British Committee
BCRU
on Radiation Units and Measurements

A GUIDE

TO THE MEASUREMENT OF

ENVIRONMENTAL GAMMA-RAY DOSE RATE

F.W. Spiers

J.A.B. Gibson

I.M.G. Thompson

A Guide to the Measurement of Environmental Gamma-Ray Dose Rate

by

F.W. Spiers, C.B.E., D.Sc., Ph.D., F.Inst.P.

Bone Dosimetry Unit
Cookridge Hospital
Leeds

J.A.B. Gibson, B.Sc., M.Inst.P.

Environmental and Medical Sciences Division
Atomic Energy Research Establishment
Harwell

I.M.G. Thompson, B.Tech.

Berkeley Nuclear Laboratories
Central Electricity Generating Board
Berkeley

Published on behalf of the

BRITISH COMMITTEE ON RADIATION UNITS AND MEASUREMENTS

by the

NATIONAL PHYSICAL LABORATORY

Teddington
Middlesex, TW11 0LW
United Kingdom

ABSTRACT

SPIERS, F.W., GIBSON, J.A.B and THOMPSON, I.M.G. A guide to the measurement of environmental gamma-ray dose rate, Teddington, Middlesex, National Physical Laboratory, United Kingdom, (on behalf of British Committee on Radiation Units and Measurements), 1981. 107 + x pages, ISBN 0 9504496 7 9.

This report provides information and guidance on the measurement of the absorbed dose rate from low-level gamma radiation, such as may occur in the neighbourhood of a nuclear installation. Apart from the technical difficulties of measuring such low absorbed dose rates, there are also problems in assessing the significance of the results in relation to the roughly comparable, but variable, dose rates from natural sources.

After an introductory chapter there is a review of the nature and variability of natural environmental gamma radiation and of cosmic radiation to which most dosemeters also respond; the level of gamma radiation from man-made sources such as fall-out and nuclear installations is also discussed briefly. Geiger—Müller counters, ionization chambers, scintillators, gamma-ray spectrometers and thermoluminescence dosimeters, and their different performance characteristics, are examined in relation to their usefulness for various types of measurements. Other chapters are concerned with the calibration of the dosemeters, measurement procedures, precautions to reduce errors, any ancillary measurements needed to derive corrections of the measured absorbed dose rates, the assessment of accuracy and the interpretation of results. In order to put into perspective the underlying reasons for making these measurements, a chapter is provided on the calculation of dose equivalent to body organs, and typical levels of the annual dose equivalent received by the population in the UK. The final chapter is a summary of the main recommendations and conclusions, and the document ends with a comprehensive list of references.

An important conclusion put forward relates to the minimum change of gamma-ray dose rate that can be regarded as significant. After examining the uncertainties in measurement procedures and typical variations experienced in the natural environmental dose rate, the authors conclude that the minimum change in measured dose rate that is significant at the 95% confidence level as an estimate of the mean environmental dose rate is 120 μGy/yr (12 mrad/yr). This has important implications in the drafting and implementing of regulations relating to nuclear installations. The Guide should be of value not only to users of monitoring instruments, but also to those who have to assess the results in relation to compliance with official regulations or recommendations.

CONTENTS

LIST OF TABLES

LIST OF FIGURES

PREFACE

The British Committee on Radiation Units and Measurements (BCRU) recognized the need for guidance on the calibration and use of instruments to measure environmental gamma radiation. This measurement is of particular importance around a nuclear installation, for which the site-licence conditions allow the subtraction of natural environmental radiation from doses attributable to workers or members of the public. A small working group was therefore set up by BCRU, consisting of Professor F.W. Spiers, Mr J.A.B. Gibson and Mr I.M.G. Thompson. The working group was invited to review the nature and variability of environmental gamma radiation, to consider the types of instruments that are suitable for measurements at these low levels, and to discuss the calibration of such instruments and the interpretation of their readings. They were asked to provide a report embodying these points and other relevant information for publication by BCRU. This Guide is the result of their deliberations.

During the preparation of this Guide, SI units were introduced into radiation measurements and so both new and old units are quoted, with the old units in parentheses. The new named units and their relationships to the old special units are summarized below.

Quantity	New SI unit and symbol	In other SI units	Old unit and symbol	Relationship between old and new units
Exposure	...	C/kg	röntgen (R)	1 R = 2.58×10^{-4} C/kg
Absorbed dose	gray (Gy)	J/kg	rad (rad)	1 rad = 0.01 Gy
Dose equivalent	sievert (Sv)	J/kg	rem (rem)	1 rem = 0.01 Sv
Activity	becquerel (Bq)	s^{-1}	curie (Ci)	1 Ci = 3.7×10^{10} Bq

In the past, environmental gamma-ray levels have been expressed either in terms of exposure rate in units of μR/h or mR/yr as appropriate, or in terms of absorbed dose rate to air in units of μrad/h or mrad/yr; an exposure of 1 R has been taken to be equal to an absorbed dose to air of 0.869 rad. In future, the inconvenient magnitude of the SI unit coulomb per kilogram will probably lead to the abandonment of exposure rate as a practical quantity, so absorbed dose rate to air has been used in most places in this Guide, in units of nGy/h or μGy/yr as appropriate. When the term dose is used, it should be assumed that absorbed dose to air free-in-air (i.e. scatter-free except for air

scatter) is meant, and that the measuring device has a wall thick enough to give electron equilibrium. Air kerma, which uses the same unit, gray, as absorbed dose, can be taken to have the same numerical value as absorbed dose to air under conditions of electron equilibrium. Exposure, air kerma and absorbed dose to air can also be converted into any appropriate dose-equivalent quantity; values for conversion factors are discussed in Chapter 7.

Just before the publication of this Guide a working party was set up by the International Commission on Radiation Units and Measurements (ICRU) to put forward proposals on the most appropriate quantity to use for measuring ambient radiation for routine protection purposes. Any recommendations of this working party may influence the choice of quantities used in future versions of the Guide.

When referring to the various components of background radiation, the following terminology has been adopted throughout this Guide.

BACKGROUND RADIATION includes both external and internal components.

ENVIRONMENTAL RADIATION is the external component of background radiation and comprises:

terrestrial gamma radiation (arising directly from radionuclides in the ground);

atmospheric beta and gamma radiations (arising directly from radionuclides in the atmosphere);

cosmic radiation;

fall-out beta and gamma radiations;

other man-made beta and gamma radiations.

NATURAL ENVIRONMENTAL RADIATION comprises all the components of environmental radiation except for fall-out and other man-made radiations.

NATURAL ENVIRONMENTAL GAMMA RADIATION comprises terrestrial gamma radiation and atmospheric gamma radiation.

ENVIRONMENTAL GAMMA RADIATION comprises natural environmental gamma radiation, and fall-out and other man-made gamma radiations.

Technical Panel 7 of the British Calibration Service suggested the need for this Guide to the BCRU. The authors wish to acknowledge the particular help of Mr J.E. Burns (Scientific Secretary, BCRU), who made many improvements to the presentation and to the quality of information. We are grateful for useful comments from Professor J.R. Greening, Chairman, and other members of BCRU; also from Mr M. O'Riordan (NRPB), Dr B.M. Wheatley (CEGB) and Mr J.A. Douglas (AERE). Finally we acknowledge the careful preparation of the text for publication by Mr D.P. Biddiscombe (NPL).

1 INTRODUCTION

Over the past twenty-five years there has been a considerable and increasing interest in the level of background radiation to which man is exposed and particularly in the fraction of the background dose rate attributable to environmental gamma radiation, whether of natural or man-made origin. The total natural background dose rate to man is usually taken as about 1 mGy/yr (100 mrad/yr) and is made up of cosmic radiation, internal radiation and environmental gamma radiation in roughly equal proportions. It is often regarded as a yardstick against which dose rates from man-made sources can be compared, especially when consideration is given to the radiation received by populations from such sources as fall-out from the testing of nuclear weapons, releases of radioactivity from nuclear power plants, or X radiations and radionuclides used in medicine (MRC, 1956, 1960; UNSCEAR, 1962, 1966, 1972, 1977). An important example of this relates to the environmental gamma-ray dose rate in the neighbourhood of a nuclear installation and the possible contribution made to it by the operation of that installation. However, it is far from easy to establish whether a significant departure from what is regarded as the natural level has occurred. Unless any increment is large compared with the natural level, when detection would be easy, its measurement will be dominated by problems of measuring the natural environmental gamma radiation itself. This Guide concentrates, therefore, on the measurement of environmental gamma radiation at or about the natural level and the factors that affect the attainable accuracy and significance of the result.

The fraction of the background dose rate to man from environmental gamma radiation is very variable and depends on factors such as the radioactivity of the local rock and soil, the nature of building materials and the construction of buildings in which people live and work. Dose rates to man from natural environmental gamma radiation therefore exhibit considerable geographical variation; they may be very much lower in areas of sedimentary rock compared with areas of igneous rock, and may be an order of magnitude higher, amounting to about 10 mGy/yr (1 rad/yr), in some parts of the world. The natural environmental gamma-ray dose rate also varies with time, and this in

itself makes it difficult to specify the exact dose rate at a particular site; it cannot be regarded as a fixed, invariable quantity.

The measurement of the low dose rates associated with environmental gamma radiation, e.g. in the range 20—150 nGy/h (2—15 μrad/h) in the UK, is not without difficulties inherent in the instrumentation. Measuring instruments may respond differently to the various components of environmental radiation, especially to cosmic radiation, which is inevitably included in the indication of the measuring instrument. Stray inputs may occur in the measurement circuit, caused by insulator leakage, grid current or 'noise' from associated electronic equipment. Radioactive contamination of the instrument may be present naturally or occur accidentally, and go unrecognized unless careful checks are made. These factors affect the response of an instrument to an extent that is not always easy to determine, yet they must be eliminated or assessed quantitatively if the environmental gamma-ray dose is to be correctly deduced from the instrument's indication. It is equally important to establish the constancy of the gamma-ray calibration, as well as the long-term stability of the instrument performance. Unfortunately there is no natural situation, and no easily fabricated zone, that is completely free of all ionizing radiation where a 'zero' adjustment could be made to offset the current, or count rate, resulting from the factors enumerated above.

Although there is no defined allowable increment above background to the dose rate around a nuclear installation, some indication of possible requirements may be inferred from national and international recommendations. The International Commission on Radiological Protection have proposed (ICRP, 1977) that the maximum annual dose equivalent to an individual in the general population should not exceed 5 mSv (500 mrem). The National Radiological Protection Board have advised (NRPB, 1978) that the annual dose equivalent to members of a 'critical group of the general public' from the disposal of radioactive wastes in the UK should not exceed 1 mSv (100 mrem). Further, the nuclear industry accepts the principle (ICRP, 1978) that the dose equivalent from man-made radiation should be kept 'as low as reasonably achievable' (ALARA). It could be suggested that one tenth of the recommended dose-equivalent levels would represent a practicable adherence to the ALARA principle, thus implying that any increase in the environmental dose rate should not exceed 100 μGy/yr or 500 μGy/yr

(10 mrad/yr or 50 mrad/yr) above a natural level of 1000 μGy/yr (100 mrad/yr). Whether such an increase in the absorbed dose rate from environmental gamma radiation in the vicinity of a nuclear installation can be established at an acceptable significance level is one of the important questions discussed in a later chapter.

This Guide describes the nature and variability of environmental gamma radiation, particularly in relation to the levels in the UK, and considers the main types of instrument and the factors that affect their responses. Methods are given that enable the response of an instrument to be determined, and the environmental gamma-ray dose rate deduced with sufficient accuracy for practical purposes. The measurement of cosmic radiation, beta radiation, neutrons, or atmospheric radon and thoron (and their daughters) is not discussed, except where any of these radiations may interfere with the gamma-ray measurement. The accuracy to which an increment in dose rate can be measured is discussed in the light of the practical factors involved. It is hoped that the recommendations will provide an acceptable guide for the measurement of low levels of gamma-ray dose rate in the environment, whatever the origin of the radiation.

2 NATURE AND VARIABILITY OF ENVIRONMENTAL RADIATION

This chapter begins with a description of the origin of natural environmental gamma radiation, its energy distribution and its variability. Most instruments also respond to cosmic radiation and to gamma radiation arising from fall-out from nuclear explosions, and these two sources of radiation are discussed. In order to put the measurements into perspective, the chapter concludes with a section on the production and magnitude of environmental gamma radiation from nuclear installations.

2.1 Composition of environmental gamma radiation

At the earth's surface environmental gamma radiation arises mainly from the long-lived radionuclides present in the primary rocks of the crust and, in lesser concentrations, in sedimentary rocks and soils; in this Guide, this component is referred to as 'terrestrial gamma radiation'. An additional small contribution is made by radon diffusing from the soil into the atmosphere in a manner that depends on local meteorological conditions, and that may increase significantly for short periods under special atmospheric circumstances; this component is referred to as 'atmospheric gamma radiation'. Certain shorter-lived radionuclides are present at the earth's surface because they are continuously generated in the upper atmosphere (and to a small extent at the surface itself) by cosmic radiation; of the dozen or so thus produced, only ^{14}C and ^{3}H are significant contributors to the dose received by human beings and other living things, but since these are low-energy beta-ray emitters they make no contribution to the environmental gamma radiation.

Radioactivity in water is variable and high levels of radon and radium are found in deep-well water depending on the geological structures in which it originates. Surface waters are, however, generally low in radioactivity, the concentration of radionuclides in sea water being about two orders of magnitude lower than in rocks and that in river waters about four orders of magnitude lower. The water content of the soil does not therefore contribute significantly to the radiation dose rate above it.

Considerable data on the occurrence of natural radioelements are given by UNSCEAR (1977) and NCRP (1976), and the following examples are based on information in these reports. Table 2.1-1 illustrates typical concentrations of the most important gamma-emitting radioelements in rocks; for the two radioactive series only the values for the parent elements uranium and thorium are given, but generally the daughter products in the series are in approximate equilibrium. Exceptions are found where the escape of radon seriously disturbs the equilibrium of

TABLE 2.1-1: Concentration of radioelements in rocks; from NCRP (1976), Table 2.6

Type of rock	Radioelements K (%)[1]	Th (ppm)[2]	U (ppm)[2]
Igneous			
Silica (e.g. granites)	3.3	20.0	4.7
Intermediate (e.g. diorites)	2.3	8.0	1.8
Mafic (e.g. basalt)	0.8	2.7	0.9
Ultramafic (e.g. dunites)	0.5	6.0	0.03
Sedimentary			
Limestones[3]	0.3	1.7	2.2
Carbonates	...	1.9	2.1
Sandstones[3]	1.2	~3.0	~1.5
Shales	2.3	11.0	3.5

[1] Potassium contains 0.0118% of K-40 (by weight)

[2] ppm—parts per million by weight

[3] Compare for example Edinburgh and Dundee (see later)

the daughter products further down the ^{226}Ra series. Table 2.1-2 gives typical radioactive concentrations of ^{40}K, ^{232}Th and ^{238}U in soil and, assuming chain equilibrium, the calculated absorbed dose rates to air at 1 m above the ground (assumed to be a semi-infinite uniform source) are shown in the right-hand columns, the total dose rate amounting to 390 μGy/yr (39 mrad/yr) or 44 nGy/h (4.4 μrad/h). Surveys in various parts of the world give average outdoor dose rates that lie generally in the range 300 μGy/yr to 1000 μGy/yr (30 mrad/yr to 100 mrad/yr) apart from a few areas of specially high radioactivity. A world-average outdoor dose rate of about 400 μGy/yr (40 mrad/yr) in air has been suggested by UNSCEAR (1977). It should be noted that the dose rate does

TABLE 2.1-2: Typical radioactive concentrations in soil and resulting dose rates at 1 m above ground; based on UNSCEAR (1977), Annex B, Tables 5 and 6, page 44

Radionuclides	Radioactive concentration		Mean absorbed dose rate to air	
	(Bq/kg)	(pCi/g)	(μGy/yr)	(mrad/yr)
^{40}K	370	10	140	14.0
^{232}Th + daughters	26	0.7	151	15.1
^{238}U + daughters	26	0.7	97	9.7
ALL NATURAL NUCLIDES	TOTAL DOSE RATE:		388	38.8

not vary much with height above ground for heights of less than 10 m; the value at ground level is only about 2% greater than at 1 m, whilst at 10 m it is lower by 11% (NCRP, 1976).

As may be expected from the nature of the contributing radionuclides, the energy distribution of environmental gamma radiation is complex; the high-energy parts of the spectrum arise particularly from the 1.46 MeV gamma rays of ^{40}K, the 1.76 MeV gamma rays of ^{214}Bi in the ^{238}U chain and 2.32 MeV gamma rays of ^{208}Tl in the ^{232}Th chain. The spectrum of environmental gamma radiation in terms of exposure rate at 1 m above ground is shown in Figure 2.1-1, as determined by Beck (1972); the mean energy, weighted by exposure, is approximately 1 MeV. (The same distribution holds for absorbed dose to air.) Extreme variations in the level of potassium in the soil will distort this distribution.

2.2 Environmental gamma-ray levels in the UK

Environmental gamma-ray dose rates vary considerably from place to place, with outdoor levels depending on the radioactivity in rock and soil. Within cities and towns the materials used for buildings and for surfacing roads and pavements also affect the dose rate. Data for the UK have been summarized in a recent NRPB Report by Taylor and Webb (1978) and available data on a world basis are given by UNSCEAR (1977).

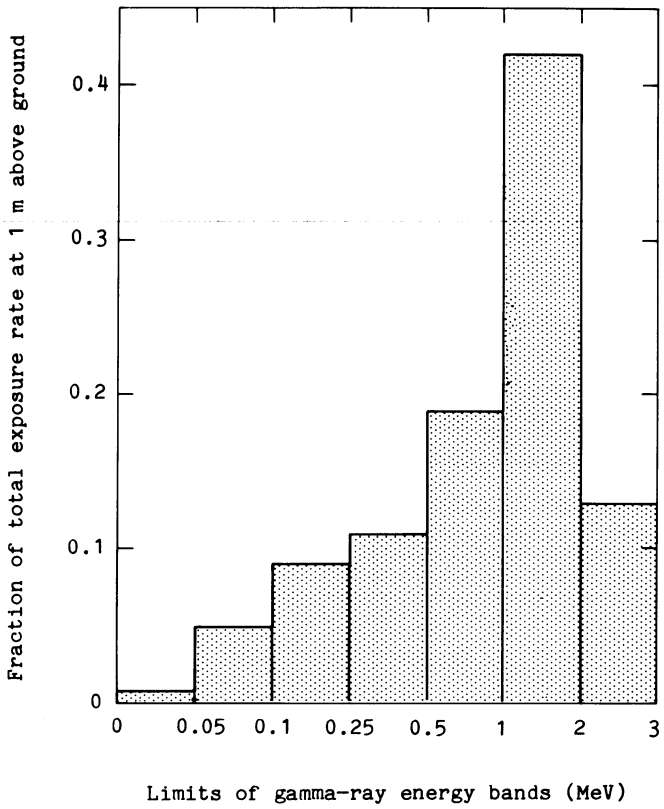

FIGURE 2.1-1: Relative contributions to the total exposure (or dose) rate at 1 m above ground from photons of various energies produced by a uniform distribution of naturally occurring radionuclides in the ground (from Beck, 1972)

As an illustration of outdoor gamma-ray levels in fairly typical UK situations, dose rates measured 1 m above ground over fields and roads in several areas of Scotland are shown in Figure 2.2-1 (MRC, 1960). Thus in Edinburgh, an area of carboniferous limestone, dose rates over the roads are low and the situation is clearly very homogeneous. In Dundee, where the surface rocks are old red sandstone, the outdoor dose rates are higher, with a greater range of dose rate. In a heterogeneous situation such as over the whole Grampian region where the geology includes a variety of both sedimentary rock and granite areas, a very wide distribution of dose rate is found. However, over fields, where the surface soils provide more uniform radioactivity, a much narrower distribution of dose rates is found; this situation is more representative of the sites where nuclear installations are located. Outdoor dose rates and their ranges in the UK are very similar to those in other countries, as for example those reported for 25 areas in the USA (NCRP, 1975). In built-up areas dose

rates can increase with urbanization; over roads near Aberdeen, for example, dose rates increase by about 50% as the centre of that 'granite city' is approached (Spiers et al., 1964).

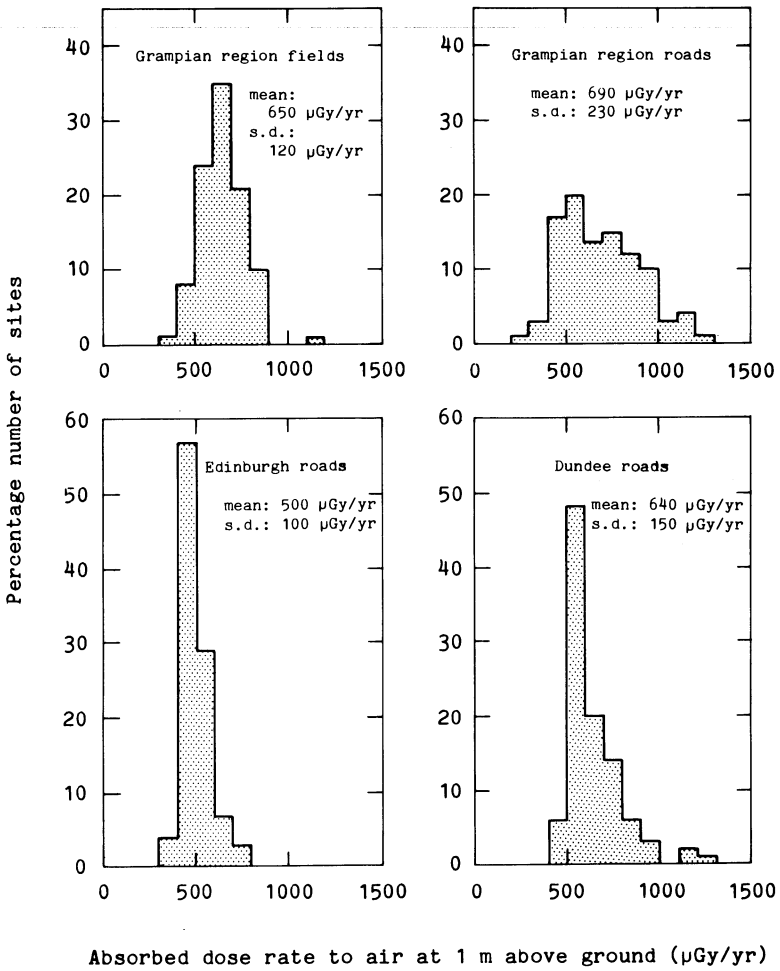

FIGURE 2.2-1: Dose rates over fields and roads in some areas of Scotland (MRC, 1960)

Indoor dose rates depend mainly on the building materials used and are less influenced by the level of radiation outside. Where buildings are of local stone, the dose rates inside houses differ little from those outside, but where buildings of brick or concrete are located in areas of sedimentary rocks and soils, the dose rates in houses may be up to 50% greater than those outside. In the USA, on the other hand, dose rates inside frame houses are about 70% of those outside, so, as rather more than 70% of the population live in such houses, the whole-body dose to the population in the USA is about 80% of the outdoor dose (NCRP, 1975). These figures are not directly applicable to the UK, but it should be borne in mind that changes in the outdoor gamma-ray dose

rate are likely to result in smaller changes to the population dose when this is largely influenced by the dose rates indoors. Some examples of dose rates in houses in relation to the types of local materials are shown in Figure 2.2-2.

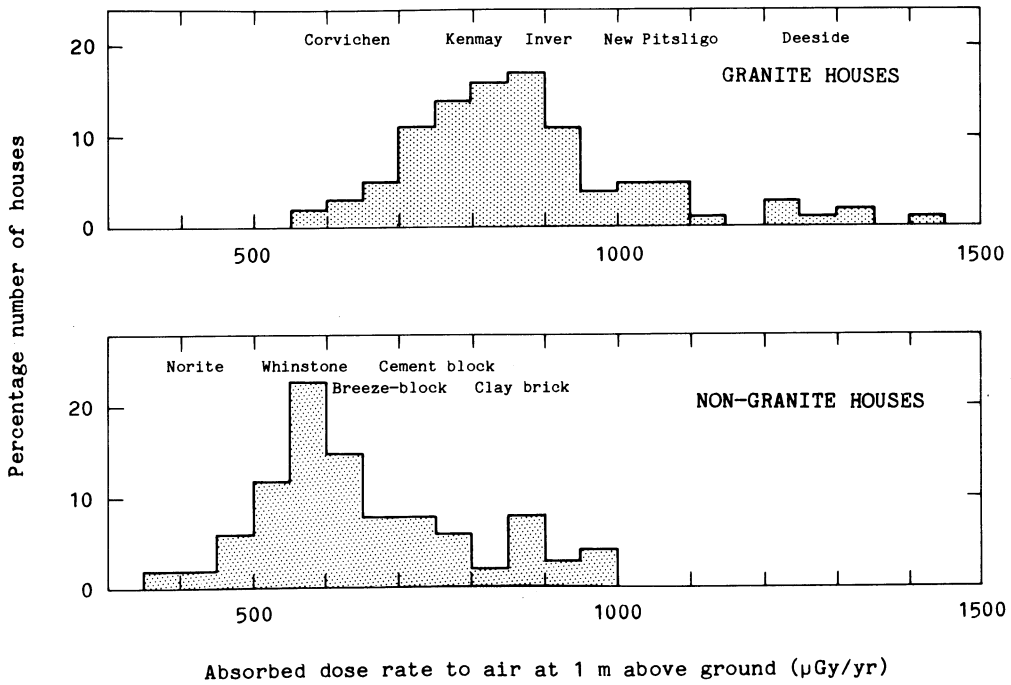

FIGURE 2.2-2: Dose rates in houses built of various local materials in Grampian region (MRC, 1960)

2.3 Effect of rain and snow

Because the radioactivity in surface waters and in rainfall is very much lower than that in rocks and soils, the gamma-ray dose rate above the soil (e.g. over grassland) depends on the water content of the soil. The water may affect the radon content in the upper soil layers by reducing its diffusion from the ground but, as this loss is normally less than 10% of the retained radioactivity, a greater effect is the increase in self-absorption of wet soil compared with dry soil and a consequent reduction in dose rate. This is illustrated in a series of continuous measurements of environmental gamma-ray dose rate with a high-pressure ionization chamber positioned 3 m above a field (Burch et al., 1964). Results for the months March to August 1961, during a period of low fall-out deposition, are shown in Figure 2.3-1. A very approximate interpretation of the data shown in Figure 2.3-1 is that a decrease of about 3% (standard deviation 0.6%) may be expected for each

centimetre of rainfall received in the previous one or two days, with an initial recovery of about 1% for each day in which no rain occurs. These suggested corrections are very rough, as may be readily seen from Figure 2.3-1 where the decrements and recovery patterns appear to be affected by the previous history of rainfall and drought. They may provide some guidance however when one measurement is being compared with an earlier one.

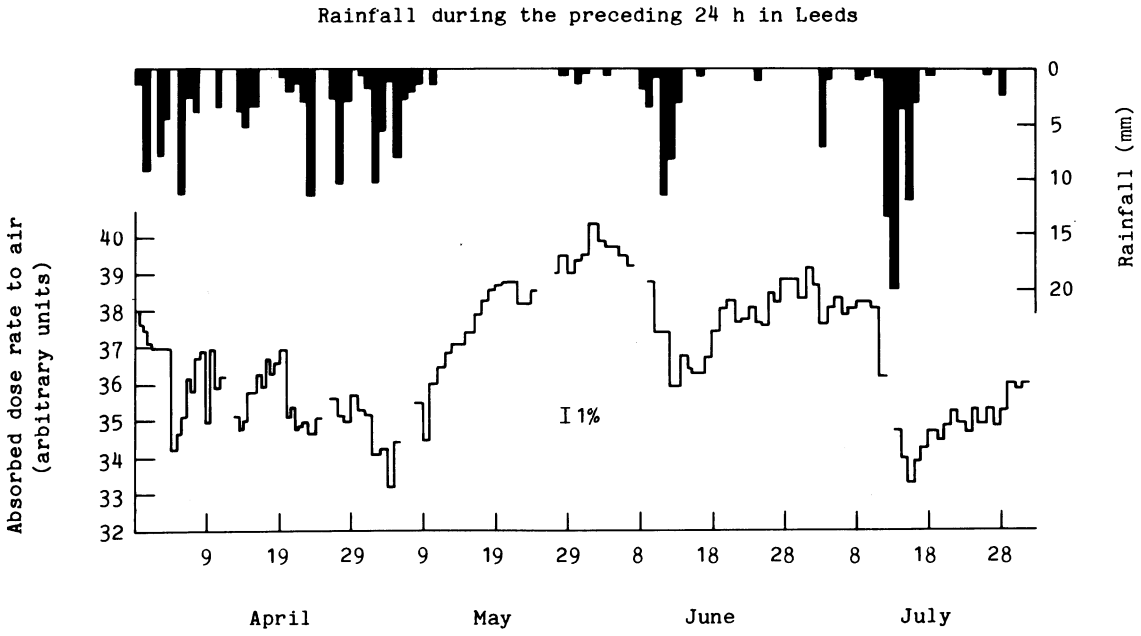

FIGURE 2.3-1: Correlation between dose rate and rainfall at a site near Leeds (Burch et al., 1964)

Snow cover also reduces the terrestrial gamma-ray dose rate and is more effective than an equivalent rainfall because it remains on the surface instead of draining immediately into the soil. A seasonal pattern of outdoor dose rate occurs, therefore, in regions subject to significant snowfall. Sievert and Hultqvist (1952) have shown that a 20 cm depth of snow reduces the environmental gamma-ray dose rate by 26% for a snow density of 0.1 g/cm^3, and by 58% for a snow density of 0.4 g/cm^3.

Normally atmospheric radon and its daughter products contribute little to the gamma-ray dose rate above ground, but if a thunderstorm occurs following a period of drought, the short-lived daughter products that have accumulated on airborne dust particles are washed out and deposited on the ground. The rather large increase in the gamma-ray dose rate that can follow such an event is shown in Figure 2.3-2; the increase is short-lived and follows the decay of the

daughter products, ^{214}Bi and ^{214}Pb. Such an event is unlikely to add significantly to the long-term dose or confuse the measurement of environmental radiation if account is taken of the meteorological conditions at the time of measurement.

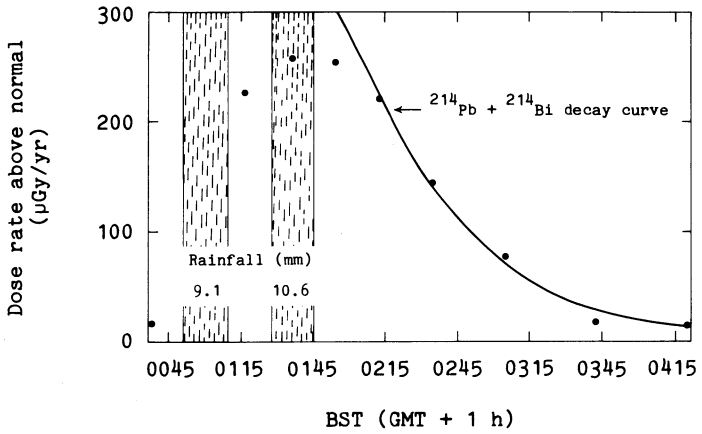

FIGURE 2.3-2: Variation in the gamma-ray dose rate from ground-deposited radon daughter products caused by a heavy summer rainstorm (Burch et al., 1964); theoretical decay curve shown for comparison

2.4 Cosmic radiation

Cosmic radiation is present at the earth's surface and increases steeply with altitude.It has a very complex composition of mu mesons, electrons, protons and neutrons, which varies with altitude and shielding. There is a 'soft' component (electrons and photon radiation), which is reduced significantly by the shielding encountered in ordinary buildings, and a 'hard' component (mu mesons), which is reduced appreciably only by great thicknesses of rock or water. Both components vary with barometric pressure and, because most radiation detectors respond to cosmic radiation, the effect of the cosmic-ray intensity must be taken into account in any measurement of the environmental gamma-ray dose rate.

Cosmic radiation at sea level is conveniently divided into these two components, hard and soft. The hard component is that which will penetrate 10 cm of lead without significant attenuation; it consists of mu mesons and produces about 75% of the cosmic-ray dose at sea level. The soft component gives the other 25% of the dose and is almost totally absorbed by 10 cm of lead; it consists mainly of

electrons and photon radiation generated in the atmosphere (as are mu mesons) by the primary cosmic radiation from the sun, the galaxy and other stars. The ratio of soft to hard components is given as a function of barometric pressure in Figure 2.4-1. The contribution of cosmic radiation to the dose rate at sea level in mid-northern latitudes is 280 μGy/yr (28 mrad/yr), which is equivalent to 32 nGy/h (3.2 μrad/h), with a dependence on barometric pressure represented by an increment of 0.2% per millibar decrease in pressure. At high altitudes the increase in cosmic-ray dose rate becomes very important, the dose rate at an altitude of 2 km, for example, being approximately twice that at sea level. In addition to variations with barometric pressure, there is a further fluctuation of about ±1% due in part to atmospheric conditions and in part to extraterrestrial events; there is a fall of 4% from the winter to the summer season (Burch et al., 1964). The dose rate from the neutron component is small, amounting to 3.5 μGy/yr (0.35 mrad/yr) at sea level. The soft component of the cosmic-ray dose rate is appreciably absorbed by building materials; an average reduction of 10% has been applied to outdoor dose rates to allow for structural shielding in US buildings (NCRP, 1975), and reductions of up to 20% in UK buildings, depending on their structure (Spiers et al., 1964).

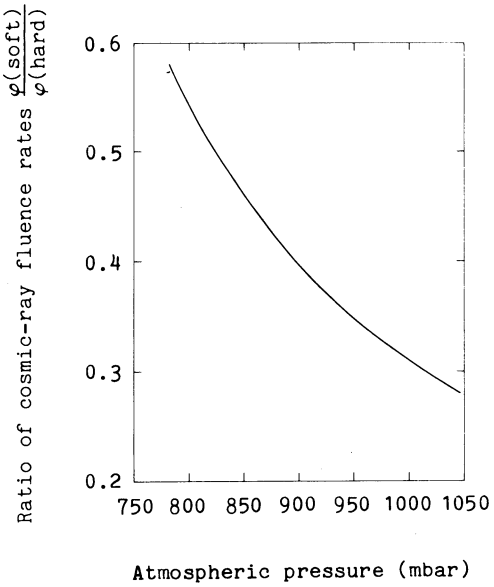

FIGURE 2.4-1: Effect of atmospheric pressure on the ratio of soft to hard cosmic-ray fluence rates (Rossi, 1948)

2.5 Fall-out from nuclear explosions

Nuclear explosions in the atmosphere produce a wide range of radioactive species, which are transported around the globe and then deposited in rain and by dry deposition on terrestrial surfaces. Although test explosions date from the 1940s, the most significant deposition occurred in the period 1958—9 and 1961—3 when the gamma-ray dose rate at 1 m above ground level in air was increased by at least 50% over that due to natural causes, and the beta-ray dose

rate was increased by more than an order of magnitude (Gibson et al., 1969). The beta radiation could have an effect on thin-walled detectors, if these were mistakenly used to measure gamma radiation (see Section 3.2).

The short-lived radioactivity of ^{140}Ba + ^{140}La (half-life 12.8 d) and ^{131}I (half-life 8.0 d) is succeeded by ^{95}Zr + ^{95}Nb (half-life 65 d), which dominates the gamma-ray dose rate in the medium term. The radionuclides ^{106}Ru + ^{106}Rh (half-life 365 d) and ^{125}Sb + ^{125}Te (half-life 2.37 yr) are the main contributors in the longer term and by 1978 the gamma-ray dose rate from fall-out had decreased to about 3 nGy/h (0.3 μrad/h), mainly from ^{137}Cs + ^{137}Ba (Gibson, 1978). The radioactivity is deposited on the surface and slowly penetrates into the ground with consequent decrease in the dose rate above (Gale et al., 1964). Soil type has some effect on the leaching rate and profiles of soil radioactivity with depth are needed for precise dose-rate calculations. To a first approximation, the activity in rain is independent of location in the UK, so the deposited radioactivity is proportional to rainfall (Peirson and Salmon, 1959) and calculations on this basis can be used to give an estimate of the local dose rate. Snow cover will reduce the gamma-ray emissions from the ground and the concentration of water in the soil also causes minor changes in density, which will result in an effect similar to that observed with natural radioactivity.

Typical variations from month to month are given in Figure 2.5-1 (Gibson, 1978) for a site at Chilton, Oxfordshire, with an average annual rainfall of about 650 mm (Cambray et al., 1978). These data have been computed from deposited fission products accumulated monthly, so that by correcting to mid-month some overestimate is made of the dose rate when short-lived fission products arrive late in the month. This was particularly evident in 1965 and 1966—7 but during the period of maximum deposition in 1961—3 good correlation was obtained with the measured dose rate (Gibson et al., 1969).

2.6 Nuclear installations

2.6.1 Main sources of gamma radiation: Contributions to the environmental gamma-ray dose rate from nuclear installations can arise from gaseous

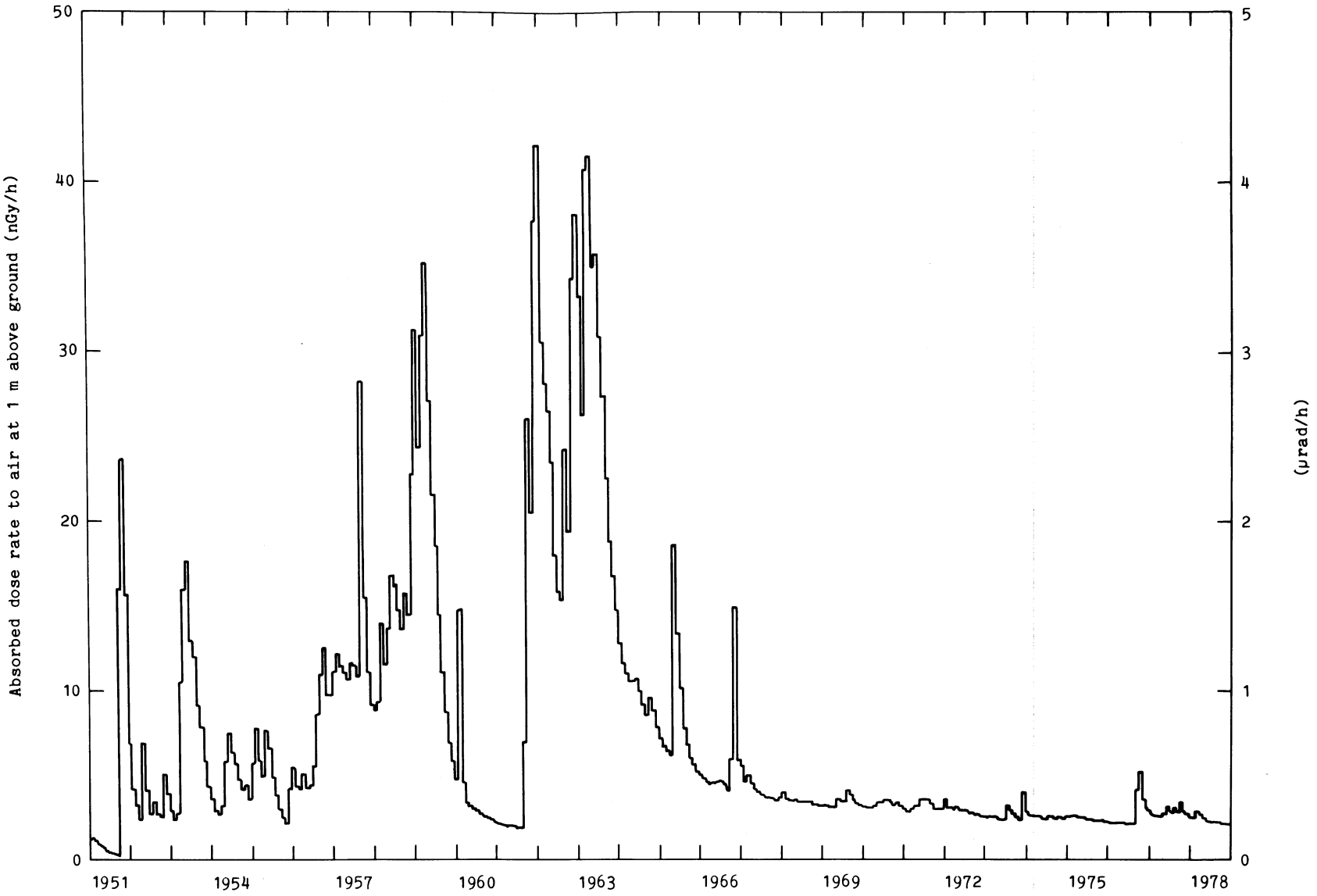

FIGURE 2.5-1: Gamma-ray dose rate calculated from deposited fall-out radioactivity at Chilton, Oxfordshire, between 1951 and 1978

or liquid effluents as well as directly from equipment and stored radioactive material. The thirteen installations now directly concerned with the nuclear-fuel cycle are situated in various parts of the UK, but their contributions to the environmental gamma-ray dose rate are normally confined to their immediate vicinities. Fuel fabrication for Magnox power reactors is undertaken at Springfields by British Nuclear Fuels Ltd (BNFL); the ^{235}U fuel enrichment for AGR reactors is carried out by BNFL at Capenhurst. Nine Magnox reactor power stations with a total installed capacity of approximately 4300 MW(electrical) and two AGR stations with 2640 MW(e) are already producing electricity; a further 3960 MW(e) from AGR stations will be commissioned during the next few years. The irradiated fuel from these power stations is reprocessed by BNFL at the Windscale and Calder works in Cumbria. Additional exposure arises from the operation of equipment and disposal of radioactive wastes from research establishments, hospitals and other industrial organizations. A detailed account of the disposal of radioactive waste in the UK and of the associated radiation exposure to the UK population is contained in a report by Taylor and Webb (1978).

2.6.2 Fuel fabrication: The airborne discharges of natural uranium from Springfields and of slightly enriched uranium from Capenhurst contribute very little to the gamma-ray dose rate. Liquid discharges of uranium are also very small.

2.6.3 Nuclear power reactors: During the operation of a nuclear reactor, radioactive fission and activation products are produced, most of which are retained within the fuel elements.

(a) Airborne effluents: Measures are taken to control discharges of gaseous radioactive effluent, but releases of certain specified radionuclides are authorized within regulatory limits. By far the most significant source of gaseous activity released from the earlier Magnox stations arises from neutron activation of ^{40}Ar present in the cooling air (1% of air) passed between the steel pressure vessel and the concrete biological shield of the reactor. The ^{41}Ar produced has a half-life of 1.83 h and emits a 1.29 MeV gamma ray in 99.2% of its disintegrations. The mean free path in air of such gamma rays is approximately 130 m and so, even with

elevated releases, significant dose rates at ground level can occur. Annual release rates for 1975 and 1976 have been determined by Wilson (1977). Clarke and Wilson (1977) calculated that the dose rate from ^{41}Ar at the site boundaries was from approximately 10 nGy/h to 100 nGy/h (1 μrad/h to 10 μrad/h), depending upon output and the dispersion conditions.

(b) Direct radiation from Magnox reactor buildings: The major source of direct radiation from the earlier Magnox reactor buildings is from the neutron activation of the oxygen of the carbon dioxide coolant gas within the reactor:

$$^{16}O\ (n,p)\ ^{16}N \xrightarrow[7\ s]{\beta} {}^{16m}O \longrightarrow \begin{matrix}\gamma\ 6.14\ \text{MeV}\ (68\%)\\ \gamma\ 7.12\ \text{MeV}\ (5\%)\\ \gamma\ 8.88\ \text{MeV}\ (1\%)\end{matrix} \longrightarrow {}^{16}O$$

This reaction has a neutron threshold of 10 MeV. Most of the dose rate is directly attributable to the 6.14 MeV photons but there are additional contributions from the continuous spectrum of Compton scattering by the materials of the heat exchanger and by the intervening air and ground. The maximum dose rate from this source occurs at Berkeley, where the reported dose rate is approximately 1.9 μGy/h (190 μrad/h) (Taylor and Webb, 1978) at a position on the perimeter fence; this is four times higher than at any other CEGB nuclear station. Fission noble-gas releases are insignificant in Magnox and AGR reactors since the fuel temperatures are low enough for their release from the fuel material to be negligible.

(c) Liquid effluents: Most of the activity in liquid effluents is produced by the spent-fuel storage ponds. Fission products that leach from the fuel as well as activated corrosion products contaminate the pond water. The liquid wastes are first treated before being released to the sea, or in the case of Trawsfynydd station to a freshwater lake after treatment. The maximum quantities that may be discharged are authorized by the appropriate government department. Details of authorized quantities for radionuclides (other than tritium) and the actual amounts discharged from 1971 to 1975 are given by Groom and Short (1979). The highest measured dose rate from the discharge of liquid effluents occurs at Trawsfynydd and is of the order of 10 nGy/h (1 μrad/h), the dose rates at other nuclear power stations being much lower.

2.6.4 Fuel reprocessing: In the UK, power reactor fuel is not reprocessed until at least 100 days after removal from the reactors. During reprocessing the uranium and plutonium in irradiated fuel are recovered for re-use.

(a) Airborne effluents: The most significant radionuclide contributing to environmental radiation is ^{85}Kr (half-life 10.76 yr) and from 1971 to 1976 typical releases per year have been 44 000 TBq (1.2 MCi) (Taylor and Webb, 1978). Bryant (1978) has derived conversion factors of gonad dose per unit activity discharge. Assuming average weather conditions, the dose rate at 1 km from a 30 m effective stack height for these releases would be of the order of 10 nGy/h (1 μrad/h).

(b) Liquid effluents: At Windscale and Calder Hall, liquid effluent is discharged into the sea 2 km from the coast. Environmental irradiation of members of the public may arise from marine sediments deposited on mud and sand along the coastline. Spectral measurements by Wilson (1978) in the vicinity of a nuclear power station, prior to its initial fuel loading, showed that the most prominent radionuclide was ^{137}Cs with a smaller contribution from ^{134}Cs. The presence of the caesium isotopes is attributed to the authorized discharges from Windscale. The contribution to the dose rate from ^{137}Cs and other fission products at the sites measured was 50 nGy/h to 230 nGy/h (5 μrad/h to 23 μrad/h), between about one half and twice the local natural environmental level there of approximately 100 nGy/h (10 μrad/h).

3 CHARACTERISTICS OF INDIVIDUAL INSTRUMENTS

In order to meet differing requirements and situations, various types of instruments may be used in the measurement of environmental gamma radiation. This chapter discusses the characteristics of Geiger—Müller (G-M) counters, ionization chambers, scintillators, spectrometers and thermoluminescence dosimeters, together with their methods of operation. The calibration of these instruments, the interpretation of the readings and the choice of instruments to suit particular circumstances are discussed later in Chapters 4, 5 and 6 respectively.

3.1 Geiger—Müller counters

3.1.1 Energy dependence: G-M counters do not measure dose rate directly and so their use is an essentially pragmatic approach to the problem. The measuring circuits are simple, the determination of the internal background and cosmic-ray response are straightforward and calibration presents no major problem. G-M counters measure the product of the fluence rate of the radiation and the interaction probability with the counter walls. The response per unit dose is thus enhanced at low energies relative to a calibration at about 1 MeV and some form of energy compensation is required (Whiting, 1965). A typical response is given in Figure 3.1-1 to demonstrate the compensation effect and to show that the detector response lies within $\pm$10% over the gamma-ray energy range of interest from 50 keV to 3 MeV. The shielding introduced by the filters for energy compensation is sufficient to eliminate any contribution from beta radiations.

3.1.2 Internal contamination: The internal background of a G-M counter is caused by traces of radioactivity in its components and is usually of similar order to the counting rate from environmental gamma radiation. By placing the test counter within a ring of G-M counters in a lead castle of thickness 10 cm, the internal background can be determined from pulses that are not in coincidence with the ring counters. Corrections are required for accidental coincidences (less than 1%) and for incomplete coverage by the ring (about 1%). The

internal background should not change significantly during the life of the counter because the radionuclides present have long half-lives, but occasional checks are advisable.

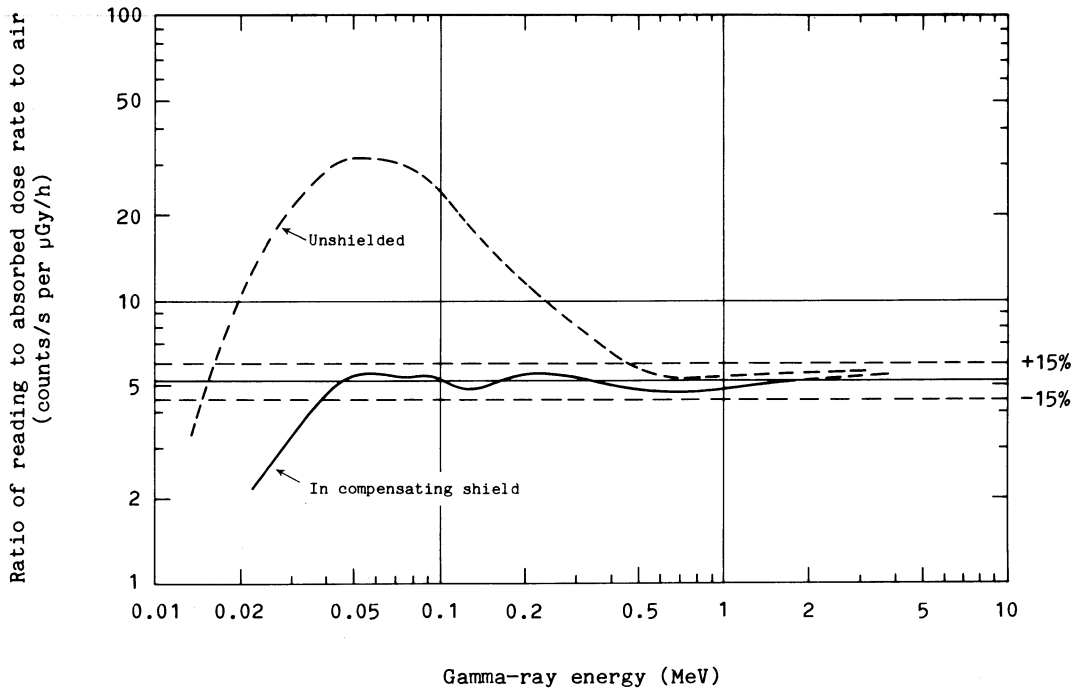

FIGURE 3.1-1: Energy dependence of a typical G-M counter (Whiting, 1965)

3.1.3 Cosmic-ray response: The hard component of cosmic radiation (mu mesons) can be determined from the same experiment as that used for measuring the internal background with similar correction terms. A small correction (about 5%) for attenuation in the 10 cm of lead is required. The soft component (electrons and photon radiation) can be determined from the curve of the ratio of soft to hard components as a function of barometric pressure (Figure 2.4-1).

The cosmic-ray fluence rate per unit solid angle is directional and is proportional to the square of the cosine of the zenith angle (Rossi, 1948), that is:

$$\varphi'_\theta = \varphi'_0 \cos^2\theta \ ,$$

where φ'_0 is the particle fluence rate per unit solid angle in the vertical direction and θ is the angle relative to the vertical (the zenith angle).

If ψ is the angle relative to magnetic north in the horizontal plane then, with no dependence upon ψ, the total fluence rate is

$$\varphi_F = \int_0^{\pi} \varphi_0' \cos^2\theta \, d\omega ,$$

where $d\omega = \sin\theta \, d\theta \, d\psi$ is the solid angle. Therefore

$$\varphi_F = \int_0^{\pi/2} \varphi_0' \cos^2\theta \sin\theta \, d\theta \int_0^{2\pi} d\psi$$

$$= \frac{2}{3}\pi\varphi_0' . \qquad (3.1\text{-}1)$$

The hard cosmic-ray dose rate, $\dot{D}_h$, is then

$$\dot{D}_h = \varphi_F(\frac{1}{\rho})(\frac{dE}{dl})_h \times 577 = \frac{2}{3}\pi\varphi_0'(\frac{1}{\rho})(\frac{dE}{dl})_h \times 577 \text{ nGy/h} , \qquad (3.1\text{-}2)$$

where $(1/\rho)(dE/dl)_h$ MeV cm^2/g is the mass stopping power for air and ρ is the air density. There is a similar expression to Equation 3.1-2 for the soft cosmic-ray dose rate $\dot{D}_s$, so that the total cosmic-ray dose rate is the sum of $\dot{D}_h$ and $\dot{D}_s$. Douglas (1981) has calculated $(1/\rho)(dE/dl)_h$ as 2.31 MeV cm^2/g and $(1/\rho)(dE/dl)_s$ as 2.00 MeV cm^2/g.

Since the efficiency for counting cosmic radiation is 100% then the count rate for a cylindrical detector is dependent upon its cross-sectional area and so for a cylindrical G-M detector placed with its axis vertical ($\theta = 0$)

$$\dot{C}(a) = \int_0^{\pi/2} \varphi_0' \cos^2\theta \, (2RL \sin\theta + \pi R^2 \cos\theta) \sin\theta \, d\theta \int_0^{2\pi} d\psi$$

$$= \frac{\pi}{2}\varphi_0'(\frac{\pi}{4} 2RL + \pi R^2) . \qquad (3.1\text{-}3)$$

With the detector axis horizontal in the direction $\psi = 0$, then its projected area is a function of the angle ϵ as shown in Figure 3.1-2, where

$$\epsilon = \cos^{-1}(\sin\theta \cos\psi) .$$

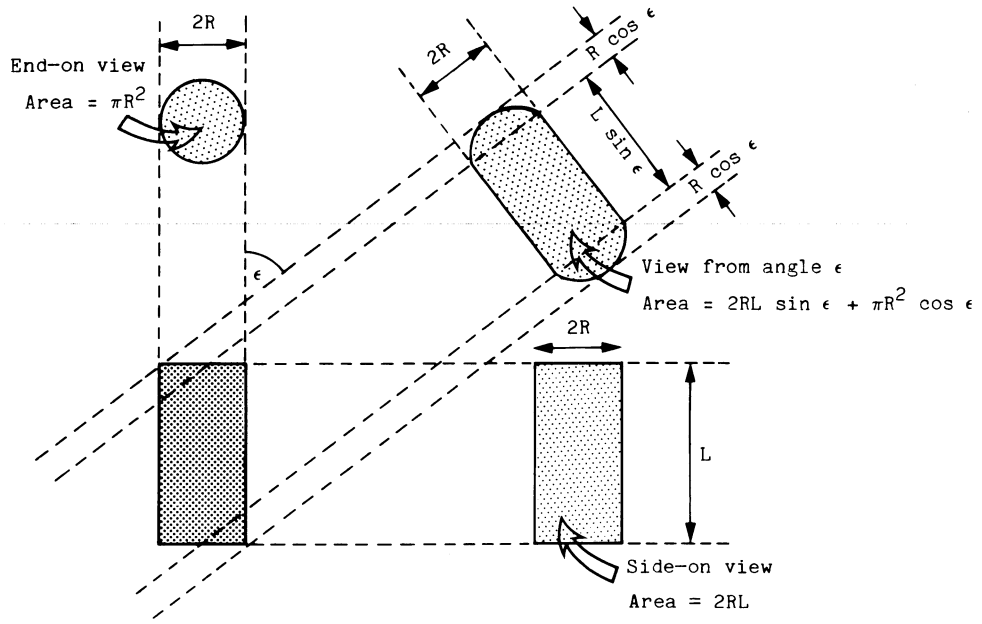

FIGURE 3.1-2: Effective area of a cylindrical detector

There is no complete analytical solution to the integral

$$\dot{C}(b) = \int_0^{\pi/2} \int_0^{2\pi} \varphi_0' \cos^2\theta \, (2RL \sin \epsilon + \pi R^2 \cos \epsilon) \sin \theta \, d\theta \, d\psi$$

but the first term may be integrated numerically to give the following expression for the count rate:

$$\dot{C}(b) = \frac{\pi}{2}\varphi_0' \, (1.243 \times 2RL + 0.5 \, \pi R^2) \, . \qquad (3.1\text{-}4)$$

Since the side area of a detector is normally much larger than the top end area, the cosmic-ray response is least when the axis of the detector is vertical.

Thus the cosmic-ray response, $K_c = \dot{C}/\dot{D}$, will depend upon the orientation of the detector and upon the barometric pressure since the value of $(1/\rho)dE/dl$ is different for hard and soft cosmic radiation. This variation with barometric pressure is small.

This theoretical approach can also be used for scintillation detectors.

3.1.4 Dead-time correction: Although dead-time corrections are negligible whilst measuring normal background dose rates, they are important during

calibration exposures. Dead time is fixed electronically by a quenching circuit and is determined by three measurements with two sources in which the measured count rates are $\dot{N}_1$, $\dot{N}_2$ counts/s for sources 1 and 2 separately and $\dot{N}_3$ counts/s for both sources counted together. If $\dot{N}_b$ is the background count rate and τ is the dead time (s), then the count rate $\dot{N}_i(0)$ from a source corrected for dead time and background will be given by

$$\dot{N}_i(0) = [\dot{N}_i/(1 - \dot{N}_i\tau)] - [\dot{N}_b/(1 - \dot{N}_b\tau)] , \qquad (3.1\text{-}5)$$

where i = 1, 2 or 3. The corrected count rate of both sources together must be equal to the sum of the corrected count rates of the separate sources, i.e.

$$\dot{N}_3(0) = \dot{N}_1(0) + \dot{N}_2(0) , \qquad (3.1\text{-}6)$$

The dead time τ can then be determined by substitution of Equation 3.1-5 in Equation 3.1-6 and solving a quadratic equation of the form

$$a\tau^2 - 2b\tau + c = 0 ,$$

where

$$a = \dot{N}_1\dot{N}_2(\dot{N}_3 + \dot{N}_b) - \dot{N}_3\dot{N}_b(\dot{N}_1 + \dot{N}_2) ,$$

$$b = \dot{N}_1\dot{N}_2 - \dot{N}_3\dot{N}_b ,$$

$$c = \dot{N}_1 + \dot{N}_2 - (\dot{N}_3 + \dot{N}_b) .$$

The dead time is then given by

$$\tau = \frac{1}{a}[b - \sqrt{(b^2 - ac)}] . \qquad (3.1\text{-}7)$$

This exact formula should be used in all calculations of τ.

When the dead time τ has been determined for a given instrument and the background count rate $\dot{N}_b$ has been measured in a given situation, the corrected count rate $\dot{N}(0)$ in that situation can be calculated from the measured count rate $\dot{N}$ by means of Equation 3.1-5.

Typical values for three G-M counters used in parallel to obtain adequate sensitivity are as follows:

gamma-ray response: 24 counts/s per μGy/h;

cosmic-ray response: 41 counts/s per μGy/h;

internal background: 1.3 counts/s;

dead time, τ: 2.0 ms.

Thus if a calibration were carried out using a gamma-ray source that produced an absorbed dose rate to air of 10 μGy/h (1 mrad/h) at the point of measurement, the count rate corrected for dead time and internal background, $\dot{N}_i(0)$, would be 24 x 10 = 240 counts/s. The background count rate from cosmic radiation and internal background, $\dot{N}_b$, would be 2.6 counts/s, and hence from Equation 3.1-5:

$$240 = \dot{N}_i/(1 - 0.002\,\dot{N}_i) - 2.6 ,$$

from which it follows that the measured count rate $\dot{N}_i$ would be only 164 counts/s, or 0.68 of the corrected count rate $\dot{N}(0)$.

This difference is quite large and emphasizes the need for an accurate measurement of τ at the count rates used for calibration. The uncertainties in τ can also be quite large unless care is taken with the counting. For example taking the above values and counting to $\pm$2% statistics (10^4 counts) for

$\dot{N}_b$ = 1 count/s,

$\dot{N}_1(0) = \dot{N}_2(0)$ = 240 counts/s and

$\dot{N}_3(0)$ = 480 counts/s,

then the uncertainty in τ would be $\pm$27% (95% confidence limits). If $\dot{N}_1$, $\dot{N}_2$ and $\dot{N}_3$ are counted to 0.2% (10^6 counts) and the uncertainty in $\dot{N}_b$ is still $\pm$2% then the uncertainty in τ is reduced to $\pm$4%.

3.1.5 Calibration: Calibration will be discussed in Chapter 4 and the only additional point to note is that a dead-time correction should be applied to the count rates used for calibration. An actual installed system could be as indicated in Figure 3.1-3; this would provide direct measurement of the dose from hard and soft cosmic radiation (with atmospheric gamma radiation), terrestrial gamma radiation and all sources. The horizontal configuration is more convenient when all the components are to be measured together. A measure of barometric pressure is also to be recommended. A simpler system may be suitable for spot measurements.

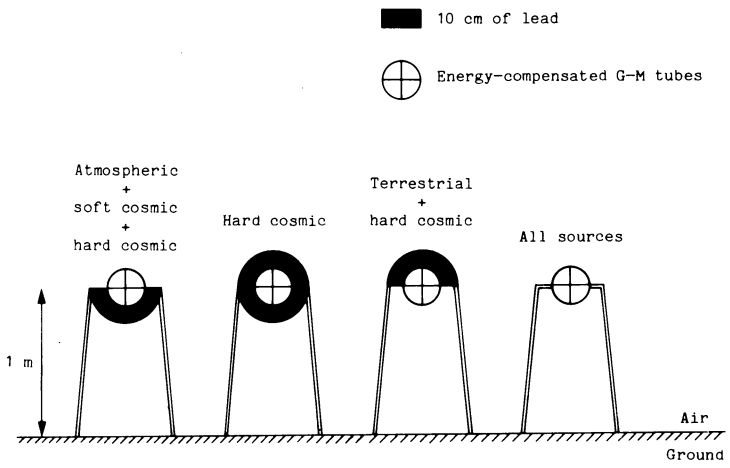

FIGURE 3.1-3: Arrangement of G-M detectors and shields for measuring environmental radiation at a fixed site

3.2 Ionization chambers

3.2.1 Types of ionization chamber: Ionization measurements of gamma-ray dose rates should be made under conditions of electron equilibrium and the wall thickness necessary for this also excludes ionization from beta particles and secondary electrons arising outside the chamber itself. In ionization chambers at atmospheric pressure, the air-equivalent or tissue-equivalent walls provide most of the secondary ionizing electrons. In high-pressure ionization chambers, the walls play a lesser part because a large fraction of the ionization arises from gamma-ray absorption in the high-pressure gas.

Air-filled ionization chambers with approximately air-equivalent walls have been used as the commonest form of dosemeter. Ionization

chambers associated with radiation protection instruments, however, are not large enough to make satisfactory measurements at dose rates of about 50 nGy/h (5 μrad/h) and resort must be made to large chambers at atmospheric pressure or to high-pressure ionization chambers. Thus Kastner et al. (1964) used a 16 litre tissue-equivalent plastic chamber, filled with a tissue-equivalent gas at atmospheric pressure, and Shamos et al. (1964) used a 74 litre chamber filled with Freon gas (CCl_2F_2) at atmospheric pressure for the measurement of background gamma radiation and cosmic radiation. High-pressure ionization chambers were used as early as 1932 to measure cosmic-ray intensity and among the more recent users of pressurized chambers are Burch (1954), Sievert and Hultqvist (1957), Burch et al. (1964) and Spiers et al. (1964). Typical of portable equipment are the chambers used by Spiers et al. (1964), 5.5 litres filled with nitrogen at 46 bar (45 atm), and by Shambon et al. (1963), 8 litres filled with argon at 30 bar (30 atm). The energy dependence of these atmospheric and pressurized chambers depends chiefly on their wall material and thickness; the plastic chambers, with walls some 6—10 mm thick, have reasonably flat energy characteristics whereas at low energies there is a considerable fall in response of the steel-walled high-pressure chambers relative to that at about 1 MeV. For example the response of the portable steel chamber used by Spiers et al. (1964) (12.5 mm steel wall) for ^{131}I gamma radiation (mean energy 0.35 MeV) is 83% of that for ^{226}Ra gamma radiation (mean energy 0.84 MeV). Provided proper calibration is carried out, this variation may well be adequate for environmental gamma-ray monitoring (see Section 4.4.3 for energy dependence).

3.2.2 Internal contamination: Internal radioactive contamination can be a serious problem with ionization chambers used at environmental dose rates. In chambers at atmospheric pressure, alpha radiation from the natural radioactivity in the walls can produce ionization amounting in a 1 litre chamber to as much as one third of that produced by external environmental gamma radiation (Shamos et al., 1964). These workers overcame the internal contamination effect by using a large chamber (since for a sphere the ratio of surface area to volume is inversely proportional to the radius) filled with a dense electron-attaching gas (Freon). This promotes columnar recombination in the alpha-particle tracks, reducing the collected ionization from this source compared

with that from the less densely ionizing secondary electrons released by the absorbed gamma radiation. In high-pressure chambers a high recombination in the alpha-particle tracks is produced by the pressure itself and the use of an electron-attaching gas such as nitrogen. The alpha-particle tracks are short and are therefore confined to small distances from the chamber wall where the collecting field is weak and recombination high.

3.2.3 Measurement of ionization current: The ionization currents at environmental levels are small even in large chambers, e.g. 10^{-13} A in a 5.5 litre chamber at 46 bar (45 atm), and their measurement requires low-drift amplifiers and very stable input resistors of high value, typically of the order of 10^{12} Ω, if long-term stability is to be achieved. Alternatively the rate of drift of voltage across a capacitance in the input circuit avoids the use of a high-value resistor and involves only the capacitor stability, which is greater and more easily achieved than that of the resistor. The rate-of-drift method can be operated via a balance circuit incorporating an electrometer valve, a vibrating-reed electrometer or a modern solid-state amplifier. Where present, grid currents that may add to or subtract from the true ionization currents must be small, e.g. 10^{-15} A or less, and the effects of battery drift must be minimized and be capable of being checked and compensated for in portable or battery-operated equipment. In operating high-pressure chambers the polarizing voltage must be very stable since true 'saturation' of the ionization current cannot be attained. Leakage across the collecting electrode insulator can easily be made negligible with modern insulators, e.g. several orders of magnitude below the ionization current, but this should not be simply taken for granted and appropriate tests should be made to exclude it. A particularly stable system was used by Kastner et al. (1964) in which a vibrating-quartz-fibre electrometer (Shonka, 1962) was coupled directly to the collecting electrode of their 16 litre tissue-equivalent chamber.

3.2.4 Cosmic-ray response: In all ionization chambers the ionization produced by cosmic radiation represents a considerable fraction of the background reading. For example, using a rate-of-drift method and a 5.5 litre chamber filled with nitrogen at 46 bar (45 atm), Spiers

et al. (1964) found the hard cosmic-ray component (mu mesons) produced a drift rate of 0.5 V/min on the measuring electrometer compared with about 1 V/min from an external gamma-ray field of 57 nGy/h (5.7 μrad/h). The total cosmic-ray drift rate (or ionization current) in a large ionization chamber can be determined by measurements made in a light wooden or plastic boat on a freshwater lake or reservoir at about a quarter to a half mile from land (Spiers et al., 1964; Kastner et al., 1964; Shamos et al., 1964). Alternatively the response to the hard component of the cosmic radiation can be measured in a small steel cubicle, which excludes any local gamma radiation (e.g. a cubicle about 100 cm x 30 cm x 30 cm with steel walls and door of thickness 12.5 cm) and the total cosmic-ray response deduced from the known ratio of soft to hard cosmic radiation (Figure 2.4-1) at a given barometric pressure (Rossi, 1948). It is important that this cosmic-ray response be subtracted from the instrument reading before applying any gamma-ray calibration factor.

3.3 Scintillators

3.3.1 Types of instruments: Scintillation detectors are widely used for environmental dose-rate measurements. As they are very much denser than gas-filled ionization chambers a significantly smaller detector gives comparable sensitivity resulting in a more portable instrument. Frequently scintillators are used for spectrometry to identify individual environmental radionuclides and to estimate their contribution to the total dose rate. (The use of scintillators as spectrometers is discussed more fully in Section 3.4.) Differences in the mass energy absorption coefficients of the scintillator materials compared with those of air produce a dose response that is energy dependent.

Scintillation detectors can be operated in either the current mode or the pulsed-operation mode. In the current mode the instrument reading will be proportional to the energy converted into light within the scintillator, which in turn will be proportional to the total energy absorbed in the detector. Most instruments are, however, operated in the pulsed mode and measure count rate or integrated counts. In the pulsed mode the fast response of the scintillator with short decay times can be utilized and all pulses will be counted irrespective of

their energy provided that they exceed a certain bias level. With such a simple counting system no information is obtained concerning the energy absorbed. To determine the energy absorbed it is necessary to employ techniques to sum the number of pulses in proportion to their pulse heights.

3.3.2 Inorganic scintillators: Inorganic scintillators such as NaI(Tl) have an enhanced response to photons with energies between approximately 30 keV and 600 keV mainly because their effective atomic number is greater than that of air. At high energies above a few MeV they have a reduced response owing to the small size of the detector. Figure 3.3-1 shows a typical energy-dependence curve for a commercial dosemeter. At 100 keV the instrument has twenty times the response, whereas at 6 MeV it has only a third of the response, that it has for ^{226}Ra gamma radiation (Thompson et al., 1971).

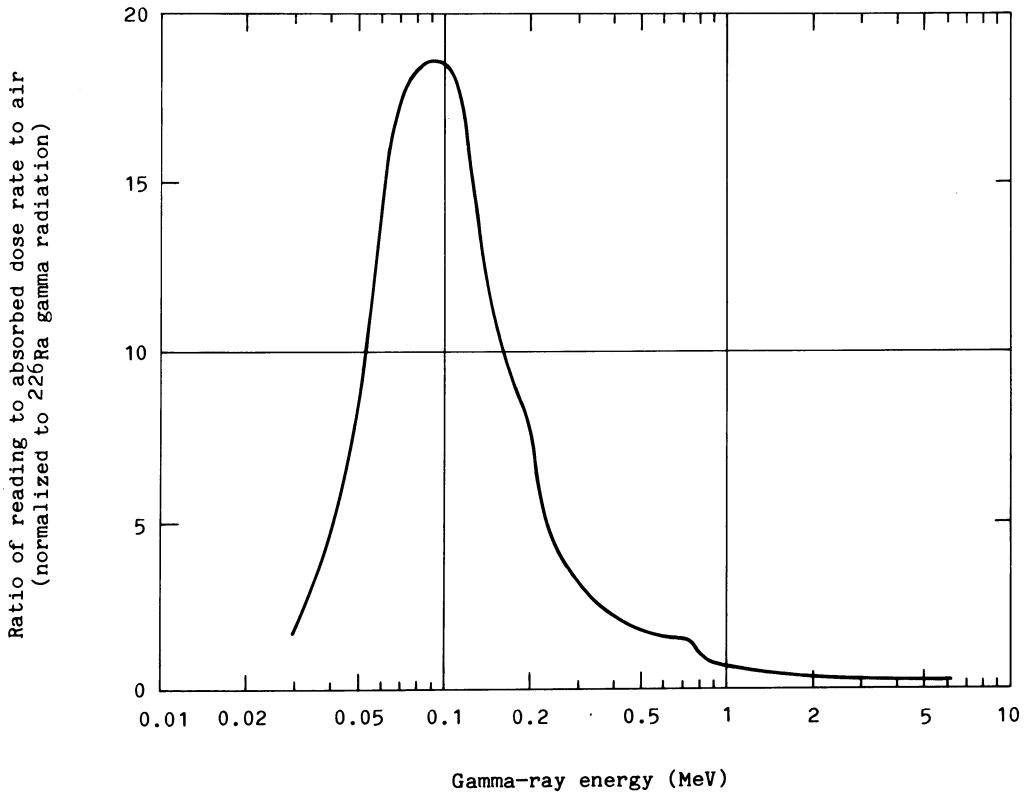

FIGURE 3.3-1: Energy dependence of a portable scintillation ratemeter based upon NaI(Tl) of size 25 mm x 38 mm dia.

Significant errors can thus occur if inorganic scintillators are calibrated using ^{137}Cs, ^{60}Co or ^{226}Ra sources and then used for natural

environmental measurements. The softer environmental spectra would lead to an overestimate of the dose rate. If ^{137}Cs or ^{60}Co is used for the calibration then the response to either of these sources is considerably smaller than that to the natural photon environment and for accurate measurements it would be necessary to correct for this difference (see Section 4.4.3). Without such corrections overestimates of field dose rate by factors of 1.5 to 2.0 are likely (Gudiksen and Crites, 1976). For measurements close to power reactors producing significant amounts of ^{16}N, the scintillator will significantly underestimate the dose from the resulting 6 MeV photons.

Inorganic scintillators have not only a short-term fading fluorescence but also a long-term phosphorescence. If they are irradiated at high dose rate prior to being used for environmental measurements the retained luminescence can lead to falsely high readings.

The response of a 44 mm x 44 mm NaI(Tl) scintillator to cosmic radiation has been investigated by Pensko (1967). At altitudes of about 100 m and in a gamma-ray background of approximately 60 nGy/h (6 μrad/h) the cosmic-ray contribution to the instrument reading was 6% (with the lower-level discriminator set at 0.3 MeV), compared with contributions of 30—40% for ionization chambers (Spiers, 1960; Herbst and Hubner, 1961). An upper-level discriminator set at about 3 MeV is normally used to cut out the larger mu-meson pulses.

3.3.3 Organic scintillators: These scintillators are more suited for dose-rate measurements, particularly for energies above 100 keV, since their mass energy absorption coefficient corresponds approximately to that of air. Below 100 keV the response falls off rapidly with decreasing photon energy but by coating the scintillator with a thin layer of high-atomic-number material such as ZnS, the air-equivalent response can be extended down to 20 keV. At low photon energies the fluorescent light is produced solely in this ZnS layer whereas above 100 keV the gamma and X radiation penetrates the ZnS layer and the light is produced chiefly in the thick organic scintillator.

Homogeneous incorporation of the ZnS within the plastic scintillator is restricted to small detectors since low-energy photons will not

penetrate to the ZnS within the inner layers of a large crystal (Belcher and Geilinger, 1957). Kolb and Lauterbach (1974) developed a ZnS(Ag)-loaded anthracene detector, which is approximately air equivalent from 20 keV to 2 MeV (Figure 3.3-2), whilst Chester et al. (1972) employed a tin-loaded scintillator with digital read-out.

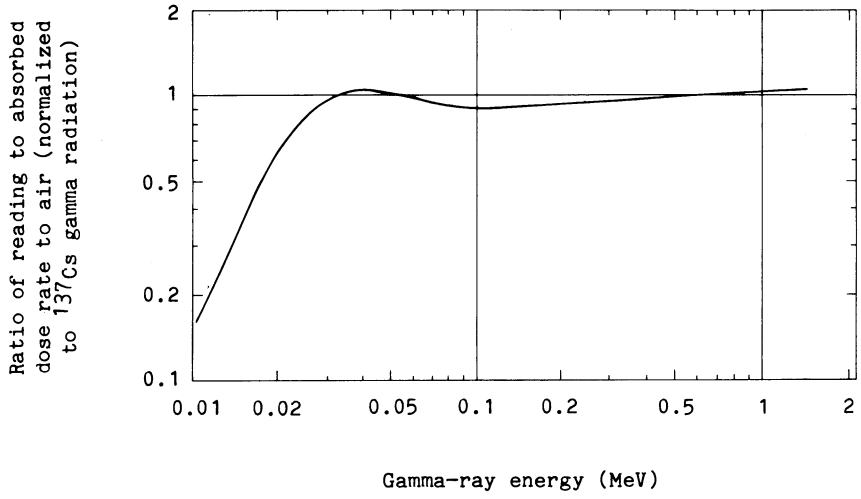

FIGURE 3.3-2: Energy dependence of an instrument based upon a plastic scintillator (NE 102) surrounded by a layer of ZnS(Ag) (Kolb and Lauterbach, 1974)

3.3.4 Lower limit of detection: In scintillation detectors the dark current of the photomultiplier tube limits the lowest dose rate that can be measured. Low-noise photomultiplier tubes should always be used, and operated so that the optimum ratio of signal to dark current is obtained. This can be achieved by careful selection of the operating voltage or by employing a cooling device. The dark current increases with rise in temperature (approximately +20% for a temperature change from +10 ^{o}C to +30 ^{o}C) and the photomultiplier amplification also increases by approximately 0.7% per degree Celsius. For maximum sensitivity temperature stabilization is desirable and Kolb and Lauterbach (1971), by cooling to +3 ^{o}C, could measure dose rates of less than 0.4 nGy/h (40 nrad/h).

A novel pulse system was developed by Roberts and Speight (1964) to reduce the effect of photomultiplier dark current. The pulses from a ZnS-loaded plastic phosphor were superimposed on a free-running triangular waveform and the discriminating bias level was set above the apex of the triangular waveform. This ensured that noise pulses were

discriminated against and only pulses arising from photons with energies above 30 keV were counted.

Inherent activity of the detector material can impose a limitation on the sensitivity. The major contribution is ^{40}K activity in the scintillator and in the glass of the photomultiplier; this can be reduced by means of special potassium-free scintillators and photomultiplier tubes, and by means of mountings with quartz windows. Aluminium should be avoided for the scintillator mounting and electrolytically purified copper, which has low inherent activity, should be used instead.

3.4 Gamma-ray spectrometers

3.4.1 Methods of use: Conventional dosemeters employing ionization chambers, scintillation crystals and G-M tubes as their detectors provide a total estimate of the absorbed dose rate from all causes. They do not usually enable the dose rates to be estimated from different components of the radiation field. Statistical techniques have been developed that permit discrimination to be made between the dose rate from nuclear-installation emissions and the contribution from natural causes (McLaughlin, 1976). To identify and determine the contribution to the total dose rate from individual radionuclides in the environment, in situ gamma-ray spectrometry must be used (Finck et al., 1976; Gogolak and Miller, 1977; Beck et al., 1972; Dickson et al., 1976; Ragaini et al., 1974).

All spectrometry measurements made to date use either NaI(Tl) or Ge(Li) detectors. Contributions from individual radionuclides to either the dose rate or the soil activity are determined from the measured areas of the total-absorption peaks in the energy spectrum. Since the ratio of the total-absorption peak to the Compton continuum increases with the atomic number and the size of the detector, it is advantageous to use large-volume detectors of high-atomic-number material. The reduction in resolution with increasing detector volume is small and is not a serious disadvantage. For the identification of most environmental radionuclides the detector and spectrometry system, amplifiers and multichannel analyser should cover an energy range up to 2.7 MeV. When contributions of the direct radiations from water-moderated or

gas-cooled reactors have to be measured it may be necessary to extend this range to provide identification up to the 8.9 MeV photons from ^{16}N (see Figure 3.4-1).

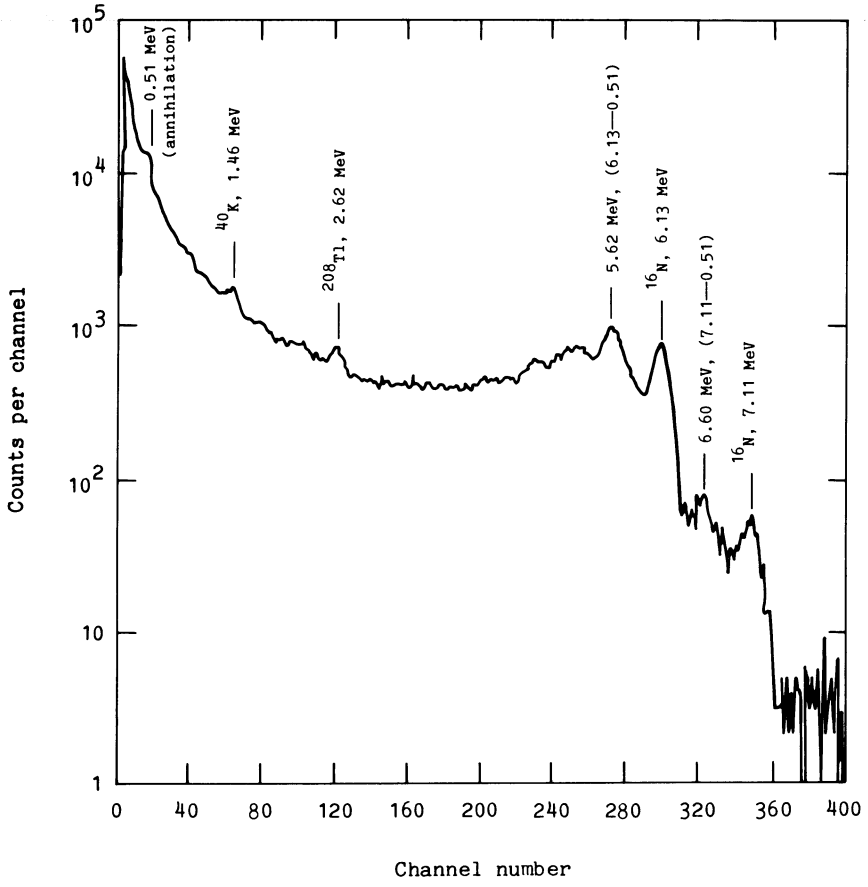

FIGURE 3.4-1: In situ NaI(Tl) spectrum 200 m from turbine of a 1600 MW(thermal) boiling-water reactor (Beck et al., 1972)

In situ gamma-ray spectrometry is most frequently used for measuring the concentrations in soil of natural or fall-out radionuclides as well as airborne and deposited radionuclides released from nuclear installations. Since an unshielded NaI(Tl) or Ge(Li) detector situated 1 m above the earth detects gamma radiation from an area of about 500 m^2 the in situ measurement requires significantly less effort than soil-sampling techniques. For the in situ soil concentration measurements information is required on soil density, moisture and the variation in concentration of radionuclides with soil depth. This Guide is, however, restricted to the measurement of dose rates, and the conversion factors used to calculate dose rate from the number of photons incident upon the detector are

less sensitive to a change of source distribution than are the factors used to determine soil activity. Beck et al. (1972) have shown that the ratio of photon fluence rate to dose rate varies by only 25—30% between a plane source and a deeply distributed source, and that it is relatively insensitive to variations in air density, to water content, to mass attenuation coefficient and to the soil density.

An undisturbed nearly flat site should be chosen for the measurements and the soil should be typical of the surrounding area. The site should be monitored with a portable dosemeter to ensure that the point of measurement is typical of the whole site.

3.4.2 Theory: Very few detectors used for gamma-ray spectrometry have a uniform response to radiation from different directions and so it is necessary to derive a calibration factor for environmental radiation since it will be different from that for a single point source located along the axis of a cylindrical detector. If the fluence rate per unit solid angle is $\varphi'(\theta, \psi)$, then for a detector placed above the ground the total fluence rate is

$$\varphi_F = \int_0^{2\pi} \int_0^{\pi/2} \varphi'(\theta, \psi) \sin \theta \, d\theta \, d\psi \, , \qquad (3.4\text{-}1)$$

where θ is the angle with the vertical and the limits of integration are taken as 0 to $\pi/2$ since there is no significant photon fluence rate from the upper half-space and only primary gamma rays are being counted; ψ is the angle in the horizontal plane (taken relative to magnetic north to be consistent with Section 3.1.3). It is assumed that there is no variation with ψ, since a site will normally be chosen to avoid anomalies, and so the integral becomes

$$\varphi_F = 2\pi \int_0^{\pi/2} \varphi'_\theta \sin \theta \, d\theta \, . \qquad (3.4\text{-}2)$$

The variation of φ'_θ with angle will depend upon the variation of radioactivity with depth below the ground. It can be shown from the data of Beck et al. (1972) (Table 6, page 48 of that reference) that for a thick source

$$\frac{d\varphi_F}{d(\cos\theta)} = \varphi'_\theta \simeq \text{constant},$$

and so some simplifying assumptions may be possible in using Equation 3.4-2.

The main purpose of using a gamma-ray spectrometer is to measure the count rate in a total-absorption peak in order to derive the dose rate $\dot{D}$ from a particular radionuclide or from a series of radionuclides in a natural decay chain such as those from uranium and thorium. The count rate $\dot{N}_F$ in a chosen total-absorption peak in the observed pulse-height distribution is related to the total dose rate through the response K_F, such that

$$\dot{D} = \dot{N}_F / K_F \ . \qquad (3.4\text{-}3)$$

The response K_F in counts/s per nGy/h has to be determined for each total-absorption peak by measurement or it can be calculated by making assumptions about the variation of fluence rate with angle. Equation 3.4-3 may be rewritten as

$$K_F = \frac{\dot{N}_F}{\dot{D}}$$

$$= (\dot{N}/\varphi)_0 \frac{(\dot{N}/\varphi)_F}{(\dot{N}/\varphi)_0} (\varphi_F/\dot{D}) \ . \qquad (3.4\text{-}4)$$

The first term $(\dot{N}/\varphi)_0$ in Equation 3.4-4 represents the efficiency of the detector expressed as the count rate in the total-absorption peak per unit fluence rate of gamma radiation of the energy of that peak along the axis of symmetry of the detector (the detectors are usually cylindrical). The variation of $(\dot{N}/\varphi)_0$ with energy can be determined with a series of standardized point sources at various distances from the detector to establish the effective detector centre to which subsequent measurement can be related. The area of the total-absorption peak is determined by fitting the continuum with an exponential or a linear function and then calculating the area above the fitted curve.

The second term in Equation 3.4-4 is used to correct for the

directional dependence of the detector and is defined by Beck et al. (1972) as

$$P_F = \frac{\dot{(N/\varphi)}_F}{\dot{(N/\varphi)}_0} . \qquad (3.4\text{-}5)$$

It is necessary to calculate P_F as a function of gamma-ray energy. With the axis of the detector vertical ($\theta = 0$),

$$P_F = \frac{1}{\dot{(N/\varphi)}_0} \frac{\int_0^{\pi/2} \dot{(N/\varphi)}_\theta \, \varphi'_\theta \sin\theta \, d\theta \int_0^{2\pi} d\psi}{\int_0^{\pi/2} \varphi'_\theta \sin\theta \, d\theta \int_0^{2\pi} d\psi} ,$$

and so, from Equation 3.4-2,

$$P_F = \frac{1}{\dot{(N/\varphi)}_0} \frac{\int_0^{\pi/2} \dot{(N/\varphi)}_\theta \, \varphi'_\theta \sin\theta \, d\theta}{\varphi_F/2\pi} . \qquad (3.4\text{-}6)$$

When the axis of the detector is horizontal ($\theta = \pi/2$), the angle ϵ with the axis of the detector is given by

$$\epsilon = \cos^{-1}(\sin\theta \cos\psi) ,$$

and P_F is a complex function of θ and ψ. Therefore it is suggested that the spectrometer is always used with its axis vertical.

The directionally dependent term P_F is normally not significantly different from unity and the uncertainties in the branching ratios and the distribution of radioactivity in the soil will introduce far larger uncertainties in the final estimate of the dose rate.

As mentioned earlier, for a uniformly distributed source the differential fluence rate φ'_θ is constant, that is

$$\frac{d\varphi_F}{d(\cos\theta)} = \varphi'_\theta \simeq \text{constant},$$

and integrating φ'_θ with respect to cos θ shows that more than 50% of the fluence rate comes from an angle greater than 60°. For freshly deposited fall-out it can be shown from Beck et al. (1972) that

$$\frac{d\varphi_F}{\tan\theta \, d\theta} = \varphi'_\theta \simeq \text{constant},$$

and more than 50% of the fluence rate comes from angles greater than 72°.

The final term in Equation 3.4-4, $\varphi_F/\dot{D}$, is the ratio of the fluence rate at the detector due to gamma radiation of a given energy to the total dose rate from a radionuclide or a radioactive series. This term is independent of the detector and calculated values produced by Beck et al. (1972) are given in Table 3.4-1(b) for natural radionuclides and in Table 3.4-2 for fall-out radionuclides. It is assumed that the depth distribution of the particular radionuclide is of the form

$$S/\rho = (S_0/\rho) \exp [-(\alpha/\rho)(\rho Z)] ,$$

where S_0/ρ is the radioactive concentration in the surface soil, Z is the depth, ρ is the soil density and $1/\alpha$ is the depth at which the radioactive concentration is reduced to e^{-1} or 0.37 of its surface value.

Natural radionuclides are assumed to be uniformly distributed so that $\alpha/\rho = 0$. Freshly deposited fall-out radionuclides are assumed to form a plane source so that $\alpha/\rho = \infty$; as the material is weathered into the soil values of α/ρ are reduced to between 0.01 m^2/kg and 0.02 m^2/kg.

TABLE 3.4-1(a): Dose rates and exposure rates at 1 m above ground per unit radioactive concentration in soil for natural radionuclides

Radionuclides	Dose rate		Exposure rate
	$(\frac{nGy/h}{Bq/kg})$	$(\frac{\mu rad/h}{pCi/g})$	$(\frac{\mu R/h}{pCi/g})$
^{40}K	0.042	0.156	0.179
^{232}Th + daughters	0.662	2.45	2.82
^{238}U + daughters	0.427	1.58	1.82

TABLE 3.4-1(b): Ratio of fluence rate of photons of energy E to total dose rate from natural radionuclides distributed uniformly in the soil, measured at 1 m above ground

Nuclide	Photon energy E (keV)	$\dot{\varphi}_F/\dot{D}$ ($\frac{\text{photons/m}^2\ \text{s}}{\text{nGy/h}}$)	
Uranium series			
^{226}Ra	186	2.19	
^{214}Pb	242	4.96	
	295	13.9	
	352	28.7	
^{214}Bi	609	45.0	46.6
	666	1.62	
	768	5.59	
	934	3.87	
	1120	20.1	
	1238	8.21	
	1378	7.12	
	1401—8	5.97	
	1510	340	
	1730	4.87	34.3
	1765	25.7	
	1845	3.78	
	2205	9.3	
	2448	3.18	
Thorium series			
^{228}Ac	129	0.90	
	210	1.79	
^{212}Pb, ^{224}Ra	239, 241	22.3	
^{228}Ac, ^{208}Tl, ^{228}Ac	270, 277, 282	3.15	
^{212}Pb	301	1.70	
^{228}Ac	338	6.71	
Mixed	328—40	9.0	
^{228}Ac	463	2.83	
^{208}Tl	510	5.94	
	583	19.7	
^{212}Bi, ^{228}Ac	727	5.74	
^{228}Ac	755	0.83	
	772	1.26	
	795	3.69	
	830 + 835 + 840	2.89	
^{208}Tl	860	3.63	
^{228}Ac	911	23.3	
	965 + 969	18.9	
	1588	3.79	
^{208}Tl	2615	51.4	
Potassium-40			
^{40}K	1464	176	

TABLE 3.4-2: Ratio of fluence rate of photons of energy E to total dose rate from fall-out radionuclides distributed exponentially in the soil, measured at 1 m above ground

Nuclide	Photon energy E (keV)	Nuclide distribution parameter α/ρ (m^2/kg)					
		0.006	0.021	0.031	0.062	0.62	∞
		$\dot{\varphi}_F/D$ (photons/m^2 s per nGy/h)[1]					
^{144}Ce	134	904	1030	1140	1290	1710	1860
^{144}Ce + ^{144}Pr	134	306	394	438	501	678	728
^{141}Ce	145	1020	1230	1320	1510	1940	2130
^{131}I	364	386	470	508	553	610	702
^{125}Sb	428	129	154	168	176	213	225
^{140}La	487	40	49	53	56	66	72
^{140}Ba + ^{140}La	487	32	39	43	45	54	58
^{103}Ru	497	362	437	459	499	597	621
^{106}Ru + ^{106}Rh	512	218	263	277	295	348	364
^{140}Ba	537	257	306	321	348	404	422
^{140}Ba + ^{140}La	537	18	22	23	24	29	32
^{125}Sb	601	90	105	111	117	139	144
^{103}Ru	610	24	28	29	31	37	38
^{106}Ru	622	112	133	139	148	172	180
^{137}Cs + ^{137}Ba	662	328	382	404	434	506	527
^{95}Zr	724	135	157	167	179	209	215
^{95}Zr + ^{95}Nb	724	41	48	51	55	64	66
^{95}Zr	757	170	200	210	230	263	271
^{95}Zr + ^{95}Nb	757	52	62	65	71	80	84
^{95}Nb	766	301	355	372	411	461	479
^{95}Zr + ^{95}Nb	766	208	245	258	285	322	331
^{140}La	816	25	29	31	32	38	40
^{140}Ba + ^{140}La	816	20	24	25	26	30	32
^{54}Mn	835	289	330	348	381	432	442
^{60}Co	1173	110	127	133	140	159	163
^{60}Co	1333	116	131	138	146	163	166
^{140}La	1597	134	144	150	152	169	176
^{140}Ba + ^{140}La	1597	108	116	122	123	137	142

[1] To convert to photons/cm^2 s per μrad/h, divide all values of $\dot{\varphi}_F/D$ by 1000

Thus the use of a gamma-ray spectrometer is mainly dependent upon a careful calibration with point sources along the axis of the detector to establish $(\dot{N}/\varphi)_0$ followed by the use of Tables 3.4-1 and 3.4-2 to calculate the dose rate.

An alternative approach is to calibrate the detector directly in terms of the dose rate such that from Equation 3.4-3

$$K_F = \frac{\dot{N}_F}{\dot{D}} = \frac{(\dot{N}/\varphi)_0}{\dot{D}/\varphi_F} \frac{(\dot{N}/\varphi)_F}{(\dot{N}/\varphi)_0},$$

that is

$$K_F = \frac{(\dot{N}/\varphi)_0}{\dot{D}/\varphi_F} P_F . \qquad (3.4\text{-}7)$$

The total dose rate can then be calculated by adding the dose rate from all significant total-absorption peaks. The uncertainties of this method are greater in that it is difficult to determine the first term in Equation 3.4-7 and allow for the scattered radiation, which does not produce an identifiable peak in the gamma-ray spectrum.

3.4.3 Ge(Li) systems: A Ge(Li) system has a resolution markedly superior to that of a NaI(Tl) system and should be used where it is necessary to discriminate between the contribution of small quantities of man-made radionuclides and that of the complex natural background spectrum (Figure 3.4-2). To analyse the data produced by a Ge(Li) detector and its associated electronics system a multichannel analyser of 4000 channels should be used. The high resolution of these systems means that gamma rays separated by only a few keV can be analysed without interference from each other and further peaks from the same transition can often be measured individually.

Disadvantages of the Ge(Li) system are its high cost, the need to maintain the detector permanently at liquid-nitrogen temperatures, and the large amount of data that may be produced.

Hyperpure or intrinsic germanium detectors have the advantage that they

can be cycled between ambient temperatures and liquid-nitrogen temperatures without damage to their performance. However since they require a lengthy cool-down period before adequate detector performance is obtained they are best kept cooled, especially if rapid use is required. Intrinsic detectors are expensive and to date have not been used for in situ measurements.

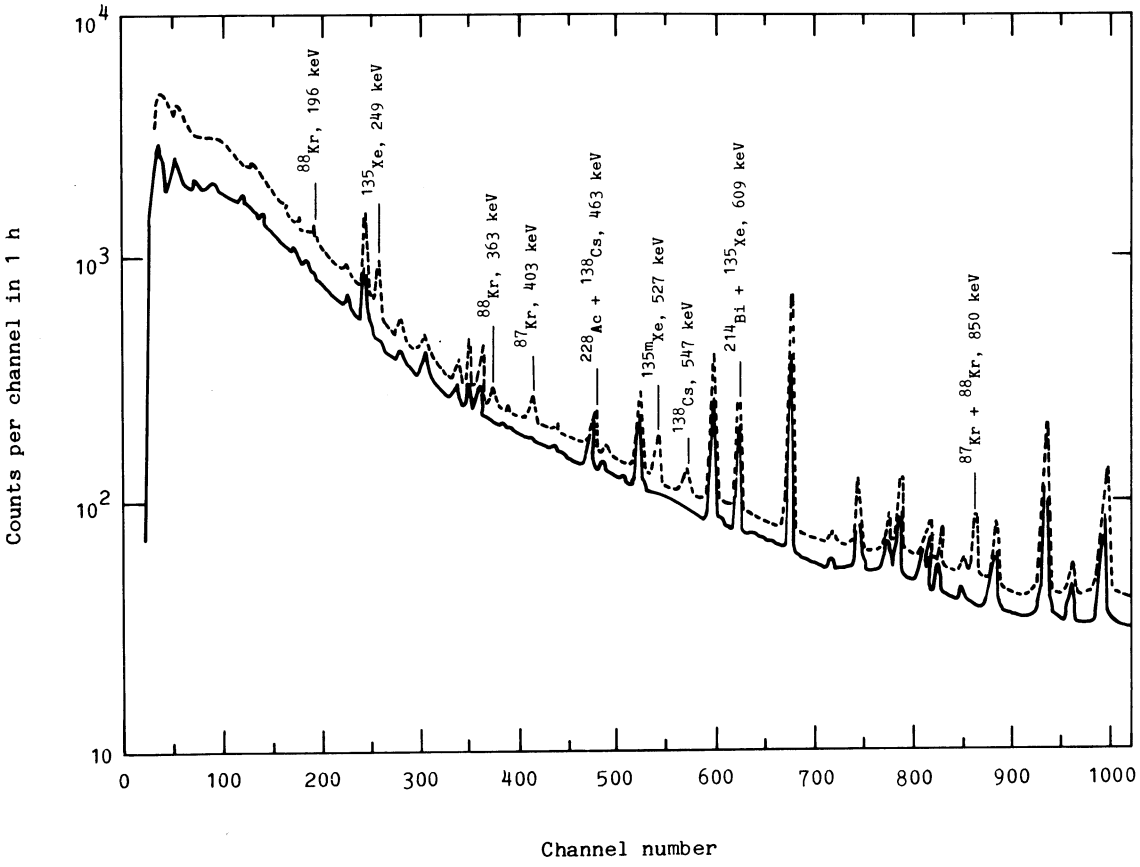

FIGURE 3.4-2: Spectra measured by a Ge(Li) detector at the fenceline of a boiling-water reactor: upper spectrum with gaseous plume overhead; lower spectrum with wind blowing from detector towards reactor (Beck et al., 1972)

The liquid-nitrogen cryostat required for Ge(Li) detectors can be positioned either above or below the crystal with the crystal mounted facing downward or upward respectively. An upward-facing crystal has practical advantages for it is more stable and also if used to measure samples these can be readily placed on the detector. The disadvantage is that it screens the detector from the sources in the soil immediately below it but the solid angle blocked is very small and most of the primary flux comes from further afield. Elongated, horizontal configurations should be avoided since they will

have a complex directional dependence and would make interpretation of the spectra difficult.

A complete analysis of all the total-absorption peaks measured need not be undertaken. In field spectrometry we have seen that the contributions to the dose rate chiefly occur from ^{238}U and ^{232}Th series, ^{40}K and a few other fall-out nuclides such as ^{137}Cs + ^{137}Ba and ^{95}Nb + ^{95}Zr. For the natural contributions Gogolak et al. (1977) selected six peaks at 186, 295, 352, 609, 1120 and 1765 keV for the ^{238}U series, and for the ^{232}Th series, peaks at 583, 911, 966 and 2614 keV. Two other peaks at 662 keV for ^{137}Cs + ^{137}Ba and 1460 keV for ^{40}K were also analysed. The principal natural contributors to the dose rate are ^{214}Bi, ^{208}Tl and ^{40}K.

Sampling times are longer than for NaI(Tl) systems because of the smaller relative sensitivity of Ge(Li) detectors, for example sampling periods of 40 minutes to one hour are used.

3.4.4 NaI(Tl) systems: NaI(Tl) systems can be operated with cheaper electronics and the poorer resolution of the system requires only a 400-channel analyser and consequently less data-handling equipment. For straightforward natural environmental measurements in situations where the operator does not have to determine the man-made contributions, a NaI(Tl) system will suffice. The higher efficiency of a NaI(Tl) detector means that sampling times can be reduced to about ten minutes. Since they do not need to be cooled they are also more portable than Ge(Li) systems.

A detailed analysis of the spectra is not possible with NaI(Tl) systems because of the inferior resolution of the detector, which does not resolve peaks separated by less than 100 keV. Nevertheless, when naturally occurring radionuclides predominate, Beck et al. (1972) demonstrated that an accurate estimate of the dose rate could be obtained by a wide-energy-band analysis. They analysed three energy bands centred on the 1.46 MeV ^{40}K peak (1.32 MeV to 1.60 MeV), the 1.76 MeV ^{214}Bi peak (1.62 MeV to 1.90 MeV) and the 2.62 MeV ^{208}Tl peak (2.48 MeV to 2.75 MeV); these bands are indicated by E_1, E_2 and E_3 in Figure 3.4-3. The small contribution due to cosmic radiation in each energy band should be subtracted. This method provides a reasonable measure of the

dose rate because the in situ spectrum above a few hundred keV is not critically dependent on the exact proportions of uranium, thorium and potassium. It cannot be used if there are any high-energy fall-out emitters contributing significantly to the dose rate.

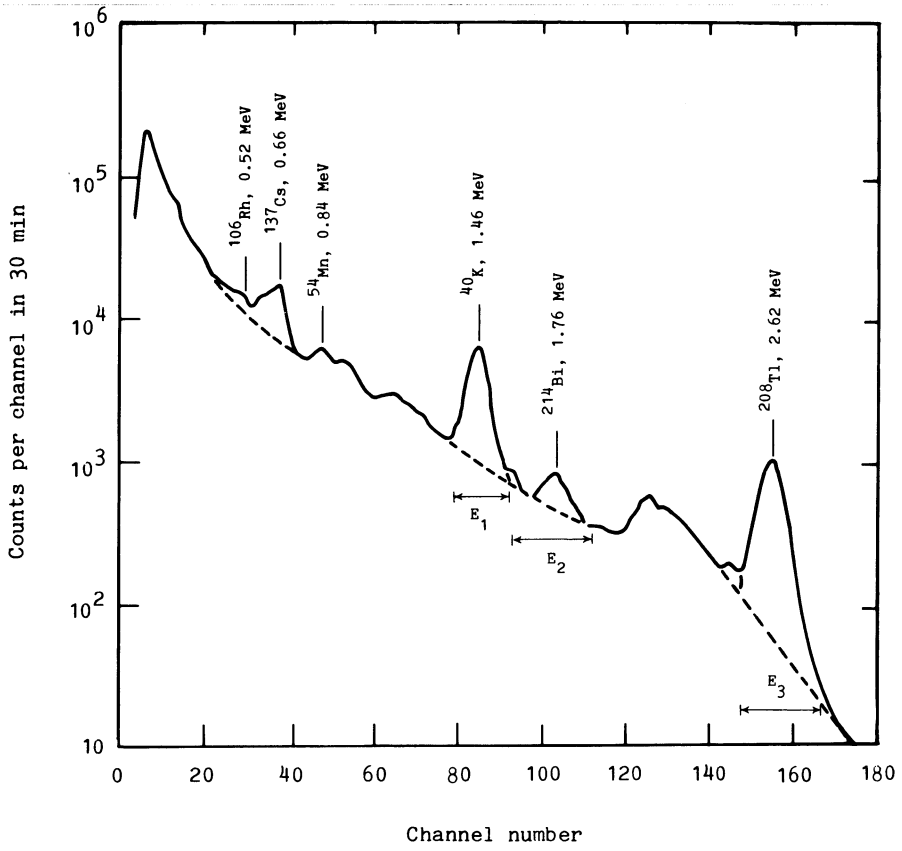

FIGURE 3.4-3: Field spectrum using an NaI(Tl) crystal (Beck et al., 1972)

3.5 Thermoluminescence dosimeters

3.5.1 Advantages and choice of system: Passive dosimeters like photographic-film or thermoluminescence dosimeters (TLDs) have the obvious advantage of being small and relatively cheap, and requiring no in situ electronics. However, they will not provide information on the variation of dose rate with time, and the dose received during their exposure is determined by reading out the dosimeter in a laboratory. Although photographic dosimeters have been used for environmental dose measurements their low sensitivity and susceptibility to adverse ambient conditions has led to their being largely supplanted by TLDs. Considerable experience has been gained on using TLDs to measure low doses and since 1974 there have been four international intercomparisons of environmental

dosimeters. Most of the dosimeters intercompared were TLDs (Gesell et al., 1976; de Planque and Gesell, 1979). The results of the second intercomparison, held in 1976, are summarized in Section 4.9.

Many organizations and countries now use TLD systems for the measurement of personal doses to radiation workers. The use of TLDs to measure the lower environmental dose rates, however, requires the application of very much more careful techniques and assessment. With such care average dose rates can be assessed with a reproducibility of ±16%.

Before selecting any particular detector and read-out system the characteristics of both should be carefully examined. Among the characteristics that should be investigated are the energy dependence and directional dependence of the detector, its response to cosmic radiation and its temperature dependence, fading, self-irradiation and sensitivity over the expected dose range. The uniformity of batches of detectors, and the reproducibility and stability of the system should also be investigated. Further problems relate to the calibration of the dosimeters. Methods of dealing with the evaluation of these characteristics have been well reviewed by Burke and McLaughlin (1974) and a summary of their findings is given below.

3.5.2 Energy dependence: The detector's response should be as close as possible to the absorbed-dose-to-air response over the range of the environmental photon-energy spectrum. Figures 3.5-1 and 3.5-2 show the energy dependence of several TLD materials and two glass-phosphors. The LiF phosphors have the lowest energy dependence whilst CaF_2 dosimeters will overestimate the dose from 30 keV photons by a factor of about 20 relative to a calibration with ^{60}Co gamma radiation. The enhanced low-energy response of CaF_2 dosimeters can be reduced by shielding the detector. A 0.25 mm tantalum filter plus 0.051 mm of lead was used by Denham et al. (1972) to produce a response for the combined CaF_2:Dy dosimeter that is constant to within ±20% from 50 keV to 1.25 MeV. This shield can also provide protection against adverse ambient conditions but it can, however, produce an undesirable directional dependence.

3.5.3 Directional dependence: During calibration the radiation field is approximately unidirectional whilst the environmental radiation field

is much more isotropic. If the TLD were calibrated in the direction of its maximum response it could considerably underestimate the environmental dose.

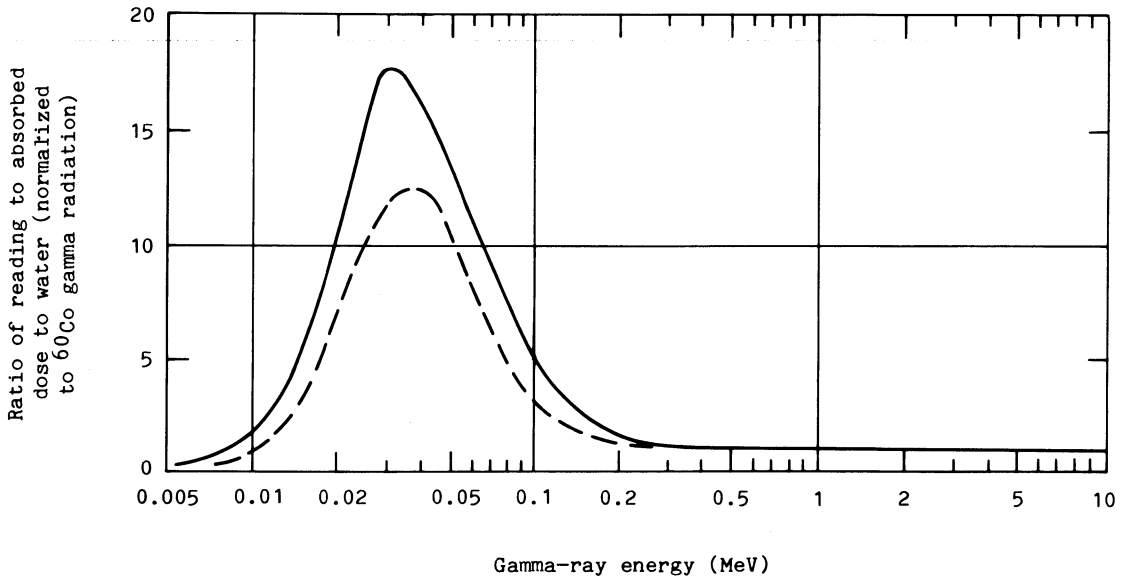

FIGURE 3.5-1: Energy dependence of CaF_2:Dy (Budd, 1976): continuous line shows average of experimental results from Almond et al. (1968), Binder and Cameron (1969), Budd (1976) and Endres et al. (1970); broken line shows theoretical response curve for CaF_2 based on Almond et al. (1968)

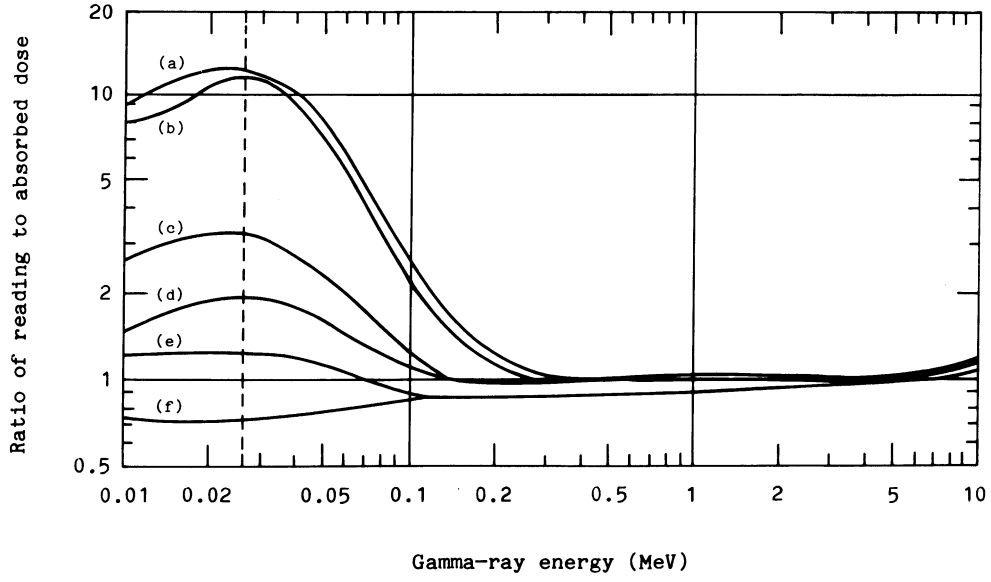

FIGURE 3.5-2: Energy dependence of different TLD materials (François et al., 1970):

(a) CaF_2;
(b) $CaSO_4$;
(c) Aluminophosphate glass;
(d) Aluminosilicate glass;
(e) LiF;
(f) $Li_2O \cdot 2B_2O_3$

3.5.4 Temperature dependence and fading: Fading, which is the de-excitation of the radiation-induced effect in the TLD material, is a function of the ambient temperature. During exposure of the dosimeter it is impracticable to stabilize the temperature. Thus if there is significant fading it is necessary to have a detailed knowledge of the variations in dose rate and temperature with time and to know the fading characteristics as a function of temperature in order to make the necessary corrections. However, the temperature and dose-rate variations during the in situ exposure may not be known and attempts must then be made to eliminate their effect from the measurement.

Fading for a particular peak can be expressed mathematically as follows:

$$R = R_0 e^{-\lambda t} , \qquad (3.5\text{-}1)$$

where R_0 is the initial reading, R is the reading after time t and λ is the fading-rate constant, which itself depends upon the ambient temperature. If the glow curve for the TLD material contains many peaks then each peak will have a different value for λ. Frequently, the fading rate decreases as the characteristic temperature of the glow peak increases and therefore this correction can be made smaller by measuring only the higher-temperature peaks.

If the in situ dose rate is reasonably constant the fading correction can be estimated by using control dosimeters at the same location. These control dosimeters are placed within a lead shield of thickness 5 cm and will therefore be irradiated at a known low constant dose rate from penetrating cosmic radiation and contamination in the shield and the dosimeter.

Lindeken et al. (1973) estimated the fading by irradiating another set of dosimeters to a known dose at the effective mean time of the field exposure period. The proportion of the decrease in reading at the end of the field exposure of this irradiated dosimeter, which has been stored under known conditions, will be approximately the same as the proportion of the decrease in reading of TLDs receiving a constant exposure rate over the entire

period, provided that the ambient temperature is constant. Except at high temperatures, fading in LiF dosimeters can be neglected. The fading rate for CaF_2 dosimeters can be 2—3% per month at normal ambient temperatures.

3.5.5 Self-irradiation: This arises from small amounts of radioactivity in the materials used for the TLD encapsulation. It can be equivalent to 10 nGy/h (1 μrad/h) or higher and can vary within a batch. An estimate of the self-irradiation can be made by reading out dosimeters that have been stored for a long time in an environment with a low, constant and known dose rate, such as in a lead box. This method is valid only if the fading can be neglected. When the phosphor has significant fading then the combined fading and self-irradiation can be determined as follows.

If the initial reading is zero and the dosimeter is exposed to a constant dose rate β for time t, then

$$R = (\beta/\lambda)(1 - e^{-\lambda t}) . \qquad (3.5\text{-}2)$$

In this equation β is the net dose rate, given by

$$\beta = \beta(\text{self-irradiation}) + \beta(\text{ambient}).$$

Again, assuming β(ambient) is known, β(self-irradiation) can be derived by fitting data obtained at a given temperature for various times t.

3.5.6 Sensitization of TLDs: TLD dosimeters can be sensitized by irradiation with gamma radiation followed by annealing while being irradiated with UV light. The energy dependence of sensitized LiF TLDs is shown in the lower curve of Figure 3.5-3 whilst the upper curve shows the variation in sensitizing factor with energy (Jones, 1977).

3.5.7 Calibration of TLDs: Chapter 4 deals with general calibration problems for all dosimeters and dose-rate meters. The specific problems associated with the calibration of TLDs are briefly discussed in this section.

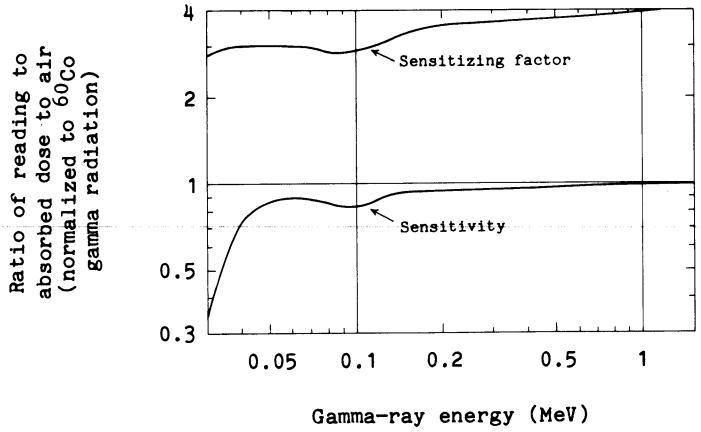

FIGURE 3.5-3: Energy dependence of sensitizing factor and sensitivity of sensitized TLDs (Jones, 1977)

Calibration of the dosimeters is required to convert the output to absorbed dose and three methods are commonly used:

(a) Batch calibration: The calibration dosimeters are selected from the same batch as those used for field measurements. During read-out of the field dosimeters the calibration dosimeters should be evenly interspersed among them to check the stability and accuracy of the read-out system.

(b) Individual calibration: Each individual field dosimeter is calibrated (Jones, 1977). The precision of conversion to dose is determined from the reproducibility of the read-out system, which can be tested by use of controlled TLDs irradiated to known doses.

(c) Semi-individual calibration: Only some of the field TLDs are used for calibration but corrections are applied to compensate for variations in sensitivity among all the field TLDs.

3.5.8 Response of TLDs to cosmic radiation: The response to cosmic radiation can be calculated from a knowledge of the composition and energy distribution of the particles and their collision stopping powers for the TLD material (O'Brien, 1978). Mu mesons account for 75% of the charged particle fluence and the theoretical collision stopping power of Bethe-Bloch for mu mesons may be used (ICRU, 1970). Alternatively a synchrotron may be used to determine the response of the dosimeter to mu mesons (Lowder and de Planque, 1977). A more practical approach is

to determine the response to cosmic radiation by irradiating the dosimeters over deep water, or by measuring their response when exposed in a shield of thickness sufficient to exclude any local gamma radiation (see Chapter 4).

3.5.9 Isolation of environmental exposure: It is necessary to subtract from the read-out of an irradiated TLD that part of the exposure accumulated both prior to and after the field exposure. A dosimeter should be read out as soon as possible after exposure and transit times should be kept to a minimum. Prior to the field exposure the dosimeters should be kept in a shielded low-background store and the control dosimeters should remain in this store during the exposure.

TABLE 3.5-1: Characteristics of thermoluminescent phosphors suitable for environmental radiation measurements; data derived from NCRP (1976)

Phosphor	Z_{eff} [1]	Room-temperature fading ($\frac{\%}{month}$)	Self-irradiation ($\frac{nGy}{h}$) {$\frac{\mu rad}{h}$}	Reported lower dose limit[2] (nGy) {mrad}	References [3]
LiF (TLD-700)[4]	8.2	negligible	negligible	7.5 {0.75}	1
$Li_2B_4O_7$:Mn	7.4	<10	none reported	... {...}	2, 3
CaF_2:Mn	16.3	5	60—180 {6—18}	10 {1.0}	4, 5
CaF_2 (natural)	16.3	negligible	80—120 {8—12}	<10 {<1.0}	6, 7
CaF_2:Dy (TLD-200)[4]	16.3	(after pre-read-out anneal)	negligible	4 {0.4}	8
$CaSO_4$:Dy	15.4	<2	none reported	4 {0.4}	9

[1] Effective atomic number

[2] Lower limits highly dependent on photomultiplier tube used

[3] References:
1—Shambon (1972); 2—Binder et al. (1968); 3—Becker (1973); 4—Burke (1972); 5—Brinck et al. (1975); 6—Aitken (1968); 7—Schulman (1967); 8—Denham et al. (1972); 9—Yamashita et al. (1971)

[4] TLD-100, 200 and 700 are Harshaw Chemical Co. designations, but are widely used; TLD-100 (natural LiF composition) and TLD-700 (enriched in Li-7) have about the same gamma-ray response

Fading, self-irradiation and temperature corrections should be applied to all the control, field and calibration dosimeters.

3.5.10 Characteristics of TLDs: Table 3.5-1 summarizes the characteristics of some of the TLD materials used for environmental measurements. The TLDs should be suitably packed to be light-tight and to provide protection from moisture, humidity and excessive temperatures. They should be exposed well away from buildings and at secure locations.

Of the various TLD materials $CaSO_4$:Dy is the most sensitive, being some 25 to 30 times more sensitive than LiF (Becker, 1972). Also it has a simple glow curve with only one low-temperature trap, and has the further advantage that its luminescent centres are not affected by the use of different annealing procedures. On the other hand, LiF has a much smaller energy dependence than $CaSO_4$:Dy. To improve the energy dependence of $CaSO_4$:Dy TLDs external filters may be added but these may affect the directional dependence, particularly for low photon energies.

3.5.11 Draft international standards on TLDs: At the time of writing this Guide two international standards on TLDs are in preparation. The International Electrotechnical Commission SC45B Working Group 6 is preparing a standard for read-out systems. The International Standards Organization TC85/SC2 Working Group 7 is preparing a standard dealing with the performance of personal and environmental dosimeters, including test methods for verifying compliance with the requirements; characteristics being covered in this standard include sensitivity, repeatability, dose range, linearity, memory, zero stability, detection threshold, photon-energy dependence, directional dependence and climatic requirements.

4 INSTRUMENT RESPONSE AND CALIBRATION

The various stages in the calibration of an instrument intended for the measurement of environmental gamma-ray dose rate are described in this chapter. Traceability to national standards and checks on the stability of instrument response are also discussed.

4.1 Instrument response

The calibration of an instrument for measurement at environmental dose rates requires a detailed knowledge of the detector's response to the different components of the background (McLaughlin, 1977). The cosmic-ray response and the internal background of each individual instrument should be determined using the methods described for each type of detector in Chapter 3.

When making a measurement of the environmental dose rate at any location, the reading, R, of an instrument may be represented by

$$R = K_e \dot{D}_e + K_c \dot{D}_c + R_i \,, \qquad (4.1\text{-}1)$$

where

K_e is the response to environmental gamma radiation;

$\dot{D}_e$ is the absorbed dose rate from environmental gamma radiation;

K_c is the response to cosmic radiation;

$\dot{D}_c$ is the absorbed dose rate from cosmic radiation; and

R_i is the contribution to the reading from internal radioactive contamination and/or electronic noise.

To estimate the environmental dose rate, either $\dot{D}_e$ or $(\dot{D}_e + \dot{D}_c)$, from such a reading, it is necessary to determine these individual factors. For non-air-equivalent detectors K_e and K_c are usually not equal, and

the value of K_e depends on the photon energy. To evaluate K_e, K_c and R_i it is necessary to measure each response separately by elimination of the other two quantities.

4.2 Determination of internal background, R_i

The reading R_i due to the internal background of any instrument may be estimated in various ways including the use of an anticoincidence ring for pulse-measuring instruments (Section 3.1.2) or by observing the instrument reading when it is taken to a large depth below ground. At a depth of 600 m the soft and hard components of the cosmic radiation are effectively eliminated and by placing the detector within a lead shield of thickness 10 cm the effect of the radiation from local rocks can also be reduced to an insignificant level.

For ionization chambers, R_i can usually be considered as due entirely to intrinsic alpha radioactivity; it can be estimated by placing the chamber in a shielded low-background facility and monitoring the electrometer output with a short-time-constant recorder (DeCampo et al., 1972). Alpha-particle pulses can be identified by large spikes produced in the recorder output. Periodic checks for an increase in the leakage current and for insulator stresses should also be made; unidirectional currents arising from stress within the insulator can be determined by making measurements with both positive and negative polarizing voltages.

The internal background of any instrument should not change significantly during its life because the radionuclides present have long half-lives, nevertheless occasional checks are advisable since the instrument may itself become contaminated.

4.3 Determination of cosmic-ray response, K_c

The determination of the cosmic-ray response, K_c, can be made either experimentally or from a theoretical calculation of the interaction of mu mesons and electrons in the detector. The experimental measurement of the total cosmic-ray response (both soft and hard components) can be made in a light wooden or plastic boat on a

freshwater lake or reservoir or at sea between a half and one kilometre from the land (Spiers et al., 1964; Kastner et al., 1964; Shamos et al., 1964). Correction must be made for the internal background of the instrument. The absorbed dose rate from cosmic radiation at sea level in mid-northern latitudes may be assumed to be 32 nGy/h (3.2 μrad/h) (see Section 2.4).

Alternatively the response to the hard component of the cosmic radiation can be measured in a small steel cubicle that largely excludes any local gamma radiation (e.g. a cubicle about 90 cm x 90 cm x 30 cm with steel walls and door 12 cm thick) and the total cosmic-ray response deduced from the known ratio of soft to hard cosmic radiation at a given barometric pressure (Figure 2.4-1). Again, the internal background reading of the instrument must be subtracted if a separate determination of the cosmic-ray response, K_c, is required.

4.4 Determination of environmental gamma-ray response, K_e

4.4.1 General procedure: The response of instruments with detectors constructed of air-equivalent materials (e.g. some types of ionization chambers) will vary only slightly over the energy range of environmental gamma radiation. For these instruments a calibration by means of a single radiation source, as described in Section 4.4.2, will provide a value for the response that can be taken as equal to the environmental gamma-ray response K_e with no further correction for energy dependence.

Many types of instruments, however, incorporate detectors whose response to gamma radiation varies significantly with photon energy. The determination of the response to environmental gamma radiation K_e will then involve a knowledge not only of the energy dependence of the detector, but also of the spectrum of the environmental gamma radiation to be measured. A procedure for calculating K_e is described in Section 4.4.3.

The calibration procedure described in Section 4.4.2 will give a value for the response that is strictly applicable only for measurements made at the same dose rate as that at the calibration. For some types of instruments it may be necessary to determine the variation of response

with dose rate by using sources of different activities or by varying the source—detector distance. This applies both to dose-rate meters (linearity of response) and to integrating instruments.

For an instrument to comply with draft IEC standards (IEC 45B SC45B/WG5 (Secretariat) 51), the relative intrinsic error shall be within ±10% between scale readings of 10% and 90% of the scale maximum angular deflection; this excludes any uncertainty associated with the determination of the dose.

An instrument with a significant directional dependence that is calibrated by unidirectional radiation may need a further correction when it is to be used to measure environmental radiation (see Section 4.5). It should be noted that both the response and directional dependence will in general depend on the radiation energy; this may be investigated by means of sources with a range of X-ray or gamma-ray energies (see Section 4.4.3).

4.4.2 Calibration procedure: This section is relevant to calibrations with either X radiation or gamma radiation. The dose rate at the position of the detector should have been determined, either from a previous measurement using a secondary-standard dosemeter whose response is known and whose calibration is traceable (see Section 4.7), or from information provided on a source certificate. In principle either a collimated beam of radiation or an unshielded source may be used, unless a particular condition is specified on the source certificate. It is recommended that scattered radiation should be reduced by positioning the source and detector so that their distances from any irradiated scattering material (such as the ground) are at least three times the source—detector distance. The distance from source to detector, however, should be at least ten times any of the dimensions of the detector.

The activities of the calibration sources (or beam currents of the X-ray generator) should be chosen so as to give dose rates at the detector as close as practicable to the dose rates of the environmental gamma radiation to be measured; this choice may however, be limited by a conflicting requirement to provide higher dose rates suitable for a secondary-standard instrument of lower sensitivity (see Section 4.7).

For calibrations at low dose rates it may be necessary to correct for background by the following method. The instrument is first exposed without the radiation source to give the background reading R_B,

$$R_B = K_e \dot{D}_e + K_c \dot{D}_c + R_i \,, \qquad (4.4\text{-}1)$$

where the symbols have the same meanings as in Equation 4.1-1, but both dose rates are for radiation at the detector under the calibration conditions.

The instrument is then exposed to the radiation source and the reading R_S is given by

$$R_S = K_s \dot{D}_s + K_e \dot{D}_e + K_c \dot{D}_c + R_i \,, \qquad (4.4\text{-}2)$$

where K_s is the response of the instrument to gamma radiation from the calibration source, and $\dot{D}_s$ is the absorbed dose rate from the calibration source at the reference position of the detector.

By subtraction of Equation 4.4-1 from Equation 4.4-2, the response to the calibration gamma radiation may be calculated to be

$$K_s = \frac{R_S - R_B}{\dot{D}_s} \,. \qquad (4.4\text{-}3)$$

This method will eliminate the contributions from cosmic radiation and environmental gamma radiation in the calibration room and from the internal background of the detector. It should be noted that it is only applicable if the scattered radiation from the source is negligible (less than 5%). Where significant scattered radiation is present then these two measurements should be replaced by one with the source present and another with a 'shadow' shield of 20 cm of lead to prevent the radiation from passing directly from the source to the detector. Subtraction of the lead-shield reading allows the response to the primary radiation from the calibration source to be determined.

4.4.3 Energy dependence: To determine the variation of K_s with photon energy, selected energies from the low-exposure-rate series of filtered X radiations proposed by the International Standards Organization, ISO,

should be used. The operating conditions for the X-ray generator and the filters that produce these reference radiations are listed in Table 4.4-1. The instrument response should be determined using the 660 keV (^{137}Cs) and 1.25 MeV (^{60}Co) reference gamma radiations (Table 4.4-2) in addition to these filtered X radiations; the 59.5 keV gamma radiation from ^{241}Am may be used if required. The conditions for the production of these radiations are given by the British Calibration Service (1977). If the instrument is to be used to measure environmental dose rates in the area surrounding a nuclear power station producing radiation up to 9 MeV, it will also be necessary to determine the response up to this energy. Working Group 2 of ISO/TC85/SC2 are at

TABLE 4.4-1: ISO reference X radiations, low-exposure-rate series

Mean energy (keV)	Resolution FWHM (%)	Generating potential (kV)	Additional filtration[1] Lead (mm)	Tin (mm)	Copper (mm)	First HVL (mm)
30	21	35	0	0	0.25	2.38 Al
48	22	55	0	0	1.2	0.25 Cu
60	21	70	0	0	2.5	0.48 Cu
87	20	100	0	2.0	0.5	1.28 Cu
109	18	125	0	4.0	1.0	2.14 Cu
148	18	170	1.5	3.0	1.0	3.67 Cu
185	18	210	3.5	2.0	0.5	4.91 Cu
211	17	240	5.5	2.0	0.5	5.89 Cu

[1] The total filtration includes an inherent filtration adjusted to 4 mm Al; dose rates are approximately 300 μGy/h at 1 m for a tube current of 1 mA

TABLE 4.4-2: Radionuclides used for ISO reference gamma radiations

Radioactive nuclide	Gamma-ray energy (keV)	Half life (yr)	Absorbed dose rate to air[1] At 1 m from 1 MBq (nGy/h)	At 1 m from 1 μCi (μrad/h)
^{60}Co	(1173.3 1332.5	5.272	308	1.14
^{137}Cs	661.6	30.1	79	0.292
^{241}Am	59.54	433	3.0	0.011

[1] The dose rates are for unshielded point sources and are therefore given only as a guide

present studying reference radiations having energies from 2 MeV to 10 MeV, and they have suggested that energy dependence should be within ±25% from 50 keV to 2 MeV and within ±35% from 2 MeV to 9 MeV.

Owing to the inherent difficulties in standardizing the measurement of low dose rates, energy-dependence tests are more conveniently carried out at dose rates near the maximum measurable by the instrument under test.

In order to calculate the response to environmental gamma radiation, K_e, three things must be known:

(a) the response K_s of the instrument to the radiation from a standard source (usually ^{137}Cs or ^{60}Co),

(b) the energy dependence of the instrument, and

(c) the spectrum of the environmental gamma radiation.

In many situations it can be assumed that the spectrum of the environmental gamma radiation to be measured is similar to the spectrum of natural environmental gamma radiation as shown in Figure 4.4-1. If this is not the case then the spectrum will have to be measured with a spectrometer (see Section 3.4).

Assuming now that the spectrum of the environmental gamma radiation is known, the following procedure may be adopted. The instrument response, determined by the method discussed in Section 4.4.2, is measured at a range of photon energies and normalized to unity for the radiation from the standard source. These normalized responses are

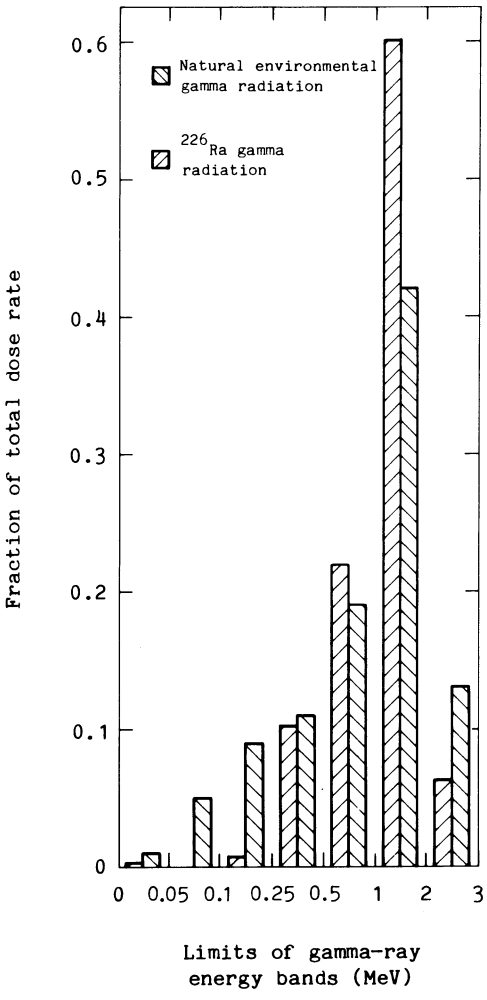

FIGURE 4.4-1: Comparison of the gamma-ray energy spectra from a Ra-226 source and natural environmental radiation (calculated from Beck, 1972)

then plotted against photon energy and the mean values k_j are estimated for each of the seven energy intervals shown in Figure 4.4-1. Each value of k_j is then multiplied by the fractional dose rate, $\dot{d}_j$, in each energy range of the environmental gamma radiation, either taken from Figure 4.4-1 or measured by a spectrometer. The energy-dependence correction factor to the response for the standard source radiation, K_s, to convert it into the response for environmental gamma radiation, K_e, is then given by M, where

$$M = \sum_{j=1}^{7} k_j \dot{d}_j \qquad (4.4\text{-}4)$$

and hence

$$K_e = K_s M \ . \qquad (4.4\text{-}5)$$

A measurement of M need be made only as a type test on one instrument of each commercial design; other instruments of the same type will then only require to be calibrated against a standard ^{137}Cs or ^{60}Co source.

It is interesting to note that, as shown in Figure 4.4-1, the spectrum from a ^{226}Ra source is similar to that of natural environmental gamma radiation. Although not specified as an ISO reference radiation, nevertheless ^{226}Ra is a useful reference source to use for calibrating instruments whose variation of response with energy is small. No correction factor M is then necessary for energy dependence.

4.5 Directional dependence

The question of directional dependence for cosmic radiation has been discussed in Section 3.1.3 and for the calibration of gamma-ray spectrometers in Section 3.4.2. Gamma-ray detectors used to measure the dose rate from radioactivity in the soil must be corrected for their response as a function of angle of incidence. Polar diagrams in the plane of the detector axis as shown in Figure 4.5-1 demonstrate a clear directional dependence and a correction is required.

For such a detector its response K_s to unidirectional radiation from a gamma-ray source can be corrected to its response K_e to omnidirectional

environmental gamma radiation by a factor L such that

$$L = \frac{\int_0^{2\pi}\int_0^{\pi} k(\theta,\psi)\dot{d}(\theta,\psi)\,\sin\theta\,d\theta\,d\psi}{\int_0^{2\pi}\int_0^{\pi} \dot{d}(\theta,\psi)\,\sin\theta\,d\theta\,d\psi}, \qquad (4.5\text{-}1)$$

where

$k(\theta,\psi)$ is the directional response of the detector to the dose rate from photons incident from a direction θ,ψ, normalized to unity for photons from the direction $\theta = 90^{\circ}$ (as in Figure 4.5-1) and $\psi = 0^{\circ}$;

$\dot{d}(\theta,\psi)$ is the absorbed dose rate to air at the detector for photons from direction θ,ψ, normalized to unity for photons from the direction $\theta = 90^{\circ}$, $\psi = 0^{\circ}$;

θ is the angle of incidence of photons with respect to the detector axis (direction of calibration source, $\theta = 90^{\circ}$); and

ψ is the angle of incidence of photons in a plane perpendicular to the detector axis (direction of calibration source, $\psi = 0^{\circ}$).

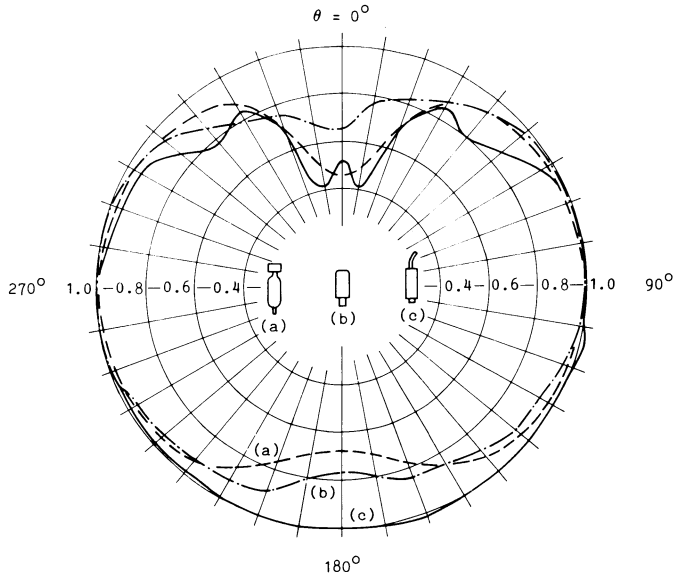

FIGURE 4.5-1: Directional dependence of three types of detectors to Ra-226 gamma radiation:
(a) High-pressure ionization chamber (Spiers et al., 1964)
(b) Cylindrical probe with six G-M tubes (Jagielak et al., 1974)
(c) 44 mm x 32 mm diameter NaI(Tl) scintillator (Jagielak et al., 1974)

For a cylindrical detector the directional response $k(\theta,\psi)$ will be constant with respect to ψ. Moreover, if the directional distribution of environmental gamma radiation is not known it may be reasonable to assume that it is isotropic, in which case Equation 4.5-1 simplifies to

$$L = \frac{\int_0^{\pi} k(\theta) \sin\theta \, d\theta}{\int_0^{\pi} \sin\theta \, d\theta} = \frac{1}{2}\int_0^{\pi} k(\theta) \sin\theta \, d\theta \, . \qquad (4.5\text{-}2)$$

The variation of $k(\theta)$ with θ can be taken from a polar diagram such as Figure 4.5-1 and L calculated from Equation 4.5-2, but it may be more convenient to read off average values of $k(\theta)$ at 20° angles and to use the equivalent summation formula:

$$L = \frac{\sum_{10^{\circ}}^{170^{\circ}} k(\theta_j) \sin\theta_j}{\sum_{10^{\circ}}^{170^{\circ}} \sin\theta_j} \, , \quad \text{with } \theta_j = 10^{\circ}, 30^{\circ}, 50^{\circ}, \ldots, 170^{\circ}$$

$$= 0.174 \sum_{10^{\circ}}^{170^{\circ}} k(\theta_j) \sin\theta_j \, . \qquad (4.5\text{-}3)$$

$\sin 10^{\circ}$ = 0.174 = $\sin 170^{\circ}$	$\sin 30^{\circ}$ = 0.500 = $\sin 150^{\circ}$	$\sin 50^{\circ}$ = 0.766 = $\sin 130^{\circ}$	$\sin 70^{\circ}$ = 0.940 = $\sin 110^{\circ}$	$\sin 90^{\circ}$ = 1.000

Using this value of the conversion factor L, then

$$K_e = K_s L \, . \qquad (4.5\text{-}4)$$

An instrument with a significant directional dependence which is calibrated with monoenergetic radiation may require further correction for the energy of the environmental radiation as in Equation 4.4-5. This should be investigated during type testing with a range of X-ray and gamma-ray energies. Since the directional correction is small then the additional energy correction may not be

significant for most detectors, particularly if a ^{226}Ra source is used to determine the directional dependence. For example, for the Leeds high-pressure ionization chamber shown in Figure 4.5-1, L = 0.94 for ^{226}Ra gamma radiation and L = 0.91 for ^{131}I gamma radiation (Spiers et al., 1964).

4.6 Type testing

In addition to the calibration and measurement of energy dependence of the detector, general test procedures have been laid down by the IEC (IEC 45B SC45B/WG5 (Secretariat) 51). These include tests to ensure the response to beta radiation is small (e.g. <1% of the response for environmental radiation measurements is proposed here in this Guide), tests on electrical, mechanical and safety characteristics, and also the response under changes of ambient temperature, relative humidity and barometric pressure. These environmental factors are particularly important for unsealed ionization chambers working at ambient atmospheric pressure.

4.7 Traceability of calibration standards

If, for the purpose of type or individual testing, calibrations are required over a range of X-ray or gamma-ray energies, then the calibration must be made traceable to national standards of exposure. Because the environmental gamma-ray dose rate is several orders of magnitude less than that used with the national standard, it is necessary to make very careful measurements in order to reduce the uncertainties in the calibration of the field instrument. Preferably this should be achieved by comparing, in the same X-ray beam, the response of the instrument with that of a secondary standard that has been calibrated by a national standardizing laboratory (NPL in the UK). This should be practicable if the instrument under test can be used up to 10 μGy/h (1 mrad/h), which is typically the most sensitive range of a secondary standard dosemeter specifically designed for protection-level measurements. If, however, the ranges of the two instruments do not overlap, then it will be necessary to introduce a tertiary standard of intermediate sensitivity into the calibration chain.

Annual calibrations using radioactive sources should be performed at dose rates between 0.01 μGy/h and 1 μGy/h (1 μrad/h and 100 μrad/h). This can be carried out by exposing the detector in specified calibration conditions to the gamma radiation from low-activity ^{60}Co, ^{137}Cs or ^{226}Ra sources. These low-activity sources should have been calibrated by comparison with higher-activity sources; a re-entrant ionization chamber, e.g. AERE-type 1383A (Dale et al., 1961), is suitable for this purpose. The output of these higher-activity sources should have been measured in the same specified calibration conditions by a protection-level secondary-standard dosemeter. Alternatively a calibrated source (supplied by the Radiochemical Centre Ltd., Amersham) can be used, provided the dose rate or exposure rate is certified. In this way, traceability to national standards of exposure will have been achieved.

It may be noted that, at the present time, national standards for radiation dosimetry measure exposure. A calibration of a secondary standard in terms of exposure is usually carried out under effectively scatter-free conditions in a photon beam of sufficient cross-sectional area to irradiate the detector uniformly. Conversion into absorbed dose to air is carried out at the secondary-standard stage using appropriate conversion factors, which assume that electron equilibrium exists at the point of measurement.

4.8 Constancy checks

A check for constancy of gamma-ray response can be made conveniently by using a small source of a long-lived radionuclide in some standard geometry in relation to the detector; a 0.5 MBq (~10 μCi) ^{226}Ra source at 1 m will give a convenient dose rate of the same order as the environmental gamma radiation. This test will check the constancy of the gas pressure in sealed ionization chambers; in G-M counters it will test the gas filling and the stability of the gas composition. Because the test is made by obtaining the difference between the instrument indication due to background and that due to background plus radiation from the source, it will not of itself give any information on the reliability of the background response as a true measure of environmental radiation.

4.9 International comparisons

An international comparison experiment was organized in conjunction with the conference on 'The Natural Radiation Environment' held at Rice University, Houston, Texas, in 1963. Three portable ionization-chamber equipments were compared and one scintillator equipment; two

TABLE 4.9-1: Results of an international comparison of environmental exposure-rate meters, 1963

Laboratory	Instrument	Exposure rates (μR/h)						
		Cosmic radiation	Gamma-ray sources					TOTAL EXPOSURE RATE
			Natural environmental			Fall-out	Total gamma	
			K	U	Th			
	Measurements at Rice University							
ANL	Ion chamber[1]	3.8	...	...	...	...	9.1	12.9
	Spectrometer	...	1.0	1.8	3.4	4.7	10.9	...
HASL	Ion chamber[1]	3.4	...	...	...	...	9.4	12.8
	Spectrometer A	3.4	0.9	0.6	3.0	5.6	10.1	13.5
	Spectrometer B	3.4	0.9	0.6	3.0	4.8	9.3	12.7
UCLRL	Scintillator[1]	3.4	...	...	...	...	9.6	13.0
	Spectrometer	...	0.9	2.0	3.4	(high)	...	...
Leeds	Ion chamber[1]	3.4	...	...	...	...	9.8	13.2
Rice	Spectrometer	...	1.3	1.6	3.4	...	...	...
	Measurements at Galveston							
ANL	Ion chamber[1]	3.8	...	...	...	...	4.1	7.9
	Spectrometer	...	0.8	1.0	0.9	1.7	4.4	...
HASL	Ion chamber[1]	3.4	...	...	...	...	3.4	6.8
	Spectrometer A	3.4	1.2	0.7	0.9	0.5	3.3	6.7
	Spectrometer B	3.4	1.2	0.7	0.9	0.5	3.3	6.7
UCLRL	Scintillator[1]	3.4	...	...	...	...	3.3	6.7
	Spectrometer	...	1.2	1.6	1.9	...	...	...
Leeds	Ion chamber[1]	3.4	...	...	...	...	3.3	6.7
Rice	Spectrometer	...	1.2	0.9	1.2	...	...	...
	Measurements at Texas City (near zircon sand)							
HASL	Ion chamber[1]	...	...	...	...	...	38.9	...
UCLRL	Scintillator	...	...	...	...	...	38.6	...
Leeds	Ion chamber[1]	...	...	...	...	...	38	...

[1] Portable

spectrometer equipments were used, one to analyse the environmental gamma radiation directly and one to operate on soil samples. Measurements were made at three sites chosen to represent low, medium and high gamma-ray levels, the last being near a large pile of zircon sand. The results are summarized in Table 4.9-1, from which it is evident that environmental exposure-rate meters, checked and calibrated according to

TABLE 4.9-2: International comparison of environmental dosimeters, 1976 (de Planque and Gesell, 1979)

Type of detector	Field measurements			Laboratory measurements		
	Number of dosimeters	Mean exposure (mR)	Standard deviation (%)	Number of dosimeters	Mean exposure (mR)	Standard deviation (%)
Thermoluminescent						
BeO	1	22.8	...	2	12.2	7.6
CaF_2:Dy	21	16.8	21.6	23	19.0	11.7
CaF_2:Mn	14	14.8	12.4	14	19.2	13.0
CaF_2, natural	0	...	...	1	15.5	...
$CaSO_4$:Dy	17	16.5	26.5	17	18.6	21.2
$CaSO_4$:Tm	5	14.8	12.3	6	17.6	12.5
LiF	42	16.8	23.0	44	19.2	22.5
$LiF:LiB_4O_7$	1	13.0	...	1	13.5	...
Mg_2SiO_4:Tb	1	16.0	...	1	21.6	...
Thermally stimulated exoelectron						
BeO	2	11.5	35.2	2	13.8	11.3
Film						
All types	3	19.7	25.0	4	22.0	33.0
Photoluminescent						
$AgPO_3$	1	18.2	...	1	20.0	...
TOTAL	108	16.4	22.8	116	18.8	20.3
Estimated exposure		17.1[1]	...		21.3[2]	...

[1] Field estimate from ionization-chamber measurements

[2] Laboratory estimate derived from an NBS-calibrated ionization chamber measuring Cs-137 gamma radiation

the principles outlined in this section, gave very concordant results even though they were of different types and calibrated in different laboratories in the USA and the UK. For the four portable instruments the spread of results was no more than ±4% of the mean for the medium-level background at Rice University. At the same location the three spectrometers showed a wider spread of about ±8%. Measurements of the very high-level exposure rate at Texas City agreed to about ±1%.

International comparisons have also been carried out on passive dosimeters. In the second of such intercomparisons 85 participants from 26 countries submitted 133 sets each containing six dosimeters; the majority were TLDs. The results are summarized in Table 4.9-2. One of the most important findings was the need to improve the calibrations of the dosimeters. The standard deviation of the results obtained by participants who calculated their calibration exposures was more than double the standard deviation of the results of those who measured their calibration exposures.

5 INTERPRETATION OF MEASUREMENTS

The estimation of uncertainties is often the most difficult part of any measurement, but it is certainly an essential part. The following sections give an estimate of the accuracy at the 95% confidence level (2 standard deviations) for systematic and random uncertainties of typical measurements. In the final section some estimate is provided of the normal variations of the background radiation from terrestrial and cosmic-ray sources. The uncertainties and normal variations can then be used to assess what is a realistic lower limit to the measurement of the additional dose rate from man-made sources.

5.1 Systematic uncertainties

5.1.1 Introduction: The accuracy with which one can measure environmental gamma radiation depends on many factors, which vary between detector types, and may not be the same for all instruments using a given type of detector. For example the internal background of a Geiger—Müller (G-M) counter can be determined quite precisely but the uncertainty on the internal background of a scintillation counter can be ten times higher owing to the larger size of detector and ^{40}K in the glass of the multiplier phototube. Also, for scintillators of different types, the energy dependence can be very large, which could significantly affect their response to different radiation fields. Thus to compare instruments it is necessary to assume that conditions have been optimized for the detector used. The data given in Table 5.1-1 show the estimated systematic uncertainty at the 95% confidence level. The individual sources of uncertainty will be discussed below with a separate section on the Ge(Li) spectrometer. Random uncertainties will be included in Section 5.2 with an estimate of normal variations in the background in Section 5.3.

5.1.2 Energy dependence: The limits of variation for the energy dependence have been defined in Section 4.4 as $\pm$25% for gamma-ray energies from 50 keV to 2 MeV and $\pm$35% from 2 MeV to 9 MeV. If an instrument is calibrated using ^{226}Ra gamma radiation, the measured background will have an uncertainty of considerably less than these maximum variations and

TABLE 5.1-1: Estimated component systematic uncertainties for different types of detectors

Source of uncertainty	G-M counter (%)	Ion chamber (%)	Scintil-lator (%)	TLD (%)
Energy dependence	±2	±2	±2	±2
Calibration	±6	±6	±6	±6
Internal background	±0.6	±0.1	±10	±2
Cosmic-ray contribution	±1	±2	±2	±2
Dead-time correction	±4	...	...	...
Directional dependence	±2	±1	±1	±0.1
TOTAL SYSTEMATIC[1]	±8	±7	±7	±7

[1] Individual components weighted by data in Table 5.1-2 and then added in quadrature

TABLE 5.1-2: Relative contributions of the three main sources to measurements of environmental dose rate by different types of detectors

Contributing source	G-M counter (%)	Ion chamber[1] (%)	Scin-tillator[2] (%)	TLD (%)
Terrestrial (45 nGy/h)	29	61	73	34
Cosmic[3] (32 nGy/h)	37	39	27	21
Internal	34	0	0	45

[1] High pressure [2] Organic [3] Both hard and soft

the estimates given in Table 5.1-1 are for typical variations in the soil content of uranium, thorium and potassium (Table 2.1-2). However it is necessary to allow for this difference in spectrum, and McLaughlin (1977) has calculated that the response of a thick-walled

ionization chamber to natural environmental gamma radiation is 3% higher than its response to gamma radiation from a ^{226}Ra source (Section 4.4). Even allowing for this factor considerable uncertainties may arise for some types of instrument and careful analysis is required in calculating the dose in a particular environment.

5.1.3 Dose-rate calibration: The accuracy with which the dose rate is known depends upon the uncertainties introduced in the primary calibration at the national laboratory and any additional uncertainties introduced in calibrating a field instrument. The assumption made in Tables 5.1-1 and 5.1-2 is that a standard ^{226}Ra source is used for calibrating the field instrument, the uncertainty therefore being the same for all instruments. It is also assumed that corrections for scattered radiation have been made and the uncertainties on these are negligible (within $\pm$0.5%).

5.1.4 Internal background: This can be readily determined for pulsed instruments by placing them in an anticoincidence ring to eliminate the effect of hard cosmic radiation and in a lead shield to remove soft cosmic radiation and environmental gamma radiation. The internal background in pressurized ionization chambers is negligible and the background for a Ge(Li) detector can be determined in the laboratory. As indicated in Section 3.3, for the highest accuracy it is necessary to measure the internal background of scintillators in a deep mine to eliminate cosmic radiation; as this may prove to be difficult the uncertainty given in Table 5.1-1 for scintillators is based upon surface estimates over water. Very careful handling of TLDs is required for measurements of environmental radiation, since 'static' generated on the surface by accidental rubbing with an insulator can induce an extra 'background'.

5.1.5 Cosmic-ray contribution: This is again reasonably straightforward if the effective area of the detector is known and the instrument is always used in the same orientation relative to the ground. The difference in calibration between the terrestrial and cosmic radiations will introduce some additional uncertainty and the data in Table 5.1-1 reflect this factor, particularly for TLDs. The uncertainty quoted is that for estimating the cosmic-ray contribution.

5.1.6 Dead-time correction: This affects pulse detectors and is significant only at the time of calibration when corrections up to 50% may have to be made at dose rates of 10 μGy/h (1 mrad/h). The correction at 50 nGy/h (5 μrad/h) is 0.25% and the additional uncertainty introduced is negligible (within ±0.1%). Because dead-time measurements involve small differences of large counts it is necessary to obtain very precise counting data. For example it can be demonstrated that with a dead time of 2 ms, a total of 10^4 counts (±2% counting uncertainty) would give an uncertainty of ±27% in the dead time (see Section 3.1.4). Reducing the counting uncertainties to ±0.2% (10^6 counts) would reduce the dead-time uncertainty to ±4% (all values are given at the 95% confidence level). This is about the uncertainty limit over which the stability of other components can be guaranteed. The uncertainty for the other systems is zero unless the pulsed mode is used, e.g. with the scintillator.

5.1.7 Directional dependence: For a single detector placed vertically at 1 m above the ground level the uncertainty due to the directional dependence will be within ±1% for ionization chambers, G-M counters and scintillators (Figure 4.5-1). This is because the majority of the gamma-ray flux is close to the horizontal. For a detector placed horizontally to ease the correction for the cosmic-ray response it is unlikely that the uncertainty will exceed ±2% but, by careful calibration and calculation, ±1% uncertainty can also be achieved for this geometry. Many TLDs have an isotropic response but if used with badge holders or energy-compensating filters they may show significant directional dependence, particularly at low photon energies.

5.1.8 Total systematic uncertainty: The total systematic uncertainty can be calculated from the quadrature sum of the individual uncertainties, since there is little correlation between the different influences. This sum can be used to provide an estimate of the precision for measuring the environmental gamma-ray dose or the total dose including cosmic radiation. However, in order to calculate this sum it is necessary to take into account the different relative contribution each source of uncertainty makes to the measurement. The relative contributions from gamma radiation, cosmic radiation and the internal background are given in Table 5.1-2 and by weighting the data of Table 5.1-1 in the proportions given in Table 5.1-2 a weighted quadrature sum

can be calculated as given in the last row of Table 5.1-1. Thus, excluding fall-out gamma radiation, a total dose rate of 77 nGy/h (7.7 μrad/h) can be determined to ±7% or ±5 nGy/h (±0.5 μrad/h) if random uncertainties are not included. This is equivalent to ±50 μGy/yr (±5 mrad/yr) on a total dose rate of 680 μGy/yr (68 mrad/yr).

5.1.9 Ge(Li) spectrometer: Systematic uncertainties for a Ge(Li) spectrometer measuring the dose rate above different rock types are given in Table 5.1-3 (UNSCEAR, 1977). The data are based upon individual

TABLE 5.1-3: Radioactive concentrations in different types of rock and resulting dose rates at 1 m above ground as measured by a Ge(Li) spectrometer; data from UNSCEAR (1977)

Type of rock	Typical radioactive concentration						Absorbed dose rate to air[1]	Uncertainty [2]
	^{40}K		^{232}Th + daughters		^{238}U + daughters			
	$(\frac{Bq}{kg})$	$(\frac{pCi}{g})$	$(\frac{Bq}{kg})$	$(\frac{pCi}{g})$	$(\frac{Bq}{kg})$	$(\frac{pCi}{g})$	$(\frac{nGy}{h})$	(%)
Igneous								
Acidic (granite)	1000	27	81	2.2	59	1.6	120	±4%
Intermediate	700	19	33	0.88	23	0.62	62	±4%
Mafic (basalt)	240	6.5	11	0.30	11.5	0.31	23	±4%
Ultrabasic	150	4.0	24	0.66	0.4	0.01	23	±5%
Sedimentary								
Limestone	90	2.4	7.0	0.19	28	0.75	20	±7%
Carbonate	...	...	7.8	0.21	27	0.72	17	±9%
Sandstone	370	10	11	0.3	18	0.5	32	±5%
Shale	700	19	44	1.2	44	1.2	79	±4%
Typical								
(Table 2.1-2)	370	10	26	0.7	26	0.7	45	±5%

[1] Using the conversion factors given in Table 3.4-1(a)

[2] Uncertainties calculated from measurements with a Ge(Li) gamma-ray spectrometer with the following uncertainties: K-40 ±6%, Th-232 ±6%, U-238 ±12% (McLaughlin, 1976)

uncertainties for the two radioactive series and ^{40}K as calculated by McLaughlin (1976). (The corresponding uncertainty for ^{137}Cs is ±10%.) Taking the 'typical' soil of Table 2.1-2, a total systematic uncertainty of ±6% including a 1% directional dependence is comparable to the uncertainty for the detectors used for measuring the total dose rate (systematic uncertainties up to ±9% are calculated for some soil types).

5.2 Random uncertainties

5.2.1 Introduction: In addition to the systematic uncertainties inherent in any system, there will also be variations in the instrument response that arise from counting statistics, mains or battery fluctuations and effects of temperature, pressure and humidity. Atmospheric conditions also produce changes in the cosmic-ray dose rate and radon level but these will be discussed in Section 5.3. Data on variations in environmental gamma radiation are provided in Table 5.2-1 for a

TABLE 5.2-1: Estimated percentage random uncertainties for different detectors for measurements of the environmental gamma-ray dose over one hour and one month at the 95% confidence level

Source of uncertainty	G-M counter		Ion chamber		Scintillator		TLD
	1 h (%)	1 month (%)	1 h (%)	1 month (%)	1 hr (%)	1 month (%)	1 month (%)
Statistics[1]	±8	±0.3	±0.4	±0.1	±8	±0.3	±6
Power fluctuations	±1	±1	0.0	±0.1	±2	±5	0
Atmospheric conditions	±0.1	±0.1	0.0	±0.1	±0.1	±0.1	±15
QUADRATURE SUMS	±8	±1.1	±0.4	±0.1	±8	±5	±16

[1] Assuming dose rates and internal background as in Table 5.1-2

short-term measurement (1 h) and for an installed instrument providing monthly mean dose rates. It is these variations that set the limit for the detection of small changes in dose rate.

5.2.2 Measurement statistics: The uncertainty for a counting system is easily calculated from the total count in the given period. For example the G-M counter producing 3.7 counts/s, of which 1 count/s is caused by environmental radiation, will have an uncertainty of ±8% (95% confidence level) in a one-hour count to determine the environmental gamma-ray component of background. The corresponding uncertainty over one month is ±0.3%. Corresponding data for a high-pressure ionization chamber have been calculated from a paper by Spiers et al. (1964) as ±0.4% in a one-hour period but for atmospheric-pressure chambers the uncertainty would be about five times higher (±2% in a one-hour measurement). Kastner et al. (1964) give a coefficient of variation of 0.5% for their atmospheric-pressure chamber over a period of six months; this is equivalent to an uncertainty of ±1.5% at the 95% confidence level. If the indication of a dose-rate meter is by means of a visual analogue display (e.g. a moving-coil meter) then uncertainties of about ±10% are possible owing to statistical fluctuations and errors in reading at reasonable values of the circuit time constant.

5.2.3 Supply voltage fluctuations: This contribution is very difficult to calculate and will vary between instruments. Normally in modern instruments fluctuations are less than ±1% but it is necessary to calculate the factor appropriate to each instrument. Facilities for battery checking are also important and variations between minimum and maximum charge should be determined.

5.2.4 Atmospheric conditions: The detection system will be subject to a wide range of conditions and some measurements at the extremes of temperature and humidity are desirable. In modern instruments the figures quoted in the Table 5.2-1 should be achieved. However, unsealed ionization chambers working at atmospheric pressure could be subject to much larger fluctuations and corrections for barometric pressure and temperature are then required. High relative humidity can affect insulators and high-value resistors or capacitors unless the system is properly sealed. Lightning can cause sudden transients if the instrument is not properly shielded. TLDs are subject to fading in high temperatures and exposure to direct sunlight is to be avoided.

5.3 Natural variability in environmental radiation

5.3.1 Introduction: In addition to variations of instrumental origin, there are other influences, such as barometric pressure, temperature and rainfall, that cause actual variations in the natural environmental radiation at a given site. These effects have been discussed in Chapter 2 but it is important to bring them together, firstly in order to provide a guide for the precision required by integrating instruments, and secondly to provide a limit below which it is unreasonable to expect measurements to be made of the additional dose rate from man-made sources. Two integrating periods of one hour and one month have been chosen to represent a single spot measurement and a typical continuous measurement respectively. A typical site based upon the soil activities given in Table 2.1-2 has been chosen as an example of the type of analysis that can be made. The detailed data are given in Tables 5.3-1(a) and (b) with a summary in Table 5.3-1(c). In these tables and the subsequent discussion the variability is expressed as two standard deviations of the appropriate measurement. The derivation of the data will be discussed below.

5.3.2 Terrestrial gamma radiation: The effects considered have been discussed in Sections 2.2 and 2.3 and the following is only a summary. The most significant influence is that of rain, which causes a reduction in dose rate of approximately 3% per centimetre of rainfall. An analysis of the monthly rainfall at Chilton, Oxfordshire, over two years shows a variability of $\pm$7 cm, which implies a variability of $\pm$20% in the dose rate. Similar variabilities have been assumed for the one-hour measurements since changes in the soil-moisture content are relatively slow.

The second major influence is the effect of barometric pressure and the general stability of the lower atmosphere on the diffusion of radon from the soil. The nuclide ^{220}Rn (^{232}Th series) is not significantly affected because of its short half-life, but the level of ^{222}Rn (half-life 3.8 d) can change quite markedly in the surface layers of the soil. The major contribution to the dose rate from the ^{238}U series is from daughters of ^{222}Rn, so any losses from the surface can result in quite large reductions in the dose rate. Using data from Kataoka et al. (1979), it can be estimated that the variability between months is

about ±13% of the dose rate from the ^{238}U series, with an hourly variability of ±45%. The difference between monthly and hourly values is based upon the variability in barometric pressure obtained from an analysis of meteorological data at Chilton, Oxfordshire. It is necessary to know the activity of ^{226}Ra in the soil to obtain an estimate of this effect; the dose rate from ^{40}K is not affected by atmospheric stability.

Snow will attenuate all terrestrial radiations and so measurements of small changes are not possible whilst snow cover is present. The snow density is a further variable and, as discussed in Section 2.3, a depth of 20 cm of snow reduces the terrestrial gamma-ray dose rate by 26% for a snow density of 0.1 g/cm^3 and by 58% for a snow density of 0.4 g/cm^3. This effect has not been included in the 'normal' variabilities.

5.3.3 Fall-out gamma radiation: Rain and snow will have effects on deposited fall-out similar to those observed for natural radioactivity in the soil. Barometric pressure and temperature will not affect the dose rate significantly. However, an analysis of the current dose rate from fall-out shows that for the years 1977—9 there was a ±20% effect caused by the continued testing of nuclear devices in the atmosphere and by seasonal variations in the transfer of long-lived radionuclides from the stratosphere to the troposphere and hence via rain to the ground.

The fall-out dose rate has been subject to large changes over the years (up to 20 times the current fall-out dose rate) and if there is a significant increase in the testing of nuclear devices in the atmosphere then the variability caused by this source could become more significant.

5.3.4 Atmospheric gamma radiation: A consequence of any reduction in the radon concentration in the surface soil will be a corresponding increase in the dose rate from atmospheric radioactivity. There is not complete compensation because the stability of the atmosphere, and the height over which it is stable, will control the radon concentration and hence the dose rate. Gale and Peaple (1958) have shown that the radon concentration varies very significantly during the day and also over the year. An analysis of their data suggests that a monthly

variability of ±150% and an hourly variability of ±200% might be expected. The data are not distributed normally but, since the absolute variability from this source is quite small relative to other effects, this lack of normality is ignored in the analysis.

5.3.5 Cosmic radiation: The effects of barometric pressure on the cosmic radiation have been discussed in Section 2.4, and UNSCEAR (1977) provides a considerable amount of data on the variability with temperature and time of year. Very occasionally solar flares will produce sudden increases in the cosmic-ray dose rate by a factor of up to five times the normal but these have been neglected in the determination of the normal variability. However, this could be important if a method using a rate of change of the dose rate were used.

5.3.6 Total natural variability: Most of the dose rate variabilities listed in Table 5.3-1(a) and Table 5.3-1(b) are correlated in some way and the influences (barometric pressure, temperature and rainfall) show a correlation as well. Some of the variables such as the dose rate from radon daughters in the soil and in the atmosphere have a negative correlation. Thus it is unreasonable to add the individual variabilities in quadrature and so, although it provides an overestimate, the arithmetic sum has been used to calculate the total natural variability in Table 5.3-1(c). Variabilities for measurements integrated or averaged over longer periods of up to a year and between years will be of a similar order to those obtained for the monthly variabilities.

The major contribution to the variability comes from the terrestrial gamma radiation and calculations on the soils derived from the different rock types listed in Table 5.1-3 show that, in general, the total variability is about ±25% of the total dose rate for a one-hour measurement and is about ±17% for the one-month average. More exact calculations can be made using the data for any locality if required.

5.3.7 Implications for measurement of environmental radiation: Comparing the systematic uncertainties of instrumental origin in Table 5.1-1, the random uncertainties of instrumental origin in Table 5.2-1 and the effect of natural variations in environmental dose rate in Table

TABLE 5.3-1: Effect of various influences on the variability of environmental radiation at one site; variability expressed as two standard deviations of the measurements

(a) Measurements of dose rate integrated or averaged over one hour

Influence, Range	Terrestrial gamma rays	Fall-out gamma rays	Atmospheric gamma rays	Cosmic rays	TOTAL
	Typical mean dose rate (nGy/h)				
	45	2.3	0.4	32	80
	Variability of dose rate (nGy/h)				
Pressure, ±70 mbar	±5.0[1]	...	±0.8[1]	±1.6	...
Temperature, ±10 °C	...	...		±0.6	...
Rainfall, ±7 cm	±9.0	±0.5		...	...
Time of year, ...	...	±0.5	...	±0.6	...
TOTAL VARIABILITY	±14.0	±1.0	±0.8	±2.8	±19

[1] Associated with a Ra-226 concentration of 26 Bq/kg (0.7 pCi/g)

(b) Measurements of dose rate integrated or averaged over one month

Influence, Range	Terrestrial gamma rays	Fall-out gamma rays	Atmospheric gamma rays	Cosmic rays	TOTAL
	Typical mean dose rate (μGy/yr)				
	400	20	3.5	280	700
	Variability of dose rate (μGy/yr)				
Pressure, ±20 mbar	±13[1]	...	±5[1]	±4	...
Temperature, ±10 °C	...	...		±6	...
Rainfall, ±7 cm	±80	±4		...	...
Time of year, ...	...	±4	...	±6	...
TOTAL VARIABILITY	±93	±8	±5	±16	±120

[1] Associated with a Ra-226 concentration of 26 Bq/kg (0.7 pCi/g)

(c) Summary of the total variability derived from the arithmetic sum of the variabilities arising from the separate influences; the typical mean dose rate is 80 nGy/h (8 μrad/h) or 700 μGy/yr (70 mrad/yr)

Time over which separate measurements are integrated or averaged	Variability of dose rate				
	(nGy/h)	(μrad/h)	(μGy/yr)	(mrad/yr)	(%)
One hour	±19	±1.9	±170	±17	±24
One month	±14	±1.4	±120	±12	±17

5.3-1(c), it is clear that instrumental uncertainties are relatively unimportant, and that any of the instruments discussed in this Guide have a performance that is adequate to measure environmental dose rate.

Smaller changes than those given in Table 5.3-1(c) can be detected if the radionuclide source is known and a Ge(Li) detector is used. The alternative method of using the rate of change of the dose rate may give misleading results after thunderstorms (Figure 2.3-2) or during solar flares, and other measurements are required if a proper interpretation is to be made.

It is doubtful whether changes of less than 120 μGy/yr (12 mrad/yr) can be considered as statistically significant and it is suggested that this variability in the natural environmental radiation be taken as a minimum value for all measurements. Single measurements over one hour cannot identify the source of changes of less than 20 nGy/h (2 μrad/h), which is equivalent to 170 μGy/yr (17 mrad/yr). These variations should be compared with the 'target' levels of between 100 μGy/yr and 500 μGy/yr (between 10 mrad/yr and 50 mrad/yr) inferred from national and international recommendations in Chapter 1. An analysis of the data given in Table 5.3-1 suggests that long-term (monthly) measurements are required if the lower levels are to be detectable and additional measurements and research are needed if changes caused by nuclear installations of less than 120 μGy/yr (12 mrad/yr) are to be detected.

6 CHOICE OF INSTRUMENTS AND CONDITIONS

When presented with the task of determining the level of environmental gamma-ray dose rate in the neighbourhood of a nuclear installation, four main decisions have to be made—the type of measuring instrument to be used, the ancillary equipment that will be required, the location of the measurement site and the experimental procedure to be adopted. These four topics are discussed in this chapter.

6.1 Type of measurements required

The choice of instrument is dependent upon a series of requirements, which the user must assess for himself. First there is the question of the type of measurement; this can range from a spot check to ensure that some legal limit is not being exceeded, to very precise and careful measurements to determine changes as small as 100 μGy/yr (10 mrad/yr). Secondly, the environment in which the detector is to be used has to be considered, particularly with regard to a wide range of temperatures and for operation in rain, snow and wind. There is no point in spending a great deal of effort in order to improve the precision of an instrument to much better than the variability of natural environmental radiation. On the other hand, improvements to other performance characteristics may be required in some circumstances. For example an energy dependence of ±25% from 50 keV to 3 MeV should be quite adequate for most measurements but if ^{16}N is present then it will be necessary to determine the energy dependence up to 9 MeV (Chapter 4).

The simplest and most reliable instrument for spot checks may incorporate either a Geiger—Müller (G-M) detector or an ionization chamber as described in Chapter 3. Normally the essential requirements are stability, portability, rapid warm-up period, ruggedness and a resistance against all weather conditions. Simple installed dosimeters based upon a TLD system to measure the integrated dose should be adequate for long-term control with reasonably good precision (Section 3.5). Long-term measurements to look for small occasional changes are probably best based on G-M counters with suitable circuits to examine the rate of change of the counting rate. This measurement would require many

supplementary determinations to eliminate natural variations caused by rain, radon daughters etc. In each case it is necessary to examine the list of uncertainties in Section 5.1 and decide which need to be reduced to make the measurement sufficiently reliable.

It is difficult to produce a completely objective comparison of different instruments in view of the varying requirements of different users but a comparison of the different types of instruments is given in Table 6.1-1 for a range of possible requirements. For each requirement the best instrument is given ten points and the others are compared on a scale between one and ten. The points for each instrument are not summed because the individual user may wish to weight individual requirements to a different extent.

TABLE 6.1-1: Relative advantages of instruments for measuring environmental gamma-ray dose rate based on a ten-point scale

Requirement	G-M counter	Ion chamber	Scintil-lator	Spectro-meter	TLD
Small systematic uncertainty	7	8	5	10	8
Small random uncertainty	6	10	6	8	8
Reliability	8	8	8	6	10
Ease of calibration	10	10	10	7	5
Ease of use	8	7	8	1	10
Portability	10	8	10	1	10
Cheapness	8	4	8	1	10
Ability to detect small changes	5	5	3	10	2
Ruggedness and weather resistance	8	8	6	4	10

6.2 Ancillary equipment

It is clear from earlier chapters that the measurement of environmental radiation is affected by many factors. In order to measure a small increase from a particular cause, additional equipment is required. Initially it is necessary to provide an energy calibration of the

detector over the range of energies of interest, e.g. 50 keV to 3 MeV or 9 MeV. This requires access to a laboratory with X-ray and gamma-ray sources and an accelerator for the higher energies. Secondly the internal background of the detector and its cosmic-ray response should be determined using a shielded facility. For pulsed instruments a measurement of the dead time is required. A facility for annual calibrations is necessary and a smaller source for regular weekly or daily checks is very useful. These facilities are required before any measurements are made.

Some measure of the cosmic-ray contribution is required for precise determinations, although an indirect estimate by means of a barometer should be adequate in most cases. Fluctuations caused by radon and thoron in the air may be small in most areas, but for very precise measurements a filter-paper sample of the air may be taken. Wind direction and speed may be important in some cases for measurements near to a nuclear installation. Soil samples returned to the laboratory for gamma-ray spectrometry are necessary if corrections are to be made for deposited fall-out from nuclear explosions. Some means of measuring precipitation including snow will assist in interpretation (see Section 2.3) and spurious influences, such as interference through the electricity supply or from radio-frequency noise, should be considered. Temperature and humidity can affect some detectors (e.g. TLDs); again the degree of sophistication to correct for these effects is dependent upon the precision required.

6.3 Choice of site

In certain cases it may be expedient to make spot measurements anywhere, but in order to obtain representative and reproducible results it is necessary to choose a site carefully, particularly for installed equipment. The site should be on a slight slope, well drained and grass-covered up to at least 30 m from the measuring point. There should be no trees, buildings or obstructions other than a small wooden hut to house the equipment and a security fence to keep out animals. As far as possible the geology should be unchanged within the site and it is advisable to take a series of soil samples for analysis of the natural radioactivity and nuclear-weapon debris. Finally it is necessary to make a survey with a field

instrument to ensure that there are no significant anomalies that might affect the measurements.

The choice of site may be influenced by other considerations such as the availability of a mains electricity supply, security of the equipment or the geographical location of a population in a prevailing wind direction. It may also be necessary to have a main monitoring site with a series of subsidiary stations with simpler integrating dosimeters such as TLDs.

6.4 Experimental procedure

Having chosen the most appropriate measuring instrument, ancillary equipment and site for the measurement, the next step is to establish an experimental procedure. In general this can be broken down as follows:

(a) effect on instrument of external and internal influences, e.g., dependence on energy, direction and temperature; internal background; cosmic-ray response; dead time etc.;

(b) calibration of instrument;

(c) method of measurement, including precautions to reduce errors;

(d) ancillary measurements, e.g. site uniformity, barometric pressure, rainfall;

(e) corrections to measurements, e.g. dead time, cosmic radiation;

(f) interpretation of measurements;

(g) presentation of results.

The procedures for measuring the instrument response for individual instruments are discussed in Chapter 3 with the general calibration procedures in Chapter 4. The method of measurement can be either a simple positioning of the instrument on a tripod for a short check (about one hour), or a permanent installation using battery-operated

equipment or with cables back to a waterproof hut. For spot checks a fixed procedure should be followed with the detector in a fixed orientation; e.g. a cylindrical detector would be aligned north—south in a horizontal plane if the cosmic-ray response has been determined in this orientation. Ancillary measurements have been discussed in Section 6.2 and these can be used to correct the observations, for cosmic-ray response for example. Interpretation may include the use of rainfall data, wind direction or a rate of change of dose rate to detect sudden changes. This latter effect can be due to sudden storms as well as the effects of nuclear installations. Finally in presenting the results it should be made clear what dose rate is being measured, e.g. terrestrial gamma radiation, natural environmental gamma radiation (including atmospheric gamma radiation) or total environmental radiation including cosmic radiation. Ideally the components should be presented separately. The uncertainty on any measurement should always be included and if the apparent difference due to a nuclear installation is presented then the significance of this measurement should be clearly stated.

7 DOSE EQUIVALENT TO BODY ORGANS

The purpose of this chapter is to show how organ dose equivalents and effective whole-body dose equivalents can be derived from field measurements and to give data on dose equivalents from the natural background radiation in the UK, against which dose equivalents from man-made gamma-ray sources can be compared.

7.1 Measurements free-in-air

Measurements of environmental gamma radiation are usually made in terms of absorbed dose rate to air, the detector being placed at a standard height of 1 m above the ground. Such an instrument will have previously been calibrated so that the reading will indicate absorbed dose rate to air 'free-in-air', i.e. the dose rate in the absence of the instrument and of any other scattering material except air. If the instrument is now replaced by a human body at the point of measurement, the absorbed dose rate to the body, or to any organ in the body, will not be the same as the instrument indication, partly because of the numerical difference between absorbed dose to air and absorbed dose to tissue, and partly because of the attenuation and scattering of the radiation by the body. There is however no difference between the numerical values of absorbed dose rate to tissue and dose equivalent rate, because a quality factor of unity applies for environmental gamma radiation.

Many instruments calibrated in terms of exposure rate free-in-air are still in use; a similar reasoning applies to these, although the numerical value of the instrument indication will be different. Methods of calculating dose equivalent to the body or body organs from the indication of an instrument calibrated free-in-air are discussed in the next section.

7.2 Calculation of dose equivalent to the body

If the relevant dose equivalent rates to populations are to be assessed, tissues or organs should be assigned that relate to specific

biological effects. As an indicator of somatic effects either the weighted mean dose equivalent to the whole body or specifically that to bone marrow could be chosen. For non-uniform irradiation of the whole body the International Commission on Radiological Protection (ICRP 1977, 1978) has recommended the following formula for the calculation of a risk-weighted mean whole-body dose equivalent:

$$H_E = \sum_T W_T H_T \qquad (7.2\text{-}1)$$

where H_E is the 'effective dose equivalent' for the whole body, and H_T is the dose equivalent to tissue T, which has a risk-weighting factor W_T allocated to it ($\sum_T W_T = 1$). In the following discussion, both the effective whole-body absorbed dose and the effective whole-body dose equivalent are averaged in this manner.

The absorbed dose rate to body organs is less than the absorbed dose rate to air, free-in-air, because overlying or adjacent tissues partially shield the organs from some or all of the incident radiation. Thus Bennett (1970) has suggested a screening factor of 0.8 for the red bone marrow and O'Brien (1975) has calculated that the absorbed dose to the various organs from environmental gamma radiation is within about $\pm$20% of the mean whole-body absorbed dose. Although there is some variation in the value of the screening factors, depending on assumptions made in the theoretical calculations and on the mode of the irradiation and phantom structure in the experimental methods, it appears that the factors for different

TABLE 7.2-1: Factors for converting measurements of absorbed dose to air free-in-air (and exposure free-in-air) into dose equivalent to body organs for 1 MeV gamma radiation; from Ashton and Spiers (1979)

Body organ	Dose equivalent to organ / Absorbed dose to air		Dose equivalent to organ / Exposure	
	(Sv/Gy)	(rem/rad)	(Sv/R)	(rem/R)
Bone marrow	0.80	0.80	7.0×10^{-3}	0.70
Whole body	0.87	0.87	7.5×10^{-3}	0.75
Gonads[1]	0.81	0.81	7.1×10^{-3}	0.71

[1] Approximate values

body organs lie within the range suggested by O'Brien. Screening factors have been measured by Ashton and Spiers (1979) using small thermoluminescence dosimeters in an anthropomorphic phantom. At an approximate mean energy of environmental gamma radiation (1 MeV) the factors determined by Ashton and Spiers (1979) were 0.87 Sv/Gy (rem/rad) for conversion from absorbed dose to air to whole-body dose equivalent, and 7.5×10^{-3} Sv/R (0.75 rem/R) for conversion from exposure to whole-body dose equivalent. Taylor and Webb (1978) used a factor of 0.70 Sv/Gy (rem/rad) for conversion from absorbed dose to air to whole-body dose equivalent.

The screening (or conversion) factors for bone marrow and the gonads are given in Table 7.2-1 for comparison with the whole-body factors. The factor for gonads is the mean of 0.70 Sv/Gy (rem/rad) for ovaries and 0.92 Sv/Gy (rem/rad) for testes and so this is only an approximate figure.

7.3 Environmental irradiation in the UK

Some examples of the average effective annual dose equivalent to the whole body from natural environmental gamma radiation in several areas of the UK are given in Table 7.3-1, as assembled by Taylor and Webb (1978), together with ranges of the estimates. The results are given in μSv/yr (0.1 mrem/yr) as population-weighted means with the quality factor taken to be unity. The calculations of the annual effective dose

TABLE 7.3-1: Annual effective dose equivalent to the population in the UK from environmental gamma radiation; from Taylor and Webb (1978)

Area	Annual dose equivalent	
	Average	Range of estimates
	(μSv/yr)	(μSv/yr)
England	370	260—460
Scotland	450	290—560
Wales	390	260—480
N. Ireland	420	280—540
WHOLE UK	380	270—490

equivalent assume that the ratio of effective dose equivalent to absorbed dose to air is 0.7, and that an individual spends 2.5 h outdoors each day. The authors estimate that the annual effective dose equivalent to individuals could range from about 200 μSv/yr to 1000 μSv/yr (20 mrem/yr to 100 mrem/yr) owing to local variations in the areas concerned.

The distribution of annual absorbed dose to red bone marrow from environmental gamma radiation calculated from measurements made by Spiers et al. (1964) at 679 different sites in the UK is shown in Figure 7.3-1; the average for each site has been

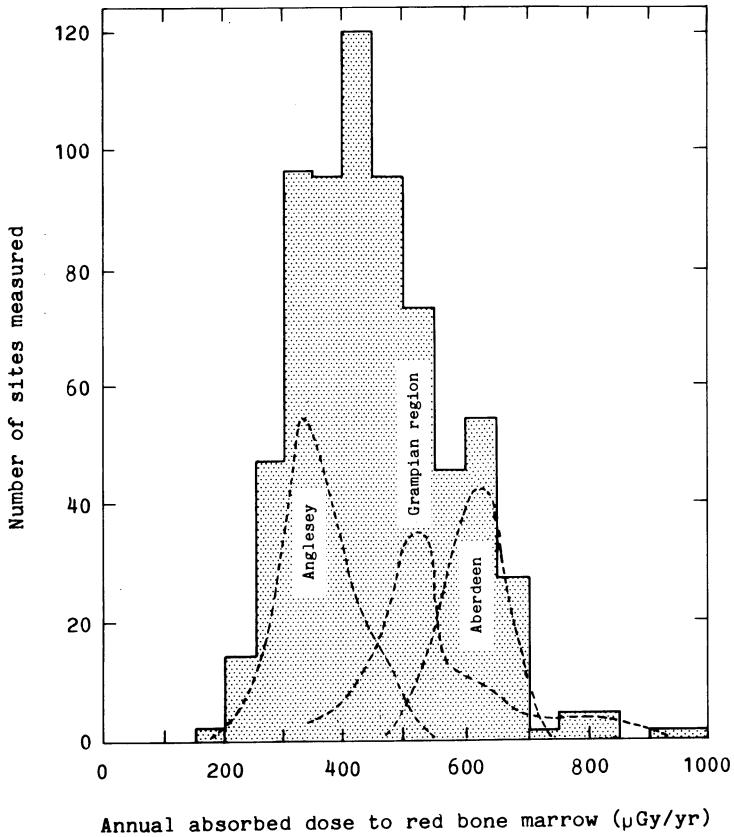

FIGURE 7.3-1: Distribution of annual absorbed dose to red bone marrow in the UK; broken curves indicate the proportions contributed to the total by the areas named (from Spiers et al., 1964, plus additional data)

weighted to allow for the proportion of time spent indoors and outdoors. The range corresponds closely to that suggested by Taylor and Webb (1978) although the doses are about 8% lower because

slightly different assumptions were made about the screening factors and the time spent outdoors.

Natural environmental gamma radiation is however only one of the significant contributions to the effective dose equivalent from all natural background radiations. It has been estimated that cosmic radiation contributes 310 μSv/yr (31 mrem/yr) and internally deposited natural radionuclides contribute 370 μSv/yr (37 mrem/yr) to the annual effective dose equivalent of an individual in the UK (Taylor and Webb, 1978). Unlike natural environmental gamma radiation, which varies from place to place, depending on local geology and building materials, the contributions from cosmic radiation and internal background radiation do not vary much with geographical location in this country.

There is also a further contribution to the dose equivalent, particularly to lung tissue, arising from inhalation of radon and its daughter products occurring naturally in the atmosphere. This contribution is very difficult to quantify. Radon concentrations vary over a range of about two orders of magnitude, depending on the nature of building materials and on ventilation, and the relation between effective dose equivalent and radon concentration is also uncertain. A recent estimate (NRPB, 1981) has suggested that the average annual effective dose equivalent from inhalation of natural radon in the UK is about 800 μSv/yr (80 mrem/yr).

8 CONCLUSIONS AND RECOMMENDATIONS

In this chapter the main conclusions are drawn together, with reference back to the relevant sections. Where appropriate, recommendations are made to emphasize good practice.

8.1 Natural environmental radiation

8.1.1 Composition: The general magnitudes of the individual components of the environmental absorbed dose rates to air are listed in Table 8.1-1 in units of μGy/yr for measurements at 1 m above the ground (Sections 2.1, 2.2, 2.3 and 2.4).

TABLE 8.1-1: Components of natural environmental radiation

Radiation	Dose rate (μGy/yr)
Terrestrial gamma radiation (~1 MeV)	200 to 1000
Atmospheric gamma radiation (~1 MeV)	0 to 10
Cosmic radiation	280 (sea level)
Fall-out gamma radiation[1] (~0.7 MeV)	20 (in 1978)

[1] Included to complete the general picture

8.1.2 Variation: The component dose rates listed in Table 8.1-1 are subject to variations that are not all capable of precise quantification. Some notes given below indicate only the general kind of variation that may be experienced in practice. Sometimes, but not always, an approximate allowance for the likely variation can be made in interpreting an observed dose rate.

(a) Site: The dose rate over a given site may vary with position; for example, a standard deviation of ±20% is possible in observations made over an area of a few square kilometres (Section 2.2).

(b) Rainfall: The dose rate above the ground depends on the water content of the soil, and observations made daily over a few months

suggest that over this period a standard deviation of a single measurement of ±10% may be expected from this cause. A rough guide to the effect of rainfall is that a decrement in dose rate of about 3% (±0.6%) may follow 1 cm of rainfall received over the preceding two to three days (Section 2.3).

(c) Snow cover: The gamma-ray dose rate above ground is also reduced by snow cover. The few data available on this suggest that the dose rate may be reduced by about 1.3% per centimetre of snow depth at a snow density of 0.1 g/cm^3 and by 3% per centimetre at a density of 0.4 g/cm^3 (Section 2.3).

(d) Cosmic radiation: Allowance for the increment in cosmic-ray mu-meson dose rate with decreasing barometric pressure can be made at the rate of 0.26% per millibar (0.35% per millimetre of mercury) under conditions of light shielding or at 0.17% per millibar (0.22% per millimetre of mercury) if the shielding is heavy, e.g. 10 cm of steel (Section 2.4).

8.2 Measuring instruments

8.2.1 General recommendations: The instruments considered in Chapter 3 can all be designed to have a sensitivity that is adequate for the measurement of gamma radiation at about the natural level. Not all commercial gamma-ray monitors may meet the requirements for measurements of environmental gamma radiation, and none can be used to provide a reliable measurement without certain precautions in their use and in the interpretation of their readings.

All the instruments will respond similarly to change in environmental gamma radiation arising from such causes as site variation, rainfall, snow or occasional depositions of the decay products of atmospheric radon. The response of instruments to cosmic radiation, however, can vary considerably. In Geiger–Müller (G-M) counters and ionization chambers, typically some 40% of a dose-rate reading at sea level can be contributed by cosmic radiation, depending, of course, on the level of the environmental gamma-ray dose rate. Cosmic-ray contributions of about 30% may also be present in the measurements made with TLD systems, and in all these cases due allowance for the cosmic-ray dose

rate and its variation should be made. However in instruments with scintillation detectors, pulse-height discrimination may keep the cosmic-ray contribution small, e.g. to a level of only about 5% of the total reading, and corrections for cosmic-ray variations are relatively unimportant. Without pulse-height discrimination, the cosmic-ray response in organic scintillators can be about 30% (Sections 3.1.3, 3.2.4, 3.3.3 and 3.5.8).

The extent to which the instrument reading is affected by internal radioactive contamination also varies with the type of instrument. The problem may be important in G-M tube instruments and TLDs but can be largely avoided or minimized in ionization chambers. Apart from the possible presence of potassium in the glass of the optical system, contamination is a lesser problem with scintillators (Sections 3.1.2, 3.2.2, 3.3.4 and 3.5.5).

Measurement of the dead time is required for all pulsed instruments. Electrical stability is important in all cases and high insulation is essential in ionization chambers. Stray currents and drifting potentials can present difficulties where d.c. amplifiers are involved. The elimination of unwanted electrical signals can only be considered in relation to the particular circuitry used (Sections 3.1.4, 3.2.3 and 3.3.4).

8.2.2 Choice of instrument: It is unnecessary and also undesirable to recommend a single instrument for the measurement of environmental gamma radiation because different purposes of measurement are met by different types of instrument. Requirements may also differ; a very flat energy characteristic, for example, may be relatively unimportant in one application, yet essential in another.

Where only a simple periodic check is required a cheap and reliable instrument may be adequate. This could be based on a G-M detector, an ionization chamber or a scintillator (Sections 3.1, 3.2 and 3.3), all of which satisfy the requirements of portability, rapid warm-up and robustness of design (Section 6.1).

For long-term measurements to detect and record changes, a more elaborate installed system is required, calling for more supplementary measurements and checks for operational stability. A suitable system

based on G-M detectors is illustrated in Section 3.1.5. Where gamma-ray-energy discrimination is also desired, an installed spectrometer instrument is needed (Section 3.4).

Simple installed dosimeters, using thermoluminescence methods, may be employed to measure the integrated dose (over one or more months) at multiple points over an area, and this method can yield adequate precision for longer-term control (Section 3.5). Such systems may of course be combined with other more precise, localized short-term methods.

8.2.3 Calibration: Instruments for measuring environmental gamma-ray dose rates must be calibrated, preferably in nGy/h or μGy/yr in air, and the calibration must carry traceability to the standards of a national standardizing laboratory (NPL in the UK). Frequently the calibration may be made by means of small gamma-ray sources and these themselves should have traceability to national standards (Chapter 4).

The energy dependence of the instrument should be determined and this should cover an energy range representative of the natural environmental gamma-ray spectrum (Sections 2.1 and 4.4). Calibration with a ^{226}Ra source only is not sufficient in itself because its energy spectrum is not a complete match for that of environmental gamma radiation. The response of the instrument to gamma radiation of about 6 MeV should also be determined, if only approximately, because possible additional radiation from some reactors may be in this energy region.

If there is asymmetry in the structure of an instrument that affects its directional dependence, this should be investigated and a 2π- or a 4π-weighted calibration factor determined (Section 4.5).

Frequent checks should be made to monitor the constancy of the gamma-ray response of the instrument and for this purpose a small source would be suitable, e.g. about 0.5 MBq or 10 μCi of ^{226}Ra (Section 4.8).

8.3 Choice of site and ancillary equipment

There should be a careful choice of measuring site in relation to a nuclear installation and any neighbouring population, in order to

provide stable conditions for serial measurements. An undisturbed, nearly flat and well drained site is desirable, grass-covered to a distance of 30 m from the point of measurement and unlikely to be disturbed by subsequent building operations. A radiometric survey of the site should be made to ensure that there are no significant anomalies that might affect the reading of the instruments (Section 6.3).

An installed system should be housed in a secure location such as a hut of light construction. If it is assumed that the instrument calibration and testing are first carried out in a suitably equipped laboratory, the main facilities required for the installed system are:

(a) the calibrated instrument itself, whether designed for hand operation or for continuous recording,

(b) a shielded radioactive source and jig for regular constancy checks (hand-operated or automatic),

(c) a shielded zone in which the cosmic-ray response of the instrument can be determined. (If, however, a once-only determination of the cosmic-ray response is made, e.g. by measurements over water, a barometer would be sufficient to enable subsequent cosmic-ray responses to be deduced),

(d) a rain gauge or soil-moisture meter for the interpretation of changes in natural gamma-ray dose rate under varying weather conditions (Section 6.2).

In addition to the installed system for monitoring changes in environmental gamma-ray dose rate, simultaneous area monitoring may well be provided at a series of subsidiary sites by simple integrating thermoluminescence dosimeters. These can provide cumulative doses over periods of one to several months to complement the short-term information provided by the central system (Section 6.4).

8.4 Monitoring programme

In order to provide day-to-day information reflecting on the operational safety of a reactor, continuous monitoring of the

environmental gamma radiation is necessary. If a recording system is not used the monitoring frequency should be daily or nearly daily. The advantage of a recording system is, however, very considerable since it ensures, for example, that the integrated dose from an escaping radioactive cloud is fully recorded. The combination of an installed recording system and an area system based on TLD offers the best total method for monitoring changes in environmental dose rate on both a temporal and spatial basis (Chapter 6).

8.5 Natural environmental dose equivalent

Although the dose rate from natural environmental gamma radiation varies from one locality to another, knowledge of the level for a neighbouring population is necessary before the effects of possible changes can be evaluated. The average dose equivalent to the body from environmental gamma radiation, weighted according to ICRP factors for different organs, has been estimated to lie in a range from 270 μSv/yr to 490 μSv/yr (27 mrem/yr to 49 mrem/yr) in the UK. Individual dose equivalents from natural environmental radiation are known to vary over a range at least as great as 200 μSv/yr to 1000 μSv/yr (20 mrem/yr to 100 mrem/yr) and, in a town or locality of moderate size, the variation between different sites may easily be some ±25% (Sections 2.2 and 7.3).

8.6 Statement, accuracy and significance of a measurement

The statement of dose should be given in absorbed dose to air in grays, with a momentary dose rate in nGy/h and a long-term dose rate in μGy/yr. In every case an estimated uncertainty (±) should be stated at the 95% confidence level and the basis of the estimation (statistical and systematic) should be given (Chapter 5 and Section 6.4).

It is unreasonable to attach significance to changes in dose rate that are comparable with the uncertainty of measurement or the common fluctuations of natural environmental radiation. From consideration of the magnitude of the natural environmental gamma-ray dose rate and its variations and the likely sources of error of measurement, a realistic estimate of what could be regarded as significant for measurements made over a period of a month would be approximately 120 μGy/yr (12 mrad/yr)

(Chapter 5). This estimate can be compared with the limits on the annual dose of low LET radiation of 500 μGy/yr (50 mrad/yr) for an individual in the general population (ICRP, 1977) and 100 μGy/yr (10 mrad/yr) for members of a 'critical group' (NRPB, 1978) (see Chapter 1 and Section 5.3). It must be emphasized that if changes in dose rate of less than 120 μGy/yr (12 mrad/yr) are to be established as arising from man-made sources, then methods considerably more elaborate than those discussed in this Guide will be required, coupled with the need for further research.

REFERENCES

AITKEN, M.J. (1968) Low level environmental radiation measurements using natural calcium fluoride. Proceedings of second international conference on luminescence dosimetry, held at Gatlinburg, Tennessee, September 1968, edited by J.A. Auxier, K. Becker and E.M. Robinson, Oak Ridge, Tennessee, US Atomic Energy Commission and Oak Ridge National Laboratory, Report No. CONF-680920, 281—290.

ALMOND, P.R., McGRAY, K., ESPEJO, D. and WATANABE, S. (1968) The energy response of LiF, CaF_2 and $Li_2B_4O_7$:Mn from 26 keV to 22 MeV. Proceedings of second international conference on luminescence dosimetry, held at Gatlinburg, Tennessee, September 1968, edited by J.A. Auxier, K. Becker and E.M. Robinson, Oak Ridge, Tennessee, US Atomic Energy Commission and Oak Ridge National Laboratory, Report No. CONF-680920, 401—423.

ASHTON, T. and SPIERS, F.W. (1979) Attenuation factors for certain tissues when the body is irradiated omnidirectionally. Phys. Med. Biol., 24, 950—963.

BECK, H.L. (1972) The physics of environmental radiation fields. The natural radiation environment, II, edited by J.A.S. Adams et al., Washington, DC, US Energy Research and Development Administration, Report No. CONF-720805, Vol. 1, 101—133.

BECK, H.L., DeCAMPO, J. and GOGOLAK, C.V. (1972) In-situ Ge(Li) and NaI(Tl) gamma-ray spectrometry, New York, US Atomic Energy Commission Health and Safety Laboratory, Report No. HASL-258.

BECKER, K. (1972) Environmental monitoring with TLD. Nucl. Instrum. Methods, 104, 405—407.

BECKER, K. (1973) Solid state dosimetry, Cleveland, Ohio, Chemical Rubber Co.

BELCHER, E.H. and GEILINGER, J.E. (1957) Improved scintillating media for radiation dosimetry. Br. J. Radiol., 30, 103—105.

BENNETT, B.G. (1970) Estimation of gonadal absorbed dose due to environmental gamma radiation. Health Phys., 19, 757—767.

BINDER, W. and CAMERON, J.R. (1969) Dosimetric properties of CaF_2:Dy. Health Phys., 17, 613—618.

BINDER, W., DISTERHOFT, S. and CAMERON J.R. (1968) Dosimetric properties of CaF_2:Dy. Proceedings of second international conference on luminescence dosimetry, held at Gatlinburg, Tennessee, September 1968, edited by J.A. Auxier, K. Becker and E.M. Robinson, Oak Ridge, Tennessee, US Atomic Energy Commission and Oak Ridge National Laboratory, Report No. CONF-680920, 43—53.

BRINCK, W., GROSS, K., GELLS, G. and PARTRIDGE, J. (1975) Special field study of the Vermont Yankee nuclear power station, Cincinnati, Ohio, US Environmental Protection Agency. Private communication.

BRITISH CALIBRATION SERVICE (1977) Supplementary criteria for laboratory approval—calibration of radiological protection level instruments, Teddington, Middlesex, British Calibration Service, Publication No. 0811.

BRYANT, P.M. (1978) The discharge to atmosphere of radionuclides from reprocessing plants and the associated radiation exposure of the public. Proceedings of seminar on radioactive effluents from nuclear fuel reprocessing plants, held in Karlsruhe, November 1977, Luxembourg, Commission of the European Communities, Report No. V/2266/78.

BUDD, T. (1976) The properties of CaF_2:Dy used as a thermoluminescent dosimeter at low doses, London, Her Majesty's Stationery Office, AERE Report No. R-8385.

BURCH, P.R.J (1954) Cosmic radiation: ionization intensity and specific ionization in air at sea level. Proc. Phys. Soc. A, 67, 421—430.

BURCH, P.R.J., DUGGLEBY, J.C., OLDROYD, B. and SPIERS, F.W. (1964) Studies of environmental radiation at a particular site with a static γ-ray monitor. The natural radiation environment, edited by J.A.S. Adams and W.M. Lowder, Chicago, University of Chicago, 767—779.

BURKE, G. de P. (1972) Investigations of CaF_2:Mn thermoluminescent dosimetry system for environmental monitoring, New York, US Atomic Energy Commission Health and Safety Laboratory, Report No. HASL-252.

BURKE, G. de P. and McLAUGHLIN, J.E. (1974) Performance criteria for environmental radiation monitoring with TLD systems. Trans. Inst. Electr. Electron. Eng. NS, 21, 444—455.

CAMBRAY, R.S., FISHER, E.M.R., PLAYFORD, K. and PEIRSON, D.H. (1978) Radioactive fallout in air and rain: results to the end of 1977. London, Her Majesty's Stationery Office, AERE Report No. R-9016.

CHESTER, J.P., CHASE, R.L. and WOOD, S. (1972) A digital environmental monitor, Brookhaven, Brookhaven National Laboratory, Report No. BNL-16922.

CLARKE, R.H. and WILSON, R. (1977) Individual and collective doses due to ^{41}Ar based on measured discharge rates from CEGB reactors. Proceedings of fourth international congress of International Radiation Protection Association, held in Paris, April 1977, Fontenay-aux-Roses, France, International Radiation Protection Association, Vol. 2, Paper No. 286, 577—580.

DALE, J.W.G, PERRY, W.E. and PULFER, R.F. (1961) A beta—gamma ionization chamber for substandards of radioactivity: I. Uses and calibration. Int. J. Appl. Radiat. Isot, 10, 65—71.

de PLANQUE, G. and GESELL, T.F. (1979) Second international intercomparison of environmental dosimeters. Health Phys., 36, 221—233.

DeCAMPO, J.A., BECK, H.L. and RAFT, P.D. (1972) High pressure argon ionization chamber systems for the measurement of environmental exposure rates, New York, US Atomic Energy Commission Health and Safety Laboratory, Report No. HASL-260.

DENHAM, D.H., KATHRAN, R.L. and CORLEY, J.P. (1972) A CaF_2:Dy thermoluminescent dosimeter for environmental monitoring, Richland, Washington, DC, Battelle Northwest Laboratories, Report No. BNWL-SA-4149.

DICKSON, H.W., KERR, G.D., PERDUE, P.T. and ABDULLA, S.A. (1976) Environmental gamma-ray measurements using in-situ and core sampling techniques. Health Phys., 30, 221—227.

DOUGLAS, J.A (1981) Setting up a station to monitor the radiation background, London, Her Majesty's Stationery Office, AERE Report No. R-10059.

ENDRES, G.W.R., KATHRAN, R.L. and KOCHER, L.F. (1970) Thermoluminescence personnel dosimetry at Hanford: II. Energy dependence and application of TLD materials in operational health physics. Health Phys., 18, 665—672.

FINCK, R.R., LIDEN, K. and PERSSON, R.B.R. (1976) In-situ measurements of environmental gamma radiation by the use of a Ge(Li) spectrometer. Nucl. Instrum. Methods, 135, 559—567.

FRANÇOIS, H., GUPTON, E.D., MAUSHART, R., PIESCH, E., SOMASUNDARAM, S. and SPURNY, Z. (1970) Thermoluminescent dosimeters. Personnel dosimetry systems for external radiation exposures, Vienna, International Atomic Energy Agency, Technical Report Series No. 109, 75—93.

GALE, H.J., HUMPHREYS, D.L.O. and FISHER, Eleanor M.R. (1964) The weathering of caesium-137 in soil. Nature, 201, 257—261.

GALE, H.J. and PEAPLE, L.H.J. (1958) A study of the radon content of ground-level air at Harwell. Int. J. Air Pollut., 1, 103—109.

GESELL, T.F., BURKE, G. de P., and BECKER, K. (1976) An international intercomparison of environmental dosimeters under field and laboratory conditions. Health Phys., 30, 125—133.

GIBSON, J.A.B. (1978) Calculation of the dose rate and air ionization from radioactive fallout deposited at Chilton, Oxon. (1951 to 1977), London, Her Majesty's Stationery Office, AERE Report No. M-2976.

GIBSON, J.A.B., RICHARDS, J.E. and DOCHERTY, J. (1969) Nuclear radiation in the environment: beta- and gamma-ray dose-rate and air ionization from 1951 to 1968. J. Atmos. Terr. Phys., 31, 1183—1196.

GOGOLAK, C.V. and MILLER, K.M. (1977) New developments in field gamma-ray spectrometry, New York, US Department of Energy, Report No. EML-332.

GROOM, D.J. and SHORT, A. (1979) Report on radioactive discharges, associated environmental monitoring, and personal radiation doses resulting from operation of CEGB nuclear sites during 1978, London, Central Electricity Generating Board, Report No. HS/R 754/79.

GUDIKSEN, P.H. and CRITES, T.R. (1976) A comparison of gamma-ray exposure-rate measurements at Bikini Atoll. Proceedings of ninth mid-year topical symposium on operational health physics, held at Denver, Colorado, February 1976, California, University of California Radiation Laboratory, UCRL-77532.

HERBST, W. von and HUBNER, G. (1961) Untersuchungen über die durchringende äusser Umgebungsstrahlung. Atomkernenerg., 6, 75—81.

ICRP (1977) Recommendations of the International Commission on Radiological Protection. Publication No. 26. Ann. ICRP, 1(3).

ICRP (1978) Stockholm meeting of the International Commission on Radiological Protection. Publication No. 28. Ann. ICRP, 2(1).

ICRU (1970) Linear energy transfer, Washington, DC, International Commission on Radiation Units and Measurements, Report No. 16.

JAGIELAK, J., GWIAZDOWSKI, B., PENSKO, J. and ZAK, A. (1974) Problems in calibration of instruments for environmental gamma exposure dose measurements. Environmental surveillance around nuclear installations, Vienna, International Atomic Energy Agency, 337—349.

JONES, A.R. (1977) The application of an automated thermoluminescent dosimetry systems to environmental γ-ray dosimetry, Chalk River, Canada, Atomic Energy of Canada Ltd., Report No. AECL-5835.

KASTNER, J., SHONKA, F.R. and ROSE, J. (1964) A muscle-equivalent environmental radiation meter of extreme sensitivity. The natural radiation environment, edited by J.A.S. Adams and W.M. Lowder, Chicago, University of Chicago Press, 655—660.

KATAOKA, T., IKEBE, Y. and MINATO, S. (1979) Influence of the distribution of ^{222}Rn concentrations under the ground on natural gamma-ray flux density and exposure rate. Health Phys., 37, 669—675.

KOLB, W. and LAUTERBACH, U. (1971) Dose rate meter for measuring environmental radiation. Proceedings of international symposium on rapid methods for measurements of radioactivity in the environment, held at Neuherberg, Federal Republic of Germany, July 1971, Vienna, International Atomic Energy Agency, STI/PUB/289, 565—572.

KOLB, W. and LAUTERBACH, U. (1974) The improved scintillation dosimeter PTB 7201. Radiation protection and environmental protection: proceedings of eighth annual meeting of the Fachverband für Strahlenschutz, held at Helgoland, Federal Republic of Germany, September 1974, Eggenstein—Leopoldshafen, Zentralstell für Atomkernenergie Dokumentation, AED-CONF-74-725-000, Vol. 2, 662—690.

LINDEKEN, C.L., PETERSON, K.R., JONES, D.E. and McMILLEN, R.E. (1973) Geographical variations in environmental radiation background in the United States. Proceedings of second international symposium on natural radiation environment, California, University of California Radiation Laboratory, Report No. UCRL-73822.

LOWDER, W.M. and de PLANQUE, G. (1977) The response of LiF thermoluminescence dosimeters to natural environmental radiation, New York, US Department of Energy, Report No. HASL-313.

McLAUGHLIN, J.E. (1976) Environmental radiation measurements. Proceedings of NBS fifth anniversary symposium on the measurements for safe use of radiation, held at Gaithersburg, USA, March 1976, edited by S.P. Fivozinsky, Washington, DC, National Bureau of Standards, SP 456, 233—248.

McLAUGHLIN, J.E. (1977) Instrument calibrations for environmental surveillance. Proceedings of international symposium on national and international standardisation of radiation dosimetry, held at Atlanta, USA, December 1977, Vienna, International Atomic Energy Agency, Report No. SM-222, 475—490.

MRC (1956) The hazards to man of nuclear and allied radiations, London, Her Majesty's Stationery Office, Cmnd. 9780.

MRC (1960) The hazards to man of nuclear and allied radiations, London, Her Majesty's Stationery Office, Cmnd. 1225.

NCRP (1975) Natural background radiation in the United States, Washington, DC, National Council on Radiation Protection Measurements, Report No. 45.

NCRP (1976) Environmental radiation measurements, Washington, DC, National Council on Radiation Protection Measurements, Report No. 50.

NRPB (1978) The application of ICRP recommendations: advice to the expert group reviewing the White Paper Cmnd. 884 'The control of radioactive wastes', London, Her Majesty's Stationery Office, ASP2 and Addendum A.

NRPB (1981) Radiation exposure of the UK population. Radiol. Prot. Bull., (39), 3—5.

O'BRIEN, K. (1975) Unpublished calculations (Health and Safety Laboratory, US Atomic Energy Commission, New York 10014, USA).

O'BRIEN, K. (1978) The response of LiF thermoluminescence dosimeters to the ground level cosmic-ray background. Int. J. Appl. Radiat. Isot., 29, 735—739.

PEIRSON, D.H. and SALMON, L. (1959) Gamma radiation from deposited fallout. Nature, 184, 1678—1679.

PENSKO, J. (1967) Absolute method of the gamma-ray background determination by means of NaI(Tl) scintillation counter. Nucleonika, 12(4), 319—330.

RAGAINI, R.C., JAMES, D.E., HUCKABAY, G.W. and TODACHINE, T. (1974) Terrestrial gamma-ray surveys at preoperational nuclear power plants using an in-situ Ge(Li) spectrometer, California, Lawrence Livermore Laboratory, Report No. UCRL-75864.

ROBERTS, W.E. and SPEIGHT, D.L. (1964) Gamma scintillation dose-rate meter type BP1, Berkeley, Gloucestershire, Central Electricity Generating Board, Report No. RD/B/N338.

ROSSI, B. (1948) Interpretation of cosmic ray phenomena. Rev. Mod. Phys., 20, 537—583.

SCHULMAN, J.H. (1967) Survey of luminescence dosimetry. Proceedings of symposium on luminescence dosimetry, US Atomic Energy Commission Symposium Series No. 8, held at Stanford University, USA, June 1965, edited by F.H. Attix, Washington, DC, US Atomic Energy Commission, 3—33.

SHAMBON, A.W. (1972) Some implications of a laboratory study of LiF dosimeters for environmental radiation measurements, New York, US Atomic Energy Commission, Report No. HASL-251.

SHAMBON, A.W., LOWDER, W.M. and CONDON, W.J. (1963) Ionization chambers for environmental radiation measurements, New York, US Atomic Energy Commission Health and Safety Laboratory, Report No. HASL-108.

SHAMOS, M.H., LIBOFF, A. and SIDEROWITZ, J. (1964) Thin-walled ion chambers for low-level radiation environment. The natural radiation environment, edited by J.A.S. Adams and W.M. Lowder, Chicago, University of Chicago Press, 627—641.

SHONKA, F.R. (1962) Vibrating quartz-fiber electrometer. Radiology, 78, 112.

SIEVERT, R.M. and HULTQVIST, B. (1952) Variations in the natural gamma radiation background in Sweden. Acta Radiol., 37, 388—398.

SIEVERT, R.M. and HULTQVIST, B. (1957) High pressure ionization chambers: some Swedish investigations of the radioactivity of the body. Br. J. Radiol. Suppl. 7, (1), 1—12.

SPIERS, F.W. (1960) Die Messung der naturlichen Umgebungs-gammastrahlung. Strahlentherapie, 3(1), 65—74.

SPIERS, F.W., McHUGH, M.J. and APPLEBY, D.B. (1964) Environmental γ-ray dose to populations: surveys made with a portable meter. The natural radiation environment, edited by J.A.S. Adams and W.M. Lowder, Chicago, University of Chicago Press, 885—905.

TAYLOR, E.E. and WEBB, G.A.M. (1978) Radiation exposure of the UK population, London, Her Majesty's Stationery Office, NRPB Report No. R-77.

THOMPSON, I.M.G., LAVENDER, A., SHIPTON, R.G. and GOODWIN, J. (1971) The energy response of some commercial beta/gamma and neutron survey instruments. Proceedings of symposium on advances in physical and biological detectors, held in Vienna, November 1970, Vienna, International Atomic Energy Agency, 505—531.

UNSCEAR (1962) Report of the United Nations Scientific Committee on the Effects of Atomic Radiation, General Assembly: 17th session, supplement 16 (A/5216), New York, United Nations.

UNSCEAR (1966) Report of the United Nations Scientific Committee on the Effects of Atomic Radiation, General Assembly: 21st session, supplement 14 (A/6314), New York, United Nations.

UNSCEAR (1972) Ionizing radiation: levels and effects, Report of the United Nations Scientific Committee on the Effects of Atomic Radiation, General Assembly: 27th session, supplement 25 (A/8725), vol. 1, New York, United Nations.

UNSCEAR (1977) Sources and effects of ionizing radiation, Report of the United Nations Scientific Committee on the Effects of Atomic Radiation, General Assembly: 32nd session, supplement 40 (A/32/40), New York, United Nations.

WHITING, F.B (1965) Private communication, October 1965. I. and A. P. Division, Atomic Energy Research Establishment, Harwell, Didcot, Oxfordshire, OX11 ORA, United Kingdom.

WILSON, R. (1977) The measurement of ^{41}Ar release at Magnox stations, Berkeley, Gloucestershire, Central Electricity Generating Board, Report No. RD/B/N3874.

WILSON, R. (1978) Measurement of environmental gamma-ray spectra near Heysham power station, Berkeley, Gloucestershire, Central Electricity Generating Board, Report No. RD/B/N4387.

YAMASHITA, T., NADA, A., ONISHI, H. and KITAMURA, S. (1971) Calcium sulfate activated thulium or dysprosium for thermoluminescence dosimetry. Health Phys., 21, 295—300.