 3 may be tolerated in the range of the occupational exposure guidelines on the assumption that it is likely that such errors will compensate each other during many monitoring periods. Maximum possible accuracy is, however, required in the lethality range for accidental exposure (~ 50 -1000 rad), and the accuracy requirements can be relaxed again in the range of supralethal overexposure.

THE DISTRIBUTION OF DOSE IN MAN

As discussed earlier, considerable attention has been directed to the problem of comparing experimental results with a view to giving increased confidence to the calculated results which extend beyond the range of direct experimental verification. In recent months, an extensive series of measurements of dose patterns have been made in human-sized, tissue-equivalent phantoms (15 cm radius, 60 cm high cylinder) for both a fission source and a 14-MeV source using a recoil proton proportional counter with a lower energy cutoff of ~ 160 keV. Calculations [9] have been made with similar mathematical models (including the 160 keV cutoff) for comparison. Data for the distribution of neutron and (n, γ) dose in the phantom for 14-MeV neutrons is given in Fig. 2. Dose due to recoils from fission neutron reactions is compared with calculated values in Fig. 3. In all cases, it is seen that the agreement is to within the probable errors of measurement and calculation.

Because of medical applications of ^{252}Cf , it has become necessary to determine to within a few percent the local dose distributions around typical implanted "needle" sources. Standard transport techniques [10] were used to calculate the local distribution of KERMA as shown in Fig. 4. Neutron isodose curves are presented as the percent of free space tissue KERMA at 1 mm from the surface at the center of the needle as a function

5

Fig. 1. Neutron Dose Equivalent as a Function of Depth in a Cylindrical Thorax Phantom of 30-cm Diameter.

14 MeV NEUTRON DEPTH DOSE

ORNL-DWG. 72-6666

Fig. 2. Comparison of Measured and Calculated Absorbed Dose Due to Recoil Ions and Neutron-Induced Gamma Rays as a Function of Depth in a 30 cm Diameter Cylindrical Phantom for 14-MeV Neutrons.

HPRR NEUTRON DEPTH DOSE, RECOILS

ORNL-DWG. 72-6668

Fig. 3. Comparison Between Measured and Calculated Absorbed Dose Due to Recoil Ions as a Function of Depth in a 30-cm Diameter Cylindrical Phantom for Fission Neutrons.

NEUTRON ISODOSE CURVES IN TISSUE NORMALIZED TO FREE SPACE KERMA

Fig. 4. Neutron Isodose Curves in Tissue Normalized to Free-Space KERMA for an Implanted 20 ml-long ^{252}Cf Source.

of position. Distributions were calculated for three tissue compositions, and exponential interpolations were done in two dimensions to produce the isodose curves for a 20 mm-long needle. Coefficients of variation ranged from 1 to 3.5% for these data so that the accuracy is adequate for the needs of the radiotherapist.

DOSIMETRY INTERCOMPARISON STUDIES

After the rash of nuclear accidents in the late 50's and early 60's, a program was begun whereby dosimetrists could evaluate the response of their personnel or area monitoring systems when exposed in mixed neutron and gamma-ray fields with previously-determined characteristics. This series of dosimetry intercomparison studies has become an annual event and has been a major success in the sense of helping establish reliable monitoring systems incorporated into nuclear safety programs. The ORNL Health Physics Research Reactor provides fission neutrons for these studies. Detectors are placed at three meters from the reactor for three separate experiments. In one case, there is no shielding between the detectors and the source, and in the other two cases, shields of 13 cm steel and 12 cm plastic are used. Results of these intercomparisons (see Table 1) indicate that yearly variations in reported values for neutron and gamma-ray dose are in the range of 5-18% of the average of all data for the three radiation fields with substantially different neutron-to-gamma ratios. As a result of this program at ORNL, a similar study series sponsored by the IAEA was started in 1970.

AN OPERATIONAL RATIONALE FOR RADIATION PROTECTION

The processes in which radiation interacts with man are so complex that one can accept an operational basis for correlating radiation fields with biological effects. An operational rationale is being sought which is not encumbered by such concepts as "permissible levels" measured in terms of absorbed dose or its modification by a myriad of factors. One may base a formulation on the following premise: If the changes brought about in a biological system are governed by the ordinary laws of physics and probability, then specification of the radiation field during the life of the system determines the probability that changes have occurred or will occur in that system due to radiation.

The radiation field will be defined by specifying every variable which could conceivably affect the biological probabilities. Thus, the radiation field $N_k(E, \phi, \vec{r}, t)$ is defined such that

$$N_k(E, \phi, \vec{r}, t) dE d\phi d^3r dt$$

is the number of particles of type k lying in the energy range between E and $E + dE$ in the solid angle $d\phi$, in the volume element d^3r , and in the time interval dt . The operational premise can be stated in a symbolic form if

ORNL NUCLEAR ACCIDENT DOSIMETRY INTERCOMPARISON EXPERIENCE

ORNL-DWG 72-7180

Date of Pulse	No Shield*				
	No. Fissions $\times 10^{-16}$	Neutron Dose (rad)	D_n /Fission $\times 10^{16}$	Gamma Dose (rad)	D_γ /Fission $\times 10^{16}$
March '65	7.74	377	48.7	52	6.7
October '65	8.6	372	43.1	53	6.2
May '67	7.2	317	44.0	45	6.2
May '67	6.9	304	44.0	48	6.9
December '67	11.7	481	41.1	66.1	5.6
July '68	6.34	256	40.4	48	5.3
July '69	7.52	334	44.4	44	5.8
July '70	7.64	310	40.6	44	5.8
May '71	9.02	376	41.7	56	6.2
Average			43.1		6.1
13 cm Steel					
May '67	7.1	109	15.4	11	1.5
May '67	6.6	109	16.3	11	1.7
July '68	7.11	97	13.6	14	2.0
July '69	7.09	116	16.4	13	1.8
July '70	7.16	106	14.8	11.5	1.6
May '71	9.1	147	16.2	16.6	1.8
Average			15.3		1.7
12 cm Lucite					
July '70	5.15	42	8.2	33.7	6.5
May '71	6.3	47	7.4	46.8	7.4
Average			7.8		7.0

10

one chooses to specify the state of a biological system with respect to effect i in species j through a quantity S_{ij} . Thus, one can write an operator equation

$$G(N(\tau), S_{ij}(t)) = P_{ij}(t > \tau). \quad (1)$$

Here $P_{ij}(t > \tau)$ is the probability as a function of time that effect i will be observed in species j at any future time $t > \tau$ after the radiation $N_k(\tau)$, and G is an operator which yields a transformation of the system from its initial state to the i th biological endpoint (for convenience, we have suppressed the arguments $E, \vec{r}$, etc.).

The idea expressed in Eq. 1 serves only as a logical framework within which a more explicit operational approach may be sought. Two different approaches to the problem are being pursued, each of which offers a degree of conceptual and mathematical beauty without total sacrifice of utilitarian purposefulness.

In the first approach, which might be described as a biological operator formulation, an operational relationship between radiation and effect is expressed by assuming that an operator exists which transforms the radiation field directly into probabilities of observing effects. Thus

$$O_{ij}N_k(E, \phi, \vec{r}, \tau) = P_{ij}(t > \tau), \quad (2)$$

where O_{ij} is an operator which leads to effect i in species j as a result of a radiation field N . The operator obviously would include integration over the multidimensional coordinate space of N . In actual measurements, the readings obtained by a detector in a radiation field would yield a set of numbers which are indirectly related to N , say $R_k(\epsilon, \phi, \vec{r}, \tau)$. Then if the instrument is adequately designed, it is possible to learn the real nature of the field through a transformation of the readings of R . Thus

$$O_d R_k(\epsilon, \phi, \vec{r}, \tau) = N_k(E, \phi, \vec{r}, \tau), \quad (3)$$

where O_d is an operator characteristic of the "detector which allows one to transform the readings into knowledge of the field. Thus, if O_d exists, then from Eq. 2

$$O_{ij}O_d R_k(\epsilon, \phi, \vec{r}, \tau) = O_{ij}N_k(E, \phi, \vec{r}, \tau) = P_{ij}(t > \tau), \quad (4)$$

and one can transform the instrument response directly into a table of probabilities, i.e., unfolding R to obtain N is unnecessary.

A second mathematical expression of the operational relationship implied by Eq. 1 has been prompted by the observation that there are no "radiation diseases". That is, radiation only enhances the rate of incidence of certain maladies which occur naturally with probabilities $B_{ij}(t)$ (i.e., effect i in species j). With this in mind, one is led to an operational relationship between radiation and effect which can be expressed in a form

$$O(N)B_{ij}(t) = P_{ij}(t > \tau) \quad (5)$$

where $O(N)$ is an operator which has some, as yet unspecified, functional dependence on the radiation field N . (P_{ij} is now the total probability of occurrence, including the natural rate; thus if $N = 0$, $O = 1$). The objective here is to "discover" the operators $O(N)$ which change the natural rate of incidence to that which occurs as a result of radiation N . Operators independent of i and j are probably outside the realm of possibility, but if it should happen that $O = O_i$, i.e., the operator is specific to effect i but independent of species, then one could easily extend biological data to humans. This would be tantamount to the discovery of a macroscopic "law" of radiation effects. If one is unable to find operators of this degree of generality, i.e., if both i and j have to be specified, the formalism, though still very useful, would require human data.

Even now, it is clear that research in physical instrumentation should increasingly involve spectroscopic approaches and multiparameter computer storage technology in contrast to simpler devices which can only provide a single parameter indication.

A joint program with the Biology Division of ORNL is under way in which an attempt is being made to find the operator $O(N)$, Eq. 5, for the case of mouse cataract incidence. The data $B_{ij}(t)$ are available, and so are $P_{ij}(t)$ for some radiation fields. Additional experiments will be performed as the effort to find operators suggest them. While the present formalism leads quite naturally to suggestions on new directions for physical instrumentation and on new experimental approaches to radiation biology, the more immediate benefits to be gained from the present investigation come in the nature of provocative and conceptual values.

ACKNOWLEDGMENT

The author acknowledges gratefully the extensive assistance of the following staff members of the Health Physics Division in the preparation of this manuscript: K. Becker, W. L. Chen, F. F. Haywood, G. S. Hurst, T. D. Jones, and J. H. Thorngate.

REFERENCES

1. J. H. Thorngate *et al.*, *Health Physics Ann. Prog. Rept. for Period Ending July 31, 1971*, ORNL-4720, p. 110.
2. P. W. Benjamin *et al.*, AWRE Report NR2/64.
3. E. F. Bennett, *Rev. Sci. Inst.* **11**, 1153-1160 (1962).
4. G. S. Hurst and R. H. Ritchie, *Health Phys.* **8**, 117-129 (1962).
5. For example, Berger at Munich, Thorngate at ORNL.
6. V. V. Verbinski *et al.*, *Nucl. Instr. and Meth.* **65**, 8-25 (1968).
7. J. H. Thorngate and J. A. Auxier, *Proceedings of Colloque d'Electronique Nucleaire et Radioprotection, Vol. I.*, Toulouse, France, March, 1968.
8. K. Becker, *Health Physics*, in press.
9. J. W. Poston *et al.*, *Health Physics Division Annual Progress Report for Period Ending July 31, 1971*, ORNL-4720, p. 84.
10. T. D. Jones and J. A. Auxier, *Phys. Med. Biol.* **17** (2), 206-17 (1972).

CAPTIONS

- Figure 1. Neutron Dose Equivalent as a Function of Depth in a Cylindrical Thorax Phantom of 30-cm Diameter.
- Figure 2. Comparison of Measured and Calculated Absorbed Dose Due to Recoil Ions and Neutron-Induced Gamma Rays as a Function of Depth in a 30-cm Diameter Cylindrical Phantom for 14-MeV Neutrons.
- Figure 3. Comparison Between Measured and Calculated Absorbed Dose Due to Recoil Ions as a Function of Depth in a 30-cm Diameter Cylindrical Phantom for Fission Neutrons.
- Figure 4. Neutron Isodose Curves in Tissue Normalized to Free-Space KERMA for an Implanted 20 ml-long ^{252}Cf Source.