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Measurement of radioactivity in the environment — Soil —

Part 3: Test method of gamma-emitting radionuclides using gamma-ray spectrometry

Mesurage de la radioactivité dans l'environnement — Sol —

Partie 3: Méthode d'essai des radionucléides émetteurs gamma par spectrométrie gamma

Reference number
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ISO copyright office
CP 401 • Ch. de Blandonnet 8
CH-1214 Vernier, Geneva
Phone: +41 22 749 01 11
Email: copyright@iso.org
Website: www.iso.org

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Foreword

ISO (the International Organization for Standardization) is a worldwide federation of national standards bodies (ISO member bodies). The work of preparing International Standards is normally carried out through ISO technical committees. Each member body interested in a subject for which a technical committee has been established has the right to be represented on that committee. International organizations, governmental and non-governmental, in liaison with ISO, also take part in the work. ISO collaborates closely with the International Electrotechnical Commission (IEC) on all matters of electrotechnical standardization.

The procedures used to develop this document and those intended for its further maintenance are described in the ISO/IEC Directives, Part 1. In particular, the different approval criteria needed for the different types of ISO document should be noted. This document was drafted in accordance with the editorial rules of the ISO/IEC Directives, Part 2 (see www.iso.org/directives).

ISO draws attention to the possibility that the implementation of this document may involve the use of (a) patent(s). ISO takes no position concerning the evidence, validity or applicability of any claimed patent rights in respect thereof. As of the date of publication of this document, ISO had not received notice of (a) patent(s) which may be required to implement this document. However, implementers are cautioned that this may not represent the latest information, which may be obtained from the patent database available at www.iso.org/patents. ISO shall not be held responsible for identifying any or all such patent rights.

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For an explanation of the voluntary nature of standards, the meaning of ISO specific terms and expressions related to conformity assessment, as well as information about ISO's adherence to the World Trade Organization (WTO) principles in the Technical Barriers to Trade (TBT), see www.iso.org/iso/foreword.html.

This document was prepared by Technical Committee ISO/TC 85, *Nuclear energy, nuclear technologies, and radiological protection*, Subcommittee SC 2, *Radiological protection*.

This third edition cancels and replaces the second edition (ISO 18589-3:2015), which has been technically revised.

The main change is:

— a correction to Formula (4);

A list of all parts in the ISO 18589 series can be found on the ISO website.

Any feedback or questions on this document should be directed to the user’s national standards body. A complete listing of these bodies can be found at www.iso.org/members.html.

Introduction

Everyone is exposed to natural radiation. The natural sources of radiation are cosmic rays and naturally occurring radioactive substances which exist in the earth and flora and fauna, including the human body. Human activities involving the use of radiation and radioactive substances add to the radiation exposure from this natural exposure. Some of those activities, such as the mining and use of ores containing naturally-occurring radioactive materials (NORM) and the production of energy by burning coal that contains such substances, simply enhance the exposure from natural radiation sources. Nuclear power plants and other nuclear installations use radioactive materials and produce radioactive effluent and waste during operation and decommissioning. The use of radioactive materials in industry, agriculture and research is expanding around the globe.

All these human activities give rise to radiation exposures that are only a small fraction of the global average level of natural exposure. The medical use of radiation is the largest and a growing man-made source of radiation exposure in developed countries. It includes diagnostic radiology, radiotherapy, nuclear medicine and interventional radiology.

Radiation exposure also occurs as a result of occupational activities. It is incurred by workers in industry, medicine and research using radiation or radioactive substances, as well as by passengers and crew during air travel. The average level of occupational exposures is generally below the global average level of natural radiation exposure (see Reference [1]).

As uses of radiation increase, so do the potential health risk and the public's concerns. Thus, all these exposures are regularly assessed in order to:

— improve the understanding of global levels and temporal trends of public and worker exposure;

— evaluate the components of exposure so as to provide a measure of their relative importance;

— identify emerging issues that may warrant more attention and study. While doses to workers are mostly directly measured, doses to the public are usually assessed by indirect methods using the results of radioactivity measurements of waste, effluent and/or environmental samples.

To ensure that the data obtained from radioactivity monitoring programs support their intended use, it is essential that the stakeholders (for example nuclear site operators, regulatory and local authorities) agree on appropriate methods and procedures for obtaining representative samples and for handling, storing, preparing and measuring the test samples. An assessment of the overall measurement uncertainty also needs to be carried out systematically. As reliable, comparable and ‘fit for purpose’ data are an essential requirement for any public health decision based on radioactivity measurements, international standards of tested and validated radionuclide test methods are an important tool for the production of such measurement results. The application of standards serves also to guarantee comparability of the test results over time and between different testing laboratories. Laboratories apply them to demonstrate their technical competences and to complete proficiency tests successfully during interlaboratory comparisons, two prerequisites for obtaining national accreditation.

Today, over a hundred International Standards are available to testing laboratories for measuring radionuclides in different matrices.

Generic standards help testing laboratories to manage the measurement process by setting out the general requirements and methods to calibrate equipment and validate techniques. These standards underpin specific standards which describe the test methods to be performed by staff, for example, for different types of samples. The specific standards cover test methods for:

— naturally-occurring radionuclides (including ^{40}K, ^{3}H, ^{14}C and those originating from the thorium and uranium decay series, in particular ^{226}Ra, ^{228}Ra, ^{234}U, ^{238}U and ^{210}Pb) which can be found in materials from natural sources or can be released from technological processes involving naturally occurring radioactive materials (e.g., the mining and processing of mineral sands or phosphate fertilizer production and use);

— human-made radionuclides, such as transuranium elements (americium, plutonium, neptunium, and curium), ^{3}H, ^{14}C, ^{90}Sr and gamma-ray emitting radionuclides found in waste, liquid and gaseous effluent, in environmental matrices (water, air, soil and biota), in food and in animal feed as a result of authorized releases into the environment, fallout from the explosion in the atmosphere of nuclear devices and fallout from accidents, such as those that occurred in Chernobyl and Fukushima.

The fraction of the background dose rate to man from environmental radiation, mainly gamma radiation, is very variable and depends on factors such as the radioactivity of the local rock and soil, the nature of building materials and the construction of buildings in which people live and work.

A reliable determination of the activity concentration of gamma-ray emitting radionuclides in various matrices is necessary to assess the potential human exposure, to verify compliance with radiation protection and environmental protection regulations or to provide guidance on reducing health risks. Gamma-ray emitting radionuclides are also used as tracers in biology, medicine, physics, chemistry, and engineering. Accurate measurement of the activities of the radionuclides is also needed for national security and in connection with the Non-Proliferation Treaty (NPT).

This document is to be used in the context of a quality assurance management system (ISO/IEC 17025).

ISO 18589 is published in several parts for use jointly or separately according to needs. These parts are complementary and are addressed to those responsible for determining the radioactivity present in soil, bedrocks and ore (NORM or TENORM). The first two parts are general in nature and describe the setting up of programmes and sampling techniques, methods of general processing of samples in the laboratory (ISO 18589-1), the sampling strategy and the soil sampling technique, soil sample handling and preparation (ISO 18589-2). ISO 18589-3, ISO 18589-4 and ISO 18589-5 deal with nuclide-specific test methods to quantify the activity concentration of gamma emitting radionuclides (ISO 18589-3 and ISO 20042), plutonium isotopes (ISO 18589-4) and ^{90}Sr (ISO 18589-5) of soil samples. ISO 18589-6 deals with non-specific measurements to quantify rapidly gross alpha or gross beta activities and ISO 18589-7 describes in situ measurement of gamma-emitting radionuclides.

The test methods described in ISO 18589-3 to ISO 18589-6 can also be used to measure the radionuclides in sludge, sediment, construction material and products following proper sampling procedure.

This document is one of a set of International Standards on measurement of radioactivity in the environment.

Measurement of radioactivity in the environment — Soil —

Part 3:
Test method of gamma-emitting radionuclides using gamma-ray spectrometry

1 Scope

This document specifies the identification and the measurement of the activity in soils of a large number of gamma-emitting radionuclides using gamma spectrometry. This non-destructive method, applicable to large-volume samples (up to about 3 l), covers the determination in a single measurement of all the γ-emitters present for which the photon energy is between 5 keV and 3 MeV.

Generic test method and fundamentals using gamma-ray spectrometry are described in ISO 20042.

This document can be applied by test laboratories performing routine radioactivity measurements as a majority of gamma-emitting radionuclides is characterized by gamma-ray emission between 40 keV and 2 MeV.

The method can be implemented using a germanium or other type of detector with a resolution better than 5 keV.

This document addresses methods and practices for determining gamma-emitting radionuclides activity present in soil, including rock from bedrock and ore, construction materials and products, pottery, etc. This includes such soils and material containing naturally occurring radioactive material (NORM) or those from technological processes involving Technologically Enhanced Naturally Occurring Radioactive Materials (TENORM) (e.g. the mining and processing of mineral sands or phosphate fertilizer production and use) as well as of sludge and sediment. This determination of gamma-emitting radionuclides activity is typically performed for the purpose of radiation protection. It is suitable for the surveillance of the environment and the inspection of a site and allows, in case of accidents, a quick evaluation of gamma activity of soil samples. This might concern soils from gardens, farmland, urban or industrial sites that can contain building materials rubble, as well as soil not affected by human activities.

When the radioactivity characterization of the unsieved material above 200 µm or 250 µm, made of petrographic nature or of anthropogenic origin such as building materials rubble, is required, this material can be crushed in order to obtain a homogeneous sample for testing as described in ISO 18589-2.

2 Normative references

The following documents are referred to in the text in such a way that some or all of their content constitutes requirements of this document. For dated references, only the edition cited applies. For undated references, the latest edition of the referenced document (including any amendments) applies.

ISO 10703, *Water quality — Gamma-ray emitting radionuclides — Test method using high resolution gamma-ray spectrometry*

ISO 11074, *Soil quality — Vocabulary*

ISO/IEC 17025, *General requirements for the competence of testing and calibration laboratories*

ISO 18589-1, *Measurement of radioactivity in the environment — Soil — Part 1: General guidelines and definitions*

ISO 20042, *Measurement of radioactivity — Gamma-ray emitting radionuclides — Generic test method using gamma-ray spectrometry*

ISO 80000-10, *Quantities and units — Part 10: Atomic and nuclear physics*

ISO/IEC Guide 98-1, *Uncertainty of measurement — Part 1: Introduction to the expression of uncertainty in measurement*

3 Terms, definitions and symbols

3.1 Terms and definitions

For the purposes of this document, the terms and definitions given in ISO 10703, ISO 11074, ISO 18589-1, ISO 20042 and ISO 80000-10 apply.

3.2 Symbols

For the purposes of this document, the symbols given in ISO 10703, ISO 11074, ISO 18589-1, ISO 20042 and ISO 80000-10 apply.

Symbol	Meaning	Unit
A	Activity of each radionuclide in calibration source, at the calibration time	Bq
a, a_c	Activity per unit of mass of each radionuclide, without and with corrections	$\mathrm{Bq{\cdot}kg^{-1}}$
ε_E	Efficiency of the detector at energy, E, with the actual measurement geometry	
f_E	Correction factor considering all necessary corrections	
$f_{\mathrm{att}}(E)$	Correction factor for self-attenuation at photon energy E	
$F_{\mathrm{att}}(E)_{\mathrm{sample}}$ $F_{\mathrm{att}}(E)_{\mathrm{standard}}$	Attenuation factor at photon energy E respectively for the sample and the standard	
f_{d}	Correction factor for decay for a reference date	
$f_{\mathrm{cl},E}$	Correction factor for coincidence losses (summing-out)	
$f_{\mathrm{su},E}$	Correction factor for summing-in effects by coincidences	
$f_{\mathrm{dt\ pu},E}$	Correction factor for dead time and pile up	
h	Height of the sample in the container	cm
λ	Decay constant of each radionuclide	$\mathrm{s^{-1}}$
$\mu_{\mathrm{m}}(E)$	Mass attenuation coefficient, at photon energy, E	$\mathrm{cm^2{\cdot}g^{-1}}$
$\mu_{\mathrm{sample}}(E)$, $\mu_{\mathrm{standard}}(E)$	Linear attenuation coefficient at photon energy E respectively for the sample and the standard	$\mathrm{cm^{-1}}$
$n_{\mathrm{N},E}$, $n_{\mathrm{N0},E}$, $n_{\mathrm{Ns},E}$	Number of counts in the net area of the peak, at energy E, in the test sample spectrum, in the background spectrum and in the calibration spectrum, respectively	
$n^{\mathrm{T}}_{\mathrm{N},E}$	Theoretical number of counts in the net area of the peak, at energy E	
$n_{\mathrm{g},E}$, $n_{\mathrm{g0},E}$, $n_{\mathrm{gs},E}$	Number of counts in the gross area of the peak, at energy E, in the test sample spectrum, in the background spectrum and in the calibration spectrum, respectively	
$n_{\mathrm{b},E}$, $n_{\mathrm{b0},E}$, $n_{\mathrm{bs},E}$	Number of counts in the background of the peak, at energy E, in the test sample spectrum in the background spectrum and in the calibration spectrum, respectively	
P_E	Probability of the emission of a gamma-ray with energy E of each radionuclide, per decay	
ρ	Bulk density, in grams per cubic centimetre, of the sample	$\mathrm{g{\cdot}cm^{-3}}$

Symbol	Meaning	Unit
S	Cross-section of surface	
t_g	Test sample spectrum counting time	s
t_0	Background spectrum counting time	s
t_i	Time between the reference time and the start of the measuring time	s
t_S	Calibration spectrum counting time	s
$u(a)$, $u(a_c)$	Standard uncertainty associated with the measurement result (without and with corrections)	Bq·kg^{-1}
u_{rel}	Relative uncertainty	
u_c	Combined uncertainty	
U	Expanded uncertainty calculated with $k = 2$.	Bq·kg^{-1}
x	Unit thickness	cm
X	Thickness of the sample crossed by a photon flux; it also represents the filling height of the sample in the measurement containers	cm
w_i	Mass fraction of element i (no unit)	
a^*, a_c^*	Decision threshold, without and with corrections	Bq·kg^{-1}
$a^\#$, $a_c^\#$	Detection limit, without and with corrections	Bq·kg^{-1}
$a^\triangleleft$, $a^\triangleright$	Lower and upper limits of the probabilistically symmetric coverage interval	Bq·kg^{-1}
$a^<$, $a^>$	Lower and upper limits of the shortest coverage interval	Bq·kg^{-1}
k_p	Quantile of the standardized normal distribution for the probability p (for instance $p = 1 - \alpha$, $1 - \beta$ or $1 - \gamma / 2$)	
k_q	Quantile of the standardized normal distribution for the probability q (for instance $q = 1 - \omega \cdot \gamma / 2$)	
ϕ	Distribution function of the standardized normal distribution	
ω	Auxiliary quantity	

4 Principle

Gamma-rays produce electron-hole pairs when interacting with matter. When a voltage is applied across a semiconductor detector, these electron hole-pairs are, after proper amplification, detected as current pulses. The pulse height is related to the energy absorbed from the gamma-photon or photons in the resolving time of the detector and electronics. By discriminating between the height of the pulses, a gamma-ray pulse height spectrum is obtained. After analysis of the spectrum, the various peaks are assigned to the radionuclides which emitted the corresponding gamma rays using an established detector energy calibration curve.

The activity of gamma-emitting radionuclides present in the soil samples is calculated using the established energy-dependent detector efficiency curve. These techniques allow the identification and the quantification of the radionuclides[2][3].

The nature and geometry of the detectors as well as the samples call for appropriate energy and efficiency calibrations[2][3]. True coincidence summing effects need to be considered, in particular when analysing samples with high activity levels or in applications with high detections efficiencies (e.g. when using Marinelli type containers or well-type detectors) or when the sample container is placed directly on the detector (see 10.1.5).

Fundamentals to gamma-spectrometry, definition and terms and generic description of gamma-spectrometry equipment are summarized in ISO 20042.

NOTE This part deals exclusively with gamma spectrometry using semiconductor detectors.

5 Reference sources

5.1 Source(s) for energy calibration

The energy calibration of the spectrometer shall be established using one or more sources containing radionuclides that emit gamma-rays that cover the energy range of interest. Sources can be of any form but the dead time of the spectrometer for the measurements shall be such that the full energy peak shape is not distorted and pulse pile-up avoided.

The number of peaks (full energy peaks) required depends on the order of polynomial needed for the energy vs. channel calibration curve; normally 5 to 10 peaks should be sufficient. Sources containing long-lived radionuclides (for example ^{152}Eu, ^{241}Am, ^{60}Co or ^{137}Cs) are recommended for this purpose. For periodical checks of the energy calibration, a smaller number of energy peaks may be used.

5.2 Reference source(s) for efficiency calibration

5.2.1 General

The general method to calibrate the spectrometer is to establish the detection efficiency as a function of energy for a defined geometry and energy range. One or more reference sources containing single or multiple radionuclides may be used for this purpose. The activity or emission rates of the radionuclide(s) in the reference source(s) shall be traceable to national or international standards.

The energies of the emitted gamma-rays shall be distributed over the entire energy range of interest, in such a way that the energy-dependent efficiency of the spectrometer for the specific geometry can be determined in a sufficiently accurate way. For most purposes, the accuracy is sufficient for an energy range of 60 keV to 1 836 keV if a multi-radionuclide source is used containing all or most of the following radionuclides: ^{241}Am, ^{109}Cd, ^{57}Co, ^{139}Ce, ^{203}Hg, ^{51}Cr, ^{113}Sn, ^{85}Sr, ^{137}Cs, ^{54}Mn, ^{59}Fe, ^{60}Co, ^{65}Zn or ^{88}Y.

For determining the activity of radionuclides emitting gamma-ray or X–rays in the energy region less than 60 keV, the spectrometry system can be calibrated using a reference source containing the radionuclides of interest.

It may be necessary to take into account true coincidence summing corrections for the calibration radionuclides (for example ^{60}Co and ^{88}Y).

5.2.2 Reference sources for laboratory gamma spectrometry systems

Reference sources for laboratory gamma spectrometry systems shall match, as closely as possible, the geometry, density and matrix composition of the samples to be measured. Reference sources may be prepared from standardized solutions or purchased as sealed sources. Only standardized solutions or reference sources that are traceable to national or international primary standards of radioactivity shall be used.

If no reference materials are available to match the samples, correction factors shall be calculated, documented and be applied to results from the measurements to take into account differences in detection efficiency due to geometry, density and matrix effects.

NOTE Reference material used for calibration should be prepared according to ISO 17034.

If a reference source is prepared by dilution from a standardized solution, the supplier's recommendation on the chemical form of the diluent shall be followed. It is also recommended that the dispensing process includes checks for possible losses of active material and on the accuracy of dispensing (for example gravimetric, volumetric and radiometric techniques should be used and cross-checked).

For this purpose, a calibration source should have the same physical and chemical properties as the sample. It might, for instance, be produced by spiking an appropriate sample of soil. In this case, it is essential to ensure the homogeneity of the spiking soil.

5.2.3 Reference sources used with numerical methods

Reference sources for gamma-ray spectrometry systems based on numerical models shall be used following the manufacturer's recommendations (see 9.3.2). The activity or the emission rates of the reference sources shall be traceable to national or international standards.

6 Gamma spectrometry equipment

6.1 General description

The operation of the measurement system is as follows: in semi-conductor detectors, freed charge (the positive and negative charge carriers, holes and electrons) is generated by the interaction of ionising radiation with the detector material (through the photoelectric effect, the Compton effect or pair production). A high-voltage supply applies a bias voltage to the detector crystal resulting in an electric field. The freed charge is accelerated by the electric field towards the detector electrodes. The collected charge is converted into an output voltage pulse by a preamplifier and the output pulse is shaped and amplified by the main amplifier.

Two types of electronic systems can be used to process the signal from the detector preamplifier; an analogue amplifier combined with digital analogue converter (ADC), or a digital signal processor (DSP) system. Both systems convert the pulse amplitude and the pulse-height histogram (spectrum) is stored using a multichannel analyser (MCA). The height of the pulse is proportional to the amount of freed charge and hence to the energy of the ionising radiation striking the detector.

The spectrum stored by the MCA shows a set of peaks (full energy peaks) superimposed on a background continuum from scattered radiation. The full energy peaks are approximately Gaussian in shape. The channel number of the peak centroid depends on the energy of the photon detected. The net full energy peak area is proportional to the number of photons of that energy that have interacted with the detector during the counting period (corrected for dead time). The net full energy peak area is normally determined in the analysis software package by one of two different techniques – summation or fitting.

For laboratory use, the spectrometer should be located in a facility with stable temperature following the manufacturer recommendations. It should be noted that changes in temperature can affect the amplifier gain, changing the energy calibration substantially.

The apparatus shall consist of the following necessary parts from 6.2 to 6.8.

6.2 Detector types

The three main geometries of germanium or other type of detectors available are planar, coaxial and well-type. Each has specific advantages depending on the circumstances. Coaxial detectors are generally used with large volume samples, whereas the well-type detectors are most efficient for small volume samples. Planar detectors can be useful for detecting photons with energies below 200 keV as they can have better energy resolution than coaxial detectors at these energies. More detailed information on the detectors is given in ISO 20042:2019, Table D.1.

Microphonics phenomena can result in an increase in the Full Width at Half Maximum (FWHM) of the full energy peak. It may be necessary to place the detector on an anti-vibration mat.

Depending on the required accuracy and the desired detection limit, it is generally necessary to use high-quality detectors whose energy resolution is less than 2,2 keV (for the ^{60}Co peak at 1 332 keV) and with a peak/Compton ratio between 50 and 80 for ^{137}Cs (see IEC 61452[6]).

Some natural radionuclides (e.g. ^{210}Pb and ^{238}U through ^{234}Th) can be measured only through gamma lines in the energy range below 100 keV. In this case, the use of an N-type detector is recommended. Low-energy, low-level detectors offered by manufacturers have been optimized for this purpose and can additionally be used in other areas of environmental monitoring (e.g. for measurements of ^{129}I and ^{241}Am in samples from the vicinity of nuclear facilities).

6.3 High voltage power supply

When in operation, the power supply should be free from spikes and dropouts, or provided with a power conditioner and/or uninterruptible power supply. Instrument earth connections should be isolated and connected to local ground at one point only to avoid ground loops that may affect the detector resolution.

WARNING — Take necessary safety precautions according to the manufacturer's instructions.

6.4 Preamplifier

The preamplifier strongly influences the quality of the entire measuring system, as both noise and energy resolution depend on the characteristics of the preamplifier.

NOTE Usually the preamplifier is located very close to germanium crystal. Cooling the input stage (FET) of the preamplifier decreases the noise level and improves the energy resolution.

6.5 Cryostat or electric cooler

The cryostat or the electric cooler should be capable of keeping the detector close to the temperature of liquid nitrogen. Operation at a low temperature is required to reduce the leakage current and electronic noise level of the detector and preamplifier; it is recommended that an automatic switch off and an alarm signal be installed which are activated in the case of an increase in the temperature of the detector (e.g. caused by malfunctioning of the cryostat or loss of liquid nitrogen).

A high purity germanium detector may be stored at room temperature, however, it shall be cooled before the bias voltage is applied.

6.6 Shielding

The detector shall be shielded from all sides (including the bottom) with lead, to reduce background signals originating mainly from naturally occurring radionuclides. If measurements in the energy region 40 keV < E < 100 keV are to be made, the internal casing should consist of three successive layers of cadmium/tin, copper and polymethylmethacrylate to achieve a low and constant background by attenuating the X-ray produced in the shielding. Too large a thickness of Cu can increase the background by bremsstrahlung between 100 keV and 200 keV.

Shielding is important to reduce background levels, especially if low activity levels shall be measured. The following measures can be taken to improve the performance of the shield:

- use low ^{210}Pb activity lead;
- limit the back scattering effect by not having shielding close to the detector if at all possible;
- carefully choose ventilation conditions, air filtration and the construction and other materials of the system, in order to reduce ambient radioactivity and achieve low levels of background radiation. Reduction of the radon inside the shield is desirable for measuring low-activity samples. This may be achieved by displacing the air inside the shield with nitrogen boiled off from the cooling Dewar.

An active shielding can also be used (such as cosmic veto or anti-compton systems).

6.7 Analogue or digital acquisition electronics

6.7.1 General

Two types of electronics can be used to process the signal from the detector preamplifier. Either an analogue amplifier combined with a digital analogue converter or a digital signal processor (DSP) system. Regardless of the system used, it shall have a good linearity to obtain a linear pulse conversion for the 40 keV to 2 MeV energy range regardless of the gain applied to the signal. Similarly, the conversion system shall allow to work on a minimum of 4 096 channels in order to obtain a good energy resolution which assists the discrimination of peaks and the deconvolution of multiplets.

6.7.2 Analogue electronic

Analogue amplifier-based systems should have an adjustable Gaussian pulse shaping system. An adjustment approximately between 4 µs and 8 µs is recommended to obtain the best resolution (FWHM). It shall also have a pole-zero setting and an automatic mechanism for restoring the continuous component. A stacking rejection device is recommended for high count rates (>5 000 s^{-1}).

An analogue amplifier is usually associated with a digital analogue converter, which converts the pulses coming from the amplifier according to their amplitude.

The data thus converted is stored in a MCA (multi channels analyser), itself connected to a computer. The PC software to view and save the spectrum reads the MCA memory. These modules can be separated in a NIM-type drawer system or integrated into a single unit.

6.7.3 Digital electronic DSP

Digital systems are generally integrated into a single unit (referred to as an MCA) comprising signal formatting electronics and scanning and storage electronics. The shaping circuit transforms the input pulses to obtain trapezoidal pulses whose characteristics can be adjusted using “rise time” and “flat top” parameters. These parameters should be adjusted to obtain the best resolution (FWHM) depending on the type of detector used and the expected pulse flow. Manufacturers generally indicate the best settings to use. This system shall also have a pole-zero setting.

6.8 Computer, including peripherical devices and software

The computer, in combination with the available hardware and software should be able to

- read the data from the MCA,
- reproduce these data on a video display, a plotter or a printer and store them,
- determine the relation between channel number and corresponding energy over the entire energy range to be studied, by making use of the appropriate reference source, energy calibration,
- determine the energy-dependent efficiency over the entire energy range to be studied, by making use of the appropriate reference source(s),
- detect peaks, determine the characteristics of the detected peaks such as the centroid, the full peak width at half maximum height, the number of net counts collected under the peak, and determine the uncertainty of this number,
- identify the radionuclides responsible for the observed full energy peaks by making use of radionuclide references (see Clause 7),
- calculate the activity concentration of the respective radionuclides on the basis of the number of counts, the counting time, the counting efficiency and the data given in radionuclide references (see Clause 10),
- calculate the combined uncertainty of the activity concentration of the identified radionuclides, and
- calculate the detection limit and the decision threshold of radionuclides to be measured but not found in the sample.

It is recommended that the results of the computer analysis of the spectrum be visually checked regularly for obvious anomalies or errors. To check the performance of the apparatus, the use of a laboratory standard is recommended. Participation in intercomparison exercises also helps to test the performance of the apparatus and analysis.

Whenever necessary, calculations and identification may be performed manually.

7 Nuclear decay data

Nuclear decay data (half-lives, gamma-ray energy and emission probabilities) are available from the scientific literature, databases and other publications. There are often differences between the values depending on the original data used and the evaluation method.

To ensure consistent results from gamma-ray spectrometry measurements, nuclear decay data used for instrument calibration or for estimating the activity content of samples, should therefore be taken from the decay data evaluation project (http://www.lnhb.fr/nuclear-data/). If no evaluation of the nuclear decay data of the radionuclide of interest is available in this database, the following databases should be consulted:

- Joint Evaluated Fission and Fusion (JEFF) Library (https://www.oecd-nea.org/dbdata/);
- National Nuclear Data Center database (https://www.nndc.bnl.gov/);
- Evaluated Nuclear Data File (ENDF) (https://www-nds.iaea.org/exfor/endf.htm).

8 Sample container

Measuring gamma radioactivity in soils requires sample containers that are suited to gamma spectrometry with the following recommended characteristics:

- be made of materials with low absorption of gamma radiation;
- have volumes adapted to the shape of the detector for maximum efficiency;
- be watertight and not react with the sample constituents;
- have a wide-necked, airtight opening to facilitate filling;
- be unbreakable.

In order to verify easily that the content of the container conforms to the standard counting geometry, a transparent container with a mark to check the filling level can be selected.

9 Procedure

9.1 Packaging of samples for measuring purposes

The soil samples packaged for gamma spectrometry measurements are usually dried, crushed, and homogenized in accordance with ISO 18589-2.

The procedure shall be carried out as follows.

a) Choose the container that is best suited to the volume of the sample so as to measure as much material as possible. To decrease self-absorption effects, the height of the contents should be minimized. Further information is given in Reference [5].

b) Fill the container to the level of the volume mark. A mechanical filling device (for example, a vibrating table) can be used to pack the sample to avoid any future losses in volume.

NOTE Compaction of the sample is not necessary if the bulk density is corrected by the analysis of the spectra.

c) Note the sample mass. This information is useful when using the measurements to express the result as specific activity and when carrying out self-absorption corrections.

d) Visually check the upper level of the sample and make sure that it is horizontal before measuring. Where applicable, add more material to the sample until the mark has been reached and adjust the noted sample mass accordingly.

e) Hermetically seal the container if volatile or natural radionuclides are being measured.

NOTE When measuring ^{226}Ra through the short-lived decay products of ^{222}Rn, it is to consider that radioactive equilibrium between ^{226}Ra and ^{222}Rn might be is disturbed because of inhomogeneous distribution of the ^{222}Rn and its daughter radionuclides in the sample. A further sample treatment procedure is recommended.

f) Clean the outside of the container to remove potential contamination due to the filling process.

If measurements are required quickly, the processing method described in ISO 18589-2 can be ignored. This shall be mentioned in the test report and the results cannot be expressed in becquerels per kilogram of dry soil.

It is recommended to achieve the radioactive equilibrium of short-lived decay products to analyse ^{226}Ra under the best conditions. When measuring ^{226}Ra through the short-lived decay products of ^{222}Rn, the sealed container shall be stored long enough (at least 23 d) to allow radioactive equilibrium to be reached between ^{226}Ra and ^{222}Rn.

9.2 Laboratory background level

As some radionuclides found in the soil (see Annex A) are the same as in building materials, the detector and sample shall be adequately shielded against natural background radiation. Frequently, it is sufficient to shield the detector in a 10 cm thick, low-background lead case wall. Reduction of radon inside the shield is desirable. Further information is given in References [2] and [3] or ISO 20042.

The natural radionuclides and their decay products occur widely and with large concentration ranges in floors, walls, ceilings, the air of the measuring rooms and in the materials of which detectors and shielding are made.

There are isotopes of the decay chain of the gas radon, whose emanation from the materials surrounding the measuring instruments depends on various physical parameters. Thus, large fluctuations in the concentration of radon and of the decay products can occur in room air and in the air of the detector shielding. This is a particular problem in basements of old buildings with defective floors. Then background of radon daughters in the background can vary depending on the ventilation. If ventilation is shut down during the weekend, the background can rise and this should be taken into account when collecting background and during sample measurement

The background of the measuring instruments shall be kept as low as possible and, in particular, as stable as possible by appropriate measures. This includes vacuuming the shielding and removing the dust by filtration. Frequent measurements of the background level permit the verification of its stability. This is necessary because the peaks of the background spectrum shall be subtracted from those of a sample spectrum.

The spectrum used for background subtraction should be collected for the same, or longer, time than the sample spectrum

9.3 Calibration

9.3.1 Energy calibration

For reliable radionuclide identification, the energy calibration should be accurate to better than 0,2 keV over the energy range of interest. The energy calibration of the spectrometer shall therefore be established using one or more sources containing radionuclides that emit gamma-rays that cover the energy range of interest. This calibration establishes the relationship between the channel number of the photopeak centroid and the known energy of the photons. This task is normally carried out using commercially-available software to determine the photopeak centroid (channel) and to match this to the energy (keV). The software also determines the full width half maximum of the full energy peaks. This information is used by the photopeak search algorithm and for the calculation of characteristic limits.

The count rate in the detector shall be such that the full energy peak shape is not distorted due to pulse pile-up; a dead time less than 10 % normally suffices. For a germanium spectrometer, the full energy peak position should be determined with an accuracy of better than 0,2 keV.

A calibration curve (energy vs channel) should be calculated by least-squares fitting, so that photopeak energies may be interpolated. A linear or quadratic function is normally sufficient but higher order polynomials may be required. A similar process should be used to determine the full width half maximum as a function of energy.

Care shall be taken when extrapolating the energy calibration curve. Photopeaks in the background spectrum or X-rays should be used to check the accuracy of the extrapolated curve, and the extrapolation should be plotted to check for large deviations from a linear function due to quadratic or higher order components.

9.3.2 Efficiency calibration

The following approaches may be considered for determining the detection efficiency:

- direct comparison with a reference source of the same radionuclide in the same matrix and geometry. This approach is appropriate if the aim of the measurement is to determine the activity of a single radionuclide or if the photon energy is in an energy region that is difficult to calibrate (typically 20 keV to 100 keV). It may also be required if true coincidence summing corrections are significant (for example, when measuring a gamma-ray emitting radionuclide with a complex decay scheme using a high efficiency detector such as a well detector);
- measurement of the full energy peak detection efficiency as a function of energy;
- calculation of the full energy peak detection efficiency as a function of energy by Monte Carlo simulation or other modelling technique. Such numerical models are sensitive to input parameters such as the detector dimensions, and therefore shall be checked using at least one reference source containing radionuclides that emit gamma-rays covering the energy range of interest. If a discrepancy is found between efficiency calculated using the model and the reference source, the discrepancy shall be investigated and corrections applied to the numerical model.

The same algorithm for analysis of the spectrum shall be used for both calibration and sample measurement.

The detection efficiency is affected by the following factors:

- the detector;
- the geometry of the sample with respect to the detector (solid angle);
- the density of the sample and the sample container characteristics;
- the sample mass and chemical composition;
- the heterogeneity of the sample matrix with respect to activity, density and chemical composition.

When any of these factors is changed, the detection efficiency shall be re-evaluated for the new conditions.

If no numerical model is used, the test sample measurement shall be performed with comparable measuring conditions as used for calibrating the gamma-ray spectrometry system. In particular, the measurement geometry, the position of the reference source in relation to the detector and the test sample and reference source matrices should be identical. Corrections should be applied to account for differences in the matrix and other factors that can affect the detection efficiency, as appropriate. If the efficiency calibration method uses a numerical model, the parameters used to describe the sample shall match the item being measured.

In most practical measurements, the instrument dead time for the calibration measurements is higher than for the sample measurements. The activity of the efficiency calibration source shall be such that count rate does not distort the full energy peak shape and that an accurate dead-time correction is applied by the analysis software; a dead time of less than 5 % normally suffices. Higher dead times may be used but evidence of the accuracy of the dead-time correction shall be required.

The full energy peaks in the spectrum shall contain sufficient counts such that the uncertainty in the full energy peak area is small in comparison with other components of uncertainty. A full energy peak area of 10 000 counts is normally sufficient. The full energy peak shall be discarded from the efficiency calibration data set if visual inspection shows that it is not Gaussian in shape.

With these conditions, the efficiency at energy E shall be calculated as given in Formula (1):

$$\varepsilon_E = \frac{n_{\mathrm{Ns},E} / t_\mathrm{s}}{A \cdot P_E} \tag{1}$$

NOTE Formula (1) assumes negligible decay of the calibration source during the acquisition time.

For an undisturbed peak at an energy E, the count, $n_{\mathrm{Ns},E}$, in the net-peak area of a γ-spectrum is calculated as given in Formula (2):

$$n_{\mathrm{Ns},E} = n_{\mathrm{gs},E} - n_{\mathrm{bs},E} \tag{2}$$

When the physical and chemical nature of the sample (chemical composition, bulk density) is different from the conditions of the efficiency calibration, a correction for the self-absorption of gamma radiation should be applied.

The efficiency for each specific radionuclide i, at energy E, is determined using Formula (3):

$$\varepsilon_{i,E} = \frac{n_{\mathrm{Ns},E} / t_\mathrm{s}}{A_i \cdot P_E} \tag{3}$$

The detection efficiency as a function of energy should be calculated and, if required, a function fitted to enable detection efficiency to be interpolated. The detection efficiency is a varying function of energy and software packages offer a choice of different functions. The selection of the fit function shall be based on an evaluation of the goodness of fit parameter (chi-squared) and visual inspection of the function to confirm that the fitted function does not deviate from a smooth curve due to using too many variable parameters. Extrapolation of the efficiency calibration function to energies below the range of the calibration points shall not be used. Extrapolation of the efficiency calibration function to energies above the range of the calibration points is not recommended unless supported by other evidence such as modelling.

Further information is given in IEC 61452[6], ISO 10703, References [7] and [8] and ISO 20042.

9.4 Correction required for the measurements of natural radionuclides

If activities of natural radionuclides in the soil are being measured, the areas of full-energy peaks used for evaluating their activities shall be corrected for the background contribution of those same radionuclides inside the detector shielding, taking into account potential differences of the duration of the sample and background measurements.

Special advice to take into account during the measurement of natural radionuclides in soil and information on spectroscopic interferences is given in Annex A.

The gamma ray of the radionuclides in the background and/or of natural radionuclides inside the sample can also interfere with measurements of artificial radionuclides and can require appropriate corrections.

9.5 Quality control

Quality control operations shall meet the requirements of ISO/IEC 17025. It is necessary to regularly check consistency of the detector efficiency, FWHM, energy calibration and background.

10 Expression of results

10.1 Calculation of the activity per unit of mass

10.1.1 General

The activity per unit of mass, a of each radionuclide present in the sample is obtained from the net count, $n_{\mathrm{N},E}$, from the peak of an individual γ-line without interference using Formula (4):

$$a = \frac{n_{\mathrm{N},E} / t_\mathrm{g}}{P_E \cdot \varepsilon_E \cdot m \cdot f_E} \tag{4}$$

where f_E is the correction factor considering all necessary corrections according to Formula (5).

$$f_E = f_d \cdot f_\mathrm{att}(E)\, f_{\mathrm{cl},E} \cdot f_{\mathrm{su},E} \cdot f_{\mathrm{dt\ pu},E} \tag{5}$$

For a peak with energy, E, and no interfering lines, the count, $n_{\mathrm{N},E}$, in the net-peak area of a γ-spectrum is calculated by Formula (6):

$$n_{\mathrm{N},E} = n_{\mathrm{g},E} - n_{\mathrm{b},E} \tag{6}$$

Thus, Formula (4) can be expressed as given in Formula (7):

$$a = \frac{n_{\mathrm{N},E} / t_\mathrm{g}}{P_E \cdot \varepsilon_E \cdot m \cdot f_E} = \frac{n_{\mathrm{g},E} - n_{\mathrm{b},E}}{P_E \cdot \varepsilon_E \cdot m \cdot f_E \cdot t_\mathrm{g}} = \left(n_{\mathrm{g},E} - n_{\mathrm{b},E}\right) \cdot w / t_\mathrm{g} \quad \text{with } w = \frac{1}{P_E \cdot \varepsilon_E \cdot m \cdot f_E} \tag{7}$$

If the net-peak area, $n_{\mathrm{N},E}$, is obtained by unfolding of a multiplet Formula (6) is valid; but, special care is needed in calculating the uncertainties according to 10.2.

If a peak is affected by an interfering γ-line of another radionuclide and cannot be resolved by unfolding methods due to the limited resolution of the detector, and if the contribution of the interfering radionuclide can be estimated from another γ-line of the interfering radionuclide, the procedure described in 10.6 should be applied.

For nuclides characterized by more than one line, the activity can be computed using several peaks taking into account the known branching ratios described by their decay scheme and the efficiency curve.

10.1.2 Dead time and pile up corrections (see ISO 20042)

The pulse processing electronics and analysis software normally apply a correction for the dead time introduced by the electronics. However, the analogue pulse may extend beyond this time period; if a second pulse arrives on the tail of the pulse, the pulse height may be distorted and a pulse may be lost from the full energy peak in the spectrum.

The effect may be seen in the spectrum as increased full energy peak widths (FWHM) and low-energy tails. The effect shall not be neglected for low-activity measurements as it may be significant for the reference source used for calibration.

The effect depends on the instrument and electronics used.

An additional correction may be needed if the dead time changes significantly during the measurement time (for example when measuring high-activity short half-life samples). Special dead-time correction systems are needed in this case[9].

Low count rates (less than a few thousand pulses per second, equivalent to dead times of a few percent) should be used when possible, by choosing appropriate sample sizes and geometry.

Short shaping times may be selected on the main amplifier to reduce pulse lengths. However, this may have the effect of increasing the FHWM of full energy peaks.

Care shall be taken to ensure that the pole-zero adjustment has been made as an incorrect setting may affect the magnitude of this effect.

For dead times less than 10 %, the correction for pile-up is typically less than 1 %, though the magnitude of the correction depends on the spectrometer type and settings. If higher dead times are used, a correction should be applied as described in ISO 20042.

10.1.3 Decay corrections

Depending on the half-life of the radionuclide being measured, the activity per unit of mass shall be corrected by f_d. To take into account the radioactive decay during the counting time and during the time between the reference instant (t = 0) and the measuring instant ($t = t_i$), f_d shall be calculated by Formula (8):

$$f_\mathrm{d}^{-1} = \mathrm{e}^{\lambda \cdot t_i} \cdot \left[\frac{\lambda \cdot t_\mathrm{g}}{1 - \mathrm{e}^{-\lambda \cdot t_\mathrm{g}}} \right] \tag{8}$$

10.1.4 Self-absorption correction

Measurement of radioactivity in soils by gamma spectrometry may involve a sample with a different matrix from that of the source used for the yield calibration. In this case, a correction factor can be applied to the result obtained depending on the type of sample, the volume of the geometry and the energy of the measured radiations. The lower the radiation energy, the greater the correction factor.

Table 1 — Examples of correction factor, f_att, values for a soil with ρ = 1,2 g/cm^3

	Sample filling height cm			
Energy keV	0,9	1,5	3,5	5,8
46	**0,86**	**0,80**	**0,68**	**0,59**
59	0,92	**0,88**	**0,81**	**0,76**
100	0,97	0,96	0,93	0,91

It appears that the measurement at low energy (E < 100 keV) in a large container induces corrections of more than 10 % up to 25 % for the measurement of ^{241}Am at 59,5 keV and 40 % for the measurement of ^{210}Pb on a geometry of the order of 6 cm in height.

For measurements of emitters above 100 keV, the absence of self-absorption correction can be taken into account by increasing the uncertainties, of the order of 10 % in the example cited.

Different techniques can be used to determine this correction factor:

1) Measurement of the linear attenuation coefficient μ by transmission (collimated flux) and use of calculation (Formula 9 and Annex B) in the case of measurements at a distance from the detector (or with an equivalent thickness for measurements in contact).

2) Measurement of the linear attenuation coefficient μ by transmission (collimated flux) and use of a calculation tool (Monte Carlo) type GESPECOR, EFFTRAN, GEANT.... In this case specify the associated uncertainty calculation.

3) Use of spiked samples (choosing this method requires ensuring that the spiking is homogeneous).

4) Use of Monte-Carlo simulation (GESPECOR, MCNP, etc.) from the elementary composition of the sample. This requires an adapted qualitative and quantitative measurement of this elementary composition with an associated uncertainty balance. The variability in composition from one sample to another should be taken into account. Indeed, the mass attenuation coefficient can vary from one soil to another from 0,28 to 0,57 cm^2/g at 46,5 keV, 0,21 to 0,37 cm^2/g at 59,5 keV and again from 0,15 to 0,21 cm^2/g at 100 keV.

For cylindrical sample containers at the detector, the value of the attenuation factor for a specific matrix $F_\text{att}(E)$, can be estimated using Formula (9) (see Annex B):

$$F_\text{att}(E)=\frac{1-\mathrm{e}^{-\mu_\text{m}(E)\cdot\rho\cdot X}}{\mu_\text{m}(E)\cdot\rho\cdot X} \tag{9}$$

The mass attenuation coefficient $\mu_m(E)$ depends on the energy of the photon and the chemical composition of the sample and expresses the exponential decrease in photon flux density with distance.

X is the filling height of the test sample in the container (cm) if the sample is measured at a distance from the detector.

The correction factor due to the self-attenuation to be applied to the detection efficiency is obtained by using Formula (10):

$$f_\text{att}(E)=\frac{F_\text{att}(E)_\text{sample}}{F_\text{att}(E)_\text{standard}} \tag{10}$$

NOTE This correction is only strictly valid when the sample is at a certain distance from the detector. For the case where the sample is measured on contact, this correction can lead to a bias inversely proportional to the energy of the photon.

When the mass attenuation coefficient is obtained from the sample composition the uncertainties associated with the self-attenuation correction factor shall be taken into account, either by means of an exhaustive study of all the matrices used or on a case-by-case basis according to the level of knowledge of the basic composition of the sample. In the case of a transmission measurement the mass attenuation coefficient shall be determined by means of a measurement supported by an analytical calculation and an adjustment, in which case a balance of uncertainties shall be achieved.

10.1.5 True coincidence summing

10.1.5.1 General

A detailed description of true coincidence summing is given in the Annex C.

10.1.5.2 Probability of true-coincidence summing

Because true-coincidence summing requires the detection of two photons from the same decay it is strongly dependent on the following:

- the detection efficiency for the photons that participate in the summing process. This is a function of the detector type and the counting geometry that is used. For example, the probability of detecting two photons as a summed event increases with increasing detector size and decreasing source-to-detector distance.
- The nuclide that is being measured. More specifically, the decay schema of the nuclide (e.g., the presence of a cascade and the mode of decay, such as internal conversion or electron capture). There are some nuclei that are unaffected (such as ^{137}Cs) and others (such as ^{60}Co and ^{88}Y) that demonstrate significant summing effects.

10.1.5.3 Consequences for source-based efficiency calibrations and for sample count

Most commercially available multi-nuclide sources include nuclides that exhibit true-coincidence summing. For example, it is common to use ^{60}Co and ^{88}Y as the high-energy photon-emitting nuclides and both of these suffer from summing-out. Using nuclides that have summing-out effects in close geometries results in peak count rates that are lower than they would have been if the nuclide didn't have summing-out effects. The result of this is that the efficiency at these energies appears to be lower than the true efficiency. If these efficiency points are used without correction, then the calculated efficiency is biased low compared to the true efficiency. As a result, when the efficiency is used to calculate the activity for a summing-free sample measurement, the result is higher than the true activity.

For sample counts, the summing-out effects cause a reduction in the measured peak count rates; the measured activities are therefore lower than the true activities. Summing-in has the effect to generate summation peak that can interfere[1]. It should be noted that summing-out effects are most common.

10.1.5.4 Reducing the effect of true-coincidence summing

Since the probability of true-coincidence summing is dependent on the detection efficiency, the effect can be reduced by moving the sample away from the detector. However, the impact of reducing the overall efficiency of the counting system can result in increased counting times (in order to maintain counting statistics) and therefore reduced productivity for sample counting.

If the coincidence summing is mainly due to X-rays (as may be the case when the nucleus decays by electron capture) then the coincidence summing can be reduced by introducing an attenuator that attenuates most low energy photons. The optimum treatment is to use specific software to quantify and correct for true-coincidence-summing effects. This approach is applicable for all measurement geometries and all commonly measured nuclides that are known to be affected.

Most of the theoretical methods to calculate summation are related with the use of transport theory and Monte-Carlo techniques (see References [10] to [15]). Given the difficulties to model the detectors, some experimental procedures can be applied for each specific situation.

Further information is given in References [3] and [8].

10.2 Standard uncertainty

According to ISO/IEC Guide 98-1 the standard uncertainty of a is calculated by Formula (11):

$$u(a)=\sqrt{(w/t_{\mathrm{g}})^2\cdot\left[u^2(n_{\mathrm{g},E})+u^2(n_{\mathrm{b},E})\right]+a^2\cdot u_{\mathrm{rel}}^2(w)} \quad (11)$$

where the uncertainty of the counting time is neglected.

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[1] Examples: a ^{152}Eu summation peak at 167,7 keV interferes with ^{201}Tl at 167,5 keV, $^{110\mathrm{m}}$Ag summation peak at 1 331,5 keV interferes with ^{60}Co at 1 332,5 keV, ^{57}Co summation peak at 136 keV interferes with ^{201}Tl at 136 keV,

The relative standard uncertainty of w is calculated by Formula (12):

$$u_{\text{rel}}^2(w) = u_{\text{rel}}^2(P_E) + u_{\text{rel}}^2(m) + u_{\text{rel}}^2(\varepsilon_E) + u_{\text{rel}}^2(f_E) \tag{12}$$

Taking Formula (1) into account, the relative standard uncertainty of ε_E is calculated by Formula (13):

$$u_{\text{rel}}^2(\varepsilon_E) = u_{\text{rel}}^2(n_{\text{Ns},E}) + u_{\text{rel}}^2(A) + u_{\text{rel}}^2(P_E) = + u_{\text{rel}}^2(n_{gs,E} - n_{\text{bs},E}) + u_{\text{rel}}^2(A) + u_{\text{rel}}^2(P_E) \tag{13}$$

where $u_{\text{rel}}(A)$ includes all the uncertainties related to the calibration source: standard certificate, preparation of the calibration source.

For the calculation of the characteristic limits (see ISO 11929-1), it is necessary to know $\tilde{u}(\tilde{a})$, i.e. the standard uncertainty of a as a function of its true value. For a true value $\tilde{a}$, from $n_{\text{g},E} = \tilde{a} \cdot t_{\text{g}} / w + n_{\text{b},E}$ and with $u^2(n_{\text{g}}) = n_{\text{g}}$, one obtains Formula (14):

$$\tilde{u}(\tilde{a}) = \sqrt{(w / t_{\text{g}})^2 \cdot \left[(t_{\text{g}} / w) \cdot \tilde{a} + n_{\text{b},E} + u^2(n_{\text{b},E}) \right] + \tilde{a}^2 \cdot u_{\text{rel}}^2(w)} \tag{14}$$

The uncertainties $u(n_{\text{N}})$, $u(n_{\text{g}})$, and $u(n_{\text{b}})$ shall be calculated in accordance with ISO/IEC Guide 98-1 taking into account that the individual counts, n_i, in channel i of a multi-channel spectrum are the result of a Poisson process and hence $u^2(n_i) = n_i$ holds. The values of n_{N}, n_{g}, and n_{b} and their associated standard uncertainties $u(n_{\text{N}})$, $u(n_{\text{g}})$, and $u(n_{\text{b}})$ can be calculated with a computer program. Since there are various methods of subtracting the background below a peak in order to derive the number of counts in the net peak area, no generally applicable formula can be given. An example of the simple case of linear background subtraction is given in Annex D.

If the net-peak area $n_{\text{N},E}$, is obtained by a software using unfolding techniques, the software should yield $n_{\text{N},E}$ and its associated standard uncertainty. When it provides both $n_{\text{b},E}$ with its associated standard uncertainty, the uncertainties can be calculated according to Formulae (11) to (14). If the code gives directly a decision threshold and a detection limit for the activity a, these characteristic limits should be calculated according to ISO 11929-3. This procedure of ISO 11929-3 based on Reference [18] is not intended for users but rather for code developers.

10.3 Decision threshold

The decision threshold, a^*, is obtained from Formula (14) for $\tilde{a} = 0$ (see ISO 11929-1). This yields Formula (15):

$$a^* = k_{1-\alpha} \cdot \tilde{u}(0) = k_{1-\alpha} \cdot (w / t_{\text{g}}) \sqrt{n_{\text{b},E} + u^2(n_{\text{b},E})} \tag{15}$$

$\alpha = 0{,}05$ and $k_{1-\alpha} = 1{,}65$ are often chosen by default.

10.4 Detection limit

The detection limit, $a^\#$, is calculated by Formula (16) (see ISO 11929-1):

$$\begin{aligned} a^\# &= a^* + k_{1-\beta} \cdot \tilde{u}(a^*) \\ &= a^* + k_{1-\beta} \cdot \sqrt{w^2 \left[(a^\# / w + n_{\text{b},E} / t_{\text{g}}) / t_{\text{g}} + u^2(n_{\text{b},E}) / t_{\text{g}}^2 \right] + a^{\#2} \cdot u_{\text{rel}}^2(w)} \end{aligned} \tag{16}$$

$\beta = 0{,}05$ and $k_{1-\beta} = 1{,}65$ are often chosen by default.

The detection limit can be calculated by solving Formula (16) for $a^\#$ or, more simply, by iteration, starting with the approximation $a^\# = 2 \cdot a^*$.

By setting $\alpha = \beta$, then $k_{1-\alpha} = k_{1-\beta} = k$ and the solution to Formula (16) is given by Formula (17):

$$a^{\#} = \frac{2a^{*} + \left(k^{2} \cdot w\right) / t_{\mathrm{g}}}{1 - k^{2} \cdot u_{\mathrm{rel}}^{2}(w)} \tag{17}$$

10.5 Limits of the coverage intervals

10.5.1 Limits of the probabilistically symmetric coverage interval

The lower, $a^{\triangleleft}$, and upper, $a^{\triangleright}$, coverage limits are calculated using Formulae (18) and (19), respectively (see ISO 11929-1 and ISO 11929-3):

$$a^{\triangleleft} = a - k_{p} \cdot u(a) \text{ where } p = \omega \cdot (1 - \gamma / 2) \tag{18}$$

$$a^{\triangleright} = a + k_{q} \cdot u(a) \text{ where } q = 1 - \omega \cdot \gamma / 2 \tag{19}$$

where $\omega = \Phi[y / u(y)]$, Φ being the distribution function of the standardized normal distribution.

If $a \geq 4 \cdot u(a)$, ω can be set equal to 1 and Formula (20) applies:

$$a^{\triangleleft,\triangleright} = a \pm k_{1-\gamma/2} \cdot u(a) \tag{20}$$

$\gamma = 0{,}05$ and $k_{1-\gamma/2} = 1{,}96$ are often chosen by default.

10.5.2 The shortest coverage interval

As described in detail in ISO 11929-1, the lower limit of the shortest coverage interval, $a^{<}$, and the upper limit of the shortest coverage interval, $a^{>}$, are calculated from a primary measurement result, a, of the measurand and the standard uncertainty, $u(a)$, associated with a, either by Formula (21):

$$a^{<}, a^{>} = a \pm k_{p} \cdot u(a); p = [1 + \omega \cdot (1 - \gamma)] / 2 \tag{21}$$

or if $a^{<} < 0$, by Formula (22):

$$a^{<} = 0; a^{>} = a \pm k_{q} \cdot u(a); q = 1 - \omega \cdot \gamma \tag{22}$$

$\omega = \Phi[y / u(y)]$, Φ being the distribution function of the standardized normal distribution;

The relations $0 \leq a^{<} < a^{>}$ apply and the approximation of Formula (20) is valid.

10.6 Corrections for contributions from other radionuclides and background

10.6.1 General

In gamma spectrometry, it is frequently necessary to correct for two types of contributions.

a) The gamma line of the radionuclide being determined contains contributions from gamma radiation of another radionuclide in the sample. The contributing radionuclide has another gamma line from which the contribution to the line in question can be estimated taking into account the emission probabilities of the gamma lines.

b) The gamma line of the radionuclide being determined occurs also in the background of the spectrometer. By measuring a background spectrum without a sample for a counting time, t_0, this contribution can be corrected, taking into account the different counting times for the two spectra.

For both cases, the activity per unit of mass can be calculated using a model given in Formula (23):

$$a_c = \left(n_{N,E} / t_g - x \cdot n_{N0,E} / t_0\right) \cdot w \tag{23}$$

where x is a factor that is a function of the type of correction and $n_{N0,E}$ is replaced by n_{N,E_2} for other radionuclides contribution.

For the both types of contributions, this model gives the necessary correction.

10.6.2 Contribution from other radionuclides

The gamma line being corrected at the energy, E_1, has the net peak area, n_{N,E_1}. The contribution of the radionuclide is calculated using the ratio of the contributing radionuclide for gamma energy, E_1. Formula (23) gives the necessary correction with $x = P_{E_1} \cdot \varepsilon_1 / P_{E_2} \cdot \varepsilon_2$ and $t_0 = t_g$. This yields Formula (24):

$$a_c = \left(n_{N,E_1} - x \cdot n_{N,E_2}\right) \cdot w / t_g \tag{24}$$

Neglecting the standard uncertainty of x, the standard uncertainty of a_c is calculated from Formula (25):

$$u^2(a_c) = \left(w / t_g\right)^2 \left\{n_{g,E_1} + u^2\left(n_{b,E_1}\right) + x^2\left[n_{g,E_2} + u^2\left(n_{b,E_2}\right)\right]\right\} + a_c^2 \cdot u_{rel}^2(w) \tag{25}$$

and with a true value $\tilde{a}_c$ of a_c, Formula (26) can be derived:

$$\tilde{u}^2(\tilde{a}_c) = \left(w / t_g\right)^2 \left\{\tilde{a}_c t_g / w + n_{b,E_1} + u^2\left(n_{b,E_1}\right) + x\left(n_{g,E_2} - n_{b,E_2}\right) + x^2\left[n_{g,E_2} + u^2\left(n_{b,E_2}\right)\right]\right\} + \tilde{a}_c^2 \cdot u_{rel}^2(w) \tag{26}$$

Then, the decision threshold, a_c^*, is given by Formula (27):

$$a_c^* = k_{1-\alpha} \cdot \left(w / t_g\right) \cdot \sqrt{n_{b,E_1} + u^2\left(n_{b,E_1}\right) + x\left(n_{g,E_2} - n_{b,E_2}\right) + x^2\left[n_{g,E_2} + u^2\left(n_{b,E_2}\right)\right]} \tag{27}$$

and the detection limit, $a_c^\#$, by Formula (28)

$$a_c^\# = a_c^* + k_{1-\beta} \sqrt{\left(w / t_g\right)^2 \left\{\frac{a_c^\# t_g}{w} + n_{b,E_1} + u^2\left(n_{b,E_1}\right) + x\left(n_{g,E_2} - n_{b,E_2}\right) + x^2\left[n_{g,E_2} + u^2\left(n_{b,E_2}\right)\right]\right\} + a_c^{\#2} u_{rel}^2(w)} \tag{28}$$

The detection limit can be calculated by solving Formula (28) for $a_c^\#$ or, more simply, by an iteration starting with the approximation $a_c^\# = 2 \cdot a_c^*$.

By setting $\alpha = \beta$, then $k_{1-\alpha} = k_{1-\beta} = k$ and the solution to Formula (28) is given by Formula (29):

$$a_c^\# = \frac{2 \cdot a_c^* + \left(k^2 \cdot w\right) / t_g}{1 - k^2 \cdot u_{rel}^2(w)} \tag{29}$$

10.6.3 Contribution from background

In this case, Formula (23) is used for the correction by setting $x = 1$ and $u(x) = 0$. $N_{\mathrm{N0},E}$ is the net peak area of the gamma line in the background spectrum and t_0 is the counting time of the background spectrum. This yields Formula (30):

$$a_\mathrm{c} = \left(n_{\mathrm{N},E} / t_\mathrm{g} - n_{\mathrm{N0},E} / t_0\right) \cdot w \tag{30}$$

The standard uncertainty of a_c is calculated by Formula (31):

$$u^2\left(a_\mathrm{c}\right) = w^2\left(n_{\mathrm{g},E} / t_\mathrm{g}^2 + n_{\mathrm{g0},E} / t_0^2 + u^2\left(n_{\mathrm{b},E}\right) / t_\mathrm{g}^2 + u^2\left(n_{\mathrm{b0},E}\right) / t_0^2\right) + a_\mathrm{c}^2 \cdot u_\mathrm{rel}^2(w) \tag{31}$$

and with a true value $\tilde{a}_\mathrm{c}$ of a_c, Formula (32) is obtained:

$$\tilde{u}^2\left(\tilde{a}_\mathrm{c}\right) = w^2\left\{\tilde{a}_\mathrm{c} / t_\mathrm{g} w + \left[n_{\mathrm{b},E} + u^2\left(n_{\mathrm{b},E}\right)\right] / t_\mathrm{g}^2 + \left[n_{\mathrm{g0},E} + u^2\left(n_{\mathrm{b0},E}\right)\right] / t_0^2 + \left(n_{\mathrm{g0},E} - n_{\mathrm{b0},E}\right) / t_0 t_\mathrm{g}\right\} + \tilde{a}_\mathrm{c}^2 u_\mathrm{rel}^2(w)$$
(32)

Then, the decision threshold, a_c^*, is given by Formula (33):

$$a_\mathrm{c}^* = k_{1-\alpha} \cdot w \cdot \sqrt{\left[n_{\mathrm{b},E} + u^2\left(n_{\mathrm{b},E}\right)\right] / t_\mathrm{g}^2 + \left[n_{\mathrm{g0},E} + u^2\left(n_{\mathrm{b0},E}\right)\right] / t_0^2 + \left(n_{\mathrm{g0},E} - n_{\mathrm{b0},E}\right) / t_0 t_\mathrm{g}} \tag{33}$$

and the detection limit $a_\mathrm{c}^\#$ by Formula (34):

$$a_\mathrm{c}^\# = a_\mathrm{c}^* +$$
$$k_{1-\beta}\sqrt{w^2\left\{a_\mathrm{c}^\# / t_\mathrm{g} w + \left[n_{\mathrm{b},E} + u^2\left(n_{\mathrm{b},E}\right)\right] / t_\mathrm{g}^2 + \left[n_{\mathrm{g0},E} + u^2\left(n_{\mathrm{b0},E}\right)\right] / t_0^2 + \left(n_{\mathrm{g0},E} - n_{\mathrm{b0},E}\right) / t_0 t_\mathrm{g}\right\} + a_\mathrm{c}^{\#2} u_\mathrm{rel}^2(w)}$$
(34)

The detection limit can be calculated by solving Formula (34) for $a_\mathrm{c}^\#$ or, more simply, by iteration with a starting approximation $a_\mathrm{c}^\# = 2 \cdot a_\mathrm{c}^*$.

When taking $\alpha = \beta$, then $k_{1-\alpha} = k_{1-\beta} = k$ and the solution of Formula (34) is given by Formula (35):

$$a_\mathrm{c}^\# = \frac{2 \cdot a_\mathrm{c}^* + \left(k^2 \cdot w\right) / t_\mathrm{g}}{1 - k^2 \cdot u_\mathrm{rel}^2(w)} \tag{35}$$

11 Test report

The test report shall conform to ISO/IEC 17025 requirements. It shall contain the following information:

a) a reference to this document i.e. ISO 18589-3:2023;

b) identification of the sample;

c) units in which the results are expressed;

d) the test result:

  1) when the activity, a, is compared with the decision threshold (see ISO 11929 series);

    — if the result is less than the decision threshold, the result of the measurement is expressed as $\leq a^*$,

  - if the result is greater than the decision threshold, the result of the measurement is expressed as $a \pm u_c(a)$ or $a \pm U$ with the associated k value,

2) when the activity, a, is compared with the detection limit;

  - if the result is less than the detection limit, the result of the measurement is expressed as $\leq a^{\#}$
  - if the result is greater than the detection limit, the result of the measurement is expressed as $a \pm u_c(a)$ or $a \pm U$ with the associated k value.

Complementary information can be provided such as:

e) the uncertainty can also be expressed as the limits of the probabilistically symmetric coverage interval $a^{\triangleleft}, a^{\triangleright}$ and/or the limits of the shortest coverage interval $a^{<}, a^{>}$;

f) probabilities α, β and $(1 - \gamma)$;

g) decision threshold and the detection limit;

h) if the detection limit exceeds the guideline value, it shall be documented that the method is not suitable for the measurement purpose;

i) mention of any relevant information likely to affect the results;

NOTE Occasionally, it is requested by the customer or regulator to compare the primary measurement result, a, with the detection limit, $a^{\#}$, in order to decide whether the physical effect is recognized or not. Such stipulations are not in accordance with ISO 11929-1. They have the consequence that it is decided too frequently that the physical effect is absent when in fact it is not absent.

Annex A
(informative)

Analysis of natural radionuclides in soil samples using gamma spectrometry

A.1 General

Among the natural radionuclides, those besides ^{40}K, belonging to a natural decay chain that can be measured using gamma spectrometry include ^{238}U, ^{226}Ra; ^{210}Pb of the uranium/radium decay chain; ^{235}U and ^{227}Th of the uranium/actinium decay chain; as well as ^{232}Th, ^{228}Ra and ^{228}Th of the thorium decay chain; see Figure A.1.

Some radionuclides of the natural decay chains (e.g., ^{238}U, ^{228}Ra, ^{228}Th) cannot be determined directly by gamma spectrometry but only by measuring their daughter radionuclides. In these cases, it is necessary to ensure that there is equilibrium between the parent radionuclide and the daughter radionuclides being measured. Radioactive equilibrium can be disrupted within the media being examined due to the different chemical or biochemical behaviour of the respective elements. For example, the radioactive equilibrium can be shifted strongly due to the different transfer behaviour of parent radionuclide and daughter radionuclides in the soil-vegetation-animal-milk chain.

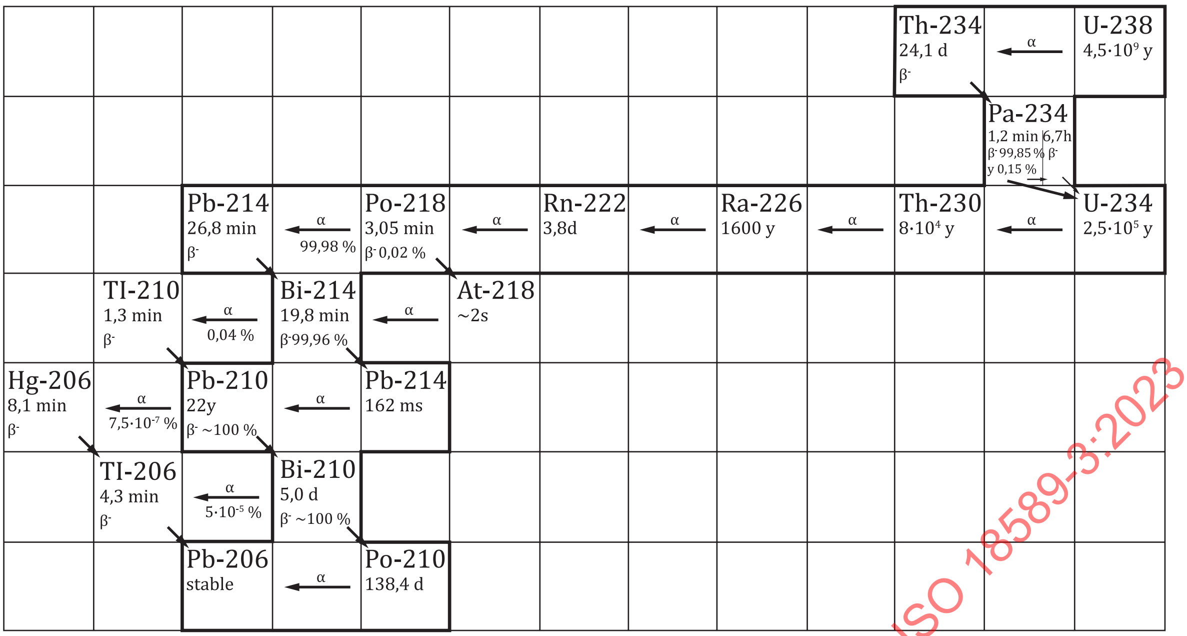
In these cases, it is recommended to keep the samples for a sufficiently long period before measuring them. On the other hand, interference of the radioactive equilibrium in the sample for measurement can result from escaping radon. In the case of measuring the short-lived decay products of ^{222}Rn, the sample material shall additionally be put into a gas-tight glass container such that the dead volume in the glass container between the sample and the lid is as low as possible, and the sample left until radioactive equilibrium has been achieved. Since, as a rule, the severity of the interference with the equilibrium is not known, to be on the safe side, one should assume, when estimating the waiting period that initially there are practically no daughter radionuclides. In the case of a long-lived parent radionuclide and a short-lived decay product, this means that the waiting period should be at least six half-lives of the decay product.

Another problem for the gamma spectrometric determination of natural radionuclides is the fact that some radionuclides show gamma lines that are identical or so near each other that it is not possible to resolve them by hardware or software means. In these cases, it is necessary to make corrections using other gamma lines; see 10.6.1 a).

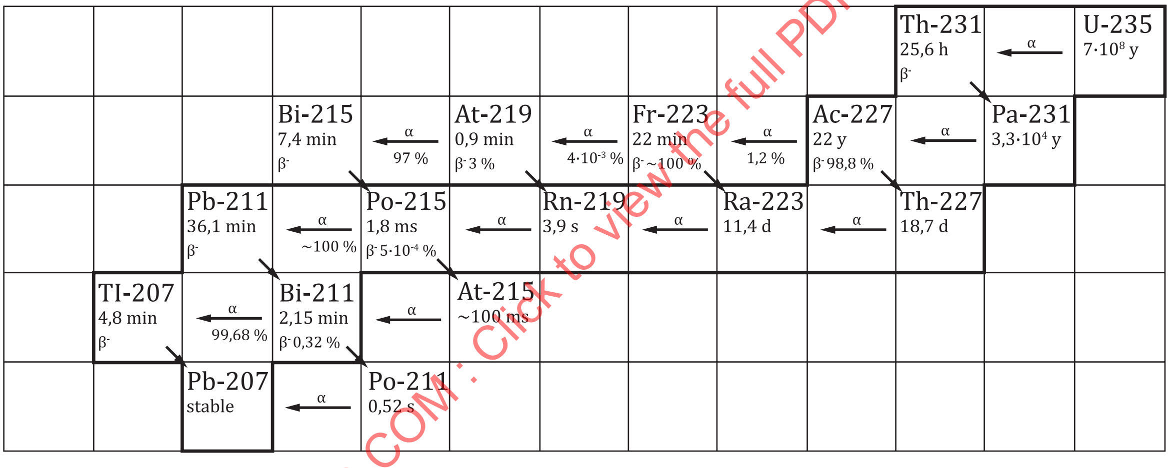
If this procedure cannot be carried out, a correction can be performed only by measuring radionuclides with other methods (e.g. alpha spectrometry or an emanation measurement). It is necessary to use these more sensitive methods when required detection limits cannot be achieved through gamma spectrometry (e.g. due to insufficient sample quantities).

A.2 to A.10 give additional explanations regarding the gamma spectrometric determination of common natural radionuclides in soil. For some radionuclides, the requirement of correcting self-absorption and summation losses are mentioned. Table A.1 lists the photon energies, E_γ, and the emission probabilities, P_E, of selected radionuclides.

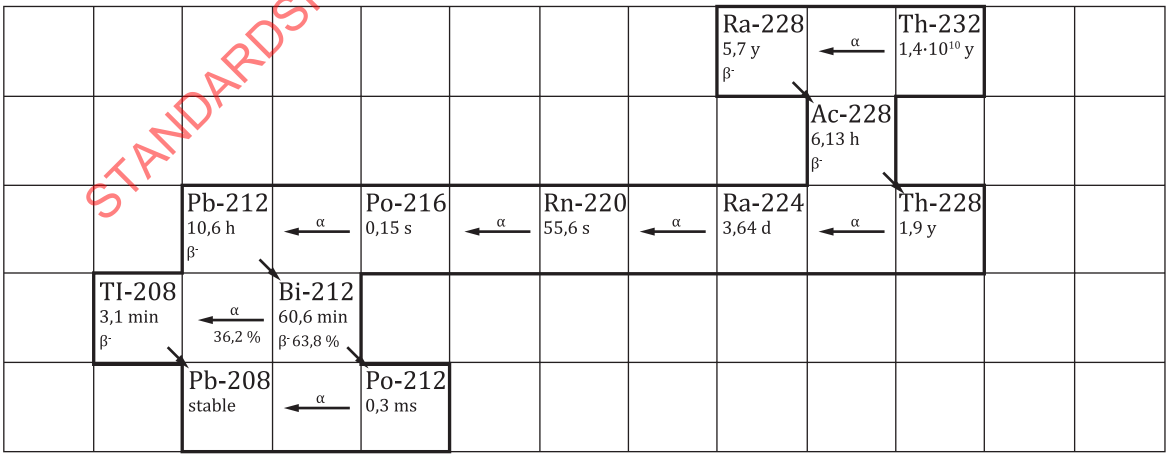
When the decay chains are in equilibrium, the listed emission probabilities refer to the decay of the parent nuclide. For example, in the determination of the ^{228}Th content through ^{208}Tl, it is not necessary to take into account that the branching ratio of the ^{212}Bi into ^{208}Tl is only 36,2 %; see Figure A.1 c). This is already considered when stating the emission probability in Table A.1.



a) **Uranium/radium** ($A = 4n + 2$)



b) **Uranium/actinium** ($A = 4n + 3$)



c) **Thorium** ($A = 4n$)

Figure A.1 — Natural decay chain

A.2 Determination of ^{238}U

^{238}U with its half-life of 4,468 · 10^9 years is the parent radionuclide in the uranium/radium decay chain. ^{238}U cannot be measured by gamma spectrometry but only through its decay radionuclides, ^{234}Th and ^{234m}Pa. Very often, the 63,28 keV-line of ^{234}Th is used. At this low energy, however, matrix variations between calibration and test sample can have considerable influence on the result due to different self-absorption and therefore it is necessary that they be taken into account. The 63,28 keV line is additionally interfered with by the 63,81 keV line of the ^{232}Th with an emission probability of 0,263 %. The use of the double line energies 92,38 keV and 92,8 keV of ^{234}Th with an emission probability of 4,33 % (2,18 % + 2,15 %) is not recommended to quantify ^{238}U.

The lines of ^{234m}Pa at 1 001,03 keV and 766,37 keV are more suitable due to their higher energies, although their emission probabilities are very low and summing coincidences have to be considered. When evaluating the 1 001,03 keV line, it is necessary to observe that the emission probability is 0,847 %. The lines suitable for evaluation have been compiled In Table A.1.

Additional information for the determination of ^{238}U is the nearly constant natural ^{238}U/^{235}U activity ratio of 21,7 in samples of natural origin.

A.3 Determination of ^{226}Ra

^{226}Ra is a long-lived radionuclide in the uranium/radium decay chain with a half-life of 1 600 years. For the gamma spectrometric determination of ^{226}Ra in soil samples, there are, in particular, two possibilities. Either ^{226}Ra is evaluated at its 186,21 keV line or the short-lived decay radionuclides ^{214}Pb and ^{214}Bi of ^{222}Rn are measured after the radioactive equilibrium between ^{226}Ra, ^{222}Rn, ^{214}Pb and ^{214}Bi has been established.

If the line at 186,21 keV is used, it is necessary to take into account the contribution of the overlapping line at 185,72 keV of ^{235}U. The contribution of this overlap can be corrected according to 10.6.1 a) if the ^{235}U activity concentration is known. The activity concentration of uranium in a sample can also be derived using the results from other uranium determinations (e.g. fluorimetry, phosphorimetry, X-ray fluorescence measurements, mass spectrometry). If radioactive equilibrium can be assumed in the uranium/radium chain, the activity concentration of ^{238}U equals that of ^{226}Ra, and in case of natural uranium, the activity concentration of ^{235}U can be calculated since the specific activity of ^{235}U is approximately 1/22 of the specific activity of ^{238}U.

If radioactive equilibrium or the natural composition of uranium cannot be assumed, the contribution of ^{235}U can be corrected only if its activity concentration is determined through other lines of ^{235}U.

For the determination of activities of the short-lived decay products, gas-tight containers are required since, otherwise, the values determined for ^{214}Pb and ^{214}Bi can be too low due to the loss of ^{222}Rn. The period between filling the sample into a gas-tight vessel and starting the measurement is determined by the half-life of ^{222}Rn (3,8 d) and should be at least 23 d. The application of special types of activated charcoal can be useful to prevent the radon losses. It is necessary to correct the summation if all detectable gamma-lines of ^{214}Pb and ^{214}Bi are used for the determination of ^{226}Ra. If the gamma line of ^{214}Pb at 351,93 keV is used, the summation effects become negligible.

A.4 Determination of ^{210}Pb

^{210}Pb is a radionuclide of the uranium/radium decay chain with a half-life of 22,3 years and a gamma energy line at 46,54 keV. This energy gamma line is usually measured by gamma spectrometry using an N-type coaxial high-purity germanium detector. Different material compositions of calibration sources and test samples play an important role because of different self-absorption effects. It is necessary to consider the material composition appropriate for the correction of these effects (see 10.1.4).

A.5 Determination of ^{235}U

^{235}U is the parent radionuclide of the uranium/actinium decay chain with a half-life of 7,037 · 10^{8} years. ^{235}U can be measured through the lines at 143,8 keV, 163,3 keV, 185,7 keV, and 205,3 keV. The line with the highest emission probability at 185,7 keV is not suitable if the ^{226}Ra activity is not known. The line with the second highest emission probability at 143,8 keV overlaps the 144,2 keV line of ^{223}Ra. The two other lines have lower emission probabilities, which are not sufficient for measuring samples with a low activity concentration of ^{235}U. Alternative measurement techniques, such as alpha-spectrometry or mass-spectrometry, can also be used.

A.6 Determination of ^{227}Ac

^{227}Ac is a radionuclide of the uranium/actinium decay chain with a half-life of 21,8 years. ^{227}Ac can be determined by gamma spectrometry through its short-lived decay radionuclides ^{227}Th, ^{223}Ra and ^{219}Rn. Generally, the 235,97 keV line of ^{227}Th is preferred, since the other lines either have low emission probabilities or overlap with lines of other radionuclides.

A.7 Determination of ^{232}Th

^{232}Th with a half-life of 1,41 · 10^{10} years is the parent radionuclide of the thorium decay chain. ^{232}Th has a line at 63,81 keV with a very low emission probability of 0,263 % which overlaps a line of ^{234}Th at 63,28 keV with a higher emission probability of 4,1 %, so that ^{232}Th cannot be determined directly by gamma spectrometry in soil samples. Determination through its decay radionuclides ^{228}Ac, ^{212}Pb and ^{208}Tl can be performed only if one assumes that these radionuclides are in radioactive equilibrium with each other and with ^{232}Th.

This assumption is not always true in soil samples because of the presence of ^{228}Ra with a half-life of 5,7 years in the decay chain between ^{232}Th and ^{228}Ac (half-life of 6,3 h), which can cause a radioactive disequilibrium different from that of thorium due to its solubility.

A.8 Determination of ^{228}Ra

^{228}Ra is a radionuclide of the thorium decay chain with a half-life of 5,75 years. ^{228}Ra is determined through measuring its decay radionuclide ^{228}Ac, which is available in radioactive equilibrium with ^{228}Ra due to its short half-life of 6,15 h. Depending on the chosen measurement geometry, considerable counting losses can be expected as a result of summation effects, which require correction.

If the derived activity concentrations for ^{224}Ra, ^{212}Pb, and ^{208}Tl do not differ from ^{228}Ac, they can also be used for the evaluation of the ^{228}Ra activity concentration.

A.9 Determination of ^{228}Th

^{228}Th is a radionuclide of the thorium decay chain with a half-life of 1,91 years. ^{228}Th can be determined through measuring its short-lived decay radionuclides ^{224}Ra, ^{212}Pb, and ^{208}Tl if there is a radioactive equilibrium. Equilibrium is established within about 22 d between ^{228}Th and its short-lived daughters.

^{212}Pb lines are subject to interference and should not be used to determine ^{228}Th. Better choices are the 727,4 keV line from ^{212}Bi and the 583 keV line from ^{208}Tl, which are not normally subject to interference.

When measuring ^{224}Ra at 240,99 keV, it is necessary to consider the contribution of the line of ^{214}Pb at an energy of 242,00 keV. Due to the half-life of 3,66 d for ^{224}Ra, radioactive equilibrium with ^{228}Th can be assumed in the soil sample so that the measured results for ^{224}Ra also apply to ^{228}Th.

A condition for the measurement of ^{212}Pb and ^{208}Tl is that ^{220}Rn remains in the test sample. This is generally given due to its low half-life of 54 s. Preferably, the 238,63 keV line of ^{212}Pb is used for evaluation. However, as a result of the summation effect, it is necessary to take counting losses into account.

A.10 Determination of ^{40}K

^{40}K is included in the isotope mixture of natural potassium (^{39}K, ^{40}K, ^{41}K) with an isotopic abundance of 0,011 9 % and decays with a half-life of $1,29 \cdot 10^9$ years to ^{40}Ca (β-decay) or ^{40}Ar (electron capture), respectively.

^{40}K is determined by gamma spectrometry using its 1 460,83 keV line.

Table A.1 — Selected gamma lines (above 25 keV) for the determination of natural radionuclides and their interferences[19][20]

Radionuclide determined	Radionuclide measured	Energy E keV	Emission probability of the radionuclide determined P_E %	Interfering radionuclide E, P_E
^{238}U	^{234}Th	63,30	3,75	^{232}Th (63,811 keV; 0,259 %)
		92,38	2,18	Th-Kα1 (93,35 keV; 5,6 %)
		92,80	2,15	—
^{238}U	^{234m}Pa	766,361	0,323	—
		1 001,026	0,847	—
^{226}Ra	^{226}Ra	186,211	3,555	^{235}U (185,720 keV; 57,0 %)
^{226}Ra	^{214}Pb	295,224	18,414	^{211}Bi (351,03 keV; 13,00 %)
		351,932	35,60	—
^{226}Ra	^{214}Bi	609,312	44,49	—
		1 120,287	14,91	—
		1 764,494	15,31	—
^{210}Pb	^{210}Pb	46,539	4,252	—
^{235}U	^{235}U	143,767	10,94	^{223}Ra (144,27 keV; 3,36 %)
		163,356	5,08	—
		185,720	57,0	^{226}Ra (186,211 keV; 3,555 %)
		205,316	5,02	—
^{227}Ac	^{227}Th	235,97	12,3	—
^{232}Th	^{228}Ac	see ^{228}Th		—
	^{228}Th			—
	^{208}Tl			—
^{228}Ra	^{228}Ac	209,248	3,97	—
		338,320	11,40	^{223}Ra (338,282 keV; 2,85 %)
		911,196	26,20	—
		968,96	15,90	—
^{228}Th	^{224}Ra	240,986	4,12	^{214}Pb (241,997 keV; 7,268 %)
^{228}Th	^{212}Pb	238,632	43,6	—
		300,089	3,18	^{227}Th (300,00 keV; 2,70 %)
				^{231}Pa (300,07 keV; 2,47 %)
^{228}Th	^{208}Tl	277,37	2,39	^{228}Ac (278,80 keV; 0,204 %)
		583,187	30,77	^{228}Ac (583,391 keV; 0,120 %)
		860,53	4,49	—
		2 614,511	36,11	—
^{40}K	^{40}K	1 460,822	10,55	^{228}Ac (1 459,131 keV; 0,87 %)

Annex B
(informative)

Self-attenuation correction[21][22]

The linear attenuation coefficient $\mu(E)$ expresses the probability of a photon at the energy E interacting with matter. It depends on the material passed through and the energy of the photons.

For a mono-energetic and mono-directional beam, the number of photons N at the energy E absorbed or diffused in a material unit thickness dx is proportional to the number N_0 of incident photons and the thickness of the material traversed. After the crossing it remains $N+dN$ by using Formula (B.1):

$$\mathrm{d}N = -\mu(E) \cdot N \cdot \mathrm{d}x \qquad \text{(B.1)}$$

The law of attenuation results from the integration on a thickness X of Formula (B.1) as given in Formula (B.2):

$$N = N_0 \cdot \mathrm{e}^{-\mu(E)X} \qquad \text{(B.2)}$$

The law is identical for photon flux express in s^{-1}(Formula B.3):

$$\phi = \phi_0 \cdot \mathrm{e}^{-\mu(E)X} \qquad \text{(B.3)}$$

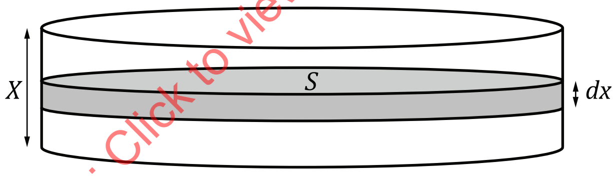


Figure B.1 — Representation of the cross-section of surface S and the height X in a cylinder