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DOSIMETRY, SHIELDING AND SCATTERING II*

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J. H. Thorngate
Health Physics Division
Oak Ridge National Laboratory
Oak Ridge, Tennessee

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Introduction

The preceding paper⁽¹⁾ gave a summary of dosimetry and some of the units used. It also gave an insight into what must be considered the central problem of dosimetry and radiation protection, that is, the correlation of observed biological effects with measured quantities. Surprising as it may seem, only recently has this begun to receive its proper share of study by health physicists. Because some of the current approaches for the solution of this problem have already been outlined, this discussion will cover other work, both past and present, being done in dosimetry. Some is related closely to basic radiation protection, and other is more typical of what research in dosimetry has generally been in the past, that is, research and development of dosimeters and related dosimetric systems. Typical research problems encountered are nuclear accidents, radiobiological and radiochemical experiments, and experiments to measure depth dose: a term used to describe dose as a function of position in an animal or its phantom equivalent.

Electroscopes and ion chambers⁽²⁾ were the first instruments used to measure ionizing radiation quantitatively. These instruments were sufficient

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when the sole radiation to be measured was x rays or radium gamma rays. In 1942, these needs were expanded drastically by the advent of nuclear reactors which are abundant sources of neutrons. Early solutions to this added complexity in the problem were simple and straightforward. Ion chambers or electroscopes were lined with some material which would yield a charged particle when irradiated with neutrons. Typical approaches were to line the detectors with boron or a hydrocarbon to provide a supply of charged particles. Natural boron contained enough ^{10}B to provide reasonable sensitivity for the detection of neutrons by the (n,α) reaction. While this approach still finds some practical applications, such as reactor control, it suffers as a dosimetry system because it merely detects neutrons. Only if the energy spectrum of the neutrons is known can dose information be obtained. In addition, nothing had been done, or could be done, to reduce the inherent sensitivity of these devices for gamma rays. Thus, two units were always operated, one with and one without the neutron-sensitive coating.

Because it was necessary to know the neutron energy spectrum to determine neutron dose, it became clear that a spectrometer was desirable. The means had been available since before the war in the form of the nuclear emulsions developed for cosmic-ray studies. Recoil protons from the hydrogenous film backing form tracks in the emulsions which can be correlated with the incident neutron spectrum and used for neutron dosimetry. Moreover, the tracks could be separated from the general clouding of the emulsion caused by gamma rays. That this system is successful is illustrated by the fact that it is still widely used for personnel monitoring, and

indeed a nuclear track film is a component in the film badges worn by Oak Ridge employees.⁽³⁾

Regardless of how successful this approach was for personnel monitoring, it has limitations, and the quest for other systems continued. One that measured spectrum, in a crude way, was developed. This was the threshold detector system in which foils of different elements were used. The most widely used system has six foils: gold, gold enclosed in cadmium, plutonium-239 enclosed in boron-10, neptunium-237, uranium-238, and sulfur.⁽⁴⁾ The first figure shows the reason for this selection. This graph represents the probability of a neutron producing an interaction of interest as a function of neutron energy. Gold is activated to produce radioactive gold-198; the sulfur undergoes an (n,p) reaction and produces phosphorous-32, a beta-ray emitter. The other elements undergo fission, producing radioactive fission products. In each case, the radiations can be measured with a scintillation detector, and, as the curve shows, this activity is related to the number of neutrons above a certain threshold level.⁽⁵⁾ Thus, when enclosed in the proper thickness of ^{10}B , ^{239}Pu interacts in a fairly uniform manner with all neutrons that have an energy greater than 1 kev. Neptunium-239 reacts with those above 750 kev, ^{238}U with those above 1.5 Mev, and ^{32}S with those above 2.5 Mev. By subtracting the number of neutrons above 750 kev from those above 1 kev, those in the range from 1 kev to 750 kev are determined. Similar calculations will produce the number of neutrons between 750 kev and 1.5 Mev, 1.5 Mev and 2.5 Mev, and above 2.5 Mev. Multiplying these fluxes by the average dose per neutron for the related energy increments

and summing provides the required fast neutron dose measurement. Gold is used to determine the number of neutrons with thermal velocities. This system is widely used as a nuclear accident dosimeter, but it has shortcomings. Because the thresholds are not sharp, the sensitivity of the foils in these regions depends upon neutron spectrum. The average cross sections and the average dose per neutron are also spectrum dependent. Thus, one has a device, the calibration of which depends on spectrum, being used for spectrum measurements. As a spectrometer, this is a severe weakness; as a dosimeter, some of the effects tend to cancel so that specific knowledge of the shape of the spectrum, before measuring it, is not required. As a result of the energy increments used, this instrument was designed to be most useful for measuring doses from normal fission spectra. Spectra with a predominance of neutrons below 0.5 Mev are difficult to evaluate with this system as are spectra with large numbers of neutrons greater than 2.5 Mev. The second problem may be alleviated somewhat by using additional foils with higher energy thresholds. This system is insensitive. Evaluation of long exposures is complicated by the complex decay of the fission products occurring concurrently with irradiation. A related problem arises from the need to count the fissile foils within a short time after exposure due to their rapid decay. Prompt recovery is not always feasible, such as after a nuclear accident or during a nuclear weapons test. Finally, some rather expensive foil counting equipment is required. The fission foil system described is shown in the second figure. A variation using thin targets of the fissionable material and a silicon surface barrier diode to detect the fission fragments is

shown in the third figure.⁽⁶⁾ This arrangement has some advantages over the basic foil system in that it may be used to measure lower levels and lengthy runs. In fact, this equipment was used to provide an accurate calibration of the standard foil counters. This system requires either a considerable amount of electronic equipment, if all three fissile foils are to be operated simultaneously, or a great deal of exposure time. A more recent variation is given in Fig. 4.⁽⁷⁾ Thin fissile foils are used, and the fission fragments are detected by the tracks they produce in a piece of glass or plastic. This detection system was developed by the people at Jülich.⁽⁸⁾ The advantages make it a quiet revolution in nuclear accident dosimetry. Only small amounts of fissile material are needed, but the good geometry for detection increases the over-all sensitivity. No expensive apparatus is required to read the glass, for when it is etched, the tracks become readily visible with an ordinary microscope. What a glass looks like after irradiation and etching with hydrofluoric acid is shown in Fig. 5. Early recovery is no longer important. Klaus Becker of Jülich estimates the half-life of the unetched tracks to be similar to the age of the earth. When used for nuclear accident dosimetry, it can be recovered at any time after an accident and sent elsewhere for interpretation. In fact, this system is so appealing that we have begun studies to determine if a similar approach could be used to replace the nuclear track film in the personnel monitoring system. In certain plastics, it is possible to detect tracks left by α particles of particular energy. This increases the possibility that a suitable personnel monitoring system may be designed around solid-state detectors.

Another solution to the problems of neutron dosimetry, developed in the early fifties, was the polyethylene-lined, ethylene-filled proportional counter. As shown in Fig. 6, Hurst found that the ratio of energy deposited per gram by fast neutrons in polyethylene to that deposited per gram in tissue remained essentially constant over a wide range of neutron energies.⁽⁹⁾ A cross-sectional drawing of the counter is given in Fig. 7. Thus, if the energy deposited in polyethylene could be measured, a simple factor would convert it to the amount of energy which would be deposited in tissue irradiated under similar circumstances. Bragg and Gray had developed a theory that the energy deposited in a small void within an equilibrium thickness of material would be related to that deposited in the material.⁽¹⁰⁾ In addition, if the void were filled with a gas with the same composition as the material, the void could be any size. Ethylene and polyethylene (C_2H_4) fulfill these requirements. Both are hydrogenous materials so that, in a neutron fluence, recoil protons will be produced equally in each for a unit mass. The protons lose energy by ionizing atoms along a fairly straight path. For that part of the path in the gas, an applied electric field separates the electrons and ions and accelerates the electrons to the anode. Near the anode, the electric field is intense and the electrons obtain enough energy from it to produce additional ionizations. The total number of ions formed is proportional to the initial number of electrons produced along the proton track which is, in turn, proportional to the amount of energy lost by the proton. Thus, the work done by the electric field in moving the positive ions out of the region near the anode is proportional to the energy deposited by the proton in the gas. A voltage

pulse developed across the capacitance to the anode is related to the work done by the electric field. By integrating the number and size of these pulses, the energy deposited by the recoil protons, and thus the incident neutrons, can be measured. An electron will lose much less energy in traversing the counter than a proton, so pulses produced by gamma rays will be smaller than those produced by neutrons. Rejecting small pulses limits the number of gamma-ray interactions recorded. Unfortunately, this also affects the minimum neutron energy which can be measured, a fact of varying importance depending upon the neutron spectrum.⁽¹¹⁾ When the actual amount of gas present in the sensitive volume of the counter is known, absolute measurements of energy deposited per gram may be made. Energy calibrations are simplified by an internal α source of known energy since the α particle will be stopped before it has traversed the sensitive volume. This detector has found wide application where careful measurements of low dose rates are required, although it is not suited to measurements of high dose rates or low energy neutrons. Because absolute measurements may be made with it, it has been used to determine the average dose per neutron from a number of commonly used neutron sources.⁽¹²⁾ A smaller counter operating on the same principle was built for measurements in a tissue-equivalent phantom. These measurements are of considerable importance to determine how neutron dose builds up and is attenuated in animals and humans. A computational program has been conducted as well to determine the neutron dose as a function of position in a man-sized tissue-equivalent phantom.⁽¹³⁾

Hurst and Ritchie have developed a generalized concept of radiation dosimetry that describes a system of operators that makes it possible to use any detector system which has an energy-dependent response to measure any type of dose.⁽¹⁴⁾ An application of this principle was used to make neutron dose measurements with the simple apparatus shown in Fig. 8.⁽¹⁵⁾ Measurements of average dose per neutron for several sources were made which agreed well with those made with the proportional counter. This particular system is not convenient for routine measurements but served to illustrate the generalized concept of dosimetry.

Finally, a system of neutron dosimetry which deserves mention, primarily due to its popularity, involves the use of a thermal neutron detector within a sphere of moderating material. Measurements with it are expressed in a dose unit theoretically related to the biological effectiveness of the radiation. This system will measure over the entire energy region of thermal, intermediate, and fast neutrons but generally overestimates the dose in the intermediate region. A variation on this technique has been used by Japanese investigators to determine the neutron dose versus distance from the nuclear detonations at Hiroshima and Nagasaki. They measured ^{60}Co activity in iron, such as reinforcing rods, buried in concrete.

Until good systems of neutron dosimetry were developed, it was not evident that gamma-ray dosimetry was perhaps the most difficult of the two, particularly for coexistent neutron and gamma-radiation fields. The problem can be stated simply. Almost all elements produce charged particles or gamma rays when irradiated with fast neutrons. Thus, gamma-ray measurements in a mixed field are complicated by the contribution caused by neutrons.

Ion chambers and electroscopes built with hydrogenous materials were recognized early as being poorly suited for gamma measurements in mixed fields. Carbon ion chambers filled with CO_2 gas were tried and considered adequate until carbon recoils were recognized as a severe problem. Thus, a solution evolved based upon the premise that if neutron interactions cannot be eliminated, perhaps a system can be devised which minimizes their contribution to the measured dose. Such a system is the Single Ion Detector shown in Fig. 9.⁽¹⁶⁾ Basically, this is a small proportional counter with a low pressure filling, which is made of materials known to have a low probability of undergoing neutron interactions. Its dimensions are such that when a gamma ray produces an electron in the sensitive volume, the ion path produced by slowing down of the electron is only one ion long. This produces a pulse of a particular small size from the detector. A large charged particle produced by a neutron interaction will leave more energy in the chamber but will also produce only a single pulse. Thus, while more dose was deposited, its contribution had been minimized. This is a difficult device to operate and use, but it led directly to what is probably the simplest device possible.⁽¹⁷⁾ As shown in Fig. 10, this is a small geiger counter. Its small size reduces the probability of unwanted neutron interactions, and, like the SID, it produces only a single output pulse, of uniform size, for every charged particle reaching its sensitive volume. What's more, the halogen-quenched tube used is stable and produces large output pulses. A graded shield encloses the detector to decrease the response at low energy. Any gamma-ray detector composed of high Z materials will have an increased response at low energy due to the increase in the

probability of a photoelectric interaction. The shield makes the detector response uniform from 100 kev to over 8 Mev.⁽¹⁸⁾ Because it is more sensitive to thermal neutrons than desirable, a ^6Li shield is used when measurements in a mixed radiation field are made. The small size of this detector make it, like the small neutron proportional counter, admirably suited for depth-dose measurements.

Other properties of materials have been exploited for use in gamma-ray dosimetry. One which has found wide use is radiophotoluminescence.⁽¹⁹⁾ In this case, a material is put into a metastable state by the gamma rays and then returned to the ground state by irradiation with ultraviolet light, with the simultaneous emission of energy. Silver-activated metaphosphate glass is the medium most often used for this type of measurement. These detectors can be quite small, such as the 1-mm-diameter-by-6-mm-length rods used as accident dosimeters in the film badge unit. This type of dosimeter, in other physical configurations, can be reasonably sensitive and also capable of high dose readings. However, as gamma-ray sensitivity increases, so does neutron sensitivity, mostly as response to thermal neutrons. This is generally minimized by shielding with ^6Li . The increased response at low energies is usually compensated by a graded tin and lead shield similar to that used with the geiger tube.

Another approach, to be mentioned only briefly, is thermoluminescence. This system is not basically different from glass except the luminescence is obtained by heating the sample. This appears to yield greater sensitivities than glass but has an uncertain response to neutrons. For some applications, the fact that readout is destructive can be a severe handicap. Until lately,

thermoluminescent materials were also generally more difficult to handle than glass, but considerable improvements in packaging have occurred. LiF and Teflon can be compressed to give rigid, easy-to-handle detectors. It is interesting to note that we use ^6LiF -Teflon containers to shield glass from excessive thermal neutrons.

An interesting application of thermoluminescence has been made by Japanese investigators.⁽²⁰⁾ Using the nonferrous portions of ground-up roof tiles, they have been able to determine the gamma doses as a function of distance from the nuclear detonations at Hiroshima and Nagasaki. Their results are in good agreement with similar data obtained by other methods.

One final note before we leave the subject of gamma dosimetry; the generalized concept of dosimetry is just as applicable to gamma dosimetry as to neutron dosimetry. One of the first tests of the principle suggested was to use a scintillation detector for dose measurements.

An important and powerful tool in measuring radiation doses is spectrometry. Measurements of the energy spectrum of the incident radiation can be used to calculate almost any dose desired, e.g., depth dose, kerma, etc. For several years, the fluid state of dose units and terminology made it appear that perhaps spectrometry would be the only satisfactory solution to the dosimetry problem. Fortunately, this situation has eased somewhat, fortunate not only because there is considerably less confusion, but also because spectrometry is a difficult problem, and to supply a practical, easily operated device for field use seems beyond the limits of present technology.

Even limited in its use, spectrometry has been particularly useful in solving dosimetry problems associated with specific experiments. In general, neutron spectrum measurements have been important due to the various energy-dependent effects of the neutron dosimeters available. Some of the spectrometers used were developed to solve specific dosimetry problems, others were borrowed from the nuclear spectroscopists. An example of the first type is shown in Fig. 11. This device was designed more for sensitivity than resolution.⁽²¹⁾ It is a thick radiator, recoil proton telescope that was built originally for use during a field experiment at the Nevada Test Site. It will be used again this fall for another experiment at the NTS. Neutrons are collimated so that they impinge normally upon the first organic phosphor. Recoil protons are produced with those scattered in the near forward direction traversing the evacuated space and being detected in the second phosphor. When a coincidence is obtained and identified as a proton by a special pulse shape selection circuit, the signals from the detectors are summed and analyzed. Thus, information about the energy deposited by the proton in leaving the radiator is not lost. As the range of the protons increases with energy, so does the useful radiator thickness and thus efficiency of this spectrometer, a rather unusual situation. Unfortunately, the recorded pulse-height spectrum has little in common with the neutron spectrum due to the shape of the efficiency curve and the nonlinear light output versus proton energy of the phosphors.

For gamma rays, spectrometry in mixed radiation fields has proven especially difficult. A spectrometer was built for use during a field

experiment in 1962 which operated by magnetically analyzing the Compton recoil electrons produced in a beryllium foil by the incident gamma spectrum.⁽²²⁾ It is shown in Fig. 12. While this yielded low neutron sensitivity, it also gave low gamma efficiency so little useful data were obtained during this experiment. This approach has warranted further investigation and development due to its wide energy range and freedom from neutron response. It shows considerable promise for high resolution measurements.

For a pure gamma-ray field, standard scintillation detectors have proven useful although they produce a complex response to a given gamma-ray energy. During the field experiment previously mentioned, the gamma spectra as a function of time after a burst of neutrons and as a function of angle from a large ^{60}Co source were measured. The next figure shows one of the curves from the ^{60}Co measurements and illustrates the complexity of data from a scintillation detector.

More sophisticated scintillation spectrometers utilizing the pair production interaction of high-energy gamma rays or Compton recoil reactions for wide energy ranges are also employed. As an example, the Compton spectrometer configuration is shown diagrammatically in Fig. 14.⁽²³⁾ Because both systems operate with some sort of coincidence analysis, neutron response is low. In fact, a PuBe neutron source, which is an abundant source of 4.43-Mev gamma rays, is commonly used to set up the pair spectrometer.

Spectrum measurements are not limited to the incident spectrum of neutrons and gamma rays. A good example of other measurements made in

dosimetry experimentation is shown in Fig. 15. This is the spectrum of recoil particles produced by 15 Mev neutrons as a function of position in a tissue-equivalent phantom.⁽²⁴⁾ The spectrometer in this case was a lithium-drifted silicon diode in the form of a cube 1.5 mm on a side. While most of the spectrum is due to protons, the increase in pulses at lower energies is due to heavy ion recoils. These data are quite recent, having been taken by a graduate student at our facility during the spring and early summer (1966).

Nuclear accident dosimetry has been an important aspect of health physics since the first probability of an accident developed. Dosimetry is important as a guide to the proper medical treatment for survivors. However, we tend to sift, vulture like, over all of the data we can obtain from the accidents which do occur in an effort to increase the store of human data dealing with radiation damage. As understanding of the effects of radiation on the human system increases, more realistic, safer limits on occupational exposures may be set. The most useful devices we have for nuclear accident dosimetry have proven to be the threshold detector systems for neutrons and glass for gamma rays. Considerable effort has been put into a biological dosimeter which everyone always has with him--his blood--through efforts to relate dose received to activation of blood sodium. If something is known or can be assumed about the incident neutron spectrum and the orientation of the victim, sodium activation is a powerful dosimetric tool. Considerable laboratory and computational work has been expended to increase our knowledge of this tool and more is continuing.⁽²⁵⁾

In 1964 we performed an experiment at Los Alamos to measure the radiation leakage of a critical assembly designed to have leakage characteristics similar to the Hiroshima weapon.⁽²⁶⁾ This input information was another basic piece of information in evaluating the radiation doses received by the survivors of the nuclear bombings of Hiroshima and Nagasaki. Careful medical records have been kept concerning these people since shortly after the war which, when coupled with good dose estimates, form the largest available pool of human data on the effects of exposure to nuclear radiation. Work towards determining these doses has continued since 1956 in many ways. Many experiments have been performed in the laboratory and during weapons tests and other experiments at the Nevada Test Site that have been concerned with transport of weapons radiation through air and shielding by Japanese-style houses. Backing up these extensive experiments have been many calculations. As a result, an equation has been developed which relates a survivor's location, with all its geometric and shielding considerations, to the dose he received.

During our 1962 field experiment, measurements were made of the effect of the air-ground interface on the attenuation of radiation as a function of distance from a nuclear reactor.⁽²⁶⁾ A typical set of data is given in Fig. 16 which compares the effect of the air-ground interface with the transmission of the radiation through an infinite media. The data presented in Fig. 17 were also taken during this field experiment and show the gamma dose as a function of angle at a distance of about 800 yards from the source.⁽²⁸⁾

A number of experiments in radiation scattering have been conducted; for example, the neutron albedo of various materials, such as concrete, soil, and water, were measured.⁽²⁹⁾ Theoretical work was done to relate the measured results to known physical phenomena. A more recent experiment provided the same sort of data for gamma rays scattered by lead, water, or concrete. Typical data for concrete and ^{60}Co gamma rays are shown in Fig. 18. It is worth noting that a great deal of this latter work was done by a graduate student and a visiting professor.⁽³⁰⁾

Obviously all of the material I have presented was not for your edification as dosimetrists but was directed toward increasing your understanding of the tools and research which has been conducted in the past, directed at solving specific problems in dosimetry. I hope that the omissions and simplifications were not overdone. Many widely used dosimetry systems were not mentioned because we have done little with them. For the sake of time, most details of our extensive field experiments were omitted. While a great deal of that discussed seems applied, such an impression is incomplete and may have resulted from the fact that my own work has been highly applied in the last few months as we prepare for another field operation. The operation itself is a large-scale, basic research problem in the transport of neutrons in air and a study of the high-energy gamma rays produced by neutron interactions in air.⁽³⁰⁾ Moreover, as more is learned about dosimetry, new approaches become more basic. In short, those of us in dosimetry find it constantly challenging and are eager to encourage others to join us.

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FIGURE CAPTIONS

Figure 1. Threshold Detector System Cross Sections.

Figure 2. TDU Fission Foil Configuration.

Figure 3. A Fissile Foil with a Semiconductor Diode Fission Fragment Detector.

Figure 4. Fissile Foils with Plastic Fission Fragment Detectors.

Figure 5. Microphotograph of Etched Glass after Pitting by Fission Fragments.

Figure 6. Ethylene and Tissue Neutron First Collision Dose vs E_n and the Ratio of These Two Doses.

Figure 7. Polyethylene-lined, Cyclopropane-filled Proportional Counter for Fast Neutron Dosimetry.

Figure 8. Detector and Electronic Equipment Used to Test the Utility of the Hurst-Ritchie Generalized Concept of Dosimetry.

Figure 9. Single Ion Detector.

Figure 10. "Phil" Gamma Dosimeter and Thermal Neutron Shield.

Figure 11. Thick Radiator Telescope Neutron Spectrometer.

Figure 12. BREN Compton γ -Ray Spectrometer.

Figure 13. Typical BREN ^{60}Co Gamma-Ray Spectra as Recorded with a 3-In.-by-3-In. NaI(Tl) Scintillation Detector.

Figure 14. NaI(Tl) Compton Spectrometer.

Figure 15. Recoil Proton Spectrum Produced by 15 Mev Neutrons as a Function of Position in a Tissue-Equivalent Phantom.

Figure 16. BREN Air-Ground Interface Data.

Figure 17. BREN γ Dose vs Angle at 750 Yards from the Base of the Tower.

Figure 18. Gamma-Ray Albedo of ^{60}Co on Concrete.

DENSITY OF B^{10} - 1.65 GRAMS PER C.C.
SECTION THROUGH SPHERE ON $\perp$
PERPENDICULAR TO PARTING LINE

Fast Neutron First Collision Curves for Tissue and for Ethylene.

ABSOLUTE FAST NEUTRON DOSIMETER

ORNL-DWG. 65-13030R

GAMMA DOSIMETER #2

ORNL-LR-DWG 45048-A

THERMAL NEUTRON SHIELD

PROBE ASSEMBLY

ORNL-LR-DWG 61705

COMPTON SPECTROMETER

AXIAL TRAVERSE

NEUTRON ENERGY - 14 MeV

DETECTOR - 1.5 mm CUBE

ERROR BARS - 1 STANDARD DEVIATION