Neutron Monitoring for Radiation Protection Purposes

VOL. I

PROCEEDINGS SERIES

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VOL. I

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FOREWORD

With the rapid increase in the use of neutrons in research, industry and medicine, neutron monitoring for radiological protection continues to be a subject of increasing general interest. The main sources of neutrons are sealed radioisotopic sources, nuclear reactors and particle accelerators, including neutron generators. They are also encountered in the enrichment of fissile material and in the processing of spent fuel elements.

Since neutrons can contribute significantly to biological damage, proper assessment of the organ dose resulting from exposure to neutrons is most important for personnel protection. In practical monitoring, neutrons present special problems. As they are uncharged particles, their detection can be based only on the products of their interaction with matter. They have wide ranges of energies from a few hundredths of an electron volt to several hundred million electron volts. The neutron interaction cross-sections show irregular variation with energy, particularly in the intermediate energy range, where sharp resonance peaks are found. Further, neutrons do not, in general, occur alone; other types of radiation, particularly gamma rays, are also usually present. All these factors give rise to practical difficulties – in the techniques of neutron monitoring, the design of monitoring instruments and the assessment of organ doses resulting from exposure to neutrons alone or to mixed radiation fields. There is therefore a need for both improved and more rapid methods of neutron monitoring and for greater accuracy in dose estimation.

The Symposium on Neutron Monitoring for Radiation Protection Purposes was one of a series of scientific meetings through which the IAEA promotes the exchange of information on all aspects of personnel and area monitoring. It concerned recent developments in neutron measurements, particularly of neutron spectra, instruments and techniques for field and personnel monitoring, comparison of various methods and monitoring systems, and standardization and calibration.

During the course of the Symposium, neutron spectrometry was seen to be a significant development, providing an excellent basis for determining neutron dose. More emphasis was being placed on Bonner sphere techniques, on albedo-neutron dosimeters and on measurements based on counting etch pits produced in plastic detectors. Many of the larger nuclear laboratories lay more stress than formerly on calibration and intercomparison. Although no truly important new approach to dosimetry has been propounded in recent years, some new neutron detectors have been introduced in the past decade.

The Symposium was attended by 132 participants from 30 Member States and 9 international organizations. All the papers, including the invited papers and a summary of the Symposium, are presented here in full, together with the discussions.

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CHAIRMAN: P.N. KRISHNAMOORTHY (India)

DEVELOPMENTS IN NEUTRON DOSIMETRY FOR RADIATION PROTECTION*

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Abstract

DEVELOPMENTS IN NEUTRON DOSIMETRY FOR RADIATION PROTECTION.

The information presented in this paper is representative of that developed in the ORNL Health Physics Division's Radiation Dosimetry programme, although 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 programmes 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 programme of calculating and measuring dose distributions in phantoms, and the formulation of national and international studies for the intercomparison of dosimetry systems.

1. INTRODUCTION

The steady development of improvements in neutron dosimeters and detectors since the introduction of the nuclear reactor has not been able to keep pace with the need for better instruments and techniques. 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 so to a limited extent, the chief effort at the Laboratory 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 programme 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 to be done by means of a simple system such as a proton recoil proportional counter. This system requires extensive use of large digital computers.

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

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

2. STATIONARY AND PORTABLE ELECTRONIC DEVICES

One of the most recent developments in neutron dosimetry concerns 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 incorporating 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 from 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]. Since 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 the 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 in which recoils from a gas heavier than hydrogen are used.

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 (for example, Berger at Munich and Thorngate at ORNL). 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 used by workers in nuclear physics. 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 [5]. As with the proportional counter, this step might be bypassed and the data converted directly to dose with an operational analyser. 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 [6]. 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.

3. 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"

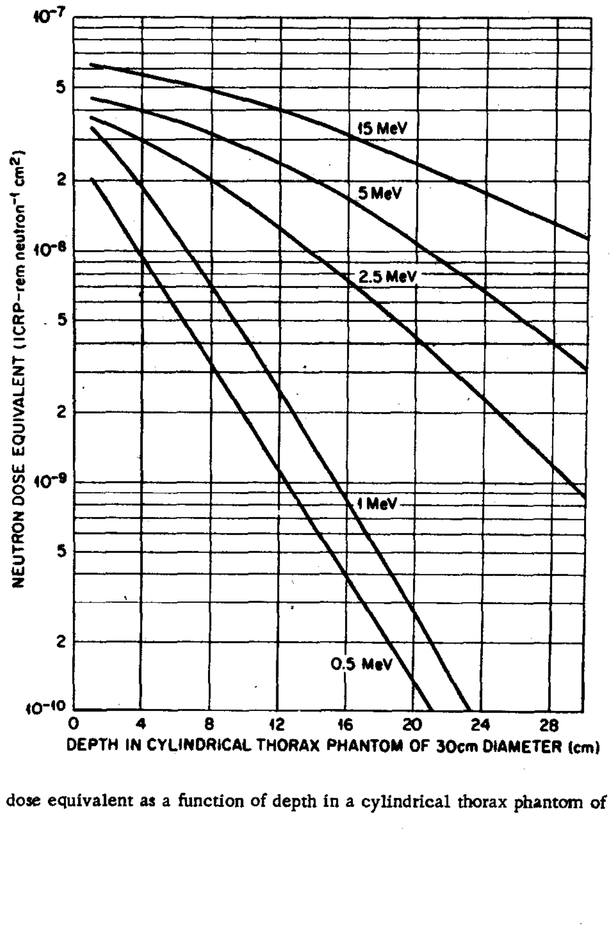

FIG.1. Neutron dose equivalent as a function of depth in a cylindrical thorax phantom of 30-cm diameter.

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. [7]). 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 multi-directional incidence of neutrons above 15 to 20 MeV in energy, that the dose indication of a detector at the body surface agrees within ± 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 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.

4. THE DISTRIBUTION OF DOSE IN MAN

As discussed earlier, considerable attention has been given to the problem of comparing experimental results with a view to increasing confidence in 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 [8] 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 per cent the local dose distributions around typical implanted "needle" sources. Standard transport techniques [9] were used to calculate the local distribution of kerma as shown in Fig. 4. Neutron isodose curves are presented as the per cent of free space tissue kerma at 1 mm from the surface at the centre of the needle as a function 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.

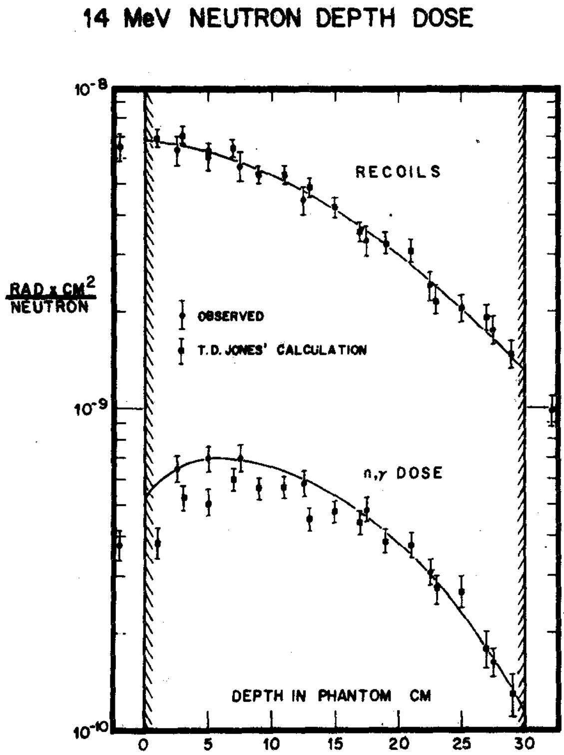

FIG.2. Comparison of measured and calculated absorbed dose due to recoil ions and neutron-induced gamma-rays as a function of depth in a 30-cm-diameter cylindrical phantom for 14-MeV neutrons.

5. DOSIMETRY INTERCOMPARISON STUDIES

After the rash of nuclear accidents in the late nineteen-fifties and early nineteen-sixties, a programme was begun whereby dosimetrists could evaluate the response of their personnel-monitoring 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 helping to establish reliable monitoring systems incorporated into nuclear safety programmes. The ORNL Health Physics Research Reactor provides fission neutrons for these studies. Detectors are placed three metres 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 of steel and 12 cm of plastic are used. Results of these intercomparisons (see Table I) 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 programme at ORNL, a similar study series sponsored by the IAEA was started in 1970.

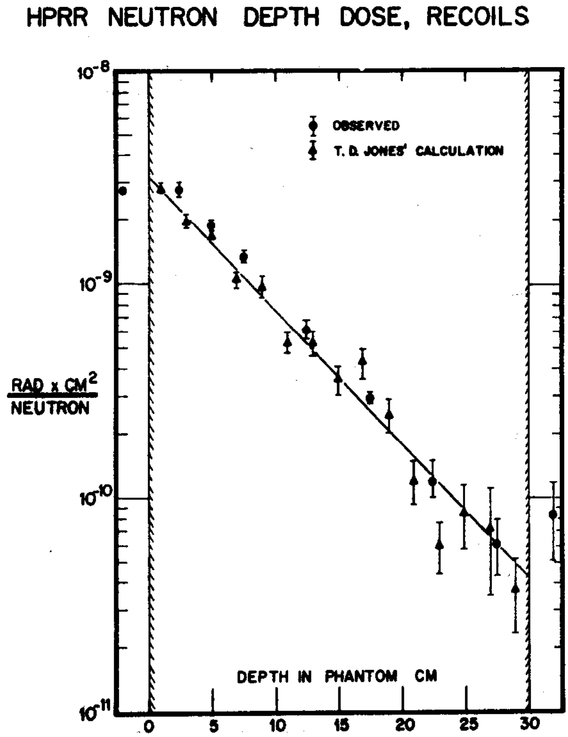

FIG. 3. Comparison between measured and calculated absorbed dose due to recoil ions as a function of depth in a 30-cm-diameter cylindrical phantom for fission neutrons.

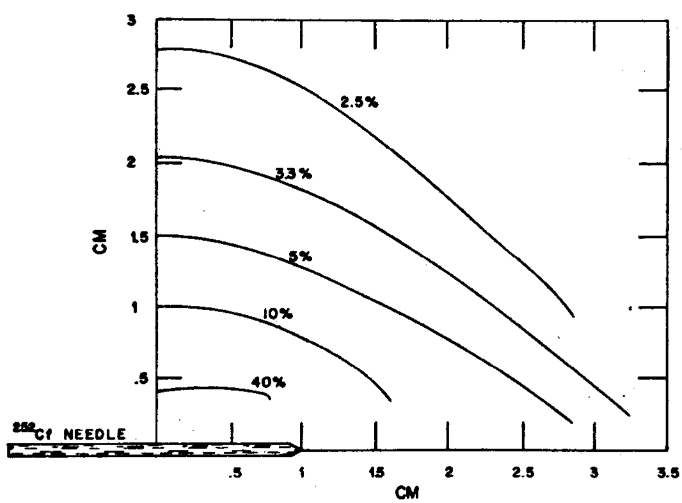

FIG. 4. Neutron isodose curves in tissue normalized to free-space kerma for an implanted 20-mm-long ^{252}Cf source.

6. 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

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

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 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

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

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 ith 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

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

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 multi-dimensional co-ordinate 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

$$O_dR_k(\epsilon, \phi, \vec{r}, \tau) = N_k(E, \phi, r, \tau) \qquad (3)$$

TABLE I. ORNL NUCLEAR ACCIDENT DOSIMETRY INTERCOMPARISON EXPERIENCE

Date of pulse	No. fissions x 10^{-16}	Neutron dose (rad)	D_n/fission x 10^{16}	Gamma dose (rad)	D_γ/fission x 10^{16}
No shield					
March 1965	7.74	377	48.7	52	6.7
October 1965	8.6	371	43.1	53	6.2
May 1967	7.2	317	44.0	45	6.2
May 1967	6.9	304	44.0	48	6.9
December 1967	11.7	481	41.1	66.1	5.6
July 1968	6.34	256	40.4	48	5.3
July 1969	7.52	334	44.4	44	5.8
July 1970	7.64	310	40.6	44	5.8
May 1971	9.02	376	41.7	56	6.2
Average			43.1		6.1
13 cm steel					
May 1967	7.1	109	15.4	11	1.5
May 1967	6.6	109	16.5	11	1.7
July 1968	7.11	97	13.6	14	2.0
July 1969	7.09	116	16.4	13	1.8
July 1970	7.16	106	14.8	11.5	1.6
May 1971	9.1	147	16.2	16.6	1.8
Average			15.5		1.7
12 cm Lucite					
July 1970	5.15	42	8.2	33.7	6.5
May 1971	6.3	47	7.4	46.8	7.4
Average			7.8		7.0