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SAFETY SERIES No. 114

SAFETY PRACTICES

# safety series

# Direct Methods for Measuring Radionuclides in the Human Body

INTERNATIONAL ATOMIC ENERGY AGENCY, VIENNA, 1996

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Basic requirements which must be satisfied to ensure safety for particular activities or application areas.

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# DIRECT METHODS FOR MEASURING RADIONUCLIDES IN THE HUMAN BODY

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# FOREWORD

The production of radionuclides as a consequence of activities in the nuclear fuel cycle and the use of radioactive materials in science, medicine and industry give rise to the possibility of internal contamination of workers and members of the public. This has prompted the development of techniques for the direct assessment of internal deposits of radioactive materials through the detection of photon fluxes emerging from the body. Such techniques were first used some 60 years ago, before the discovery of nuclear fission, the development of particle accelerators for isotope production and the general availability of artificially produced radionuclides.

The IAEA has always recognized monitoring for internal contamination as an important element in radiological protection, and it has promoted the development of methods of assessment by supporting research in Member States and by organizing meetings on the subject. The proceedings of four symposia (in Vienna in 1961, Heidelberg in 1964, Stockholm in 1971, and Paris in 1984) contained contributions relevant to the monitoring and evaluation of internal radiation dose, including many on the development and application of techniques for direct assessment of radionuclides in the body. More recently, awareness of the value of these techniques has been extended owing to their well publicized application in assessing internal contamination in members of the public following the nuclear accident at Chernobyl in Ukraine and the radiological accident at Goiânia in Brazil.

Other useful information is available in various journals and technical reports; however, its presentation is often complex and detailed. For this reason the IAEA has decided to produce the present Safety Practice.

The Publication is the result of the efforts of numerous people who have provided material and drafted and reviewed the report. The IAEA gratefully acknowledges the assistance of all these contributors.

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# 1. INTRODUCTION

## 1.1. BACKGROUND

Occupational exposure leading to intakes of internally incorporated radionuclides can occur as a result of various activities. This includes work associated with the different stages of the nuclear fuel cycle, the use of radioactive sources in medicine, scientific research, agriculture and industry, and occupations which involve exposure to enhanced levels of naturally occurring radionuclides.

In 1987 the IAEA published a Safety Guide on basic principles for occupational radiation monitoring [1] which set forth principles and objectives of a strategy for monitoring exposures of workers. Since drafting of the present Safety Practice commenced, the International Basic Safety Standards for Protection against Ionizing Radiation and for the Safety of Radiation Sources (BSS) have been issued [2]. On the basis of the principles laid down in the BSS, the 1987 Safety Guide is also being revised, and recommendations on the assessment of the occupational intake of radioactive materials are to be added.

The present Safety Practice, which deals with direct measurement of radionuclides in the human body, is the first to be published in this area.

## 1.2. OBJECTIVE

This Safety Practice provides information on the establishment and operation of facilities for the measurement of body activity by direct methods, both in general application and in a range of specific situations.

## 1.3. SCOPE

The emphasis in this publication is on measurements of body radioactivity made in programmes of internal dosimetry for occupationally exposed personnel, or in investigations following incidents. The less rigorous systems for rapid screening of employees leaving controlled areas are not considered. The publication is also not concerned with the design of monitoring regimes, which must reflect local conditions and comply with national regulations, or with the use of biokinetic models to derive intakes of radionuclides from the results of monitoring.

The publication is also relevant to assessments (as distinct from initial rapid screening) of larger groups following general contamination of the environment. Strategies for the initial screening of such groups are discussed elsewhere [3]. It may also be useful to those concerned with applications in clinical medicine or biomedical

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research. At the technical level, the publication is addressed primarily to those experienced in the more common applications of gamma ray spectrometry, concentrating on techniques specific to the measurement of radioactive material in the whole body or in particular organs.

### 1.4. STRUCTURE

The report comprises a general text in five sections, followed by eight annexes. The general text describes equipment, techniques for investigation and measurement of radionuclides in different organs and the interpretation of the results.

The annexes contain more detailed descriptions of different methods of investigation for specified radionuclides, including isotopes of uranium, $^{241}Am$, $^{60}Co$, $^{137}Cs$, $^{125}I$ and $^{131}I$, as well as mixtures of radionuclides.

## 2. CONDITIONS FOR USING DIRECT METHODS FOR MEASURING RADIONUCLIDES IN THE HUMAN BODY

### 2.1. GENERAL

In most in vivo counting applications, photon detectors are positioned at selected points on or near the body. Usually, at least partial shielding of the detector and/or of the subject will be needed to reduce the interfering response from ambient radiation. In some cases anticoincidence techniques may be required to achieve sufficient discrimination. Electrical signals from the detectors must be amplified and processed, leading to a gamma ray spectrum which will most conveniently be stored on magnetic media in digital form. Procedures are necessary to separate the response attributable to a given nuclide in the body from that due to ambient radiation and to components associated with other sources of body radioactivity. The extracted response must be converted into an assessment of body or organ content through appropriate calibration procedures (see Section 4.4).

### 2.2. SENSITIVITY AND ACCURACY

The *sensitivity* of a procedure is influenced by the size of random variations affecting a measurement. The most important of these will generally concern counting statistics but fluctuations in the level of background radiation and spurious effects from unstable equipment may contribute. Moreover, the ability to detect small

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changes in a deposit between successive measurements will be impaired if the conditions of measurement are not adequately standardized. The required sensitivity varies over a wide range depending on the nature of the contamination, on the purpose of the investigation and, if the technique is to be used for routine monitoring, on the frequency of measurement. For many common fission and activation products, minimum detectable activities (*MDA*s) (see Annex I) of several kilobecquerels may suffice; indeed, in certain phases of an emergency monitoring programme, higher *MDA*s could be accepted [3]. By contrast, with long lived alpha particle emitters, the residual organ deposit following occupational intake at the annual limit may be as low as 10 Bq.

The *accuracy* depends on systematic errors in the assessment. In general these will relate mainly to the validity of the calibration procedure (see Section 4.4), but they may arise also from errors in the adopted values of reference sources and, in some cases where organ rather than whole body contamination is to be estimated, as a result of interference from deposits in adjacent regions. The accuracy required depends on the expected or assessed level of contamination in relation to action levels: a result valid only as an order of magnitude may be wholly adequate if it implies only a trivial intake, while greater accuracy should obviously be expected if the estimate approaches or exceeds established action levels.

## 2.3. EQUIPMENT

### 2.3.1. Detectors

Depending on the application, important considerations in specifying the type and quality of the detector may include intrinsic detection efficiency, availability in sufficient size and at acceptable cost, energy resolution, background response in spectral regions of particular interest, potential for anticoincidence background suppression and long term stability of operation.

*Inorganic scintillators*

Inorganic crystals of high atomic number materials, usually thallium activated sodium iodide or NaI(Tl), are most commonly used in assessing internal contamination by energetic ($> 100$ keV) photon emitters, including many fission and activation products. They are readily available in sizes of up to 300 mm in diameter and 150 mm in thickness, although much smaller crystals suffice for the likely requirements in radiological protection. They can satisfy most of the basic requirements, provided that care is taken to ensure that the crystal, its housing and associated components (photomultiplier, dynode chain, etc.) are acceptably free of radioactive

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material. The isotope $^{40}K$, and radium and thorium are often present and may elevate the background response substantially. The energy resolution is generally adequate, provided that deconvolution procedures (Section 4.3) are available to separate overlapping spectral contributions from multiple contaminants. However, these procedures are unlikely to be able to cope with highly complex spectra, such as those from fresh fission product mixtures.

The inclusion of a second inorganic scintillator (usually thallium activated caesium iodide) as an anticoincidence guard, interposed between a thin (1–10 mm) NaI(Tl) crystal and the photomultiplier tube and optically coupled to both, improves detection limits for certain nuclides emitting only photons of low energy, such as those of plutonium (13–20 keV) and $^{241}Am$ (60 keV) [4]. Such a device is commonly known as a 'phoswich'. The anticoincidence is effected through differences in the decay times of the two scintillators, and consequently in the rise times of the pulses at the anode of the photomultiplier tube, which result ultimately in different pulse shapes at the output from the preamplifier. The anticoincidence gating reduces substantially the low energy background of the outer crystal and also its response to scattered photons originating from the body's content of natural $^{40}K$. It can provide similar benefits in reducing interference from any other energetic photon emitters present. Thin entrance windows of low atomic number material are necessary to avoid excessive attenuation of low energy photons. Beryllium is most commonly used, in thicknesses of up to 1 mm, but frequently contains unacceptable levels of uranium series nuclides emitting photons that may cause interference in critical spectral regions.

*Organic scintillators*

Solid organic scintillators can be made by impregnating plastic materials with anthracene. They can be made in very large sizes (e.g. 60 cm × 40 cm × 10 cm [5]) but require several photomultiplier tubes to achieve even a modest energy resolution, and in consequence have not been widely adopted for radiological protection purposes. They could be considered only where the interest was in a single nuclide, or in a mixture whose composition was reliably known, and where interference from the body's natural $^{40}K$ could be either neglected or inferred from measurements prior to the subject's exposure. Organic scintillators can also be incorporated into liquid solvents. Geometries approaching $4\pi$ can be produced with such solutions contained in annular tanks [6], but they suffer from the same restrictions as plastic scintillators.

*Semiconductor detectors*

Semiconductor detectors have major advantages in energy resolution, allowing: (i) the unambiguous identification of various photon emitting radionuclides;

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(ii) the quantitative assessment of mixtures of nuclides whose spectral features may be separated by a few kiloelectronvolts or less; and (iii) the elimination of major uncertainties in the level of a continuum of scattered radiation underlying a spectral peak. In the assessment of nuclides emitting photons of energy of up to 200 keV which are present in localized regions of the body, these advantages more than compensate for the inconvenience of having to maintain cooling to liquid nitrogen temperatures.

In this respect, high purity germanium detectors, which require the provision of cooling only when a bias voltage has to be applied to the diode for the purposes of a measurement, are preferable to lithium drifted types which need continuous cooling whether or not the bias voltage is applied. For whole body counting, their current availability only in fairly small sizes, often only 60 mm diameter, is a disadvantage, making it difficult to achieve both adequate sensitivity and uniformity of response with a single detector. However, examples of such use are given in Annexes II and III and Annex IV illustrates how these conflicting requirements can be reconciled by combining detectors in arrays. Semiconductor detectors may also be useful as an adjunct to scintillation counters to facilitate the identification of contaminants.

Compact arrays of three to six detectors are becoming common in the monitoring of contamination in specific organs such as the lungs and liver [7, 8]. Used individually, semiconductor detectors have been applied to the investigation of radioactive deposits in small organs or regions, such as the thyroid or the site of a contaminated wound. As with scintillation counters, detectors fabricated from low activity materials are preferable. Otherwise, peaks may occur in the background spectrum and these may complicate the evaluation of nuclides contributing in the same energy region. From the operational standpoint, a major disadvantage of these detectors is that they are prone to sudden failures in their cooling and/or vacuum systems. It is important to maintain spare units as replacements.

*Gas filled detectors*

Large area proportional counters with anticoincidence guard layers can offer energy resolutions intermediate between those of scintillation counters and semiconductor detectors and, with acceptable detection efficiency at energies below 30 keV, they were at one time seen as the most profitable approach to the assessment of plutonium in lungs. Several designs were produced by individual laboratories [9–11] but fell into disuse with the development of the phoswich, which offered greater sensitivity and robustness. There has been a recent attempt to refine these earlier designs in a multilayer detector, with six-sided anticoincidence and other advanced background suppression features originally developed for X ray astronomy. However, this gave no major improvement over phoswich detectors in the assessment of pulmonary plutonium [12].

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### 2.3.2. Electronics

The term 'electronics' refers to the ancillary equipment required to extract, amplify and sort electrical signals from the detector, converting them ultimately into a pulse height distribution where appropriate. Facilities for the long term storage of spectra and for their processing and analysis will be discussed separately (Section 2.3.3). The term therefore includes: preamplifiers (which are often built into the detector housing, and whose primary function is to provide a low impedance source enabling the signals to be transmitted through long cables without loss or degradation); main amplifiers and analog-to-digital converters (ADCs); units with specialized functions required for phoswich detectors (Section 2.3.1), such as pulse rise time discriminators, linear gates, etc., or for controlling the movement of detectors scanning the body (Section 3.2.4); and modules providing DC power supplies for other units and for the operation of the detectors themselves, which may require up to 5 kV according to type.

The general requirements for such equipment do not differ from those applying in other areas of gamma ray spectrometry. Compatibility of impedances and input/output signal characteristics is required, and flexibility in the selection of shaping time constants for optimum signal to noise ratios is important, particularly in maximizing the energy resolution available from semiconductor detectors. An additional consideration deserves emphasis for users who do not have specific experience of thick scintillation counters: amplifiers should have good overload characteristics if the large signals resulting from minimum ionizing cosmic ray particles traversing the detector are not to result in spurious effects distorting the recorded background gamma ray spectrum. The satisfactory operation of all electronic equipment will depend on the quality of the mains electricity supply, the elimination of ground loops and the proper provision of housing in a suitable, stable environment.

### 2.3.3. Storage and processing of data

At its simplest, the acquisition of data may entail no more than manually noting the reading of a rate meter connected to the output of a single channel analyser replacing the ADC envisaged earlier (Section 2.3.2), although this procedure would become tedious and subject to error if assessment of more than one radionuclide were contemplated. In general, there will be a need to record, in a secure and readily retrievable form, photon energy spectra up to at least 1.6 MeV. This will accommodate the contributions from the body's content of $^{40}K$, which may need to be taken into account in assessing those from other nuclides. This upper limit may need to be extended, particularly if assessment of certain natural radioelements is anticipated. The necessary channel widths vary from several hundred electronvolts to several tens of kiloelectronvolts, according to the type of detector and the particular application. The output from the ADC is routed to some intermediate storage

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medium for accumulation of the spectrum. This intermediate storage is most conveniently the memory of a small computer, from which the spectrum can be transferred at the end of a measurement to long term storage on magnetic media. The multi-megabyte hard disks common in personal computers, either as integral features or peripheral equipment, could accommodate several years' output for many laboratories.

The use of such computer based arrangements facilitates both elementary manipulations of spectra, such as background subtraction, and also more ambitious operations such as adjustment for instrumental drift, resolution of a spectrum into its several components by linear regression analysis (Section 4.3) and peak search, evaluation and identification procedures. Less conveniently, the spectrum may be accumulated by a multichannel analyser possessing limited in-built processing facilities, with the more demanding operations performed off-line by a separate small or mainframe computer. Whatever form of retention is adopted, appropriate arrangements should be made for duplicate storage to avoid loss of data through equipment malfunction or other mishaps. Facilities for rapid retrieval of spectra from long term storage and for displaying them in digital and graphical printed form are desirable.

In some applications multiple spectra may need to be accumulated independently during a measurement (e.g. the spectra from individual detectors in an array, or those from the two scintillators in a phoswich detector). In such cases 'multiplexing' arrangements are required with the ADC, to route signals to the appropriate area of memory.

### 2.3.4. Shielding of the installation

*Purpose*

The purpose of shielding is threefold. First, to reduce the background radiation from cosmic rays and from radioactive materials naturally present in the local environment to the level necessary for the sensitivity required; second, to minimize or eliminate interference from even the movement of radioactive materials in or near the laboratory; and third, to reduce perturbations in the counter background response which occur because the subject's body distorts the ambient radiation field through absorption, scattering and other processes. This third effect cannot be wholly eliminated, since a residual flux will be present inside the most massive and elaborate shield, as a result of the hard component of cosmic rays. Unless semiconductor detectors are employed, this flux makes it necessary to substitute an inactive phantom for the subject in assessing the relevant background response. However, adequate shielding will minimize, and for most purposes make negligible, the effect on this background of variations in body size.

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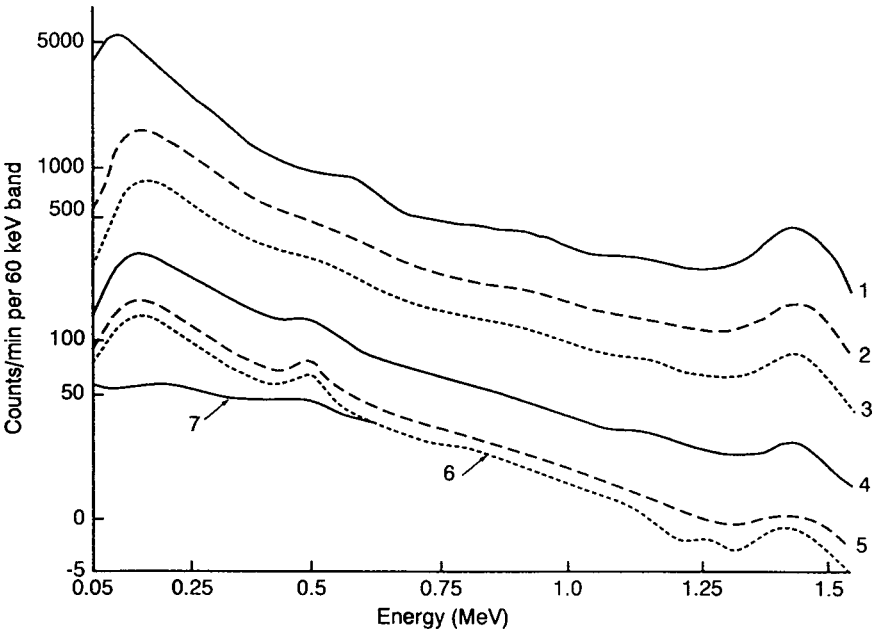


*FIG. 1. Reduction in gamma ray background of a 203 × 102 mm NaI(Tl) crystal shielded by various thicknesses of iron and lead. Internal dimensions of shield 2.4 m × 2.2 m × 1.9 m. (1) — no shield; (2) — 19 mm Fe; (3) — 38 mm Fe; (4) — 76 mm Fe; (5) — 140 mm Fe; (6) — 203 mm Fe; (7) — 203 mm Fe + 3 mm Pb.*

### *Choice of materials*

The requirements for primary shielding materials are: high attenuation of gamma rays, requiring high atomic number and density; freedom from unacceptable concentrations of natural or artificial radionuclides; and suitable mechanical properties for fabrication and assembly. The aim should be for optimal background reduction in relation to weight and cost. Steel or lead are most commonly used. The effect on counter background of all-round shielding with various thicknesses of steel (Fig. 1) has been reported [13, 14]. If lead is chosen, there will be characteristic X rays, induced by ambient radiation or by the subject's gamma ray emissions; if these interfere in a critical energy region, they may be removed through an inner lining of a few millimetres of cadmium or tin, with the ensuing Cd or Sn X rays eliminated if necessary by a further lining of steel or copper. For major installations (well screened shadow shield or shielded room), typical thicknesses for the primary material are 50–100 mm lead or 100–200 mm steel. Smaller thicknesses of these materials may suffice if external, supplementary shielding is present, for example

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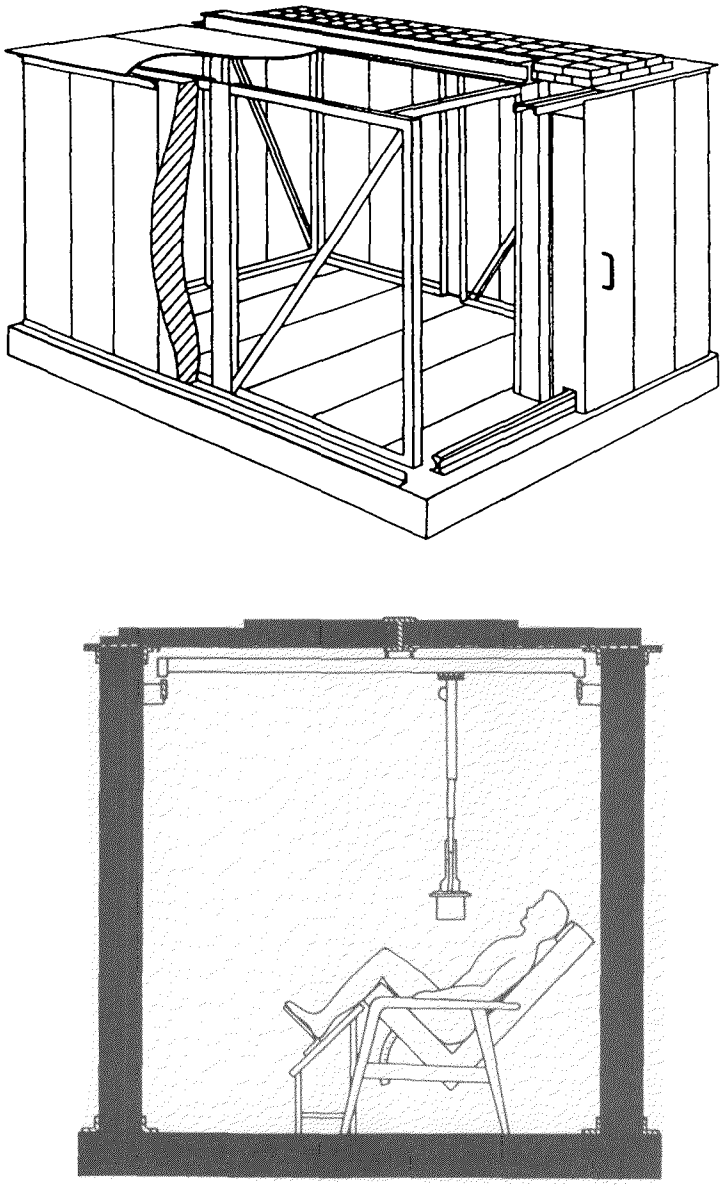

*FIG. 2. Shielded room.*

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with water or low activity sand or by virtue of the laboratory's location underground or in the basement of a multistorey building; in these latter situations much will depend on the natural radioactivity of the local strata.

*Design*

The most effective, convenient arrangement is a wholly shielded enclosure to accommodate both the subject and the detector(s) used to assess the radioactivity of the whole body and/or specific organs or regions (Fig. 2). It is also the most flexible for setting up special equipment to investigate unusual cases of internal contamination, readily allowing, for example, the substitution of smaller detectors, or of more distant measurement geometries, in order to avoid saturation of the electronics by high count rates. A supply of filtered air with several changes hourly is recommended (Section 2.4). For a given thickness of the chosen shielding material, this design offers the greatest reduction in background, except that, where there is a major component from gaseous sources, the large volume of contaminated air within the shield may offset the benefit; it offers also the smallest dependence of background response on body size. The cost and weight of such shielding (perhaps 50 t or more) may be prohibitive. Occasionally, subjects may react adversely to isolation in a shielded room; such instances are much rarer where access to the counter is via an open shielded labyrinth rather than through massive hinged or sliding doors.

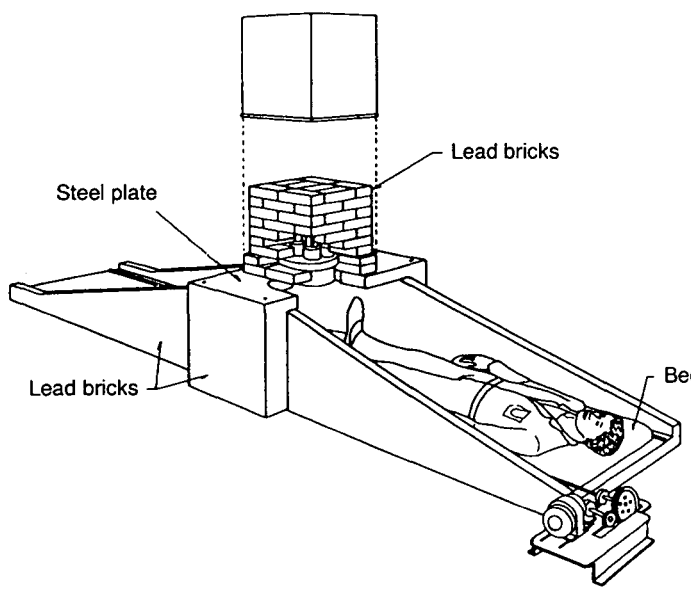


*FIG. 3. Shadow shield whole body counter.*

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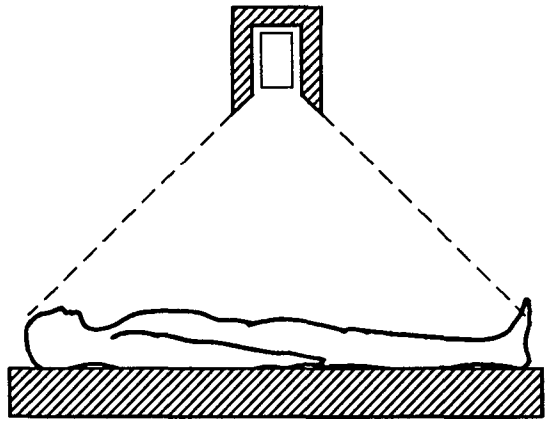

*FIG. 4. Shadow shield installation with single detector and subject in supine position.*

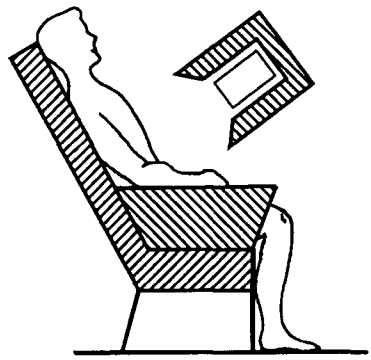

*FIG. 5. Shadow shield installation with single detector and chair technique.*

Some installations comprise more open structures which eliminate direct paths for radiation between the detector and the laboratory. Examples are the 'shadow shield' design used for assessing whole body radioactivity: Fig. 3 shows one design in which the subject lies on a bed moving under a fixed detector in a central turret, while Fig. 4 shows a simpler, static arrangement. Other arrangements embodying the same principle can be devised to assess the radioactive content of individual organs or regions. A small NaI(Tl) detector recessed within a lead cylinder and directed at the thyroid may be used to assess its content of radioiodine, with additional local shielding (lead bricks or steel plate) provided behind the neck. Figure 5 shows another simple arrangement for the investigation of radioactive deposits in

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larger regions. With all partially shielded counters, the background response below 200 keV is likely to be much larger than in a shielded room, because the counters respond to photons scattered by the subject into the detector. For this reason a shielded room is essential for the sensitive assessment of low energy photon emitters in vivo.

The internal surfaces of the shield, especially the floor, should be such as to permit easy cleansing in the event of inadvertent contamination.

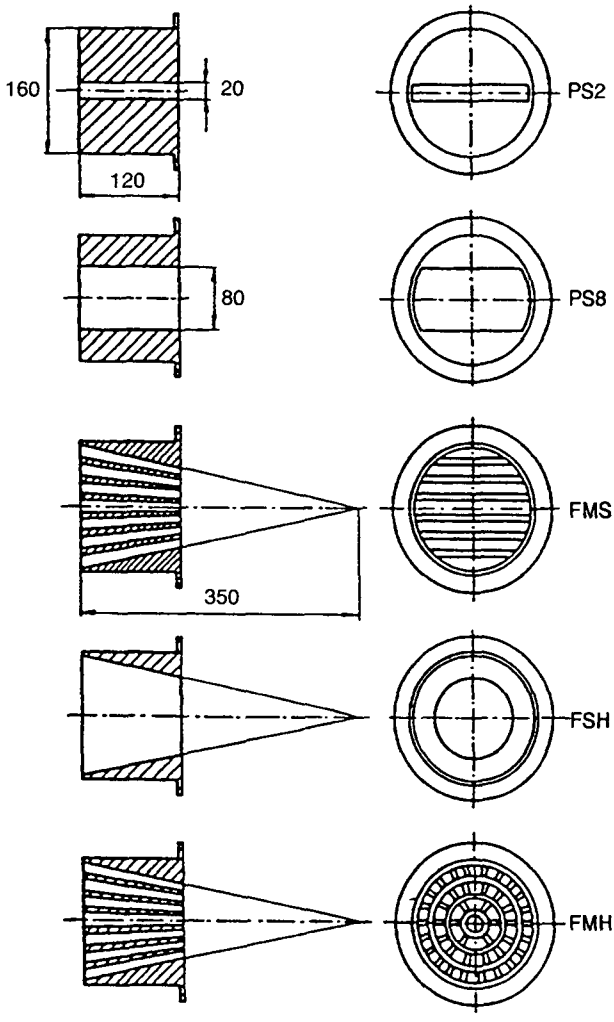


*FIG. 6. Collimators for profile and rectilinear scanning: PS2 — parallel slit (2 cm); PS8 — parallel slit (8 cm); FMS — focusing multislit; FSH — focusing single hole; FMH — focusing multihole.*

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### 2.3.5. Ancillary equipment

It will sometimes be necessary to establish the location of a contaminant in the body. Some such information may arise gratuitously during investigations with certain types of whole body counter (Section 3.2.4). Alternatively, or in addition, collimators, usually of lead, can be fitted to one or more detectors of an installation to allow mapping of the patterns of photon flux over the posterior and/or anterior surfaces of the body (Section 3.4). The most suitable devices for such exploratory studies will generally be slit or hole collimators (Fig. 6) restricting the detector's field of view to a roughly wedge shaped or conical volume. The design will depend on the size of the detector, on the energy of the radiation and on the spatial resolution which is desired, consistent with the photon flux available [15–17].

Access will be required to equipment for monitoring surface contamination on the subject.

## 2.4. ACCOMMODATION AND PLANT

The monitoring facilities should ideally be situated in a building remote from other laboratories or operations giving rise to emissions of radioactive materials or penetrating radiation which could interfere with measurements. If a new building is to be erected for the purpose, low activity materials should be used for its construction. The application of epoxy paints and/or certain plastics [18] to internal surfaces will impede the release of radon. The monitoring area, containing shielded detectors and associated electronic equipment, would normally occupy a ground floor or basement location in view of floor loading requirements and the advantages of additional shielding provided by any upper storeys. In addition to the main laboratory area, requirements are for office space, for changing and showering facilities for subjects prior to measurement, and for separate areas where radioactive solutions for use in calibration can be stored and dispensed. Depending on the anticipated level of demand for the service, a reception/waiting area may be useful. Shoe change facilities may be required for staff and others entering the monitoring area or its surroundings.

A constant temperature, preferably maintained to within 1 °C, and adequate control of humidity are required for the stable operation of photomultiplier tubes and other electronic equipment. A temperature of 21 °C will ensure the comfort of subjects wearing only the lightweight clothing typically worn during monitoring (Section 5.1.1). The arrangements for this temperature control are logically to be combined with those for filtering the incoming air.

Submicrometre filtration of incoming air is necessary to secure the greatest benefit from a heavily shielded room. This will promote stability of the background by removing radon decay products attached to airborne particles; it will be of further

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advantage if airborne radioactive particles may arise from the operation of neighbouring plants or laboratories. Filtration will not of course remove radon itself, but a flow sufficient to give 5–10 changes per hour will remove most of the daughter products formed within the shield before they themselves decay. The filtered supply should be derived from outside the building, with the outgoing air discharged through the surrounding laboratory to maintain a positive pressure within both areas and thus inhibit the ingress of airborne contaminants into either. Air from the laboratory should not be recirculated through the filter and shield unless its radon level is known to be acceptable.

Some organizations have commissioned mobile facilities, generally with shadow shields [19–21], which are useful where the requirement is for intermittent routine monitoring at dispersed locations.

## 2.5. PERSONNEL

The requirements for staff depend very much on the scope of the service which is contemplated. If only routine investigations are envisaged, as part of a regular monitoring regime which can be interrupted at times of staff shortage, it may suffice to employ one or more technicians, according to demand, supervised by a professional radiation spectrometrist whose functions would include certification of the results. If an emergency service is envisaged, for the investigation of occasional cases of potentially serious contamination, then additional professional staff, including medical practitioners, should be available to provide cover at all times. Ancillary staff, engaged in administration, reception and record keeping, will be required for large scale operations.

Difficulties may arise in the recruitment and retention of highly qualified personnel. Competence in the measurement of body radioactivity requires a working knowledge in areas of anatomy and biokinetics as well as of radiation physics and electronics. The expertise and skill required to direct the assessment of a complicated case are developed over many years. However, people with the required skills are unlikely to tolerate for long working with a large component of routine monitoring, in which these attributes are largely untested. A solution may be to combine the laboratory's functions in operational radiological protection with others in areas such as biomedical research [22].

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# 3. TECHNIQUES OF INVESTIGATION

## 3.1. GENERAL

Gamma ray spectrometry applied to radionuclides in the living subject involves considerations which are additional to those in its more common use in studies of inanimate analytical material. The sample is of large mass, and irregular and variable in its size, shape and composition. Moreover, the spatial distribution of the radionuclide is not accurately reproducible for calibration purposes. The sample is essentially inviolable: it cannot be processed to circumvent these and other complications, and the possibilities for manipulation to standardize the measurement geometry are limited. It cannot be subdivided for analysis, and changes can occur in the content and location of the deposited radionuclide; nor can it be stored in controlled conditions to prevent contamination between measurements. The living subject will not invariably adopt the passive role of inanimate samples during measurement. Consequently, truly replicated analyses may not be possible. The availability will generally be strictly limited, restricting the possibilities of compensating for a low sample activity by extending the measurement period.

The following discussion of techniques will concentrate on systems giving acceptable energy discrimination and capable under favourable circumstances of producing the accuracy envisaged (Section 2.2). It will exclude proprietary monitoring devices for screening employees after each work shift, although certain such systems may be capable of quantitative performance sufficient for a formal personnel monitoring programme. Of the various types of detector already discussed, only semiconductor detectors and NaI(Tl) scintillation counters, the latter combined in some cases with a CsI(Tl) scintillator as an anticoincidence guard, are commonly used, and only systems embodying these devices are considered.

## 3.2. MEASUREMENT OF RADIONUCLIDES IN THE WHOLE BODY

### 3.2.1. Purpose

The whole body content of a nuclide may in itself be of limited dosimetric significance. For satisfactory assessment of the hazard accruing in an individual case, the contents of specific organs will often be more relevant. Whole body counting is nevertheless useful both as a means of routine control and in the investigation of known contamination: in conjunction with biokinetic models and with assumptions about the timing of exposure, it can permit an estimate of the intake of a radionuclide, which is the quantity to be compared with a stipulated annual limit of intake in deriving an index of compliance.

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### 3.2.2. Specific objectives

In whole body counting, the requirements (Section 2.2) for adequate sensitivity and acceptable accuracy may conflict. For detector(s) of given type and size with given shielding arrangements, the first of these needs may require location of the detector(s) close to the subject, making the detection efficiency oversensitive to the pattern of distribution of the radionuclide in the body and to the subject's physique, and creating the likelihood of a systematic error. This error will arise because, in the derivation of a calibration factor for the nuclide concerned, its presumed disposition relative to the detector(s) can only be simulated roughly. The errors are at their most serious for low energy photon emitters ($< 100$ keV), and investigations of such nuclides tend to focus on the emissions from localized regions in which the pattern of deposition can be more readily simulated (Section 3.3). The procedure, in designing a whole body counter, should be: (i) to select the type of detector; (ii) to decide on the deployment of the detector or detectors in relation to the subject, giving a response which is adequately independent of the distribution of activity within the body and of the subject's physique; and (iii) to achieve the required sensitivity through the selection of appropriate shielding and size of detector. It is not feasible to issue firm guidance in any of these matters to cater for all applications and situations; perusal of the IAEA's compilation of data on such installations [23] is recommended. Decisions on the optimum siting of detectors may be aided through the use or adaptation of methods for calculating the response from localized or distributed sources in attenuating media [24–27]. In practice, the design will often be constrained by the availability of pre-existing resources, and ingenuity may produce satisfactory adaptations of equipment installed for other purposes, e.g. gamma cameras [28].

### 3.2.3. Single detector geometries

Whole body counters with only a single, stationary detector offer certain advantages in simplifying the requirements for signal processing and supporting mechanisms, but they are the most prone to systematic error if the subject–detector configuration has to be optimized for sensitivity.

*Arc geometry*

The arc arrangement (Fig. 7) is capable of high accuracy if the levels of internal radioactivity and the space available inside a shielded room are sufficient. The subject lies on a curved frame forming the arc of a circle centred at the detector, so that all parts of the body are roughly equidistant from it. Uniformity of response will be optimized if: (i) the radius of curvature is large (1.5–2 m); (ii) in the common case of a cylindrical detector, it is mounted so that radiation from the subject

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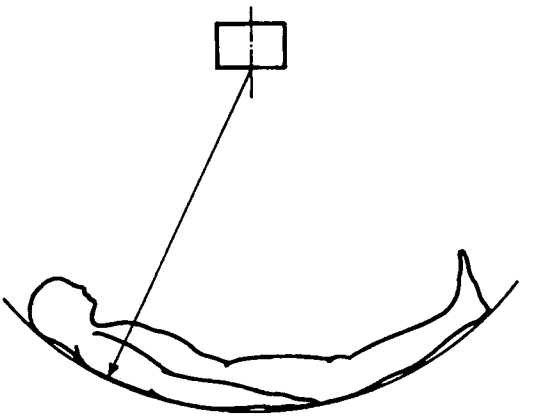

*FIG. 7. Arc geometry.*

impinges predominantly on the curved surfaces (rather than on the exposed plane surface as in Fig. 7); and (iii) two measurements are made, one with the subject lying as in Fig. 7 and a second in the reverse posture, so that the posterior surfaces are proximal to the detector. With these conditions satisfied, an estimate based on a statistically precise mean response in the two positions can be made accurate to within 10% for photon energies >200 keV, and greater accuracy is possible if rigorous calibration procedures are followed (Section 4.4.2). The detection efficiency is of course poor and even with a large detector (e.g. 230 mm diameter × 150 mm NaI(Tl)) in a heavily shielded room, the technique will not generally be applicable to the determination of activities below several kilobecquerels; it would seldom be feasible with semiconductor detectors. In the context of radiological protection, therefore, the arc method will be used primarily in the investigation of established cases of internal contamination rather than as a regular means of control.

*Chair geometry*

In the chair geometry arrangement (Fig. 8) the subject reclines in a tilted chair with the detector supported typically 0.4 m above the abdomen [29]. The detector will commonly be an NaI(Tl) scintillator, e.g. 200 mm diameter × 100 mm, and in a heavily shielded room would be capable of detecting as little as 50 Bq of most common fission and activation products in a counting time of 15 min. Similar arrangements have been employed with semiconductor detectors (Annexes II and III). The response will, however, depend markedly on the location of the radioactive deposit: depending on the separation of the detector from the body, the efficiency for detecting activity with a high local concentration, e.g. in the lungs or

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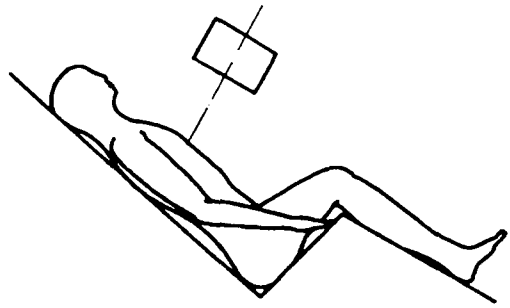

*FIG. 8. Chair geometry.*

liver, may differ by a factor of two or more from that applying to material widely dispersed in the body. The potential for systematic error is accordingly much greater than with the arc technique, but this will be unimportant in many routine applications. A modified design (Ref. [30, 31], see also Annex III, Fig. III-1) is reported to display improved uniformity of response.

*Scanning with a single detector*

With the single detector design of counter, the response is accumulated from a single detector which traverses the subject's length at a fixed distance above or below the supine body or, in some versions, in a corresponding disposition relative to the erect body. Alternatively, the supine subject may be moved in relation to a fixed detector, as in Fig. 3. Accuracy will be improved if a second traverse is performed with the subject's posture reversed and the evaluation is based on the combined response in the two positions. In that situation the system shares many of the attributes of the more convenient two detector scanning arrangements (Section 3.2.4).

### 3.2.4. Multidetector geometries

*Stationary arrays*

An alternative approach, preserving the good detection efficiency given by the chair technique but offering improved uniformity of response, is to adopt a 'stretcher' geometry, with the subject in a supine posture and several detectors distributed about the body (Fig. 9). Arrangements with up to 54 detectors [32] have been constructed but much more commonly 4–8 are employed, disposed above and below the stretcher so that their combined response is acceptably independent of the

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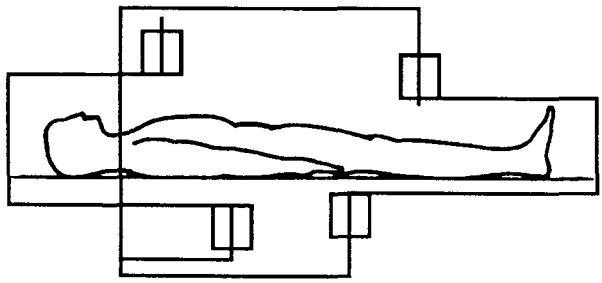

*FIG. 9. Stretcher geometry.*

source distribution. Complete uniformity of efficiency is of course unattainable: the combined response from a bolus of contamination passing through the gut, for example, may vary easily by a factor of two according to its current location in relation to a detector close to the abdomen. However, for activity which is not concentrated in small organs or regions, such an array can, with straightforward calibration methods (Section 4.4.2), yield results for energetic ($> 100$ keV) photon emitters that are accurate to within 20% or better. For the same total mass of detector within similar shielding, the efficiency and sensitivity should be comparable with those for a chair geometry (Section 3.2.3).

Disadvantages are the need to provide several independently adjustable supporting mechanisms for the various detectors, and the requirement of a larger shielded room than would generally be necessary to accommodate only a chair. The processing of multiple output signals requires additional amplifiers, etc., and the task of frequently ensuring that each spectrometer is matched for gain may be found tedious. If the individual spectra, rather than their sums, are to be stored, multiplexing arrangements will be necessary within the analog–digital converter. Recording the discrete data in this way may permit broad inferences about the distribution of a contaminant or tracer in the body.

*Scanning devices*

In scanning device arrangements, most commonly employing two detectors (one above and one below the supine subject), a more representative sample of the photon flux is obtained by moving the supine subject relative to the counters during accumulation of the spectral data. Figure 10 depicts detectors moving along a static subject (the obvious arrangement where space is restricted) but the roles are sometimes reversed. More elaborate arrangements have been devised (Annex V), including one [33] in which a rotating annular array of detectors scanned the supine subject, thus approximating to uniform cylindrical geometry.

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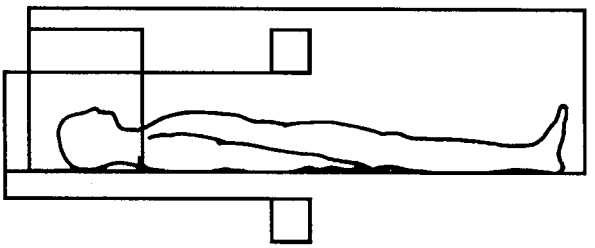

*FIG. 10. Scanning bed geometry.*

However, while scanning arrangements can provide a more uniform detection geometry compared with that given by most other configurations, they may not yield significantly greater accuracy than a well designed array of stationary detectors (Section 3.2.4). This is because the emergent photon flux generally depends as much on attenuation within the body as on purely spatial considerations [25]. Disadvantages include the complex mechanical arrangements and the need to ensure that the carriage or stretcher moves at a constant rate in detector live time, rather than in clock time. Useful indications of the distribution of a radionuclide within the body may be obtained if the system can display a profile of the response according to position. If the levels of contamination are sufficient, additional traverses with the detectors collimated (Section 2.3.5) can provide clearer indications of distribution, provided that the structure and/or drive mechanism can cope with the additional load.

## 3.3. MEASUREMENT OF RADIONUCLIDES IN INDIVIDUAL ORGANS OR REGIONS

### 3.3.1. Purpose and feasibility

The need to assess the radioactive content of a specific organ or region, rather than of the body as a whole, may arise in two situations. It may be that the radionuclide is known, or may reasonably be assumed, to be concentrated wholly or largely in the organ or region, but that whole body monitoring would be insufficiently sensitive. In that case improved sensitivity can be secured through the use of static detectors located close to the site of interest. Alternatively, the deposit may be present in two or more organs, and techniques are required to infer the partition between each. In some instances both situations may exist, for example in the assessment of lung deposits of several tens of becquerels of $^{241}$Am, accompanied by comparable hepatic and skeletal deposits.

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The feasibility and practice of a range of approaches to the assessment of organ deposits, at various activity levels from kilobecquerels to megabecquerels, were reviewed by an IAEA Panel [34].

### 3.3.2. Mechanical arrangements

In whole body counting, the detector–subject geometry is often fixed and flexibility in the counting geometry, while it may be desirable to accommodate specific requirements, is not a primary need. By contrast, when the radioactive content of a particular anatomical region is of interest, the efficiency of detection is often maximized through location of the detectors in close proximity to the organ or region. This will frequently require the detectors to be mounted in supports offering several degrees of freedom (including rotational as well as linear, lateral and vertical displacements) to achieve this objective for every subject, irrespective of anatomical configuration. Scales provided for each such movement may assist in securing reproducibility of counting geometry in serial investigations. If the use of collimators is contemplated in assessing the distribution of the nuclide (Section 3.4) or to minimize interference from deposits in adjacent regions, their masses will need to be taken into account in the design of the mountings.

### 3.3.3. Monitoring procedures

*Monitoring of radionuclides in wounds*

An extreme but fairly common example of a localized deposition is the presence of poorly soluble radioactive material at the site of a puncture wound, investigated shortly after an accident, before appreciable quantities have become systemic. With fission or activation products giving abundant and energetic photon emissions and with high limits of intake, improvised arrangements employing any spectrometrically suitable scintillation or semiconductor detector are likely to be satisfactory; the requirements will for the most part coincide with those common in the measurement of small, prepared samples in radiochemical analysis.

There are more stringent requirements for sensitivity and accuracy when alpha emitters are involved, notably actinides emitting photons of low energy ($< 100$ keV) and often of low abundance. A semiconductor detector will be more suited than a scintillation counter of similar efficiency. For the highest sensitivity, the detector should be located as close as possible to the site of the injury. It should be of small diameter (10–20 mm) and no thicker than required for efficient interception of the radiation concerned. However, with good shielding (as provided for example by conducting the investigation within the shielded enclosure of a sensitive whole body counter), the larger detectors installed for other purposes may suffice, for example

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one of an array of high purity germanium detectors (typical size 50 mm diameter × 20 mm) provided for assessment of actinides in lungs should permit detection of $^{241}$Am or of high burnup plutonium in becquerel quantities. Portable semiconductor detectors designed for this purpose that can be operated in a medical unit or elsewhere are commercially available. In these, use of a smaller crystal partially compensates for lack of substantial shielding.

In cases where part of the contamination has migrated from the wound and the residue is to be assessed, the surrounding areas may need to be screened from the detector by covering them with suitable shielding, graded if necessary. A similar approach may be required in localizing contamination at the site of injury. In cases needing sensitive measurement, and particularly where scintillation counters rather than semiconductor detectors are used, an appropriate background spectrum may be required, allowing for the effects of other sources of body radioactivity. This may be achieved by recording the response with the detector viewing an uncontaminated region of appropriate mass.

*Measurement of radionuclides in the thyroid*

Isotopes of iodine, and also $^{99}$Tc$^{m}$, may concentrate in the thyroid gland. The range of photon energies encountered is from 27 keV ($^{125}$I) to several hundred kiloelectronvolts. Some high purity germanium detectors used for the assessment of actinides in lungs are of suitable diameter (approximately 50 mm) in relation to the size of the thyroid, and are large enough to provide adequate detection efficiency over most or all of this energy range. Alternatively, a planar germanium detector or thin NaI(Tl) crystal may be adopted for photon energies of up to 100 keV. Thicker crystals, either NaI(Tl) or coaxial germanium detectors, may be used if necessary to secure efficient photon detection at high energies (Annex VI).

If the requirement is simply to demonstrate that the thyroid's content is well below some stipulated recording level, it may suffice to locate the detector close to the organ, with minimal shielding against background radiation or against spectral contributions from the activity of neighbouring tissues, e.g. blood. With such an approach, evaluation of the result would need to take account of uncertainties in calibration. More accurate assessment of an easily detectable deposit would require better shielding, with the detector recessed in a suitable collimator to reduce interference from extra-thyroidal deposits and sufficiently distant from the organ to permit reliable calibration. The IAEA [35] has provided advice, including suggestions on collimator design, for the assessment of $^{131}$I in medical diagnosis.

*Measurement of radionuclides in the lungs*

Inhalation is the most common route for intake in occupationally exposed personnel, with the respiratory tract the site of initial deposition. If the deposit persists

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for a sufficient time, monitoring of pulmonary activity may offer the most sensitive and reliable means of assessing the intake. Indeed, in the case of certain actinides which subsequently relocate to other organs which absorb virtually all low energy photon emissions, it offers the only remotely practicable means of assessing an intake by external counting.

Measurement of the lung deposit of an energetic photon emitter may be achieved by whole body counting (Section 3.2) if it is known that most or all of the contamination is pulmonary. If mechanical scanning is employed (Section 3.2.4) and the 'profile' of response according to position is recorded, the pulmonary component may be resolved from that in other organs unless, as with the liver, they are in too close proximity. The separation will be more effective if the detectors are fitted with slit collimators (Section 3.4.2). A large diameter (150–300 mm) stationary NaI(Tl) detector recessed in a cylindrical collimator may also be used. The response would preferably be recorded in two locations, the detector viewing in turn the anterior and posterior surfaces of the thorax. Ideally, if two such detectors were available, the measurements could proceed simultaneously. Arrays of collimated detectors have been designed specifically for studies of aerosol deposition in the respiratory tract [36, 37].

The sensitive measurement of pulmonary deposits of low energy photon emitters ($< 100$ keV) is most commonly required for $^{241}$Am and for isotopes of uranium and plutonium. It requires equipment giving a better signal to background ratio than is provided by NaI(Tl) detectors. This is achieved either through partial suppression of background response (as in the phoswich detector) or through the improved energy resolution offered by semiconductor detectors (Section 2.3.1). Phoswich detectors of large area are available and, when used in pairs (Annexes VII and VIII), can provide adequate coverage of the anterior surfaces of the chest. The most common arrangement is of paired 120 mm diameter units viewing the anterior aspects of the upper thorax, one over each lung. Corrections (Section 4.3) are required for the effects of scattered radiation from the subject's natural $^{40}$K and any other interfering nuclide. With poor energy resolution, a result, even if statistically significant, may not be unambiguously attributable to a specific nuclide. The good resolution of high purity germanium detectors simplifies these corrections and often permits a clear identification of the contaminant. However, they are much more expensive than phoswiches and are available only in sizes of up to about 80 mm diameter. They are therefore used in clusters, typically of three such detectors viewing each lung (Annexes VII and VIII). The sensitivity is comparable with that given by phoswich detectors in typical situations, and somewhat better when there is major interference from other nuclides. Uncertainties may arise with either type in corrections for the contributions from systemic activity (Section 4.4.3).

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*Monitoring of other organs*

The two other sites of deposition which frequently require specific attention are bone and the liver. Arrays of detectors viewing the skull have been employed to assess skeletal deposits of $^{210}$Pb (47 keV photons) and $^{241}$Am (60 keV) [38, 39]. With such an approach, assumptions have to be made concerning the proportion of the total skeletal mass viewed by the detectors and the uniformity of concentration in the skeleton. The knee is an alternative region in which judicious location and/or collimation of detectors can lead to satisfactory discrimination for registering skeletal activity, in the presence of deposits in disseminated soft tissues or in other organs. The liver is most commonly monitored for accumulations of $^{241}$Am translocated from the lungs, and the same detectors as installed for investigations of pulmonary deposits would generally be used. In the siting of such detectors for maximum response, the lateral asymmetry of the organ would be taken into account.

The levels of contamination in liver and skeleton may be of interest with regard to their interfering contributions when lung deposits are assessed (Section 4.4.3), as well as in their own right.

## 3.4. INVESTIGATIONS OF DISTRIBUTION

### 3.4.1. Purpose

Indications of the pattern of distribution of a contaminant may be helpful in several contexts. They are relevant in determining the appropriate strategy in a dosimetry programme for the siting of detectors viewing specific organs; they may be required to establish whether the metabolism of the nuclide conforms to some particular model adopted in translating an estimate of whole body content into one of effective dose equivalent; and, where the response of a whole body counter depends markedly on the distribution of the contaminant, the information may enable the actual distribution to be roughly simulated in the process of calibration (Section 4.4).

### 3.4.2. Methods

With appropriate systems to record separately the signals from members of a multidetector array or the response as a function of position of detectors scanning the body (Section 3.2.4), some information on the distribution of a nuclide may be available adventitiously during the course of a measurement of whole body radioactivity. This may point to the source as being widely dispersed (as in the case of nuclides present in bone or in the disseminated soft tissues) or concentrated in some general anatomical region, such as the upper or lower thorax (suggesting possible deposition of radioactive material in the lungs or radionuclides passing through the

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gastrointestinal tract). It may also point to the presence of surface contamination which has escaped removal through the recommended procedures (Section 5.1.1). Additional inferences are sometimes possible without the use of special equipment, for example the liver may be identified as a site of major deposition if a detector in contact with the right lateral aspect of the mid-thorax shows a much greater response than in a corresponding location on the left side.

However, the most useful information, provided that the levels of contamination make the approach both technically feasible and administratively justified, will be obtained through the use of collimators fitted to detectors scanning the body either on motor driven carriages or manually on a point by point basis. Although the optimum design must depend on numerous factors, including the levels of contamination, the gamma ray energy and the distribution itself, for general use a slit collimator constructed of lead 30–50 mm thick, with the long dimension of the aperture orthogonal to the direction of movement, is recommended (Fig. 6). If the width of the aperture can be adjusted through the provision of a series of inserts, the conflicting requirements of adequate spatial resolution and acceptable counting statistics are more readily balanced. A second factor in this balance is the energy range over which the spectrum is integrated: the inclusion of too much scattered radiation will impair this resolution, since its origin will often lie outside the defined field of view. It should be noted that profile scanning with simple slit collimated detectors will often give reliable indications of the relevant sites of deposition, but that only a rough evaluation of that deposition can be obtained if a single detector viewing only one aspect of the body is employed. Quantitative assessment of radionuclide distribution by profile scanning requires either paired detectors or, if only one is available, separate traverses of the anterior and posterior surfaces, and further improvements may result if focused collimators [15, 16] are used and computer aided evaluation techniques [17] are applied.

# 4. ANALYSIS OF DATA FROM MEASUREMENTS

## 4.1. OUTLINE OF PROCEDURES

Analysis of a photon energy spectrum of body radioactivity will initially require the identification of radionuclides responsible for its individual features. The next stage will generally involve resolution into the constituent components. In a further process, the response attributable to a particular contributor will be translated into an estimate of body or organ content. This is accomplished by reference to a spectrum representing a known quantity of the radionuclide (or sometimes of another

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nuclide emitting photons sufficiently similar in energy) in some appropriate geometry and absorbing medium. In one common approach (linear regression analysis (see Section 4.3)) the second and third stages are in effect combined.

## 4.2. IDENTIFICATION OF NUCLIDES

When a contaminant is present in sufficient quantity to produce well defined spectral features, its identity can often be established or confirmed from the corresponding energies of these features and, in some cases, from the relative responses in multiple peaks. This is most easily accomplished with semiconductor detectors giving peaks which are well differentiated both from underlying scatter components and from neighbouring features. The peak search and identification routines incorporated into proprietary software may be helpful in this context, but as a supplement to (rather than as a substitute for) visual examination. Ambiguities may occur in the analysis of poorly defined spectra from low level contaminants investigated with scintillation counters. As in other applications of radiation spectrometry, reliable identification naturally depends on accurate energy calibration, by reference to the locations of features in spectra from known sources.

## 4.3. ANALYSIS OF SPECTRA

Methods of unfolding photon energy spectra of body radioactivity do not differ in principle from those applied in X ray and gamma ray spectrometry generally, except that account must often be taken of the effect on spectral shape of scatter in a large attenuating mass. The process is at its simplest in the estimation of peak areas from semiconductor detectors. The good energy resolution of such instruments allows the effective background response underlying a spectral peak (both that due to the ambient photon flux and that arising from scattered radiation from other sources of body radioactivity) to be reliably deduced from the adjacent continuum. The procedure is straightforward where peaks are well defined. Additional useful techniques are available in other situations [40].

For scintillation counters, the poor resolution of spectral peaks often makes this approach inapplicable except for the purposes of giving a rough indication. It will generally be necessary first to subtract an appropriate spectrum of counter background (Section 5.1.2). A more rigorous analytical procedure will then generally be required, unless a spectrum has been obtained from the subject prior to the intake of interest, when it may be valid to use this as a 'background' spectrum.

When unfolding or deconvolution of spectra from scintillation counters is required, a 'stripping' process is sometimes used. Reference spectra (Section 4.4) are derived for each nuclide present, each representing the response from known

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amounts of the nuclide in an appropriate geometry and in relevant absorbing media. The reference spectrum containing the peak with the highest energy (frequently that of $^{40}K$) is selected, and it is normalized to the subject's spectrum on the basis of count rate in an energy region where only that nuclide contributes. Subtraction of the normalized spectrum gives a residue representing the remaining components, which is treated in the same way. The content of each nuclide is calculated directly from the fraction of its reference spectrum which must be subtracted. In principle, the process can be repeated until the residue consists of the response from a single nuclide only. In practice, unacceptable errors are likely to accumulate if the number of stages exceeds two or three, particularly in relation to minor components in a spectrum. Moreover, the method will generally be inapplicable when the dominant peaks of different components overlap.

A variant of this procedure is sometimes employed when scintillation counters are used in assessing lung deposits of actinides, through detection of L X rays or 60 keV gamma rays from $^{241}Am$, and a correction is required for the component of scattered quanta from $^{40}K$ (and any other energetic photon emitters) in the spectral region of interest. This is derived by reference to a correlation established in unexposed subjects, linking the relative responses in that region and in some other range of higher energies, typically 80–100 keV, with some parameter reflecting physique, such as the weight/height ratio [41].

A more satisfactory procedure in many situations is to adopt a method of linear regression analysis, to derive the proportions of each reference spectrum which, when combined, give rise to the best fit to the subject's spectrum. Facilities for such analyses are embodied in several commercially available computer programs for processing gamma ray spectra. Alternatively, they can be developed locally [42, 43]. By utilizing a much larger portion of the spectrum, instead of the restricted regions successively considered in the stripping process, this method gives improved statistical accuracy in the estimates of the various components; moreover, realistic estimates of this accuracy may be derived in the matrix inversion procedures. A further attribute is that, if the spectrum contains a component which has been neglected in the analysis, the omission will generally be apparent on comparing the 'input' spectrum of the subject's body radioactivity with the 'fitted' spectrum, i.e. the sum of the proportions of the various reference spectra giving rise to the best fit.

As with other methods of deconvolution, this approach has its limitations. In particular, it demands stability of the spectrometer during the measurement, especially if the nuclides present possess overlapping spectral features. It is also important that the locations of peaks in the subject's spectrum should coincide with those in the relevant reference spectra. Where minor drifts occur between the measurements, adjustments can often be made prior to the analysis if appropriate routines are available [42, 43]. However, the validity of the analysis depends also on the accordance of the spectral shapes of the reference standards with those of the corresponding components in the subject's spectrum. This is sometimes difficult to

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achieve at energies below 400 keV, a region where the intensity and energy distribution of scattered photons can depend strongly on the size and shape of the body and on the location of the radionuclide within it. For this reason it may be expedient to restrict the analysis to data in the range 400—1600 keV or higher, supplying reference spectra only for those components contributing within the selected region; this ensures that any mismatch at lower energies will not affect the assessment of those components. Cruder methods may then be applied to the 'residual' spectrum (i.e. the difference between the 'input' and 'fitted' spectra) below 400 keV in evaluating any peaks present in that region. These may introduce systematic errors, the importance of which will depend on the size of the peaks concerned in relation to the underlying continuum.

## 4.4. CALIBRATION AND REFERENCE SPECTRA

The approach to calibration must depend on several factors, including

- the type of detector employed and the subject–detector configuration;
- the decay properties and distribution of the nuclide;
- the need to separate the spectrum into individual components.

It may not be necessary to acquire calibration data specifically for all nuclides anticipated or encountered. It may suffice to interpolate from data obtained with a series of nuclides together possessing an appropriate range of photon energies. As a general rule, spectra for calibration purposes should be of good statistical precision (say $<3\%$) to avoid contributing to systematic error. This may be particularly important if they are to serve as reference spectra in analysis by linear regression: statistical fluctuations in neighbouring channels of a calibration spectrum may lead to overestimation of some components in the subject's spectrum at the expense of others. However, the count rates should not be so high as to produce pulse pile-up in the electronics and so distort the spectrum.

### 4.4.1. Precautions in the use of radioactive solutions

When active solutions are used in calibration procedures, precautions are required to prevent the contamination of equipment through leakage. It is important also to keep the solution sufficiently acid, and to provide sufficient inactive carrier, in order to discourage precipitation and/or plating of the radionuclide on the internal walls of the phantom. As well as possibly invalidating the calibration, this could present difficulties if the phantom were required for re-use with another nuclide.

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### 4.4.2. Procedures applicable in whole body counting

*Tissue substitutes*

It was indicated above (Section 3.2.2) that whole body counting as such was generally concerned with the investigation of energetic photon emitters (> 100 keV). For such nuclides, many materials are available for use as tissue substitutes in calibration procedures, simulating adequately the attenuation properties of human tissues [44].

*Calibration with a point source*

Calibration with a point source may be adopted for whole body counters whose 'geometrical' counting efficiency shows little dependence on the location of the source, namely those disposed in distant arc geometry (Section 3.2.3) and in certain scanning arrangements (Section 3.2.4). With the arc geometry, calibration accurate to within a few per cent can be achieved with a standardized point source suspended in a tank of water or located between stacked plates of a solid absorber with comparable attenuation properties [45]. The appropriate location of the source in the tank or stack is deduced from examination of the spectral shape and evaluation of the relative photon fluxes emerging from the subject's anterior and posterior surfaces. An alternative approach is to record the response from a point source without surrounding absorber and to modify it according to the measured transmission of its photons through the subject [46]. Procedures for point source calibration of scanning whole body counters are well documented [25]. The greatest accuracy is obtained when evaluation is based on the response in a wide energy band (i.e. including scattered radiation as well as that producing a specific peak), but this may not be feasible if the spectrum is too complex. Moreover, reference spectra derived from measurements of point sources in a single location will seldom match those of the same radionuclides in the body, to the extent required in separating a complex spectrum into its constituents (Section 4.3). The use of a point source to calibrate either method of measurement will therefore often be restricted to assessment of intakes of a single nuclide, in situations where the pre-existing body radioactivity is either known from prior investigation or can be neglected.

*Calibration with anthropomorphic phantoms*

In whole body counters comprising a limited number of stationary detectors close to the subject, or using a detector scanning one side of the body only, calibration with a single standardized point source is unreliable, unless (i) the deposit is highly localized at a known site or (ii) a dispersed distribution is simulated by averaging the response in repetitive measurements with the source occupying numerous

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## *Description*

Assemblies of ten cylindrical polyethylene bottles that can be filled with water and other liquid based tissue substitutes. The complete phantoms represent torso, pelvis, head and neck, arms and legs for adults and children.

## *Physical dimensions*

### Original phantom

Total assembly represents a human, 180 cm tall (*a* 30 cm; *b* 10 cm; *c* 40 cm; *d* 20 cm; *e* 40 cm; *f* 40 cm)
Cylinders of circular and elliptical cross-section

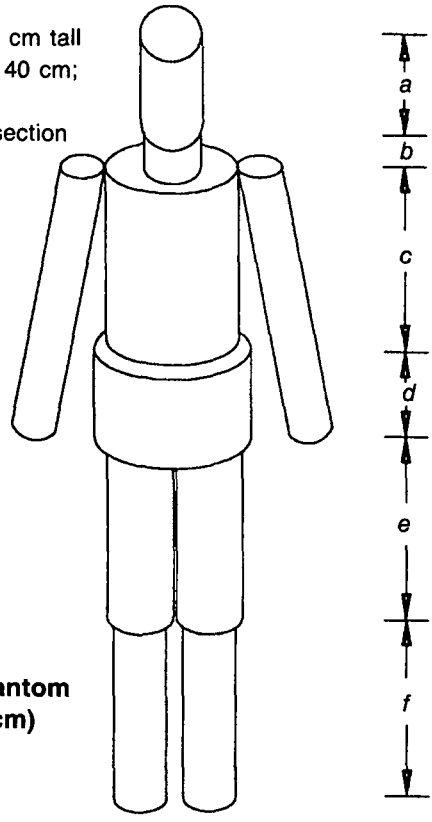
### Reference adults and children

| | Total height (cm) |
|---|---|
| Adult male | 169.5 |
| Adult female | 160.2 |
| 4 year old child | 105.3 |
| 10 year old child | 140.0 |

## *Materials*

**Containers** Polyethylene

**Filling** Water or appropriate water based solution




**Original BOMAB Phantom (Adult, height: 180 cm)**

## *Applications*

(1) Calibration of whole body counters for uniformly distributed gamma emitting nuclides, including fission and activation products.
(2) Calibration of systems used to assess blood sodium activation following a nuclear criticality accident. The phantom is filled with solution of a sodium salt and exposed to neutrons from a reactor or similar source. Aliquots of the solution can then be used to assay the activated $^{24}Na$.

*FIG. 11. BOMAB (bottle manikin absorber) whole body calibration phantom.*

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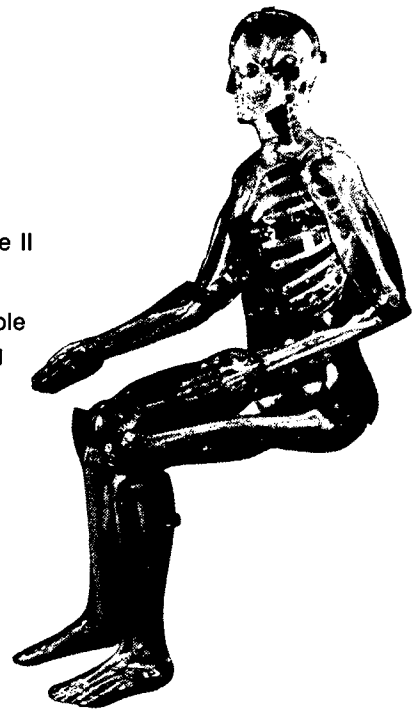
## *Description*

Anthropomorphic plastic shell containing a complete skeleton. Phantom is filled with water or other liquid based tissue substitute.

## *Physical dimensions*

| | |
|---|---|
| Height | 175 cm |
| Mass | 73.5 kg |
| Shell thickness | 2 to 3 mm |

## *Photograph*



## *Materials*

| | |
|---|---|
| Shell | Vacuum formed Tenite II with solvent bonding |
| Skeleton | Human bones, available with neoprene coating |
| Filling | Water or appropriate water based solution |

## *Applications*

(1) Phantom for absorbed dose distribution studies. Dosimeters, such as thermoluminescent dosimeters, are placed at selected points in an internal network of plastic tubes.
(2) Calibration of systems used to assess blood activation following a nuclear criticality accident. The phantom is filled with solution of a sodium salt and exposed to neutrons from a reactor or similar source. Aliquots of the solution can then be used to assay the activated sodium-24 activity.
(3) Calibration of whole body counters for uniformly distributed gamma emitting nuclides, including fission and activation products.

*FIG. 12. The REMAB (radiation equivalent manikin absorption) whole body dosimetric/calibration phantom.*

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## Description

Anthropomorphic plastic shell with cavities for major organ lungs, thyroid, heart, kidneys, spleen, pancreas, stomach, bladder and simplified lower intestinal tract. Phantom and organs (other than lungs) are filled with water or other liquid based tissue substitutes and radioactive solutions.

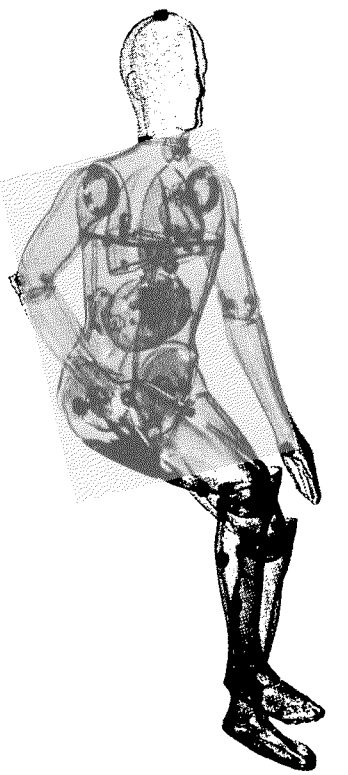
## Physical dimensions

| | |
|---|---|
| Height | 175 cm |
| Mass | 73.5 kg |
| Shell thickness | 2 to 3 mm |

## Materials

| | |
|---|---|
| Shell | Vacuum formed Tenite II with solvent bonding |
| Lungs | Rando lung substitute |
| Other organs | Vacuum formed Tenite II with solvent bonding |
| Filling | Water or appropriate water based solution |

## Photograph



## Applications

Calibration of whole body counters, or nuclear medicine systems. Selected organs are filled with solutions of the radionuclide of interest, while the remainder of the phantom, including other organs, can be filled with water or other non-radioactive solution to simulate the surrounding tissue for counting purposes.

*FIG. 13. The REMCAL (radiation equivalent manikin calibration) whole body calibration phantom.*

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## Description

An anthropomorphic phantom representing the torso of an adult North American male without the head or arms, and terminated just above the pelvis. The phantom contains a simulated rib cage and removable lungs, liver, heart and major tracheobronchial lymph nodes. Chest plate overlays provide representation of individuals with different chest wall thicknesses. An extended "Fission Product" version includes a complete torso, head and neck (with thyroid), hollow arms and legs. Radioactivity can be added in the form of solutions to hollow organs, as powders dispersed throughout moulded materials, or as discrete capsules fitted into holes located in selected organs.

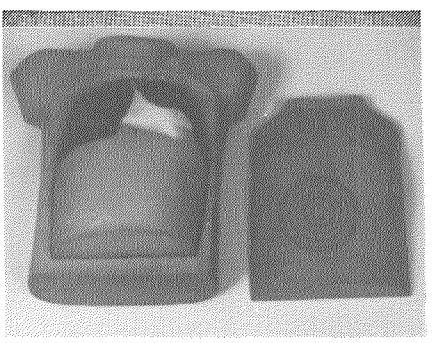
## Physical dimensions

Torso of adult North American male of height: 175 cm; mass: 76 kg; chest circumference: 100 cm.

## Materials

| | |
|---|---|
| Torso | Polyurethane based muscle substitute |
| Skeleton | Polyurethane based bone and cartilage tissue substitutes |
| Lungs | Foamed polyurethane lung substitute |
| Chest overlays | Representative of muscle (100); muscle (50), adipose (50); muscle (13), adipose (87) (% by mass) |
| Arms, legs | Cellulose acetate butyrate shells. Choice of liquid filling left to user but water is most commonly used. |

## Photograph



## Applications

Calibration of radiation detectors used for in vivo measurement of $^{239}Pu$, $^{241}Am$ and other transuranic radionuclides deposited in organs of the upper torso.

*FIG. 14. Livermore torso calibration phantoms for transuranic radionuclides.*

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positions in a suitable attenuating matrix. In practice, systemic radionuclides tend not to be concentrated in a single anatomical region, and such counters are generally calibrated, at least in the first instance, with the aid of whole body phantoms simulating the human form and containing a standardized aqueous solution of the relevant nuclide. The most convenient general purpose phantom consists of a collection of polyethylene vessels of circular or elliptical cross-section (Fig. 11). These are available commercially, or can be made by a workshop with experience of plastics. Appropriate proportions for each section have been given by Bush [47]. It is useful to have available two or three such phantoms with dimensions suitably scaled to represent individuals of different sizes; for intermediate physiques calibration factors may be derived by interpolation according to functions of anatomical parameters, e.g. weight/height ratio. Alternatively makeshift arrangements, for example using plastic reagent bottles, may suffice, and in some situations they may be more versatile in simulating specific physiques and postures. Conversely, much more elaborate whole body phantoms (Figs 12–14) can be purchased, some of them provided with discrete organs which can be labelled independently of a dispersed deposit. Such complicated devices would ordinarily be required only in the calibration of equipment for assessment of deposits in specific organs (Section 4.4.3), or of 'whole body' counters whose response was oversensitive to the distribution.

*Calibration with volunteers and patients*

On several occasions, known amounts of radionuclides have been administered to volunteers expressly in order to calibrate whole body counters, for example $^{132}Cs$ [48], $^{137}Cs$ [49] and $^{42}K$ [50]. This would now only be possible under strict supervision with ethics committee approval. It may also sometimes be possible to make use of subjects who have received intakes of radionuclides in medical diagnosis or other metabolic studies. If the measurement is delayed after intake of the tracer, for example to allow its distribution to stabilize, excreta voided in the interim may need to be collected and assessed for radioactive content. Alternatively, the content of such a patient, or of a worker accidentally contaminated at a sufficient level, may be determined reliably by counting (for example) in distant arc geometry (Section 3.2.3). Intercalibration measurements with other equipment may then be performed.

*Calibration by calculation*

It is possible to estimate by calculation the detection efficiency for a nuclide deposited in any assumed pattern relative to the detector(s), taking into account geometrical and attenuation factors [24–26]. It is advisable to validate efficiencies derived in this way through comparisons with values measured in a phantom for at

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least two well separated gamma ray energies. The approach is perhaps most useful in deriving adjustments to calibration data observed under specific conditions (photon energy, size of phantom, etc.) to allow for small variations in those conditions.

### 4.4.3. Procedures applicable to investigation of specific sites or individual organs

*Wounds*

Calibration with a suitably located point source will generally be appropriate in the evaluation of local deposits acquired via puncture wounds, unless there has been migration of the deposit beyond the immediate vicinity. Most commonly, the contamination will lie within a few millimetres of the surface, and there will be no need to simulate the small attenuation effects applying when energetic photon emitters are involved. However, when the contamination comprises actinides emitting photons of low energy, it may be important to interpose appropriate thicknesses of a suitable tissue substitute [44] between the source and the detector. A realistic thickness of underlying material may also be required, since the response from scattered radiation will not be readily separated from that due to the uncollided flux. It may be possible to assess the effective depth of a deposit of plutonium or of a Pu/$^{241}$Am mixture from the relative fluxes recorded from individual lines in the uranium L X ray spectrum or from the relative responses in the 13–20 keV and in the 60 keV regions. Radiography may help in the location of a contaminated foreign body.

*Thyroid*

These investigations relate mainly to isotopes of iodine, and occasionally to $^{99}$Tc$^{m}$. Commercially available phantoms may be used (e.g. Fig. 15), reproducing roughly the shape and size of the organ and embedded in a cylinder representing the neck, but an adequate substitute (Fig. 16) may be produced locally [35]. The nuclide may be present in solution inside a hollow phantom thyroid, or incorporated in a casting. Except where $^{125}$I (27–35 keV photons) is to be monitored, many tissue substitutes [44] will be acceptable and, from the standpoint of attenuation effects, the thickness of material overlying the phantom organ will not be critical. For $^{125}$I, both the nature and thickness of the overlying soft tissue will need to be more carefully simulated. A mixture of $^{133}$Ba and $^{137}$Cs is sometimes used as a long lived, convenient substitute for $^{131}$I in calibration measurements; this practice has been discouraged [35] in the clinical context, but should not lead to unacceptable error in monitoring for internal contamination.

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## Description

The phantom comprises a head, neck and shoulder region (without arms), fitted with a snap-in thyroid shell and cover-plate. An artificial skeleton is embedded within the phantom. The hollow thyroid is made in one piece from a clear plastic and has posterior ports for rapid filling with a radioactive solution and thorough flushing after use.

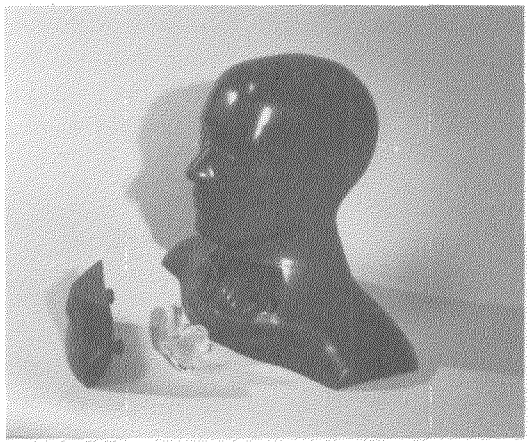
## Physical dimensions

Representative of North American adult.

## Materials

| | |
|---|---|
| Soft tissues | Cast material, cured to solid, soft tissue substitute |
| Skeleton | Artificial (includes cortical bone, spongiosa and cartilage) Earlier humanoid version has human bones |
| Thyroid | Pre-cast in transparent plastic; leak-proof for containment of radioactive solutions |

## Photograph



## Applications

For calibration of scanning and thyroid uptake equipment used in nuclear medicine and radiation protection.

*FIG. 15. RSD (Radiology Support Devices, Incorporated, USA) thyroid calibration phantom for radioiodine measurements.*

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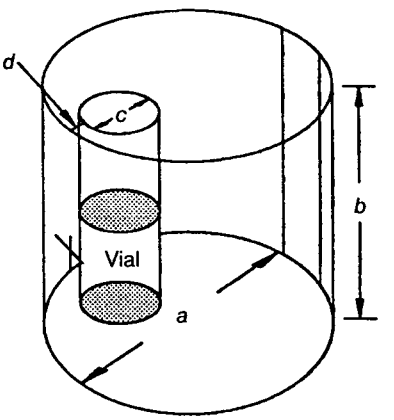
## *Description*

Clear plastic cylinder with cylindrical channel for source vial. The design was originated by the IAEA in 1962, and was reassessed by ANSI in 1973.

## *Diagram*




| *Physical dimensions* | | *Material* |
|---|---|---|
| **Cross-section** (*a*) | | |
| (circle) | 12.7 cm diameter | Acrylic |
| **Length** (*b*) | 12.7 cm | |
| **Source via channel** (*c*) | 3.0 cm diameter | |
| **Thickness** (*d*) to nearest surface of vial | 5 mm | |

## *Applications*

Calibration of radiation detectors used to measure thyroid burden of $^{131}I$.

*FIG. 16. IAEA/ANSI (American National Standards Institute) neck calibration phantom for radioiodine ($^{131}I$) measurements.*

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*Lungs*

Elaborate phantoms may not be required in the calibration of detectors viewing the thorax for purposes of assessing lung deposits of energetic photon emitters. It will often suffice to use an arrangement of stacked plates of an appropriate absorber [44], with gaps to reproduce roughly the bulk density of the lungs and in which point sources may be located. Commercially produced thorax phantoms are available, with cavities accommodating imitation lungs made of a foam such as polyurethane, which can be loaded with the relevant nuclide before casting.

By contrast, large errors may arise in assessing pulmonary deposits of low energy photon emitters, notably when isotopes of plutonium (X ray energies of 13–20 keV) are concerned. The differential attenuation in various categories of tissue (muscle, adipose tissue, lung and bone) must be taken into account [51]. Moreover, the phantom must be anatomically realistic with regard both to the profile of the chest wall and to the configurations of the lungs and other internal structures. These considerations exclude makeshift arrangements and also certain proprietary phantoms designed originally for applications in radiotherapy with photons of much higher energy. The latter performed poorly in evaluating independently known pulmonary deposits of a 20 keV photon emitter [52]. Because the appropriate calibration factor depends strongly on the nature and thickness of the soft tissues of the chest wall, it is necessary both to estimate these variables in the individual subject (for example by ultrasound or by magnetic resonance imaging) and to use a phantom whose design allows for their effects on calibration to be assessed. Only two proprietary designs potentially satisfy the requirements of tissue equivalence, anatomical validity and flexibility in application to a variety of male physiques. One (the chest section of the phantom shown in Fig. 14) is a variant of the phantom designed by the Lawrence Livermore National Laboratory [53]. Similar phantoms were shown on average to predict satisfactorily the 16 keV photon fluxes from independently known pulmonary deposits of $^{92}Nb^{m}$ inhaled by volunteers [54]. The other (available from Kyoto Kagaku Hyohon Co. Ltd, Japan) is intended to represent the Asian male thorax [55]. Both phantoms were designed specifically for assessment of inhaled plutonium, but can equally accommodate lungs loaded with $^{241}Am$, isotopes of uranium or indeed any other nuclide. However, they are suitable only in the calibration of detectors for viewing the anterior aspects of the thorax.

Aside from issues of tissue equivalence and anatomical validity, systematic errors may arise in the assessment of plutonium in lungs by X ray counting, because its pattern of deposition within the lungs will in most cases be indeterminate, and the common assumption of a uniform deposit may not be valid. Many laboratories prefer to assess pulmonary plutonium by recording the 60 keV photons from associated $^{241}Am$, for which calibration presents fewer uncertainties and which is more readily detected. Such an approach presupposes that the radioactive composition of the

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inhaled contamination is known reliably, and that there is no possibility of differential clearance of the two elements from the lungs.

*Other organs and tissues*

Makeshift arrangements, such as matrices of point source locations in stacks of tissue equivalent plates, can be used in assessing deposits of energetic photon emitters in discrete organs, such as liver and kidney, viewed by detectors local to those regions. In defining the matrix, due account would have to be taken of anatomical factors, including, as in the case of the liver, any marked asymmetry of shape and/or position. Alternatively, some of the proprietary phantoms employed in radiotherapy, or produced specifically for use in monitoring internal contamination with fission products, have removable structures representing individual organs, and may be used or adapted for this application.

Extrapulmonary systemic deposits of very low energy photon emitters (such as isotopes of plutonium) are not readily detectable except following intakes which exceed annual limits by orders of magnitude, but there may be a requirement to monitor hepatic or skeletal contamination. In the former application, the two thorax phantoms (Section 4.4.3) developed primarily for assessment of plutonium in lungs will be most useful, since they provide for the inclusion of labelled imitation livers. The assessment of bone deposits of nuclides emitting 40–80 keV photons is generally effected by extrapolation to the entire skeleton of the response of detectors located over specific sites such as the skull or patellae (Section 3.3.3). A series of phantoms labelled with $^{241}$Am, and others labelled with $^{210}$Pb, have been produced by the New York University Medical Center [56, 57], and have been used by several laboratories in the USA, South America and Europe. They were produced from components of human skeletons whose internal and external surfaces were coated with standardized quantities of the radionuclide. The cavities were subsequently filled with tissue equivalent material to simulate the marrow, and external layers of other tissue equivalent material were applied to provide the attenuation expected in overlying soft tissues.

*Interference from radioactive material deposits in adjacent organs*

Situations arise in which assessment of an organ deposit is complicated by the additional presence of the radionuclide in adjacent organs or tissues. In some instances the interference may be substantially eliminated through the use of local shielding, for example in measurements of thyroidal radioiodine, sheet lead covering the neck and leaving only the organ exposed can satisfactorily remove contributions to the response from activity in the blood. Alternatively, interference from the radioactivity of blood or other disseminated soft tissue may in some cases be adequately assessed through a 'background' measurement made on some other part of the body

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which is remote from the organ of principal interest but which possesses the appropriate cross-section. It is frequently found that assessments of pulmonary contamination shortly after an intake are complicated by the presence of material in the upper respiratory and gastrointestinal tracts. A delay of one or two days will allow these to clear and permit a more reliable assessment to be made of the potential long term deposit.

The greatest difficulties are likely with contamination deposited in discrete organs which are closely adjacent; the most common such example in radiological protection relates to accumulations in the lungs and liver. Attempts can be made to separate the two components by observing the responses of a detector or detectors sited close to each organ in turn, and relating them to data recorded in corresponding locations on a phantom, first with the nuclide present solely in one organ and then with only the other loaded. However, the extent of 'cross-talk' will depend on individual anatomical factors which may not be reproduced adequately in the phantom, allowing systematic errors to arise. Moreover, the process involves the solution of simultaneous linear equations which may be ill conditioned. In the case of inhaled $^{241}$Am, the monitoring of lung and liver deposits may be further complicated by an accumulation in the skeleton. Efforts may be made to correct the response of detectors viewing the thorax for the presence of $^{241}$Am in the ribs and other skeletal structures. The approach often depends on monitoring in an additional, bony region, such as the head, where the contributions from lungs and liver are either negligible or can be removed by shielding. Some relationship must then be assumed between the response in that region and the count rate from skeletal $^{241}$Am when the lungs or liver are monitored. This may depend on measurements with bone phantoms such as those already described (Section 4.4.3), or alternatively on measurements of subjects with long standing deposits of $^{241}$Am assumed to be wholly skeletal.

Because of the uncertainties commonly attending the evaluation of interfering components, it may be expedient to calculate the organ content as though all of the measured response derived from the organ itself, and to enter that quantity in the records, explicitly as an upper limit of the contamination present.

# 5. PRACTICAL DETAILS

Irrespective of the scale of monitoring operations, procedures should be established for the processing of subjects, for the execution of measurements, for the documentation and reporting of measurements and data and for adequate quality control.

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## 5.1. IMPLEMENTATION OF MEASUREMENTS

### 5.1.1. Preparation of subjects for measurement

It is important to remove surface contamination if this can readily be achieved. Surface contamination on a subject may invalidate a measurement and, if transferred to the shielded enclosure or its environs, may also prejudice subsequent measurements. The decay products of natural radon deposited on the person may also prejudice the assessment.

Subjects should therefore take a shower, paying particular attention to washing their hair, and should be provided with fresh clothing before entering the monitoring area. Some synthetic textiles are unsuitable because accumulation of electrostatic charge encourages the deposition of radon decay products. Disposable paper garments are favoured by some laboratories.

An explanation of the measurement procedures, in lay terms, may allay any anxieties on the part of those unfamiliar with the process. Some larger dosimetry services provide illustrated brochures in advance of an appointment.

### 5.1.2. Execution of measurements

In whole body counting, the subject should to the extent possible be located on the bed or other support according to some predefined scheme, to ensure reproducibility of the measurement geometry in serial investigations. Reproducibility may be particularly important, although sometimes less readily achieved, in the monitoring of individual organs. Location of the detector(s) by reference to anatomical landmarks may be adequate, or, for repeated measurements in the short term, marking of the skin is a possibility. Where there is known to be irremovable surface contamination of the subject, local shielding of the affected region may eliminate or minimize ensuing errors in the assessment.

Ordinarily, measurements of counter background should be made as close as possible in time to the measurements on a subject. The importance of this requirement will depend on the experience of variability in the background level in relation to the levels of internal contamination required to be detected. Variations in the cosmic ray component of the background will certainly occur according to changes in barometric pressure and, despite the provision of a fast flow of filtered air (Section 2.3.4), variable interference from the decay products of radon may complicate assessments under adverse meteorological conditions. Difficulties may arise from radioactive emissions from local nuclear installations or the movement of sources.

A suitable phantom may be required as a substitute for the subject during measurement of the background (Section 2.3.4). In whole body counting, this may be of the type shown in Fig. 11, with its vessels filled with inactive distilled water; for

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part body investigations, an appropriate component of the same phantom could be used. Sugar filled bags have been used as one alternative.

Some means of contact should be provided for subjects in enclosed shielding. Distractions such as radio programmes or recorded music are possible measures to alleviate boredom or anxiety.

## 5.2. REPORTING AND DOCUMENTATION

Monitoring data from incorporated radionuclides contribute to an individual's exposure history, and will generally be accorded the same status as those relating to external exposure. Aside from any national statutory requirements, the accurate reporting of results and the secure retention of the basic data may be recommended for medico-legal and other reasons.

### 5.2.1. Reporting

Reports should indicate at least the date of monitoring and time of day, together with, for each nuclide assessed, the region monitored (organ, anatomical region or whole body), and the derived activity and its estimated uncertainty from counting statistics (Annex I). Results deemed to be of no statistical significance may be reported as below a minimum significant measured activity (Annex I). However, the actual result (whether numerically positive or negative), with its estimated statistical standard deviation, should be preserved in the records (Section 5.2.2). This will permit any necessary comparisons with earlier or later data for the same individual, and also allow the result to be included in statistical analyses of exposure in a workforce.

Estimates carrying small statistical uncertainties will nevertheless be subject to relatively large systematic errors, reflecting such factors as the validity of the calibration process (Section 4.4), the accuracy with which a spectrum has been resolved into its constituent components (Section 4.3) and errors in allowing for interference from activity in neighbouring regions (Section 4.4.3). Where assessments of the potential systematic errors can be made, they should be indicated in the report, either individually or in combination. The potential errors from all sources will often be small in relation to others which arise in the use of biokinetic models to translate a result into an assessment of intake.

### 5.2.2. Documentation

The information to be retained should include the following:

(i) Unambiguous identification of the subject through name and, in addition, date of birth and/or (for example) social security number.

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(ii) Other personal information relevant to the measurements, such as weight, height, sex and any history of medical administration of radionuclides in cases where unexpected nuclides were found.
(iii) The reasons for referral and the date, time and duration of the measurement.
(iv) The equipment and detection geometry employed, with details of any necessary deviations from standard documented procedures, including measures to deal with any known non-removable surface contamination.
(v) The basic measurement data (gamma ray spectra of subject and of relevant background, or integrated counts over specified energy ranges where only these are recorded).
(vi) The origin and basis of calibration and the methods used to calculate the reported result.
(vii) Statistical uncertainties in calculated values of radionuclide content (Annex I) and any assessments of potential systematic error (Section 5.2.1).
(viii) Copies of reports issued.

## 5.3. QUALITY CONTROL

Quality control in a measurement process is important to ensure that assessments of intakes are as reliable as possible. Evidence of the validity of such assessments may be required for legal and/or regulatory purposes. However, procedures for quality control, as the term is generally understood in relation to analytical processes, are to a large extent inapplicable to body radioactivity measurements, insofar as it is rarely possible to process a test sample whose radioactive content is reliably known. Nor, in some instances, is it possible to quantify reliably the uncertainty or potential systematic error in an assessment where such errors arise predominantly in calibration procedures or in allowances made for major interference from deposits external to the organ of interest.

It follows that quality control procedures will generally be restricted to regular checks on the performance of equipment and to other practices established for the more common applications of gamma ray spectrometry. These may include:

— Checks on the stability of the efficiency, energy calibration and resolution of detector systems by recording spectra of suitable reference point sources.
— Checks on the validity of dead time corrections by following the decay of an initially sufficiently active sample of a short lived radionuclide over several half-lives.
— Checks on the stability of the background response through repeated measurements overnight and at other times when the equipment is not otherwise in use.
— Establishing that a procedure for assessing internal contamination is free from bias by applying it to representative groups of people who have no likelihood

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of exposure to the nuclide concerned. The expense of such a programme, in terms of subjects' time alone, may make this an infrequent exercise.

— Collaboration with other laboratories in programmes of measuring the same sample, which could be a phantom with known radioactive loading, or a subject whose radioactive content may or may not be reliably known. Exercises with phantoms are of limited value since they may have little relevance to the performance of a laboratory when presented with a radioactive person. However, a result well outside the range reported by other laboratories should prompt a careful review of procedures, for example those relating to the use of locally developed phantoms. Such programmes of intercomparison may have additional functions in motivating and broadening the experience of staff whose duties may be otherwise of a routine nature, and in promoting information transfer between participants.

— Establishing and documenting the relationship of locally used standardized radioactive sources to accepted national or international reference standards.

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# Annex I

## INDICES OF STATISTICAL SIGNIFICANCE AND SENSITIVITY

*This annex has been prepared from material provided by staff of the Centre of Radiation Hygiene, National Institute of Public Health, Prague, Czech Republic.*

An appreciation of certain basic statistical concepts in radioactivity measurement is necessary in two related areas: in assessing the statistical reliability of some particular observation of body or organ content, etc., and in expressing the sensitivity of a procedure in terms of the smallest deposit which it may be expected to detect. These concepts are discussed extensively elsewhere [I-1–I-3] and are treated here only in summary. The two most relevant are:

- Minimum significant activity (*MSA*), sometimes referred to as the decision limit $L_C$;
- Minimum detectable activity (*MDA*), otherwise known as minimum detectable true activity or detection limit $L_D$.

The *MSA* is related to the smallest recorded signal which is regarded as significantly in excess of the background response. It corresponds to the level of a randomly fluctuating background response which, in the absence of a radioactive sample, will be exceeded only with some low probability $\alpha$. Conventionally, $\alpha$ is taken to be 0.05, so that a net signal corresponding to deposited activity at the *MSA* level may be taken to indicate the presence of activity with 95% probability. If, as will usually be the case, random fluctuations in the net counts follow a normal distribution, the *MSA* will correspond to $1.64\sigma$, where $\sigma$ is the standard deviation for the distribution. Reports of assessments which lie below this level may legitimately state that activity was 'not detected'. It should be noted that activity genuinely present at the *MSA* level will not necessarily be detected according to this criterion: in that situation there would be a 50% probability that a net recorded signal would be obtained corresponding to a deposit at or below the *MSA*.

The *MDA* corresponds to the deposit which is required to ensure, with some chosen probability $\beta$, that the net signal recorded will be detected according to the designated criterion, i.e. that the signal will correspond to activity exceeding the *MSA*; obviously $MDA > MSA$. The mathematical treatment is simplified if $\beta = \alpha$, and by common convention 0.05 is adopted for both. The *MDA* characterizes a priori the measurement procedure and allows predictions of the sensitivity attainable with specific combinations of equipment, detection geometry, measurement times, etc.

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## Evaluation of MSA and MDA

The equations to be quoted may be derived as in Refs [I-1] and [I-2] or by analogous treatments. The first relate to simple comparisons of sample and background count rates, with no spectrometry performed. In what follows, only variations associated with counting statistics are considered; $n_b$ is the background count rate; $t_s$ and $t_b$ are respectively the count times for the sample and for an associated measurement of the background; $F$ is a calibration factor (count rate per unit activity in the sample); and 95% confidence intervals are assumed to apply, i.e. $\alpha = \beta = 0.05$.

$$MSA = \frac{1.64}{F}\sqrt{\frac{n_b}{t_s}\left[1 + \frac{t_s}{t_b}\right]} \tag{I-1}$$

$$MDA = \frac{2.71}{Ft_s} + 2MSA \tag{I-2}$$

The first term in this expression for *MDA* can be neglected if, as generally applies:

$$n_b t_s \gg \frac{0.7}{[1 + t_s/t_b]} \tag{I-3}$$

Where spectrometry is involved, the assessment of uncertainties and detection limits may become much more complex if there is the additional requirement to evaluate the effects of interfering spectral components; this situation will most commonly occur when scintillation counters, rather than semiconductor detectors, are used. Commercially available or locally developed software for spectral analysis will frequently indicate the uncertainties in calculated activities, on which estimates of *MSA* and *MDA* may be based. In the case of procedures embodying regression analysis, uncertainties may be derived from the matrix algebra and should in effect reflect the sensitivity of the quality of fit to small changes in the assumed composition of the mixture; thus they may include uncertainties arising from instrumental instabilities, etc., as well as those originating purely from counting statistics. When imported software is used for spectral analysis, it is important to establish what the procedures for calculating uncertainties are and to assess whether they are likely to be realistic.

A major simplification is possible if high resolution detectors are employed. In this case, the spectral region of interest will generally be narrow; hence the combined effects of counter background and underlying scatter contributions from other nuclides can be assessed reliably from the recorded response immediately above and/or below the relevant peak channels. In this way, the requirements both for measurement of the counter background and for deconvolution of multiple components

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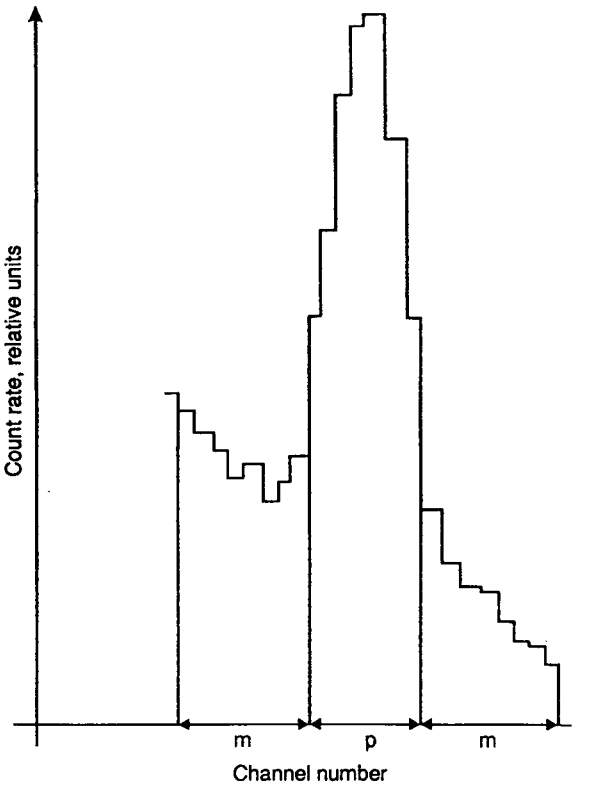


*FIG. I-1. Regions selected for analysis of peaks in a high resolution spectrum (see Eq. (I-4)).*

are removed. In the equations below, $n_s$ is the net count rate attributable to the contaminant in the $p$ channels encompassing the peak, superimposed on an underlying 'baseline' contribution determined from the count rates $n_l$, $n_h$ each recorded in $m$ channels below and above the peak (Fig. I-1); $t$ is the counting time.

$$MSA = \frac{1.64}{F}\sqrt{\frac{(n_l + n_h)p}{2mt}\left[\frac{p}{2mt} + 1\right]} \tag{I-4}$$

$$MDA = \frac{2.71}{Ft} + 2MSA \tag{I-5}$$

and again the first term in this expression for *MDA* can be neglected in most practical situations.

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If the background of a semiconductor detector contains a peak in the spectral region of interest, a separate measurement of this background is necessary. In this case

$$MSA = \frac{1.64}{F}\sqrt{\frac{(n_l + n_h)p}{2mt}\left[\frac{p}{2m} + 1\right] + \frac{b}{t} + \sigma_b^2} \qquad \text{(I-6)}$$

where $b$ is the area of the interfering peak falling in the $p$ channels and $\sigma_b^2$ is the variance in the estimate of this peak count rate.

Situations will occur where the baseline contributions must be determined from the response over unequal band widths above and below the peak of interest, or where it is appropriate to employ a single reference band above the peak. These may arise where excessive forward scatter or interfering peaks are present on the low energy side. In such cases appropriate modifications are required to Eqs (I-4–I-6). When the deconvolution of multiple peaks is required, the values of *MSA* and *MDA* inevitably exceed those applying to isolated peaks.

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## Annex II

# USE OF A SEMICONDUCTOR DETECTOR IN THE INVESTIGATION OF INTERNAL CONTAMINATION WITH COMPLEX MIXTURES OF RADIONUCLIDES

*This annex has been compiled from material describing procedures adopted at the Centre of Radiation Hygiene, National Institute of Public Health, Prague, Czech Republic.*

## II-1. EQUIPMENT AND MEASUREMENT PROCEDURES

A single, *n*-type high purity germanium detector (crystal size 50 mm diameter × 53 mm thick) is used, fitted with a 0.5 mm thick beryllium window. The photopeak efficiency, relative to that of a 75 mm diameter × 75 mm thick NaI(Tl) crystal, is 22%, measured for the 1.33 MeV photons from a source of $^{60}$Co at 250 mm; the energy resolution (full width at half maximum) at 1.33 MeV is 1.9 keV. The detector is installed inside a room with 2.5 m × 2.0 m × 2.0 m internal dimensions and with 210 mm thick walls of armour plate. The room is fed with filtered air giving 12 changes hourly [II-1]. The subject is seated before the detector as in Fig. II-1.

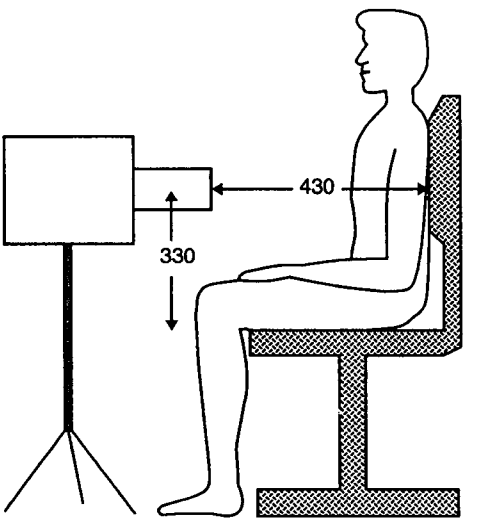


*FIG. II-1. Subject-detector configuration.*

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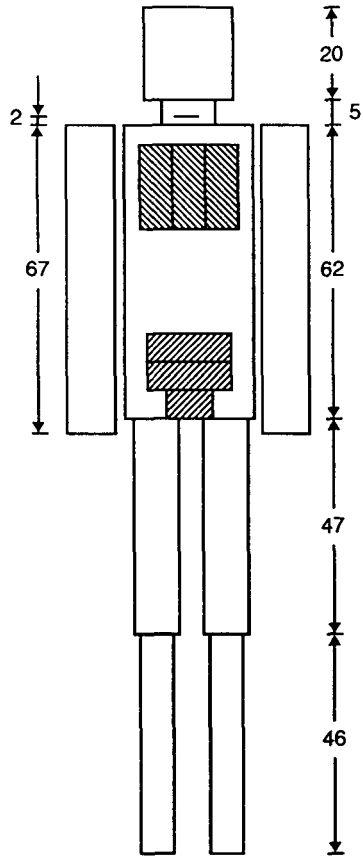
## II-2. ANALYSIS OF SPECTRA

The analysis employs the peak search and quantification routines in the application software 'Spectran F' [II-2], to indicate and evaluate peaks significant with 95% confidence. This treatment is supplemented by visual examination, with manual evaluation of peaks expected from nuclides known to be present but not found by the automated process.




Cross-section of various components of the phantom

| Part of the phantom | Diameter |
|---|---|
| Head | 20 |
| Neck | 12 |
| Trunk | 29 x 19* |
| Arms | 9 |
| Thighs | 12 |

* axes of ellipse

*FIG. II-2. Phantom with simulated thyroid, lungs and intestines (dimensions in cm).*

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## II-3. CALIBRATION

Two types of phantom have been used. For nuclides presumed to be homogeneously distributed, a BOMAB phantom (Fig. 11) is filled with the radionuclide in solution. To provide calibration data for radionuclides specifically present in the lungs, intestines or thyroid, a modified design of total volume 51 L has been developed (Fig. II-2). Within the trunk component there are facilities for locating model lungs or intestines containing the appropriate radionuclide dispersed either in solution or in a polymer matrix. For the lungs three 1 L vessels of rectangular cross-section (dimensions 65 mm × 85 mm × 185 mm) are used, located as in Fig. II-2. For the intestines, two such vessels, with a third consisting of a 200 $cm^3$ cylindrical laboratory wash bottle, are arranged at the base of the trunk, again as in Fig. II-2. The volumes and locations of these models were chosen roughly on the basis of the relevant dimensions in the MIRD mathematical phantom [II-3]. The space remaining in the trunk, and the other components of the phantom, are filled with water as in the BOMAB design. In calibration for deposits in the thyroid, a point source is embedded 23 mm below the frontal surface of the neck component.

Figure II-3 shows the relationship between efficiency in the peak ($\eta_{WB}$, counts per photon emitted) and energy $E$ (keV), determined with standardized quantities of a series of radionuclides dispersed throughout the BOMAB phantom. The data are fitted by:

$$\eta_{WB} = 0.013E^{-0.727} \qquad [140 \leq E(keV) \leq 2760] \qquad \text{(II-1)}$$

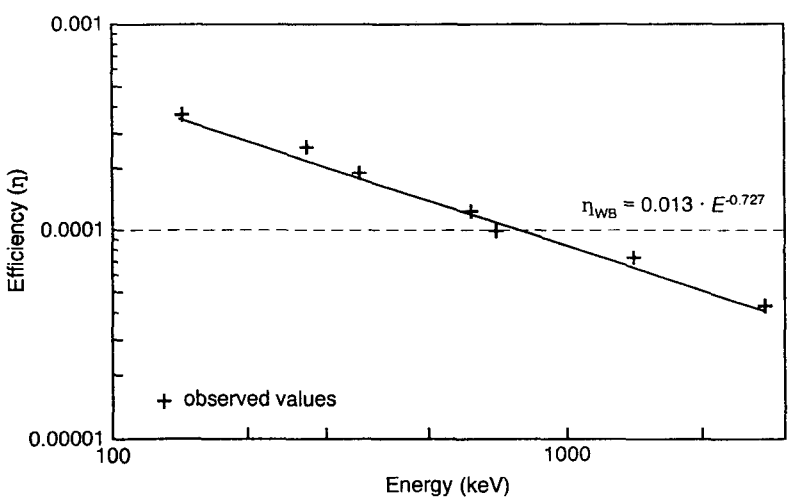


*FIG. II-3. Calibration curve for radionuclides homogeneously distributed in the body.*

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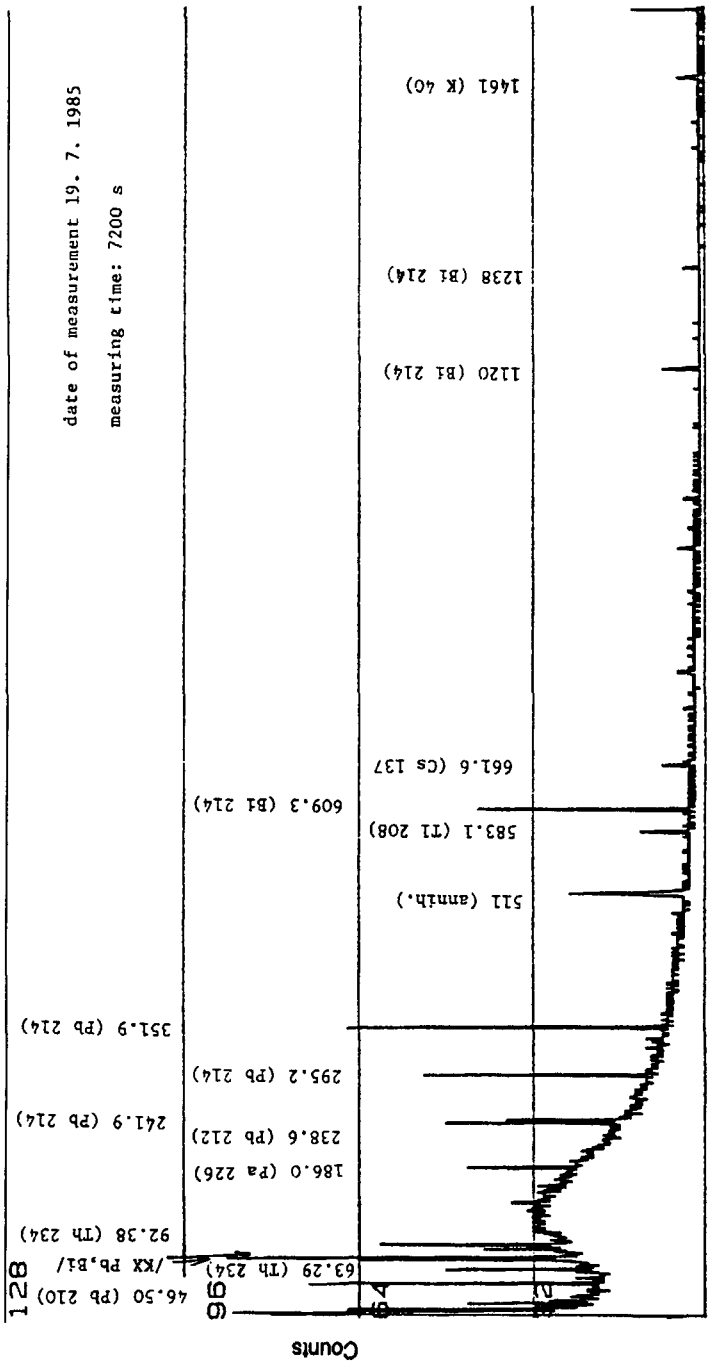


*FIG. II-4. Spectrum of background in the whole body counter shielding prior to the accident at Chernobyl.*

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TABLE II-1. MINIMUM DETECTABLE ACTIVITIES (Bq) FOR NUCLIDES EMITTING PHOTONS WITH 100% ASSUMED ABUNDANCE[a]

| | Whole body | Lungs | Thyroid | Intestines |
|---|---|---|---|---|
| Normal background (Fig. II-4) | 70–130 | 80–150 | 100 | 70–170 |
| Elevated background (Fig. II-5) | 90–160 | 140–270 | 150 | 100–210 |

[a] The estimates relate to photon energies in the range 140–2760 keV, except for the thyroid, where the range is 284–637 keV.

The corresponding functions describing the efficiencies $\eta_L$, $\eta_I$ and $\eta_{TH}$, for detecting deposits in the lungs, intestines and thyroid are:

$$\eta_L = 0.0034E^{-0.559} \quad [140 \leq E(\text{keV}) \leq 2760] \tag{II-2}$$

$$\eta_I = 0.0223E^{-0.813} \quad [140 \leq E(\text{keV}) \leq 2760] \tag{II-3}$$

$$\eta_{TH} = 0.0071E^{-0.659} \quad [284 \leq E(\text{keV}) \leq 637] \tag{II-4}$$

## II-4. MINIMUM DETECTABLE ACTIVITY

Under conditions of normal background (Fig. II-4), *MDA* values ($\alpha = \beta = 0.05$, Annex I) are as in the upper set of figures in Table II-1. They are calculated according to Eqs (I-5) and (I-6), i.e. for the case where interfering peaks occur in the background, with assumed measurement times of 20 min. Ranges of *MDA* are given for each type of assessment for a series of photon energies in the range shown against the relevant function in Section II-3; in all cases 100% gamma ray abundance is assumed. Within these energy ranges, the *MDA* does not show a smooth relationship with energy, being affected by the size of any interfering peak at a given energy.

## II-5. EXAMPLE OF APPLICATION

A spectrum (Fig. II-6) was recorded [II-4, II-5] from a male subject on 5 May 1986, immediately following his return from one month's stay in Kiev and consequent exposure to airborne contamination from the Chernobyl accident on 26 April 1986. The subject had showered and changed clothes to remove any residual surface contamination. At the time of measurement, the background was elevated by the widespread environmental contamination from the accident and was

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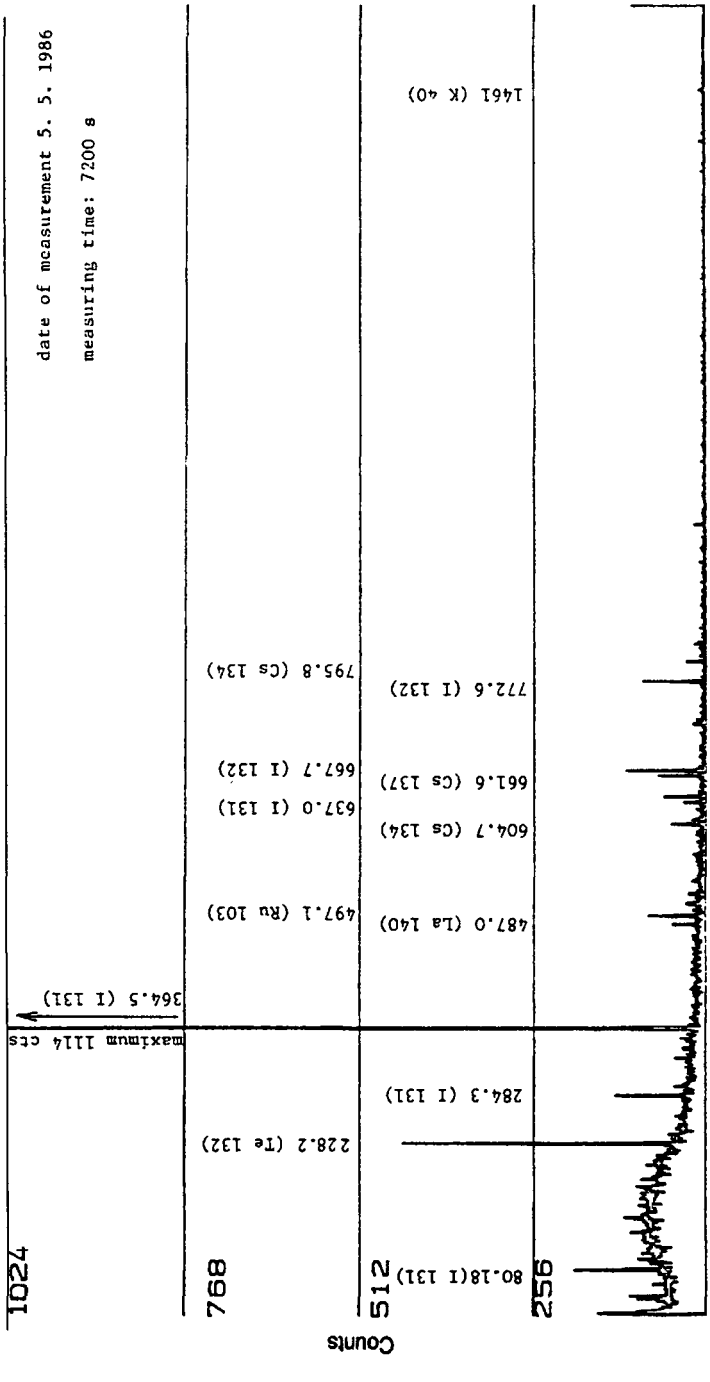


*FIG. II-5. Spectrum of background in the whole body counter shielding 10 days after the accident at Chernobyl.*

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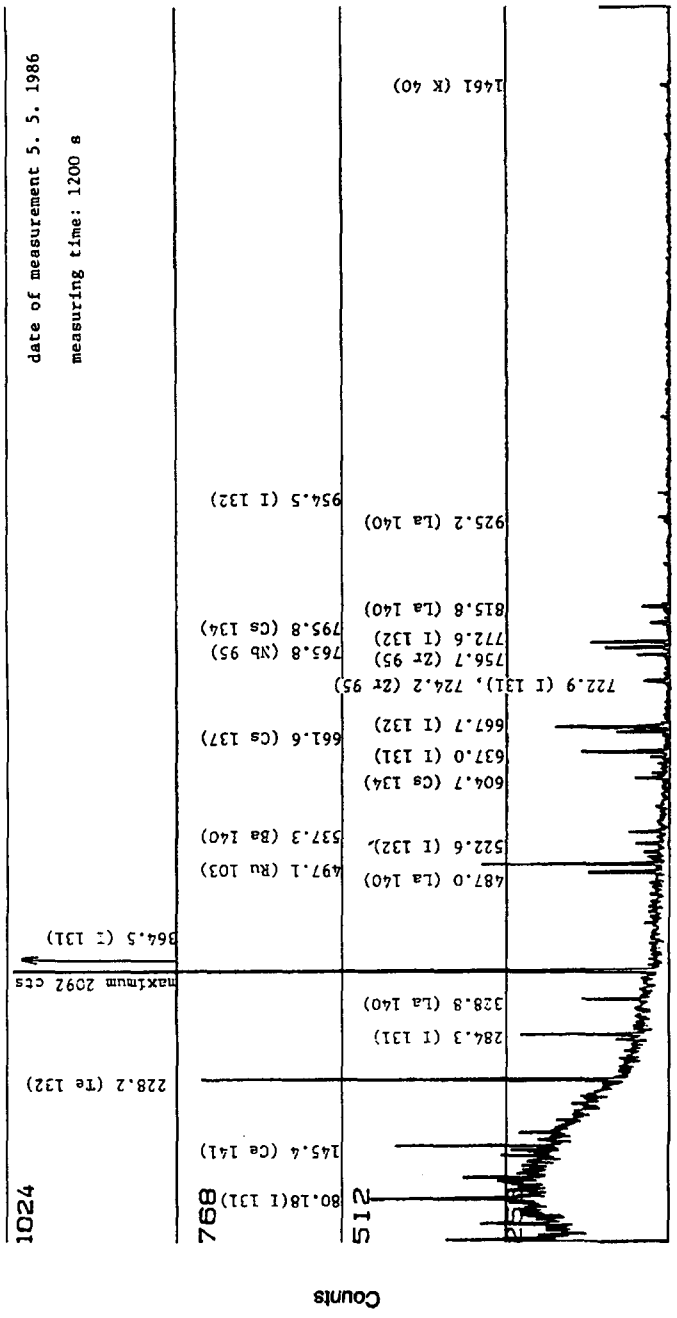


*FIG. II-6. Spectrum of subject internally contaminated with radionuclides released following the accident at Chernobyl.*

TABLE II-2. RADIONUCLIDES IDENTIFIED IN A SUBJECT EXPOSED TO MIXED FISSION AND ACTIVATION PRODUCTS DERIVED FROM THE CHERNOBYL ACCIDENT

PART I

| Radionuclide | Half-life | Main gamma energy (keV) | $f\gamma$[a] | *MDA* (whole body) (kBq) | Activity (calibration for lungs) (kBq) | Activity (calibration for whole body) (kBq) |
|---|---|---|---|---|---|---|
| K-40 | $1.28 \times 10^9$ a | 1461 | 0.107 | 1.08 | — | 3.63 ± 0.90 |
| Zr-95 | 63.98 d | 724.2 | 0.445 | 0.23 | 1.71 ± 0.17 | 1.35 ± 0.13 |
| | | 756.7 | 0.556 | | | |
| Nb-95 | 35.15 d | 765.8 | 1.000 | 0.14 | 2.03 ± 0.15 | 1.61 ± 0.12 |
| Mo-99 | 66 h | 73.9 | 0.122 | [b] | | |
| Tc-99m | 6.02 h | 140.5 | 0.889 | 0.28 | <0.63 | <0.33 |
| Ru-103 | 39.28 d | 497.1 | 0.864 | 0.25 | 4.82 ± 0.31 | 3.56 ± 0.23 |
| Ru-106 | 368.2 d | | | [b] | | |
| Rh-106 | 29.9 s | 511.8 | 0.206 | 1.20 | <1.93 | <1.48 |
| | | 621.8 | 0.0981 | | | |
| Ag-110m | 249.9 d | 657.7 | 0.947 | 0.14 | <0.19 | <0.15 |
| | | 884.7 | 0.729 | | | |
| Sb-125 | 2.77 a | 427.9 | 0.295 | 0.53 | <0.90 | <0.64 |
| | | 600.6 | 0.176 | | | |
| Te-132 | 78.2 h | 228.2 | 0.882 | [b] | | |
| I-132 | 2.3 h | 667.7 | 0.987 | 0.18 | 3.05 ± 0.02[c] | 2.27 ± 0.04 |
| | | 772.6 | 0.762 | | | |
| I-131 | 8.04 d | 364.5 | 0.812 | 0.62 | 31.80 ± 0.56[c] | 22.00 ± 0.39 |
| | | 637.0 | 0.0727 | | | |
| I-133 | 20.8 h | 529.9 | 0.863 | 0.17 | <0.31 | <0.23 |
| Cs-134 | 2.06 a | 604.7 | 0.976 | 0.11 | | |
| | | 795.8 | 0.854 | | 0.65 ± 0.06 | 0.51 ± 0.04 |

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PART II

| Radionuclide | Half-life | Main gamma energy (keV) | $f\gamma$[a] | *MDA* (whole body) (kBq) | Activity (calibration for lungs) (kBq) | Activity (calibration for whole body) (kBq) |
|---|---|---|---|---|---|---|
| Cs-136 | 13.1 d | 818.5 | 0.997 | 0.14 | <0.19 | <0.15 |
| Cs-137 | 30 a | 661.6 | 0.850 | 0.29 | 1.36 ± 0.12 | 1.06 ± 0.10 |
| Ba-140 | 12.74 d | 537.3 | 0.244 | [b] | | |
| | | 162.6 | 0.0621 | | | |
| La-140 | 40.272 h | 487.0 | 0.459 | 0.39 | 3.22 ± 0.63 | 2.47 ± 0.17 |
| | | 1596 | 0.954 | | | |
| Ce-141 | 32.501 d | 145.4 | 0.480 | 0.54 | 2.40 ± 0.61 | 1.26 ± 0.32 |
| Ce-144 | 284.3 d | 133.5 | 0.108 | 2.42 | <4.56 | <2.28 |
| Pr-144 | 17.28 min | 696.5 | 0.0148 | | | |
| Np-239 | 2.35 d | 228.2 | 0.107 | 1.22 | <2.03 | <1.50 |
| | | 277.6 | 0.141 | | | |

[a] $f\gamma$ is the probability of emission of a photon per disintegration.

[b] Measured through the daughter or parent radionuclide.

[c] Efficiency for activity in lungs is roughly the same as for thyroid (see Eqs (II-2) and (II-4)).



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subject to rapid change. The background spectrum in Fig. II-5, obtained two hours after the measurement of the contaminated subject, was chosen as likely to be the most representative of that applying during the measurement. This elevated background led to increased *MDA*s, as indicated by the lower set of values in Table II-1.

Table II-2 lists the decay characteristics, MDAs and estimated radionuclide contents from analysis of the spectrum in Fig. II-6.

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[II-1] INTERNATIONAL ATOMIC ENERGY AGENCY, Directory of Whole Body Radioactivity Monitors. 1970 Edition, IAEA, Vienna (1970).

[II-2] CANBERRA INDUSTRIES, Technical Reference Manual for Spectran F, Version 2, CI-SE-525 (1981).

[II-3] MEDICAL INTERNAL RADIATION DOSE COMMITTEE, J. Nucl. Med., MIRD, Suppl. 3, Pamphlet 5 (August 1969).

[II-4] INSTITUTE OF HYGIENE AND EPIDEMIOLOGY, Report of Radiation Situation in CSSR After the Chernobyl Accident, Prague (1986).

[II-5] MALÁTOVÁ, I., BUČINA, I., ČEŠPÍROVÁ, I., DRÁBOVÁ, D., THOMAS, J., Committed effective dose equivalent from internal contamination of the Czechoslovak population after the Chernobyl accident, Radiat. Prot. Dosim. **28** (1989) 291-301.

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## Annex III

# INVESTIGATION OF WHOLE BODY $^{137}$Cs WITH MOBILE EQUIPMENT

*This annex has been prepared from material describing procedures developed at the Finnish Centre for Radiation and Nuclear Safety, Helsinki.*

## III-1. RADIOLOGICAL CHARACTERISTICS

$^{137}$Cs decays by $\beta^-$ emission with a half-life of 30.2 a. Its incorporation in the body is readily assessed by detection of abundant 662 keV photons emitted by its daughter $^{137}$Ba$^m$ (half-life 2.6 min).

## III-2. EQUIPMENT AND MEASUREMENT PROCEDURES

A single high purity germanium detector of volume 120 cm$^3$ is used. Its efficiency, relative to that of a 75 mm diameter × 75 mm thick NaI(Tl) crystal, is 22% measured for the 1.33 MeV photons from a source of $^{60}$Co at 250 mm; the energy resolution (FWHM) at 1.33 MeV is 1.95 keV. The lead shielding (thickness 70 mm minimum) and subject–detector disposition [III-1, III-2] are as shown in Fig. III-1. The entire equipment (mass 3 t) is housed in a 10 t capacity truck which also carries office and changing facilities.

## III-3. ANALYSIS OF SPECTRA

A locally developed computer program is employed, providing peak search and identification from a library with 92 radionuclides and 486 photon energies. The background spectrum varies considerably according to the location. This is illustrated in Fig. III-2, obtained with the vehicle parked in a garage, showing contributions from $^{40}$K, annihilation radiation (511 keV) and lead X rays (73 keV). There may also be prominent contributions from radiocaesium, depending on the level of local environmental contamination; in such cases an appropriate background has to be determined separately and taken into account in the processing of the data.

## III-4. CALIBRATION

Calibration is effected with appropriate configurations of the phantom shown in Fig. III-2, containing a standardized solution of $^{137}$Cs.

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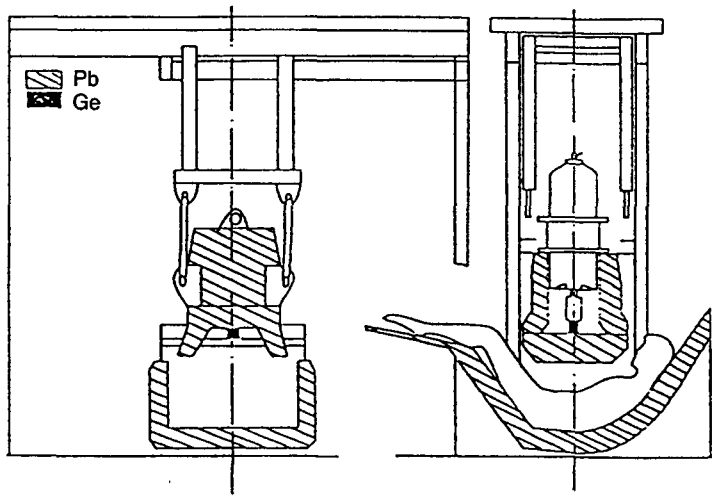


*FIG. III-1. Mobile whole body counting system with high purity germanium detector.*

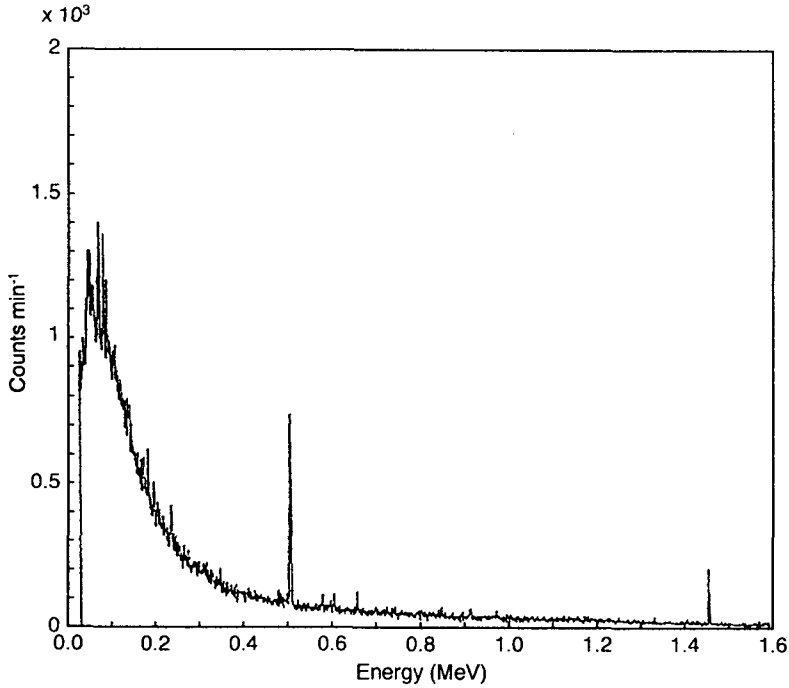


*FIG. III-2. Background spectrum obtained with the vehicle parked in a garage.*

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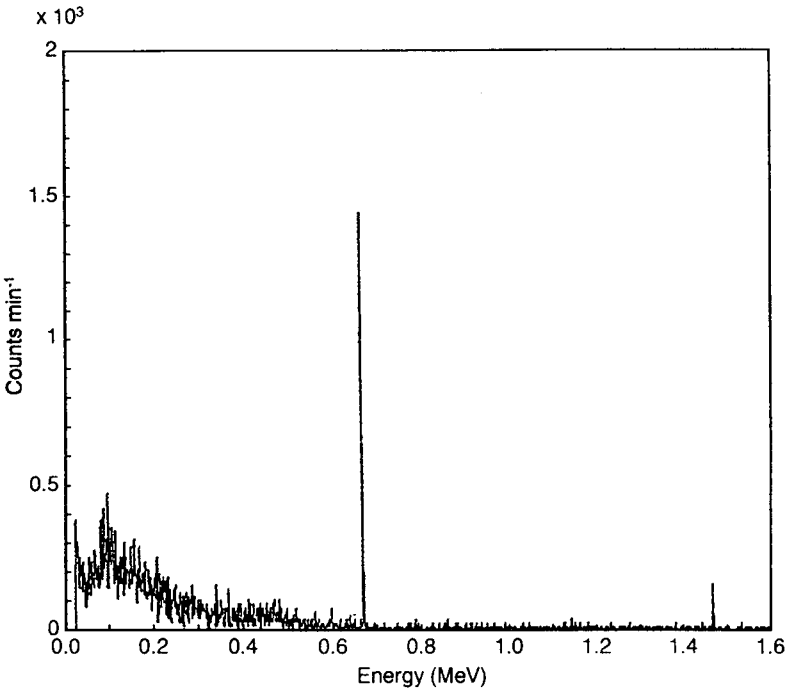


*FIG. III-3. Spectrum recorded in field conditions from a reindeer herder on 7 April 1986.*

## III-5. MINIMUM DETECTABLE ACTIVITY

The minimum detectable activity of $^{137}Cs$ is 50 Bq, for a measurement time of 10 min with the counter in a location where the background level is similar to that shown in Fig. III-2 and contains no interfering contributions from $^{137}Cs$. This value corresponds to $3.29\sigma$, where $\sigma$ is the statistical uncertainty in a determination of zero activity, i.e. 95% confidence limits are assumed.

## III-6. EXAMPLE OF APPLICATION

Figure III-3 shows a spectrum recorded in field conditions from a reindeer herder on 7 April 1986, in a location where no $^{137}Cs$ was detected in the background. Analysis of the spectrum as in III-3 gave the following results:

$^{137}Cs$: 3.0 ± 0.4 kBq
$^{40}K$: 3.7 ± 0.8 kBq

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The uncertainties relate to counting statistics only. In addition, it is estimated that systematic errors of 20% may occur in the calibration procedure. The subject's content of $^{137}Cs$ was much greater than the contemporary average for the Finnish population, and this was attributed to an atypically large dietary intake of reindeer and fish contaminated with $^{137}Cs$ from weapons tests.

## REFERENCES TO ANNEX III

[III-1] SUOMELA, M., Whole Body Counter Studies in Radiation Protection and Clinical Research, Rep. STL-A45, Institute of Radiation Protection, Helsinki (1983).

[III-2] RAHOLA, T., SUOMELA, M., HENTELA, R., OJALA, J., "Advantage of using a semiconductor detector for determining internal radioactive contamination in nuclear power plant workers", Assessment of Radioactive Contamination in Man (Proc. Symp. Paris, 1984), IAEA, Vienna (1985) 107–113.

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# Annex IV

# INVESTIGATION OF INTERNAL CONTAMINATION WITH $^{60}$Co

*This annex has been compiled from material describing procedures at the Battelle Pacific Northwest Laboratories, USA.*

## IV-1. RADIOLOGICAL CHARACTERISTICS

$^{60}$Co decays by beta particle emission with a half life of 5.27 a, with two coincident gamma rays at 1.17 and 1.33 MeV emitted with 100% abundance.

## IV-2. EQUIPMENT AND MEASUREMENT PROCEDURES

A prime objective in this annex is to illustrate the usefulness of semiconductor detectors in the investigation of fission and activation products in vivo, despite certain operational disadvantages (Section 2.3.1). The illustration will be by reference to a fairly common requirement, the assessment of lung deposits of $^{60}$Co. The much improved energy resolution of semiconductor detectors, compared with that available with scintillation counters, is of major benefit in several respects, notably when the contamination is present at low level relative to other nuclides. These benefits include: unambiguous identification of the contaminant; major reductions in the relative interference from scattered radiation associated with other components in the spectrum; and much more convenient and reliable corrections for this interference.

Figure IV-1 shows an array of four coaxial germanium detectors used at the Battelle Pacific Northwest Laboratories in the assessment of internal $^{60}$Co. The crystals are each of about 68 mm diameter; the thickness varies from 68 mm to 89 mm among the four. Typically, the energy resolution at 1.33 MeV is 2–3 keV for a single detector, deteriorating to 5–7 keV when, as here, the outputs from four are combined. Together with their associated liquid nitrogen reservoir, the detectors are mounted in line as shown, with axes vertical, on a motor driven carriage. The equipment is housed inside a cell whose walls, floor and ceiling are made of 0.3 m of steel with inner linings of lead, cadmium and copper. Whole body $^{60}$Co can be assessed by accumulating the response during scans of the anterior and/or posterior surfaces; alternatively, a lung deposit can be estimated with arrays of stationary detectors viewing either or both aspects. Combining the response recorded from both surfaces will provide a detection efficiency which is less dependent on the distribution of the activity within the body or lungs than in the case of a single measurement.

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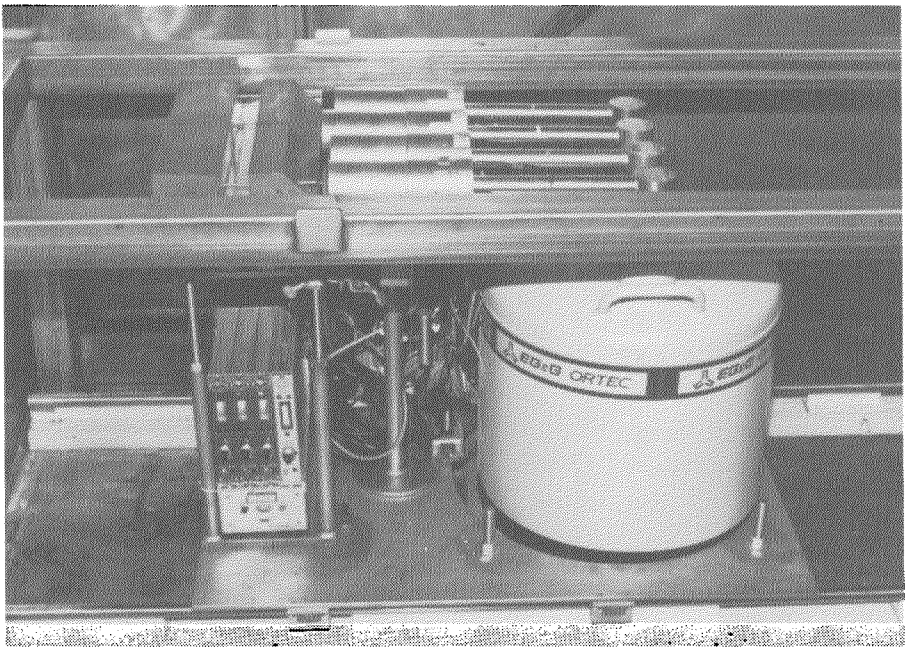


*FIG. IV-1. Array of four semiconductor detectors used at the Battelle Pacific Northwest Laboratories for investigation of whole body or organ deposits of fission and activation products.*

Interference from extrapulmonary $^{60}Co$ can complicate assessment of the lung deposit. This may arise in the early stages following exposure, from material in transit through the gut following initial rapid clearance from the respiratory tract; generally, the significant quantity, from the standpoint of lung dosimetry, is the residual deposit on completion of this phase, and a subsequent measurement will ordinarily remove the ambiguity. Systemic uptake may also occur; systemic $^{60}Co$ is assumed to be deposited in the liver and uniformly in all other tissues in the ratio 4:1 [IV-1]. If a systemic deposit is indeed present and its existence is ignored, the lung content will be exaggerated. The error may be deemed of no consequence if the exaggerated result is nevertheless trivial in relation to established action levels, not warranting the effort and expense of quantifying the interference from extrapulmonary deposits. In other circumstances, collimators may be fitted to detectors viewing the lungs to reduce the interference to an acceptable level. Alternatively, ancillary investigations, including scans with collimated detectors, may be possible to assess the amount and distribution of the systemic component, and the extent of the interference in the assessment of the lung deposit.

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## IV-3. ANALYSIS OF SPECTRA

An estimate of pulmonary $^{60}Co$ may be based on the recorded response in either of the two spectral peaks (1.17 and 1.33 MeV); precision may be improved, when counting statistics are poor, if a suitably weighted mean calculated from both count rates is adopted.

Analysis of a spectrum involves separation of the peak area attributable to the contaminant from the underlying 'baseline' contributed by scattered quanta derived from other sources of body radioactivity and/or the counter background. At its simplest, this underlying continuum is derived by averaging the response over a group of channels (typically three times as many as those encompassing the peak) and scaling this average according to the relative numbers of channels. Alternatively, a two stage peak search process can be adopted to indicate the presence in the spectrum of statistically significant peaks [IV-2]. In this procedure, the content of each channel in the spectrum is subjected to a variance stabilizing transformation, giving unit weighting for each, and allowing the statistical significance of spectral features to be assessed rigorously. A non-linear moving median function is then applied to the untransformed data in regions between the identified features, to establish the continuum for subtraction from the transformed spectrum.

## IV-4. CALIBRATION

Calibration for the scanning procedure is effected with (i) a whole body phantom similar to that in Fig. 11, but containing a standardized quantity of $^{60}Co$ dispersed in a polyurethane filling rather than in solution, and (ii) with a phantom chest (the thorax component of that illustrated in Fig. 14) whose lungs contain $^{60}Co$ uniformly distributed. Both phantoms give similar calibration factors. Calibration when the detectors are in a fixed location relative to the thorax is effected with the phantom chest alone. The same phantom has provision for incorporating a labelled liver; this is useful if ancillary investigations are required to derive adjustments allowing for interference from hepatic deposits, by processes analogous to those described for $^{241}Am$ in Annex VII.

In these procedures, the phantom or relevant organ does not necessarily have to be loaded with $^{60}Co$. Calibration functions applicable over a wide energy range can be established by labelling with an appropriate mixture of nuclides, for example a mixture of $^{154}Eu$, $^{155}Eu$ and $^{125}Sb$ will cover the energy range 87–1274 keV. An empirical combination of linear, power and/or exponential functions can invariably be found to represent adequately the energy dependence of the calibration factor, expressed here as counts per photon emitted in the body or organ.

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## IV-5. MINIMUM DETECTABLE ACTIVITY

For the scanning procedure with a count time of 20 min, the minimum significant measured activity (*MSA*) and the minimum detectable true activity (*MDA*) of whole body $^{60}$Co are respectively 20 Bq and 48 Bq in typical cases, expressed according to Eqs (I-4) and (I-5). The corresponding quantities for the determination of $^{60}$Co specifically in lungs will depend on the extent of interference from deposits elsewhere in the body.

## IV-6. EXAMPLE OF APPLICATION

In routine screening with a linear array of NaI(Tl) detectors, indications were found of $^{60}$Co activity in an employee's chest. A spectrum (Fig. IV-2) was recorded in 20 min with the four semiconductor detectors scanning the posterior surfaces, as

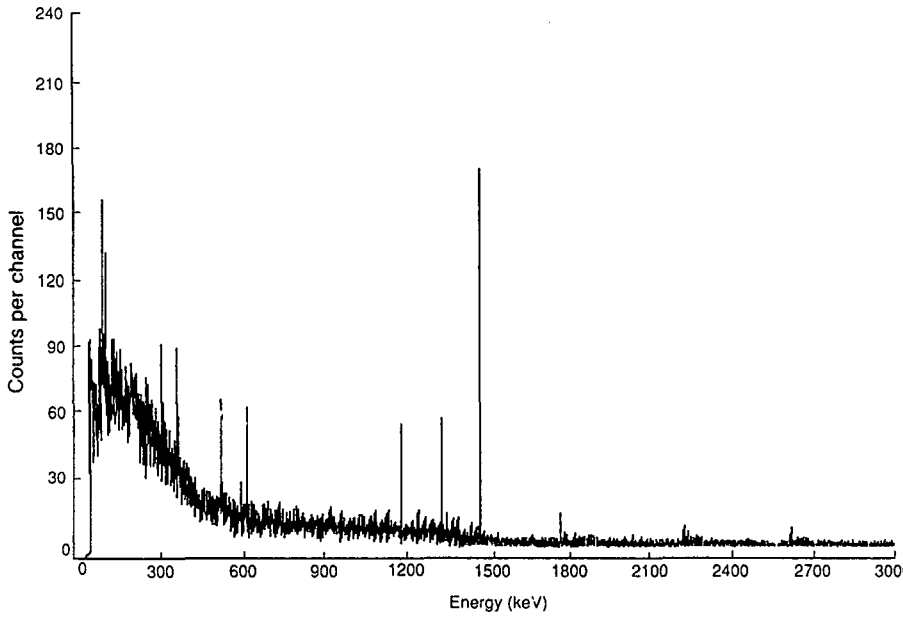


*FIG. IV-2. Gamma ray spectrum recorded during the 20 min scan of a subject contaminated with $^{60}$Co. Prominent peaks include those due to $^{40}$K (1.46 MeV) and $^{60}$Co (1.17, 1.33 MeV) in the subject, and to annihilation radiation (0.511 MeV) in the background. Others arise in the background, at various energies, from radon daughters.*

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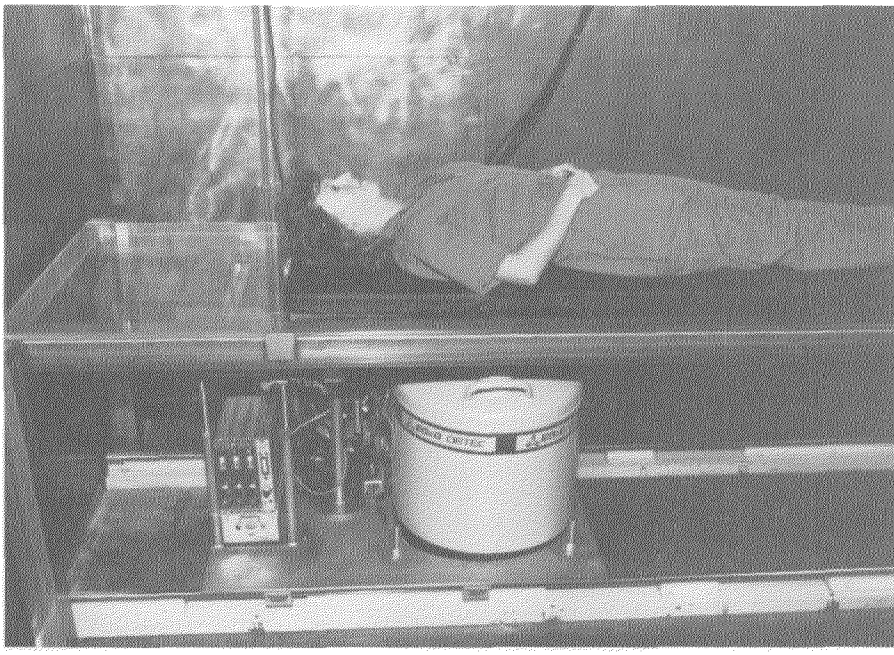

*FIG. IV-3. The four-detector array scanning the posterior surfaces of the supine subject in an investigation of internal $^{60}Co$.*

in Fig. IV-3. The most prominent feature of Fig. IV-2 is the peak at 1.46 MeV, arising from the subject's content of natural $^{40}K$; there are peaks also in the background from annihilation quanta (0.511 MeV), from radon progeny and from Pb X rays. The peaks from $^{60}Co$ (1.17 and 1.33 MeV) are of much lower intensity than that from $^{40}K$ but are readily identified and quantified, with an accuracy not attainable with scintillation counters. The computed activity of $^{60}Co$ in the lungs was 73 $\pm$ 13 Bq, where the uncertainty ($1\sigma$) relates to counting statistics.

The computed quantity was deemed to be of no dosimetric significance; it did not justify efforts at refinement, for example to correct for interference from any deposit in the liver. If higher levels of contamination made the operation both radiologically justified and technically feasible, the distribution in the body would first be investigated by scanning with a suitably collimated 406 mm long NaI(Tl) detector of 102 mm $\times$ 102 mm square cross-section. The outcome would determine the strategy for investigating deposition in individual organs with collimated detectors.

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## REFERENCES TO ANNEX IV

[IV-1] INTERNATIONAL COMMISSION ON RADIOLOGICAL PROTECTION, Limits for Intakes of Radionuclides by Workers, Publication 30, Pergamon Press, Oxford and New York (1979).

[IV-2] SPITZ, H.B., BUSCHBOM, R.L., RIEKSTS, G.A., PALMER, H.E., A new method for analyzing high-resolution spectra from whole body counter in vivo measurements, Health Phys. **49** (1984) 1085–1096.

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## Annex V

# ESTIMATION OF WHOLE BODY $^{137}Cs$ WITH A SCANNING ARRAY OF NaI(Tl) DETECTORS INSTALLED FOR ASSESSMENT OF FISSION AND ACTIVATION PRODUCTS

*The annex has been prepared from material illustrating procedures at the Finnish Centre for Radiation and Nuclear Safety, Helsinki.*

## V-1. RADIOLOGICAL CHARACTERISTICS

$^{137}Cs$ decays by $\beta^-$ emission with a half-life of 30.2 a. Its incorporation in the body is readily assessed by detection of abundant 662 keV photons emitted by its daughter $^{137}Ba^m$ (half-life 2.6 min).

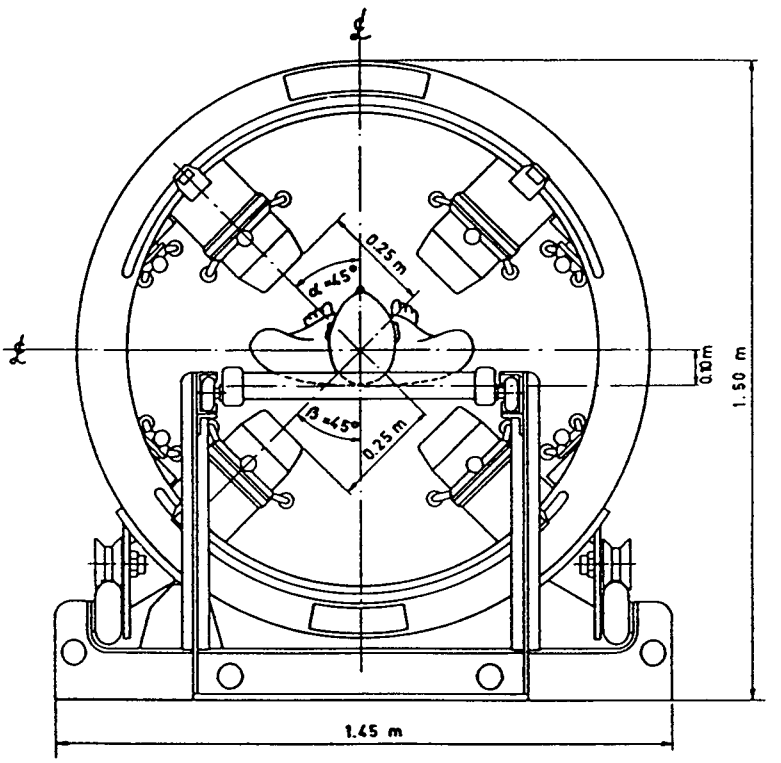


*FIG. V-1. Design of scanning whole body counter used for investigation of internal contamination with fission and activation products.*

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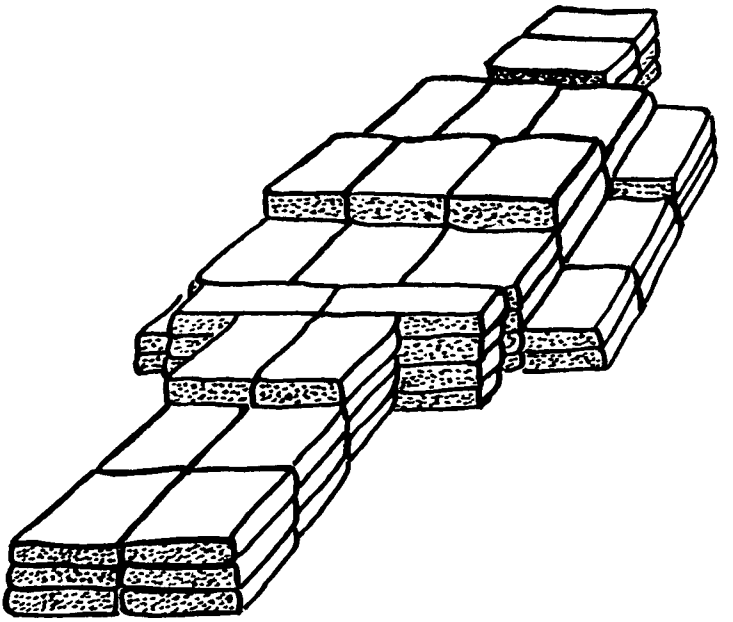

*FIG. V-2. Calibration phantom improvised by assembling solution filled vessels of rectangular cross-section.*

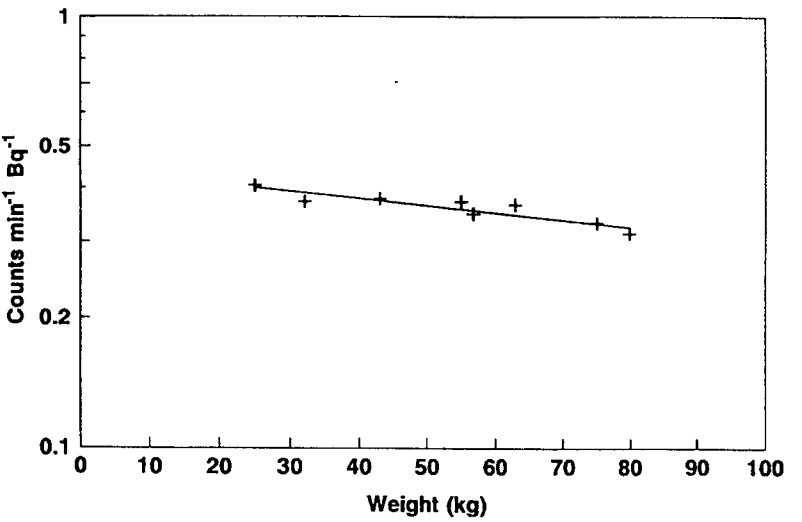


*FIG. V-3. Calibration factors for $^{137}Cs$ observed with eight phantoms of weight 25–80 kg.*

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TABLE V-1. CALIBRATION FACTORS AND *MDA*s FOR DIFFERENT NUCLIDES MEASURED WITH THE IRMA 1 WHOLE BODY COUNTER[a]

| Nuclide | Energy band (keV) | Calibration factor (counts/min per Bq) | *MDA* (Bq) |
|---|---|---|---|
| K-40 | 1335–1585 | 0.0241 | 360 |
| Cr-51 | 270–370 | 0.0457 | 360 |
| Mn-54 | 755–915 | 0.314 | 34 |
| Co-58 | 455–565 | 0.126 | 110 |
| Fe-59 | 975—1185 | 0.303 | 33 |
| Co-60 | 1080–1260 | 0.305 | 28 |
| Se-75 | 80–450 | 1.61 | 20 |
| Ag-110m | 810—960 | 0.316 | 29 |
| Sb-124 | 1570–1810 | 0.110 | 54 |
| Cs-137 | 585–735 | 0.289 | 43 |

[a] The weight of the calibration phantom was 70 kg and the measuring time 30 min for both subject and background. The *MDA* values are based on $3.29\sigma$, where $\sigma$ is the statistical uncertainty in a single determination of zero activity.

## V-2. EQUIPMENT AND MEASUREMENT PROCEDURE

Four NaI(Tl) detectors, each 127 mm diameter × 102 mm thick, are used in a scanning bed geometry (Section 3.2.4) arranged radially about the long axis of the body as shown in Fig. V-1. The frame holding the detectors is driven at a constant speed along the bed over a distance of 1.7 m. The equipment is housed in a steel room with walls 150 mm thick and inner linings of 3 mm lead and 1 mm electrolytic copper [V-1]. Filtered air is supplied to the room, giving 4–5 changes hourly.

## V-3. ANALYSIS OF SPECTRA

The method of analysis, illustrated in Section V-6, involves successive stripping (Section 4.3) of normalized reference spectra.

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## V-4. CALIBRATION

Phantoms, either of the BOMAB design (Fig. 11) or improvised from a series of vessels of rectangular cross-section (Fig. V-2) are filled with solutions containing known concentrations of relevant radionuclides, and are scanned in the same way as the human subjects. Figure V-3 shows the calibration factor (counts/min per becquerel in the energy range 585–735 keV) for $^{137}Cs$ determined for a series of phantoms weighing 25–80 kg. Calibration factors for $^{137}Cs$ and various other nuclides, appropriate to a phantom weighing 70 kg, are given in Table V-1.

## V-5. MINIMUM DETECTABLE ACTIVITY

Estimates of minimum detectable activity (*MDA*) are given in Table V-1 for various nuclides; equal counting times of 30 min are assumed for subject and background. The estimates correspond to $3.29\sigma$, where $\sigma$ is the statistical uncertainty in a determination of zero activity, i.e. 95% confidence limits are assumed. These *MDA*s apply to the situation where a single radionuclide contributes in the energy range considered; higher values will apply if other components interfere.

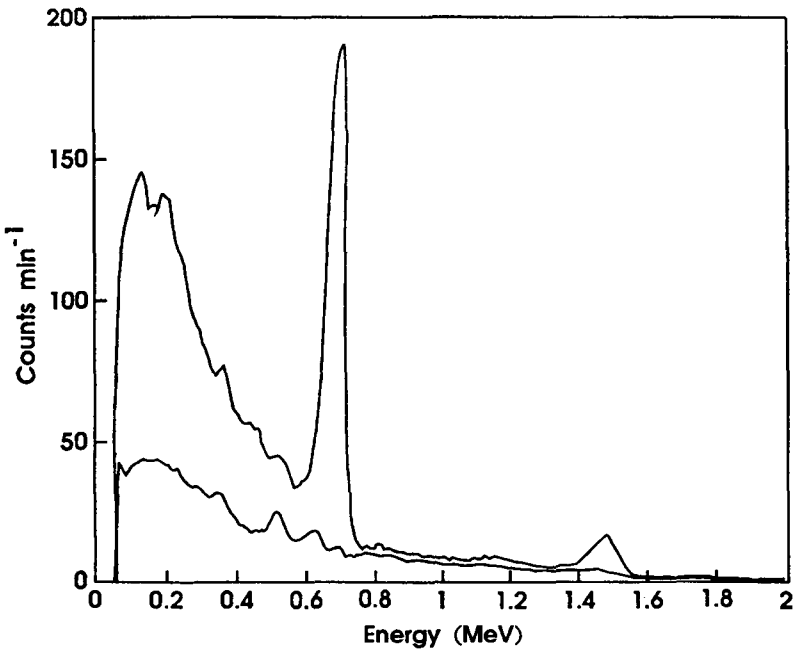


*FIG. V-4. Spectra recorded from subject contaminated with $^{137}Cs$ (upper plot) and background measured in presence of an inactive phantom (lower plot).*

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## V-6. EXAMPLE OF APPLICATION

Figure V-4 shows (i) a spectrum (duration 33 min) recorded from a male subject weighing 84 kg who had become accidentally contaminated with $^{137}Cs$, and also (ii) a background spectrum recorded over 60 min in the presence of an inactive, water filled phantom. The subject's contents of $^{40}K$ and $^{137}Cs$ were derived as follows:

*(a) Estimation of $^{40}K$*

| | |
|---|---|
| Observed count rate (1335–1585 keV) with subject present: | 223.5 ± 2.6 counts/min |
| Background count rate (1335–1585 keV) with phantom present: | 91.1 ± 1.2 counts/min |
| Net count rate: | 132.4 ± 2.9 counts/min |
| Calibration factor for 84 kg body weight: | 0.0241 counts/min per Bq |
| $^{40}K$ content: | 5.49 ± 0.12 kBq |

*(b) Estimation of $^{137}Cs$*

| | |
|---|---|
| Observed count rate (585–735 keV) with subject present: | 1251 ± 6 counts/min |
| Background count rate (585–735 keV) with phantom present: | 191 ± 2 counts/min |
| Net count rate: | 1060 ± 6 counts/min |
| Contribution from $^{40}K$[1]: | 38 ± 1 counts/min |
| Net count rate with $^{40}K$ response subtracted: | 1022 ± 6 counts/min |
| Calibration factor for 84 kg body weight: | 0.290 counts/min per Bq |
| $^{137}Cs$ content: | 3.52 ± 0.02 kBq |

In addition to the uncertainty from counting statistics (0.02 kBq) in the estimated $^{137}Cs$ content, other uncertainties, notably those inherent in the calibration procedure, may contribute errors of up to 15%.

[1] From relative responses in the bands 585–735 keV and 1335–1585 keV recorded from the potassium phantom, and net response from subject in the 1335–1585 keV band.

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## REFERENCE TO ANNEX V

[V-1] INTERNATIONAL ATOMIC ENERGY AGENCY, Directory of Whole Body Radioactivity Monitors, 1970 Edition, IAEA, Vienna (1970).

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# Annex VI

# ASSESSMENT OF $^{125}I$ AND $^{131}I$ IN THE THYROID

*This annex has been prepared from material describing procedures followed at GSF, Forschungszentrum für Umwelt und Gesundheit GmbH, Frankfurt am Main, Germany.*

## VI-1. RADIOLOGICAL CHARACTERISTICS

$^{125}I$ decays by electron capture with a half-life of 60 d, emitting a complex spectrum of X and $\gamma$ radiation (energy range 27–35 keV) with a total abundance of 146%. $^{131}I$ is a $\beta$ emitter with a half-life of 8.0 d and principal photon emissions at 364 keV (abundance 81%).

## VI-2. EQUIPMENT AND MEASUREMENT PROCEDURES

The rigorous assessment of thyroidal radioiodine requires a detector with a restricted field of view to avoid interference from deposits elsewhere in the body [VI-1]. The equipment installed at GSF, with less restrictive collimation, has nevertheless been found adequate where the measurement occurs at a sufficient time after intake for this interference to be unimportant.

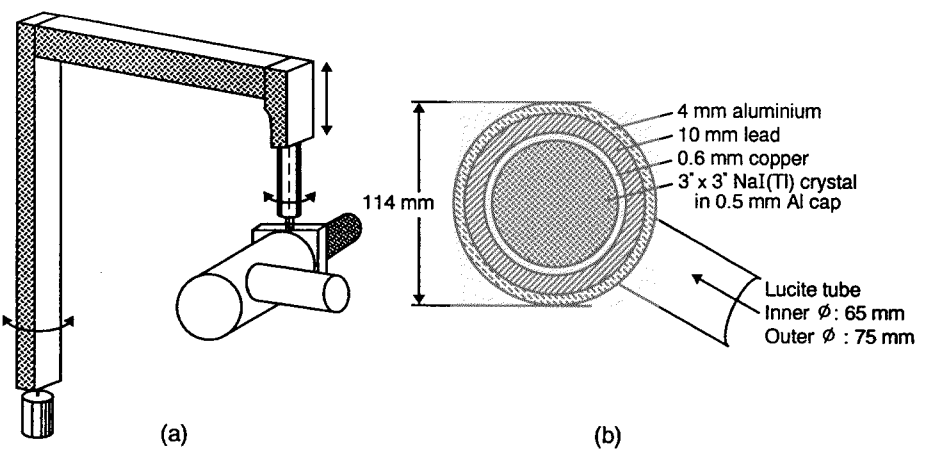


*FIG. VI-1. Movable thyroid monitoring system.*
*(a) General view*
*(b) Mounting and shielding of the 3" × 3" NaI(Tl) crystal.*

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The detector and its support (Fig. VI-1) allow investigations of thyroid deposits to take place in a shielded room [VI-2], simultaneously with assessments of whole body radioactivity in a chair geometry (Fig. 8). The detector is an NaI(Tl) crystal, 76 mm diameter × 76 mm thick, canned in alumnium 0.5 mm thick. The detector and photomultiplier tube housing are contained in a 10 mm thick lead cylinder with an inner lining of 0.6 mm copper, totally enclosed except for an aperture of 76 mm diameter in the *cylindrical* surfaces proximal to the thyroid. A series of interchangeable lucite tubes, recessed into the aperture and perpendicular to the axis of the detector, act as spacers to locate the crystal at the required distance from the surface of the neck (usually 120 mm). Photon energy spectra are accumulated with a multichannel analyser; typical measurement times are 10 min for the subject and 100 min for the counter background.

## VI-3. ANALYSIS OF SPECTRA

For $^{125}I$, estimates of the thyroid content are based on the response in the energy band 20–36 keV, containing most of the response from the unresolved K X ray and $\gamma$ radiations (Fig. VI-2). A contribution in this region might be expected

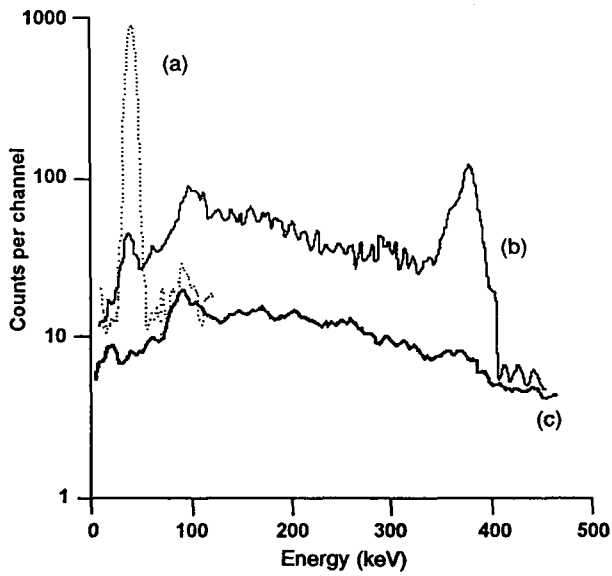


*FIG. VI-2. Gamma ray spectra measured with the thyroid device.*
*(a) subject with 1300 Bq $^{125}I$ thyroidal activity,*
*(b) subject with 440 Bq $^{131}I$ thyroidal activity,*
*(c) background spectrum without subject.*

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through detection of scattered quanta from $^{40}K$. However, measurements of uncontaminated subjects have shown no systematic elevation of the background response recorded in the subject's absence, presumably because the $^{40}K$ component compensates for the attenuation of ambient photons by the subject. Consequently, in the absence of other interfering contamination, no manipulation of the subject's spectrum is performed, beyond the subtraction of a background spectrum, which is recorded without an inactive phantom. An additional summation, over the range 50–66 keV, representing signals from coincident X and $\gamma$ rays, may be useful in matching the calibration factor (Section VI-4) to the individual subject, with considerable improvement to the accuracy of the estimated deposit.

Estimates of $^{131}I$ are based on the recorded response in the 364 keV peak (Fig. VI-2, range 332–397 keV), again after background subtraction; in this case signals from scattered radiation (range 100–250 keV) may also be integrated as an aid in the calibration procedure (Section VI-4). Contributions from $^{40}K$ are assessed from correlations observed in uncontaminated subjects, connecting body weight and intensity of scattered radiation.

## VI-4. CALIBRATION

A phantom neck (Fig. VI-3) is employed. It consists of a lucite cylinder, 120 mm diameter × 200 mm. Paired cylindrical access channels are provided as shown; into these it is possible to insert vessels of various wall thicknesses containing

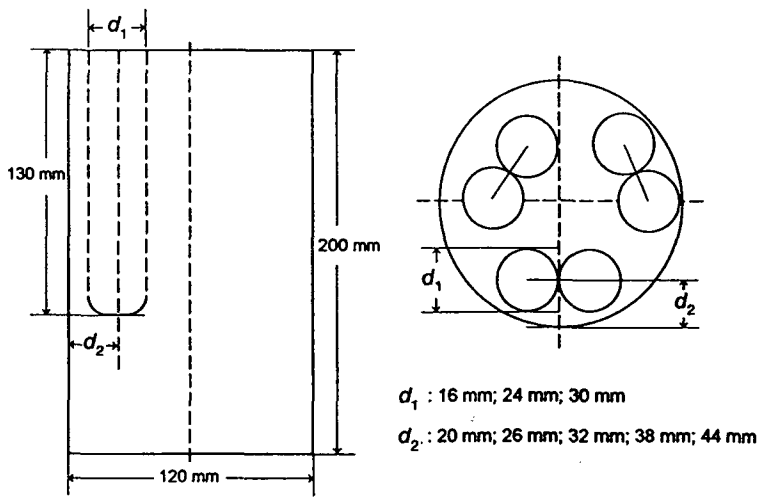


*FIG. VI-3. Scheme of thyroid–neck phantom made out of a block of lucite and used for calibration and intercomparison measurements. Two cylinders are available, so that mean organ depth $d_2$ can be varied between 20 and 44 mm and thyroid masses between 2 and 55 g respectively.*

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TABLE VI-1. COUNTING CHARACTERISTICS OF A TYPICAL THYROID MONITORING DEVICE

*(Shielded 3 in. × 3 in. NaI(Tl) detector system. Distance between detector and neck 120 mm)*

| | I-125 | I-131 |
|---|---|---|
| Calibration factor ($F$) [counts/min × Bq] | 0.46 | 0.29 |
| Background count rate ($n_b$) [counts/min] | 5 | 14 |
| Counting time persons ($t_s$) [min] | 20 | 20 |
| Counting time background ($t_b$) [min] | 100 | 100 |
| *MSA* [Bq] | 2 | 5 |
| *MDA* [Bq] | 4 | 11 |

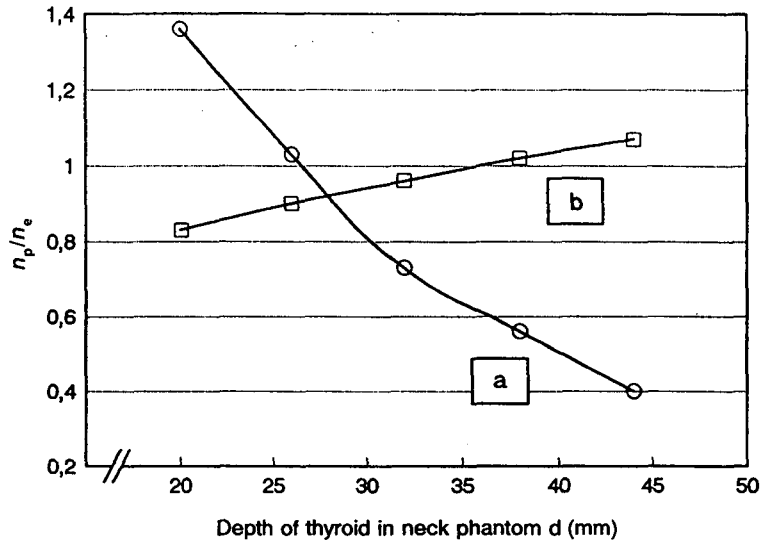


*FIG. VI-4. The relative response in the primary peak and coincidence summation regions from a detector–neck distance.*

*(a) Ratio $n_p/n_c$ of net counts in the photopeak region ($n_p$ = 20–36 keV) to net counts in the coincidence peak region ($n_c$ = 50–66 keV) for $^{125}I(\times 10^{-2})$*

*(b) Ratio $n_p/n_c$ of net counts in the photopeak region ($n_p$ = 332–397 keV) to net counts in the Compton region ($n_c$ = 100–250 keV) for $^{131}I$.*

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the calibration source either in liquid form or as a gel [VI-3]. In this way, calibration factors could be derived for a series of organ masses between 2 and 55 g at mean depths in the range 20–44 mm. The detection efficiency for $^{131}I$ was found to decrease by only 5% as the mass was increased within this range, for a given mean depth $d_2$ (Fig. VI-3) of the thyroid; for $^{125}I$ this variation was 10%. Variation of the detector–neck distance by $\pm 5$ mm affected the efficiency by about 5%; displacement of the phantom relative to the axis of the detector changed the response by $<5\%$, provided that the source did not extend beyond the projected area of the exposed crystal surface.

By contrast, the response was found to depend strongly on the mean depth $d_2$, with changes of $\pm 10$ mm affecting the efficiency for $^{131}I$ by roughly $\pm 20\%$ and that for $^{125}I$ by about $\pm 40\%$. For an assumed mass of 20 g at a mean depth of 32 mm, the calibration factors were as given in Table VI-1. Use of these factors may result in systematic errors of up to 30% ($^{131}I$) and 50% ($^{125}I$), arising mainly from variations in actual mean depths about that assumed. These errors may be reduced if the actual depth can be estimated by ultrasonic methods or, in some instances, by inference from the recorded spectra where these exhibit good counting statistics, as follows.

Figure VI-4(a) shows that with a much reduced detector–neck distance (30 mm) the relative response in the primary peak (20–36 keV) and coincidence summation (50–66 keV) regions for $^{125}I$ depends strongly on $d_2$; however, the use of such relationships requires a thyroid deposit of $>1$ kBq. Figure VI-4(b) shows a systematic increase with depth in the relative scatter component (100–250 keV) for $^{131}I$, which may be used to derive a refined calibration for deposits of 1 kBq or more.

## VI-5. MINIMUM DETECTABLE ACTIVITY

Estimates of minimum significant activity (*MSA*) and of minimum detectable activity (*MDA*), calculated according to Eqs (I-1) and (I-2), are given in Table VI-1. They apply only in the absence of interference from other contaminants.

## VI-6. INTERLABORATORY COMPARISONS

The neck/thyroid phantom (Fig. VI-3) has been employed as the test object in a series of interlaboratory exercises [VI-4] intended to assess the variability of estimates of $^{125}I$ and $^{131}I$ in the thyroid made by different methods. In one of these comparisons, a standardized quantity of $^{131}I$ was incorporated into a dummy thyroid of mass 20 g and located at a mean depth of 32 mm in the phantom. The independent assessments of 27 laboratories in five European countries, with their estimated limits of random and systematic error, are shown in Fig. VI-5, relative to the known content. Large differences are evident, in some cases well in excess of expectation from

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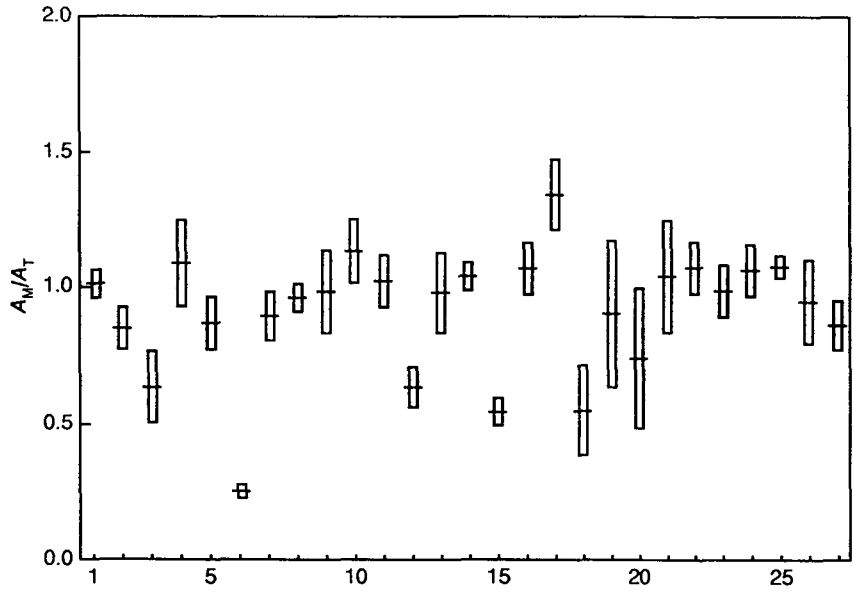


*FIG. VI-5. In vitro intercomparison of $^{131}I$ thyroid activity. Dummy thyroid mass: 20 g. Mean depth of the thyroid in the neck: 32 mm. X-axis: Laboratory code; Y-axis: Ratios of measured activity ($A_M$) to true activity ($A_T$) as obtained by the 27 participating laboratories. Measured values and range of uncertainties as estimated by the laboratories. Detector systems applied include NaI(Tl), HP-Ge, and others.*

the quoted uncertainties. Even larger discrepancies were found for $^{125}I$, both present in a phantom and deposited in human subjects. These findings point to the usefulness of such exercises as a component of quality assurance procedures (Section 5.3).

## REFERENCES TO ANNEX VI

[VI-1] INTERNATIONAL ATOMIC ENERGY AGENCY, Report of consultants' meeting on the calibration and standardisation of thyroid radioiodine uptake measurements, Brit. J. Radiol. **35** (1962) 205–210.

[VI-2] INTERNATIONAL ATOMIC ENERGY AGENCY, Directory of Whole Body Radioactivity Monitors, 1970 Edition, IAEA, Vienna (1970).

[VI-3] OLENDER, G., KRAMER, G.H., Evaluation of the Properties of NORSOCRYL A1-S and B-65 as Tissue Substitute Gelling Agents for the BRMD BOMAB Phantom Family, Bureau of Radiation and Medical Devices, HMLTD-92-6, Ottawa, Ontario (1992).

[VI-4] WERNER, E., ROTH, P., HANSEN, Ch., ALT, P., Vergleichende Untersuchungen zur Ermittlung der Schilddrüsenaktivität bei Inkorporation von Jodisotopen mit Teil- und Ganzkörperzählern, Rep. BNU-St-Sch-1112, J.W. Goethe-Universität, Frankfurt am Main (1992).

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# Annex VII

# INVESTIGATION OF INTERNAL CONTAMINATION WITH $^{241}$Am

$^{241}$Am is a radiologically important component of irradiated nuclear fuel and it will accumulate in plutonium which has been chemically separated during fuel reprocessing, as a decay product of $^{241}$Pu. Its technical applications include uses in X ray fluorescence, in moisture, thickness and liquid level gauges, in bone densitometers and in smoke detectors. Investigations may therefore be required to assess internal $^{241}$Am either as a contaminant in its own right, or alternatively in cases where its presence in inhaled plutonium offers a valid and more sensitive means of assessing contamination with the matrix material than is possible through the detection of uranium L X rays from the decay of plutonium. The initial site of deposition will most commonly be the lung, with the likelihood of subsequent translocation to systemic sites, including liver, bone and lymph nodes associated with the lung [VII-1], although there are reported cases showing no detectable redistribution [VII-2]. A rigorous investigation may require the acquisition of spectra recorded in several anatomical regions, complemented by a programme of excretion analysis.

Sections VII-1 and VII-2 describe two approaches to the investigation of internal $^{241}$Am through body radioactivity measurement. The first is one of the methods practised at the Kernforschungszentrum Karlsruhe GmbH (KFK), Germany, using large area phoswich detectors; the second, with arrays of semiconductor detectors, is favoured at Battelle Pacific Northwest Laboratories (BNL), Richland, USA, and various other laboratories in the USA. It will be noted that the two approaches, and the interpretation of the measurements described in each, are based on premises which conflict in certain respects.

## VII-1. INVESTIGATION OF INTERNAL CONTAMINATION WITH $^{241}$Am WITH PAIRED PHOSWICH DETECTORS

*This section has been compiled from material describing procedures followed at the Kernforschungszentrum Karlsruhe GmbH, Germany.*

### VII-1.1. Radiological characteristics

The isotope $^{241}$Am decays by alpha particle emission with a half-life of 432 a. The decay is accompanied by gamma ray emission at 59.6 keV (abundance 36%). L X and gamma rays of lower energy (range 14–26 keV) are also produced but are not favoured as a means of detection. The reasons are that they are severely attenuated in the body and that their contributions cannot be resolved from those given by X rays due to the decay of any associated plutonium.

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TABLE VII-1. COUNTING EFFICIENCY (COUNTS IN 3000 s PER Bq IN ENERGY RANGE 20–80 keV) FOR DETECTING $^{241}$Am WITH PAIRED PHOSWICH DETECTORS

| Region viewed | Detection efficiencies for $^{241}$Am in | | | |
|---|---|---|---|---|
| | Lungs | Lymph nodes | Liver | Bone |
| Skull | 6.4 | 4.3 | 1.07 | 22.5 |
| Lungs | 114.6 | 73.9 | 32.1 | 10.7 |
| Liver | 16.1 | 10.7 | 128.5 | 10.7 |
| Knees | <0.1 | <0.1 | <0.1 | 38.6 |

### VII-1.2. Equipment and measurement procedures

Paired phoswich detectors (See Section 2.3.1) are used inside a large shielded room with walls of 150 mm steel and inner linings of 3 mm lead, 1.5 mm tin and 0.5 mm copper. The detectors are of diameter 200 mm; the primary element consists of NaI(Tl) (thallium doped) 1 mm thick and the rear guard crystal of 50 mm thick CsI(Tl). For the investigation of pulmonary contamination the detectors are mounted as close as possible to the anterior aspects of the upper thorax, one viewing each lung; for assessment of hepatic $^{241}$Am appropriate locations are selected over the lower thorax; and in the study of skeletal activity, separate measurements are made with the detectors located symmetrically close to the skull and patellae. When an initial pulmonary deposit has become partially systemic, the response in most of these locations will arise from $^{241}$Am in several organs (Table VII-1) and deconvolution procedures (Section VII-1.7) are required to assess the deposit in each.

### VII-1.3. Typical spectra

Figure VII-1 shows spectra of the combined response of the paired detectors viewing the lungs of (a) a subject occupationally exposed to airborne $^{241}$Am (upper plot) and (b) an unexposed subject of similar physique and similar body contents of potassium and radiocaesium. In neither case has a counter background spectrum been subtracted. The response with the unexposed subject arises from local environmental background within the shield and from scattered radiation associated with the body's contents of natural $^{40}$K and radiocaesium resulting from weapons tests and/or released at Chernobyl. With the intrinsically poor energy resolution of the phoswich detector, the elevation in the response from the exposed subject occurs over a 60 keV

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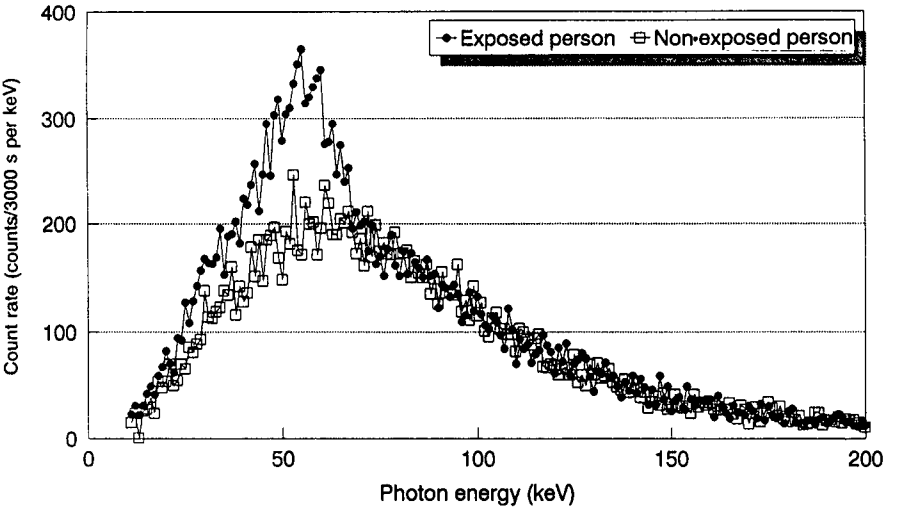


*FIG. VII-1. Spectra recorded in 3000 s with paired phoswich detectors viewing the lungs of a worker contaminated with $^{241}Am$ (upper plot) and of an unexposed subject (lower plot).*

wide energy range, rather than as clearly identifiable, discrete peaks within that range. Consequently, low level pulmonary contamination (a few becquerel) with $^{241}Am$ will generally not give rise to a statistically significant elevation, and when a significant increase is observed the elevation may not be unambiguously attributable to $^{241}Am$.

### VII-1.4. Analysis of the spectrum

One of two approaches may be followed to isolate the component due to $^{241}Am$ in a spectrum such as that shown (Fig. VII-1, upper plot). The spectrum may be compared with that of an appropriate, uncontaminated subject of similar physique and similar contents of potassium and radiocaesium; the validity of the latter spectrum may be judged by how well it matches that from the exposed subject at energies exceeding 80 keV. Sometimes a spectrum from the exposed subject may be used, recorded prior to the exposure, if there have been no intermediate changes in physique or in contents of nuclides other than $^{241}Am$. In general there is no suitable spectrum for direct comparison, and the following relationship is adopted to estimate the count rate $Z_{B1}$ in the energy range 20–80 keV from the combined effects of counter background and the body's contents of $^{40}K$ and radiocaesium:

$$Z_{B1} = Z_2^{3-x} Z_3^{x-2} \qquad \text{(VII-1)}$$

where $Z_2$ is the count rate for the same subject in the range 80–120 keV, $Z_3$ is the

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count rate for the same subject in the range 130–200 keV, and $x$ is a parameter specific to the detector (1.08 and 0.91 for the left and right detectors respectively).

The parameter $x$ was estimated by reference to the spectra of 50 occupationally uncontaminated people of various physiques and with radiocaesium burdens in the range 30–1000 Bq; its validity at higher levels of radiocaesium contamination cannot be presumed. For paired detectors viewing the lungs or liver, the scatter (1 sd) of the observed values of $Z_{B1}$ in these 50 subjects about those predicted from Eq. (VII-1) is about $\pm 375$ counts/3000 s for each detector, or $\pm 530$ counts/3000 s for the pair. Corresponding uncertainties when the detectors view the skull or knees are lower by a factor of about 1.5. For all locations this scatter is about three times that expected from counting statistics.

### VII-1.5. Calibration

For deconvolution purposes the detectors have to be calibrated for each combination of measurement position and deposition site. To simulate deposits in lungs, liver and tracheo bronchial lymph nodes, the thorax section of a phantom (Fig. 14) is employed, with the appropriate organs uniformly labelled in turn with standardized quantities of $^{241}$Am. The thickness and composition of the phantom's anterior chest wall can be varied. In calibrations for skeletal $^{241}$Am, a set of labelled bone phantoms, representing the skull, thorax and knee [VII-3] was borrowed from the New York University Medical Center. The response matrix is given in Table VII-1, for a subject with a 25 mm thick chest wall whose soft tissues consist equally of muscle and adipose tissue. The calibration factors for bone refer to the activity of the entire skeleton, whose $^{241}$Am content is assumed to be distributed as found for Case No. 102 of the US Transuranium Registry [VII-4].

TABLE VII-2. MINIMUM DETECTABLE ACTIVITIES ($3.29\sigma$) OF $^{241}$Am IN LUNGS AND AT SYSTEMIC SITES, VIEWED BY PAIRED 200 mm DIAMETER PHOSWICH DETECTORS

| Deposition site | Region viewed | *MDA* (Bq) |
|---|---|---|
| Lungs | Lungs | 15 |
| Lymph nodes | Lungs | 22 |
| Liver | Liver | 13 |
| Skeleton | Skull | 52 |
| Skeleton | Knees | 30 |

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### VII-1.6. Minimum detectable activities

The minimum detectable activity (*MDA*) of $^{241}$Am in a specific organ will depend on the extent of interference in the response of a detector viewing that organ arising from deposits in other organs. Table VII-2 contains estimates of *MDA* applicable to the notional situation of no such interference, i.e. with the activity confined to a single organ. The basis of these estimates is as follows.

For small deposits of $^{241}$Am, the major uncertainty in assessing the response from the contaminant will arise from the uncertainty in $Z_{B1}$, i.e. in the count rate arising from counter background and body radioactivity other than that due to $^{241}$Am. These uncertainties have been indicated in Section VII-1.4 above for the various detector–subject configurations. They were translated into uncertainties in the estimated organ deposit by reference to the appropriate calibration factors given in Table VII-1. The *MDA*s quoted (Table VII-2) are $3.29\sigma$, i.e. they represent 95% confidence limits and are therefore expressed on a similar basis to that advocated in Annex I (Eq. (I-2)) for the more common situation where the background response is separately measurable.

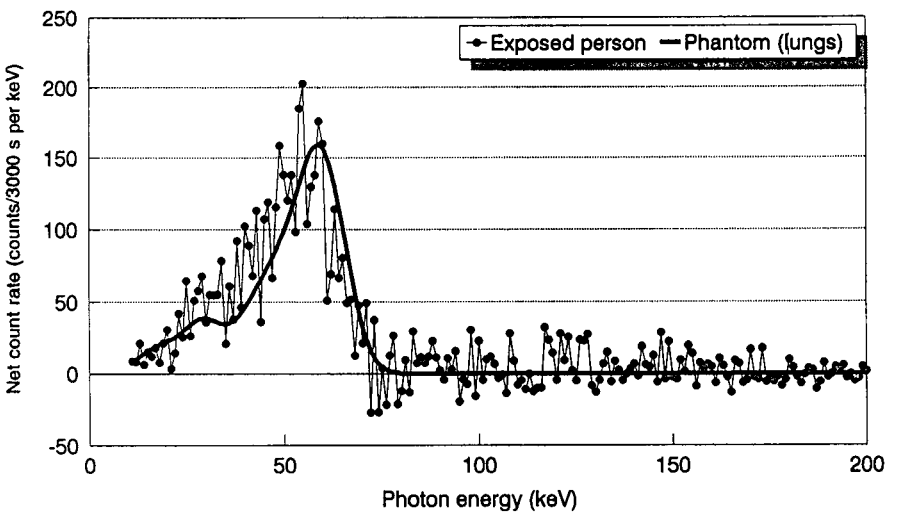


*FIG. VII-2. Points: spectrum of response attributed to $^{241}$Am in the contaminated subject of Fig. VII-1, obtained by subtraction of the spectrum given by the matching unexposed subject. Curve: spectrum of response from $^{241}$Am uniformly deposited in the lungs of a humanoid phantom.*

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### VII-1.7. Example of application

The exposed subject of Fig. VII-1 is believed to have inhaled $^{241}$Am on one or more occasions 1–2 years previously; in this instance a spectrum, also shown in Fig. VII-1, was available from an unexposed subject of similar physique and with roughly the same contents of potassium and radiocaesium as the contaminee. In Fig. VII-2 the response from $^{241}$Am in the exposed subject, obtained from the difference between the two spectra of Fig. VII-1, is compared with that from a suitably scaled spectrum of $^{241}$Am in the lungs of the calibration phantom. A shift in the peak position is apparent; no such shift was observed when comparison was made with a spectrum of $^{241}$Am in the phantom's tracheo bronchial lymph nodes. This was interpreted as indicating that most of the $^{241}$Am present in the upper thorax was deposited in the lymph nodes, and in the ensuing analysis it was assumed that the contamination was confined to lymph nodes, liver and bone.

Although in this instance a suitable 'background' spectrum was available from an uncontaminated subject, these background contributions will be assessed according to Eq. (VII-1), for purposes of illustration. The subject's chest wall thickness was estimated as 25 mm from published correlations of this quantity with weight and height; consequently, the calibration factors in Table VII-1 were adopted.

TABLE VII-3. OBSERVED COUNT RATES (COUNTS/3000 s) OF PAIRED 200 mm DIAMETER PHOSWICH DETECTORS VIEWING FOUR ANATOMICAL REGIONS OF A SUBJECT CONTAMINATED WITH $^{241}$Am[a]

| Region viewed | Count rate (counts/3000 s) | | |
|---|---|---|---|
| | $Z_1$ | $Z_2$ | $Z_3$ |
| Lungs | 23008 ± 152 | 10911 ± 104 | 6132 ± 78 |
| Liver | 25119 ± 158 | 10422 ± 102 | 5562 ± 75 |
| Skull | 7779 ± 88 | 3682 ± 61 | 2199 ± 47 |
| Knees | 9537 ± 98 | 4492 ± 67 | 2980 ± 55 |

[a] Data are given both for $Z_1$, the response in the energy range 20–80 keV, and for $Z_2$ and $Z_3$, the count rates in the two higher energy bands 80–120 keV and 130–200 keV respectively.

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Table VII-3 shows the measured response of the paired detectors in the three relevant energy ranges (Section VII-1.4). Table VII-4 lists the net count rates attributed to $^{241}Am$; these were derived from Table VII-3 by using Equation (VII-1) to calculate the expected $Z_{B1}$ values for an uncontaminated subject and subtracting them from those observed in the exposed subject (Section VII-1.4).

TABLE VII-4. DERIVED COUNT RATES (20–80 keV) ATTRIBUTABLE TO $^{241}Am$ (COUNTS/3000 s) IN THE CONTAMINATED SUBJECT

| Region viewed | Count rate (counts/3000 s) |
|---|---|
| Lungs | 3160 ± 527 |
| Liver | 5473 ± 579 |
| Skull | 1585 ± 347 |
| Knees | 2701 ± 347 |

The efficiencies and count rates in Tables VII-1 and VII-4, with the detectors viewing the skull, lungs and liver, enable three simultaneous linear equations to be constructed to determine the $^{241}Am$ content of the skeleton, lymph nodes and liver; a second, independent value for the skeleton is obtained directly from the response over the knees, since the elements of the response matrix (Table VII-1) are essentially zero for the other sites of deposition. Solution of the equations leads to the following estimates of organ content:

| | |
|---|---|
| Lymph nodes: | 17 ± 8 Bq |
| Liver: | 35 ± 5 Bq |
| Skeleton: | 68 ± 8 Bq |

where the value for skeleton is the mean of consistent estimates obtained with the detectors viewing the skull and the knees.

The uncertainties quoted against these values relate to random effects only. Systematic errors, potentially much greater, may be present because of the various assumptions made in this analysis. Aside from neglect of a possible pulmonary deposit and the presumed uniform deposition pattern in the liver, the elements of the assumed response matrix (Table VII-1) may be in error if the subject's internal anatomy differs from that of the phantom, or if the distribution of his skeletal $^{241}Am$ differs from that in Case No. 102 of the US Transuranium Registry.

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## VII-2. INVESTIGATION OF INTERNAL CONTAMINATION WITH $^{241}$Am WITH MULTIPLE SEMICONDUCTOR DETECTORS

*This section has been compiled from material describing procedures followed at the Battelle Pacific Northwest Laboratories, USA.*

### VII-2.1. Radiological characteristics

See Section VII-1.1.

### VII-2.2. Equipment and measurement procedures

The equipment consists of six high purity planar 51 mm diameter germanium detectors, mounted in paired arrays of three and housed inside a heavily shielded room. Spectra from individual elements of the arrays are accumulated independently. In the recording of spectra of pulmonary contamination the detectors are ordinarily located on the anterior surfaces of the thorax as in Fig. VII-3, with the subject in a semireclining posture [VII-5]. Additional measurements may be made to assess skeletal and/or hepatic deposits of $^{241}$Am and the extent to which these

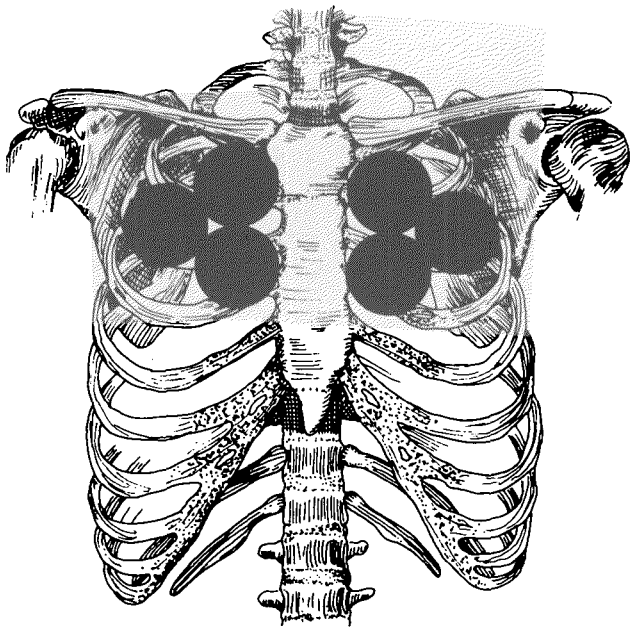

*FIG. VII-3. Anatomical location of detectors in assessment of $^{241}$Am in lungs.*

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may interfere in the assessment of the lung deposit [VII-6]. Skeletal activity is estimated with two of the detectors viewing the skull, with lead shielding worn by the subject to minimize interference from pulmonary deposits. Hepatic $^{241}$Am is investigated with three detectors suitably located against the anterior surface of the lower thorax. Translocation to the tracheo bronchial lymph nodes cannot be quantified without the use of collimated detectors providing adequate discrimination against radiation from residual lung deposits [VII-7].

## VII-2.3. Analysis of spectra

As has been indicated in the main text (Section 4.3), evaluation of the spectral contribution from a specific radioactive contaminant is facilitated by the good energy resolution of semiconductor detectors. The procedure in its simplest form, applied to determination of the peak area due to $^{241}$Am in vivo, is illustrated in Fig. VII-4: a horizontal pedestal, whose level is deduced from the average response in an adjacent energy range above the peak, is subtracted from the observed response in the channels containing the 60 keV peak. This procedure takes account of the underlying

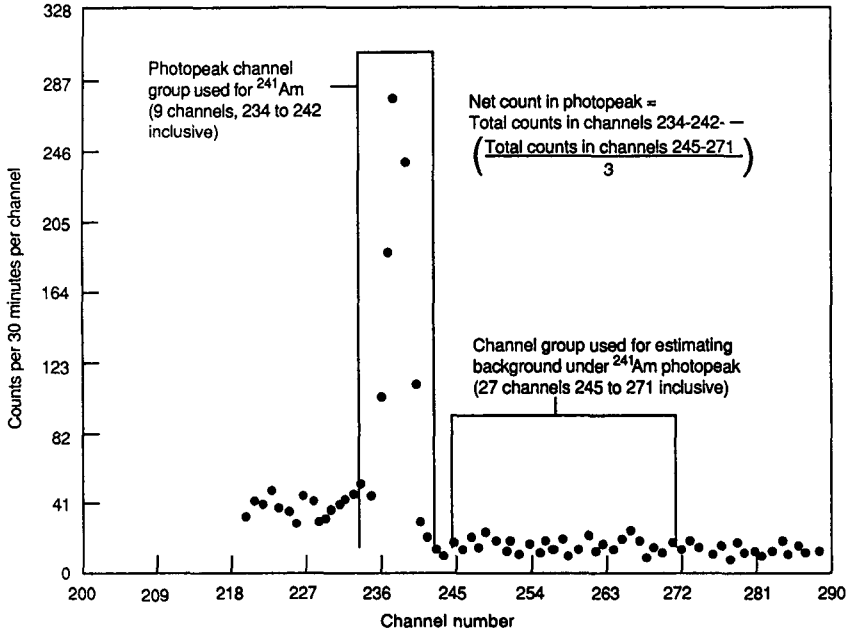


*FIG. VII-4. Determination of 60 keV peak area due to $^{241}$Am.*

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continuum from all sources: ambient background as well as other radioactive deposits in the subject. A separate spectrum of counter background is unnecessary except where this background shows interfering peaks in the energy regions of interest. An alternative procedure [VII-8] is available to estimate the net response objectively in situations where the observed counts are close to background; this relies on a data transformation and smoothing function applied to the adjacent continuum to calculate the interference under the peak.

## VII-2.4. Calibration

A response matrix, analogous to that in Section VII-1.5 is constructed, with each assembly of detectors (arrays of two, three or six) located over the appropriate regions of suitable phantoms.

The six detector array is calibrated for pulmonary deposits by reference to spectra recorded from standardized quantities of $^{241}$Am uniformly distributed in the lungs of a model thorax. The phantom most commonly used is the proprietary article illustrated in Fig. 14. A most useful feature in this application is its ability to simulate chest wall thicknesses ranging from 16 mm to 40 mm or greater. There are facilities also for varying the composition of the chest wall, to allow for differing proportions of adipose tissue, but the effect of these variations is negligible for 60 keV photons. The same phantom, with the $^{241}$Am loaded lungs in situ, is used to assess interference from pulmonary activity when the three detector array is employed to investigate $^{241}$Am in the liver.

Conversely, the three detector array is calibrated for hepatic deposits by recording spectra from known amounts of $^{241}$Am distributed throughout a model liver which can be incorporated into the same phantom thorax; the same combination of labelled liver and phantom thorax makes it possible to assess interference from hepatic $^{241}$Am when the six detector array is used to assess lung deposits.

The paired detectors used to investigate skeletal activity are calibrated through measurements of a phantom head whose skull contains a known quantity of $^{241}$Am [VII-9]; it is assumed that the skull contains 15% of the total skeletal $^{241}$Am. Pulmonary and hepatic deposits do not interfere in the detection of $^{241}$Am in the skull, because of the local shielding provided on the subject's body (Section VII-2.2). However, skeletal $^{241}$Am interferes in the investigation of pulmonary and hepatic $^{241}$Am, and so the three and six detector arrays viewing the liver and lungs are calibrated in terms of skeletal $^{241}$Am with the aid of a thorax phantom containing suitably labelled bone structures. (The skull and thorax phantoms used for these purposes contain parts of a human skeleton with a known deposition of $^{241}$Am; they are shown in Fig. VII-5, together with similarly labelled phantoms representing an arm and a leg.)

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## Description

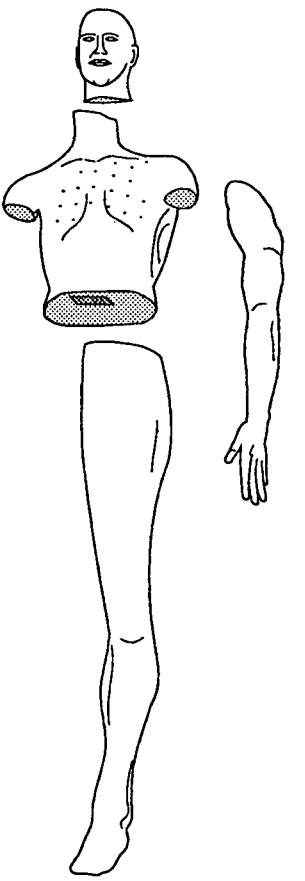
This four-piece phantom incorporates a half skeleton from a voluntary donor to the US Transuranium Registry. The donor incurred an acute accidental exposure to $^{241}$Am via a wound. The other half of the skeleton was radiochemically analysed for $^{241}$Am.

## Physical dimensions

The living and healthy subject had

| | |
|---|---|
| Height | 182 cm |
| Mass | 65 kg |

## Photograph



## Materials

Polyurethane based tissue substitutes.

## Applications

Measurement of the skeletal content of $^{241}$Am; interlaboratory comparisons of measurement techniques. The phantom is available on loan, both in and outside the USA for whole body counter applications.

*FIG. VII-5. Americium calibration phantom for whole body counters.*

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### VII-2.5. Uncertainties in estimates of internal deposits

In a typical case where the contamination produces only a marginal elevation of the background response, the major uncertainties in estimates of the pulmonary deposit arise from (i) statistical variability in the response attributable to $^{241}$Am, and (ii) from calibration errors which may occur in the use of the phantom, with a uniform distribution of $^{241}$Am in its lungs, to represent the true geometrical configuration. Other important uncertainties arise (iii) in adjusting the response recorded over the lungs for interference from systemic $^{241}$Am, and (iv) in assessing the subject's chest wall thickness, which influences the appropriate calibration factor. These uncertainties are summarized in Table VII-5. They cannot usefully be combined in quadrature since some, such as (i), are random in effect and others, like (ii), are systematic in any given subject; a further category, for example (iii), contains errors of both types. At higher levels of internal contamination, the uncertainties associated with counting statistics become relatively less important. Analogous considerations apply in assessing errors in estimates of hepatic $^{241}$Am and total skeletal deposits through detection of 60 keV photons emitted from the skull. In the latter case an important and generally unquantifiable error occurs if the distributions of $^{241}$Am between and within the subject's various bones differ from those applying in the phantom used for calibration.

### VII-2.6. Minimum detectable activity

The *MDA* is indicated here only for the simplified situation in which the contamination is confined to the lungs; higher values will clearly apply if pulmonary $^{241}$Am is to be assessed in the presence of interfering spectral contributions from systemic deposits. A typical uncontaminated subject, whose zero lung deposit is assessed with the array of six planar germanium detectors in a standard count time of 2000 s, will give the following count rates in the two relevant channel ranges indicated in Fig. VII-4:

| | | |
|---|---|---|
| (1) | Channels 245–271 (reference region above 60 keV): | 15.75 ± 0.69 counts/min |
| (2) | Channels 234–242 (containing any response from $^{241}$Am): | 5.25 ± 0.40 counts/min |
| (3) | Channels 234–242 (baseline from scaling (1)): | 5.25 ± 0.23 counts/min |
| (4) | Channels 234–242 (due to $^{241}$Am, from (2) − (3)): | 0 ± 0.46 counts/min |

Hence the uncertainty ($1\sigma$) from counting statistics, associated with an estimated zero lung deposit, is 0.46/0.35 = 1.3 Bq, where 0.35 counts/min per becquerel is the calibration factor appropriate to a subject with 20 mm chest wall thickness. The *MDA*, calculated as $3.29\sigma$, is then 4.3 Bq.

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## VII-2.7. Example of application

This example shows how a lung deposit of $^{241}$Am is calculated when deposits in liver and bone interfere. We assume that a subject whose chest wall thickness is 24 mm gave the following count rates attributable to $^{241}$Am in the counting periods indicated:

| | | |
|---|---|---|
| Six detectors viewing lungs: | (2000 s) | 2.9 counts/min ($C_p$) |
| Two detectors viewing skull: | (3000 s) | 0.73 counts/min ($C_s$) |
| Three detectors viewing liver: | (3000 s) | 1.3 counts/min ($C_h$) |

The first step is to adjust the count rates $C_p$ and $C_h$ for interference from skeletal $^{241}$Am. Previous studies with the labelled bone phantoms (Section VII-2.4) had indicated count rates for these three detector/subject configurations in the ratio 1.4:1:0.42 respectively. Hence the adjusted count rates from the subject become:

$$C'_p = 2.9 - (0.73 \times 1.4) = 1.9 \text{ counts/min}$$
$$C'_h = 1.3 - (0.73 \times 0.42) = 1.0 \text{ counts/min}$$

$C'_p$ still contains a component from hepatic $^{241}$Am and $C'_h$ includes a response from the pulmonary deposit. Derivation of the further adjusted 'true' count rates $C''_p$ and $C''_h$' entails the solution of simultaneous equations. For a chest wall thickness of 24 mm, the relative count rates from the $^{241}$Am labelled liver in the chest phantom were 0.31:1 for the six and three detector arrays viewing the lung and liver respectively. Similarly, the relative count rates given by the two arrays when the $^{241}$Am loaded lungs were substituted were 1:0.042.

TABLE VII-5. ESTIMATED UNCERTAINTIES FOR IN VIVO MEASUREMENTS OF $^{241}$Am IN LUNGS NEAR THE MINIMUM DETECTION LEVEL FOR AN ARRAY OF SIX PLANAR GERMANIUM DETECTORS

| Origin of uncertainty | Estimated magnitude |
|---|---|
| Distribution in lungs | +100%<br>−50% |
| Contributions from systemic deposits | ±50% |
| Measurement of chest wall thickness | ±20% |
| Statistical variability in response attributable to $^{241}$Am at or near detection limit of 4 Bq | ±1.3 Bq |

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It follows that:

$$C'_p = C''_p + 0.31\, C''_h = 1.9 \text{ counts/min}$$

and

$$C'_h = 0.042\, C''_p + C''_h = 1.0 \text{ counts/min}$$

whence:

$$C''_p = 1.6 \text{ counts/min}$$
$$C''_h = 0.9 \text{ counts/min}$$

The calibration factor for the six detector array with a 24 mm thick chest wall is 0.29 counts/min per becquerel of $^{241}$Am in lungs. The estimated lung content is then 1.6/0.29 or 5.5 Bq, subject to the uncertainties indicated in Table VII-5.

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# Annex VIII

# ASSESSMENT OF URANIUM IN THE LUNGS

*This annex has been prepared from material describing procedures adopted by the Swedish Radiation Protection Institute, Stockholm.*

## VIII-1. COMPOSITION AND RADIOLOGICAL CHARACTERISTICS OF NATURAL URANIUM

All three of the long lived isotopes of uranium are alpha emitters and the daughters $^{234}$Th, $^{234}$Pa$^{m}$ and $^{231}$Th are assumed to be in radioactive equilibrium in vivo with their respective parent uranium nuclide.

## VIII-2. EQUIPMENT AND MEASUREMENT PROCEDURES

The approach to assessment depends on the isotopic composition of the uranium. For natural and enriched uranium, detection of the abundant 186 keV photons from $^{235}$U will provide a more sensitive means of assessment, and one which is less prone to calibration errors, than estimates based on quanta in the range 63–93 keV from $^{235}$U and $^{238}$U. This approach may, however, be complicated if backscattered 662 keV photons from $^{137}$Cs in the subject interfere substantially. With substantially depleted uranium, measurements of the 63 and/or 93 keV lines will be required.

Two procedures are available at this laboratory. The first uses paired 127 mm diameter phoswich detectors in contact with the frontal surfaces of the upper thorax, one viewing each lung; the detectors are housed in a shielded room with floor, walls and ceiling made of 40 mm lead with an outer layer of 0.6 m concrete selected for low natural radioactivity. The curved surfaces of the detectors are additionally shielded with 2 mm of tin, surrounded by 20 mm of lead to reduce interference from radioactive deposits outside the lungs.

The second procedure employs three 50 mm diameter × 10 mm thick planar germanium detectors; two of these are sited over the right lung and the third over the left lung. These detectors are installed in a room shielded on all sides by 0.18 m of steel, with an inner lining of 2 mm lead to reduce scattered radiation derived from the subject's internal radionuclides. There is local shielding of the semiconductor detectors, with tin and lead, similar to that provided locally for the phoswich counters. The semiconductor detectors provide less coverage of the lungs than do the phoswiches but their good energy resolution allows low levels of contamination to be identified more reliably, and indeed is essential if attempts are to be made to assess $^{235}$U and $^{238}$U independently.

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Special care was exercised in the selection of shielding and of materials sited close to the detectors. A flow of filtered air is provided to control background variations arising from fluctuating accumulations of natural radon decay products inside the shield. Precautions are sometimes required to reduce interference from radon decay products deposited in the lungs; these require keeping the subject in a low radon environment for one hour prior to measurement.

## VIII-3. ANALYSIS OF SPECTRA

For spectra recorded with phoswich detectors, it is first necessary to subtract a background spectrum, recorded with an inactive phantom substituted for the subject. The relevant energy intervals chosen are 78–106 keV (band A) when the contaminant is depleted uranium or, more commonly, 166–204 keV (band B) when a deposit of natural or enriched uranium is suspected. The response in a third, reference region 216–400 keV (band C) is also computed; this is used as a measure of the scattered photon flux from the body's contents of $^{40}K$ and $^{137}Cs$, whose estimated contributions in bands A and B must be subtracted from the recorded response. These contributions are assessed by reference to the mean relative responses in bands A, B and C found in groups of subjects with no occupational exposure. This procedure may be invalidated if other radionuclides are present or

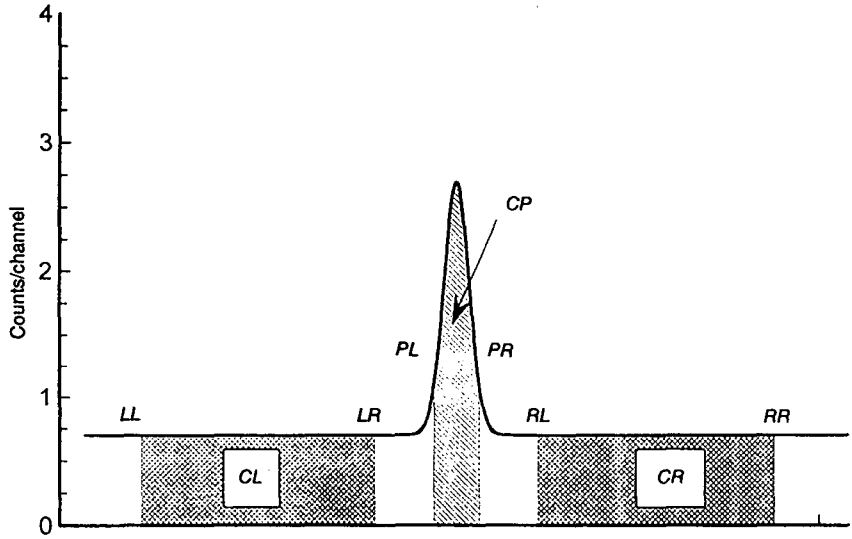


*FIG. VIII-1. Derivation of peak area in spectra obtained with semiconductor detectors: illustration of notations used in Section VIII-6.2.*

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if the ratio $^{137}Cs/^{40}K$ in the subject differs markedly from that in the reference group. In such instances investigations with the germanium detectors would be recommended, when much narrower regions could be chosen for integration and the scatter components could be more reliably estimated from the adjacent continuum.

The assessment of peak areas in spectra recorded with semiconductor detectors involves the subtraction of 'baseline' contributions derived by scaling the response in selected regions of the adjacent continuum (Section VIII-6.2 and Fig. VIII-1).

## VIII-4. CALIBRATION

The calibration factors currently adopted are derived from measurements with the thorax section of a humanoid phantom (Fig. 14) containing lungs loaded uniformly with standardized amounts of natural or enriched uranium. (Previously, the lungs were simulated by plastic bags filled with uranium impregnated sawdust, and these were surrounded by sugar filled bags to represent the other tissues of the thorax.)

The accuracy of the calibration factor may be no better than 40%, because of possible errors relating to the validity of the assumed distribution (uniform) in the lungs, and to the design of the phantom as an adequate anatomical model for the subject concerned.

The calibration factor appropriate to subjects of average physique is 15 counts/s per kilobecquerel of $^{235}U$ in band B for the paired phoswich detectors; for the array of three semiconductor detectors it is 1.59 counts/s per kilobecquerel of $^{235}U$ in the 186 keV peak.

## VIII-5. MINIMUM DETECTABLE ACTIVITY

*MDA*s depend strongly on the presence of other radionuclides in the subject. For a subject containing no radionuclides other than $^{40}K$, $^{137}Cs$ ($<0.3$ kBq) and isotopes of uranium, the *MDA* of $^{235}U$ is 7–14 Bq with the paired phoswich detectors and 4–6 Bq with the array of planar germanium detectors. These estimates are based on $3.3\sigma$, where $\sigma$ is the uncertainty in a single determination of zero activity; in the case of phoswich detectors they include the uncertainty in assessing the scatter contribution from observations in the reference group of uncontaminated subjects.

## VIII-6. EXAMPLE OF APPLICATION

The results to be presented were obtained for the same subject following exposure to enriched uranium, from measurements with each system of detectors.

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TABLE VIII-1. DATA USED IN THE INVESTIGATION WITH PHOSWICH DETECTORS

| | Counts/s ± 1σ | |
|---|---|---|
| | Band B (166–204 keV) | Band C (216–400 keV) |
| Subject | 0.664 ± 0.017 | 1.213 ± 0.022 |
| Background (with phantom) | 0.235 ± 0.002 | 0.476 ± 0.003 |
| Subject minus background | 0.429 ± 0.017 | 0.737 ± 0.022 |

### VIII-6.1. Investigation with phoswich detectors

The data shown in Table VIII-1 were employed.

In a group of 31 subjects without occupational exposure to uranium, the mean relative response in bands B and C was: 0.474 ± 0.033 (1 sd).

Hence the response in band B, in excess of that attributable to $^{40}K$ and $^{137}Cs$, was:

$$[0.429 \pm 0.017] - [(0.474 \pm 0.033) \times (0.737 \pm 0.022)] \text{ counts/s}$$
$$= 0.080 \pm 0.031 \text{ counts/s}$$

The excess is highly significant and, if due wholly to uranium detected with the efficiency indicated in Section VIII-4 above, would correspond to a lung deposit of 5.3 ± 2.2 Bq $^{235}U$.

### VIII-6.2. Investigation with the array of semiconductor detectors

The assessment is to be based on the response at 186 keV from $^{235}U$. Because the background response also shows a small peak at 186 keV from contaminated materials in the detector, the analysis is performed independently on spectra recorded in the presence of the subject and of an inactive phantom, and the net response is derived by difference. The method of analysing each spectrum is illustrated in Fig. VIII-1.

The count peak rate net (*CPRN*) in the 186 keV peak is calculated as:

$$CPRN = CP/T - ((CL + CR)\cdot(PR - PL + 1)/(LR - LL + RR - RL + 2))/T$$

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and the standard deviation from counting statistics is calculated as:

$$\sigma(CPRN) = [(CP/T^2) + ((CL + CR)\cdot(PR - PL + 1)^2)/ (LR - LL + RR - RL + 2)^2/T^2]^{1/2}$$

where

$T$ is the measuring time
$CP$ is the number of pulses in the peak window
$CL$ is the number of pulses in the left window
$CR$ is the number of pulses in the right window
$PL$, $PR$ are the left and right channel numbers in the peak window
$LL$, $LR$ are the left and right channel numbers in the left window
$RL$, $RR$ are the left and right channel numbers in the right window.

With an energy calibration of 0.2 keV/channel:

$PL$, $PR$ = 918, 938
$LL$, $LR$ = 868, 908
$RL$, $RR$ = 940, 962

In a 2400 s measurement with the subject present:

$CP$ = 63 counts; $CL$ = 84 counts; $CR$ = 45 counts

so $CPRN_s$ = 0.00861 counts/s
and $\sigma(CPRN_s)$ = 0.00365 counts/s,
where $CPRN_s$ is the count peak rate net in measurement of subject.

In the 61200 s measurement with the phantom present,

$CP$ = 534 counts; $CL$ = 1104 counts; $CR$ = 516 counts

so $CPRN_b$ = 0.00004 counts/s
and $\sigma(CPRN_b)$ = 0.00043 counts/s,
where $CPRN_b$ is the count peak rate net in measurement of inactive phantom (background measurement).

The net response at 186 keV attributed to $^{235}U$ in the subject is $CPRN_s - CPRN_b = 0.00857 \pm 0.00368$ counts/s. Applying the calibration factor indicated in Section VIII-4 gives an estimated content of $5.4 \pm 2.3$ Bq $^{235}U$, consistent with the result obtained for this subject with the phoswich detectors.

The precision of an estimate for uranium could be improved if additional peaks (Table VIII-2) in the spectrum were evaluated.

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TABLE VIII-2. ISOTOPIC COMPOSITION OF NATURAL URANIUM AND RELEVANT DECAY PROPERTIES OF PARENT AND DAUGHTER NUCLIDES [III-1]

| Nuclide | Half-life (a) | Abundance (wt%) | Specific activity (kBq/g U) | Photon emissions[a] |
|---|---|---|---|---|
| $^{238}U$[b] | $4.5 \times 10^9$ | 99.28 | 12.1 | 63.3[3.8], 92.4[2.7], 92.8[2.7] |
| $^{234}U$ | $2.5 \times 10^5$ | 0.0056 | 12.8 | 53.2[0.12] |
| $^{235}U$ | $7.0 \times 10^8$ | 0.72 | 0.6 | 25.6[14.8][c], 84.2[6.5][c], 93.3[4.3], 143.8[10.5], 163.3[4.7], 185.7[54], 205[4.7] |

[a] Energy (keV), followed by % abundance.
[b] Photon emissions are from $^{234}Th$ daughter.
[c] Emission from $^{231}Th$ daughter.

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