The Series of Environmental Radioactivity Measuring Methods
No.17

Method for Measurement of Environmental Gamma-rays with a Continuous Monitor

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Radiation Monitoring Division
Radiation Monitoring Department
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Chapter 1 Introduction

The first edition of this measuring method was established in 1982 to measure environmental gamma-ray dose rates by an installed continuous monitor, mainly as part of environmental radiation monitoring around nuclear facilities. It was revised in 1996, and it is still being developed.

After the accident at Tokyo Electric Power Company's Fukushima Dai-ichi Nuclear Power Station (hereinafter referred to as the “Fukushima Dai-ichi NPS accident”) caused by the Great East Japan Earthquake on March 11, 2011, the existing monitoring framework was reviewed. In light of these circumstances, the present situation is reflected in the current revision with the basic concept and objectives inherited, considering the recent development of measuring instruments, as well as the experience acquired, and problems found after the Fukushima Dai-ichi NPS accident.

In the past, the role of continuous monitoring was to provide information that would contribute to the monitoring of any abnormalities caused by nuclear facilities and the evaluation of external exposure doses. However, after the Fukushima Dai-ichi NPS accident, it may be used as information necessary for determining whether further protective measures should be taken for the residents, etc., in case of a nuclear disaster.

From the experience of the Fukushima Dai-ichi NPS accident, it turned out that, once a nuclear disaster occurs, the effects of released radionuclides would have extensive and long-term impacts. Because conventional “Installed-type” continuous monitoring is insufficient, various methods have been developed and utilized for efficient and rapid monitoring. Based on the current situation, this measuring method also includes “transportable” continuous monitoring and “Mobile surveys.” In addition, the gamma-ray energy spectrum, together with the environmental gamma-ray dose rate, provides very useful information for estimating the causes of abnormal values. Therefore, part of the content of the radioactivity measurement series No. 20 “Environmental gamma-ray spectrometry” is provided for integration.

“Measurement of the environmental gamma-ray dose rate” described in this measuring method must be applicable to various situations, because uncertainties are involved every time a measurement is taken. Sorting the uncertainties involved in the measurements in advance and understanding their characteristics will enable responding to a nuclear disaster such as the Fukushima Dai-ichi NPS accident. This measuring method should be used as a basis for determination when users perform appropriate measurement in their own situation.

The main users of this measuring method include the local government personnel responsible for overseeing nuclear facilities.

Chapter 2 Glossary

This chapter explains the terms used in this measurement series, and describes the important terms used in the environmental gamma-ray dose rate measurement.

Natural radioactive nuclides (natural radionuclides) (Reference 1)
Naturally occurring radionuclides.

Artificial radionuclides (Reference 1)
Radionuclides artificially produced by nuclear fission, activation, etc.

Energy characteristics
The characteristics of the response or calibration constant of a radiation detector that depend on the energy of radiation.

Directional characteristics
The characteristics of the response or calibration constant of a radiation detector that depend on the incident direction of radiation to the detector.

High-energy components and low-energy components
Environmental radiation can be mainly classified into cosmic rays, radiation from radioactive materials existing in the environment, and radiation from radioactive materials released from nuclear facilities, and the energy of radiation is mostly 3 MeV or less, except for cosmic rays. In this measuring method, the energy above 3 MeV is classified as a high-energy component, and the energy below 3 MeV is classified as a low-energy component.

Response
The ratio of an indicated value of the monitor to the amount to be measured. As sensitivity usually includes a sensitivity limit and is confusing, we call it response here. It is the inverse of a calibration constant.

Pulse-height distribution
A pulse-height distribution of the spectrum of the pulse as measured.

Corrected pulse-height distribution
A distribution determined by correcting the pulse-height distribution such that the ratio of the peak value to energy is constant.

Incident gamma-ray spectrum
A spectral distribution, based on the corrected pulse height distribution, determined by applying a response function to transform the spectrum to the gamma-ray energy one.

Self-contamination
Self-contamination becomes evident when a small number of radionuclides are contained in the materials of radiation detector, and the measuring system is affected; this would create a problem in low-level radiation measurements.

Nuclear Emergency Preparedness Guideline
A guideline presented by the Nuclear Regulation Commission after the Fukushima Dai-ichi NPS accident, providing a concept of monitoring during a nuclear disaster.

UPZ

The areas where emergency protective actions are to be prepared as specified in the Nuclear Emergency Preparedness Guideline. This stands for the Urgent Protective Action Planning Zone. This is applied to the areas within approximately 30 km of nuclear facilities.

Tissue weighting factor (Reference 1)

A factor that multiplies the equivalent dose of a tissue or an organ, to account for the relative probabilistic damage resulting from each exposure. The value of this factor is assigned to each tissue or organ.

Radiation weighting factor (Reference 1)

A relative factor that is multiplied by the absorbed dose, to account for differences in the effects of different types of radiation impacting on health.

Nondestructive inspection (Reference 1)

A generic term for a method of inspecting an object without destroying it. Some inspections use radiation, ultrasound, electromagnetic induction, fluorescent dyes, etc. It is also called a nondestructive test.

Radiopharmaceuticals (Reference 1)

Medicine labeled with a radionuclide and used in diagnosis and treatment.

Folding (Reference 1)

A mathematical method for determining the output pulse-height distribution of a radiation detector from the energy response function of the detector and the energy spectrum of incident radiation.

Unfolding (Reference 1)

A mathematical method for determining the energy spectrum of the original incident radiation from the output pulse-height distribution of a radiation detector and the energy response function of the detector.

Response function

A function used to determine the spectrum of a gamma ray incident on the detector from the detected pulse-height distribution, including the efficiency of the detector in terms of the detector's response at the time of monoenergetic photon incidence.

Successive approximation method

An analytical method to minimize inaccuracies that occur in (deconvolution or unfolding) when determining a gamma-ray incident spectrum from multiple response functions (a response matrix), by successively increasing the order of approximation.

Geometry (Reference 1)

A term used to express the spatial arrangement of various elements. The term “geometric arrangement” is also used.

Tissue-equivalent material (Reference 1)

A material with an effective atomic number equal to that of a biotissue.

Air-equivalent material (Reference 1)

A material with an effective atomic number equal to that of air.

Phantom (Reference 1)

A human phantom used to simulate and measure the attenuation, scattering, or distribution of radioactive materials in a living body.

ICRU Sphere (Reference 1)

A spherical homogeneous phantom with a diameter of 30 cm, consisting of tissue-equivalent materials with a density of 1 g/cm^3. Its mass composition is 76.2% oxygen, 11.1% carbon, 10.1% hydrogen, and 2.6% nitrogen.

Pulse amplifier (Reference 1)
An amplifier with wide-band characteristics for amplifying a pulse signal.
Preamplifier (Reference 1)
An amplifier connected immediately after a radiation detector and used to improve the impedance matching and signal-to-noise ratio.
Proportional amplifier (Reference 1)
A pulse amplifier or a DC amplifier that outputs a signal of a peak value proportional to the peak value of the input signal. This is generally connected after a preamplifier and used as the main amplifier.
Analog-to-digital converter (Reference 1)
A device for converting an analog signal into an output in digital quantities. This is used in a pulse-height analyzer, etc. Abbreviated as ADC.
Pulse-height analyzer (Reference 1)
A device for measuring the frequency distribution of the peak value of a pulse signal.
Single-channel pulse-height analyzer (Reference 1)
A type of pulse-height analyzer that selects input pulses within a given pulse-height range and outputs logical signals to enable counting. Abbreviated as SCA.
Multichannel pulse-height analyzer (Reference 1)
A type of pulse-height analyzer with a large number of storage elements that accumulates input signals for each pulse height to enable counting. This is also called a multichannel pulse-height analyzer. Abbreviated as MCA.
Photomultiplier tube (Reference 1)
A vacuum tube for converting light into an electrical signal, which is essentially composed of a photocathode and an electron multiplier. Abbreviated as PMT.
Detector window (Reference 1)
The part of a detector designed to facilitate the permeation of the radiation to be detected.
Environmental radiation monitoring equipment (Reference 1)
A generic term for equipment used to measure radiation dose rates, etc. near nuclear or radiation facilities. This includes equipment to measure the concentrations of radioactive materials contained in environmental samples such as soil and sea water, and meteorological observation facilities.
Monitoring station (Reference 1)
A centralized facility for radiation measurement equipment that measures exposure dose rates and radioactivity of airborne dusts in the field around nuclear facilities or radiation facilities.
Monitoring post (Reference 1)
An installed facility in the field that continuously measures exposure dose rates, neutron fluence rates, etc. in the environment.
Monitoring car (Reference 1)
An automobile for mobile measurement, equipped with a device to measure radiation or radioactivity in the environment.
Scintillation (Reference 1)
Light emission based on the excitation of atoms by ionizing radiation, with a duration of several microseconds.
Scintillation decay time (Reference 1)
The time that the photon emission rate of scintillation takes to decay to 1/e of its initial value (e is the base of a natural logarithm) after a single excitation.
Effective center (of a detector) (Reference 1)

A point in the effective volume that can be regarded as a center used for determining the output characteristics of the detector relative to the exposure dose or exposure dose rate. Depending on the conditions, a non-negligible difference may be produced from the geometric center.

Counting loss (Reference 1)

When counting pulse signals, the measured count value might be lower than the actual value owing to the dead time of the detector or counter, resolving time, pile-up of pulses, etc.

Dead time (Reference 1)

The time during which detection and counting capabilities are lost after the counter tube, counter circuit, etc. operate once. Counting cannot be performed for the incident radiation generated during this time.

Resolving time (Reference 1)

The shortest time difference necessary for distinguishing two subsequent pulses or ionizing events as two independent pulses or events.

Pile-up (Reference 1)

A phenomenon in which the pulse-height indication becomes inaccurate as a result of a subsequent pulse overlapping the tail of the preceding pulse.

Traceability (Reference 2)

The property of a measured result or a standard value that can be linked to a given standard through a chain of contiguous comparisons in which all uncertainties are expressed. National or international standards are usually adopted as the standard.

Reference gamma-ray source

A gamma-ray source for which traceability has been established with national standards for the dose rate certification. The exposure dose rate at a point of 1 m from the center of the source is usually certified.

Certified equipment with a display

Equipment certified by the system of design certification and specific design certification. The equipment is sold by manufacturers and importers and can be used by filing a brief report to the competent authority (Nuclear Regulation Authority).

Weathering effect

Attenuation of radiation dose rate and radioactivity concentrations owing to natural factors such as wind and rain.

Chapter 3 Configuration of a continuous monitor

3.1 Outline

Various measurement principles and signal processing methods are adopted in the equipment used for continuous monitoring, and the output measurements are the resulting numerical values containing various components. The components in the measured environmental gamma-ray dose rate are classified into natural radiation and artificial radiation, each of which is affected by temporal fluctuation characteristics, energy spectral distribution, and meteorological factors.

The user is requested to fully understand the above description. For the measurement principles and signal processing methods of the equipment used for continuous monitoring, refer to this chapter; for the various components contained in the measured values, their fluctuation factors, and spectral analysis, refer to Chapter 7 “Analysis and evaluation of measurement results” and Chapter 10 “Processing of environmental gamma-ray spectrum measured data by a NaI monitor.”

Note that the detectors and various systems described in this measuring method are representative examples of the equipment necessary for conducting environmental gamma-ray dose rate measurement, and handling of other equipment is not rejected. Especially in the event of a nuclear disaster, optimal monitoring should be performed with limited personnel, time, and equipment. Therefore, it is important to meet the specifications necessary to achieve the given objectives under the respective circumstances.

General matters on environmental gamma rays measured by a continuous monitor are described in Explanation A.

3.2 Continuous monitor detector

This section describes the features of each monitor detector (Reference 3).

3.2.1 Ionization chamber-type detector

An ionization chamber-type detector is designed to detect radiation using the ionization of radiation, and it is a very primitive measuring instrument that has been used since the discovery of radiation. With the incidence of radiation (gamma rays), secondary electrons are generated by the interaction between the radiation (gamma rays) and the ionization chamber wall. When the generated electrons pass through the gas in the ionization chamber, the filler gas is ionized to positive ions and electrons, which are collected on the cathode and anode, respectively, by applying high voltage; then, a current flows. The dose rate is expressed in an analog output of a current–voltage conversion by passing a current through a high resistance, or in a digital output of a charge by current integration at fixed time interval.

There are two types of ionization chamber dosimeters, atmospheric air type (1 atmosphere) and pressurized type. The latter is widely used as a continuous monitor. The atmospheric air type (1 atmosphere) was not common because of its larger detector compared with the pressurized type; however, after the Fukushima Dai-ichi NPS accident, it has been used for measurement in areas with high dose rate. In terms of the detector energy characteristics and dose rate linearity, the atmospheric air type (1 atmosphere) has good energy characteristics. The response of the pressurized type depends on the material of the detector wall and the filler gas. The range of dose rate measurement can be large for both the atmospheric air type (1 atmosphere) and the pressurized type. Details are provided in Explanation B.

(1) Atmospheric air ionization chamber (1 atmosphere)

This type of detector has a sealed structure using a wall material (resin, etc.) of approximately 5 mm in thickness, which is considered to be equivalent to air, with an effective volume of approximately

20 L.

(2) Pressurized ionization chamber

This is an ionization chamber in which the gas filled in the radiation detection section is pressurized to atmospheric pressure or higher. For example, an ionization chamber of this type has aluminum or stainless steel wall, in which argon or nitrogen gas with a volume equivalent to 20 L or more at atmospheric pressure is filled at a pressure of 8–20 kgf/cm^2 (approximately 8–19 atmospheres), and high sensitivity is achieved in a small-sized chamber.

The energy characteristics and the dose rate linearity are as described in Explanation B.1; when comparing the types using aluminum and stainless steel as the wall materials, a large difference is identified in the range of 100 keV or less. This means that, when the proportion of ^{133}Xe (81 keV), which is subject to measurement in case of an emergency, in the radioactive plume to the total dose rate is large, caution should be taken in handling the measurements.

3.2.2 NaI (Tl) scintillation detector

Because a NaI (Tl) scintillation detector is highly sensitive to gamma rays, it is useful for measuring low-level dose rates. However, it has a characteristic of showing excessive response in the low-energy region (around 100 keV) owing to the influence of energy characteristics, and the G(E) function method is employed as a typical tool for correcting the excessive response. A description of the G(E) function method is given in Explanation B.2.

With the incidence of radiation, the excitation emission (fluorescence) generated in the NaI (Tl) crystal is converted into photoelectrons by a photomultiplier tube and amplified to output a pulse proportional to the intensity of the light. Because the output pulse signal is proportional to the energy of the incident radiation, qualitative or quantitative determination of radionuclides can be achieved by analyzing the gamma-ray spectrum.

The characteristic of the detector is that the absorption reaction of radiation is not directly measured in the form of ionization or energy, but is indirectly measured by light, and the energy resolution is low because of a small number of photoelectrons generated owing to the low efficiency of the opto-electron conversion. However, because the electron multiplication factor of a photomultiplier tube is very high, the signal and noise can be distinguished and separated by a pulse-height discriminator in spite of the very low signal; thus, the radiation detection can be performed with high stability.

3.2.3 Silicon semiconductor detector

A silicon semiconductor detector has an equivalent measurement principle to that of an ionization chamber detector and is sometimes called a “solid ionization chamber.”

Secondary electrons are generated as a result of the interaction between incident radiation and silicon crystals. By the ionization of secondary electrons, an electron–hole pair is generated, each of which is collected on the anode and the cathode, respectively, by applying a high voltage; then, a current flows. A signal is output as a pulse, and the dose rate is calculated by multiplying with a conversion factor.

The characteristics of a silicon semiconductor detector, including the energy characteristics, are described in Explanation B.3.

Further, the sensitivity of a silicon semiconductor detector is inferior to that of a NaI (Tl) scintillation detector. This feature of low sensitivity is exploited in a detector used for high-dose-rate regions.

3.2.4 CsI (Tl) scintillation detector

A CsI (Tl) scintillation detector has the same measurement principles as the NaI (Tl) scintillation detector. This detector has a feature of high sensitivity to gamma rays, as with the NaI (Tl) scintillation detector. Generally, a scintillation detector is composed of a scintillator, which is phosphor, and a photodetector, etc. A typical photodetector is a photomultiplier tube. However, the fluorescence pulselength differs depending on the type of scintillator, and the maximum-sensitivity pulselength also differs depending on the type of a photodetector it is combined with. At present, with the development in electronic technology, an optimum combination is achieved for commercialization, considering these features.

3.3 Recording of measurements

The recording of the measured values in each monitor can be performed in a digital format, in which an electronic file is stored in the measuring device itself, or in a format in which an analog output is committed to a chart paper or similar. In many cases, data are sent to a “telemeter system” at a certain transmission interval and are also recorded in the system.

3.4 Telemeter system

A telemeter system collects data from each continuous monitor and performs storage, analysis, and distribution in a centralized manner. An example of the functions of a telemeter system is shown below:

(1) Data collection

- [1] Continuous monitor (an installed measurement station) * Weather data can also be collected, if necessary
- [2] Transportable monitoring post
- [3] Monitoring car
- [4] Simplified electronic dosimeter
- [5] Continuous monitors provided by affiliates (different monitoring agencies)
- [6] Neighboring municipalities

(2) Transmission method, etc.

An appropriate transmission method should be selected from wired, satellite, and analog wireless lines, according to the installation environment. In addition, server multiplexing or multisite integration should be implemented in the system.

(3) Data distribution

If there is an information sharing system among organizations related to the monitoring institution, data are distributed to the relevant systems. Data are also distributed to neighboring municipalities. In addition, because information sharing and distribution to related organizations may be required from time to time, the specifications should be applicable to various situations.

(4) Data display

Capabilities should be provided for uploading the collected data to the home page for public use in real time and outputting the data to display devices installed in relevant places. As with “(3) Data distribution,” the specifications should be applicable to various situations.

(5) Filing and storage of data

Capabilities should be provided for storing data in a database for a long time and transferring the past data. Because storage of information is essential despite its huge volume, memory expansion should also be achievable, considering an increase in storage capacity.

(6) Monitoring and analysis of data

Capabilities should be provided for enabling real-time monitoring by a GIS[*1] map and outputting of various forms, graphs, and reports. In addition, a capability of performing a real-time spectral analysis should be provided. Note that average, maximum, and minimum values should be arbitrarily set and output, because these values contribute to extracting any abnormal values that may be contained in various forms.

(7) Notification function

When an abnormal value is detected, a notification function by telephone, e-mail, or alarm lamp will enable an early response.

(8) Collection interval

The minimum interval of collecting dose rate data and meteorological data shall be 2 min or less. However, the interval shall be possible to be set to 10 min, 1 h, etc., at any time during normal operation. The interval between collections of spectrum data should be set to approximately 10 min or more, depending on the type and size of the detector.

In addition, it is necessary to arbitrarily set an optimum collection interval while considering the amount of data and power consumption, depending on whether the situation is normal operation or in case of an emergency.

[*1] This stands for the Geographic Information System.

Chapter 4 Measuring system using a continuous monitor

4.1 Outline

The detectors described in Chapter 3 are used in a variety of systems as continuous monitors. Each detector is unique, with different advantages and disadvantages. The detectors utilized in each system are selected so as to exploit their advantages.

This chapter presents examples of how the detectors described in Chapter 3 are exploited in each system.

Note that the transportable monitoring posts described in Section 4.3 have been installed and operated in the same manner as the installed continuous monitors since the Fukushima Dai-ichi NPS accident.

4.2 Installed continuous monitor

4.2.1 Ionization chamber monitor

The pressurized ionization chamber described in Item 3.2.1 is used as an installed continuous monitor, mainly for the measurement in high-dose-rate regions. The specifications of ionization chamber monitors vary slightly from manufacturer to manufacturer, but they are nearly identical. An ionization chamber monitor consists of a detector, thermal insulation cover, temperature controller, and measuring section. The factors that affect the measured values, such as energy characteristics, are mainly the wall materials and their thickness, as well as the species of filler gases and their pressure.

The measurement principle is that charged particles incident on the ionization chamber and charged particles generated secondarily ionize gas molecules as they pass through the gas in the ionization chamber, and a current generated by a potential difference between the generated electrons and ion pairs is detected. The current value is converted into an irradiation dose (C/kg or R), and then converted into an air kerma (kinetic energy released per unit mass), air absorbed dose, ambient dose equivalent, etc., to calculate the dose rate.

Therefore, in terms of the structure, it is necessary to consider that the wall thickness of the ionization chamber is sufficient to establish the charged particle equilibrium, the strength is sufficient, and no large attenuation of incident gamma rays is identified.

In conducting the measurement of environmental gamma-ray dose rates, the contribution of self-contamination attributed to the raw materials used for manufacturing the detector, may have a significant impact on the measured values, depending on the quality of the raw materials. In terms of the presence of self-contamination, the user should understand the contribution of self-contamination in advance by asking the manufacturer or a third-party organization when the equipment is delivered.

The shapes of ionization chamber dosimeters used as continuous monitors are mainly spherical. This is intended to mitigate the influence of directional dependence on the measured values.

Table 4.1 illustrates examples of the specifications of an ionization chamber monitor.

Table 4.1 Specification examples of an ionization chamber monitor

	Specification examples of a pressurized ionization chamber			
Wall materials, thickness, capacity, and shape of ionization chamber	2 mm of stainless steel, 1.2 mm of iron, 14 L, spherical	3 mm of aluminum, 14 L, spherical	2 mm of aluminum, 14 L, spherical	3 mm of aluminum, 14 L, spherical
Species of filler gases and pressure	Argon or (and) nitrogen at 4 atmospheres	Argon or nitrogen at 4 atmospheres	Argon and nitrogen at 8 atmospheres	Argon at 6 atmospheres
Energy region of detected gamma rays	60 keV–∞			
Measurable dose rate range	10 nGy/h – 100 mGy/h			
Temperature control system	Heating or blasting type		Heating type	
Materials and thickness of a detector insulation cover	3 mm of resin	1.2 mm of aluminum or 3 mm of resin	1 mm of aluminum	1.5 mm of resin
Energy characteristics	approximately 80 keV – 3 MeV ± 30%	approximately 80 keV – 3 MeV ± 10%		
Dose rate linearity	approximately 0.4 μGy/h – 1 mGy/h ± 5%			

4.2.2 NaI monitor

This is mainly used for the measurement of the low-dose rate region; in the past, the DBM method[*1] was widely used, in most cases, for automating the dose conversion by the G(E) function method, using electronic circuits. The DBM method[*1] has a feature that the sensitivity is lowered owing to counting loss that may occur at a high counting rate, because a long time is required to perform operations in the circuit. Consequently, the effect of counting loss is currently suppressed by automatically digitally processing the G(E) function method.

The widely used NaI (Tl) scintillation detectors installed in installed continuous monitors most commonly have cylindrical shapes of 2 in. diam × 2 in. and 3 in. diam × 3 in., with a feature of producing an maximum excessive response of approximately 10% when the incident angle of a photon is approximately 30°–120°.

Table 4.2 illustrates examples of the specifications of a NaI monitor.

[*1] This stands for the Discrimination Bias Modulation method.

Table 4.2 Specification examples of a NaI Monitor

		Specifications of a NaI Monitor		
Dimensions and and shape of detector		2 in. diam × 2 in., cylindrical	3 in. diam × 3 in., cylindrical	3 in. diam, spherical
Energy compensation		• Digital processing (the G(E) function method) • DBM method		
Temperature compensation circuit		Yes		
Temperature controller		• Heating type • Blasting type		
Materials of the insulation cover		• 3.0 mm of resin • 2.5 mm of resin • 1.5 mm of resin • 2.0 mm of aluminum and 2.0 mm of resin • 1.2 mm of aluminum • 1.0 mm of aluminum		
Discri.	LLD	50 keV		
	ULD	3000 keV		
Measurable dose rate range		10 nGy/h — 10 μGy/h		
Materials for the NaI crystal optics window		• Borosilicate glass • Low-K glass • Quartz		
Materials for the photomultiplier tube window		• Borosilicate glass • Low-K glass • Quartz		

* Discri.: Abbreviation for discriminator (a pulse height discriminator)

4.2.3 Simplified electronic dosimeter

A simplified electronic dosimeter originally worked as an personal exposure dosimeter (an integrating dosimeter) and has been extensively used. Most simplified electronic dosimeters are equipped with a silicon semiconductor detector. As specified in the Nuclear Emergency Preparedness Guideline issued by the Nuclear Regulation Authority after the Fukushima Dai-ichi NPS accident, environmental radiation monitoring is required in a wide range of up to approximately 30 km from nuclear facilities, as a UPZ, and this type of dosimeter has been used as a continuous monitor in case of an emergency. In general, the dose rate calculation method converts the collected pulses into a dose rate using the conversion factor obtained at 662 keV of ^{137}Cs. Note that this approach does not allow for nuclide identification and compensation of energy characteristics, owing to the loss of energy information. Table 4.3 illustrates examples of the specifications of a simplified electronic dosimeter.

Table 4.3 Specification examples of a simplified electronic dosimeter (Reference 4)

	Specifications of a simplified electronic dosimeter
Detector	Silicon semiconductor detector
Measured dose rate range	0.2 μSv/h – 10 mSv/h (the ambient dose equivalent rate)
Measured energy range	60 keV – 1.5 MeV, gamma (X) rays
Measuring time	2 min or less (in case of an emergency) to 10 min. (during normal operation)
Working temperature environment	±20%, −10 °C to +40 °C
Operating humidity environment	±20%, 40%–90% RH, 35 °C as the standard
Energy characteristics (^{137}Cs as the standard)	−50%–30%, 60 keV to below 100 keV ±30%, 100 keV – 1.5 MeV
Dose rate characteristic (^{137}Cs as the standard)	±20%, 0.2 μSv/h – 10 mSv/h
Measurement accuracy (^{137}Cs as the standard)	±10%, 1μSv – 999.99 mSv * Only when measurement is carried out by the main body of the electronic dosimeter, using the integration method.
Directional characteristics (^{137}Cs as the standard)	±30%, 0° ± 60°
Temperature characteristic	±20%, * Within the working temperature range, +20 °C as the standard
Power source	• Primary power source: 100 VAC • Secondary power source: Battery (must be operational for 1 week or more) * Photovoltaic power generation facilities can be installed.

4.3 Transportable continuous monitor

A transportable continuous monitor is a device to be used as a continuous monitor by temporarily fixing transportable monitoring posts, etc. that can be moved and transported without being installed. Transportable monitoring posts are intended to be used in case of a nuclear disaster. The measurable dose rate range is supposed to be from a background level to hundreds of mGy/h (hundreds of mSv/h); these monitoring posts are intended to be used in areas with no installed continuous monitors installed, or used as a substitute for a measuring instrument with which it is impossible to take measurements for some reason. After the Fukushima Dai-ichi NPS accident, they have been utilized as continuous monitors, even when monitoring during normal operation, by equipping them with an external power source and a transmission system.

4.3.1 Transportable monitoring post

(1) NaI (Tl) scintillation detector

The G(E) function method or the DBM method is adopted for dose calculation.
Some of the typical devices use a NaI (Tl) scintillation detector to take measurements in the low-dose-rate regions, and use other types of detectors, such as a silicon semiconductor detector, to take measurements in the high-dose-rate regions. This is a specification to compensate for the

disadvantage of the NaI (Tl) scintillation detector that dose rates cannot be accurately evaluated owing to the effects of pile-up or counting loss when a high dose rate is detected. However, other types of devices can take measurements over the entire range of measurable dose rates using only a NaI (Tl) scintillation detector. In this case, a dose calculation method such as the G(E) function method is adopted in the low-dose-rate region. In the high-dose-rate region, the dose calculation is performed based on the current value instead of the pulse count processing, such that the measurements up to the high-dose-rate regions can be taken without being affected by counting loss. However, note that energy information is lost in this case; thus, energy characteristics cannot be compensated by means of nuclide identification, a G(E) function, or a DBM circuit.

(2) Silicon semiconductor detector

As explained in (1), there are some devices that are equipped with a NaI (Tl) scintillation detector for the low-dose-rate region and a silicon semiconductor detector for the high-dose-rate region. There are also other devices that are equipped with only a silicon semiconductor detector. Because such devices are lighter and more power-efficient than the NaI (Tl) scintillation detector, they are designed to enable also taking measurements from the low-dose-rate region by increasing the number of elements and improving the detection efficiency.

The dose calculation method is performed in the same manner as for a simplified electronic dosimeter.

Table 4.4 illustrates examples of specifications of a transportable monitoring post.

Table 4.4 Specification examples of a transportable monitoring post

	Type A		Type B		Type C	
	Low dose	High dose	Low dose	High dose	Low dose	High dose
Detector	NaI (Tl) scintillation detector	Silicon semiconductor detector	NaI (Tl) scintillation detector		Silicon semiconductor detector	
Dimensions of detector	2 in. diam × 2 in. cylindrical	-	2 in. diam × 2 in. cylindrical		-	
Measured dose rate range	BG–11 μGy/h	9 μGy/h – 100 mGy/h	0.01–500 μGy/h	300 μGy/h – 100 mGy/h	BG–100 mGy/h	
Measured energy range	50 keV – 3 MeV	50 keV–∞	50 keV – 3 MeV	50 keV–∞	50 keV–∞	
Dose calculation method	G(E) function method by digital processing	Pulse conversion system	DBM	Current measuring system	Pulse conversion system	
Collection interval	1 min or 10 min		1 min or 10 min		1 min or 10 min	
Time constant	10 s, 100 s		10 s, 60 s, 100 s		Auto toggle	
Recording system	Electronic memory system		Electronic memory system		Electronic memory system	
Power source	100 VAC and built-in battery * Solar power generation with external batteries is optionally available		100 VAC and built-in battery * Solar power generation with external batteries is optionally available		100 VAC and built-in battery * Solar power generation with external batteries is optionally available	
Dimensions	approximately 345 × 348 × 205 mm^3		approximately 670 × 440 × 450 mm^3		approximately 600 × 480 × 250 mm^3	
Weight	approximately 10 kg		approximately 15 kg		approximately 30 kg	

4.4 Mobile survey system

A mobile survey system can efficiently monitor a wide range of areas by exploiting its mobility. Mobile survey systems are classified into two types: an installed type in which a detector is installed to a vehicle called a monitoring car, and a portable type in which the system including a detector can be easily removed and installed, even in a different vehicle. The following are the detectors used in the installed type and the portable type.

Note that, in a special usage of a mobile survey, the specifications (the measurement interval, time constant, directional characteristics, etc.) suitable for the mobile survey shall be identified and utilized. For more information, see Explanation F.

4.4.1 Installed type

(1) NaI (Tl) scintillation detector

The type of devices commonly used are those with the same specifications as the installed continuous monitors described above. The typical shape of the detector is also cylindrical, with dimensions 2 in. diam × 2in.

(2) Silicon semiconductor detector

This is used for measuring the high-dose-rate areas, as with transportable monitoring posts.
The specifications are the same as those of transportable monitoring posts.

4.4.2 Portable type

Currently, CsI (Tl) scintillation detectors are used as one of the representative types of transportable mobile survey systems. The complication of installing this type of device is eliminated by exploiting its features of light weight and small size, and housing the unit, including a transmission system, in a small, robust, and dedicated case.
Tables 4.5 and 4.6 illustrate examples of specifications of the installed type and the portable type.

Table 4.5 Specification examples of a mobile survey system (Installed type)

	Specifications of a mobile survey system (Installed type)	
	Low dose	High dose
Detector	2 in. diam x 2 in. cylindrical NaI (Tl) scintillation detector	Silicon semiconductor detector
Measured dose rate range	10 nGy/h – 10 μGy/h	10 μGy/h – 100 mGy/h
Measured energy range	50 keV – 3 MeV	50 keV–∞
Calculation method for energy compensated dose rates	G(E) function method	Pulse conversion
Housing for the detector	Detector cover, including a mounting table for the source calibration jig	
Measuring time	1–99999 s, arbitrary setting is allowed	
Temperature control	Circulating the vehicle interior air	
Data collection system	(1) PC for data collection (2) Data collection and processing software (3) Printer (4) Communications system for data transmission (5) GPS	
Measures against vibration	Installation of a vibration isolator	
Power source	Equipped with a generator for measuring instruments * Output is approximately 4500 W.	
Distribution panel	A breaker is installed by separating the power sources for the main breaker and for each measuring system.	

Table 4.6 Specification examples of a mobile survey system (portable type)

	Specifications of a mobile survey system (portable type)
Detector	CsI (Tl) scintillation detector 13 mm × 13 mm × 20 mm
Photo detection element	Multi-pixel semiconductor
Measured dose rate range	1 nSv/h – 300 μSv/h
Measured energy range	30 keV – 2 MeV
Counting efficiency	40 cpm or more at 0.01 μSv/h of ^{137}Cs
Energy resolution	8 % ^{137}Cs 662 keV
Calculation method for energy compensated dose rates	G(E) function method
Housing for the detector	Aluminum storage box 175 mm × 345 mm × 240 mm With an exhaust fan for heat removal
Measuring time	1–60 s, arbitrary setting is allowed
Operating temperature range	−10 °C to 50 °C
Data collection system	(1) Embedded computer (2) Data collection and processing software (3) Communications system for data transmission (4) GPS
Power source	12 V / 24 VDC power source * Power supply shall be allowed from an automobile accessory socket and an accessory power source.

Chapter 5 Installation of a measuring system

5.1 General considerations of the installation of a continuous monitor

The following points should be considered when installing a continuous monitor.

(1) Select a location for installing a monitor considering the site of the nuclear facility, the distribution of wind direction, the distribution of towns and villages, topography, etc.

(2) Select a location where the surroundings are open, which is easily accessible, and where the drainage is good. In addition, from the viewpoint of drainage, a location with adjacent paddy fields should be avoided as much as possible. Because paddy fields are filled with water or dried depending on the seasons, the environmental gamma-ray dose rate might fluctuate. Even in fallow areas, owing to the characteristics of paddy fields, the water retention rate is high, and the soil may remain moist for a long time because of rainfall.

(3) Avoid locations in adverse environments with high temperature, high humidity, etc.

(4) Minimize the shielding effect caused by the topography, buildings, and structures around the detector.

(5) When multiple continuous monitors are used concurrently at a same point, they must be installed as close as possible, to the extent that they do not interfere with each other, and the conditions such as the ground height must be uniform.

(6) The location where a continuous monitor is installed should be surrounded with fences so that the dose rate does not fluctuate owing to human effects.

(7) When data transmission is performed using radio waves, sufficient attention must be paid to the site of the radio station, directivity of the antenna, and preventive measures against the sneak of radio waves from the power supply circuits and cables, considering the effect of the emitted radio waves.

(8) In case of a nuclear disaster involving a large-scale natural disaster, the infrastructure such as the power supply may be destroyed. Assuming such a case, two or more power supply systems should be provided by installing a private power generator or a solar power generation system together. However, generators need to secure fuel (light oil, LPG, etc.) that can be continuously supplied, and solar power generation needs to secure a backup power source during periods of insufficient sunlight (at night, in bad weather, etc.). In addition, satellite communications, etc., which are less affected by disasters, should also be provided, assuming that the means of data transmission might be cut off.

(9) Unexpected lightning may cause the malfunction of electrical systems of the measuring equipment through AC power sources. To avoid such a situation, it is desirable to install an arrester or similar.

(10) During the Great East Japan Earthquake, continuous monitors were damaged by the tsunami after the large-scale earthquake and became unusable; thus, when continuous monitors are installed at a site near the sea, consideration should be given to installing them on a hill not affected by a tsunami. However, if a village is located along the coast, it would be unavoidable to install a continuous monitor at a site near the sea. As a result, if the monitor should become unusable, a transportable continuous monitor is required as a substitute.

(11) In terms of the ground height for installing a detector, it should be considered as a matter common to all types of detectors and all installation conditions, in addition to considering the purpose of measurement, that a detector should be installed at a height allowing easy maintenance, because the adjustment of equipment is required, including periodical gain adjustment.

5.2 Installed continuous monitor

5.2.1 Ionization chamber monitor

Ionization chamber monitors, which are used as installed continuous monitors, have certain points to be considered specific to them, other than those described in Section 5.1. The following are the considerations specific to an ionization chamber monitor.

(1) An atmospheric pressure-type ionization chamber should be installed in an instrument shelter or similar, to avoid direct sunlight.

(2) If a monitor that was installed fails to measure, a transportable monitoring post must be used for compensation.

(3) The ground height for installing an ionization chamber monitor should be decided by selecting an installation environment suitable for each purpose, with clear intention.

[1] In an alley, at a height of 1 m above the ground

A height of 1 m above the ground is the basic ground height for measurement that contributes to the evaluation of exposure dose to the human body. This mainly intends to measure gamma rays from radioactive materials deposited on the ground surface.

[2] In an alley, at a height of approximately 1.5–1.8 m above the ground

This is a ground height with the same purpose as the height of 1 m above the ground, as mentioned in [1]. However, in heavy snow areas, it is assumed that detectors will be buried in the snow. As a result, this ground height is set for ensuring effective measurement, even in winter.

[3] On a monitoring building

A monitor is installed at a height of 0.6–1.0 m above the roof of a monitoring building (at a height of approximately 3 m above the ground). The height above the monitoring building is selected to avoid gamma rays from radioactive materials floating in the sky being shielded, which occurs when a monitor is installed in an alley beside the building. Installing a monitor on the building has the advantages that the area occupied can be reduced and the monitor can sensitively respond to radioactive plumes floating in the sky. However, note that the measurements include the contribution of radioactive materials contained in the roof materials of the building and a wide range of areas in the alley.

[4] At a height of dozens of meters above ground

Among the continuous monitors installed so far, some have been installed on the roofs of buildings of height, for example, three to five stories above the ground, in order to take average measurements over a wide range of areas, especially for measuring fallouts. This has the advantage of being sensitive to radioactive plumes, as with the case of installing on a monitoring building, as mentioned in [3].

Figures 5.1 through 5.4 show examples of the installation of an installed continuous monitor under various conditions.

5.2.2 NaI monitor

The NaI monitor, which is used as an installed continuous monitor, has the same points to be considered as those mentioned in Items (2) to (3) of Section 5.2.1, in addition to those of Section 5.1.

Figure 5.1 Example of installation on an alley, at a height of 1 m above the ground (Aichi Prefecture)

Figure 5.2 Example of installation on an alley, at a height of 1.5–1.8 m above the ground (Aomori Prefecture)

Figure 5.3　Example of installation on a monitoring building (Ibaraki Prefecture)

Figure 5.4　Example of installation at a height of dozens of meters above the ground (Aichi Prefecture)

5.2.3 Simplified electronic dosimeter

A simplified electronic dosimeter, which is used as an installed continuous monitor, has specific points that must be considered, in addition to those mentioned in Section 5.1. The following are the points specific to a simplified electronic dosimeter to be considered.

Simplified electronic dosimeters have been introduced to cover areas where monitoring is not conducted, because the monitoring scope has been expanded based on the experience acquired after the Fukushima Dai-ichi NPS accident.

(1) Rack

Use a robust and durable material such as stainless steel. However, because it acts as a scatterer and shield, consideration must be given to making it as thin as possible. The rack is equipped with a detector storage box, transmission equipment, data logger, battery storage box, and photovoltaic power generator. A storage box containing a detector is installed at the front of the rack so that the effective center of the detector is positioned at a height of 1 m above ground. Therefore, the battery storage box should not be installed on the front side of the rack, so as not to act as a shield. Photovoltaic power generation equipment must be installed so as not to cover the top of the detector.

(2) Data transmission

Generally, the time series measured data taken during a short period of time must be stored in the measuring system, and it must be separately output as a data file. However, to be used as a continuous monitor, data should be collected in real time, as with the installed continuous monitor as used in an ionization chamber monitor and a NaI monitor, such that the measurements taken can be confirmed. As a result, if a data transmission system is not provided, additional equipment is required. Examples of a data transmission system are shown below:

- Mobile phone lines
- Satellite mobile phone lines

In using a mobile phone line, if the installation status of the radio wave base station and the coverage for communications were understood in advance, data transmission would be ensured in a relatively stable radio wave condition. Because each provider has its own base station for mobile phone lines, the radio wave condition depends on the installation environment. In the current situation of infrastructure, where mobile phones are widely used, a relatively stable radio wave condition can be achieved in densely populated areas such as urban areas, regardless of the provider selected. However, in sparsely populated areas such as mountainous areas, you may not be able to acquire stable radio wave conditions, depending on the providers. Additionally, you may not be able to use your mobile phone line, even when you select any provider.

In such a situation, a satellite mobile phone line that can be used even in case of an emergency, regardless of the existence of a radio wave base station, should be equipped or installed together as an alternative.

Because satellite mobile phone lines are expensive, compared with mobile phone lines, the data transmission method should be selected based on the radio wave condition of each line.

(3) Driving power source

A simplified electronic dosimeter can usually be driven by providing an external AC power source. A model equipped with built-in batteries can be driven within a time period corresponding to their capacity. However, if you want to use it as a continuous monitor, you need equipment that can continuously supply power for a certain period of time. Because a power supply is also required in an attached system such as a data transmission system, it is difficult to supply power only with the built-in battery. Consequently, the following power supply methods should be used in combination.

[1] AC power source

An AC power source is advantageous in that it can stably supply power. However, it may not be easy to secure AC power source in some cases. In addition, even in a situation where AC power source can be used, when power is supplied from an outdoor power outlet used in a general household, it is desirable to install an arrester, such as those described in Section 5.1.

[2] External battery

An external battery should be provided as a standby power source so that it can be used if an AC power or other supply power should be unavailable. Batteries currently in use, such as ship batteries, are of a large capacity and some of them have a maximum driving capacity of approximately 7 days by connecting two batteries in series. Even if an external battery becomes exhausted, a function to charge it when the other power supply is restored should be provided. Since an external battery is physically large, consideration should be given to its positioning, such as placing it directly under the measuring instrument body, so that it does not become a shield for gamma rays to be measured.

[3] Photovoltaic power generation

Photovoltaic power generation is a method that ensures a relatively stable power supply over a long period of time even, in areas where no AC power supply is available, such as in mountainous areas with no power cables, or in situations where the infrastructure is unusable owing to power outages, etc. When this method is employed, sunlight (ideally direct sunlight) must be secured. In terms of measurement, an open environment is desirable for installation. However, even if there were no objects to block the sunlight at the time of installation, the situation may change depending on the growth of trees and construction of new buildings. When installing a generator, attention should be paid to the period of installation and prediction of changes that may occur in the surrounding conditions.

Figure 5.5 shows an example of the installation of a simplified electronic dosimeter.

Figure 5.5　Example of the installation status of a simplified electronic dosimeter (Shimane Prefecture)

5.3 Transportable continuous monitor

Based on the fact that, after the Fukushima Dai-ichi NPS accident, transportable monitoring posts among transportable continuous monitors have been installed and operated in the same manner as installed continuous monitors, the points to be considered in installing them are summarized as follows:

5.3.1 Transportable monitoring post

A transportable monitoring post, which is used as a transportable continuous monitor, has points to be considered specific to them, apart from those described in Section 5.1. The following are the points to be considered specific to transportable monitoring posts:

(1) Rack

The transportable monitoring posts currently on the market are structured to be installed at a height of 1 m above the ground by installing a rack or similar at the lower part of the equipment. Because the attached rack is simple, a more robust rack is required when the monitor is used as a continuous monitor for a long period. However, if an excessively robust rack is installed in the lower part of the measuring instrument or in its vicinity, it would act as a scatterer and shield of gamma rays from radioactive materials deposited on the ground surface. Therefore, it must be made as thin as possible. When selecting a rack, the influence of the shielding effect should be considered; it is also important to understand the degree of the shielding effect of the rack by comparing with the measured values taken before installation and after the whole system is installed, including the rack.

(2) Data transmission

This is performed in the same manner as a simplified electronic dosimeter.

(3) Driving power source

This is performed in the same manner as a simplified electronic dosimeter.

When a photovoltaic power generation facility is employed, the measuring instrument itself would receive direct sunlight by giving priority to the sunshine condition. Although a transportable monitoring post is equipped with a temperature compensation circuit, it is rarely equipped with a temperature control device, unlike an installed continuous monitor. As a result, the response may significantly vary owing to the failure of the temperature compensation circuit to respond to a sudden temperature change. To secure a stable power supply for a long term, solar power generation facilities should be effectively exploited, and sunlight is essential. However, this is not a desirable condition for the measuring instrument.

Consequently, it is important to understand that there may be a certain effect of temperature changes as a variation factor of the measured values. Details are given in Explanation E.3.

Items apart from those above may be considered to be used in the same way as with the installed continuous monitor. Figure 5.6 shows an example of the installation status of a transportable monitoring post.

Figure 5.6 Example of the installation status of a transportable monitoring post (the positioning of the detector: 1 m above the ground, with solar power generation and external battery facilities)

5.4 Mobile survey system

The system used for mobile surveys has some specific points to be considered. The following points are specific to the mobile survey system:

In this section, the points to be considered are described by giving a typical monitoring car as an example of the installed type and giving a measuring device as an example for the portable type, which was developed after the Fukushima Dai-ichi NPS accident and has been utilized since then.

(1) Installed type

An installed mobile survey system refers to a so-called “monitoring car.” Therefore, the detector is classified into two types: an exterior type to be installed above the vehicle, and an interior type to be installed at a predetermined position in the vehicle, so as to be installed at a height of 1 m above the ground.

Figure 5.7 shows a typical example of the external installation.

The detectors outside the vehicle are often installed at a height of approximately 2 m above the ground.

The power supply should be secured when installing measuring equipment. A typical method is to supply power from a vehicle battery, but in this case, the battery capacity needs to be larger than that used only for the vehicle. In addition, because the power supply voltage is unstable when the battery is being charged from the power generation function (dynamo) of the vehicle, the battery is not suitable as is, as the power source for a precision instrument. In such a case, an uninterruptible power supply (UPS) or similar should be connected to ensure a stable power supply voltage.

The interior-type detector is installed at a height of approximately 1 m the above ground in the vehicle.

Because the detector is installed in the vehicle, the shielding effect of peripheral equipment and the vehicle body itself should be evaluated. The geometrical conditions of the detector and its peripheral equipment must always be the same to ensure that the influence of the shielding effect is constant, and objects that can act as shields should not be loaded while taking measurements. The matters concerning the power supply are the same as those of the exterior-type detector mentioned above.

Figure 5.7 Example of an installed type (exterior-type) (Reference 5)

(2) Portable type

The portable travelling survey system has widely been in use after the Fukushima Dai-ichi NPS accident, and it is expected that more monitoring organizations will consider introducing it in the future.

A typical example of the portable type is shown in Figure 5.8.

Detectors used in the portable type include a system in which a commercial survey meter is temporarily installed in the vehicle, and a system in which it is stored in a special container for the purpose of mobile surveys and which can be loaded on a variety of vehicles. In both systems, the matters concerning the installation are based on the following concept:

The system, including the detector, should be installed and installed at any position in the vehicle. In order to use the measurements taken in the vehicle as the dose rate outside the vehicle, it is necessary to use the correction coefficient of both the outside and inside of the vehicle in accordance with the installation status, and the detector should be installed while taking measurements, to avoid changing of its position. When a commercial survey meter is used, the power supply source is should be used in accordance with the specifications of the survey meter. Generally, it is driven by dry batteries. In addition, for systems designed for mobile surveys, some mobile survey systems are currently powered from the vehicle battery by connecting to the vehicle’s accessory socket.

Figure 5.8 Example of a portable type (left: when installed; right: inside of the storage case) (Reference 6)

Chapter 6 Measurement and calibration

6.1 Precautions for use

A continuous monitor has a complex mechanism, is left unattended for long periods in harsh outdoor environments, and is furthermore required to minimize missing measurements. To satisfy these strict conditions, a well-tested product should be used, and daily thorough maintenance and management is also required.

Of the many items to be considered when using a continuous monitor, the following are the major points:

(1) Effectively understand the characteristics of the monitor to be used (energy characteristics, directional characteristics, temperature characteristics, etc.), and acquire sufficient information on the structure of the detection unit and the functions of the electronic circuit from the manufacturer, if possible. In the event that the performance of the monitor is seems inadequate for the purpose of monitoring, appropriate actions shall be taken.

(2) Handle the monitor correctly according to the instructions.

(3) Conduct inspections and calibrations at regular intervals, as necessary.

(4) Because the detection unit is generally very vulnerable to impacts, extreme care should be taken when handling.

(5) As the detection unit is supplied with a high voltage, the high-voltage terminals and similar should be periodically cleaned, because they may cause electrical leakage owing to moisture and dust adhesion, thereby causing noise. If a desiccant is used, replace it periodically.

(6) For a model containing an arrester (a lightning arrester) to prevent accidents caused by lightning strikes, regular inspections should be conducted, especially in areas where lightning strikes occur frequently.

(7) The measurement results of a continuous monitor include various elements such as the installation status and measurement condition of the monitor; therefore, clarifying these elements together with the measurement results will give us the information contained in the measurement. Examples of recording formats for installed and transportable continuous monitors are illustrated in Table 6.1.

In addition to the items common to the monitors mentioned above, the following items are to be noted with respect to the monitors for each method.

[1] Ionization chamber monitor

The ionization chamber monitor periodically checks for zero points (some are automated). Additionally, pay attention to pressure changes due to the breakage of the sealing. Changes in pressure can be detected by a significant decrease in the readings in a pressurized ionization chamber, and by a dependence on atmospheric pressure in an atmospheric pressure chamber.

[2] NaI monitor

Because a NaI monitor can generally exhibit certain temperature characteristics, attention should be paid to diurnal and seasonal changes in the indications, and appropriate actions should be taken if necessary.

[3] Simplified electronic dosimeter

Attention must be paid to temperature characteristics. Many of the simplified electronic dosimeters also have a silicon semiconductor detector mounted. In this case, it should also be noted that the effect of the energy characteristics is larger than that of other detectors.

[4] Transportable monitoring post

This has temperature characteristics depending on the type of the measuring instrument installed, as with a NaI monitor and simplified electronic dosimeter. The effect of the characteristics on the measured values is described in Explanation E.3. Owing to the nature of the transportable monitoring post, this is a simplified device that is not equipped with a temperature control device. Therefore, a variation factor due to temperature changes is included as a representative factor for measured values.

[5] Mobile survey system

Unlike a transportable monitoring post and others, because the detector and the ambient temperature of the detector, which affect the measurements, are controlled to some extent by circulating air in the vehicle, they do not greatly affect the measured values. Because of the nature of the survey purpose using a mobile survey system, the coverage for measurement or the directional characteristics accounts for an important variation factor. It is important to carry out the measurement with the understanding of the directional characteristics, including the vehicle that is the means of travel.

In addition, it can be easily imagined that a mobile survey system is utilized in the middle of an emergency. In such a case, if the vehicle is used for monitoring, the tire house of the vehicle and the air filter may become contaminated, and the background may increase. To reduce this effect, take a measurement at a same point where the dose rate does not vary before starting measurement and after finishing it, obtain the contamination level of the vehicle, and conduct an evaluation by subtracting it from the measured value, as required.

Table 6.1 Example of the recording format for installed and transportable continuous monitors

Report on the survey results of the environmental radiation dose rate by continuous monitoring

Measurement location	Address	
	Latitude	: :
	Longitude	: :
Name of the point		
Identification of the monitor		NaI monitor, ionization chamber monitor, others ( )
Height above ground (m)		
cosmic rays (with / without)		With / without

Survey year	
Research laboratory in charge	
Chief investigator	
Contact	

Date of measurement	Weather	Environmental dose rate					Remarks
		Average value	Standard deviation	Unit at the time of measurement	Measuring time	Counts of data	
				nGy/h (air absorbed dose rate)			
				nSv/h (ambient dose equivalent rate)			
				nGy/h (air absorbed dose rate)			
				nSv/h (ambient dose equivalent rate)			
				nGy/h (air absorbed dose rate)			
				nSv/h (ambient dose equivalent rate)			
				nGy/h (air absorbed dose rate)			
				nSv/h (ambient dose equivalent rate)			
				nGy/h (air absorbed dose rate)			
				nSv/h (ambient dose equivalent rate)			
				nGy/h (air absorbed dose rate)			
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				nSv/h (ambient dose equivalent rate)			
				nGy/h (air absorbed dose rate)			
				nSv/h (ambient dose equivalent rate)			
				nGy/h (air absorbed dose rate)			
				nSv/h (ambient dose equivalent rate)			

6.2 Installed continuous monitor

6.2.1 Calibration and adjustment

"Adjustment" refers to the act of changing various settings appropriately to keep the equipment healthy, and as a result, JIS specifies that a measured value is acceptable if it is within the allowable range of the "relative reference error ± 20%."

On the contrary, "calibration" is the act of providing a calibration constant necessary to correct the difference between a measured value and a certified value from the ratio between the measured value and the certified value by securing traceability with national standards.

As mentioned above, adjustment and calibration are different acts, and adjustment does not substitute for calibration. However, because it is difficult to conduct calibration on a continuous monitor that is already installed, adjustment is performed instead, in practice.

As a requirement for the measured values of a continuous monitor, if it is treated as a "monitor," the integrity for satisfying this requirement is ensured by performing adjustment. However, if the data contributing to external exposure dose assessment are supposed to be obtained, calibration should be conducted, by which traceability with national standards can be secured. Reference gamma-ray sources used for calibration work are described in Explanation C.

(1) Calibration

At the place where the monitor is installed, it is difficult to perform a calibration matching the one conducted at the irradiation field, or conduct a test that corresponds to the acceptance test by the manufacturer. However, at the place where the monitor is installed, it is possible to conduct a calibration by using a standard with clear traceability by national standards, or conduct a practical calibration by employing the source method using the reference gamma-ray source stipulated in JIS Z 4511: 2005, or the inverse square law.

For the reference gamma-ray source, the value of the irradiation dose rate at a point 1 m from a normal source is certified. If the line connecting the source and the detector center is parallel to the ground surface or to the floor surface, the lower the height of the source and the detector from the ground surface, and the larger the distance of the detector from the source, the larger will be the contribution rate of scattered gamma rays, within a certain range. On the contrary the higher the height and the smaller the distance, the smaller will be the contribution rate (see Explanation D). However, when the distance becomes very short, the change in the effective center cannot be ignored.

Major calibration conditions and methods are described below:

[1] The nuclides for the reference gamma-ray source used for monitor calibration shall be ^{60}Co or ^{137}Cs. However, ^{226}Ra may also be permitted.

[2] The calibration direction shall be horizontal or vertical to the central axis of the detector. Further, a direction with less effect from scatterers in the vicinity should be selected.

[3] The source should be installed at a distance equal to the calibration distance of the reference gamma-ray source above the calibration direction from the effective center of the continuous monitor detector. In addition, the contribution of scattered rays should be considered in accordance with the height and distance, and corrections should be made if necessary (see Explanation D).

[4] The distance between the reference gamma-ray source and the detector is preferably 1 m or more. Typically, the distance between the source and the detector is 1 m.

[5] Weather conditions that usually affect calibration results should be avoided.
Specifically, weather conditions such as rainfall, in which the background dose rate varies during a calibration operation, must be avoided.

[6] Ensure that the performance of the monitor is sufficiently stable.

[7] The calibration constant K is calculated from the following equation:

$$K = \frac{\dot{D}}{\dot{Q} - \dot{Q}_b}$$

where, $\dot{D}$: Dose rate of a standard gamma-ray source during calibration

$\dot{Q}$: Indicated value during irradiation with a standard gamma-ray source

$\dot{Q}_b$: Indicated value by dose rate of environmental gamma rays

[8] Calibrations are desirably conducted four times a year or every quarter, considering the changes in the weather and environment of the four seasons. However, if four times a year is not practicable, the integrity of the equipment can be ensured by calibrating at as short intervals as possible using a confirmation source.

[9] The date and time of calibration, calibration conditions, and results shall be recorded. When calibration is performed at many points, the effect of scattered ray varies owing to the difference in the surrounding environment at each point. Therefore, monitoring organizations should preferably conduct a comparative calibration with the monitor to be calibrated at the installation site, by using a well calibrated dose (rate) meter as the reference measuring instrument in an environment with little scattering, if possible.

(2) Adjustment

In adjusting a monitor, individual equipment and settings should be checked. Items of inspection and adjustment are shown below:

[1] Visual inspection and cleaning of components

[2] Insulation resistance

[3] Temperature indication accuracy and alarm operation of the heating system

[4] Accuracy of power supply voltage

[5] Checking of the accuracy of dose rate indication and the scaler operation

[6] Confirmation of zero points

[7] Gain adjustment

[8] Source irradiation test

6.2.2 Ionization chamber monitor

(1) Determination of the measuring time

Ionization chamber monitors are often set up for measuring high dose rates. Approximately 2 min is sufficient for the integrated count time for one lot of data of the measured values taken at a high dose rate, and approximately 2 min is also desirable for identifying the variation in measured values in case of an emergency. During normal operation, a maximum of approximately 10 min is appropriate as the integrated count time for one lot of data, considering the dose rate level of natural radiation in Japan.

(2) Data processing

[1] Recording

Data recording is performed as described in Section 3.3. The recorded data contain the measurements for dose rate.

[2] Processing

Data processing is performed as described in Section 3.4. Mean, maximum, and minimum values are calculated and analyzed, and outliers are extracted.

6.2.3 NaI monitor

(1) Determination of the measuring time

When a 3 in. diam × 3 in. NaI (Tl) scintillation detector is used, the measuring time of a few minutes is sufficient to obtain adequate accuracy for the integrated count time for one lot of data of the measured values, judging from the variation in statistical errors in the count values. However, in practice the measuring time should be determined by considering the variability.

To distinguish the variations between natural radiation and facility-induced radiation based on the difference in the cycle of variations, it is appropriate to determine the integrated count time for one lot of data, in the range of a few minutes to 10 minutes, for the reasons described in Explanation B.

To reproduce the variation pattern of dose rates more faithfully, a method can be employed in which the integrated time is set to approximately 2 min and a smoothing or averaging operation is performed in the subsequent data processing.

(2) Data processing

[1] Recording

Data recording is performed as described in Section 3.3. The recorded data contain measurements for dose rates, count values, etc. In the case of a system for obtaining spectral information, spectral data are also recorded.

[2] Processing

Data processing is performed as described in Section 3.4. Mean, maximum, and minimum values are calculated and analyzed, and outliers are extracted.

6.2.4 Simplified electronic dosimeter

(1) Determination of the measuring time

The concept of the measuring time in a simplified electronic dosimeter is the same as explained in Subsection 6.2.3.

(2) Data processing

[1] Recording

Data recording is performed as described in Section 3.3. The recorded data contain the dose rate measurements. If power is supplied by an external battery, monitoring of battery voltage is also essential, and this information should also be transmitted and recorded.

[2] Processing

Data processing is performed as described in Section 3.4. Mean, maximum, and minimum values are calculated and analyzed, and outliers are extracted.

6.3 Transportable continuous monitor

6.3.1 Calibration and adjustment

The concept of calibration and adjustment in a transportable continuous monitor is the same as explained in Subsection 6.2.1. In the case of a system in which energy information is not available, adjustment, etc. are performed in the comparison of measured values with certified values for the source (the irradiation value).

6.3.2 Transportable monitoring post

(1) Determination of the measuring time

The concept of the measuring time for a transportable monitoring post is the same as described in Subsection 6.2.3.

(2) Data processing

[1] Recording

Data recording is performed as described in Section 3.3. The recorded data contain measurements of dose rates, count values, etc. In the case of a system in which spectral information is obtained, spectral data are also recorded. If power is supplied by an external battery, monitoring of battery voltage is also essential, and this information should also be transmitted and recorded.

[2] Processing

Data processing is performed as described in Section 3.4. Mean, maximum, and minimum values are calculated and analyzed, and outliers are extracted. However, the transmission interval of data during normal operation may be set by considering the power consumption and the allowable amount of power provided.

6.4 Mobile survey system

Owing to the nature of a mobile survey, measurements are often taken while moving at all times, rather than at rest. One way of using a monitoring car is to take measurements at installed points while stopped; however, because the concept of the installed-point measurement is similar to that of an installed continuous monitor and a transportable continuous monitor, only the method of taking measurements while moving is mentioned here.

6.4.1 Calibration and adjustment

If a survey meter or similar is used, periodic calibrations must be conducted as with the case of normal usage. In addition, the adjustment required to secure the integrity of the measuring instrument is performed in the same manner as with the equipment used in an installed station, etc.

In the case of a system in which a measuring instrument is installed in a vehicle, the measured value obtained must be converted into a dose rate of 1 m above the ground outside the vehicle. Examples of the calculation method for correction factors of both the inside and outside a vehicle are provided in Chapter 7, “Analysis and evaluation of measurement results.”

6.4.2 Determination of the measuring time

In the case of a mobile survey, the shorter the interval between measuring times, the closer the intervals at which the measured values on the travel route can be obtained. This is because the shorter the interval between measuring times, the more sensitively the variation in dose rates in the vicinity of the travel route can be identified. However, as the interval between measuring times becomes longer, the average dose rate distribution in the vicinity of the travel route can be obtained more accurately.

In practice, the mobile speed during a mobile survey is set to the speed limit stipulated by law. The relationship between the mobile time and the migration length is described in Explanation F.2. Because the detection efficiency differs depending on the type of the detector, variation in the measured values obtained is important information to set the measuring time. Explanation F.2 shows the variation in the measured values, which is defined in the general measurement specifications used in a mobile survey system.

6.4.3 Processing of data

With regard to the systems developed as monitoring cars, the measured values are generally checked inside the vehicle in real time, using a rate meter[*1] or similar. These data are accumulated as a digital file and stored in a data collection system and a PC for control. In addition, with the recent improvements in the accuracy of a GPS[*2] accompanied by the advances in equipment, real-time mapping has been widely used. The mapping data are collected and managed by a cloud server or similar, such that the data can be checked anytime and anywhere, as long as an accessible tool is available at hand.

If a commercial survey meter is used, a transportable mobile survey system is also available, in which data can be checked anytime and anywhere, with an accessible tool available at hand, by collecting and managing the mapping data with a cloud server, etc., as with the monitoring car mentioned above, in addition to checking the data with the display of the survey meter.

[*1] A device that displays the dose rate and counting rate information in analog quantities on the meter.
[*2] GPS: Abbreviation for "the Global Positioning System"

Chapter 7 Analysis and evaluation of measurement results

7.1 Outline

As the concept common to all systems, it has been general until now that continuous measurement records of environmental gamma rays were analyzed by extracting significant variations against the mean values. This chapter describes typical analysis and evaluation methods and the concept during normal operation, as well as the handling of the monitoring data. In addition, during and after the Fukushima Dai-ichi NPS accident, certain situations have been identified in which this concept does not hold true; thus, analysis and evaluation methods for emergency and post-accident monitoring are also described.

7.2 Analysis and evaluation during normal operation

The "Environmental Radiation Monitoring Guideline" (Reference 7), which was provided by the former Nuclear Safety Commission, states that the purpose of monitoring during normal operation is to "protect the health and safety of residents living in the vicinity of nuclear facilities, to confirm that the environmental exposure dose of residents living in the vicinity of nuclear facilities due to radioactive materials or radiation is sufficiently below the annual dose limit, and to provide the results to the residents living in the vicinity." In addition to this purpose, "Grasping the status of accumulation of radioactive materials in the environment," "Early detection of unexpected releases of radioactive materials or radiation from nuclear facilities and assessment of their impact on the surrounding environment," and "Establishment of a system for environmental radiation monitoring in case of an incident or an emergency" are described. To achieve these objectives, examples of analysis and evaluation methods are presented below:

7.2.1 Calculation of mean values and standard deviation

Continuous measurement records of environmental gamma rays extract and analyze significant changes against the mean values. Therefore, with regard to the measurement results over a long period of time, the average and dispersion of the measured values per unit time are obtained. The unit time shall be set to 1 h and shall be shortened as necessary.
Take an arithmetic average of the measurements in the unit time. When each of the n measurements is calculated by mi , the average $\bar{m}$ is determined by $\bar{m} = \sum_{i=1}^{n} mi/n.$
The variance σ^2 is calculated by $\sigma^2 = \sum_{i=1}^{n}(\bar{m} - mi)^2/(n-1)$, and the square root σ is set as the standard deviation.
The average and standard deviation depend on which population is focused upon. The selection of a population is described in Explanation G.1.

7.2.2 Extraction of outliers

Outliers are identified by determining whether the measured value in the unit time is excessive or insufficient with respect the average value according to a given standard, considering the distribution of these values. There are many ways to adopt a standard, but in this measurement method, a measured value exceeding $\bar{m} \pm 3\sigma$ is identified as an outlier. As described in Explanation G.2, the probability of exceeding this value varies depending on the form of the population distribution; for example, the probability is 0.3% for a normal distribution and 2% for an exponential distribution.
In addition, because many NaI monitors, which are now widely used, can acquire energy information, it is also effective to extract outliers by using this information.

As a specific example, focusing on the fact that most of the artificial radionuclides released in the case of a nuclear disaster are radionuclides with photon energy in the low-energy region (approximately 80 –1400 keV), the ratio between the counting rate and the total dose rate (the total dose rate/the counting rate in the low-energy region) between these energy regions is always calculated, and if the value is lower than the ratio calculated in the normal period, the effect of the artificial radionuclides is sometimes observed; however, if the value is higher than the ratio calculated in the normal period, the effect of natural radionuclides due to rainfall, etc. is observed.

In addition, in a system in which the dose rate calculation is carried out by the G(E) function method using a NaI monitor, a measure of the increase in the dose rate owing to the contribution of artificial radionuclides can be understood, by weighting the count values of the energy pulse-height distribution obtained by the NaI detector according to the G(E) function, in which the conversion coefficient differs for each energy in order to compensate for the energy characteristics, and using the “pass rate” calculated by comparing the count values obtained before and after the weighting operation.

It is possible to confirm that the existing monitor is operating satisfactorily and obtaining the correct measured values through a comparative measurement using a portable measuring instrument (NaI (Tl) scintillation spectrometer, etc.) that performs calibration and adjustment or secures sufficient integrity. A description of the comparative measurement is given in Explanation H.

7.2.3 Analysis of the cause of outliers

Measured values that are identified as outliers because of exceeding 3σ include normal values at the rate as shown above. In addition, in continuous observation, environmental radiation fluctuates for a specific period owing to natural phenomena and artificial factors, and outliers are observed at a rate exceeding the statistical probability. This should be considered when analyzing the causes of outliers. In addition to the statistical probabilities that depend on the size and distribution of a population, the main factors that produce outliers include: (i) precipitation, (ii) snow cover, (iii) other natural phenomena such as variations in radiation levels due to inversion layers, etc., (iv) nuclear explosion experiments in the atmosphere, (v) artificial radiation sources such as nuclear facilities, and (vi) failures in measuring instruments.

To analyze these factors, the following should be done.

(1) Obtain information on each factor.

(2) Select a measuring time interval.

(3) Select a population.

Explanation G gives examples of selecting a population, the causes of major outliers, and defining a time interval required for these analyses. Figure 7.1 shows an example of a flowchart for outlier analysis.

The recently developed continuous monitors have, in some cases, a capability of obtaining spectral information concurrently with the dose rate measurement. In such a case, the dose rate contribution for each nuclide can also be evaluated by spectral analysis. Analysis and evaluation methods are described in Chapter 10.

7.3 Analysis and evaluation in case of an emergency

This section describes the data that may be measured in situations where radioactive materials may have been released from nuclear facilities.

7.3.1 Nuclides to be measured in case of an emergency and after an accident

In case of an emergency, the nuclides to be measured are different between when a radioactive plume passes through and after radioactive materials such as radioactive cesium are deposited on the ground by fallout, etc. Although the final goal of environmental gamma-ray dose rate measurement is not to evaluate the concentrations and dose rate for each nuclide, it should be understood that the measured values may be overestimated owing to the difference between the nuclide and its energy, because the detector has energy characteristics.

Recently, with the improvement in electronic equipment, monitors can increasingly obtain energy spectrum information. In case of an emergency, this energy spectrum information can be effectively used to identify the artificial radionuclides that contribute to the dose rate.

Explanation I presents an example of a situation in which the dose rate has increased in case of an emergency and after an accident.

After an accident, the dose rate decreases with time owing to the physical attenuation of artificial radionuclides, weathering effect, and penetration of the soil in the direction of vertical distribution. Therefore, outliers can be quickly extracted by performing the evaluation shown below:

- Update the threshold at an interval of approximately 1 month, considering the decrease in dose rate due to physical attenuation, etc.
- Acquire the integrity of the measuring system through comparative measurement

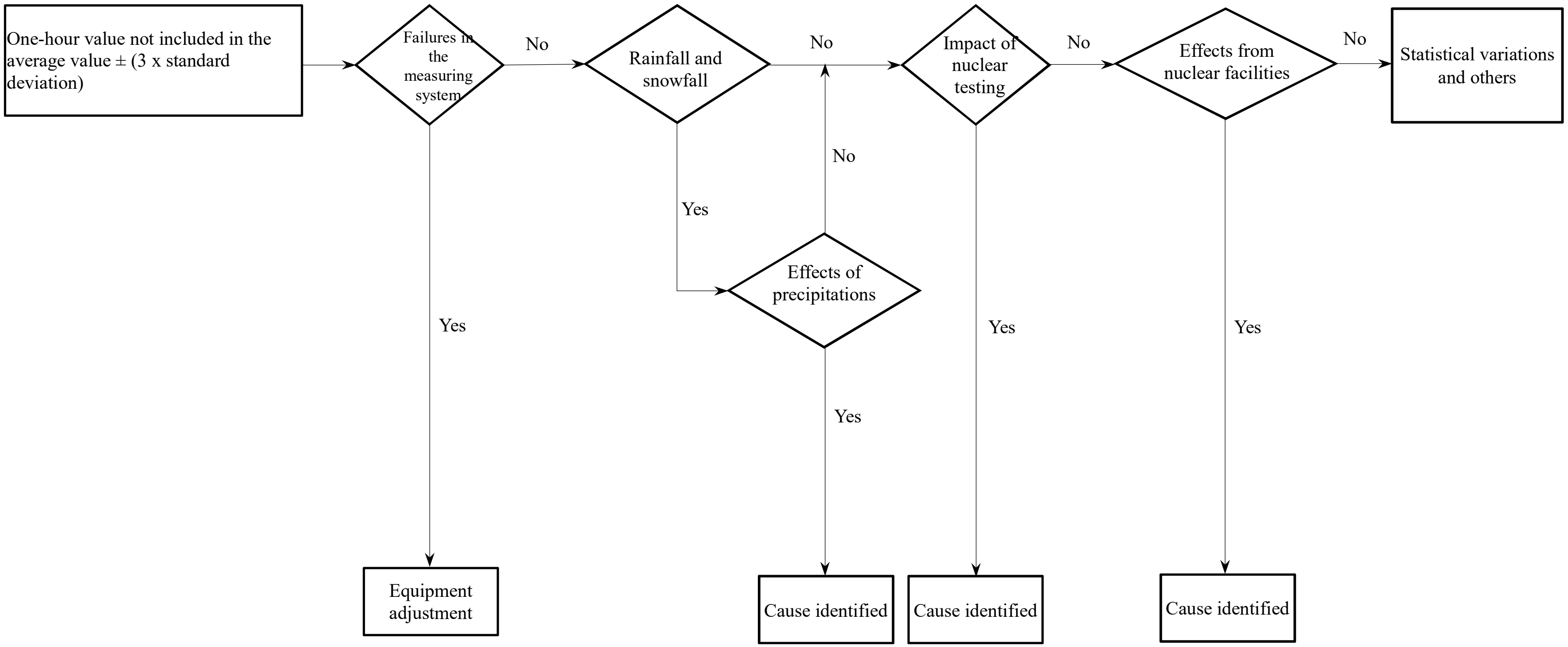


Figure 7.1 Example of a flowchart for outlier analysis

7.4 Dose

There are various definitions for environmental gamma-ray dose rates, including “air absorbed dose (Gy)” and “ambient dose equivalent (Sv).”

Doses should be selected according to the purpose of monitoring, and the measurer is requested to clearly state the dose together with the observation results when reporting.

Table 7.1 Definitions relating to doses

Dose	Indication	Description
Irradiation dose	Coulomb/kilogram (C/kg)	Dose that is the basis of the electric charge dose generated by radiation at 1 kg of air Former unit is Roentgen (R)
Absorbed dose	Gray (Gy)	Indicates the energy of radiation absorbed by a substance. 1 Gy = 1 J/kg In environmental radiation monitoring, air absorbed dose is measured.
Kerma	Gray (Gy)	Defined as the sum of the initial kinetic energy of charged particles generated by radiation per unit mass of a substance. If the acting substance is air, it is called the “air kerma (Gy)” and the air kerma and air absorbed dose may be considered equal.
Equivalent dose	Sievert (Sv)	In assessing the effects on the human body, the type and energy of radiation are considered. It is calculated for each tissue or organ by multiplying the absorbed dose of the relevant tissue or organ by a radiation weighting factor.
Effective dose	Sievert (Sv)	In assessing the effects on the human body, the sites exposed to radiation are considered. It is calculated by multiplying the equivalent dose of the tissue or organ by the tissue weighting factor and summing the total dose for the whole body. During normal operation, environmental radiation monitoring is calculated as: 1 Gy (air absorbed dose) × 0.8 = 0.8 Sv in case of an emergency, 1 Gy = 1 Sv Defined in the Environmental Radiation Monitoring Guideline as the protection amount for which an exposure dose evaluation is supposed to be performed.
Ambient dose equivalent (1 cm dose equivalent)	Sievert (Sv)	This is the dose equivalent at a depth of 1 cm from the surface of a 30-cm sphere (the ICRU sphere) with density and elemental composition simulating the human body.

7.5 Analysis and evaluation of each measuring system

Matters common to all measuring systems are described in and before Section 7.4, and this section describes the matters that differ in the method and the concept of analysis and evaluation for each system.

7.5.1 Installed continuous monitor (an ionization chamber monitor, a NaI monitor, and a simplified electronic dosimeter)

(1) Ground height for measurement

As described in Chapter 5, an installed continuous monitor is measured at different heights for various purposes, and the measurement height used should be clearly stated together with the observation results, when reporting.

(2) Evaluation of results according to the detector

As described in Explanation I.2, when a radioactive plume passes through, the contribution ratio of low-energy (81 keV) photons due to ^{133}Xe is large, which is assumed to be the main factor resulting in a high dose rate. In this situation, it is necessary to select an appropriate detector, such as an ionization chamber monitor without the effect of counting loss.

However, after radioactive materials have been deposited on the ground due to fallout, etc., measured values can be accurately evaluated by selecting a measuring instrument, considering the measurable dose rate range.

Refer to Explanation B for the characteristics of detectors required to select a measuring instrument depending on the situation.

A simplified electronic dosimeter has directional characteristics of high sensitivity to the front of the detector. In other words, because the sensitivity to the rear of the detector is lower than that to the front, when the coverage for measurement is set, the direction of installing the detector must be considered.

7.5.2 Transportable continuous monitor (a transportable monitoring post)

(1) Extraction of outliers

The location of a transportable monitoring post is described as “sites where an installed observation station is not operating owing to the effects of natural disasters or where an installed observation station is not installed” in the “Emergency Monitoring (Supplementary reference materials for the Nuclear Emergency Preparedness Guideline)” (Reference 8). The statement of “the sites where environmental radiation dose rates need to be additionally identified, depending on the source conditions” is also included.

If a planned installation site can be selected in advance during normal operation, the reference data for acquiring the background level must be accumulated during normal operation.

Note that, if the equipment is installed at “a site with no installed observation station installed,” outliers shall be able to be extracted based on the same concept as that of the installed station.

(2) Understanding the effect of temperature characteristics

As discussed in Chapter 5, a transportable monitoring post is equipped with a temperature compensation circuit but not a temperature control system. In addition, a relatively open situation in the upward direction is desirable as the installation condition, and solar power generation is often adopted as a power supply system. In such a case, the detector itself may be exposed to direct sunlight, causing a response change that cannot be compensated by the temperature compensation circuit. Therefore, it is important to understand the change in response due to temperature, when measurements are taken at the same point for a certain period of time.

7.5.3 Mobile survey system

(1) Calculation of a correction factor considering vehicle shielding, etc.

When an installed type is installed outside the vehicle, the primary radiation from radioactive materials around the vehicle can be directly measured. However, the contribution to the measured values is reduced within a few meters from the immediate vicinity of the vehicle, owing to the shielding effect by the vehicle body. Therefore, in the case of the exterior type in which the detector is installed on top of the vehicle, the measurement coverage is extended from a position slightly away from the vehicle to an outward range. Vehicles equipped with a detector normally installed to the roof (thrusting over the vehicle) are suitable for the measurement when a radioactive plume is passing, and they have been calibrated. When this is used to measure the dose rate from radioactive materials deposited on the ground, the evaluation method should be considered, taking into account the geometry such as the shielding and height of the vehicle.

If a measuring instrument is installed in a vehicle, even though it is an installed type or a portable type, measurements should be evaluated by obtaining the correction coefficients of both the inside and outside of the vehicle in advance. The calculation procedures of the correction coefficients are shown below:

[1] Park the vehicle in a flat open area and at a point where the dose rate is uniform around the vehicle.
[2] Perform the measurement using a portable measuring instrument installed at a predetermined position in the vehicle.
[3] Move the vehicle, and perform measurement with the detector at a height of 1 m above ground, at the point used in [2] where the vehicle is no longer positioned.
Perform these measurements at several points with different dose rates.
[4] Calculate the ratio of the results obtained in [2] and [3] above, and use it as the correction coefficient for the inside and outside of the vehicle.

The correction coefficient calculated by this method includes a correction term for converting the shielding effect of the height and the vehicle body.

If possible, it is desirable to obtain the measurement for calculating the correction coefficient at multiple points with different dose rates and to use it as a regression equation from the dose rates of both the inside and outside of the vehicle.

In addition, the above concept is based on the assumption that the measured values taken outside the vehicle, which are supposed to be the basis, are correct. Consequently, national standards and traceability must be secured for the measuring instrument to be used.

By applying the concept of the correction coefficient for both the inside and outside the vehicle to the measured values obtained by an installed-type mobile survey system in which the detector is installed outside, the evaluation considering the geometry such as the shielding and height of the vehicle can also be solved.

When employing the correction coefficients for the inside and outside of the vehicle described above, a validity check must be conducted by calculating the correction coefficients for inside and outside the vehicle at the same point in advance, using the measurement point that was previously used for calculating the correction coefficients as the reference point.

The concept of the correction coefficients for inside and outside the vehicle explained in this section is that the correction coefficients for inside and outside the vehicle must be obtained for the situations of normal operation and in the case of a nuclear disaster separately, because the energy distribution and the incident angle of incident photons are different between normal operation and in the case of a nuclear disaster. If it is difficult to calculate the correction coefficient for inside and outside the vehicle in the case of a sudden nuclear disaster, the correction coefficient for inside and outside the vehicle of “1.3,” which was tentatively adopted after the Fukushima Dai-ichi NPS accident, should be used, and a correction coefficient for inside and outside the vehicle should be calculated later.

(2) Extraction of outliers

To extract outliers in a mobile survey system, the dose rate in the background on a regular route should be measured during normal operation, in order to acquire the level. Numerical values for the natural environmental gamma-ray dose rate of (the background) vary greatly depending on the surrounding environment. Therefore, it is important to grasp the background level from the normal time, in order to help to extract outliers. Explanation G.3 shows the variation in environmental gamma-ray dose rates owing to the surroundings (nature).

Further, when measuring dose rates using several different vehicles or methods, it is important for the evaluation of the dose rate distribution to conduct comparative measurements at the same points in advance, and to understand the relationship between different measurement conditions.

In case of an emergency, monitoring cannot always be conducted on regular routes where investigations are carried out during normal operation. In such a case, absolute values at each point should be evaluated at the initial stage of the accident. In addition, when the dose rate has decreased owing to physical attenuation and human activities after several years have elapsed since the accident, evaluations should be performed by discriminating the contribution of natural radionuclides from the contribution of artificial radionuclides deposited owing to the impact of the accident.

Chapter 8 Measurement of environmental gamma-ray spectrum by a NaI monitor

8.1 Outline

By analyzing the gamma-ray spectrum obtained using a NaI monitor, the contribution dose rate, from uranium series nuclides, thorium series nuclides, and ^{40}K in the case of natural radionuclides, can be evaluated. For artificial radionuclides, the contribution dose rate due to specific radionuclides can also be calculated by the resolution of a NaI (Tl) scintillation detector, if the energy pulse-height distribution is not so complicated. The calculation of the contribution dose rate is performed roughly as follows:

(1) Environmental gamma-ray spectra are measured with an adjusted NaI monitor.
(2) Energy calibration of the pulse-height distribution obtained is performed.
(3) Unfolding is performed by the stripping method or the response matrix method using the response function prepared in advance, and dose analysis is performed after conversion into an incident gamma-ray spectrum is carried out.

After the environmental gamma-ray spectrum acquired using a NaI monitor is transmitted to the telemeter system, spectral analysis is carried out at any time by the software program incorporated in the system, and the effects of artificial radionuclides, if any, can be confirmed by evaluating the contribution dose rate for each nuclide.
The following sections describe the basic principles and methods of environmental gamma-ray spectrum analysis.

8.2 System configuration required for environmental gamma-ray spectrum measurement

Many NaI monitors currently in place are used to measure gamma-ray spectra.
The NaI monitor developed as a continuous monitor consists of a NaI (Tl) scintillation detector (a scintillator, photomultiplier tube, and preamplifier), main amplifier, ADC circuit, dose rate arithmetic circuit, data display unit, data processing unit, recorder, and MCA. A typical system diagram of the configuration of a NaI monitor is shown in Figure 8.1.
In conducting a environmental gamma-ray spectrum measurement, analyses are performed on the basis of the pulse-height distribution accumulated by the MCA. After being collected by the data processing unit or the telemeter system, the pulse-height distribution accumulated by the MCA is converted into an energy spectrum and analyzed for the dose calculation and so on.

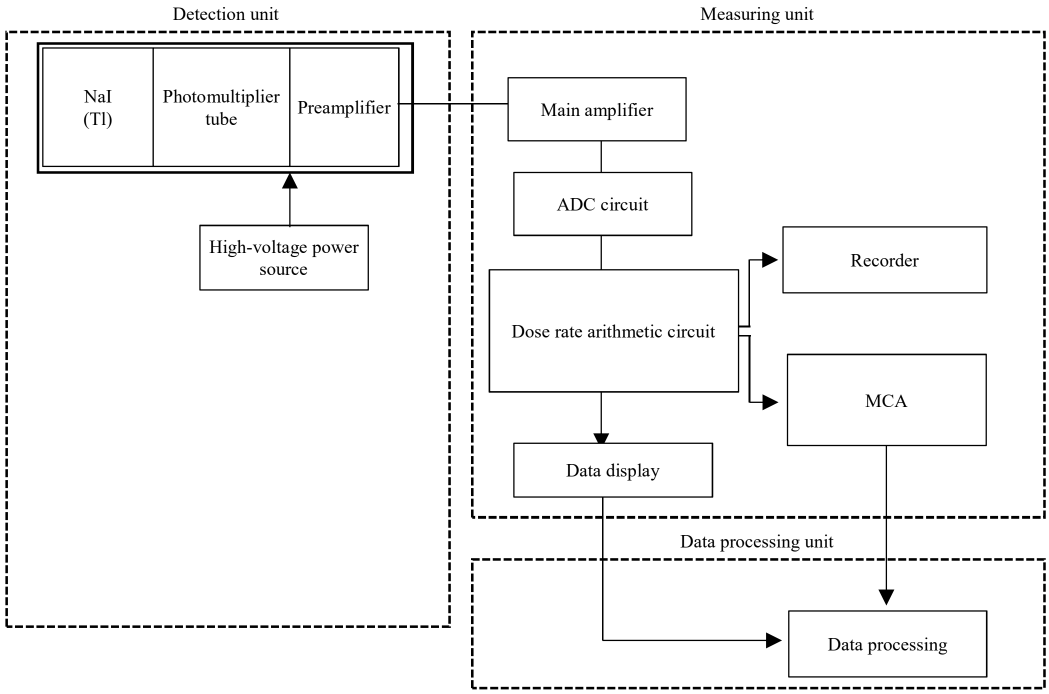


Figure 8.1 Example of the configuration of a NaI monitor

8.3 Preparation for measurement

To carry out environmental gamma-ray spectrum measurements as part of the measurement by a continuous monitor, data are continuously and automatically collected, usually without taking special measurements separately.

Here, as a basic concept for environmental gamma-ray spectrum measurement, the basic procedures for separately measuring with a portable NaI spectrometer are shown.

The measuring system consists of: (1) a radiation detector, (2) preamplifier and main amplifier for amplifying and shaping radiation signal pulses, (3) pulse-height analyzer, and (4) recorder for the measurement results. To obtain reliable measurement results, it is necessary to fully understand the function and characteristics of each component of the system and combine the components for use such that they can operate well. The key parts of adjustment and selection for determining the system requirements are:

- [1] Applied voltage of the detector,
- [2] gain of the main amplifier, waveform shaping conditions, and DC level of the DC reflex circuit,
- [3] and the conversion gain of the ADC for the pulse-height analyzer and the number of channels used.

The waveform shaping conditions are not only related to the pulse resolution but also change the temperature dependence of the pulse-height values in relation to the decay time of the scintillation pulse, which must be considered when a condition is selected. The waveform shaping time constant is usually chosen in the range of 0.5–2 μs.

The number of channels used for the MCA shall be 500–1000 channels, considering that the energy width per channel of the spectrum obtained by the NaI (Tl) scintillation detector is several to 20 keV, and usually 5–10 keV/ch is appropriate; further, this number can be set such that cosmic ray information can be obtained at 3 MeV or more.

Once a policy has been established for each of the above points, check and adjust the following items while confirming that each unit is operating satisfactorily.

[1] DC level adjustment of the DC reflex circuit
[2] Noise level testing
[3] Testing and confirmation of the measured energy range
[4] Check for any malfunctions in MCA
[5] Testing of the detector energy resolution (depending on the half-value width)
[6] Linearity testing of pulse-height versus a channel number

For the adjustment in [1] (not available in some devices), a reference source (e.g., 662-keV gamma rays of ^{137}Cs) shall be used by observing the pulse waveform with an oscilloscope.

For [2], this test shall be conducted in parallel with the test in [3]. The measured energy of gamma rays ranges from 40 keV at minimum to 3 MeV or more, and the levels of a lower-level discriminator (LLD) and an upper-level discriminator (ULD) of a pulse-height analyzer shall be set within this range. If a strong noise were produced in the range of close to 30–40 keV, not only would the measurement accuracy of low-energy gamma rays be decreased, but the resolution of the entire pulse-height would also be impaired; thus, measures of eliminating possible causes must be considered. The LLD shall be set as low as possible without noise effects, and the ULD shall be set as high as possible so as to obtain more information about cosmic ray components.

In [4], a monoenergetic gamma ray shall be used to check the shape of the entire pulse-height distribution, the symmetry of the total absorption peak[*1], etc., for confirming abnormalities, if any. In addition, when an excessive signal input such as a pulse caused by cosmic rays is identified, an abnormal signal may be generated in the gamma-ray region owing to the amplification circuit and other transient characteristics; thus, an adequate inspection is required.

For [5], 662-keV gamma rays of ^{137}Cs are used to determine the quality with the resolution of the total absorption peak. Normally, approximately 7%–9% is the range of normal values, and any abnormalities should be checked against the manufacturer's performance report, etc.

For [6], testing shall be conducted by distinguishing the characteristics of the electrical circuit only from those of the NaI (Tl) crystal, including the light emission characteristics. The characteristics of the electrical circuit section shall be examined by using a pulsar[*2] or similar, based on the relationship between various outputs through an attenuator[*3] and the channel numbers of the MCA corresponding to them. Because the channel number corresponding to the pulse-height value of zero is different from that obtained with reference to the gamma ray energies, these relationships shall be accurately determined in relation to the energy calibration of the pulse-height distribution (energy calibration will be discussed later).

8.4 Primary processing of measured data

In preparation for energy calibration, the data of photo-peak channels corresponding to gamma-ray energies are arranged, based on the pulse-height distribution data obtained by a reference source and that obtained by this measurement. Because the relationship between the energy and the channel number is almost linear, it can be determined that a normal measurement has been made if there is no large difference observed in the slope for each measurement.

[*1] A peak that is produced in the gamma-ray spectrum where all the energy of the incident gamma rays is absorbed by the detector
[*2] A device that electrically generates pulses at arbitrary energies (channels)
[*3] An attenuator is an electronic device that attenuates pulses to an appropriate level

8.5 Data acquisition for energy calibration

Perform energy calibration as necessary, in accordance with the energy calibration method that is established in advance. The energies of gamma rays used for calibration should be obtained in detail, over the entire energy range of the gamma rays to be measured; however, for the purpose of operation, calibration is carried out at two points; at 1461 keV of ^{40}K and at the energy level corresponding to the zero points, which are confirmed in advance. See Chapter 10 for details.

If a gamma-ray source can be used without special restrictions, the following methods will be applicable.

Because of the limited number of reference sources available, a method of obtaining necessary information from a few sources should be considered in advance. In terms of the calibration points, at least one point should be selected in the range of 50–200 keV, and one point in the range of 1–2 MeV. For this purpose, 0.122 MeV of ^{57}Co is suitable as a calibration point on the low-energy side, and 1.46 MeV of ^{40}K as a natural radionuclide or 1.33 MeV of a ^{60}Co source is suitable as a calibration point on the high-energy side. If an energy other than these is used to perform a calibration, a high-precision energy calibration shall be performed, with reference to Chapter 10.

Chapter 9 Response function used by NaI monitor for measured data

9.1 Description of a response function

To obtain the incident gamma-ray spectrum on the detector from a pulse-height distribution that is output from the detector, knowledge of the response indicated by the detector when a monoenergetic photon is incident on the detector is required. If a pulse-height distribution P(h) is obtained while observing a photon with an energy spectrum N(E) (E is energy), this can be generally expressed in the integral equation shown below. In this equation, ε(E) refers to the efficiency of the detector. K (E,h) or ε(E)K(E,h) in this equation is called a response function. The former is obtained by normalizing the area of a pulse-height distribution group obtained at the time of a monoenergetic photon incidence to 1, and the latter means the pulse-height distribution group obtained at the time of a monoenergetic photon incidence.

$$\mathrm{P(h)} = \int_{0}^{E} \varepsilon\,(\mathrm{E})\mathrm{K}(\mathrm{E},\mathrm{h})\mathrm{N}(\mathrm{E})\mathrm{dE} \qquad (9.1)$$

A response function is determined by an experimental method or a Monte Carlo simulation. A response function depends on the target radiation source and the arrangement of the detector used. Because radiation in the environment is generally not isotropic, special consideration should be given to directional dependence, when a small direction-dependent detector (spherical) or a cylindrical detector is used.

9.2 Creation of a response function

A response function is obtained by measuring pulse-height distribution groups of the detector for monoenergetic photons with several different energies and dividing them by their corresponding incident photon fluences. The irradiation conditions are that the source shall be placed in a direction perpendicular to the axis of the cylindrical detector, with less contribution of scattered rays, and also close to the parallel beam. The pulse-height distribution of arbitrary energies shall be obtained by interpolation. A Monte Carlo simulation method is also available (see Explanation J.2).

The segmentation width (or channel) of a response function should be selected as appropriate according to the purpose of the analysis, by setting it to 10, 20, 50, 100 keV, and so on. A response function may also be used, as represented in a matrix of 10 rows and 10 columns, or 50 rows and 50 columns.

When a response function is represented as a matrix, a value obtained by adding each element of the column (or the row) provides the energy efficiency corresponding to the column (or the row) when a single photon is incident. The validity of the response function (the matrix) created can be checked by comparing this value with the efficiency η_t, which is theoretically calculated using equation (9.2) for the detector. It is desirable that they be nearly identical or the difference between them be as small as possible.

$$\eta_t = 1 - \frac{2}{(\mu d)^2} + 2e^{-\mu d}\left\{\frac{1}{(\mu d)^2} + \frac{1}{\mu d}\right\} \qquad (9.2)$$

where μ is the linear attenuation coefficient of the scintillator, and d is the diameter of the detector.

In addition, a P/T ratio can be calculated using a response function, so as to compare it with an experimental value. In this case, an experimentally determined P/T ratio is reduced by approximately 5%–10% (see Appendix 1 “Calculation examples of a response function”), because an experimentally obtained pulse-height distribution contains the contribution of scattered rays.

9.3 Application of a response function

To analyze a pulse-height distribution in which monoenergetic photons with different energies are observed mixed at various intensities, and to determine the photon energy and intensity individually, the stripping method (the peel-off method) is applied. This entails focusing on the total (energy) absorption peak identified at the highest-energy position, applying the response function sequentially, and then stripping from the highest-energy side. In doing so, the following methods can be used: a method to subtract one after another in accordance with the response function and the peak on the highest-energy side, a method to apply the least-squares fitting to minimize the square of the subtracted residual, and a method to subtract from the highest-energy side on the basis of the total energy absorption peak and the P/T ratio.

In the case of environmental radiation observed when several types of monoenergetic photons and scattered rays are overlapped, the incident gamma-ray spectrum can be obtained by analytical methods such as the inverse matrix method or the successive approximation method, in addition to the stripping method. Among the abovementioned methods, the response matrix, in which a response function is represented as a matrix, is used.

The accuracy of the incident gamma-ray spectrum as an analysis result is determined not only by statistical errors in a pulse-height distribution and the analysis method, but also by the compatibility of the response function (or the response matrix) and the response generated by the measuring system in practice.

Chapter 10 Processing of data on environmental gamma-ray spectrum measured by NaI monitor

The processing of a pulse-height distribution obtained by a NaI monitor is, in many cases, performed by pulling the entire pulse-height distribution back to the incident spectrum and treating several channels as a monoenergetic block, rather than focusing on a particular peak.
The details of the processing include the energy calibration of a measured pulse-height distribution, conversion to an incident spectrum, and several analyses performed to use the results for specific applications.

10.1 Energy calibration of the measured pulse-height distribution

In a measured pulse-height distribution, the relationship between energy and the pulse-height or the number of channels is not generally linear. If a linear relationship has been established, you may skip this section. If this relationship is not established, the analysis result would be biased owing to the low resolution of the detecting system. Therefore, this must be calibrated. To do this, the relationship between the channel and the energy shall be clarified, and the pulse-height distribution shall be converted into one usually having a constant energy width (channel width) per channel. To perform this calibration process, the relationship between the channel and the energy of a pulse-height distribution is converted using a polynomial that produces a best approximate point among several sets of points of the center channel of the peak in the spectrum and the energy corresponding to it. As a polynomial, a linear (straight line) type is usually used, which approximates three points of the energy around a zero point (for example, 60 Kev of ^{241}Am, 122 keV of ^{57}Co, or the specific energy of an input signal zero (for example, −17 keV)), 1461 keV of ^{40}K, and 2614 keV of ^{208}Tl of the thorium series (Figure 10.1). To further refine the calibration, a higher-order equation is used for the transformation, which requires selecting a corresponding peak energy in the required number (if the n^{th} expression is used, the required number is $(n + 1)$ or more). Once the energy/channel relationship is obtained, each channel can be easily converted for the energy calibration. The converted spectrum is called the corrected pulse-height distribution. In the case of a linear approximation, the channel width for a pulse-height distribution is set to 10, 20, 50 keV, and so on, per channel, and a value of 100 or 200 keV may be selected, though it is not usual. Inaccuracy in this correction, if any, would directly produce uncertainties in the final results of analyses; thus, precautions must be taken.

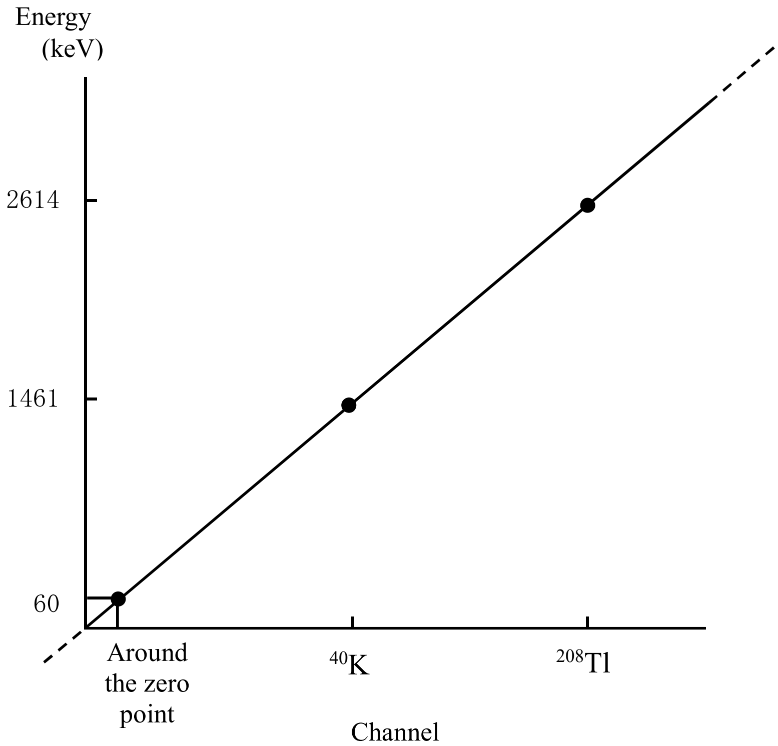


Figure 10.1 Application example of linear approximation of three points

10.2 Conversion to an incident spectrum

Conversion to an incident spectrum is a process of data processing for obtaining an incident gamma-ray energy spectrum from the corrected pulse-height distribution, and is performed by applying a response function as described in Chapter 9. In this case, that which is contained in a response function must be clarified in advance. In other words, attention should be paid to the conditions under which the response function was created, the difference from the conditions under which the response function was measured in the field, and especially, the incident direction of radiation. The conversion methods are roughly divided into the following two: the stripping method (the peel-off method) and the response matrix method.

(1) Stripping method

In the stripping method, a response function for a single peak corresponding to the high-energy channel of the corrected pulse-height distribution is subtracted from the corrected pulse-height distribution, and the value in the high-energy channel is reduced to almost zero; then, this operation is sequentially applied to the low-energy side. As a result, the response function for obtaining a distribution in the low-energy region and the method used to introduce the constant at that time affects the accuracy of the incident gamma-ray spectrum that is obtained after the processing is completed. Therefore, the shape and numerical values of the response function to be stripped must be clarified. It is also necessary to provide the conditions used for creating the response function.

An example of the stripping method is given in Appendix 2.

(2) Response matrix method

The response matrix method is a solving method in which a specific photon energy is determined in advance, a response function matching the corrected pulse-height distribution (a numerical matrix for each energy region) is created, and a matrix of each energy is solved by a simultaneous equation for the actually measured corrected pulse-height distribution; two methods are used: one is to solve by creating a conjugate matrix in advance, and the other is to solve by the successive approximation method. In this method, as with the stripping method, it is desired that the measurement conditions used when the response matrix is created be the same as those employed in the field. Differences between the conditions, if any, would contribute to uncertainties in the results.

An example of unfolding by the successive approximation is presented in Appendix 3.

In applying both methods, the accuracy of the response function used is essential. When the detected body for which a response function is determined and the detected body actually used for environmental measurement are different, a difference may be produced in the result due to various differences in the detected bodies, in particular the differences in the structure of a scintillator such as the size, shape, weight, and casing, and differences in resolution. The difference in weight can be corrected as the difference in cross-sections. In converting the energy spectrum of incident gamma rays, it may be necessary to further consider the contribution of cosmic rays from the higher-energy region and the background derived from the detected body itself.

At present, response functions that can be applied to scintillation spectrometers with different resolutions for each shape and size of the detector have been published, and they are incorporated into a commercially available analysis software.

10.3 Representation of the results

There are many expressions for the analyzed measurement results. These include: (1) the total counting rate value, (2) total count rate value for a corrected pulse-height distribution (3) gamma-ray flux density (within the total energy range and specific regions), (4) irradiation dose rate (within the total energy range and specific regions), (5) absorbed dose rate of NaI (Tl), and a graphical display specified for those above. The statistical errors of the count values, precision, details associated with channel calibration, functions used and their precision, dimensional structure of the detected body, constant used, basic constants, data processing technique, processing program, and handling of the background are also included. Of these, (1), (2), (3), and (4) mentioned above, and the ratios of (2) and (3) to (4), are often adopted as typical expressions of the results.

Note that the flux density of cosmic rays obtained from a high-energy region and its estimated dose are helpful for understanding the aspects of radiation in the environment. In doing so, if the manner in which components other than gamma rays in the spectrum are treated were not specified, the results would not be interpreted clearly. If the relationship between the irradiation dose rate across the entire energy region and the photon flux density or the counting rate is constant, the irradiation dose rate can be immediately obtained from the counting rate by the photon flux density. In this case as well, because the incident gamma-ray spectrum is already obtained, the irradiation dose rate can be obtained for each energy region.

10.4 Calculation of dose contribution by radionuclide

In measuring a environmental gamma-ray spectrum method using a spectrometer, the dose contribution to the environment for each radionuclide component is determined. A NaI (Tl) scintillation spectrometer is used only when limited types of artificial radionuclides (e.g., ^{131}I, ^{137}Cs, etc.) are present, because high resolution is not expected, unlike a germanium semiconductor detector. On the contrary, in ordinary environmental radiation fields, the dose contributions by natural radionuclides such as potassium, uranium, and thorium series are typically separated.

10.4.1 Use of count values in the peak area

This data analysis is performed by applying the peak area present in the spectrum, the energy and emission rate of the corresponding nuclides, and the detection efficiency of the detector.

Because it is generally not easy to select the peaks corresponding to the respective energies of the emitted gamma rays in a NaI (Tl) scintillation spectrometer, this is applied only to a limited number of nuclides that emit relatively simple gamma rays.

10.4.2 Determination of dose contribution by nuclide by the matrix method

In a scintillation spectrometer, for the reasons already mentioned, it is not easy to separate the dose contributions of each nuclide that emits gamma rays of various energies. Therefore, it is common that the energy regions covering a wide range are determined in advance, and a matrix analysis is conducted. This method is used for measuring normal environmental radiation, mainly composed of natural radionuclides; the method is explained below:

The dose contribution of natural radionuclides in the environment is usually generated by potassium, uranium, and thorium series, and they are typically separated; however, in some cases this is followed by the contribution of artificial radionuclides.

To separate the dose contributions for these three main natural components, the energy regions corresponding to each component are determined in advance, the contributions contained in each region per unit dose rate (dose rate, counting rate, flux density, etc.) are prepared as basic data for each component, and then the matrix is solved. For the energy regions employed in this method, the following range is defined as a standard. An example of the matrix constant representing the dose rate contribution between regions is illustrated in Table 10.1.

Potassium region (1.34–1.60 MeV)
Uranium series region (1.61–2.30 MeV)
Thorium series region (2.31–3.00 MeV)

Table 10.1 Contribution coefficients between regions for solving a matrix (the irradiation dose rate in each region per μR/h; the numbers in parentheses represent the value in "cps" for a 3-in spherical scintillator)

	K-40 region	U series region	Th series region
K-40	0.5 (1.5)	0 (0.0)	0 (0.0)
U series	0.106 (0.508)	0.255 (0.565)	0.024 (0.0236)
Th series	0.043 (0.689)	0.124 (0.674)	0.275 (0.343)

(Note) These coefficients are based on the following measured data and are expressed in R, which is the measurement unit used.

K-40 : Standard source (KCl)

U series : Cave filled with radon in Houston

Th series : Kerala District in India

The sum of the respective values obtained using these coefficients must correspond to the measured total dose rate value. However, differences in the matrix coefficients, counting errors, and conditions used in the measurement environment may produce a difference of approximately 10%, and if a difference exceeding this value is produced, it must be examined in detail. It is desirable to refrain from adopting the results when the difference between these items exceeds 10%.

Explanation

Explanation A General matters on environmental gamma rays measured by a continuous monitor

The monitor focused on in this manual is called “continuous monitor,” which can measure gamma rays in the energy range of tens of keV to several MeV among different types of ionizing radiation in the general environment, continuously obtain count values with short time intervals, and output the measured values in real time (to an analog or digital display). Many types of continuous monitors for environmental gamma rays are currently used, reflecting the historical background and owing to technological advances; their characteristics and performance are quite different. However, regardless of their type, the measurements obtained from these monitors are mainly the air absorbed dose rate (μGy/h, etc.), ambient dose equivalent rate (μSv/h, etc.), or detector counting rate (cps, cpm, etc.) acquired at the location of the detector.

Because continuous real-time data can be obtained using a continuous monitor, changes in dose rates can be tracked with time, and along with meteorological observation data, the cause of these changes can be clarified more easily. This an important characteristic of continuous monitors, when compared with a thermoluminescence dosimeter (TLD) or fluoroglass dosimeter (RPLD), which provide integrated doses over a long time. In some cases, it is also possible to discriminate natural radiation from artificial radiation by analyzing continuous recording pattern, and in such cases, the contribution from nuclear facilities can be estimated. In addition, many of the systems currently in use can acquire energy information, which can be used for a more accurate discrimination.

However, it is difficult to install many continuous monitors because of the complicated mechanism, considerable labor required for maintenance and management, high cost, high power consumption, etc. Therefore, detailed dosimetry of the surrounding areas has generally been left to TLD and RPLD; however, transportable monitoring posts and simplified electronic dosimeters may currently play this role.

Owing to the differences in the type of detector, structure around the detector, characteristics of the device, signal processing circuitry, and so on, differences may be produced between the measurements taken by various types of monitors, even those taken at the same points concurrently. Many problematic matters are identified in the vicinity of the detector, such as the contribution of cosmic rays, shielding effects, directional characteristics, and differences in the energy characteristics. As a matter of special attention required for monitoring during normal periods, regarding the contribution of cosmic rays, for example, in the G(E) function method using a scintillator as a detector, high-energy components including cosmic rays of 3 MeV or more are involved in the general use. However, those of 3 MeV or more are separated in the currently prevailing usage,; thus, a full understanding of the specification details is important. The dose rate indicated by a monitor using a scintillator as a detector may be considered to be lower than that indicated by one using an ionization chamber as a detector, by a value corresponding to the normal contribution of cosmic rays. In addition, even if a detector of the same model is used, depending on the differences in the production lots, the difference in the contribution dose rate due to self-contamination makes it difficult to compare the measured values with each other. With regard to the contribution dose rate due to self-contamination, one available measuring method is to conduct in a basement or similar environment, where the contribution of the environmental gamma-ray dose rate and cosmic ray dose rate is reduced, and to evaluate from the peak counting rate of ^{40}K; another method is to use a representative value for each of the constituent materials of the detector.

If the measured values are corrected considering these factors, the measured values of different detector models can be compared; however, in the current situation, such corrections are not easy. Therefore, when different detectors and different measuring devices are installed at the same point, attention should be paid when comparing the measured values taken at the background level with each other.

Explanation B Principle, configuration, and characteristics of each monitor

Explanation B.1 Ionization chamber monitor

The ionization chamber detectors for continuous ionization chamber monitors currently in practical use include a metal spherical ionization chamber, in which pure argon gas or nitrogen gas is injected with high pressure. To enhance the sensitivity, the volume is increased, or the injected gas is pressurized. An ionization chamber monitor displays the dose rate as an analog output of a current–voltage conversion by passing an ionization current through a high resistance, or as a digital output of charge by current integration at regular time intervals. An ionization chamber monitor has good energy and directional characteristics, and it can have a wide range for dose rate measurements, depending on the specifications. The energy characteristics and dose rate linearity of various ionization chambers are shown in Figures B.1 and B.2, respectively.

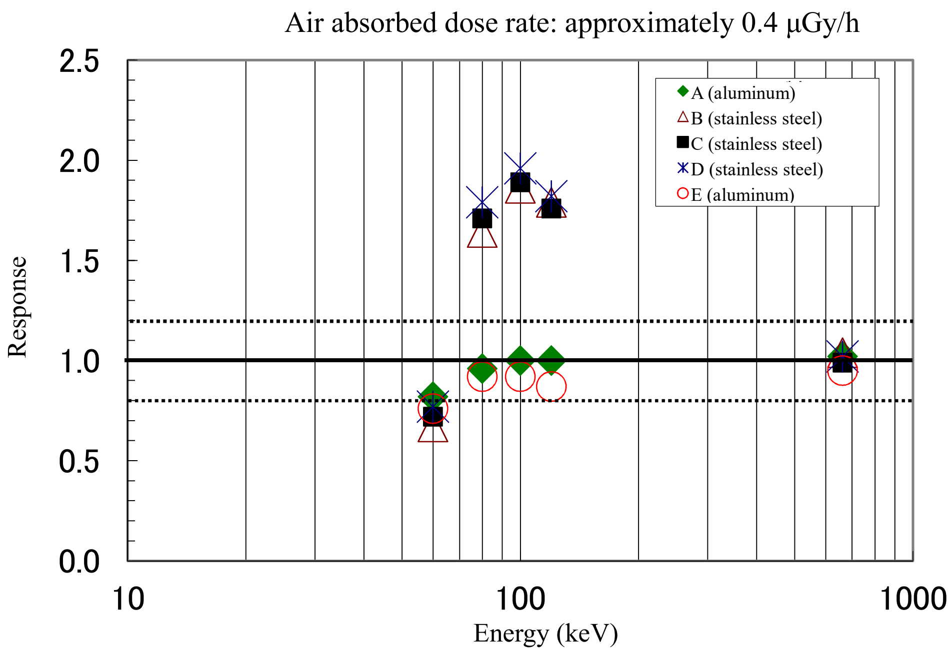
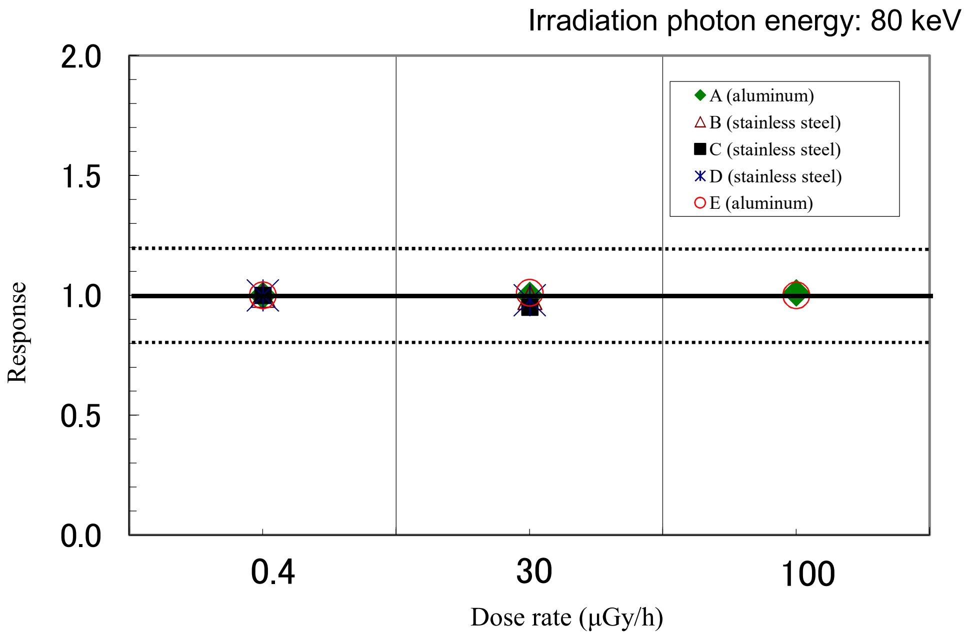


Figure B.1 Example of energy characteristics of various pressurized ionization chambers



(* Standardized on a comparison value of 0.4 μGy/h)

Figure B.2 Example of dose rate linearity of various pressurized ionization chambers

As described in Chapter 4 of the main text, the factors affecting the measured values, such as the energy characteristics of an ionization chamber dosimeter, are mainly the wall material and its thickness, as well as the type of gas contained and its pressure.

For the purpose of dosimetry, it is ideal that the materials of a detector can be selected depending on the dose to be measured. Ideally, a substance equivalent to air should be used to measure the absorbed dose of air, and a substance equivalent to tissues should be used to measure the absorbed dose of tissues. An air-equivalent ionization chamber is most suitable for environmental gamma-ray monitoring because it measures the air absorbed dose, but, in many cases, the sensitivity is insufficient for low-dose-rate measurement. In an ionization chamber used as a continuous monitor, argon gas or similar is pressurized and injected into an aluminum or stainless-steel container to enhance the sensitivity. Because the atomic number and density of stainless steel and argon gas are higher than those of air, the mass–energy absorption coefficient becomes remarkably large in the low-energy region, despite the high sensitivity, and the response increases. Further, in the low-energy region, gamma rays are absorbed by the stainless-steel wall, resulting in a rapid decrease in the response. The energy characteristics of an ionization chamber can be expressed by reflecting these factors combined.

Explanation B.2 G(E) Function method by a NaI (Tl) scintillation monitor

B.2.1 Principles and configuration of the monitor

One of the methods of evaluating the absorbed dose and ambient dose equivalent from a NaI (Tl) scintillation spectrum distribution is to use a spectrum-dose conversion operator for the dose calculation by directly applying a weighting function to the pulse-height spectral distribution of gamma rays. This is the G(E) function method.

The G(E) function method uses a spectrum-dose conversion operator (G(E)) to evaluate dose rates directly from the gamma-ray pulse-height distribution.

Assuming that the pulse-height distribution is P(E) and the in-situ dose rate is D, D is linked as follows, through the spectrum-dose conversion operator G(E). The first line of the equation represents the principle of the G(E) function method, and the second line represents the application of the method used when the dose rate is calculated from the pulse-height distribution measured by a multichannel pulse-height analyzer.

$$\begin{aligned} D &= \int_{E_{min}}^{E_{max}} P(E) \bullet G(E) dE \\ &= \sum_{I=I_{min}}^{I=I_{max}} P(I) \bullet G(I) \qquad \text{(B.1)} \end{aligned}$$

where

D	: Dose rate (nGy/h)
$P(E)$	: Pulse-height distribution (cpm/keV)
$P(I)$	: Pulse-height distribution measured by a multichannel pulse-height analyzer (cpm/channel)
E	: Pulse-height (keV)
I	: Channel number
E_{min}	: Lower limit of measurement of a pulse-height (keV)
E_{max}	: Upper limit of a dose rate evaluation (keV)
I_{min}	: Channel number corresponding to E_{min}
I_{max}	: Channel number corresponding to E_{max}
$G(E)$, $G(I)$	: Spectrum-dose conversion operator (nGy/h/cpm)

A method of performing a dose calculation based on the above equation using an analog signal of pulse-height and an electronic circuit is called the DBM method, and a monitor corresponding to the DBM method is called a DBM monitor. Recent advances in computer technology have made it possible to perform small-scale and high-speed operations; thus, the method of digitizing a pulse-height and converting it into a dose by multiplying directly by the G(E) function is now prevalent.

E_{min} and E_{max} are the lower and upper limits of the energy range for dose rate evaluation, respectively, and when environmental gamma rays are to be measured, the values of E_{min} = 30–50 keV and E_{max} = 3000 keV are usually used. As can be seen from this equation, the G(E) function acts as a type of energy-dependent weighting function. This method has an advantage that the dose rate can be evaluated without converting a pulse-height distribution to an incident gamma-ray energy spectrum. The G(E) function applied to each detector is determined separately, depending on the shape and size

of the detector. An example of the G(E) function is shown in Figure B.3.
In the case of a NaI (Tl) detector, the most important factors that determine the response function and the G(E) function are the shape and size of the NaI (TI) crystal, and the shape of the crystal container.
In the case of NaI (Tl) detectors, if the shape and dimensions are the same, the response characteristics are also the same; therefore, once the G(E) function for one type of NaI (Tl) detector is determined, the G(E) function can be directly applied to other NaI (Tl) detectors of the same type as well.
The G(E) function method was originally developed by considering the measurement of the gamma-ray dose (rate) by a NaI (Tl) detector; however, it is also applicable to a detector that can obtain the energy information (pulse-height distribution) of incident radiation, as with a NaI (Tl) detector (References 9 and 10).

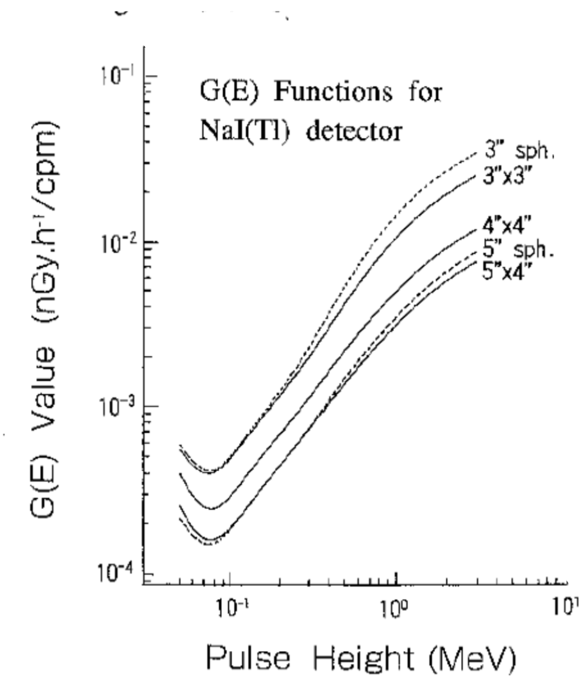


Figure B.3 G(E) function of a NaI (Tl) detector

B.2.2 Characteristics of the monitor

(1) Stability

[1] Detector

The most important aspect related to the changes in response in this type of monitoring is the stability of the detector and circuit with respect to the changes in ambient temperature in the environment. The instability of the detector against temperature changes can be divided into two factors: the temperature dependence of the luminous efficiency of NaI (Tl) scintillation and the decay time of light, and the temperature dependence of photoemission of the photoelectric surface of a photomultiplier tube. The overlapping part of these two changes is usually the temperature characteristic of the detector, and a combination of this with the pulse shaping time constant of the electronic circuit (a digital type is less sensitive to temperature changes than an analog type) causes the temperature dependence of the monitor.

[2] Electronic circuit

The temperature characteristics of a high-voltage power source are also important. In general, the gain change of a photomultiplier tube used in radiation measurement is close to 10 times the rate of variation of the high voltage, and when stability within a range of ±1% of the gain change is required, the stability of the high-voltage power source must be less than ±0.1% (±1 V equivalent or less) of the gain change.

[3] Measures for stabilization

When gamma rays are monitored in an environment with a wide range of temperature fluctuations throughout the year, special consideration should be given to the temperature characteristics; stability should be secured at least within a range of ±(3–5%) throughout the year. Many of the devices currently on the market take various measures for stabilization, such as thermal insulation of probes, isothermal treatment, and temperature compensation by an electronic circuit, and adequate stability is ensured throughout the year. However, this is the case of a fixed station or the specifications corresponding to it; the fluctuation range becomes large when measures for isothermal treatment cannot be applied owing to the nature of the system for a transportable monitoring post, etc.

(2) Energy characteristics

The energy characteristics of a monitor using the G(E) function method are usually within ±20% in the range of 0.1–3 MeV (Figure B.4), which is sufficient, considering the energy distribution of environmental gamma rays. Note that because these characteristics also vary owing to the absorption of gamma rays by the cover of the probe, an unnecessarily thick cover must not be used. Therefore, the response at 60 keV is low, as shown in Figure B.4.

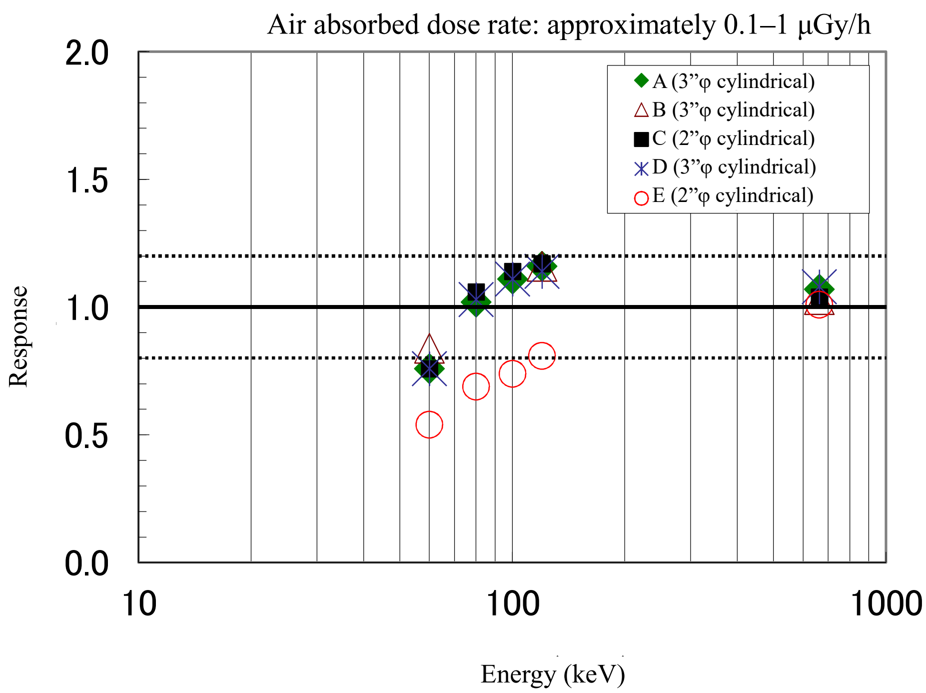


Figure B.4 Energy characteristics of a NaI monitor using the G(E) function method

(3) Response and counting errors

This type of measuring instrument exhibits a close relationship between the energy compensation range and the response; when the compensation range is narrowed, the response (in this case, the counting rate per unit dose rate) increases, while, when the compensation range is widened, the response decreases. Regarding the energy range of environmental gamma rays, the maximum energy range is 2.62 MeV for natural radiation, and no significant radiation of energy exceeding 3 MeV is present, except for gamma rays from ^{16}N, which are attributable to the facility. Therefore, energy compensation up to 3 MeV is sufficient.

The standard deviation of a measured value can be calculated considering the response and measuring time shown above. Figure B.5 shows an example of a calculation of measurement errors in 10-min measurements taken with scintillators of various sizes.

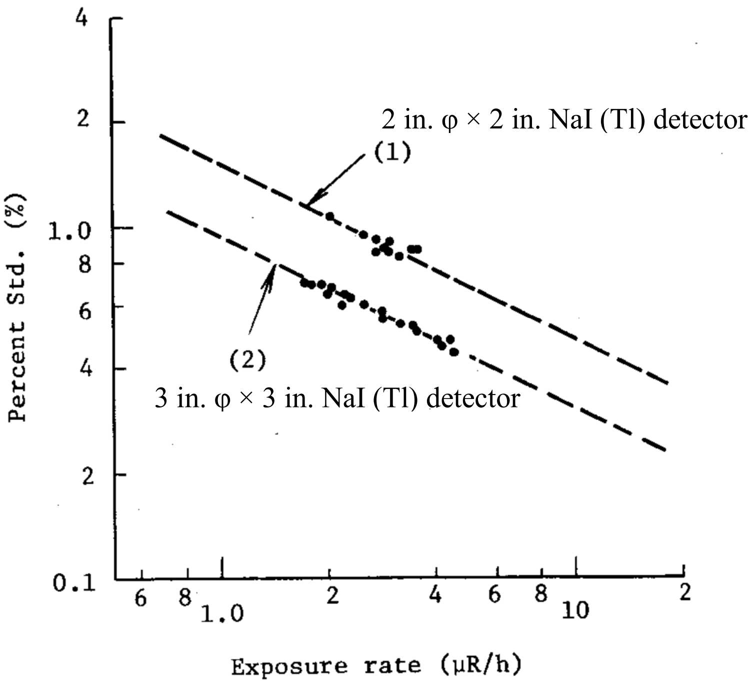


Figure B.5 Statistical errors (1σ) in a 10-min measurement of environmental gamma rays by the G(E) function method (Reference 11)

(Note) Because the figure is cited from Reference 11, the former unit R is used.

(4) Directional characteristics

NaI (Tl) scintillators have different directional characteristics depending on their shape, but the actual characteristics vary further depending on the structure of the probe in use. Spherical scintillators have the most excellent directional characteristics, and cylindrical scintillators develop exhibit their best directional characteristics when the ratio of the diameter to height is 4:3. In the case of a scintillator in which the diameter and height are equal, a difference of approximately 8% is produced in the response between the axial and lateral directions. However, because of differences in the structures of individual probes in use, the precise directional characteristics should be determined by measurements. Figures B.6 and B.7 show examples of the directional characteristics when the axial responses of a 3 in. φ × 3 in. cylindrical and 3 in. φ spherical scintillator are set to 1.0.

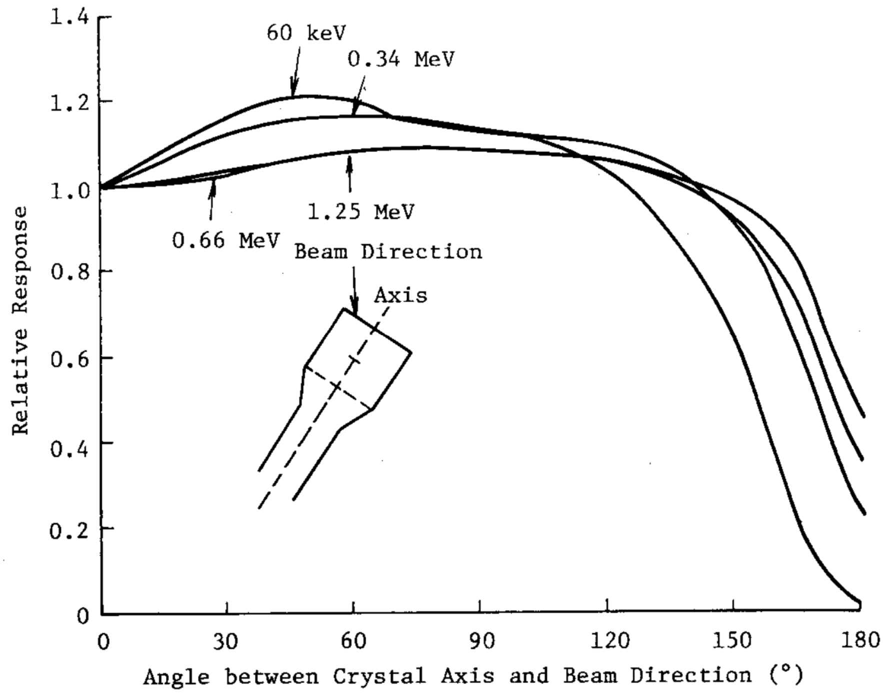


Figure B.6 Example of directional characteristics of NaI (Tl) scintillator (3 in. φ × 3 in. cylindrical) (Reference 11)

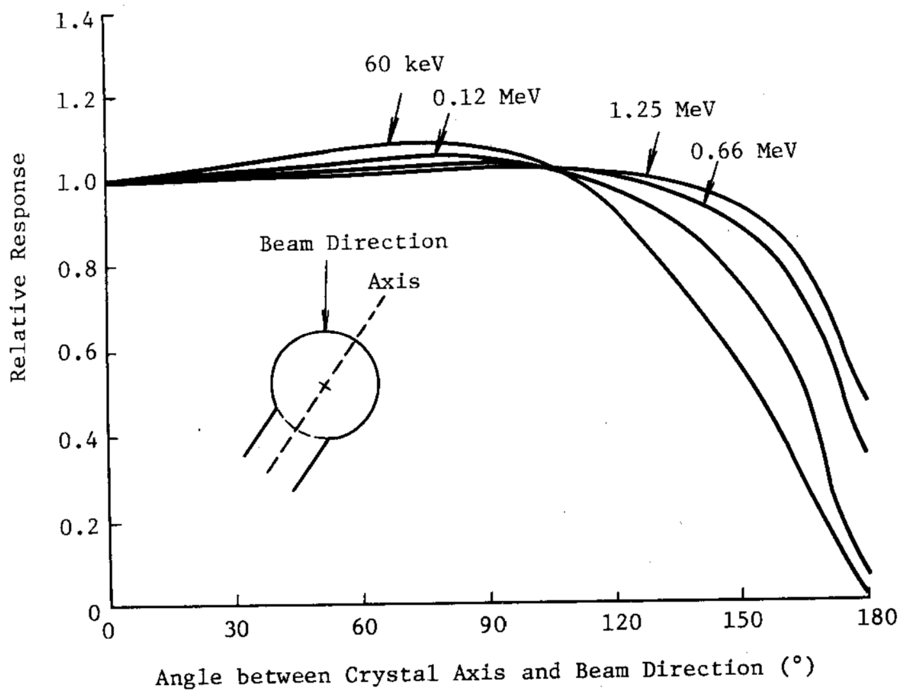


Figure B.7 An example of directional characteristics of NaI (Tl) scintillator (3 in. φ spherical) (Reference 11)

(5) Contribution of cosmic rays and self-contamination

[1] Contribution of cosmic rays

It is reasonable to treat cosmic ray components by dividing them into pulses of 3 MeV or less and those of 3 MeV or more at the pulse-height. It is very difficult to carry out a separation valuation on the ground for the components classified as 3 MeV or less, because they overlap with the environmental gamma-ray spectrum. They are usually evaluated by taking measurements on a sea or lake not affected by gamma rays from the Earth's crust, or by extrapolating high-altitude observation values taken by an airplane to the ground.

The cosmic ray component in the lowland area of Japan based on the data obtained by high-altitude observations is approximately 2 nGy/h equivalent at 3 MeV or less, when a 3 in. φ × 3 in. cylindrical NaI (Tl) scintillation detector is used. Components of greater than 3 MeV vary depending on the conversion factor, but they are typically equivalent to approximately 3 nGy/h (corresponding to 1.2×10^2 cpm). The cosmic ray intensity varies with latitude and altitude, but when a 3 in. φ × 3 in. cylindrical scintillator is used, it is calculated as 0.02 nGy/h per 1 cpm counting rate at above 3 MeV.

[2] Contribution of self-contamination

With regard to the contribution of self-contamination, radiation from ^{40}K contained in the glass materials of the detector is the only factor to be considered. The contribution of ^{40}K is, for the purpose of practical applications, calculated by multiplying the peak counting rate of 1.46-MeV gamma rays of ^{40}K measured in a shield, which is practically unaffected by ^{40}K from outside, by 0.16 nGy/h/cpm or 0.04 nGy/h/cpm for scintillators of dimensions 3 in. φ × 3 in. or 2 in. φ × 2 in., respectively (Table B.2). Usually, in the case of 0.9 nGy/h or less, and when glasses with little potassium are used for the window surface of a NaI (Tl) scintillator and photomultiplier tube, the contribution of ^{40}K is rarely observed.

Table B.2 Contribution of contamination by ^{40}K in a NaI (Tl) scintillation monitor

Dimensions of detector	Conversion factor from the total absorption peak counting rate of 1.46 MeV (nGy•h^{-1}/cpm)
2 in. φ × 2 in.	1.6×10^{-1}
3 in. φ × 3 in.	4.0×10^{-2}

B.2.3 Measurement and calibration

(1) Considerations of stability

What is most important for this type of monitor is to consider the temperature characteristics and circuit adjustment. The annual temperature of the environment varies very widely from approximately −10 °C to + 35 °C. Therefore, measures against temperature changes should be taken when a monitor is purchased or before it is used.

Because the temperature characteristics also vary depending on the calculus time constant of the pulse amplifier, it should be used under the pulse processing condition in which a highest stability can be secured in a range in which the time resolution is not impaired.

(2) Configuration of the monitor

The basic configuration of a monitor is shown in Figure B.8. However, when the discrimination between natural radiation and facility radiation using information on the energy distribution and the separation measurement of cosmic ray components are considered, the system shall be configured with a multichannel analyzer (MCA) or a single channel analyzer (SCA) combined.

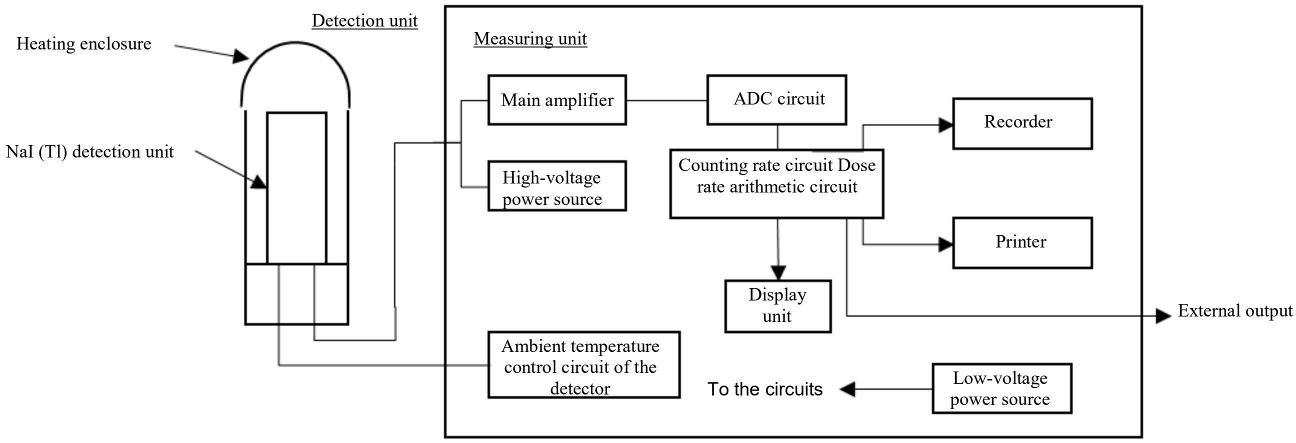


Figure B.8 Block diagram of a NaI (Tl) scintillation monitor by G(E) function method

(3) Data recording system

A data collection and processing device is one of the large components of auxiliary equipment of the monitor. In monitoring environmental gamma rays, a long-term continuous observation is usually conducted, and the results are analyzed to carry out a separate evaluation of dose contributions by cause. In many of the monitors currently in use, information such as digitally processed pulse-height distributions, dose rates, and counting rates in arbitrarily set energy ranges are stored on electronic media. In addition, the information mentioned above is output on a chart paper or something similar as a backup.

(4) Determination of the measuring time

When a 3 in. φ × 3 in. cylindrical NaI (Tl) scintillation detector is used to obtain the cumulative counting time for one data lot of measured values, a measuring time of several minutes is sufficient to obtain adequate accuracy from the statistical changes in the count values. In determining the measuring time, the temporal patterns of precipitation, fallouts, and the contribution of nuclear facilities must be considered. In general, natural radiation does not decay faster than the effective half-life of 35–40 min for descendant nuclides of radon, even when the dose rate rises rapidly owing to rainfall, snowfall, etc., and the fluctuation is usually relatively slow. On the contrary, the contribution of facilities usually indicates a rapid rise and fall and a short fluctuation period of several minutes to tens of minutes. However, there are various patterns that cannot be determined as attributable to facilities. It is desirable to use a device that can appropriately select the measurement time, considering these characteristics. For example, several to approximately 10 min should be selected to obtain the cumulative counting time for one lot of data, such that the variation of natural radiation and that of radiation attributable to facilities can be distinguished based on the difference in fluctuation periods. To reproduce the patterns more faithfully, one method is available in which the cumulative time is set to approximately 1 to 3 min and a smoothing or average operation is carried out in the subsequent data processing; it is relatively easy in the currently employed systems to rearrange patterns after they are collected by the telemeter.

For cases where the device is used in a traveling survey system, a different concept from the above is presented in Explanation F.

(5) Sensitivity adjustment and calibration

Because the sensitivity of a monitor using the G(E) function method varies as the gain of the amplifier changes, an accurate ratio of pulse-height to energy (e.g., 1.50 V/0.662 MeV) that is suitable for the measurement system must always be provided. Hence, the gain is adjusted prior to calibration. This adjustment is generally performed by the manufacturers, and the method is outlined here for reference.

[1] Sensitivity adjustment

Set the device to the operating state and adjust the gain such that the pulse-height value corresponding to the total absorption peak of ^{137}Cs 0.662-MeV gamma rays and the total absorption peak are established as specified.

At present, information on the pulse-height distribution spectrum can be obtained in many continuous monitoring systems. This information can be used for gain adjustment. In addition, some systems are currently available that can automatically perform the adjustment mentioned above by placing, for example, a source of ^{137}Cs in the vicinity of the detector. In this case, performed process is not to adjust the gain but to raise or lower the applied voltage for the adjustment. This method of adjustment is acceptable in many cases; however, when more a detailed adjustment is required, the gain needs to be adjusted.

[2] Calibration

There are two methods of calibration: A: a method using a reference gamma-ray source, and B: a method using the spectrum-dose conversion operator method to evaluate a reference gamma-ray source at an arbitrary location. In either case, the monitor is first set to its normal operating state using a reference source, prior to calibration.

A Method using a reference gamma-ray source

In this case, the conversion factor is obtained from the dose rate and the counting rate using the inverse square law of distance. The more scattered the rays, the more inaccurate the calibration becomes; therefore, special attention should be paid to scattered rays during calibration and efforts should be made to maintain the accuracy (for the evaluation of scattered rays, see Explanation D).

B Method of evaluating the reference gamma-ray dose using the spectrum-dose conversion operator method

In this case, the effect of the errors in certified values and the scattered rays (these are included in the evaluation) is small compared with the reference gamma-ray source method. However, the error of the function used for the conversion and the difference in the conditions between the detector used in determining the function and the calibrated detector, for example, the difference in absorption due to the difference in the thickness of the probe container, contribute to reducing the accuracy of the evaluation.

Sensitivity matching and sensitivity confirmation after calibration are all carried out using the reference source that was used for sensitivity adjustment. These sources are preferably available in more than one type. In addition, when a source is used, the amplifier output is provided to a pulse-height analyzer to ensure that there is no drift in the total absorption peak channel.

(6) Measurable dose rate range

[1] G(E) function method

Monitors of environmental radiation are required to be capable of measuring as wide a dose-rate range as possible, from natural radiation levels to high dose-rate levels in case of an accident. In a "continuous monitor," which is a fixed type, a NaI monitor and an ionization chamber monitor are installed together, with each monitor responsible for its respectively suitable dose-rate range; thus, a wide range of dose rates of environmental radiation is covered.

The flux density per unit dose rate varies greatly depending on the energy of the gamma rays, and in particular, the upper limit dose rate saturated owing to the temporal resolution of a pulse at low energy levels is lower than that at higher energy levels. This is because counting loss is produced during the dead time associated with the resolution of the pulse amplifier. This characteristic is related to the detector size, photon energy, and the counting rate (dose rate). The relationship between the air absorbed dose rate and the detector is shown in Figure B.9. When the photon energy is around 100 keV in the cylindrical detectors of dimensions 2 in. φ × 2 in. and 3 in. φ × 3 in., a maximum of approximately 2 μGy/h would be needed in order to achieve an accuracy of approximately 10% for the response relative to the air absorption dose rate. The level of switching to an ionization chamber monitor may be determined by referring to the photon energy and the counting rate, rather than the dose rate.

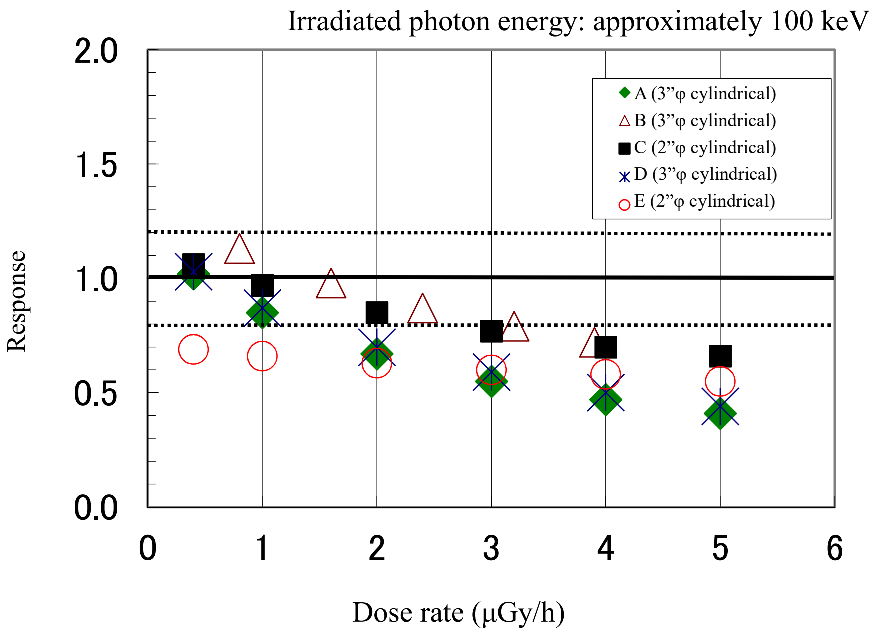


Figure B.9 Example of dose rate dependence in a NaI (Tl) scintillation monitor

Explanation B.3 Silicon semiconductor detector

B.3.1 Principles and configuration of the monitor

The measurement principle of the silicon semiconductor detector is similar to that of the ionization chamber dosimeter. However, because the information obtained is a pulse, a dose rate can be calculated by multiplying the pulse by a factor used to convert into a dose rate. Note that, owing to the measurement principle mentioned above, it is not possible to obtain energy information as with the ionization chamber dosimeter.

B.3.2 Characteristics of the monitor

(1) Dose rate linearity

Owing to the characteristics of the detector, it is suitable for measuring a high dose rate. Therefore, this offers a measurable dose rate range up to hundreds of mGy/h, which is difficult to measure with a NaI (Tl) scintillation detector.

An example of the dose rate linearity of 10 μGy/h to 100 mGy/h is shown in Figure B.10.

The dose rate linearity of a silicon semiconductor detector is good within a range of ±10% for the irradiation values at all dose rates, and there is no decrease in sensitivity owing to counting loss, which is observed in a NaI (Tl) scintillation detector.

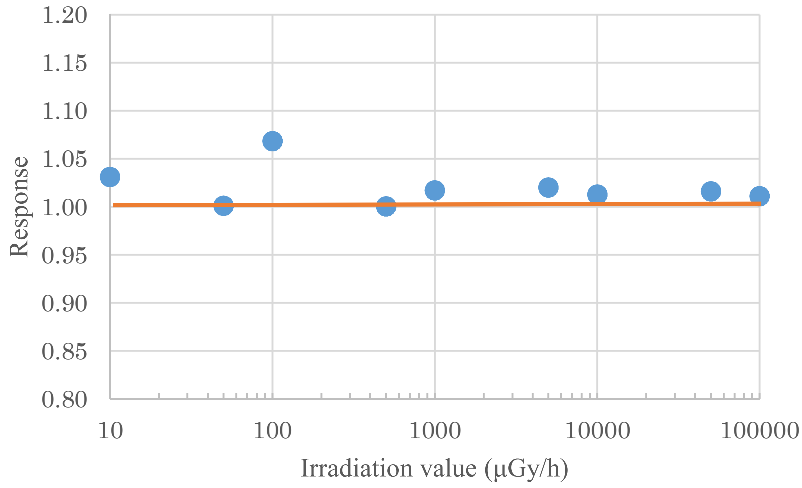


Figure B.10 Example of the dose rate linearity in a silicon semiconductor detector

(2) Directional characteristics

The directional characteristics of a silicon semiconductor detector vary, depending on the configuration of the measurement system, shape of the detector, etc. A graph showing the directional characteristics of a system is presented in Figure B.11 as an example. This graph shows the directional characteristics, with the irradiation direction from the effective center of the detector to the vertical upward direction set to 0°, including the directions up to ±90° in steps of 30°. The response of each angle is within a range of ±10%, indicating a good response. In addition, this is within a range of ±20% of the standard of JIS Z 4325: 2008 and meets the standard.

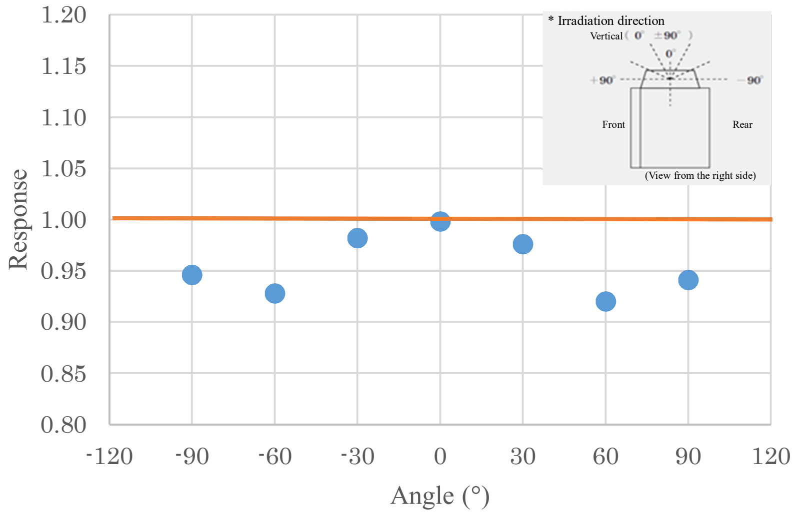


Figure B.11 Example of the directional characteristics in a silicon semiconductor detector

(3) Energy characteristics

Figure B. 12 shows the response of each energy normalized by the response of ^{137}Cs 662-keV gamma rays. The air absorbed dose rate for each energy is approximately 1000 µGy/h.

The response of each energy has decreased by an energy of approximately 50 keV, which is considered to be due to the detector cover acting as a shield. Additionally, an excessive response is observed around 80 keV, which is affected by the photoelectric effect. The overall energy characteristics are mostly within the range of the standard of JIS Z4325: 2008, except for 50 keV, which was affected by the detector cover acting as a shield.

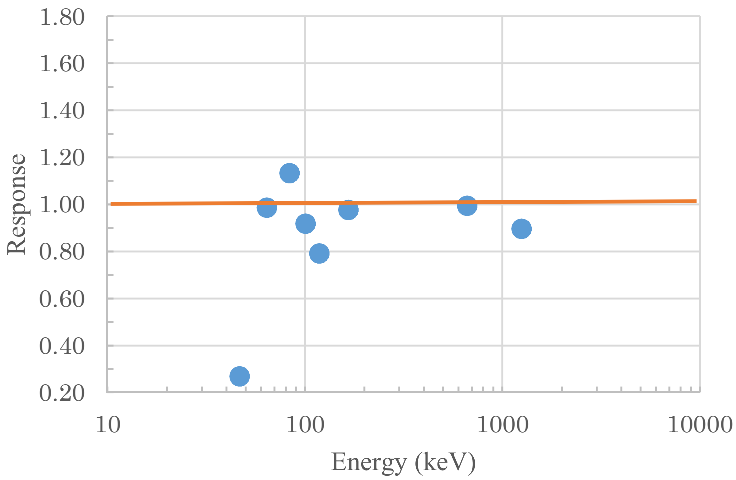


Figure B.12 Example of the energy characteristics in a silicon semiconductor detector

Explanation B.4 CsI (Tl) scintillation detector

B.4.1 Principles and configuration of the monitor

As with a NaI (Tl) scintillation monitor, the G(E) function method is used to evaluate the absorbed dose and ambient dose equivalents. In the monitor using a CsI (Tl) scintillation detector, the range of measurable energy varies depending on the type of photodetector it is combined with. The specifications should be selected in accordance with each application and the features of the monitor. Table B.3 lists the features of typical scintillators and photodetectors used in gamma-ray detectors. The main feature of a CsI (Tl) scintillation detector is its higher density compared with that of a NaI (Tl) scintillation detector. This means that the detector has a high sensitivity despite its small size, and a very compact system configuration is possible by combining a photodetector with a compatible wavelength sensitivity. However, the attenuation constant is larger than that of a NaI (Tl) scintillation detector. This means that after a gamma-ray photon is incident on the detector and the scintillator emits light, a long time is required for the light to attenuate. For this reason, when measuring at a high dose rate (a high counting rate), incident photon counting loss occurs during the attenuation. Therefore, because of possible underestimation, attention should be paid to the handling of the measured value

when the dose rate (counting rate) is high. Especially in case of an emergency, because it is assumed that the dose rate is high owing to the contribution from nuclides of the low-energy side, attention should be paid to the setting of an upper limit of the measurable dose rate, depending on the ratio of each nuclide that accounts for the contribution of the dose rate. However, owing to advances in electronic technology, some systems are currently available that have improved the counting characteristics by reducing the time required for in-circuit processing.
It should be noted that the CsI (Tl) scintillation detector described here is utilized in a system for obtaining spectral information, while in some systems for other applications, energy information cannot be obtained, and the energy range to be measured may be different from that described here.

Table B.3 Features of a scintillator and a photodetector

Phosphor	Density[a)] (g/cm³)	Attenuation constant[a)] (ns)	Deliquescent[a)]	Maximum fluorescent wavelength[a)] (nm)	Maximum sensitivity wavelength of a photodetector (nm)	
					PMT[b)*1]	MPPC[c)*2]
NaI(Tl)	3.67	230	Yes	410	420	500
CsI(Tl)	4.51	1000	No[*3]	565		

a) Reference 12; b) Reference 13; c) Reference 14

*1 PMT: Abbreviation for a photomultiplier tube

*2 MPPC: Abbreviation for multi-pixel photon counter “MPPC” is a registered trademark owned by Hamamatsu Photonics Co., Ltd.

*3 Some references suggest that it is slightly deliquescent.

Because this is assumed to be used as a continuous monitor (particularly as a traveling survey system), this section describes a typical example of a monitor that can also measure the low-energy side, which is essential in an emergency. The following descriptions are cited from the report on the KURAMA-II traveling survey system (Reference 15).
When a photon is incident on the CsI (Tl) crystal, the optical signal is amplified via multi-pixel photon counter (MPPC), a multi-pixel semiconductor photodetection element, which enables high magnification at low voltage. Then, the pulse-height data or dose rate data processed in the circuit on the substrate can be output via a USB cable, and when directly connected to a personal computer, the pulse-height spectrum data can be acquired together with the dose rate using a dedicated software package. Power is supplied via the USB, and the energy range of gamma rays to be measured is from 30 to 2000 keV. The configuration of the measuring instrument is shown in Figure B.13. In this system, the G(E) function method is adopted for calculating the dose rate, with the specification of calculating the dose rate while compensating the energy.

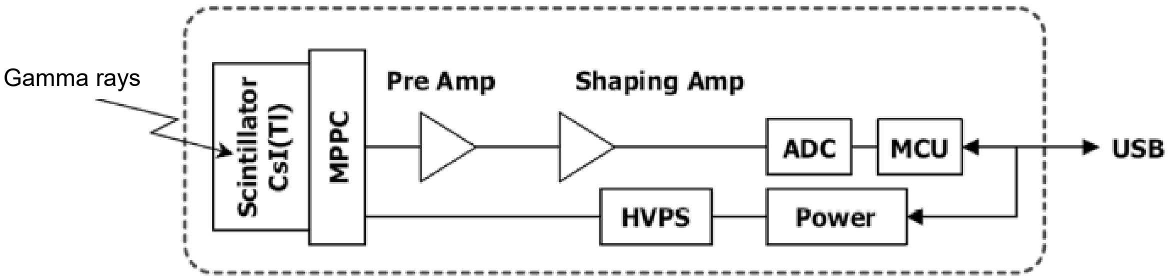


Figure B.13 Example of the configuration of a CsI (Tl) scintillation detector

B.4.2 Characteristics of the monitor

(1) Dose rate linearity

Owing to the structure or nature of the measuring instrument, the sensitivity decreases by the effect of counting loss during a high dose rate (a high counting rate), as with the NaI (Tl) scintillation detector. In the range of 0.2 µSv/h (the ambient dose equivalent rate) to 30 µSv/h at 662 keV of ^{137}Cs, the response change is approximately ±15%. In addition, when the dose rate increases further, the indicated value decreases by 30% at 100 µSv/h, and by approximately 50% at 200 µSv/h. Figure B.14 shows a graph of the dose rate linearity.

With the improvement of electronic circuits, the processing speed has increased, and the dose rate linearity has been improved as a result. The latest characteristics must be determined from the specifications of the employed equipment.

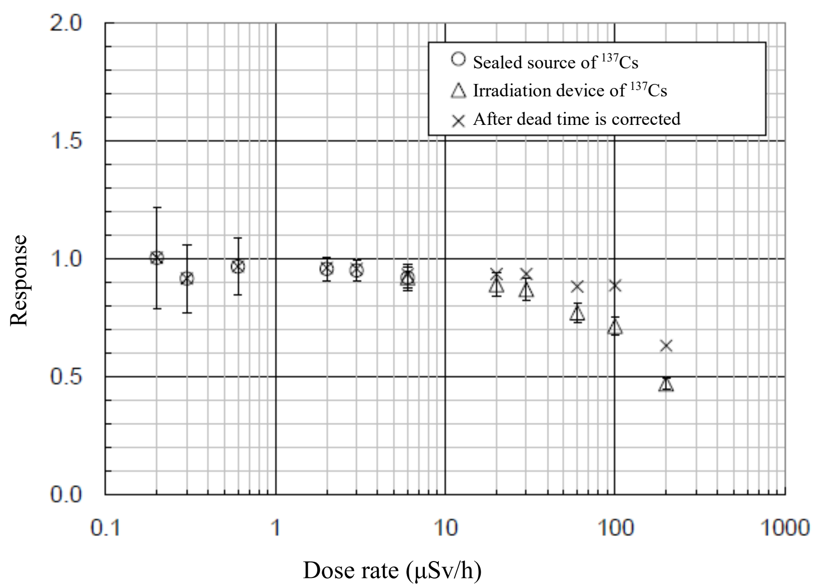


Figure B.14 Dose rate linearity in a CsI (Tl) scintillation detector

(2) Variation in the indicated values

When the variation in the indicated value is evaluated from the mean value (the indicated value) and the coefficient of variation by repeatedly measuring at each dose rate with a measuring time of 3 s, a value of less than 15%, which is the standard of JIS Z 4333: 2014, is obtained at a dose rate of approximately 0.2 μSv/h or more. The relationship between the mean value (the indicated value) and the coefficient of variation is shown in Figure B.15.

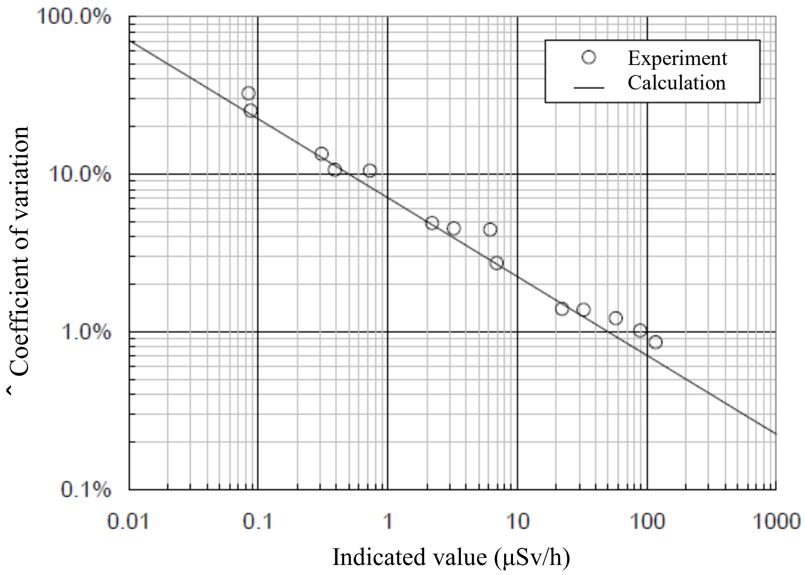


Figure B.15 Variation in the indicated value in a CsI (Tl) scintillation detector (calculated as 40 cpm per 0.01 μSv/h)

(3) Incident directional characteristic

When normalized by the response in the direction of 0°, the sensitivity slightly decreases from the rear of the detector (180°) to the direction in which the substrate for data processing and transmission is placed (270°), but the response change in all directions is in the range of approximately −23% through +10%. Figure B.16 shows a diagram representing the directional characteristics.

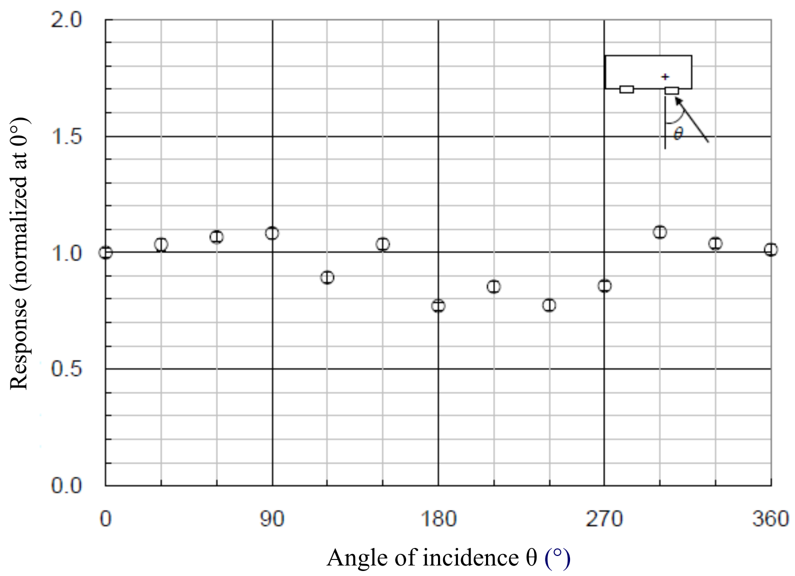


Figure B.16 Directional characteristics of a CsI (Tl) scintillation detector (this directional characteristic exists when the detector is placed in a container dedicated to the traveling survey system KURAMA-II.)

(4) Energy characteristics

Figure B.17 shows the relationship between the gamma-ray energy (effective energies are 60 keV of ^{241}Am, 340 keV of ^{133}Ba, 662 keV of ^{137}Cs, and 1250 keV of ^{60}Co) and the response when gamma-ray photons of each energy are irradiated in a measuring system combined with a CsI (Tl) scintillation detector and the G(E) function method. Simulation results obtained from Monte Carlo calculations are also presented. Although a slight decrease in the response of ^{241}Am is observed because the enclosure acts as a shield, relatively good energy characteristics are exhibited at other energies; thus, this effect is not necessarily considered when using the detector.

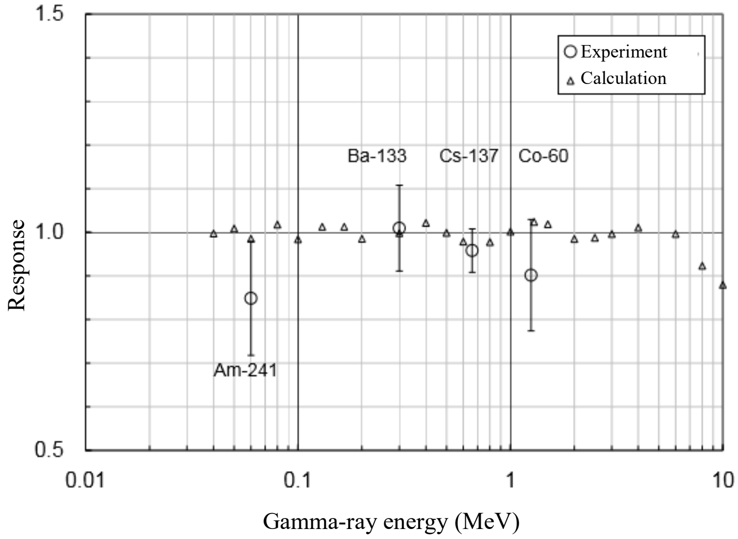


Figure B.17 Energy characteristics in a CsI (Tl) scintillation detector

singleExplanation C Reference gamma-ray source

Explanation C.1 Application of a reference gamma-ray source
A reference gamma-ray source is a gamma-ray source that has a certified irradiation dose rate at a specified distance (normally 1 m) from the source and well-defined traceability; this is used to calibrate the monitor.

Explanation C.2 Reference gamma-ray sources and dose rates
Gamma rays in the environment have an energy spectrum ranging from tens of keV to several MeV, and the dose rate is approximately 50 nGy/h. Considering these facts, the five nuclides, ^{241}Am, ^{57}Co, ^{133}Ba, ^{137}Cs, and ^{60}Co, described in the main text were selected as the reference gamma-ray sources for the calibration of a continuous monitor, based on the conditions of having appropriate half-lives and easy availability. The sources of ^{241}Am, ^{57}Co, and ^{133}Ba were intended for checking the energy characteristics in the low-energy region, and the radioactivity of approximately 10 MBq was specified. The reference gamma-ray sources are easy to move when used in a certified device with a display, and thus are convenient for calibrating a monitor installed in the field.

Explanation C.3 Accuracy of calibration of dose rates
For reference gamma-ray sources of ^{137}Cs and ^{60}Co, the accuracy of dose rate calibration is approximately ±10%, except in special cases, and approximately ±30% for those of ^{241}Am, ^{57}Co, and ^{133}Ba.

Explanation C.4 Obtaining a reference gamma-ray source
Reference gamma-ray sources are supplied from the Japan Radioisotope Association. The Japan Radioisotope Association receives high-precision primary references from the National Institute of Advanced Industrial Science and Technology, and it certifies and supplies reference gamma-ray sources.

Explanation C.5 Certification of dose rates for gamma-ray sources
C.5.1 Certification by the comparison method
Reference gamma-ray sources are certified by a specific secondary measurement standard calibrated by a national standard. However, in some cases, the dose rates of other gamma-ray sources can be calibrated by the reference gamma-ray source at the facility and used as a practical reference gamma-ray source. In this case, the reference gamma-ray source and the gamma-ray source to be calibrated should be of the same nuclide. An intermediary detector is not particularly restricted, as long as it has good stability and energy characteristics. As shown in Figure C.1, take a certain distance L from the detector, and use Qs as the indicated value of the reference gamma-ray source. Next, assume that the indicated value is Q, when substituted by the gamma-ray source to be calibrated. Assuming that the dose rate at 1 m from the reference gamma-ray source is $\dot{X}s$, the dose rate $\dot{X}$ at 1 m of the gamma-ray source to be certified is obtained by the following equation.

$$\dot{X} = \dot{X}s \frac{Q}{Qs} \tag{C.1}$$

Because the above equation is independent of L, a distance can be selected according to the dose rate. Measurements should be repeated at least twice and the results should be averaged. When the gamma-ray source is approximately 10 MBq, the monitor currently in use can be used as an intermediary detector.

C.5.2 Certification by the reference measuring instrument

A gamma-ray source whose dose rate is certified by a high-precision ionization-chamber-type dosimetry meter, which is the reference measuring instrument, can be specified as a reference gamma-ray source.

See Explanation D for the contribution of scattered rays.

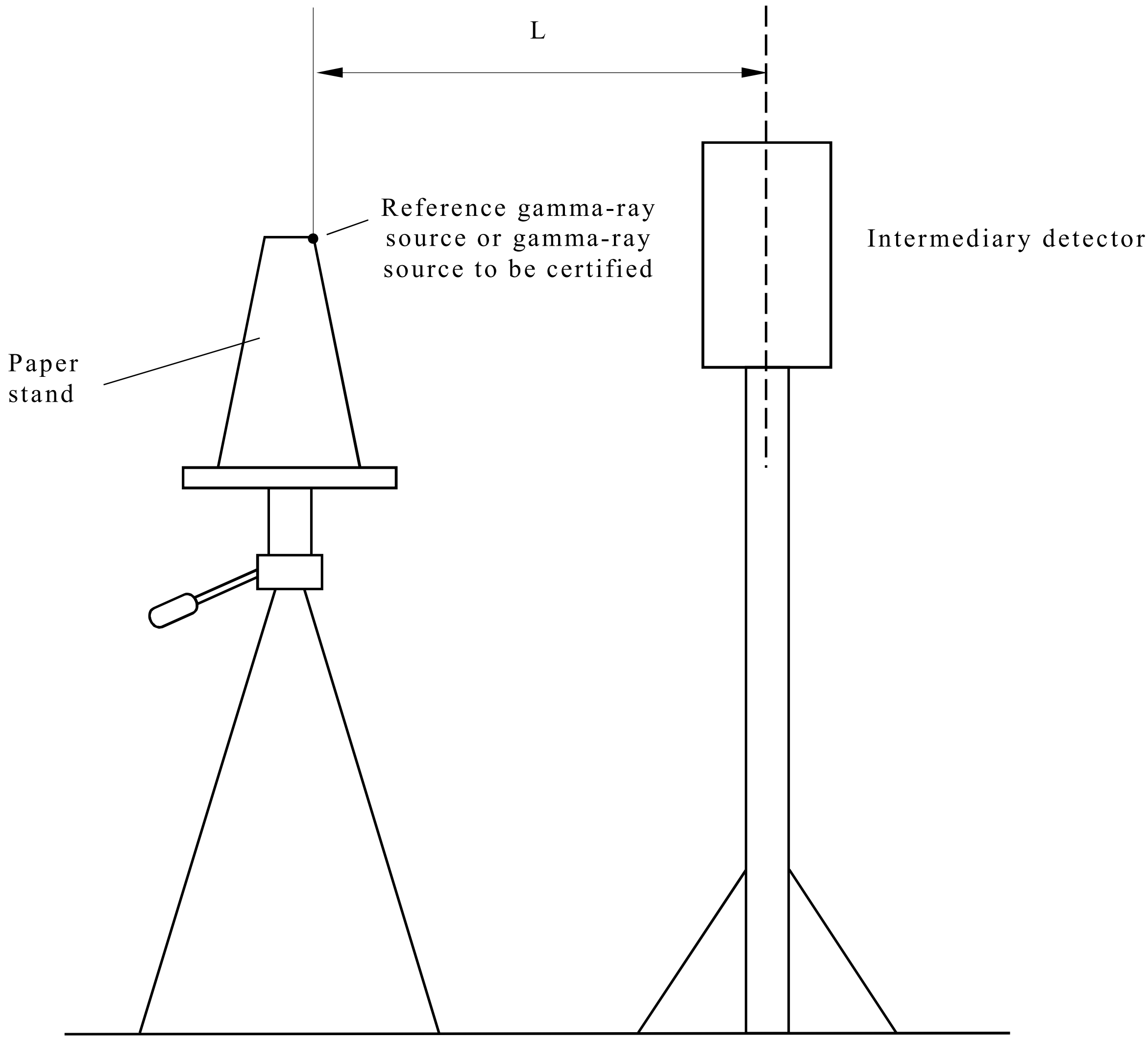


Figure C.1 Example of the comparison method for certifying the dose rate of a gamma-ray source

Explanation D Countermeasures against scattered rays during calibration

The dose rate $\dot{X}$ at the time of calibration by a reference gamma-ray source is the sum of the dose rate $\dot{X}$d due to direct gamma rays from the source and the dose rate $\dot{X}$b due to scattered gamma rays from the surroundings.

$$\dot{X} = \dot{X}\mathrm{d} + \dot{X}\mathrm{b} \qquad \text{(D.1)}$$

The contribution of scattered gamma rays $\dot{X}\mathrm{b}/\dot{X}\mathrm{d}$ depends on the distance between the source and detector, the height of the source and detector from the ground surface (or the floor surface), the surrounding objects, etc.; within a certain range, the closer the source to the detector, the higher the source and detector are from the ground surface, and the fewer and farther the surrounding objects, the smaller is the contribution of scattered gamma rays.
To conduct a calibration, the extent of the contribution of scattered gamma rays should be known. Additionally, the energy of the scattered gamma rays is generally considerably lower than that of the direct gamma rays, and the effect of the scattered gamma rays may be seemingly large depending on the characteristics of the detector. Calibration should be preferably conducted in a state where the contribution of scattered gamma rays is small.
A continuous monitor may be installed in various situations, but it is assumed in this case that the detector is installed at a certain height above a wide ground surface. The calibration direction (a line connecting the source and the detector) is supposed to be parallel or perpendicular to the ground surface (Figure D.1).
Figures D.2 and D.3 show data for the contribution of scattered gamma rays when the calibration direction is parallel to the ground surface. The curve depicted in Figure D.2 is obtained semi-experimentally for the case where the ground surface is concrete, and a ^{60}Co source is used. Experimental results obtained using a ^{60}Co source under various conditions are also shown in this figure. Table D.1 also includes the results for differences between the source and detector heights, as well as the results for a ^{137}Cs source.
In addition, according to the results of a measurement taken by placing gamma-ray sources at a height of 1 m from the detector on the roof of the monitoring post, the contribution ratio of scattered gamma rays was 2.3% for an iron roof and 2.5% for a concrete roof when a ^{60}Co source was used, and 2.6% and 3.5%, respectively, when a ^{137}Cs source was used.
Available methods for measuring the contribution ratio of scattered gamma-ray sources include the shadow shield method and the spectrum measurement method, among others. Here, the shadow shield method, which is based on a simple principle, is explained.
In the shadow shield method, as shown in Figure D.4, a lead block is placed between a radiation source and a detector, and scattered gamma rays from the surroundings are measured by blocking the direct radiation. The lead block should be thick enough to attenuate the direct radiation (to a level of approximately 1/1000). The support base of the lead block should be of a light material, such as foamed polystyrene (so that it would not act as a scatterer).
Assume that the indicated value without a lead block is Id, and the indicated value with a lead block is Is; then, the contribution ratio η of the scattered gamma rays to the direct gamma rays is given by:

$$\eta = \mathrm{Is}/(\mathrm{Id} - \mathrm{Is}) \qquad \text{(D.2)}$$

The scattering from the support base of the lead block should be checked. If it is not negligible, obtain η by the following equation. In other words, if Io is the value indicated when only the support base is placed, it is expressed as (Is and Id are the same as in equation (D.2)):

$$\eta = \frac{Is - (Io - Id)}{Io - Is} \tag{D.3}$$

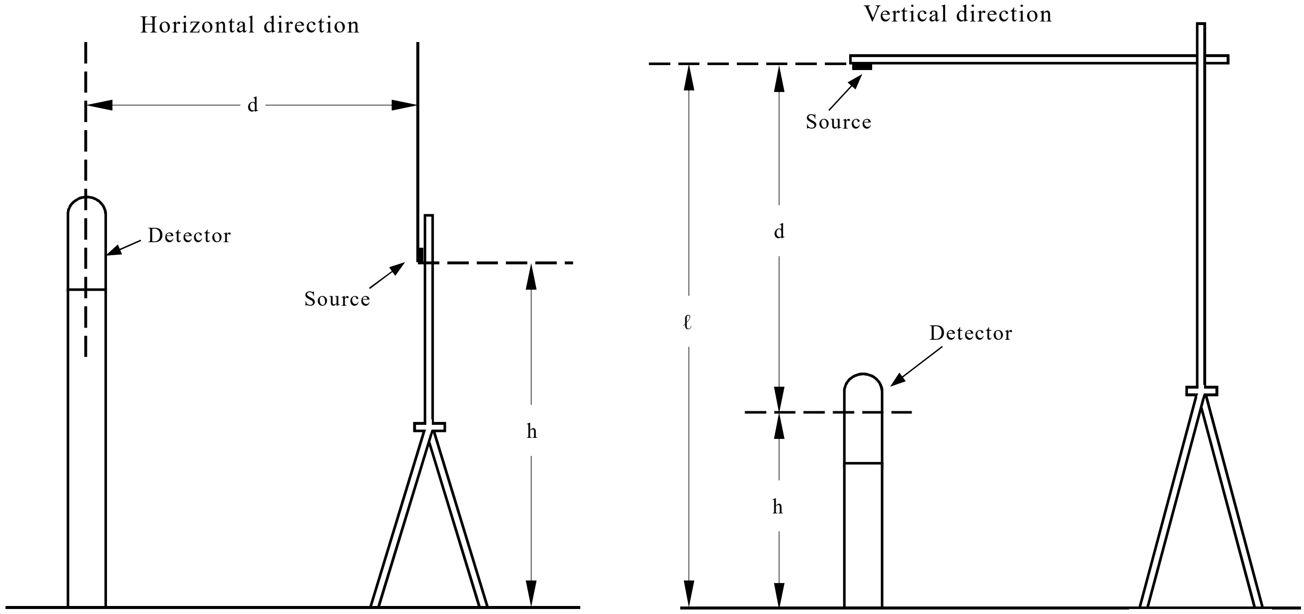


Figure D.1 Placement of the source and the detector during calibration

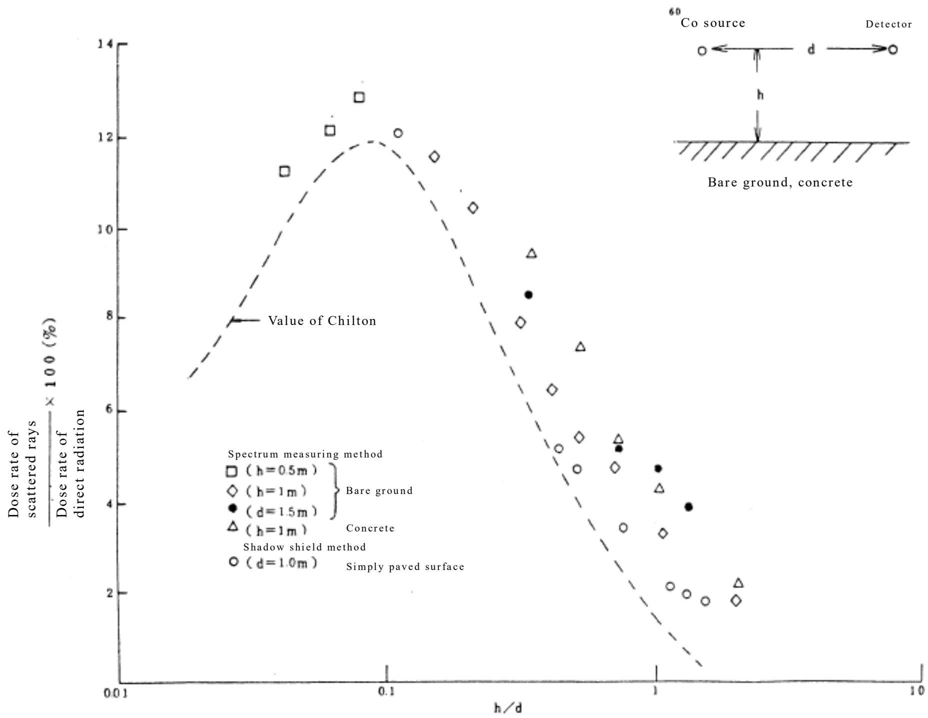


Figure D.2 Semi-experimental results of Chilton, et al. (the curve) and the measurement results

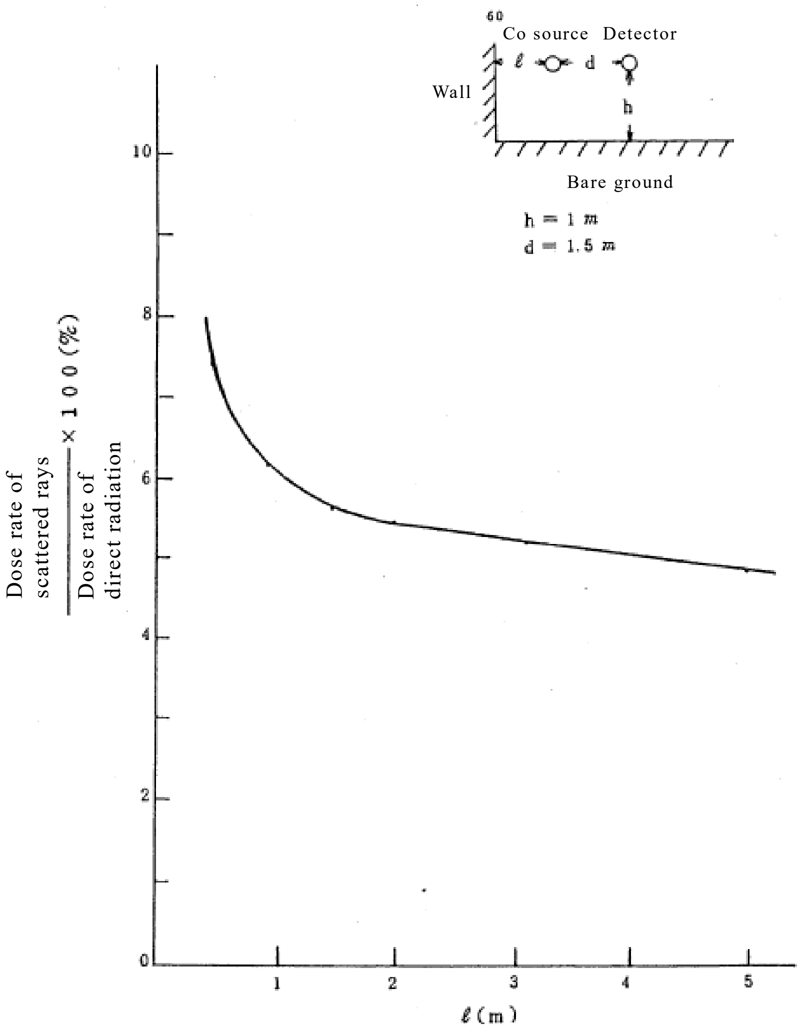
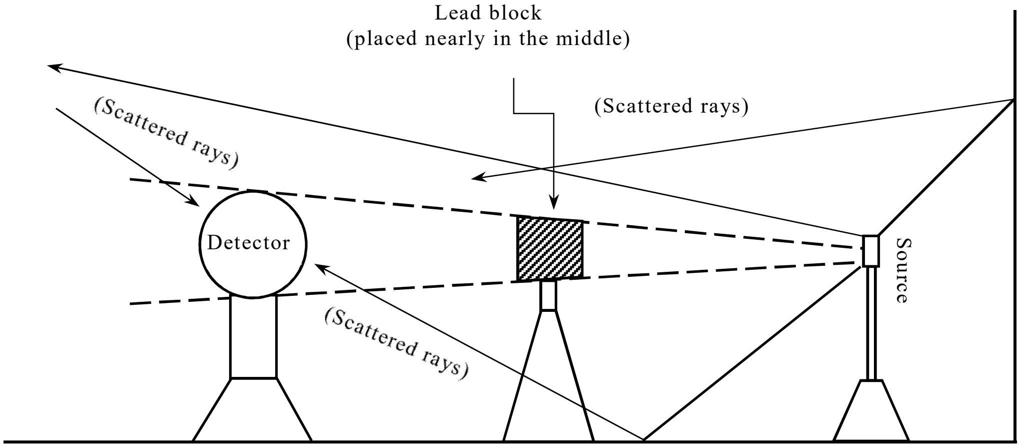


Figure D.3 Contribution of scattered rays in the presence of light-weight block walls

Table D.1 Contribution of scattered gamma rays by concrete or similar materials $\left(\frac{\text{Dose rate of scattered rays}}{\text{Dose rate of direct radiation}}\right)$

h/ℓ \ d/h	0.25	0.5	1.0	2.0	4.0	8.0	16.0	32.0	64.0
	^{137}Cs (0.662 MeV)								
1	0.002	0.007	0.024	0.062	0.107	0.134	0.118	0.083	0.053
2	0.017	0.026	0.049	0.088	0.127	0.132	0.102	0.068	0.042
4	0.052	0.060	0.081	0.113	0.138	0.125	0.090	0.059	0.036
8	0.087	0.094	0.109	0.132	0.142	0.118	0.082	0.053	0.032
16	0.115	0.119	0.129	0.145	0.143	0.113	0.077	0.049	0.030
32	0.133	0.135	0.142	0.152	0.142	0.109	0.074	0.047	0.029
64	0.144	0.146	0.150	0.156	0.141	0.106	0.072	0.046	0.028
128	0.150	0.152	0.155	0.158	0.140	0.105	0.071	0.045	0.027
∞	0.161	0.162	0.164	0.163	0.140	0.105	0.071	0.045	0.027
^{60}Co (1.25 MeV)									
1	0.001	0.004	0.014	0.038	0.073	0.109	0.110	0.083	0.054
2	0.010	0.015	0.029	0.057	0.093	0.115	0.099	0.069	0.043
4	0.030	0.035	0.050	0.076	0.108	0.113	0.088	0.059	0.037
8	0.052	0.057	0.069	0.092	0.116	0.109	0.081	0.053	0.033
16	0.070	0.074	0.083	0.103	0.119	0.105	0.076	0.049	0.030
32	0.083	0.085	0.093	0.110	0.120	0.102	0.072	0.047	0.029
64	0.091	0.093	0.099	0.114	0.119	0.099	0.070	0.046	0.028
128	0.095	0.097	0.103	0.117	0.119	0.098	0.069	0.045	0.028
∞	0.103	0.105	0.111	0.121	0.120	0.097	0.069	0.044	0.027

h: height of the detector, ℓ: height of the source, d: horizontal distance between the source and the detector (Reference 16)



Scatterers (walls, floors, bare ground, etc.)

Figure D.4 Layout of the shadow shield method

Explanation E Considerations of continuous measurement using a transportable monitoring post

Explanation E.1 Key points to be considered

As described in the main text, when continuous measurement is carried out using a transportable monitoring post, the following key points are to be considered.

[1] Understanding of the measurement method
[2] Temperature characteristics
[3] Power source
[4] Selection of the installation site
[5] Maintenance and management

Of these points, [3] and [4] are described in the text; thus, [1], [2], and [5] are explained here.

Explanation E.2 Understanding of the measurement method

A transportable monitoring post was originally commercialized for the purpose of having one device to temporarily act as continuous monitor (a NaI monitor and an ionization chamber monitor) as an installed station, at a site where continuous monitoring is required in case of an emergency. This means that a wide range of measurable dose rates is required, ranging from the normal low dose rate to a high dose rate for which an ionization chamber monitor is responsible. Therefore, each manufacturer addresses this problem using a different measuring method and a different detector. Table E.1 summarizes the various measurement methods.

Dose rate measurements can be taken over a wide range, depending on which combination of detector and measurement method is selected. However, the features of such combinations are diverse, and some of their effects are advantageous, whereas others are disadvantageous. When purchasing a device, the specifications should be examined, considering the necessity and cost of acquiring energy information. In addition, the features of the device must be understood, when used, and data should be evaluated considering the specifications.

Explanation E.3 Temperature characteristics

As mentioned in the text, a temperature controller is not usually equipped. Essentially, a temperature controller is designed to suppress the change in response due to temperature characteristics, using a temperature compensation circuit. However, these have not yet successfully followed rapid changes in temperature due to direct sunlight, etc. Figures E.1 and E.2 show examples of the temperature characteristic test results taken by the manufacturer before shipping, and Figures E.3 and E.4 show the effect of temperature changes during continuous measurements when the device is installed outdoors. Note that because the devices used for Figures E.1 and E.2, and Figures E.3 and E.4 are not from the same manufacturer, they cannot be completely compared.

In the NaI detector shown in Figure E.1, the sensitivity tends to be enhanced with a rise in temperature and decrease when the temperature falls. In the silicon semiconductor detector shown in Figure E.2,

the sensitivity tends to decrease when temperature rises, and increase when the temperature falls. A similar trend is observed in Figures E.3 and E.4, which show the effect of temperature changes during continuous measurements. However, while the change during temperature characteristic testing is approximately ±5%, the response to the effect due to a sudden temperature change outdoors may contribute to a maximum change of approximately 30%.
Therefore, the following points should be considered for the temperature change in a transportable monitoring post: (1) the tendency of changes in sensitivity is exactly opposite between a NaI detector and a silicon semiconductor detector, and (2) changes in sensitivity may be greater when exposed to a large temperature change outdoors, than during temperature characteristic testing (temperature shock).

Explanation E.4 Maintenance and management

As mentioned above, diverse variables are identified because of a “transportable” type. If this system is installed for a long period of time and continuous measurements are carried out, the following maintenance methods will be helpful.

(1) Comparative measurement

To confirm that a transportable monitoring post is operating properly, perform measurements at the same ground level in the vicinity of the detector of the transportable monitoring post, using another sound measuring instrument (such as a survey meter) that has been calibrated and inspected, and compare the measured values. However, if there is significant nonuniformity in the distribution of ambient dose rates, caution should be exercised in the comparison.

(2) Inspection and calibration

Inspection and calibration are effective for maintaining the integrity of measuring equipment, but in the case of a transportable monitoring post susceptible to temperature, inspection and calibration should be carried out quarterly, considering the seasonal temperature conditions.

Table E.1 Various measuring methods and their features

Upper row: Low dose rate areas
Lower row: High dose rate areas

Detector		Measuring method	Features
NaI	NaI(Tl) scintillation detector	• G(E) function method • DBM method	• Energy characteristics are good • Energy information can be obtained • Pay attention to the lowering of sensitivity owing to counting loss
		• Current system	• Measurements up to a high dose rate are enabled using the same model of detector • Dose rate dependency is good • Energy information cannot be obtained
NaI + semiconductor	NaI(Tl) scintillation detector	• G(E) function method • DBM method	• Energy characteristics are good • Energy information can be obtained • Pay attention to the lowering of sensitivity owing to counting loss
	Silicon semiconductor detector	• Pulsed system	• Measurements up to a high dose rate are enabled • Energy information cannot be obtained • Note changes in response owing to switching of a detector
Semiconductor	Silicon semiconductor detector	• Pulsed system	• Low dose rates can also be measured by adding elements • Because it is sensitive to cosmic rays, caution should be exercised during measurement of a low dose rate • Energy information cannot be obtained • Measurements up to a high dose rate are enabled

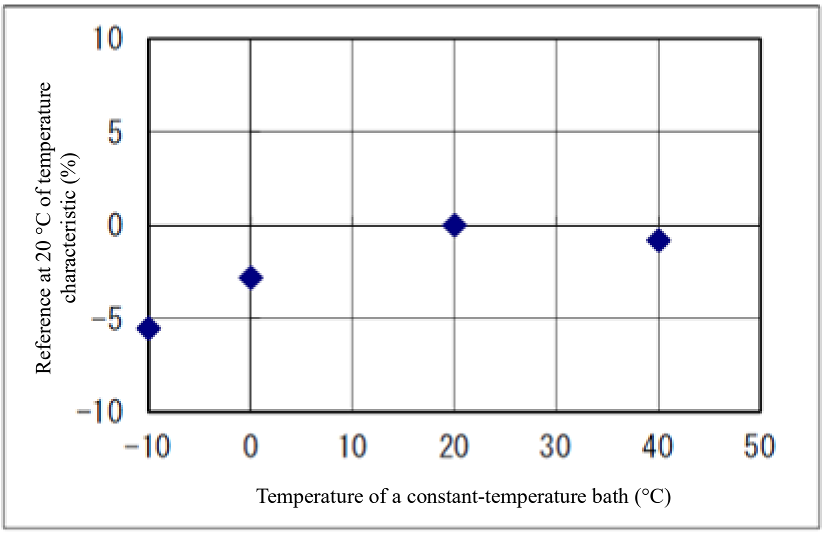


Figure E.1 Example of temperature characteristic test results of a transportable monitoring post (NaI)

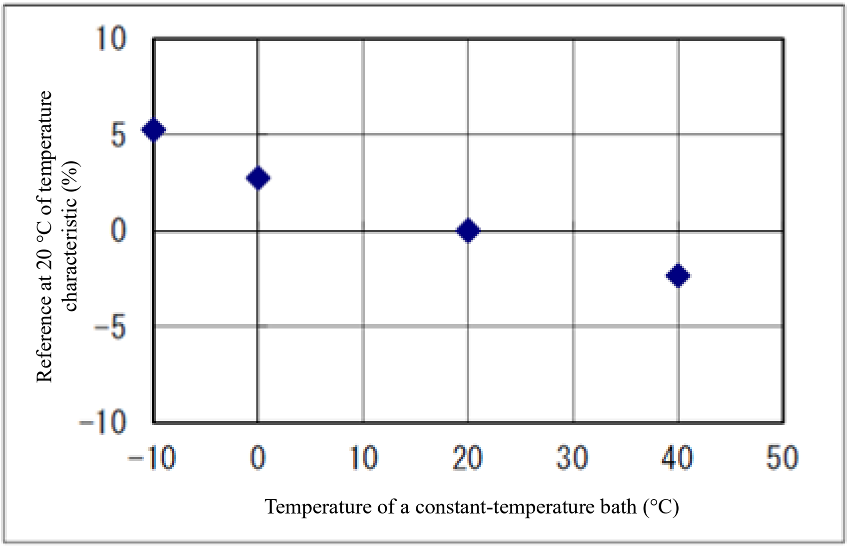


Figure E.2 Example of temperature characteristic test results of a transportable monitoring post (silicon semiconductor)

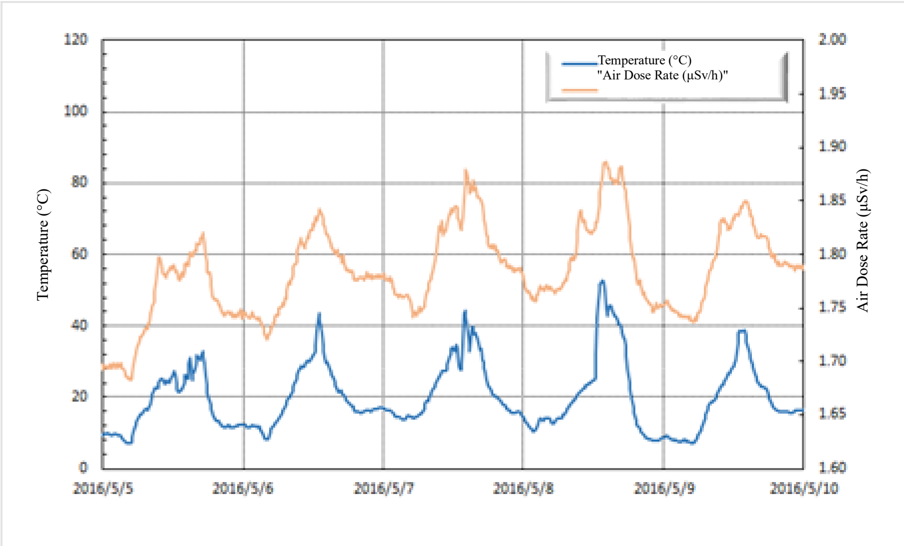


Figure E.3 Effect of temperature changes during continuous measurement by a transportable monitoring post (NaI)

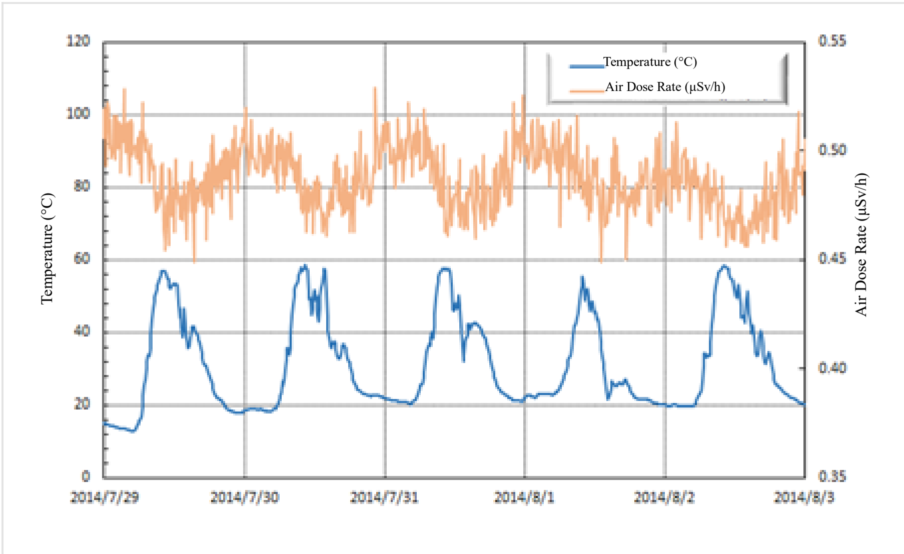


Figure E.4 Effect of temperature changes during continuous measurement by a transportable monitoring post (silicon semiconductor)

Explanation F Considerations of the traveling survey system

Explanation F.1 Overview

Monitoring using a traveling survey system is a measuring method for efficiently conducting monitoring over a wide area while moving by using a vehicle loaded with measuring equipment, which has a function of measuring the environmental gamma-ray dose rate.

From the data obtained by this method, it is possible to know the dose rate distribution mainly on and along the road, as well as to detect a point with a locally high dose rate (a hotspot) with relative ease. These data are helpful in assessing the exposure dose across a region, and in facilitating the decontamination of hotspots.

Explanation F.2 Considerations of conducting monitoring

The following points should considered when conducting monitoring using a traveling survey system.

F.2.1 Measuring time and intervals for collection

Because this monitoring is carried out while moving, unlike an installed monitor, the point of measurement (the target range) is always changing. To correctly grasp changes in the point of measurement (the target range), the interval of collecting measured data should be short. This leads to the acquisition of the mapping resolution to be used in preparing a dose rate distribution map and the data in which the points are subdivided to find hotspots.

In the monitoring currently conducting by a traveling survey system, the vehicle is not traveling at a very low speed that would impede the traffic of other vehicles, but it is running at a legal limit. Table F.1 presents the relationship between the elapsed time and the travel distance when traveling at a speed of 40 km/h.

Table F.1 Relationship between the elapsed time and the travel distance

Elapsed time (s)	Travel distance (m)
1	11
3	33
5	56
10	111
20	222
30	333
60	667

(When travelling at a speed of 40 km/h)

Although the measurement ranges are not necessarily the same, if the detector is installed at a height of 1 m above the ground in a direction perpendicular to the ground surface, the contribution ratio of the gamma-ray dose rate from within a radius of 10 m is 60–80%, as shown in Figure F.1. From Table F.1 and Figure F.1, if the elapsed time is 1–3 s (the measuring time and the interval of collection), the travel distance is short, and subdivided data can be obtained.

Because there are various installation conditions in each traveling survey system, it is desirable to know in advance the extent of the contribution ratio to each installation condition.

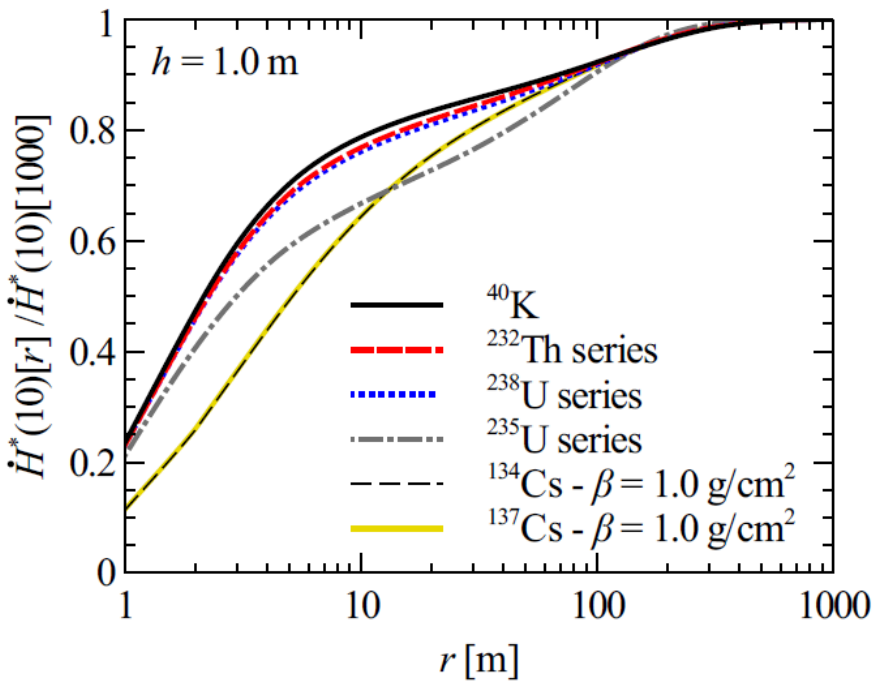


Figure F.1 Contribution ratio from the surroundings with regard to the ambient dose equivalent at 1 m above ground when natural radionuclides are uniformly distributed in the ground and ^{134}Cs and ^{137}Cs are exponentially distributed (β: 1.0 g/cm^2) (Reference 17)

F.2.2 Measurement accuracy

In “F.2.1 Measuring time and intervals for collection,” it was stated that a measuring time of 1–3 s is sufficient to obtain subdivided data, but if the measuring time is short, the measurement accuracy naturally decreases. Explanation B.4 shows the coefficient of variation for measurements repeatedly conducted at each dose rate for a measuring time of 3 s. As a result, it is understood from the experiences acquired after the Fukushima Dai-ichi NPS accident that, in a would-be target area for monitoring with a traveling survey system, if its dose rate is approximately 0.2 μSv/h, measurements could be conducted using a coefficient of variation of approximately 15%.

To establish measurement conditions for conducting a monitoring using a travelling survey system, refer to the description above.

In addition, Figure F.2 shows an example of a comparison of the results from measurements taken at certain fixed points along the road, under the conditions of 3-s measurement times, for reference. From these results, a good correlation is exhibited between the traveling survey results and the air dose rate of 1 m above ground at some fixed points, and the measured values of 1 m above ground at these fixed points are approximately 1.2 times higher than those of the traveling survey results. In other words, the monitoring results by the traveling survey system, whose target areas include an area on the road, are apt to show lower values than the fixed points in the surrounding areas, owing to the “weathering effect” and the like, which is relatively large on the road. The impact of this effect depends on the number of days elapsed since the occurrence of a nuclear disaster. It must be understood that measured values contain various information as mentioned above.

Incidentally, the following explanation in this section is quoted from “Improvement of measurement infrastructure and application to actual measurements using a KURAMA-II travelling survey system” (Shuichi Tsuda, et al.), JAEA-Technology 2013 - 037 (Reference 15).

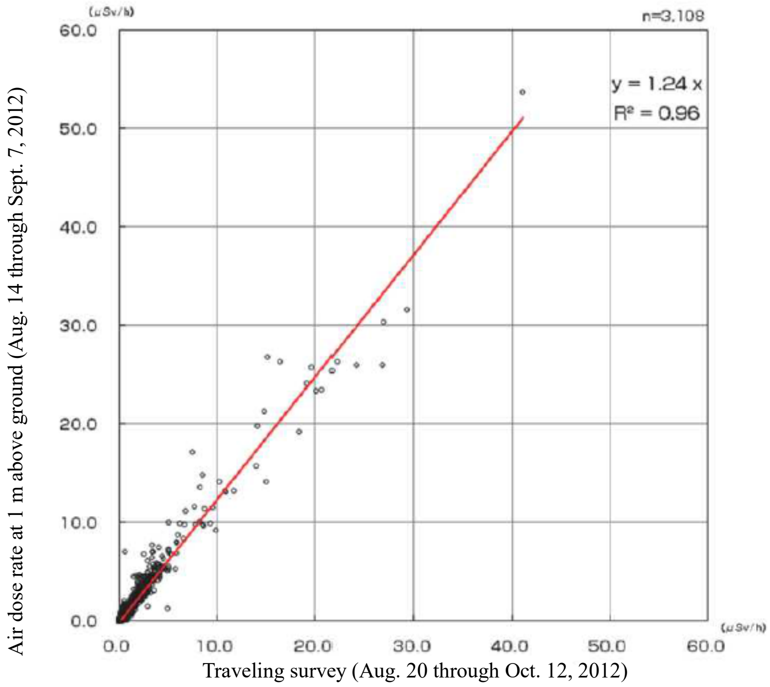


Figure F.2 Relationship between the measurement results by a traveling survey system (mean values in a 100-m square mesh) and the dose rate around the travel route (Reference 15)

F.2.3 Effect of shielding by the vehicle body

When a monitoring with a vehicle is conducted, the effect of shielding by the vehicle body should be recognized. This section explains the effect of shielding. In addition, when a detector is installed in a vehicle, the correction coefficient for inside and outside the vehicle must be calculated, which is also explained.

When a measurement is performed with the detector installed in the vehicle, the measured value is usually lower than that taken outside the vehicle. This is due to the shielding by the vehicle body. Figure F.3 shows an example of the results of comparing measurements taken inside and outside the vehicle.

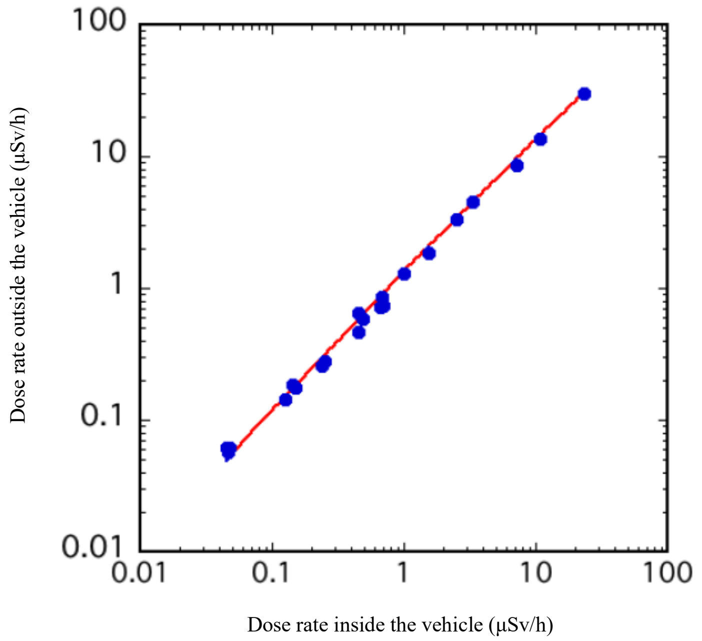


Figure F.3 Relationship between dose rates measured inside and outside the vehicle (Reference 15)

From the comparison results, a good correlation is considered to be obtained between the dose rates inside and outside the vehicle. However, when the dose rate decreases, a larger variation may occur. Because the magnitude of the influence on the variation in dose rates varies depending on the type of vehicle and the measurement environment, the influence on the variation in dose rates should be understood in advance according to the conditions used for measurement. Figures F.4 and F.5 show the results of comparisons made between vehicle types, for reference.

To confirm the relationship between dose rates inside and outside the vehicle, select multiple points with different dose rates, where radioactive materials are considered to be deposited (distributed) broadly and uniformly, take measurements by installing a detector in the predefined location in the vehicle, measure the dose rate outside the vehicle at a height of 1 m above ground, and compare the measured values.

In addition, this method of confirming the relationship between the dose rates inside and outside the vehicle can also be applied to cases at a normal environmental level. However, when the measurement is conducted at the points with low dose rate, the correction coefficient for inside and outside the vehicle can be utilized by increasing the statistical accuracy of the measurement; for example, by increasing the measuring time, although it depends on the size of the detector.

The influence of the shielding effect on the gamma rays from the downward direction, owing to the difference in the position where an in-vehicle detector is installed, can be confirmed by setting a gamma-ray source at the lower part of the vehicle body and performing measurements for the cases both when the vehicle body is present and absent. In addition, by acquiring a large set of data, not only from one position, but from various positions where the gamma-ray source is set, the detector can be installed at an appropriate position in the vehicle, where the shielding effect is low.

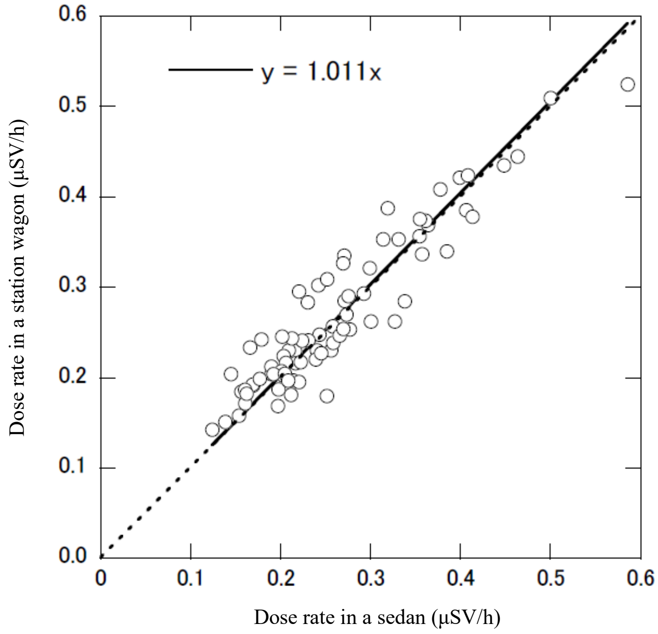


Figure F.4 Comparison of dose rates between vehicle types (a station wagon and a sedan) (Reference 15)

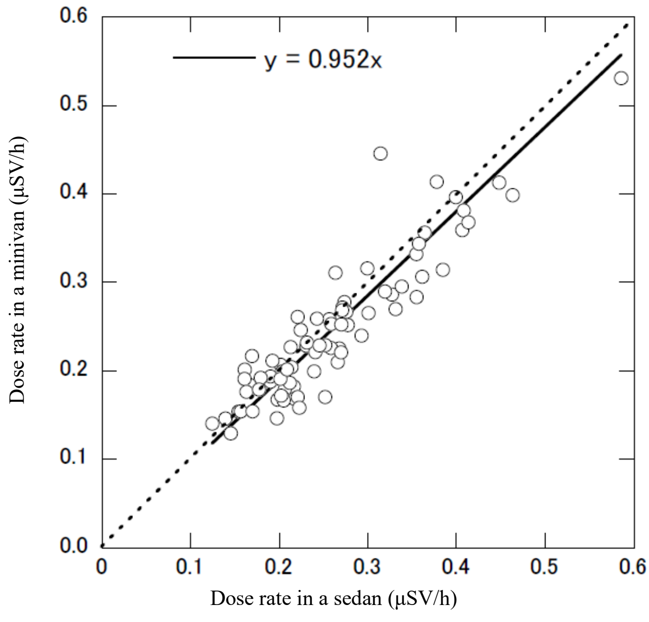


Figure F.5 Comparison of dose rates between vehicle types (a minivan and a sedan) (Reference 15)

Explanation G Analysis of measurement results

The time variation of the monitor readings is summarized in Table G.1. In other words, the causes of the variation in environmental gamma rays are roughly classified into natural phenomena, radionuclides for medical and industrial purposes, and those attributable to nuclear facilities. The analysis of measurement results is to extract these variations, examine their causes, determine the causes of variations, and perform a dose assessment. While the pattern of variations in measurement results, usually obtained by continuous measurement, is characterized by the rate and periodicity of variations, the cause of a variation is often known by its correlation with other information.
However, monitor readings involve essential variations, apart from the following, which are attributable to statistical phenomena, and they are roughly classified into those related to the meter and those related to the radiation source (the former includes pure counting statistics).
The analysis of measurement results includes the selection of populations, calculation of mean values and standard deviations, extraction of outliers, etc.; the selection of populations is particularly important. In recent years, it has become feasible to determine the outliers by analyzing energy information, such as a gamma-ray spectrum, obtained concurrently with the measured dose rate.

Explanation G.1 Selection of a population

In selecting a population, the following matters should be considered.

G.1.1 Measuring time and time constant

The measuring time is a time interval that is arbitrarily set to calculate the dose rate from information such as the measured and acquired counting. This is the measuring time that is described in this measurement method.
Apart from this, the time constant is defined for the radiation measurement equipment. The time constant is important when the instantaneous value (a value obtained in real time) being measured is read by the rate meter. The survey meter is a typical measuring instrument, and the NaI monitor and the ionization chamber monitor, which are used as continuous monitors, also have specified time constants, in addition to the measuring time. However, the time constant is not directly involved in the measuring time.
A measured value reaches 95% of the original value at a specified position after a period equal to three times the time constant elapses, and it reaches 99% after a period equal to five times the time constant elapses. Figure G.1 shows the relationship of the time change between the time constant and the measured value, using a survey meter as an example.
Many measuring systems are designed to select several settings for a time constant. If the setting of a time constant is short, the measured value can sense the original dose rate of the position as described above. However, a short setting of the time constant means that the variation in measured values increases, because of a lack of information, such as the counting, obtained by the measuring instrument at a specified interval of the time constant. In addition, a long setting of the time constant can suppress the variation in the measured values to be relatively small, in contrast to the above, but the time until a measured value is obtained increases. Table G.2 presents the variation in the measured values for each time constant at each dose rate.

Table G.1 Time variation in monitor readings

Cause of variations		Pattern of variations	Frequency of variations	Increase in amount
Variations due to natural phenomena	Rainfall and snowfall	• Gradual variation during rainfall • Increases and decreases are complicatedly mixed	Regional differences are produced approximately 100 times per year	≈ 100 nGy/h ≈ 50 μGy/y
	Lightning	Increases rapidly and decreases with a half-life of approximately 30 min	Regional variations are identified (often in winter on the Sea of Japan side)	In some cases, approximately tens of nGy/h at maximum
	Snow cover	Shielding effect of snow cover	Regional differences are identified	Reduced by approximately 10 to 30 nGy/h
	Other meteorological phenomena	Diurnal period due to the inversion layer	Common in winter	≈ 10 nGy/h
		Absorption of radiation by ground surface moisture		Reduced by approximately 2 nGy/h
Atmospheric nuclear explosion test		Variation is observed a few days after the test, and the increase is almost proportional to the elapsed time.		The shorter the number of days elapsed, the greater the increase, and in the case of a few days elapsed, the increase may be several times the normal level.
Nuclear facility		Not constant; especially on the leeward axis, an increase in dose rate and a short period of variation are observed.		
Characteristics of the measuring instrument		Mainly due to temperature change	Daily and yearly temperature changes	In some cases, it may range from a few percent to 10%, depending on temperature.
Failure of the measuring instrument		An excessive or underestimated value is indicated		

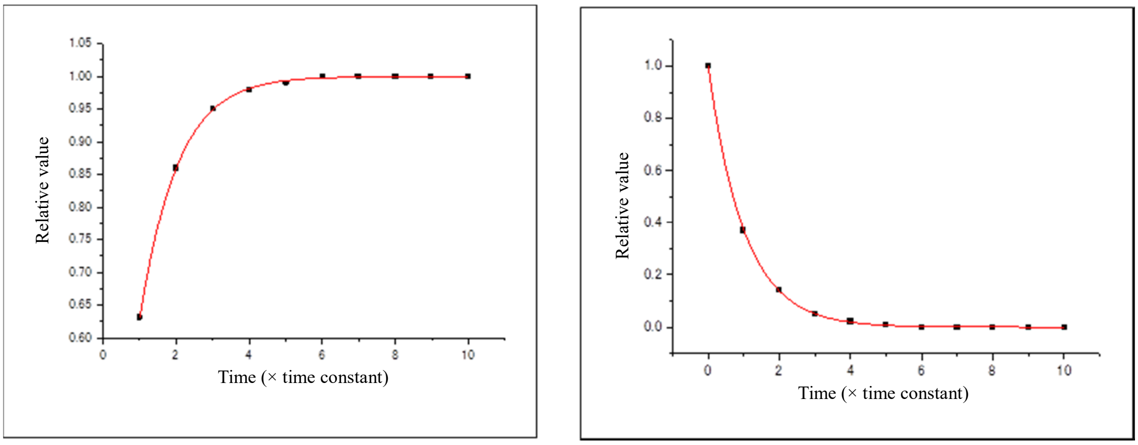


Figure G.1 Relationship of time change between the time constant and a measured value

Table G.2 Variation in measured values for each time constant and dose rate

τ: time constant

Dose rate (nGy/h)	Coefficient of variation (%)		
	τ (3 s)	τ (10 s)	τ (30 s)
10	77	42	24
20	54	30	17
50	34	19	11
100	24	13	8
200	17	9	5
500	11	6	3
1000	8	4	2
2000	5	3	2
5000	3	2	1
10000	2	1	1
20000	2	1	1

(1 in. φ × 1 in. cylindrical NaI (Tl) scintillation survey meter)

G.1.2 Time interval

The cumulative counting time for one lot of data or the time interval for a data printout must be commensurate with the rate of change in the dose rate of the event in question. The time interval used for natural phenomena such as rainfall ranges from several minutes to tens of minutes. When the variation based on nuclear facilities is considered, an interval of as short as a fraction or less is applied; however, in the case of fallout events, an interval of as long as several times or more is applied (note that natural phenomena include one-day and one-year cycles). Therefore, the data extraction interval for continuous measurement shall be specified as 5, 10, or 15 min as a standard, and if a short interval is required, 1, 2, 3, or 5 min shall be applied.

In the continuous measurement currently being conducted in Japan, the statistical error of the counting obtained at this standard interval is 2–5%, which is negligible for the variation to be detected. However, in the case of a few minutes or less, counting statistics (variations in the numerical values of the measurement results) should be considered.
These data are used to determine hourly values for subsequent analyses.

G.1.3 Quantity of data to be included in a population

The more data that are contained in a population, the closer it approaches to a true distribution; thus, it is desirable that more than dozens of data points are contained. It is easy to obtain such a quantity of data, except in the case of a population that consists only of precipitation data.
Populations are roughly divided into normal times, precipitation, snow cover, immediately after a nuclear test, and others.

(1) During normal times, it may also be useful to separate them by specific time and month, considering daily and annual variations.
(2) Precipitation can be classified into thunderstorm, light rain, typhoon, snowfall, and the nature of the front, but it is advisable to classify it into thunderstorm, other precipitation, and snowfall. In addition, because the radiation level decreases during snow coverage, a separate population should be defined for snow coverage.
(3) If an increase is observed immediately after a nuclear test, a different population should be prepared. Usually, because the fallout in the early stage of a nuclear test is attenuated at $t^{-1.2}$, where t is the elapsed time since the test was conducted, a separate population should be used for the contribution of the fallout by setting a time interval of a fraction of t, for example, by collecting the data every one day when five days have elapsed since the test, every two days when 10 days elapsed since the test, and so on, until the data fit within three times the standard deviation of the normal value.

Explanation G.2 Extraction of outliers and investigation of causes

An outlier is defined as a value that exceeds three times the standard deviation in the distribution specific to each population, and the following points should considered when investigating the cause of an outlier.
Normal values, with regard to the distribution of observed values during no rainfall, are usually expressed as a normal distribution (the distribution of observed values during no rainfall tends to spread around the bottom of the distribution in practice, owing to variations in soil moisture and atmospheric radioactivity (approximately 2 nGy/h at 3.7 Bq/m^3 of ^{222}Rn)). However, during rainfall, the high-dose-rate side is distributed almost in an exponential relationship. The standard deviation σ′ during rainfall is generally larger than the standard deviation σ during no rainfall.
The probability that observed values during no rainfall exceed the mean ± 3σ is approximately 0.3%, and the probability that observed values during rainfall exceed the mean ± 3σ' is approximately 2%.
Even given these considerations, thunderstorms can be cited as a natural phenomenon that still exceeds the range of three times the standard deviation.
When a monitor gives an excessive (or underrated) indication value, the process of extracting the outlier and investigating its cause usually starts with inspecting the monitor.
If it is confirmed that the monitor is operating normally, an examination should be made to determine whether it may be attributed to any natural phenomena such as precipitation. In doing so, compare the distribution with that of a population consisting of the data in normal times including precipitation, etc., or that of a population consisting of only the data of precipitation, if such data are available and sufficiently accumulated. The contribution of precipitation to the annual dose in Japan is approximately 10 μGy.

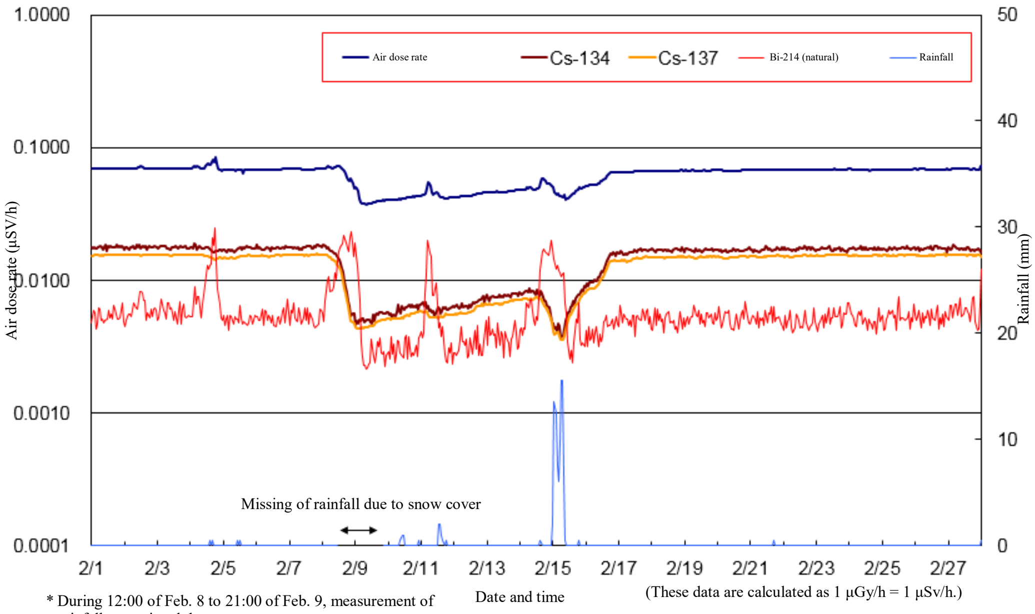
It is also conceivable that cosmic ray neutrons cause an error in the software of the semiconductor device.
If the causes have not yet been determined through the above processes, the influence of the nuclear facility should be suspected. In this case, examine the weather conditions (wind direction, wind speed, atmospheric stability, etc.), and review the correlation between the indicated values of multiple monitors located in different directions relative to the stack position of the facility, in order to determine whether the indicated values of the monitors located in the downwind direction are excessive. Needless to say, it would be preferred if information on the operational status of the facility could be obtained.
When it turns out that the cause is a plume from the stack, a relatively short cycle of fluctuation is often observed; therefore, it is useful for determining the causes to know the fluctuation pattern; however, to do so, the time interval of data extraction must be shorter than that for standard values. In addition, when the outlier is attributable to the facility, the energy distribution of gamma rays is usually different from that of normal times, and energy information may act as an influential factor.
If the cause is still unknown after the above procedure is executed, records must be kept and documents must be prepared for later examination.

Explanation G.3 Examples of variations in the environmental gamma-ray dose rate

G.3.1 Variations in dose rates due to natural changes in weather conditions, etc.

The discussion above is based on a statistical approach for the variation in the environmental gamma-ray dose rate, but in this section, some examples of variations typically observed in practice are given for reference.
It is well known that the environmental gamma-ray dose rate increases temporarily during rainfall and snowfall.
This is due to the following two phenomena: one is called “washout,” in which the radiation dose rate temporarily rises when descendant nuclides of radon, an inert gas floating in the atmosphere, fall on the ground surface along with raindrops or snow; the other is called “rainout” in which, at the stage of generating a cloud that brings about rainfall or snowfall, the cloud takes in radon and descendant nuclides of radon on the upward current, and raindrops containing radon and its descendant nuclides fall on the ground surface, thereby temporarily raising the dose rate.
In addition, when there is a large amount of rainfall or snowfall, a temporary decrease in the dose rate may be observed (in some cases, the effect of snow shielding may be prolonged), because gamma rays emitted from the earth crust are shielded by the moisture retained on the ground surface. Figures G.2 and G.3 show examples of variations observed in the dose rate due to meteorological phenomena.
After the Fukushima Dai-ichi NPS accident, it has been confirmed that, at the site with extremely high radioactive cesium concentrations deposited on the ground surface, the decrease in the dose rate due to the shielding effect of the ground surface water is more remarkable than the increase in the dose rate due to the fall of radon, etc., even in a small amount of rainfall. In the case of variations caused by natural phenomena in addition to those caused by weather such as rainfall, the changes in response due to changes in temperature have been confirmed. Refer to the description in Explanation E.3 for examples that depend on temperature changes.

Figure G.2 Example of an increase in the air dose rate due to rainfall



* During 12:00 of Feb. 8 to 21:00 of Feb. 9, measurement of rainfall was missed due to snow cover.

Figure G.3 Example of a decrease in the air dose rate due to snow cover

G.3.2 Variations in dose rates due to changes in the surrounding environment

Changes in the surrounding environment can bring about various effects on the measured values of a continuous monitor.

During monitoring in normal times, a temporary decrease in the dose rate due to the shielding effect of vehicle(s) parking near the continuous monitor, changes in the dose rate due to the construction of new concrete buildings, changes in the dose rate due to asphalt laying, etc. may be produced. Figure G.4 shows examples of such variations.

It can be seen from this figure that the dose rate also varies with changes in the surrounding environment, such as (1) soil (lawn), (2) asphalt, (3) concrete, and (4) other buildings. This is caused by the concentrations of natural radionuclides contained in materials used in the surrounding environment. Therefore, even if the surrounding is “(1) soil,” it does not necessarily exhibit the same trend in dose rate, and the dose rate also varies depending on the concentrations of natural radionuclides contained in the soil. Incidentally, granite (granitic rocks) is often used for stone monuments and similar structures built in shrines and temples that fall under “(4) other buildings.” It is generally known that such granitic rocks contain relatively high concentrations of natural radionuclides, and this effect may result in a slightly higher dose rate.

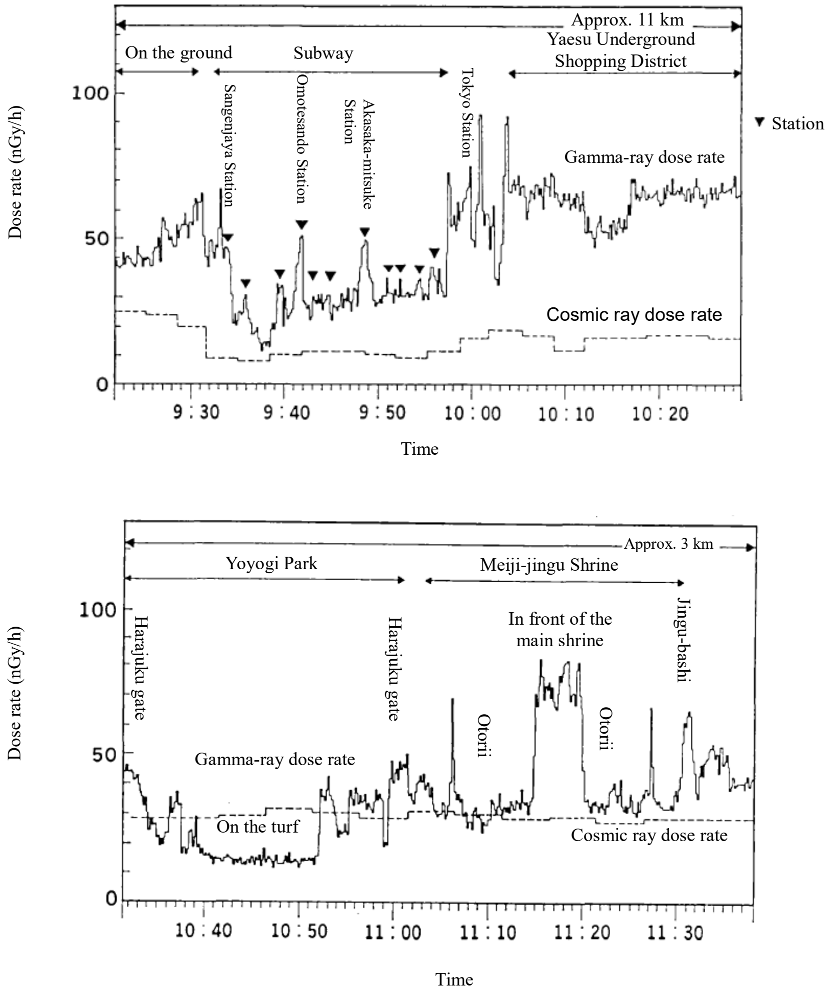


Figure G.4 Examples of variations in dose rate due to changes in the surroundings (Reference 18)

In addition, after the Fukushima Dai-ichi NPS accident, possible outcomes are a decrease in the dose rate due to decontamination, temporary increase in the dose rate due to temporary placement of decontamination waste in the vicinity of the continuous monitor, and temporary increase or decrease in the dose rate due to the installation and operation of a plant facility for treatment of decontamination waste liquid near the continuous monitor.

G.3.3 Variations in dose rate due to other factors

(1) Radioactive materials for medical use

At present, in Japan, radioactive materials are utilized in medicine by administering to the body and confirming their behaviors. As a result, some cases were reported in the past in which the dose rate increased when a person to whom medical radioactive material had been administered passed through in the vicinity of a continuous monitor. Table G.3 lists the major gamma-ray emitting nuclides used in medical applications. In addition, Figure G.5 shows a case of detection due to the influence of a person to whom medical radioactive material had been administered.

Figure G.5 shows the gamma-ray spectrum detected when a user with medical ^{67}Ga administered approached a continuous monitor (a NaI monitor). ^{67}Ga emits photons of multiple energies, as explained in Table G.3. In Figure G.5, a total of four peaks around 95 keV can be seen. In such a case, the nuclides corresponding to the respective gamma-ray photon energies can be confirmed for identification.

(2) Nondestructive inspection

For nondestructive inspection, industrial gamma-ray equipment is used for gamma-ray transmission testing. The major gamma-ray emitting nuclides in wide use are ^{192}Ir (0.317 MeV, etc.), ^{60}Co (1.332 MeV, etc.), ^{169}Yb (0.198 MeV, etc.), and others.

In addition, X-ray generators are sometimes used and a continuous monitor that is sensitive to X-rays often detects the effects. Figure G.6 shows an example of detecting the effects in a nondestructive inspection.

Some industrial X-ray generators conduct nondestructive inspections by irradiating X-rays in a low-energy region (tens of keV to 300 keV). The spectrum shown in Figure G.6 is obtained when X-rays in this low-energy region are detected. The energy of X-rays depends on the tube voltage applied to the X-ray tube in the X-ray generator, and the value varies depending on the purpose of each usage, among other factors. In addition, the energy distribution of the bremsstrahlung used for this nondestructive inspection is called a continuous spectrum, which has a wide energy distribution without a sharp peak such as that of gamma rays.

It should be noted that, unlike the bremsstrahlung, “the characteristic X-ray,” which is often mentioned when performing a gamma-ray spectrometry, shows an energy distribution with a peak similar to that of gamma rays.

Apart from the major variation factors described here, various other factors have been identified.

In environmental radiation monitoring, it is necessary to investigate the factors from the data showing variations and to confirm whether the contributions are due to artificial radionuclides.

To do so, the physical conditions at each point must be constantly acquired, and efforts should be made to obtain the accompanying weather data as much as possible. In addition, if a system capable of acquiring energy information is provided, the factors of variations can be quickly identified by closely monitoring the information along with the dose rate.

Table G.3 Examples of major gamma-ray emitting nuclides used as radiopharmaceuticals (References 19 and 20)

Nuclide	Half-life	Energy (MeV) and emission rate of major gamma rays
^{51}Cr	27.70 d	0.320 - 9.92% 0.0050 - 19.7% V-K_{α}
^{67}Ga	3.261 d	0.0933 - 39.2% 0.185 - 21.2% 0.300 - 16.8% 0.0086 - 50.3 % Zn-K_{α} 0.0096 - 6.8 % Zn-K_{β}
^{81}Rb	4.576 h Daughter ^{81m}Kr	0.190 - 64.0% ^{81m}Kr 0.466 - 23.2% 0.510 - 5.3%
^{99}Mo	65.94 h Daughter ^{99m}Tc	0.181 - 6.0% 0.739 - 12.1%
^{99m}Tc	6.01 h Daughter ^{99}Tc	0.141 - 89.1% 0.0184 - 6.1 % Tc-K_{α}
^{111}In	2.805 d	0.171 - 90.2% 0.245 - 94.0% 0.0232 - 67.8% Cd-K_{α} 0.0261 - 14.5% Cd-K_{β}
^{123}I	13.27 h Daughter ^{123m}Te	0.159 - 83.3% 0.0275 - 70.7% Te-K_{α} 0.0310 - 16.0% Te-K_{β}
^{131}I	8.021 d Daughter ^{131m}Xe	0.284 - 6.1% 0.364 - 81.7% 0.637 - 7.2%
^{133}Xe	5.243 d	0.0810 - 38.0% 0.0310 - 40.3% Cs-K_{α} 0.0350 - 9.4% Cs-K_{β}
^{201}Tl	72.91 h	0.167 - 10.0% 0.0708 - 73.7% Hg-K_{α} 0.0803 - 20.4% Hg-K_{β} 0.00999 - 46.0% Hg-L

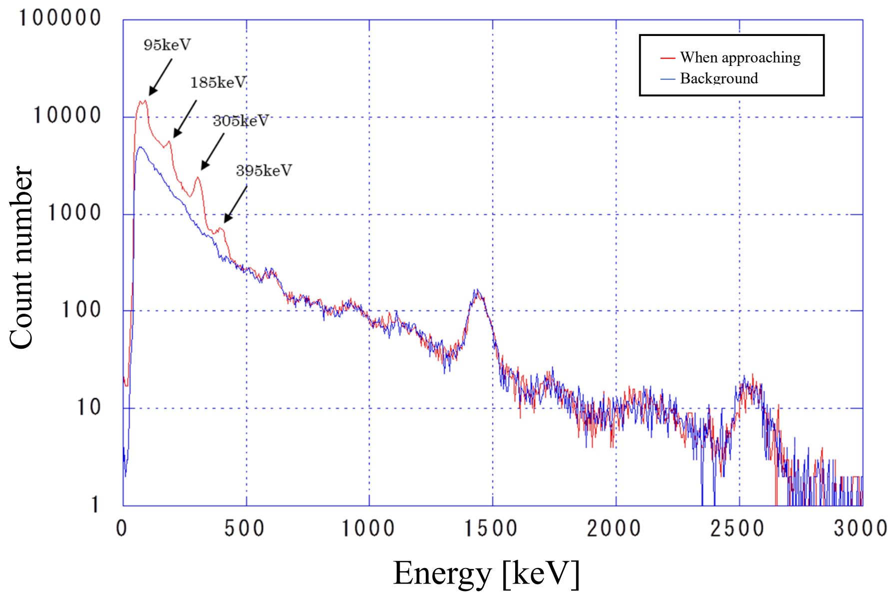


Figure G.5 NaI spectrum obtained when a person with medical radioactive material (^{67}Ga) administered approaches the detector

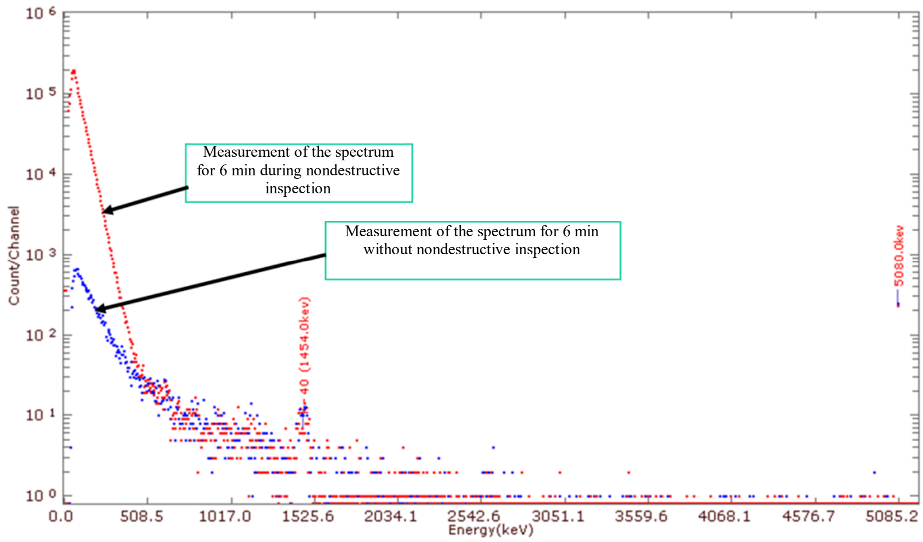


Figure G.6 Example of detection of the effect by X-ray generator during nondestructive inspection

Explanation H Integrity check of a continuous monitor by comparative measurement

Explanation H.1 Overview

Continuous monitors constantly perform measurements under conditions such as wearing weather, changing temperature, high humidity, and in some cases, even in poor environments. However, considering the significance of environmental radiation monitoring, the integrity of the measurement system must be maintained even under poor conditions. In addition, it is essential to maintain the integrity not only for the purpose of monitoring any abnormalities attributable to nuclear facilities, but also of utilizing the data obtained from continuous monitors for the exposure dose assessment after an accident, in order to help in assessing the exposure dose of the residents living in the vicinity. In this section, some examples are given to confirm whether an appropriate measurement value has been obtained as a representative of the relevant field, rather than identifying a clearly unsound situation including unmeasurable conditions and physical damages.

Explanation H.2 Method

To confirm whether the measured values obtained by the installed continuous monitor appropriately indicate the dose rate in the field, take measurements for comparison at the same place, using another measuring device that is properly adjusted. However, it is not easy to remove a continuous monitor that has already been installed. Therefore, the comparison and confirmation can be performed by analyzing data obtained through a comparative measurement using the following method.

(1) Selection of measurement points

Draw a straight line along at least three axes, such that it passes through the center of the continuous monitor installed. Select measurement points on this straight line, at an interval of approximately 0.25–0.5 m from the center of the continuous monitor.

(2) Measurement

Use a sound NaI (Tl) scintillation spectrometer with gain adjustment and electrical zeros confirmed and adjusted, and take measurements at the measurement points selected in (1). In doing so, a type of the measuring device that is capable of assessing the dose rate contribution of cosmic rays, with self-dose required, should be employed.

The selection of the points (1), and the results of the measurements taken at these points (2) are shown as examples in Figures H.1 and H.2.

(3) Analysis

The following explains how to perform an analysis based on the measurement results obtained.

[1] Calculate first, second, and third approximate equations for each axis.

[2] From each of the calculated approximate equations, obtain the value of the point where the continuous monitor is installed, through interpolation.

[3] For each axis, adopt an approximate equation whose error in the dose rate is smallest, which is obtained through interpolation at the point where the continuous monitor is installed.

[4] From the approximate equation adopted for each axis, obtain a weighted average of the values at the point where the continuous monitor is installed.

[5] Subtract the dose rate of the cosmic ray contribution and the self-dose from the obtained weighted average value, and calculate the pure environmental gamma-ray dose rate at the point where the continuous monitor is installed.

[6] Add the dose rate of the cosmic ray contribution and the self-dose, which are included in the measured values of the continuous monitor, to the calculated environmental gamma-ray dose rate, and compare it with the measured values of the continuous monitor.

Explanation H.3 Availability of comparative measurement

Comparative measurements are conducted on the assumption that the environmental gamma-ray dose rate does not fluctuate while measurements are being taken in the vicinity. Therefore, they must be carried out in a situation where no variations in the dose rate due to rainfall or similar phenomena and no changes in the surrounding environment (movement of vehicles, etc.) are observed.

In addition, as a condition for comparison, the dose rate of the cosmic ray contribution and self-dose must be evaluated. Therefore, the specifications of the measuring equipment should be fully understood.

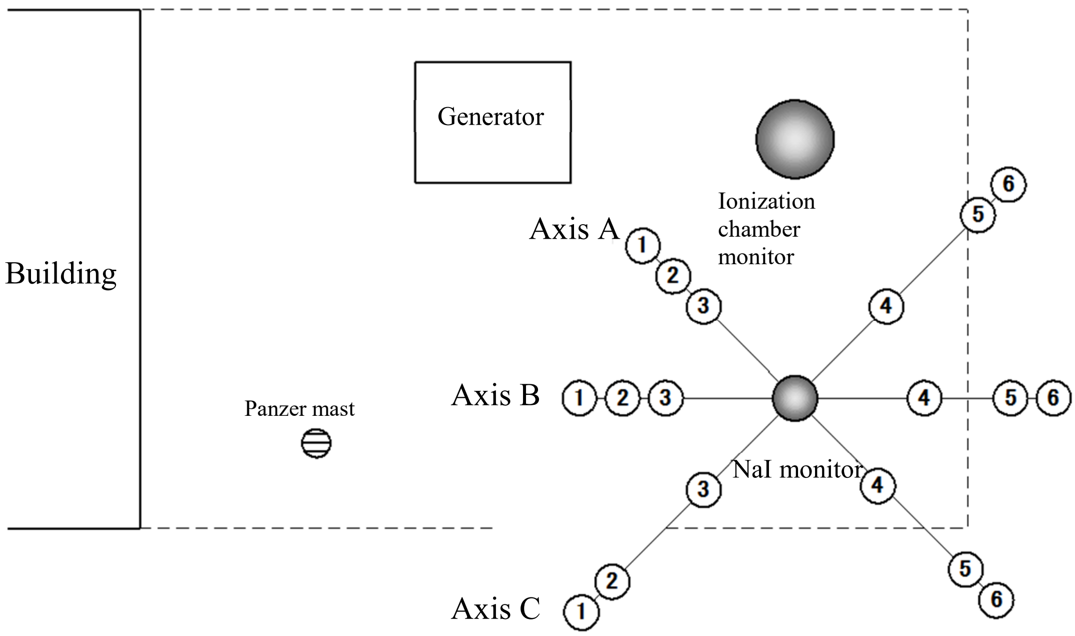


Figure H.1 Example of measurement points during comparison and schematic diagram (circled numbers indicate the numbers of the measurement points of each axis)

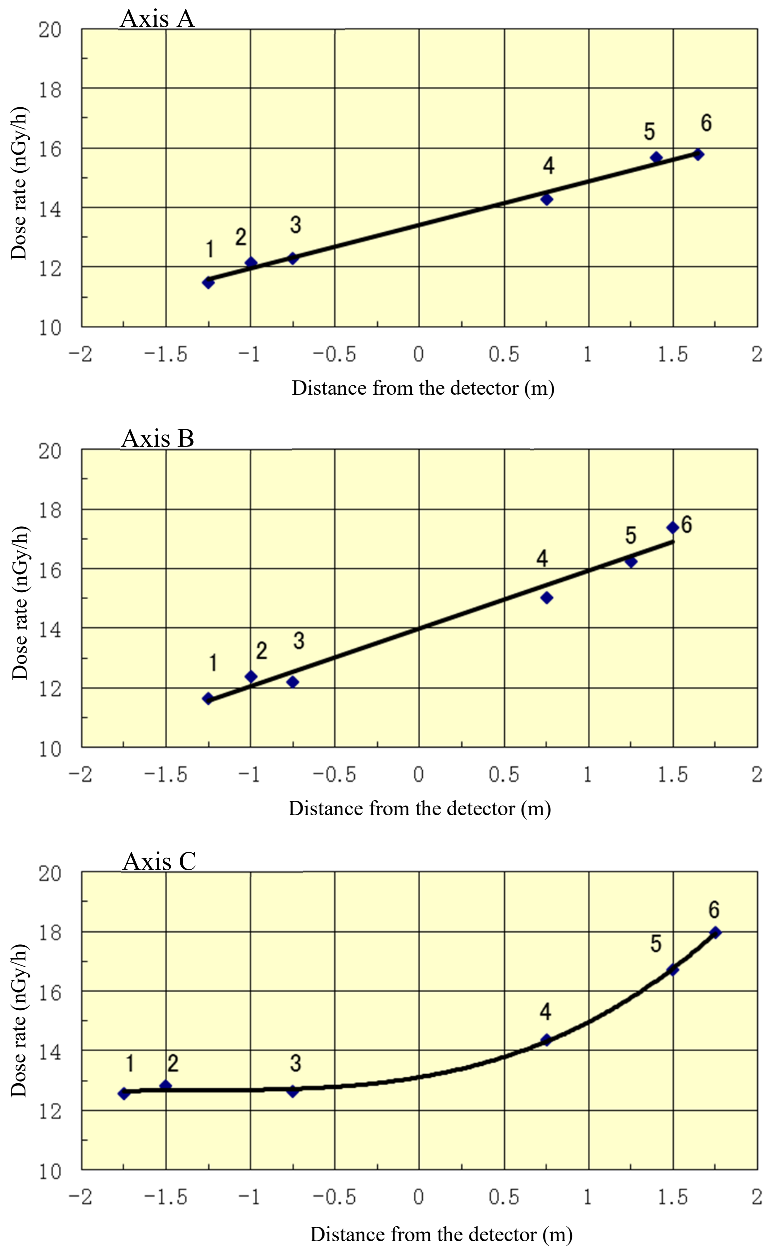


Figure H.2 Measurement results at measurement points shown in Figure H.1

Explanation I Contribution ratio of the dose rate during an accident

Explanation I.1 Overview

The main contributions of the air dose rate are the gamma rays from natural radionuclides present in the Earth's crust in normal times and the natural radionuclides contained in surrounding structures. However, in case of an emergency due to a nuclear accident, the contribution ratio will change. This section explains the contribution ratio that was observed when dose rates increased during the Fukushima Dai-ichi NPS accident.

Explanation I.2 When a radioactive plume passes through

Figure I.1 is a trend graph showing the air dose rates from March 15, 2011, when the Fukushima Dai-ichi NPS accident occurred, to March 31, 2011.

On or around March 15, when the radioactive plume shown in Figure I.1 is believed to have passed through, it can be seen that artificial gamma-ray-emitting nuclides such as ^{133}Xe (approximately 5 days), ^{132}I (approximately 2 hours), ^{131}I (approximately 8 days), ^{132}Te (approximately 3 days), ^{134}Cs (approximately 2 years), and ^{137}Cs (approximately 30 years) are present in the order of their contribution to the dose rate under the meteorological conditions with no rainfall. The periods in parentheses indicate the physical half-lives. Except for ^{134}Cs and ^{137}Cs, most of these nuclides have short half-lives of ten days or less.

Figure I.2 shows the contribution ratio of each nuclide to the dose rate rose on or around March 15, when radioactive plumes released into the atmosphere by pressure vents or hydrogen explosions were supposed to pass through at the Fukushima Dai-ichi NPS of TEPCO.

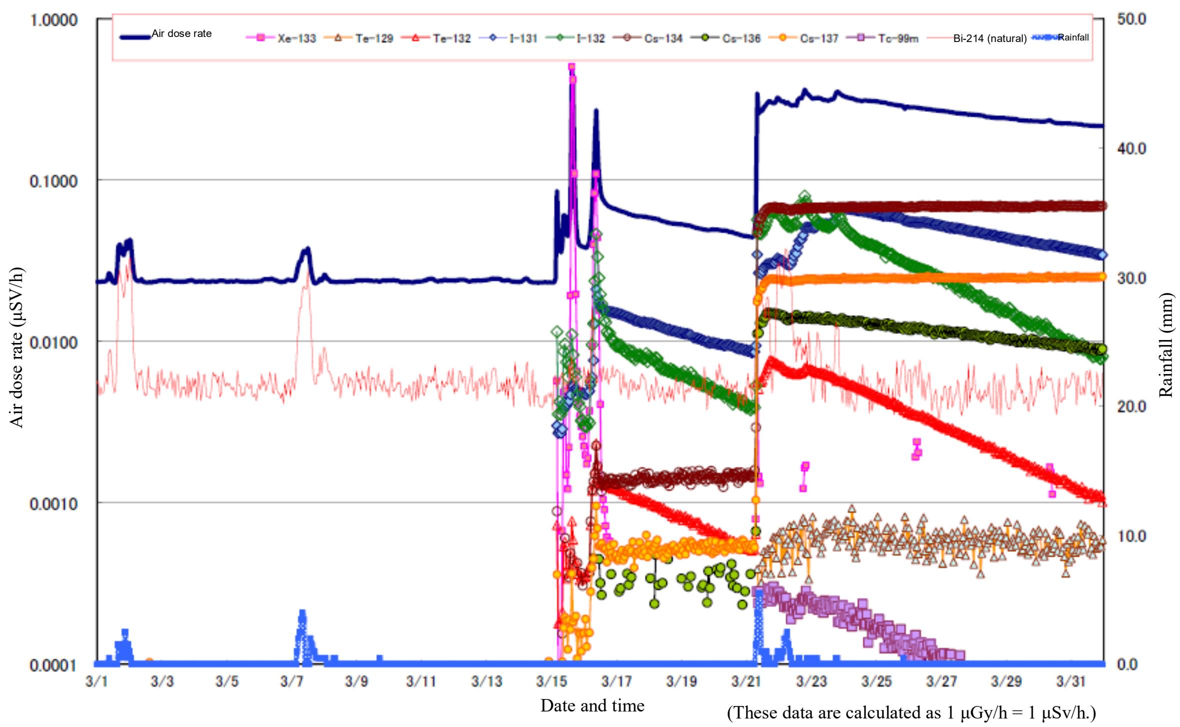


Figure I.1 Air dose rate and contribution dose rate of each nuclide (March 2011) (Reference 21)

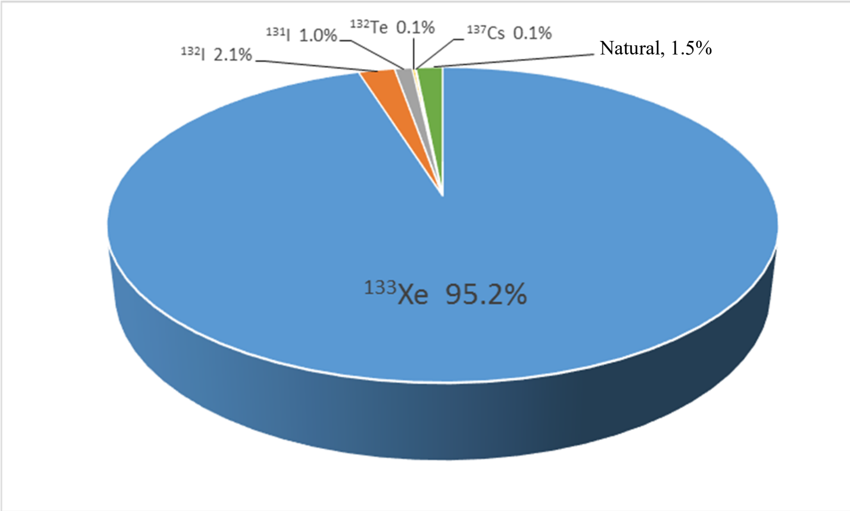


Figure I.2　　Example of dose rate contribution ratio when a radioactive plume passes through

As can be seen from this figure, the dose rate contribution ratio of ^{133}Xe (81 keV) accounts for more than 90% of the total when a radioactive plume passes through. Therefore, in case of an emergency, especially at the early stage of an accident in which a radioactive plume passes through, the energy characteristics in the low-energy region should be acquired in advance, because of the high dose rate to be produced by the low-energy gamma-ray-emitting nuclides.

Explanation I.3　　During (or after) fallout

As explained in Explanation I.2, it is conceivable that a radioactive plume passed through during the early stage of the accident. After that, as shown in Figure I.1, radioactive cesium and others, which had been radioactive nuclides suspended in the atmosphere, were deposited on the ground surface together with raindrops owing to the rainfall of March 21, (that is, the fallout). Figure I.3 shows the contribution ratio of the dose rate when this fallout was generated.

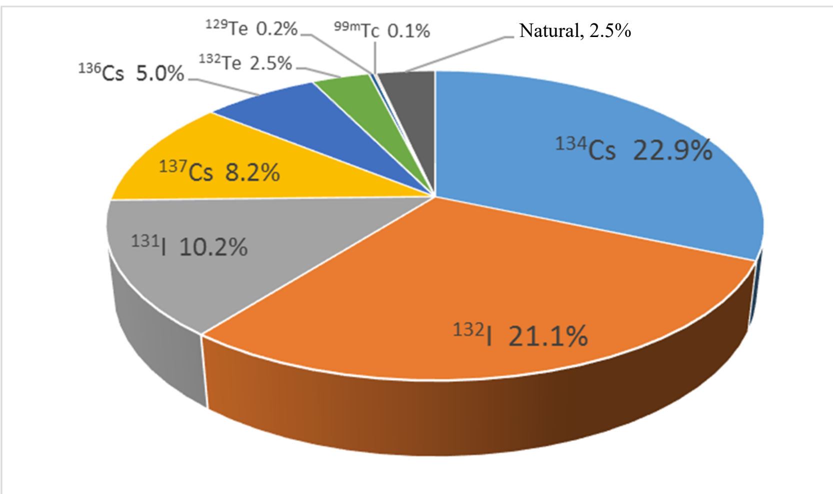


Figure 1.3　　Example of dose rate contribution ratio during fallout

The contribution ratio, after 60 days have elapsed since the fallout is shown in Figure I.4.

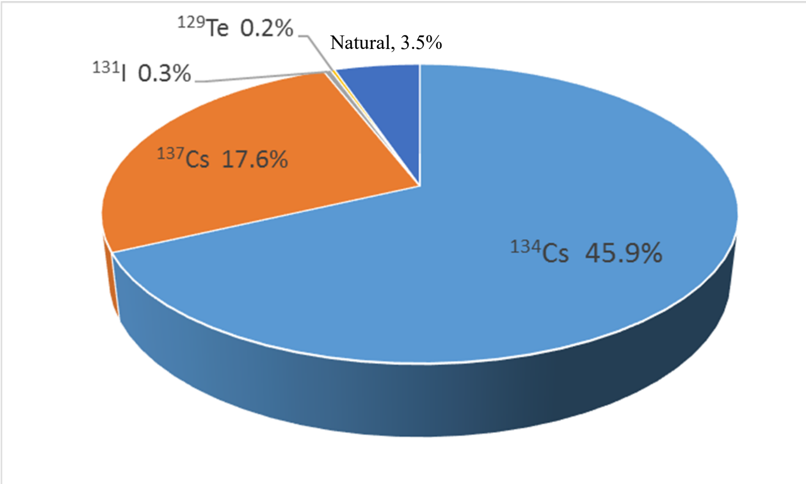


Figure I.4 Example of the dose rate contribution ratio after 60 days have elapsed since fallout

Figure I.3 shows that the contribution ratios of radioactive cesium and radioactive iodine are large, and Figure I.4, showing the condition after 60 days have elapsed since the fallout, tells us that the contribution ratio of radioactive cesium is overwhelmingly large. This is because the release from the power plant has stopped, and the contribution ratio of the radionuclides with relatively longer half-lives has increased, because they have been attenuated by the physical half-life and the weathering effect of the deposited radionuclides.

These results show that, in case of an accident, the situation is divided into three stages: (1) when a radioactive plume passes through, (2) when fallout is generated, and (3) after the release has stopped, and dose rate contribution ratios commensurate with each situation are identified. Therefore, it is extremely important for those who are engaged in continuous monitoring to understand the evaluation of the measured values corresponding to the gamma-ray energy of each contributing nuclide and that the data contain such information.

Explanation J Creation of a response function

Explanation J.1 Meaning of a response function

When an unfolding is performed to observe the output pulse-height distribution from the detector and obtain an energy spectrum of radiation incident on the detector (this method focuses on photons), the shape of the response from the detector to monoenergetic radiation (the response function) should be known. The agreement between the response function and the response of the detector considerably affects the analysis result.

Assuming that the spectrum of the incident photon is expressed as *N*(*E*) (*E* is energy) and the number of pulses observed between the pulse height h and h + ⊿h is P(h) ⊿h, then, given that the response function is *K*(*E*,*h*) and the efficiency of the detector is ε(*E*), the following equation is established:

$$\mathrm{P(h)}\varDelta\mathrm{h} = \varDelta\mathrm{h}\int_{o}^{E}\varepsilon\,(E)K(E,h)N(E)dE \qquad \text{(J.1)}$$

For a monoenergetic photon, *N*(*E*) is regarded as a delta function. In this case, ε(*E*) is a constant, and the above equation changes as follows:

$$\mathrm{P(h)}\varDelta\mathrm{h} = \varDelta\mathrm{h}\int_{-\infty}^{\infty}\varepsilon\,(E)K(E,h)\delta(E-x)dx \qquad \text{(J.2)}$$

$$\mathrm{P(h)} = \varepsilon(E)K(E,h) \qquad \text{(J.3)}$$

A response function is obtained by dividing the pulse-height distribution by ε(*E*).

Further, if ε(*E*)*K*(*E*,*h*) is collectively expressed as *R*(*E*,*h*) in equation (J.1), the pulse-height distribution P(h) relative to the spectrum *N*(*E*) of the incident photon is expressed as follows:

$$\mathrm{P(h)}\varDelta\mathrm{h} = \varDelta\mathrm{h}\int_{0}^{\infty}R\,(E,h)N(E)dE \qquad \text{(J.4)}$$

R(*E*,*h*) is the probability that an incident photon with energy of *E* produces a pulse of height *h* in the detector.

When the pulse-height distribution obtained by measuring a monoenergetic photon is divided by the flux density incident on the detector, a pulse-height distribution when monoenergetic photons of unit flux density are incident on the detector is obtained, and this represents *R*(*E*,*h*); thus, it is easy to understand the meaning of the latter response function.

In equation (J.1), *K*(*E*,*h*) represents a distribution in which the area of the pulse-height distribution is 1 for every different photon energy *E*.

A response function is specific to the radiation source, as well as the configuration and shape of the detector. Table J.1 illustrates some examples of relatively common response functions or response matrices. Apart from those shown in the table, many other response functions have been created. Heath developed and compiled a method of interpolating a response function for any energy based on a response function that is experimentally determined using a specific radiation source and a specific detector configuration (References 22, 23, 24, and 25).

Explanation J.2 Creation of a response function

A response function is specific to the arrangement of the detection system and the source. There are two methods to determine a response function: (1) the experimental method and (2) the Monte Carlo simulation method. The former is the method represented by Heath, as mentioned before. In the latter method, additional conditions such as the equipment in use and its geometry must be considered; however, such procedures are cumbersome and a simulation is often performed under the condition that photons are incident on a bare detector.

(1) Experimental creation method

This method starts with obtaining an output pulse-height distribution of the detector for photons of various monoenergetics. This is advantageous because the contribution to the pulse-height distribution under the conditions specific to the detection system used, such as forward- and backward-scattered rays, can all be included in the response function. If the entire detection system, including the shield, is always kept in the same condition, they can be regarded as one detector. To evaluate a photon spectrum in the natural environment, a group of pulse-height distributions by monoenergetic photons should be measured under the condition of a broad incident beam. Photon radiation sources of monoenergetic photons commonly used to create a response function are listed in Table J.2. It is not necessary to measure all of these sources. In taking measurements, a radiation source that is structured to emit as few scattered rays as possible should be used.

Table J.1 Response functions and response matrices of a NaI (Tl) detector

Shape	Producer or user organization	Number of rows and columns	Conditions (energy range, bin width, target, etc.)	References
(a) 3 in. φ × 3 in. cylindrical	(I) Heath	-	0.335–2.75 MeV; interpolate a response function of any energy	a)
	(II) Zerby–Moran	-	0.679–6 MeV; perform a Monte Carlo calculation for an arbitrary pulse-height distribution	b)
	(III) Berger–Seltzer	-	2 - 20 MeV; perform a Monte Carlo calculation of a pulse-height distribution for each 1 MeV	c)
	(IV) Kyoto Univ., Engineering	20 × 20	0–1.44 MeV; 72 keV/bin, equal division; for transmission spectrum study	d)
	(V) Nagoya Institute of Tech., Lab.	22 × 22	0–2.61 MeV; unequal division by monochromatic energy peak width; for the natural environment	e)
	(VI) Nagoya Institute of Tech.	46 × 46	0–9.2 MeV; 200 keV/bin, equal division; for the environment in the vicinity of the reactor	f)
	(VII) JAERI	-	Response matrixes of arbitrary width can be calculated for gamma rays of up to 10 MeV.; intended for various shapes and sizes of detectors	g)
(b) 3 in. φ spherical	(I) RIKEN	Continuous, with an interval of 10 keV	0–3.0 MeV; 10 keV/bin, equally divided BG (intrinsic background, cosmic ray distribution is subtracted); for the natural environment	h)
	(II) JAERI	60 × 60 295 × 295	0.05–3 MeV; 50 keV or 10 keV/bin, equal division; for the natural environment	i)
	(III) Reactor of Kyoto Univ.	34 × 34	0–0.1 MeV; 20 keV/bin, 0.1–3 MeV 100 keV/bin; for the natural environment	j)
	(IV) Nagoya Institute of Tech., Lab.	22 × 22	0–2.614 MeV; unequal division by monochromatic energy peak width; for the natural environment	k)
	(V) Nagoya Institute of Tech.	22 × 22	0.01–3 MeV; equal division of a logarithmic pulse-height axis; for the natural environment	l)
(c) 2 in. φ × 2 in. cylindrical	(I) Hitachi	14 × 14	0–1.4 MeV; 100 keV/bin, equal division; inverse matrix; for the study of scattered rays	m)
	(II) JAERI	60 × 60	0.05–3 MeV; 50 keV/bin, equal division	n)

References

a): Reference 22 R.L. Heath; IDO-16880-1 (1964)
b): Reference 26 C.D. Zerby and H.S. Moran; ORNL-3169 (1961)
c): Reference 27 M.J. Berger and S.M. Seltzer; Nuclear Instr. Meth., 104, 317–332 (1972)
d): Reference 28 T. Hyodo and F. Makino; Memoi. Fac. Engin., Kyoto Univ., 14, 291 (1962)
e): Reference 29 湊進; 名工試報告, 27, 384(1978)
f): Reference 30 Y. Nakashima; Thesis. Fac. Engin., Nagoya Univ., (1980)
g): Reference 31 森内茂,堤正博,斎藤公明; 保健物理, 42, 71–83, (2007)
h): Reference 32 M. Okano; Natural Radiation Environment III, Symposium Series DOF51 (CONF-780422) 867 (1980)
i): Reference 33 S. Moriuchi; Personal Communication
j): Reference 34 I. Urabe, T.T. Sujimoto, K. Yamazaki and K. Katsurayama; Jour. Rad.

Res., 19, 163 (1978)
k): Reference 35 S. Minato; Personal Communication
l): Reference 36 明野吉成: 卒業論文; 名大工, 原子核工学科, Mar. (1982)
m): Reference 37 石松健二; 日本原子力学会誌, 4, 24 (1962)
n): Reference 38 S. Moriuchi; Personal Communication

Table J.2 Nuclides and gamma-ray energies that can be used to create a response function (the circles indicate commonly used nuclides)

Nuclide	Energy (MeV)	Nuclide	Energy (MeV)
○ ^{241}Am	0.060	○ ^{85}Sr	0.513
^{129m}Te	0.109	^{91m}Y	0.555
○ ^{57}Co	0.122	○ ^{137}Cs	0.662
^{47}Sc	0.155	^{95}Nb	0.764
^{139}Ce	0.166	○ ^{54}Mn	0.835
^{97}Ru	0.213	^{92}Nb	0.930
^{135}Xe	0.250	^{86}Rb	1.080
○ ^{203}Hg	0.279	○ ^{65}Zn	1.114
○ ^{51}Cr	0.320	^{41}Ar	1.290
^{115m}In	0.335	^{42}K	1.510
^{195}Au	0.411	^{26}Al	1.780
^{7}Be	0.478	^{24}Na	1.369, 2.754

The resulting group of pulse-height distributions for monoenergetic photons is normalized relative to the number of incident photons, taking into account the efficiency of the detector. That is, the group of pulse-height distributions obtained by measurement is normalized such that every area becomes 1, and then the efficiency $\varepsilon\ (\mathrm{E}) = 1 - \exp\left\{-\mu(E)\bar{\ell}\right\}$ is applied to each pulse-height distribution, considering the energy of each incident photon. Here, μ(E) is the linear attenuation coefficient of the detected body for the photon of energy E, and $\bar{\ell}$ is the average passing distance of photons in the detected body. The response function for an incident photon of arbitrary energy can be obtained by graphical interpolation, using the group of pulse-height distributions for monoenergetic photons that was previously obtained.

Heath developed a graphical interpolation method and created a computer program to perform this; its principle is as follows. When a group of pulse-height distributions for monoenergetic photons measured under the same condition is arranged in three dimensions and covered with a smooth curved surface, a response curved surface is formed.

The total energy absorption peak of the response curved surface thus obtained is interpolated by the following Gaussian distribution.

$$y = y_0 \exp\left\{-\frac{(X - X_0)}{b_0}\right\} \qquad \text{(J.5)}$$

In this equation, y_0 is the peak height of the Gaussian distribution and b_0 is an amount related to the half-width, which can be determined experimentally as a function of the energy $E\gamma$ of the incident photon. Next, with respect to the pulse-height distribution due to the Compton effect, the energy of the Compton edge, that is, the maximum energy generated by the Compton effect, is obtained from the following equation.

$$E_c = E\gamma - E\gamma \,/\, \left\{1 + \frac{2E\gamma}{m_0 C^2}\right\} \quad (J.6)$$

In addition, the energies of the peak E_b due to backscattering, the single escape peak E_{PS} of the annihilation photon due to the reaction of pair production, and the double escape E_{PD} are determined as follows:

$$E_b = E\gamma / \left\{1 + \frac{2E\gamma}{m_0 C^2}\right\} \quad (J.7)$$

$$E_{PS} = E\gamma - 0.511(\text{MeV}) \quad (J.8)$$

$$E_{PD} = E\gamma - 1.022(\text{MeV}) \quad (J.9)$$

The pulse-height distribution is divided into five intervals of L_1, L_2, ..., L_5 according to the positions of E_C, E_b, E_{PS}, and E_{PD} (see Figure J.1). Naturally, in the case of $E\gamma < 1.02$ MeV, it can be divided into three intervals of L_1, L_2, and L_3 (see Figure J.2). The pulse-height distribution is expressed by the following polynomial in each interval.

$$\text{g}\,(X) = a + bX + \textstyle\sum_{K=1}^{N} b_k \sin\frac{k\pi X}{M-1} \quad (J.10)$$

That is, the pulse-height distribution $g(X)$ of the interval L_i can be expressed as above. X is the peak value of the pulse. In this equation, M is the number of data points in the interval, and N is the number of terms in the development, which is usually approximately $M/2$. The entire pulse-height distribution of the measured monoenergetic photons is fitted using the above equation. The coefficients a, b, and b_K are interpolated with respect to $E\gamma$ for each division interval L_i. By using the obtained values of a, b, and b_K and the previously obtained b_0 and y_0, the pulse-height distribution of the incident photons of arbitrary Eγ relative to the detector can be calculated by equations (J.4) and (J.10).
To display these groups of pulse-height distributions in a matrix, i, j, and the element R_{ij} of the response matrix relative to any given energy E_j can be determined by the following equation:

$$R_{ij} = \textstyle\int_{x_{j-1}}^{x_j} R\,(X, E_i)dx \quad (J.11)$$

A response matrix should be created to have an appropriate number of rows and columns, according to the shape of the photon spectrum, the analysis method to be applied, the data processing capability, etc. When analyzing a continuous spectrum, it is not necessary to have many rows and columns, whereas, when a spectrum consisting of a large number of monoenergetic photon groups is expected, many rows and columns are needed.

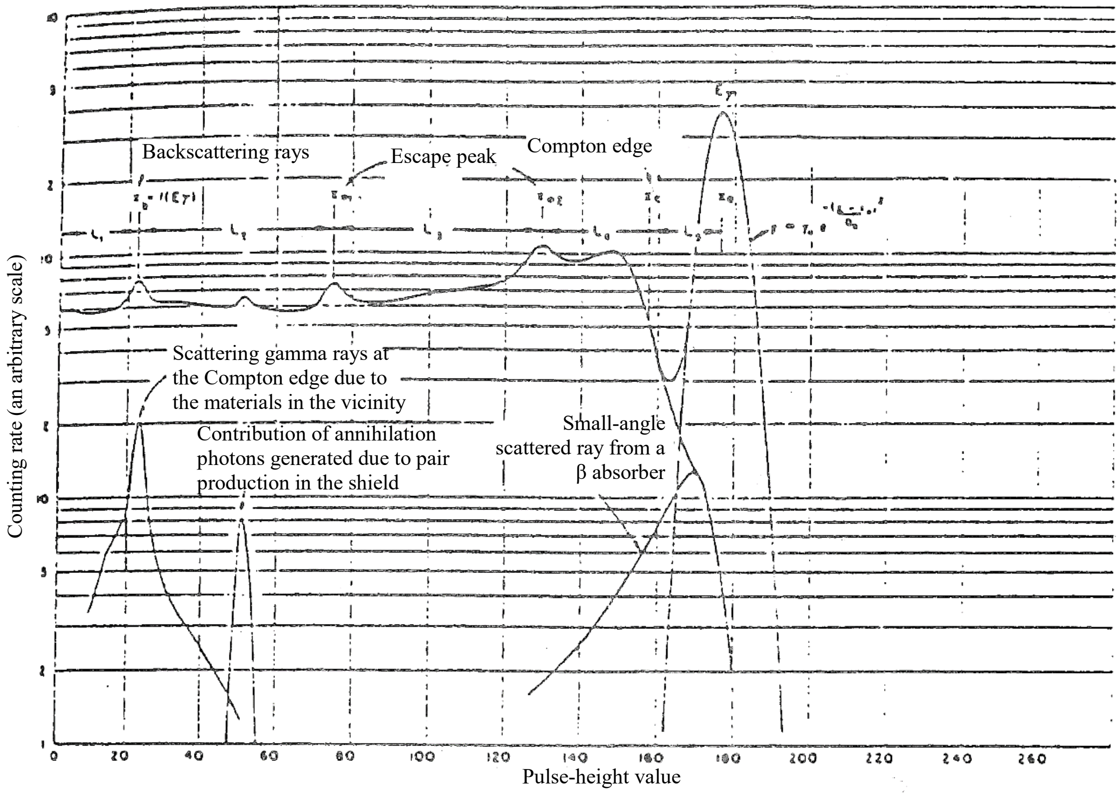


Figure J.1 Response of a 3 in. × 3 in. NaI (Tl) detector to 1.78-MeV gamma rays

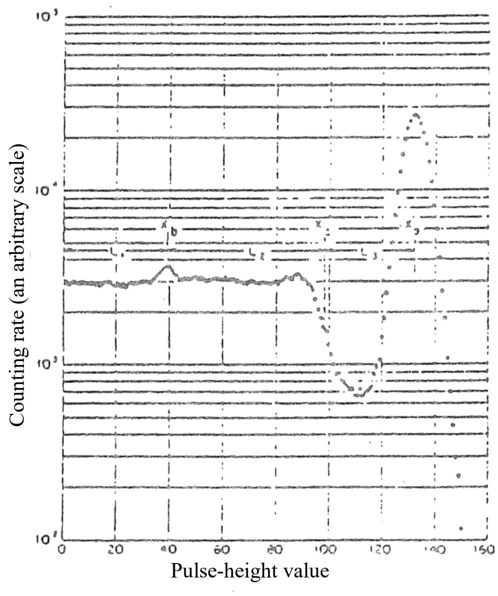


Figure J.2 Response of a 3 in. × 3 in. NaI (Tl) detector (in the case of below 1.02 MeV)

(2) Monte Carlo simulation (References 39 and 40)

When a photon enters the detector, a secondary electron is generated through interaction with the detected body. The secondary electron expends its energy in the detected body. The main interaction between the photon and the detected body is caused by the action of photoelectric effect, Compton effect, and electron pair generation. The Monte Carlo simulation uses random numbers to track the history of an incident photon in the detected body. A secondary electron is generated every time a photon collides, and the energy of the secondary electron is lost through ionization and excitation. However, when the incident photon has an energy of several MeV or less, the range of the secondary electron generated in the detected body is short; thus, all the energy of the secondary electron is treated as absorbed by the detected body. However, when the photon energy is high, secondary electron tracking is required. The scattered photons are tracked again until they lose all their energy in the detected body, fall below a certain cutoff energy, or escape from the detected body.

When the energy of the generated secondary electrons is large, the Monte Carlo simulation program becomes more complex, including two subprograms, one that track photons and one that tracks secondary electrons.

Figure J.3 shows an outline of a simple simulation (without electron tracking) for determining a response function using the Monte Carlo simulation.

First, the following is given as input data: the shape of the detector (radius and length), density, sections of energy, number of rows and columns of the response matrix, resolution of the detector, value of the attenuation absorption coefficient owing to each interaction, constant associated with electron tracking (if required), number of histories, and so on. Because each process is probabilistic, random numbers (a uniform random number that is usually (“0”,“1”)) are used to track each process. As the initial condition for a photon, determine the position and direction of incidence relative to the detector. Next, determine the distance that the photon travels through the detected body and the coordinates of the interaction point before interaction occurs. If the interaction point is outside the detected body, return to the beginning and start tracking the next photon. If the interaction point is inside the detected body, determine whether interaction is a photoelectric effect, a Compton effect, or an electron pair generation. When a photoelectric effect occurs as the interaction, determine the section of energy for recording the amount of the total energy of the photon, which is assumed as absorbed by the detected body, and proceed to track the next photon. When a Compton effect occurs, determine the energy and scattering direction of the scattered photon. In this case, the difference between the energy of the original photon and the energy of the scattered photon is specified as the absorbed energy. The scattered photon is further tracked in the scintillator as before, to determine the travel distance and the coordinates of the next interaction point, until they fall below the cutoff energy or escape from the detector.

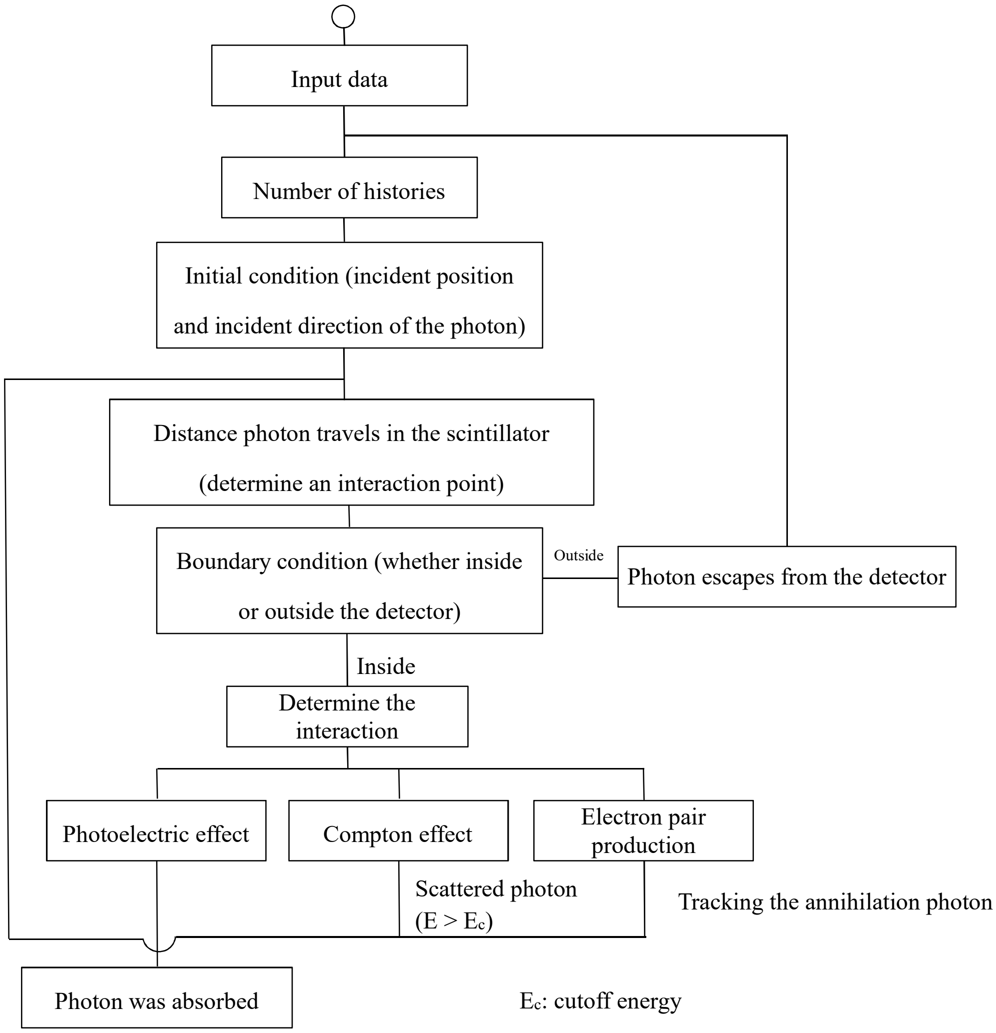


Figure J.3 Flowchart of creating a response function by the Monte Carlo simulation method (simplified)

Now, add the energy applied to the secondary electron before it reaches the cutoff energy or the energy applied to the secondary electron before it escapes from the detected body, and record the sum in the corresponding energy section. When an electron pair is generated, the difference between the energy of the original photon and 1.022 MeV is taken as the absorbed energy. Assuming that two photons of 0.511 MeV are generated in the directions separated by 180° at the interaction point, track each photon separately, add the energy absorbed in the detected body, and record the sum in the corresponding energy section.

Determine the energy absorbed by the detected body for each photon in the above process, and specify a bin in the energy section for recording, which was initially given as the input data by the absorbed energy, considering the energy resolution of the detected body. A response function is determined by repeating the above process for the number of histories, for each of the energy sections initially given.

Explanation J. 3 Problems in creating a response function

Unlike the case in which a response function is experimentally created by fixing the detector and the radiation source, many contributions of radiation from various directions are present when measuring environmental radiation; thus, many matters should be considered to create a response function.

In the Monte Carlo simulation, it is a greatly burdensome work to create a program that includes all of the following: (1) absorption and scattering by the reflection from the aluminum case of the detector and the MgO contained therein, (2) scattering by the scintillator window and the light guide, (3) scattering by the photoelectric surface of the photomultiplier tube, (4) escape of K X-rays from iodine, (5) directional dependence, (6) pulse-height resolution, (7) efficiency of the detector, (8) peak-to-Compton ratio, and so on. In addition to these factors, the nonlinearity between the amount of light emission and the secondary electron energy should also be considered. This is summarized as follows:

The contribution mentioned in (1) is due to the attenuation of the incident primary photon flux and the buildup by the scattered ray. That is, in the case of a monoenergetic photon, the primary photon flux decays according to $\exp\{-\mu'(E)\bar{d}\}$, but some fraction of it becomes incident on the detected body as a scattered ray. Here, $\mu'(E)$ is the linear attenuation coefficient of the absorber (the case, etc.), and $\bar{d}$ is the average traveling distance of photons in the absorber. Therefore, to evaluate an accurate energy spectrum, not only the attenuation of the primary photon flux but also the contribution of such scattered rays must be included in the response function. (2) denotes the effect of a glass of thickness several millimeters or similar, which is present at the exit of the emitted light in the detected body. When a primary photon is incident from the front of the detected body, the fraction of photons transmitted through the detected body, which is backscattered at the front and absorbed again by the detected body, should be considered. When a monoenergetic photon is measured, the peak observed around the range of 180 keV is contributed by such scattered rays. The contribution mentioned in (3) is due to the component in which the photon transmitted through the body as described in (2) is further backscattered on the photoelectric surface of the photomultiplier tube and reabsorbed by the detected body. The contribution of this component is considered to be smaller than that of (2). The contribution mentioned in (4) is related to the structure of the detected body, and it is important for low-energy photons when an interaction occurs on the surface of the detected body. In the case of (6) and (8), not only the detected body, but the characteristics of the whole system including the measurement system, are involved.

For accurately evaluating the contributions of (1) through (5), there is no other way than either incorporating it into a Monte Carlo simulation program or correcting it with an experimental analytical method. In the former case, because of the structure of the detected body, a simplified boundary condition is inevitably established owing to partially overlapped aluminum plates and nonuniformity of the reflective material. In the latter case, a single low-energy source is required because the lower the energy of the photons, the more likely that these contributions are not negligible. Because the case is generally thin, it is sufficient to consider the contribution of the scattered rays by the case only once. Directional dependence is related to the structure of the case and is particularly remarkable in the low-energy region. In summary, a form R_m close to the actual system can be obtained by multiplying the response matrix R created for the bare detected body by the matrix Ms representing the contribution of the absorptive scattering and the matrix D for correcting the directional dependence to modify the response matrix.

$$R_m = \mathrm{D} \times \mathrm{Ms} \times \mathrm{R} \tag{J.12}$$

However, the errors in the result are only a few percent, even if corrections are not made for (1) through (5).

Corrections for (6) through (8) are important for adapting the response function (matrix) created in the bare detected body to the actual system, or for adapting the response function already published to the employed detection system. To correct the resolution, it is sufficient to adapt only a portion of the total absorption peak to the resolution of the system to be used, considering the peak-to-Compton ratio, because the distribution due to the Compton effect is continuous. Because the response function (matrix) thus corrected, whose area (element) is statistically calculated for each incident photon energy, should produce the total efficiency, the details can be remodified in comparison with the total efficiency $\varepsilon\,(\mathrm{E}) = 1 - \exp\left\{-\mu(E)\overline{\ell}\right\}$, which is theoretically obtained using the attenuation coefficient of the detected body.

Regarding a bare 3 in. φ NaI (Tl) detector, an example of an unevenly divided response function (matrix) composed of 22 rows and 22 columns, corrected by calculating the effect of the MgO reflector and the Al case by the plate approximation for the response function (matrix) obtained by a Monte Carlo simulation, is illustrated in Table J.3. The number of histories for each energy is 50000. Additionally, if the numerical value of each element is divided by 1000, a response to a monoenergetic single photon incidence can be obtained.

Table J.3 Example of a 22 × 22 response matrix of a 3 in. φ NaI (Tl) scintillator for a uniformly incident gamma-ray field

E \ E	0.1	0.2	0.3	0.4	0.5	0.6	0.7	0.8	0.9	1.0	1.1	1.2	1.32	1.465	1.615	1.765	1.970	2.205	2.410	2.615	2.860	3.100
0.1	979.7	0.0																				
0.2	27.0	916.6	0.7																			
0.3	71.7	17.3	759.6	1.6	0.0																	
0.4	80.8	66.4	15.6	593.6	10.1	0.0																
0.5	92.1	73.7	58.1	17.8	473.4	13.8	0.0															
0.6	83.3	73.0	66.7	51.5	20.6	381.8	17.8	0.0														
0.7	73.5	66.4	60.7	59.2	45.8	22.6	318.8	17.8	0.0													
0.8	63.4	60.5	55.7	55.0	56.6	40.6	23.0	270.0	21.5	0.0												
0.9	54.6	52.4	49.1	48.2	47.8	52.6	34.6	26.5	234.8	23.2	0.0											
1.0	46.0	44.7	44.3	42.5	41.0	44.3	49.3	32.7	27.0	210.3	25.0											
1.1	42.5	40.3	37.3	38.6	37.7	38.6	39.9	48.9	31.1	27.4	188.1	24.6	0.0									
1.2	36.8	34.0	33.8	34.0	34.2	34.4	34.9	38.2	48.0	28.7	28.5	168.2	26.1	0.0								
1.32	31.8	30.4	29.2	28.3	29.4	29.2	28.5	30.2	35.3	43.0	35.1	17.3	180.9	11.8	0.0							
1.465	26.8	27.8	25.0	23.9	24.8	24.3	24.3	25.2	26.5	30.3	38.4	41.9	24.3	169.7	10.5	0.0						
1.615	22.4	21.3	21.9	21.3	21.2	24.6	20.6	22.6	22.1	23.5	29.4	33.5	58.1	27.0	152.4	11.8	0.0					
1.765	18.4	18.4	18.4	18.4	18.4	18.4	21.3	21.7	18.6	19.7	20.0	25.7	44.7	57.7	29.8	143.6	12.7	0.0				
1.970	15.1	15.1	15.1	15.1	15.1	15.1	15.1	15.1	18.2	19.2	16.4	17.5	26.3	43.9	53.7	34.0	157.2	1.5	0.0			
2.205	11.8	11.8	11.8	11.8	11.8	11.8	11.8	11.8	11.8	11.8	14.9	21.7	22.1	24.6	33.5	47.8	72.4	140.0	5.3	0.0		
2.410	9.9	9.9	9.9	9.9	9.9	9.9	9.9	9.9	9.9	9.9	9.9	9.9	21.7	27.0	21.1	26.8	84.6	50.0	126.5	8.6	0.0	
2.615	8.3	8.3	8.3	8.3	8.3	8.3	8.3	8.3	8.3	8.3	8.3	8.1	11.8	15.6	31.6	21.5	50.4	70.3	46.7	122.4	7.0	0.0
2.860	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.0	7.2	9.2	11.8	16.0	23.0	45.8	37.7	68.0	53.3	131.6	2.6
3.100	5.9	5.9	5.9	5.9	5.9	5.9	5.9	5.9	5.9	5.9	5.9	5.9	7.2	8.8	9.2	9.4	35.3	20.9	32.5	70.4	73.0	105.0

Appendix

Appendix 1 Calculation Examples of a Response Function

The components of a response function consist of a Compton distribution component generated from the primary incident photon, a KX-ray escape component, an annihilation gamma-ray spectrum component, and the total absorption component. However, the detector actually relays a shape specific to each probe, because of the influence of the NaI (Tl) crystal, as well as scattered gamma-rays generated from the case that encloses it, and from the photomultiplier tube. Although it is ideal to obtain a response function in a strict sense by means of a standard gamma-ray-based experiment in which absorption and scattering components are not included, it is impossible to obtain such a source in practice. Even if this was possible, high accuracy cannot be expected under normal experimental conditions. However, in considering a determination method that does not include such an error factor, it is possible to employ a theoretical calculation method based on the actual shape of the detector, anticipating that the determination of a response function with high accuracy would be feasible. In this case, given that it is assumed that the spectral analysis described here should be performed by means of a simple calculation code without impairing accuracy, the shape of the detector is modeled so that any response functions can be calculated by a semi-experimental method.

Based on the assumption that the spectral distribution considered here is supposed to consist of a Compton component, a KX-ray escape component, an annihilation gamma-ray spectral component, and a total absorption component, it is determined by calculating each component first and then synthesizing a result. Normalization is performed using the peak efficiency, which is obtained based on an experiment on the total absorption component and other components, as well as the overall efficiency.

The approximation is outlined as follows:

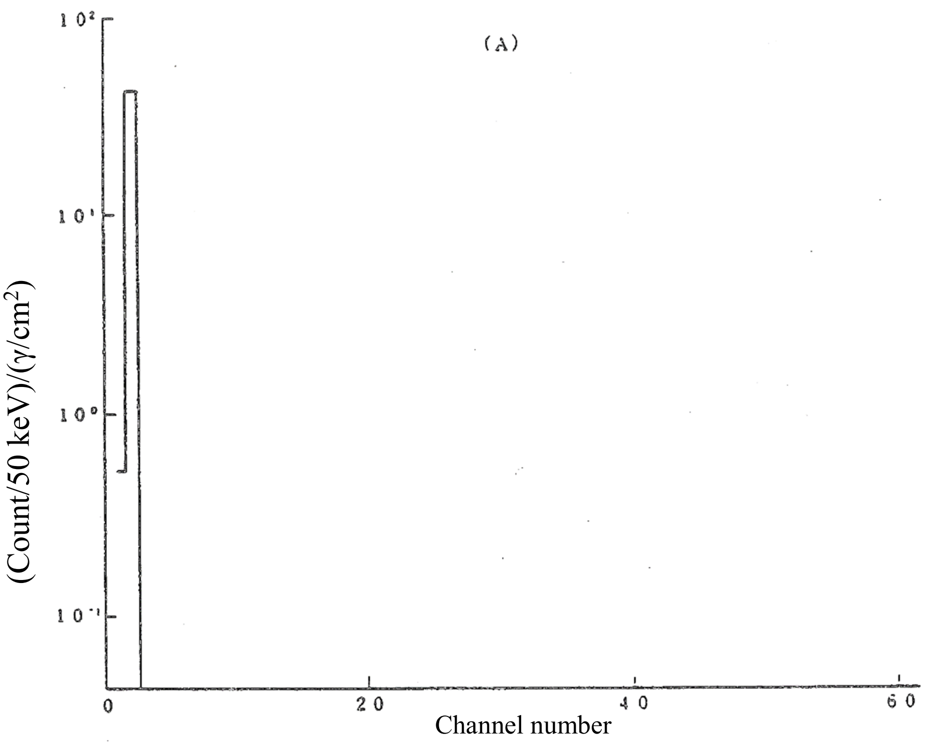
[1] Compton spectral distribution
Synthesis by means of a rectangular distribution and an exponential function.

[2] Spectral distribution produced by the KX-ray escape of iodine
Spectrum synthesis involving an escape using the escape rate of the KX-ray of iodine.

[3] Spectral distribution by means of annihilation gamma-rays
This is synthesized by calculating a probability based on the response function of 0.51 MeV.

[4] Total absorption component
This is synthesized as a δ function, including the total absorption component produced by annihilation gamma-rays.

A complete response function is obtained by superimposing [1] to [4]. Figures 1.1 and 1.2, and Table 1.1 show examples of the response functions that are actually calculated from a 3 in. φ spherical NaI (Tl) detector.



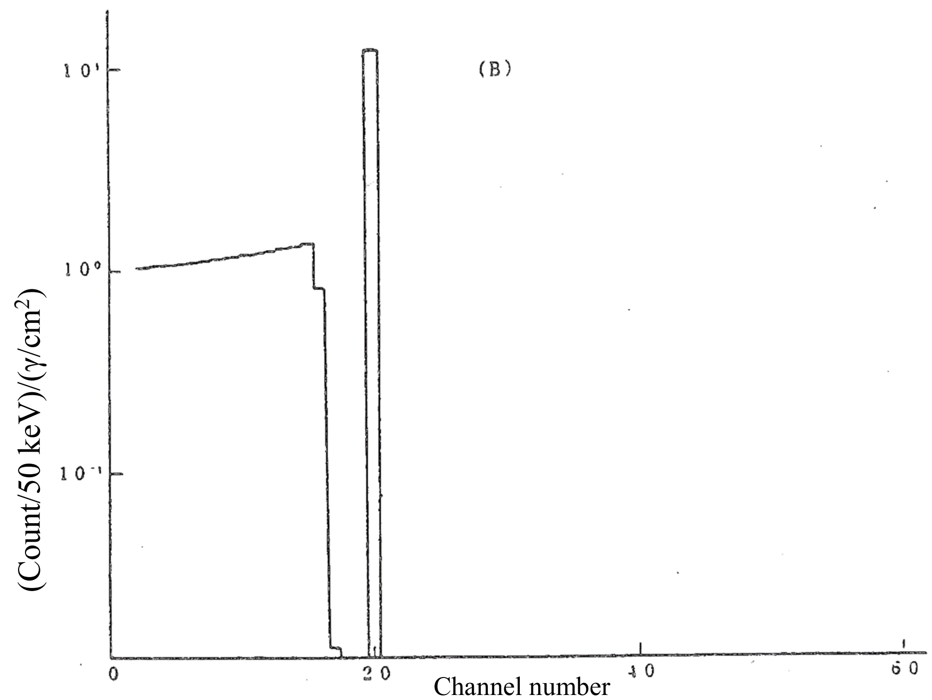


Figure 1.1 Calculation examples of a response function

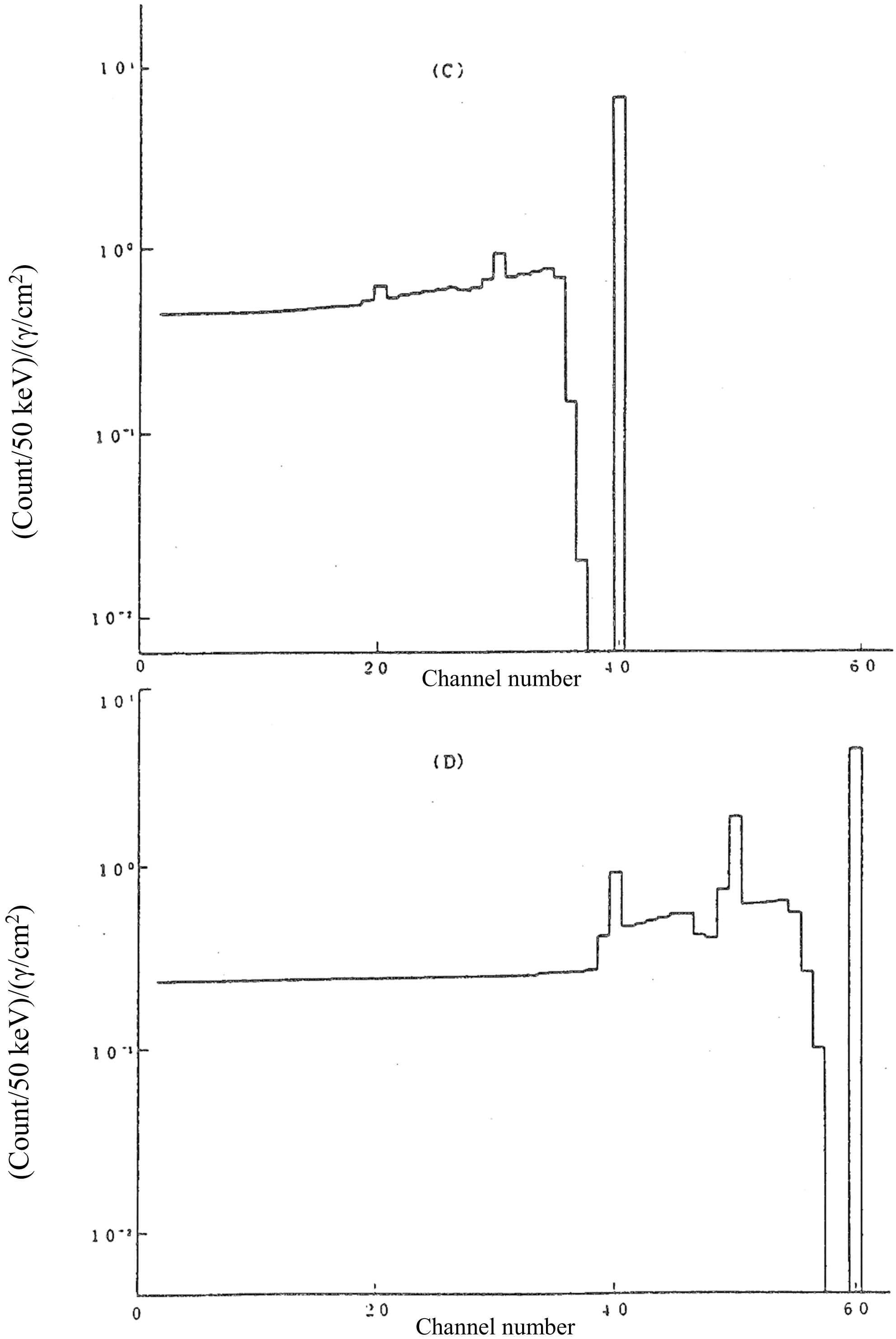


Figure 1.2 Calculation examples of a response function

Table 1.1 Examples of Response Functions Generated by Synthesis

(for a 3 in. φ spherical scintillator, 50 keV/bin)

ENERGY (60) 2950 KEV - 3000 KEV

	1	2	3	4	5	6	7	8	9	10
0	2.3928E-01	2.3936E-01	2.3946E-01	2.3956E-01	2.3967E-01	2.3980E-01	2.3994E-01	2.4010E-01	2.4027E-01	2.4046E-01
10	2.4067E-01	2.4091E-01	2.4117E-01	2.4146E-01	2.4178E-01	2.4213E-01	2.4252E-01	2.4296E-01	2.4344E-01	2.4397E-01
20	2.4456E-01	2.4522E-01	2.4594E-01	2.4675E-01	2.4764E-01	2.4863E-01	2.4972E-01	2.5093E-01	2.5227E-01	2.5376E-01
30	2.5541E-01	2.5723E-01	2.5925E-01	2.6149E-01	2.6398E-01	2.6673E-01	2.6977E-01	2.7315E-01	5.6835E-01	8.5604E-01
40	4.8389E-01	5.0346E-01	5.2408E-01	5.4586E-01	5.6894E-01	5.4610E-01	4.1633E-01	4.0640E-01	1.1145E+00	1.6290E+00
50	6.4348E-01	6.4679E-01	6.5194E-01	6.6651E-01	5.7949E-01	2.6361E-01	6.2622E-02	0.0	0.0	4.6382E+00

ENERGY (59) 2900 KEV - 2950 KEV

2.4556E-01	2.4566E-01	2.4576E-01	2.4588E-01	2.4601E-01	2.4615E-01	2.4631E-01	2.4649E-01	2.4669E-01	2.4690E-01
2.4715E-01	2.4741E-01	2.4771E-01	2.4804E-01	2.4840E-01	2.4880E-01	2.4925E-01	2.4974E-01	2.5029E-01	2.5089E-01
2.5156E-01	2.5231E-01	2.5313E-01	2.5404E-01	2.5505E-01	2.5617E-01	2.5742E-01	2.5879E-01	2.6032E-01	2.6200E-01
2.6387E-01	2.6595E-01	2.6824E-01	2.7079E-01	2.7361E-01	2.7673E-01	2.8019E-01	5.6549E-01	8.4406E-01	4.8456E-01
5.0377E-01	5.2402E-01	5.4544E-01	5.6815E-01	5.4655E-01	4.2163E-01	4.1258E-01	1.0970E+00	1.5956E+00	6.4372E-01
6.4774E-01	6.5364E-01	6.6874E-01	5.8070E-01	2.5560E-01	6.0546E-02	0.0	0.0	4.6974E+00	

ENERGY (58) 2850 KEV - 2900 KEV

2.5259E-01	2.5270E-01	2.5282E-01	2.5295E-01	2.5310E-01	2.5326E-01	2.5345E-01	2.5365E-01	2.5387E-01	2.5412E-01
2.5439E-01	2.5470E-01	2.5503E-01	2.5541E-01	2.5582E-01	2.5628E-01	2.5678E-01	2.5735E-01	2.5797E-01	2.5866E-01
2.5942E-01	2.6027E-01	2.6121E-01	2.6224E-01	2.6340E-01	2.6467E-01	2.6608E-01	2.6765E-01	2.6939E-01	2.7131E-01
2.7344E-01	2.7580E-01	2.7841E-01	2.8131E-01	2.8451E-01	2.8807E-01	5.6323E-01	8.3245E-01	4.8595E-01	5.0479E-01
5.2469E-01	5.4575E-01	5.6809E-01	5.4779E-01	4.2786E-01	4.1975E-01	1.0800E+00	1.5624E+00	6.4487E-01	6.4967E-01
6.5638E-01	6.7206E-01	5.8146E-01	2.4791E-01	5.8417E-02	0.0	0.0	4.7589E+00		

ENERGY (57) 2800 KEV - 2850 KEV

2.5371E-01	2.5383E-01	2.5396E-01	2.5411E-01	2.5427E-01	2.5446E-01	2.5466E-01	2.5488E-01	2.5513E-01	2.5541E-01
2.5571E-01	2.5605E-01	2.5642E-01	2.5684E-01	2.5730E-01	2.5781E-01	2.5837E-01	2.5899E-01	2.5969E-01	2.6045E-01
2.6130E-01	2.6224E-01	2.6329E-01	2.6444E-01	2.6572E-01	2.6714E-01	2.6871E-01	2.7045E-01	2.7238E-01	2.7452E-01
2.7688E-01	2.7951E-01	2.8241E-01	2.8563E-01	2.8920E-01	5.5389E-01	8.1342E-01	4.8010E-01	4.9846E-01	5.1785E-01
5.3838E-01	5.6017E-01	5.4100E-01	4.2599E-01	4.1858E-01	1.0537E+00	1.5192E+00	6.3668E-01	6.4191E-01	6.4905E-01
6.6489E-01	5.7311E-01	2.3927E-01	5.6235E-02	0.0	0.0	4.8330E+00			

ENERGY (56) 2750 KEV - 2800 KEV

2.6571E-01	2.6585E-01	2.6601E-01	2.6618E-01	2.6637E-01	2.6658E-01	2.6682E-01	2.6708E-01	2.6736E-01	2.6768E-01
2.6804E-01	2.6843E-01	2.6886E-01	2.6934E-01	2.6987E-01	2.7046E-01	2.7112E-01	2.7184E-01	2.7264E-01	2.7353E-01
2.7451E-01	2.7560E-01	2.7681E-01	2.7815E-01	2.7964E-01	2.8128E-01	2.8310E-01	2.8512E-01	2.8735E-01	2.8983E-01
2.9257E-01	2.9561E-01	2.9898E-01	3.0271E-01	5.5685E-01	8.0665E-01	4.8704E-01	5.0512E-01	5.2424E-01	5.4451E-01
5.6606E-01	5.4839E-01	4.3878E-01	4.3253E-01	1.0425E+00	1.4912E+00	6.4499E-01	6.5131E-01	6.5959E-01	6.7638E-01
5.7974E-01	2.3156E-01	5.4001E-02	0.0	0.0	4.8944E+00				

ENERGY (55) 2700 KEV - 2750 KEV

2.7337E-01	2.7353E-01	2.7370E-01	2.7390E-01	2.7412E-01	2.7436E-01	2.7463E-01	2.7492E-01	2.7525E-01	2.7561E-01
2.7601E-01	2.7646E-01	2.7695E-01	2.7750E-01	2.7811E-01	2.7878E-01	2.7952E-01	2.8035E-01	2.8126E-01	2.8227E-01
2.8339E-01	2.8463E-01	2.8601E-01	2.8753E-01	2.8922E-01	2.9109E-01	2.9316E-01	2.9546E-01	2.9801E-01	3.0082E-01
3.0395E-01	3.0741E-01	3.1124E-01	5.5451E-01	7.9427E-01	4.8861E-01	5.0630E-01	5.2504E-01	5.4492E-01	5.6608E-01
5.4981E-01	4.4556E-01	4.4030E-01	1.0243E+00	1.4555E+00	6.4633E-01	6.5348E-01	6.6263E-01	6.8005E-01	5.8081E-01
2.2261E-01	5.1715E-02	0.0	0.0	4.9563E+00					

ENERGY (54) 2650 KEV - 2700 KEV

2.8026E-01	2.8044E-01	2.8064E-01	2.8086E-01	2.8111E-01	2.8138E-01	2.8169E-01	2.8202E-01	2.8239E-01	2.8281E-01
2.8326E-01	2.8377E-01	2.8433E-01	2.8495E-01	2.8563E-01	2.8640E-01	2.8724E-01	2.8817E-01	2.8921E-01	2.9036E-01
2.9163E-01	2.9303E-01	2.9459E-01	2.9632E-01	2.9824E-01	3.0036E-01	3.0271E-01	3.0532E-01	3.0820E-01	3.1140E-01
3.1494E-01	3.1886E-01	5.5101E-01	7.8048E-01	4.8905E-01	5.0634E-01	5.2465E-01	5.4412E-01	5.6485E-01	5.4996E-01
4.5116E-01	4.4688E-01	1.0043E+00	1.4176E+00	6.4618E-01	6.5412E-01	6.6409E-01	6.8208E-01	5.8144E-01	2.1384E-01
4.9378E-02	0.0	0.0	5.0434E+00						

ENERGY (53) 2600 KEV - 2650 KEV

2.9194E-01	2.9215E-01	2.9238E-01	2.9264E-01	2.9292E-01	2.9324E-01	2.9359E-01	2.9397E-01	2.9440E-01	2.9487E-01
2.9540E-01	2.9598E-01	2.9663E-01	2.9734E-01	2.9813E-01	2.9901E-01	2.9998E-01	3.0106E-01	3.0225E-01	3.0358E-01
3.0504E-01	3.0666E-01	3.0846E-01	3.1045E-01	3.1266E-01	3.1511E-01	3.1782E-01	3.2082E-01	3.2414E-01	3.2783E-01
3.3191E-01	5.5277E-01	7.7181E-01	4.9501E-01	5.1197E-01	5.2998E-01	5.4913E-01	5.6956E-01	5.5624E-01	4.6317E-01
4.6007E-01	9.9051E-01	1.3858E+00	6.5317E-01	6.6220E-01	6.7331E-01	6.9223E-01	5.8636E-01	2.0543E-01	4.6990E-02
0.0	0.0	5.1177E+00							

ENERGY (52) 2550 KEV - 2600 KEV

2.9869E-01	2.9892E-01	2.9919E-01	2.9948E-01	2.9980E-01	3.0016E-01	3.0055E-01	3.0099E-01	3.0147E-01	3.0201E-01
3.0260E-01	3.0326E-01	3.0399E-01	3.0480E-01	3.0570E-01	3.0669E-01	3.0779E-01	3.0901E-01	3.1036E-01	3.1186E-01
3.1352E-01	3.1535E-01	3.1739E-01	3.1964E-01	3.2214E-01	3.2491E-01	3.2798E-01	3.3138E-01	3.3514E-01	3.3931E-01
5.4858E-01	7.5687E-01	4.9491E-01	5.1143E-01	5.2899E-01	5.4769E-01	5.6767E-01	5.5578E-01	4.6837E-01	4.6628E-01
9.6900E-01	1.3456E+00	6.5230E-01	6.6213E-01	6.7407E-01	6.9355E-01	5.8533E-01	1.9548E-01	4.4553E-02	0.0
0.0	5.2039E+00								

ENERGY (51) 2500 KEV - 2550 KEV

3.0715E-01	3.0742E-01	3.0771E-01	3.0804E-01	3.0841E-01	3.0882E-01	3.0926E-01	3.0976E-01	3.1031E-01	3.1092E-01
3.1160E-01	3.1235E-01	3.1318E-01	3.1410E-01	3.1512E-01	3.1625E-01	3.1750E-01	3.1889E-01	3.2043E-01	3.2213E-01
3.2402E-01	3.2611E-01	3.2842E-01	3.3099E-01	3.3383E-01	3.3698E-01	3.4047E-01	3.4433E-01	3.4861E-01	5.4609E-01
7.4345E-01	4.9664E-01	5.1275E-01	5.2989E-01	5.4818E-01	5.6773E-01	5.5735E-01	4.7577E-01	4.7476E-01	9.4930E-01
1.3069E+00	6.5379E-01	6.6453E-01	6.7741E-01	6.9758E-01	5.8570E-01	1.8601E-01	4.2067E-02	0.0	0.0
5.2922E+00									

ENERGY (50) 2450 KEV - 2500 KEV

3.1753E-01	3.1783E-01	3.1817E-01	3.1855E-01	3.1897E-01	3.1943E-01	3.1995E-01	3.2052E-01	3.2115E-01	3.2184E-01
3.2262E-01	3.2347E-01	3.2442E-01	3.2548E-01	3.2664E-01	3.2793E-01	3.2936E-01	3.3095E-01	3.3270E-01	3.3465E-01
3.3681E-01	3.3919E-01	3.4184E-01	3.4477E-01	3.4802E-01	3.5162E-01	3.5561E-01	3.6002E-01	5.4552E-01	7.3177E-01
5.0043E-01	5.1615E-01	5.3290E-01	5.5080E-01	5.6997E-01	5.6117E-01	4.8558E-01	4.8573E-01	9.3166E-01	1.2700E+00
6.5786E-01	6.6961E-01	6.8356E-01	7.0454E-01	5.8986E-01	1.7796E-01	3.9536E-02	0.0	0.0	5.3823E+00

ENERGY (49) 2400 KEV - 2450 KEV

3.2860E-01	3.2896E-01	3.2935E-01	3.2978E-01	3.3026E-01	3.3079E-01	3.3137E-01	3.3202E-01	3.3275E-01	3.3355E-01
3.3443E-01	3.3541E-01	3.3650E-01	3.3770E-01	3.3904E-01	3.4051E-01	3.4215E-01	3.4397E-01	3.4598E-01	3.4820E-01
3.5067E-01	3.5340E-01	3.5643E-01	3.5979E-01	3.6351E-01	3.6763E-01	3.7219E-01	5.4552E-01	7.2047E-01	5.0487E-01
5.2021E-01	5.3658E-01	5.5410E-01	5.7289E-01	5.6572E-01	4.9625E-01	4.9761E-01	9.1447E-01	1.2332E+00	6.6281E-01
6.7563E-01	6.9069E-01	7.1254E-01	5.9411E-01	1.6748E-01	3.6961E-02	0.0	0.0	5.4766E+00	

ENERGY (48) 2350 KEV - 2400 KEV

3.4254E-01	3.4294E-01	3.4339E-01	3.4389E-01	3.4444E-01	3.4506E-01	3.4573E-01	3.4648E-01	3.4731E-01	3.4824E-01
3.4926E-01	3.5039E-01	3.5164E-01	3.5303E-01	3.5457E-01	3.5627E-01	3.5816E-01	3.6025E-01	3.6257E-01	3.6514E-01
3.6798E-01	3.7113E-01	3.7463E-01	3.7850E-01	3.8278E-01	3.8753E-01	5.4856E-01	7.1207E-01	5.1251E-01	5.2751E-01
5.4356E-01	5.6077E-01	5.7926E-01	5.7383E-01	5.1065E-01	5.1334E-01	9.0083E-01	1.1997E+00	6.7190E-01	6.8596F-01
7.0233E-01	7.2526E-01	5.9994E-01	1.5751E-01	3.4345E-02	0.0	0.0	5.5719E+00		

ENERGY (47) 2300 KEV - 2350 KEV

3.5519E-01	3.5566E-01	3.5617E-01	3.5675E-01	3.5738E-01	3.5808E-01	3.5886E-01	3.5972E-01	3.6067E-01	3.6172E-01
3.6289E-01	3.6419E-01	3.6563E-01	3.6722E-01	3.6898E-01	3.7093E-01	3.7310E-01	3.7550E-01	3.7815E-01	3.8109E-01
3.8435E-01	3.8797E-01	3.9197E-01	3.9640E-01	4.0132E-01	5.4986E-01	7.0172E-01	5.1839E-01	5.3302E-01	5.4870E-01
5.6554E-01	5.8368E-01	5.7995E-01	5.2310E-01	5.2707E-01	8.8472E-01	1.1634E+00	6.7864E-01	6.9385E-01	7.1143E-01
7.3532E-01	6.0697E-01	1.4806E-01	3.1692E-02	0.0	0.0	5.6750E+00			

ENERGY (46) 2250 KEV - 2300 KEV

3.6377E-01	3.6430E-01	3.6488E-01	3.6553E-01	3.6624E-01	3.6704E-01	3.6791E-01	3.6889E-01	3.6997E-01	3.7116E-01
3.7249E-01	3.7395E-01	3.7558E-01	3.7738E-01	3.7938E-01	3.8159E-01	3.8404E-01	3.8675E-01	3.8976E-01	3.9309E-01
3.9678E-01	4.0087E-01	4.0540E-01	4.1042E-01	5.4626E-01	6.8627E-01	5.1929E-01	5.3345E-01	5.4866E-01	5.6502E-01
5.8267E-01	5.8054E-01	5.2994E-01	5.3504E-01	8.6234E-01	1.1204E+00	6.7893E-01	6.9506E-01	7.1361E-01	7.3817E-01
6.8647E-01	1.9133E-01	2.9007E-02	0.0	0.0	5.7892E+00				

ENERGY (45) 2200 KEV - 2250 KEV

3.7869E-01	3.7930E-01	3.7997E-01	3.8071E-01	3.8154E-01	3.8245E-01	3.8346E-01	3.8458E-01	3.8582E-01	3.8720E-01
3.8872E-01	3.9041E-01	3.9228E-01	3.9436E-01	3.9665E-01	3.9920E-01	4.0202E-01	4.0514E-01	4.0860E-01	4.1243E-01
4.1668E-01	4.2139E-01	4.2660E-01	5.4972E-01	6.7784E-01	5.2749E-01	5.4131E-01	5.5619E-01	5.7223E-01	5.8957E-01
5.8923E-01	5.4517E-01	5.5165E-01	8.4859E-01	1.0860E+00	6.8867E-01	7.0609E-01	7.2600E-01	7.5168E-01	6.9750E-01
1.8287E-01	2.6294E-02	0.0	0.0	5.9009E+00					

ENERGY (44) 2150 KEV - 2200 KEV

3.8920E-01	3.8989E-01	3.9065E-01	3.9149E-01	3.9243E-01	3.9347E-01	3.9461E-01	3.9589E-01	3.9730E-01	3.9886E-01
4.0059E-01	4.0251E-01	4.0463E-01	4.0699E-01	4.0960E-01	4.1249E-01	4.1569E-01	4.1924E-01	4.2317E-01	4.2752E-01
4.3235E-01	4.3769E-01	5.4795E-01	6.6400E-01	5.3032E-01	5.4370E-01	5.5813E-01	5.7374E-01	5.9063E-01	5.9196E-01
5.5432E-01	5.6201E-01	8.2821E-01	1.0446E+00	6.9147E-01	7.0992E-01	7.3091E-01	7.5739E-01	7.0082E-01	1.7265E-01
2.3559E-02	0.0	0.0	6.0248E+00						

ENERGY (43) 2100 KEV - 2150 KEV

4.0907E-01	4.0987E-01	4.1075E-01	4.1173E-01	4.1282E-01	4.1402E-01	4.1536E-01	4.1683E-01	4.1847E-01	4.2028E-01
4.2229E-01	4.2452E-01	4.2699E-01	4.2972E-01	4.3275E-01	4.3610E-01	4.3982E-01	4.4393E-01	4.4850E-01	4.5355E-01
4.5915E-01	5.5662E-01	6.6068E-01	5.4393E-01	5.5705E-01	5.7124E-01	5.8662E-01	6.0332E-01	6.0660E-01	5.7571E-01
5.8497E-01	8.2068E-01	1.0163E+00	7.0808E-01	7.2808E-01	7.5072E-01	7.7863E-01	7.2006E-01	1.6805E-01	2.0810E-02
0.0	0.0	6.1422E+00							

ENERGY (42) 2050 KEV - 2100 KEV

4.2234E-01	4.2325E-01	4.2426E-01	4.2538E-01	4.2662E-01	4.2799E-01	4.2951E-01	4.3120E-01	4.3306E-01	4.3513E-01
4.3742E-01	4.3996E-01	4.4277E-01	4.4589E-01	4.4934E-01	4.5317E-01	4.5741E-01	4.6210E-01	4.6730E-01	4.7307E-01
5.5765E-01	6.4956E-01	5.4965E-01	5.6238E-01	5.7610E-01	5.9118E-01	6.0750E-01	6.1252E-01	5.8817E-01	5.9874E-01
8.0367E-01	9.7814E-01	7.1456E-01	7.3573E-01	7.5960E-01	7.8849E-01	7.2779E-01	1.5878E-01	1.8054E-02	0.0
0.0	6.2761E+00								

ENERGY (41) 2000 KEV - 2050 KEV

4.3646E-01	4.3750E-01	4.3865E-01	4.3993E-01	4.4134E-01	4.4291E-01	4.4464E-01	4.4656E-01	4.4869E-01	4.5105E-01
4.5367E-01	4.5656E-01	4.5977E-01	4.6333E-01	4.6727E-01	4.7163E-01	4.7647E-01	4.8182E-01	4.8776E-01	5.5951E-01
6.3927E-01	5.5619E-01	5.6853E-01	5.8196E-01	5.9658E-01	6.1254E-01	6.1932E-01	6.0152E-01	6.1342E-01	7.8768E-01
9.4100E-01	7.2203E-01	7.4441E-01	7.6954E-01	7.9944E-01	7.3698E-01	1.5210E-01	1.5301E-02	0.0	0.0
6.4158E+00									

ENERGY (40) 1950 KEV - 2000 KEV

4.5415E-01	4.5534E-01	4.5667E-01	4.5813E-01	4.5975E-01	4.6155E-01	4.6354E-01	4.6575E-01	4.6820E-01	4.7091E-01
4.7391E-01	4.7724E-01	4.8092E-01	4.8500E-01	4.8953E-01	4.9454E-01	5.0009E-01	5.0624E-01	5.6531E-01	6.3303E-01
5.6675E-01	5.7878E-01	5.9190E-01	6.0625E-01	6.2193E-01	6.3056E-01	6.1939E-01	6.3274E-01	7.7673E-01	9.0909E-01
7.3468E-01	7.5845E-01	7.8506E-01	8.1619E-01	7.5327E-01	1.4434E-01	1.2563E-02	0.0	0.0	6.5587E+00

ENERGY (39) 1900 KEV - 1950 KEV

4.6735E-01	4.6870E-01	4.7021E-01	4.7187E-01	4.7371E-01	4.7576E-01	4.7802E-01	4.8053E-01	4.8331E-01	4.8639E-01
4.8980E-01	4.9358E-01	4.9776E-01	5.0240E-01	5.0754E-01	5.1324E-01	5.1955E-01	5.6602E-01	6.2173E-01	5.7200E-01
5.8363E-01	5.9637E-01	6.1033E-01	6.2563E-01	6.3595E-01	6.3118E-01	6.4581E-01	7.5974E-01	8.7109E-01	7.4057E-01
7.6546E-01	7.9326E-01	8.2533E-01	7.5802E-01	1.3448E-01	9.8509E-03	0.0	0.0	6.7157E+00	

ENERGY (38) 1850 KEV - 1900 KEV

4.8545E-01	4.8701E-01	4.8873E-01	4.9063E-01	4.9275E-01	4.9509E-01	4.9768E-01	5.0055E-01	5.0373E-01	5.0726E-01
5.1117E-01	5.1550E-01	5.2029E-01	5.2561E-01	5.3149E-01	5.3801E-01	5.7221E-01	6.1613E-01	5.8277E-01	5.9411E-01
6.0656E-01	6.2024E-01	6.3529E-01	6.4741E-01	6.4909E-01	6.6514E-01	7.4976E-01	8.4054E-01	7.5344E-01	7.7969E-01
8.0894E-01	8.4223E-01	7.7376E-01	1.3005E-01	7.1802E-03	0.0	0.0	6.8747E+00		

ENERGY (37) 1800 KEV - 1850 KEV

5.0561E-01	5.0739E-01	5.0937E-01	5.1156E-01	5.1398E-01	5.1667E-01	5.1965E-01	5.2295E-01	5.2661E-01	5.3066E-01
5.3515E-01	5.4012E-01	5.4563E-01	5.5174E-01	5.5850E-01	5.8076E-01	6.1317E-01	5.9586E-01	6.0696E-01	6.1918E-01
6.3267E-01	6.4754E-01	6.6148E-01	6.6954E-01	6.8707E-01	7.4323E-01	8.1391E-01	7.6939E-01	7.9712E-01	8.2792E-01
8.6257E-01	7.8979E-01	1.2279E-01	4.5674E-03	0.0	0.0	7.0394E+00			

ENERGY (36) 1750 KEV - 1800 KEV

5.2398E-01	5.2601E-01	5.2827E-01	5.3077E-01	5.3354E-01	5.3661E-01	5.4001E-01	5.4378E-01	5.4796E-01	5.5258E-01
5.5771E-01	5.6339E-01	5.6968E-01	5.7665E-01	5.8737E-01	6.0853E-01	6.0676E-01	6.1758E-01	6.2955E-01	6.4278E-01
6.5742E-01	6.7307E-01	6.8725E-01	7.0614E-01	7.3475E-01	7.8572E-01	7.8241E-01	8.1144E-01	8.4364E-01	8.7947E-01
8.0750E-01	1.1721E-01	2.0316E-03	0.0	0.0	7.2172E+00				

ENERGY (35) 1700 KEV - 1750 KEV

5.2883E-01	5.3110E-01	5.3361E-01	5.3639E-01	5.3948E-01	5.4289E-01	5.4668E-01	5.5087E-01	5.5551E-01	5.6066E-01
5.6636E-01	5.7268E-01	5.7968E-01	5.8743E-01	5.9602E-01	6.0554E-01	6.1609E-01	6.2777E-01	6.4072E-01	6.5506E-01
6.7096E-01	6.8856E-01	7.0807E-01	7.2968E-01	7.5363E-01	7.8016E-01	8.0955E-01	8.4212E-01	8.7820E-01	8.0356E-01
1.1127E-01	0.0	0.0	0.0	7.4187E+00					

ENERGY (34) 1650 KEV - 1700 KEV

5.4467E-01	5.4724E-01	5.5009E-01	5.5325E-01	5.5674E-01	5.6062E-01	5.6491E-01	5.6966E-01	5.7493E-01	5.8077E-01
5.8724E-01	5.9440E-01	6.0234E-01	6.1114E-01	6.2089E-01	6.3168E-01	6.4365E-01	6.5690E-01	6.7159E-01	6.8786E-01
7.0588E-01	7.2585E-01	7.4798E-01	7.7250E-01	7.9966E-01	8.2975E-01	8.6309E-01	9.0004E-01	8.2301E-01	1.1478E-01
0.0	0.0	0.0	7.6114E+00						

ENERGY (33) 1600 KEV - 1650 KEV

5.7140E-01	5.7437E-01	5.7766E-01	5.8130E-01	5.8534E-01	5.8982E-01	5.9477E-01	6.0026E-01	6.0635E-01	6.1309E-01
6.2056E-01	6.2883E-01	6.3800E-01	6.4816E-01	6.5941E-01	6.7188E-01	6.8569E-01	7.0100E-01	7.1796E-01	7.3675E-01
7.5756E-01	7.8063E-01	8.0618E-01	8.3449E-01	8.6385E-01	9.0060E-01	9.3911E-01	8.5969E-01	1.1919E-01	0.0
0.0	0.0	7.8017E+00							

ENERGY (32) 1550 KEV - 1600 KEV

5.8507E-01	5.8841E-01	5.9212E-01	5.9622E-01	6.0077E-01	6.0581E-01	6.1139E-01	6.1758E-01	6.2443E-01	6.3203E-01
6.4044E-01	6.4976E-01	6.6009E-01	6.7153E-01	6.8421E-01	6.9825E-01	7.1382E-01	7.3106E-01	7.5016E-01	7.7132E-01
7.9477E-01	8.2075E-01	8.4953E-01	8.8142E-01	9.1676E-01	9.5590E-01	9.5770E-01	1.6188E-01	0.0	0.0
0.0	8.0219E+00								

ENERGY (31) 1500 KEV - 1550 KEV

6.0178E-01	6.0556E-01	6.0975E-01	6.1439E-01	6.1953E-01	6.2523E-01	6.3154E-01	6.3854E-01	6.4629E-01	6.5487E-01
6.6438E-01	6.7492E-01	6.8660E-01	6.9953E-01	7.1387E-01	7.2975E-01	7.4734E-01	7.6683E-01	7.8843E-01	8.1236E-01
8.3887E-01	8.6824E-01	9.0078E-01	9.3684E-01	9.7678E-01	9.7901E-01	2.5249E-01	0.0	0.0	0.0
8.2509E+00									

ENERGY (30) 1450 KEV - 1500 KEV

6.1861E-01	6.2288E-01	6.2762E-01	6.3286E-01	6.3867E-01	6.4511E-01	6.5224E-01	6.6014E-01	6.6890E-01	6.7860E-01
6.8935E-01	7.0125E-01	7.1445E-01	7.2906E-01	7.4526E-01	7.6320E-01	7.8308E-01	8.0510E-01	8.2950E-01	8.5654E-01
8.8649E-01	9.1968E-01	9.5645E-01	9.9719E-01	9.9891E-01	2.5440E-01	0.0	0.0	0.0	8.4957E+00

ENERGY (29) 1400 KEV - 1450 KEV

6.4807E-01	6.5298E-01	6.5843E-01	6.6446E-01	6.7115E-01	6.7856E-01	6.8676E-01	6.9585E-01	7.0593E-01	7.1709E-01
7.2945E-01	7.4315E-01	7.5833E-01	7.7514E-01	7.9378E-01	8.1442E-01	8.3729E-01	8.6263E-01	8.9070E-01	9.2181E-01
9.5627E-01	9.9445E-01	1.0368E+00	1.0379E+00	2.6288E-01	0.0	0.0	0.0	8.7407E+00	

ENERGY (28) 1350 KEV - 1400 KEV

6.4113E-01	6.4765E-01	6.5489E-01	6.6290E-01	6.7179E-01	6.8162E-01	6.9252E-01	7.0460E-01	7.1798E-01	7.3281E-01
7.4923E-01	7.6743E-01	7.8760E-01	8.0994E-01	8.3469E-01	8.6211E-01	8.9249E-01	9.2615E-01	9.6345E-01	1.0048E+00
1.0505E+00	1.1013E+00	1.1088E+00	2.8075E-01	0.0	0.0	0.0	9.0165E+00		

ENERGY (27) 1300 KEV - 1350 KEV

6.7526E-01	6.8213E-01	6.8976E-01	6.9820E-01	7.0755E-01	7.1791E-01	7.2939E-01	7.4211E-01	7.5621E-01	7.7182E-01
7.8912E-01	8.0829E-01	8.2953E-01	8.5306E-01	8.7912E-01	9.0801E-01	9.4001E-01	9.7546E-01	1.0147E+00	1.0583E+00
1.1065E+00	1.1118E+00	2.8505E-01	0.0	0.0	0.0	9.3084E+00			

ENERGY (26) 1250 KEV - 1300 KEV

7.1313E-01	7.2040E-01	7.2844E-01	7.3736E-01	7.4724E-01	7.5818E-01	7.7031E-01	7.8374E-01	7.9862E-01	8.1511E-01
8.3338E-01	8.5363E-01	8.7605E-01	9.0090E-01	9.2843E-01	9.5894E-01	9.9273E-01	1.0302E+00	1.0717E+00	1.1176E+00
1.1194E+00	2.8681E-01	0.0	0.0	0.0	9.6195E+00				

ENERGY (25) 1200 KEV - 1250 KEV

7.5603E-01	7.6374E-01	7.7227E-01	7.8172E-01	7.9219E-01	8.0379E-01	8.1665E-01	8.3089E-01	8.4667E-01	8.6415E-01
8.8352E-01	9.0498E-01	9.2876E-01	9.5510E-01	9.8429E-01	1.0166E+00	1.0525E+00	1.0921E+00	1.1361E+00	1.1346E+00
2.8872E-01	0.0	0.0	0.0	9.9511E+00					

ENERGY (24) 1150 KEV - 1200 KEV

8.0108E-01	8.0923E-01	8.1827E-01	8.2829E-01	8.3938E-01	8.5168E-01	8.6530E-01	8.8039E-01	8.9711E-01	9.1563E-01
9.3616E-01	9.5890E-01	9.8409E-01	1.0120E+00	1.0429E+00	1.0772E+00	1.1152E+00	1.1572E+00	1.2038E+00	3.9091E-01
0.0	0.0	0.0	1.0310E+01						

ENERGY (23) 1100 KEV - 1150 KEV

8.7502E-01	8.8394E-01	8.9381E-01	9.0475E-01	9.1687E-01	9.3030E-01	9.4517E-01	9.6166E-01	9.7992E-01	1.0002E+00
1.0226E+00	1.0474E+00	1.0749E+00	1.1054E+00	1.1392E+00	1.1766E+00	1.2181E+00	1.2640E+00	5.1618E-01	0.0
0.0	0.0	1.0674E+01							

ENERGY (22) 1050 KEV - 1100 KEV

9.1611E-01	9.2544E-01	9.3578E-01	9.4723E-01	9.5992E-01	9.7398E-01	9.8956E-01	1.0068E+00	1.0259E+00	1.0471E+00
1.0706E+00	1.0966E+00	1.1254E+00	1.1573E+00	1.1927E+00	1.2319E+00	1.2753E+00	5.1635E-01	0.0	0.0
0.0	1.1115E+01								

ENERGY (21) 1000 KEV - 1050 KEV

9.8586E-01	9.9590E-01	1.0070E+00	1.0194E+00	1.0330E+00	1.0481E+00	1.0649E+00	1.0835E+00	1.1040E+00	1.1268E+00
1.1521E+00	1.1801E+00	1.2111E+00	1.2454E+00	1.2835E+00	1.3257E+00	5.3825E-01	0.0	0.0	0.0
1.1577E+01									

ENERGY (20) 950 KEV - 1000 KEV

1.0598E+00	1.0706E+00	1.0826E+00	1.0958E+00	1.1105E+00	1.1268E+00	1.1448E+00	1.1648E+00	1.1869E+00	1.2114E+00
1.2385E+00	1.2686E+00	1.3020E+00	1.3389E+00	1.3798E+00	5.5062E-01	0.0	0.0	0.0	1.2097E+01

ENERGY (19) 900 KEV - 950 KEV

1.1346E+00	1.1462E+00	1.1590E+00	1.1732E+00	1.1889E+00	1.2063E+00	1.2256E+00	1.2470E+00	1.2707E+00	1.2969E+00
1.3260E+00	1.3582E+00	1.3939E+00	1.4334E+00	6.9931E-01	0.0	0.0	0.0	1.2693E+01	

ENERGY (18) 850 KEV - 900 KEV

1.1879E+00	1.2000E+00	1.2134E+00	1.2282E+00	1.2447E+00	1.2629E+00	1.2831E+00	1.3055E+00	1.3303E+00	1.3578E+00
1.3882E+00	1.4219E+00	1.4593E+00	8.8839E-01	0.0	0.0	0.0	1.3506E+01		

ENERGY (17) 800 KEV - 850 KEV

1.2782E+00	1.2912E+00	1.3056E+00	1.3216E+00	1.3393E+00	1.3589E+00	1.3807E+00	1.4047E+00	1.4314E+00	1.4610E+00
1.4937E+00	1.5300E+00	9.2927E-01	0.0	0.0	0.0	1.4289E+01			

ENERGY (16) 750 KEV - 800 KEV

1.3822E+00	1.3963E+00	1.4119E+00	1.4291E+00	1.4483E+00	1.4695E+00	1.4930E+00	1.5190E+00	1.5479E+00	1.5799E+00
1.6153E+00	9.7867E-01	0.0	0.0	0.0	1.5160E+01				

ENERGY (15) 700 KEV - 750 KEV

1.5022E+00	1.5175E+00	1.5345E+00	1.5533E+00	1.5741E+00	1.5971E+00	1.6227E+00	1.6510E+00	1.6823E+00	1.7170E+00
1.3220E+00	0.0	0.0	0.0	1.6135E+01					

ENERGY (14) 650 KEV - 700 KEV

1.6406E+00	1.6573E+00	1.6758E+00	1.6963E+00	1.7190E+00	1.7442E+00	1.7721E+00	1.8030E+00	1.8372E+00	1.4116E+00
7.3296E-02	0.0	0.0	1.7245E+01						

ENERGY (13) 600 KEV - 650 KEV

1.8002E+00	1.8185E+00	1.8388E+00	1.8614E+00	1.8863E+00	1.9139E+00	1.9445E+00	1.9784E+00	1.6009E+00	7.8516E-02
0.0	0.0	1.8509E+01							

ENERGY (12) 550 KEV - 600 KEV

1.9827E+00	2.0029E+00	2.0253E+00	2.0501E+00	2.0775E+00	2.1080E+00	2.1417E+00	1.9072E+00	2.5606E-01	0.0
0.0	1.9969E+01								

ENERGY (11) 500 KEV - 550 KEV

2.1882E+00	2.2105E+00	2.2351E+00	2.2625E+00	2.2928E+00	2.3264E+00	2.1677E+00	2.7753E-01	0.0	0.0
2.1675E+01									

```
ENERGY ( 10 )   450 KEV -   500 KEV

2.4109E+00  2.4355E+00  2.4627E+00  2.4928E+00  2.5262E+00  2.4562E+00  6.0662E-01  0.0        0.0        2.3703E+01

ENERGY (  9 )   400 KEV -   450 KEV

2.6449E+00  2.6718E+00  2.7017E+00  2.7347E+00  2.7714E+00  8.8260E-01  0.0        0.0        2.6044E+01

ENERGY (  8 )   350 KEV -   400 KEV

2.8657E+00  2.8949E+00  2.9273E+00  2.9631E+00  1.3043E+00  0.0        0.0        2.8742E+01

ENERGY (  7 )   300 KEV -   350 KEV

3.0209E+00  3.0517E+00  3.0858E+00  1.9885E+00  0.0        0.0        3.1808E+01

ENERGY (  6 )   250 KEV -   300 KEV

3.0287E+00  3.0596E+00  2.4769E+00  1.2830E-01  0.0        3.5062E+01

ENERGY (  5 )   200 KEV -   250 KEV

2.7496E+00  2.6739E+00  4.6531E-01  0.0        3.8185E+01

ENERGY (  4 )   150 KEV -   200 KEV

2.0801E+00  8.7951E-01  3.5822E-01  4.0634E+01

ENERGY (  3 )   100 KEV -   150 KEV

7.1101E-01  1.1478E+00  4.3048E+01

ENERGY (  2 )    50 KEV -   100 KEV

4.5812E+00  4.1723E+01
```

Appendix 2 Example of Data Processing of NaI Monitor by Stripping Method[*1]

This data processing is a calculation procedure to analyze the pulse-height distribution obtained using a 3 in. φ spherical NaI (Tl) scintillator, to obtain information on gamma-rays and cosmic rays in the range of 0–3 MeV.

It can be generally classified into (1) the process of converting measurement results to 10 keV/ch, (2) selection and introduction of input data necessary for processing, (3) calculation performed based on the input data, and (4) display of calculation results, and others.

(1) Process of converting measurement results to 10 keV/ch

This conversion process standardizes the energy calibration, i.e., the relationship between the channel and the energy of the pulse-height distribution obtained (i.e., creation of a uniform channel width among all energy regions). The calibration is performed by interpolating the numerical values between channels. Usually, three points of a zero point, ^{40}K (1461 keV), and ^{208}Tl (2614 keV) are used to make corrections in the first order. However, to correct the distortion in the higher orders, fitting is performed in the nth order using more than (the order + 2) the number of sets of the relationship between the channel and the energy of the measured pulse-height distribution. The interpolation of the numerical values between channels is determined by means of proportional distribution. Given that the result after conversion to 10 keV/ch is used as the basis for data processing performed in and after (2), the correctness of this calibration affects the accuracy of the result.

(2) Input data required for processing

Considering the input data required for processing, the P/T ratio of the detector (NaI (Tl) scintillator) is the only measured data necessary for this analysis. This value is obtained using a point source of monochromatic gamma-rays, as shown in Figure 2.1 and Table 2.1.

In addition, the data of the overall counting efficiency required for processing is obtained by calculating the literature values for specific constants such as the active cross-section of gamma-rays and the scintillator.

(3) Calculation processing based on input data

The processing of the measurement results obtained from conversion to 10 keV/ch includes (a) subtraction of the Compton contribution and (b) application of peak efficiency. Peak efficiency (PE) can be determined from the following equation:

$$\mathrm{PE} = \left\{1 - \frac{2}{(\mu d)^2} + 2e^{-\mu d}\left[\frac{1}{(\mu d)^2} + \frac{1}{\mu d}\right]\right\}(PTTR)$$

using the P/T ratio (PTTR), the diameter “d” of the spherical scintillator, as well as the attenuation coefficient μ of the NaI (TI) scintillator. The peak efficiency values calculated for the 3 in. φ spherical NaI (Tl) scintillator are shown in Table 2.2. The peak area count N_p and the count N_c of the Compton continuum are as shown in the following equation:

$$N_C = N_P(1 - PTTR)/(PTTR)$$

[*1] As this is quoted from the Radioactivity Measurement Series No. 20 "Environmental Gamma-ray Spectrometry" (1990), the former units of “R” and “rad” are used.

N_c is obtained from N_p using the preceding equation and N_c is subtracted by distributing it evenly to each channel of the Compton continuum.

This corresponds to a rectangular approximation of the Compton continuum (Figure 2.2). In practice, the channel corresponding to 3 MeV (10 keV width) is first regarded as the total absorption peak (the area N_p count) of the photon of energy $E\gamma$, and N_c obtained from this N_p is distributed evenly to the corresponding channels from the Compton edge E_c (MeV) (= $E\gamma/(1 + 0.511/2\ E\gamma)$) to 0. Subsequently, this operation is repeated sequentially from the high-energy side to the low-energy side. In this way, the spectrum excluding the Compton contribution is obtained and the photon spectrum expressed as the gamma-ray flux density is determined by applying the peak efficiency of each energy and the attenuation rate of the gamma-rays caused by the scintillator case for each channel. Furthermore, the exposure dose rate for each energy is also obtained by applying the conversion factor of the photon flux density-exposure dose to each energy.

(4) Reliability of the results

The errors produced by this method include statistical errors such as the counting errors, including calculation processes, and systematic errors, such as counting losses, errors in conversion to 10 keV/ch, P/T ratio, the subtraction method of a Compton continuum (differences between a rectangle and the actual shape), selection of basic constants, and differences in background data.

(5) Analysis and representation of results

Normally, the range of 0–3 MeV is treated as the information on gamma-rays, and the range of 3 MeV or more is mainly treated as the information on cosmic rays.

i) Analysis and representation of the region of 0–3 MeV

[1] Representation of the results of converting the measured data to 10 keV/ch.
[2] Representation of the spectrum in which a Compton contribution is subtracted.
[3] Representation of the results of calculating the flux density using the photon peak efficiency for the aforementioned spectrum, i.e., the representation of the photon spectrum.
[4] Representation of the results for converting the photon spectrum to the exposure dose spectrum.
[5] For [1], [3], and [4], the results of integrating the values of 3 Mev–10 keV into the low energy region individually.

The standard representations are as follows. Representations of [1], [2], [3], and [4] are generally made using a semilogarithmic plot. For the integration result of [4], a linear expression in percentage is used for integral values up to the specified energy (generally, up to 30 keV) set to 1. It is also possible to represent gamma ray and cosmic ray doses together with data numbers as shown in Figure 2.3.

ii) Treatment of the region above 3 MeV

The information on the region above 3 MeV is mainly based on cosmic rays and in a 3 in. φ spherical NaI (Tl) scintillator, the cosmic ray dose rate equivalent can be obtained by multiplying the counting rate above 3 MeV by the constant (N_3) of 2.12 (μR/h)/(cps).

However, in an environment in which gamma-rays with energy above 3 MeV are present (in the vicinity of accelerator facilities, reactor facilities, etc.), high-energy gamma-rays may be overestimated as cosmic ray contributions. For measurements recorded at such a site, it is necessary to use an energy region with little influence due to artificial radiation, for example, a region of 10 MeV or more, to calculate the cosmic ray dose. In this process, N_{10} = 3.5 (μR/h)/(cps) should be used as the aforementioned constant. Although the ratio (N_{10}/N_3) of both of the constants varies as 1.3–1.87 depending on the cosmic ray components in the environment, 1.65 is used here.

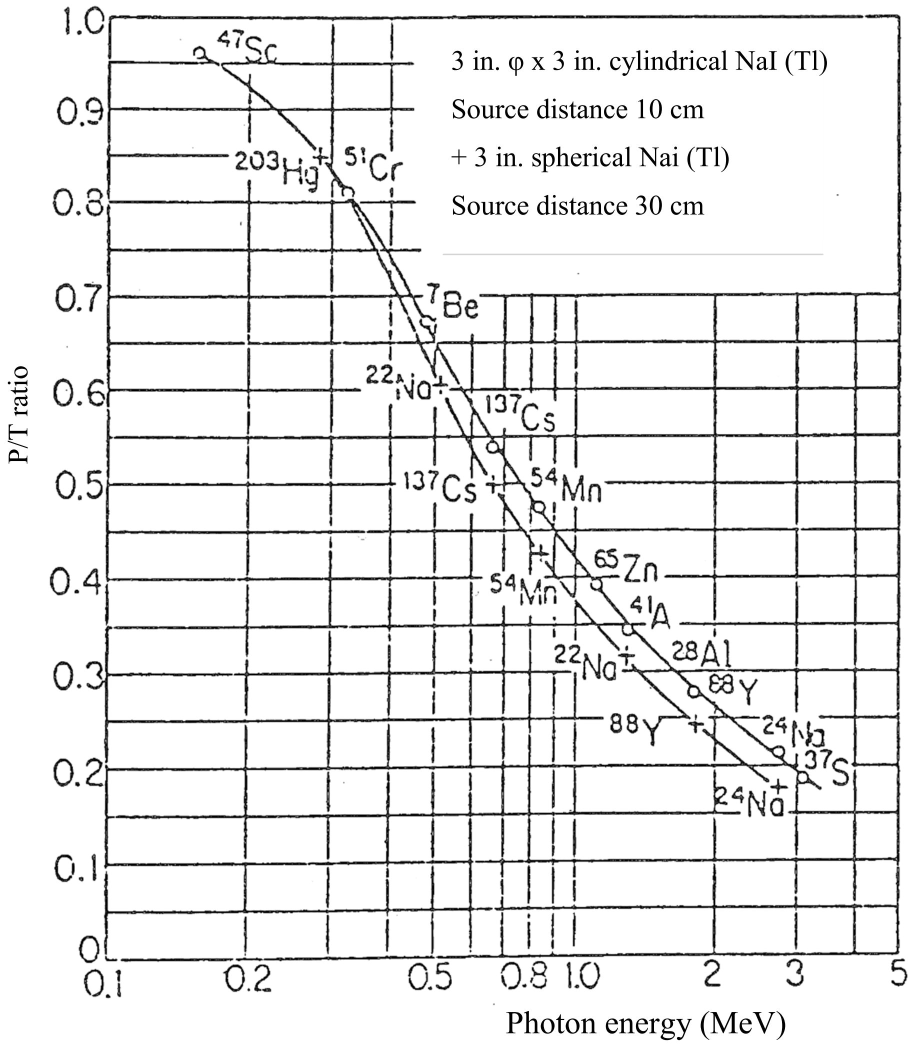


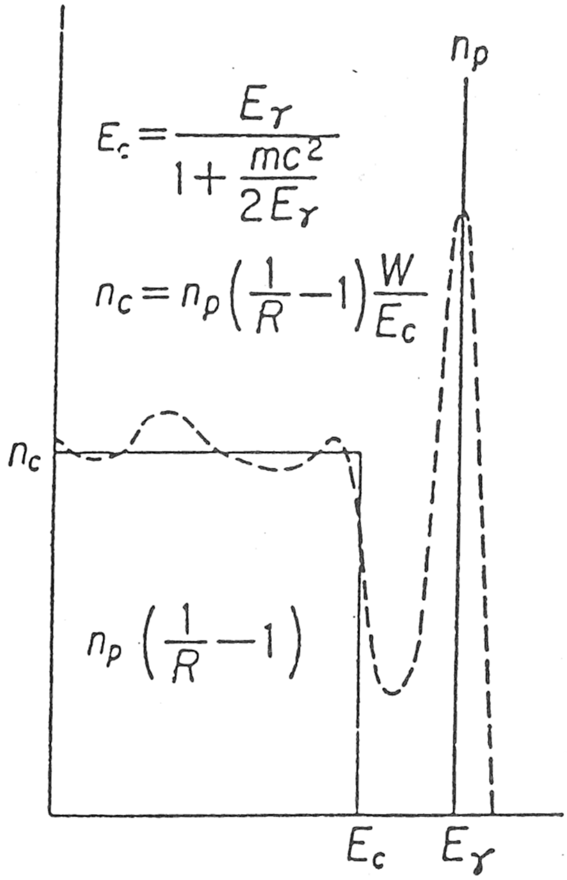
Figure 2.1 P/T ratio of NaI (Tl) scintillator

Table 2.1 P/T ratio of 3 in. φ spherical NaI (Tl) scintillator

keV	0	10	20	30	40	50	60	70	80	90
0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0
100	0.99	0.985	0.98	0.97	0.962	0.955	0.945	0.94	0.933	0.925
200	0.916	0.907	0.9	0.89	0.882	0.872	0.862	0.853	0.843	0.83
300	0.82	0.806	0.795	0.785	0.77	0.76	0.746	0.735	0.725	0.713
400	0.703	0.691	0.682	0.67	0.66	0.65	0.641	0.631	0.621	0.612
500	0.604	0.596	0.586	0.578	0.57	0.562	0.555	0.547	0.541	0.535
600	0.527	0.522	0.516	0.51	0.505	0.5	0.495	0.49	0.485	0.48
700	0.475	0.47	0.466	0.461	0.457	0.453	0.449	0.445	0.441	0.438
800	0.434	0.431	0.428	0.425	0.421	0.417	0.414	0.411	0.408	0.405
900	0.401	0.398	0.396	0.393	0.391	0.387	0.385	0.382	0.379	0.376
1000	0.374	0.371	0.369	0.366	0.364	0.361	0.359	0.357	0.355	0.353
1100	0.35	0.348	0.346	0.344	0.342	0.34	0.338	0.336	0.332	0.33
1200	0.33	0.328	0.326	0.324	0.322	0.32	0.318	0.316	0.314	0.312
1300	0.31	0.308	0.307	0.305	0.304	0.302	0.3	0.298	0.296	0.295
1400	0.294	0.292	0.291	0.29	0.289	0.287	0.285	0.284	0.283	0.281
1500	0.28	0.278	0.277	0.276	0.275	0.273	0.272	0.271	0.27	0.269
1600	0.267	0.266	0.265	0.264	0.263	0.261	0.26	0.259	0.258	0.257
1700	0.255	0.254	0.253	0.252	0.251	0.25	0.249	0.248	0.247	0.246
1800	0.245	0.2446	0.243	0.242	0.241	0.24	0.239	0.238	0.237	0.236
1900	0.235	0.234	0.233	0.232	0.231	0.23	0.229	0.228	0.227	0.227
2000	0.226	0.225	0.224	0.224	0.223	0.222	0.221	0.22	0.22	0.219
2100	0.219	0.218	0.217	0.216	0.215	0.214	0.213	0.212	0.211	0.211
2200	0.21	0.209	0.209	0.208	0.208	0.207	0.207	0.206	0.205	0.204
2300	0.204	0.203	0.203	0.202	0.202	0.201	0.2	0.199	0.198	0.197
2400	0.197	0.196	0.196	0.195	0.195	0.194	0.193	0.193	0.192	0.191
2500	0.191	0.19	0.19	0.189	0.189	0.187	0.187	0.186	0.186	0.185
2600	0.185	0.184	0.184	0.183	0.182	0.181	0.181	0.18	0.18	0.179
2700	0.179	0.178	0.178	0.177	0.177	0.176	0.176	0.175	0.175	0.174
2800	0.174	0.173	0.173	0.172	0.172	0.171	0.171	0.17	0.17	0.169
2900	0.169	0.168	0.168	0.167	0.167	0.166	0.165	0.165	0.164	0.164

Table 2.2 Peak efficiency of 3 in. φ spherical NaI (Tl) scintillator

keV	0	10	20	30	40	50	60	70	80	90
0	1.00	1.00	1.00	1.00	1.00	1.00	1.00	1.00	0.999	0.999
100	0.988	0.982	0.976	0.964	0.954	0.945	0.932	0.924	0.913	0.900
200	0.887	0.873	0.861	0.845	0.831	0.815	0.799	0.784	0.768	0.749
300	0.734	0.716	0.700	0.685	0.666	0.652	0.634	0.620	0.607	0.592
400	0.579	0.565	0.554	0.540	0.528	0.516	0.506	0.495	0.484	0.474
500	0.466	0.457	0.447	0.438	0.430	0.421	0.414	0.406	0.400	0.393
600	0.386	0.380	0.374	0.368	0.363	0.358	0.353	0.348	0.343	0.338
700	0.333	0.328	0.324	0.319	0.315	0.311	0.307	0.304	0.300	0.296
800	0.293	0.289	0.286	0.283	0.280	0.276	0.273	0.270	0.267	0.264
900	0.261	0.258	0.256	0.253	0.251	0.248	0.246	0.243	0.241	0.238
1000	0.236	0.234	0.232	0.229	0.227	0.225	0.223	0.221	0.219	0.218
1100	0.215	0.213	0.212	0.210	0.208	0.206	0.204	0.202	0.200	0.198
1200	0.197	0.196	0.194	0.192	0.191	0.189	0.187	0.186	0.184	0.182
1300	0.181	0.179	0.178	0.177	0.176	0.174	0.173	0.171	0.170	0.169
1400	0.168	0.166	0.165	0.164	0.163	0.162	0.160	0.160	0.159	0.157
1500	0.156	0.155	0.154	0.153	0.153	0.151	0.150	0.150	0.149	0.148
1600	0.147	0.146	0.145	0.144	0.144	0.142	0.141	0.141	0.140	0.139
1700	0.138	0.137	0.136	0.136	0.135	0.134	0.134	0.133	0.132	0.131
1800	0.131	0.130	0.129	0.128	0.128	0.127	0.126	0.126	0.125	0.124
1900	0.123	0.123	0.122	0.121	0.121	0.120	0.119	0.118	0.118	0.118
2000	0.117	0.116	0.116	0.116	0.115	0.114	0.114	0.113	0.113	0.113
2100	0.112	0.112	0.111	0.111	0.110	0.109	0.109	0.108	0.108	0.107
2200	0.107	0.106	0.106	0.106	0.105	0.105	0.105	0.104	0.104	0.103
2300	0.103	0.102	0.102	0.102	0.102	0.101	0.100	0.0997	0.0991	0.0986
2400	0.0985	0.0979	0.0978	0.0972	0.0972	0.0966	0.0960	0.0959	0.0953	0.0948
2500	0.0947	0.0942	0.0941	0.0935	0.0935	0.0924	0.0923	0.0918	0.0917	0.0912
2600	0.0911	0.0906	0.0905	0.0899	0.0894	0.0888	0.0887	0.0882	0.0881	0.0875
2700	0.0875	0.0869	0.0869	0.0864	0.0863	0.0857	0.0857	0.0852	0.0851	0.0846
2800	0.0845	0.0840	0.0840	0.0834	0.0834	0.0829	0.0828	0.0823	0.0822	0.0817
2900	0.0817	0.0811	0.0811	0.0805	0.0805	0.0800	0.0794	0.0794	0.0788	0.0788



n_p: Peak count

$E\gamma$: Peak energy

E_c: Energy at a Compton end

R: P/T ratio

W: Channel width

n_c: Count/channel

Figure 2.2 Subtraction method for a Compton continuum

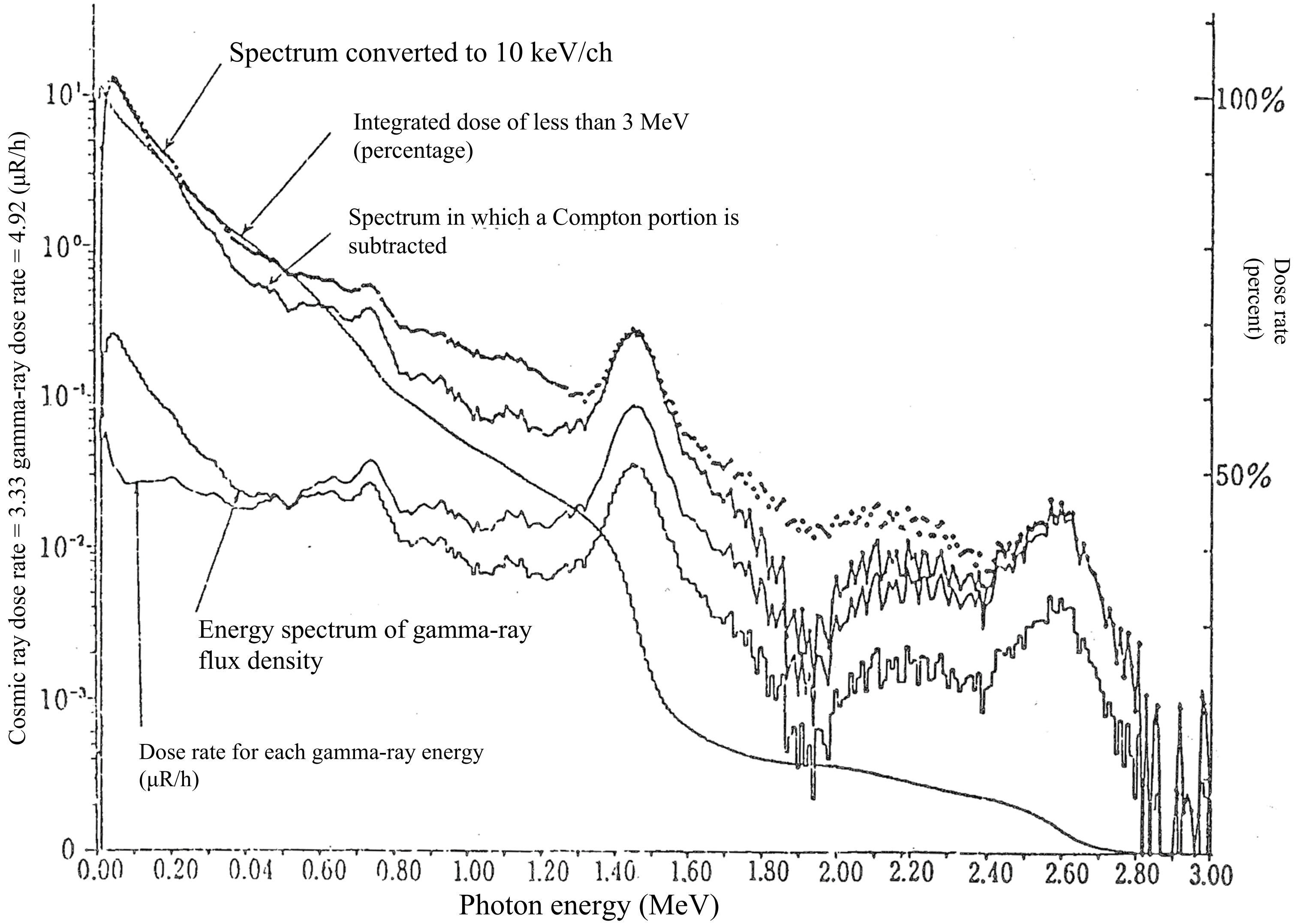


Figure 2.3 Representation example of spectrum analysis results using a 3 in. φ spherical NaI (Tl) scintillation detector

Table 2.3 Representation example of spectrum analysis results using a 3 in. φ spherical NaI (Tl) scintillation

DATA NO. *C*	CHANNEL	KEV/CH	TIME	LAST-CH	LIVE-TIME	A (I E-CNT	COS-DOSE(UR/H)	GAMMA(UR/H)(CH)		WEIGHT	E/CH
262097.0	1024.0	10.00	2003.40	446.2	2400.0	3443.0	3.041	0.632E+01	3	820.0	6.73

ORIGNL	RESULTS	PER D	0-50	50-100	100-250	250-500	500-1000	-2000	-3000		
DATA	212.08CPS	33.5+	-0.296	5.548	51.618	23.424	12.102	6.876	0.730	II(C)	3443.0
FLUX	5.84C	0.913	-3.418	1.881	39.038	21.322	15.537	19.984	4.335	JJ(C)	3020.0 R
DOSE	6.467	1.00	-2.611	0.303	10.273	13.111	19.431	43.414	13.840	3-MEV	3443.0
ADUS	5.403	0.855	-0.030	1.460	23.113	22.031	23.409	25.331	4.686	5-MEV	3020.0 R
										TDUS	0.0484

NO.	DATA(CPS)	PEAK(CPS)	FLUX(CM-2)	DOSE(MMR/H)	SUM OF DATA	SUM OF FLUX	SUM OF DOSE
1	-0.764E-03	-0.174E+01	-0.382E-01	-0.118E+00	0.212E+03	0.577E+01	0.632E+01
2	-0.376E-01	-0.178E+01	-0.390E-01	-0.267E-01	0.212E+03	0.581E+01	0.644E+01
3	-0.108E+00	-0.185E+01	-0.406E-01	-0.118E-01	0.212E+03	0.585E+01	0.647E±01
4	-0.223E+00	-0.190E+01	-0.416E-01	-0.707E-02	0.212E+03	0.589E+01	0.648E+01
5	-0.264E+00	-0.184E+01	-0.405E-01	-0.515E-02	0.212E+03	0.593E+01	0.649E+01
6	-0.384E+00	-0.191E+01	-0.418E-01	-0.528E-02	0.213E+03	0.597E+01	0.649E+01
7	-0.477E+00	-0.189E+01	-0.414E-01	-0.520E-02	0.213E+03	0.601E+01	0.650E+01
8	-0.413E+00	-0.177E+01	-0.388E-01	-0.486E-02	0.214E+03	0.605E+01	0.650E+01
9	0.324E+01	0.198E+01	0.435E-01	0.605E-02	0.214E+03	0.609E+01	0.651E±01
10	0.980E+01	0.859E+01	0.188E+00	0.289E-01	0.211E+03	0.605E+01	0.650E+01
10	0.980E+01	0.859E+01	0.188E+00	0.289E-01	0.211E+03	0.605E+01	0.650E+01
20	0.596E+01	0.520E+01	0.127E+00	0.451E-01	0.122E+03	0.425E+01	0.608E+01
30	0.278E+01	0.224E+01	0.655E-01	0.374E-01	0.764E+02	0.322E+01	0.562E+01
40	0.158E+01	0.114E+01	0.424E-01	0.333E-01	0.541E+02	0.266E+01	0.525E+01
50	0.861E+00	0.533E+00	0.246E-01	0.244E-01	0.427E+02	0.236E+01	0.498E+01
60	0.748E+00	0.468E+00	0.261E-01	0.308E-01	0.348E+02	0.211E+01	0.472E+01
70	0.473E+00	0.226E+00	0.147E-01	0.199E-01	0.277E+02	0.185E+01	0.439E+01
80	0.384E+00	0.177E+00	0.131E-01	0.199E-01	0.235E+02	0.171E+01	0.420E+01
90	0.345E+00	0.161E+00	0.134E-01	0.226E-01	0.199E+02	0.159E+01	0.400E+01
100	0.327E+00	0.171E+00	0.158E-01	0.293E-01	0.165E+02	0.144E+01	0.373E+01
110	0.230E+00	0.911E-01	0.916E-02	0.184E-01	0.138E+02	0.132E+01	0.351E+01
120	0.236E+00	0.117E+00	0.129E-01	0.280E-01	0.113E+02	0.120E+01	0.325E+01
130	0.141E+00	0.854E-01	0.103E-01	0.237E-01	0.939E+01	0.109E+01	0.301E+01
140	0.161E+00	0.129E+00	0.167E-01	0.402E-01	0.814E+01	0.984E+00	0.276E+01
150	0.394E+00	0.365E+00	0.509E-01	0.128E+00	0.507E+01	0.611E+00	0.184E±01
160	0.504E-01	0.259E-01	0.383E-02	0.102E-01	0.302E+01	0.358E+00	0.119E+01
170	0.481E-01	0.257E-01	0.405E-02	0.113E-01	0.259E+01	0.328E+00	0.111E+01
180	0.455E-01	0.243E-01	0.407E-02	0.118E-01	0.215E+01	0.292E+00	0.101E+01
190	0.263E-01	0.807E-02	0.143E-02	0.428E-02	0.179E+01	0.265E+00	0.929E+00
200	0.235E-01	0.881E-02	0.164E-02	0.508E-02	0.157E+01	0.255E+00	0.900E+00
210	0.317E-01	0.197E-01	0.383E-02	0.123E-01	0.132E+01	0.232E+00	0.829E+00
220	0.320E-01	0.214E-01	0.439E-02	0.146E-01	0.101E+01	0.194E+00	0.702E+00
230	0.186E-01	0.101E-01	0.215E-02	0.739E-02	0.753E+00	0.161E+00	0.593E+00
240	0.110E-01	0.796E-02	0.177E-02	0.625E-02	0.611E+00	0.144E+00	0.533E+00
250	0.114E-01	0.115E-01	0.265E-02	0.963E-02	0.513E+00	0.124E+00	0.463E+00
260	0.366E-01	0.368E-01	0.884E-02	0.329E-01	0.296E+00	0.727E-01	0.273E+00
270	0.744E-02	0.752E-02	0.188E-02	0.718E-02	0.231E-01	0.575E-02	0.215E-01
280	-0.663E-03	-0.663E-03	-0.172E-03	-0.670E-03	-0.124E-01	-0.332E-02	-0.133E-01
290	-0.862E-03	-0.862E-03	-0.231E-03	-0.923E-03	-0.470E-02	-0.129E-02	-0.526E-02
300	-0.124E-02	-0.124E-02	-0.346E-03	-0.141E-02	-0.124E-02	-0.346E-03	-0.141E-02

Description of Table 2.3

* DATA NO. Data number

The measured data number is expressed as six numbers consisting of ABCDEF. A indicates the data for the year 197A or 198A, and BC is the order of the tape in the year A; thus, ABC indicates the tape number. The data number of the ABC tape is assigned to DEF.

* "B" Meaning of this representation

This indicates that the calculation is performed using the data obtained by subtracting the background specific to the detector, or specific data as the background.

* "C" Meaning of this representation

The background subtraction, in addition to the aforementioned, is a calculation performed by obtaining the contribution of cosmic rays, etc. from the count above 3 MeV (ABOVE COUNT) and using the subtracted result.

* "D" Meaning of this representation

This indicates that the calculation was performed using the raw data without subtracting these backgrounds.

CHANNEL Number of channels

The number of channels used in the measured data; either 1024 or 2048 is mostly used in this program.

KEV/CH Channel width

Indicates the channel width for the numbers shown in NO. Usually, 10 keV/ch is used.

TIME The time of recording

Indicates the time for which the recording is normally performed. The first two digits indicate the hour, and the following digits indicate the minutes containing the decimal point, respectively. In some cases, the recorded data number is set to ADEF.XY, indicating that the data is collected in the number XY by DEF.

LAST-CH Last channel

Indicates the last channel used when data analysis was performed. A channel corresponding to 3 MeV is usually indicated.

LIVE-TIME Measurement time

Indicates the time (s) for which measurement is made using 0 channels of the spectrum. In some cases, this time is added separately and in other cases, a number is used which is obtained by multiplying the number of 0 channels several times (to correct counting losses). In these cases, the 0 channels of the spectrum (the total counting hours) are shown in parentheses.

ABOVE-CNT Excessed count

Total count values ranging from the last channel for which spectrum analysis was performed (LAST CH) to the last channel actually used.

COS-DOSE (μR, H) Cosmic ray dose

This is calculated as cosmic ray dose (ABOVE CNT/LIVE TIME) x K, which is calculated from ABOVE CNT. The value 2.12 is used for K. If a different value is used, it is shown in the last term.

GAMMA (μR, H) Gamma-ray dose

The gamma-ray dose is generally calculated from the spectrum up to 3 MeV.

(CH)

This is the lower limit of the energy region employed in the preceding dose calculation. If this value is 3, it indicates that integration up to 30 keV has been performed. For this parameter, 2 - 5 is usually used. In this example, a value of 10 keV is used.

WEIGHT Weight of the detector

For the weight of the detector, (NaI (Tl)) was used. By predefining this selection, the average absorbed dose rate of the scintillator is obtained from the sum of the product of the counting rate corresponding to each energy of the pulse-height distribution and the energy ($\Sigma_i N_i \times E_i$).

E.CH Channel width

Indicates the channel width of the measured data. The measured spectrum is corrected to 10 keV/ch using an n^{th} order equation and the coefficient of the first order equation obtained in advance is shown.

RESULTS Calculation results

DATA is the counting rate (cps) of 10 keV - 3 MeV.

FLUX is the total flux density (γ/cm^2) of 10 keV - 3 MeV.

DOSE is the exposure dose rate (μR/h), obtained by integrating the values of 10 keV - 3 MeV. ADOS is the average absorbed dose rate (μrad/h) in the scintillator.

PER D

This is obtained by dividing each dose of RESULTS by the value of the exposure dose; however, the DOSE value (this should be 1.00) is obtained by dividing the value of the channel region set to (CH) × 10 keV - 3 MeV by the value of 10 keV - 3 MeV. Therefore, the value is not always 1.00.

0 - 50, 50 - 100, 100 - 250... up to 3000

This value is a percentage between the energy regions of 0 - 50 keV, 50 - 100 keV, 100 - 250 keV, 250 - 500 keV, 500 - 1000 keV, 1000 - 2000 keV, and 2000 - 3000 keV, calculated for each result, respectively.

NO. Channel number

Data is output as the value of each channel number (10 keV/ch) but the content to be output can be selected. Generally, when data is output on one page, the values of 1 - 100 KeV are output for every 10 keV/ch, and the values of 100 keV - 3 MeV are output for every 100 keV.

* DATA (cps)

The counting rate value (cps/10 keV) for each channel (10 keV) is calculated for a specified channel by correcting the measured data to 10 keV/ch using the n^{th} order equation.

* PEAK (cps)

This is the result of subtracting the Compton contribution from DATA (cps). In the case of "B" and "C," a calculation is made by subtracting the background spectrum, which is separately predefined.

* FLUX ($\gamma \bullet cm^{-2} \bullet s^{-1}$)

This is the true photon flux density obtained by dividing the PEAK (cps) result by the peak efficiency of the detector.

* DOSE (μR/h)

The exposure dose rate calculated from the above photon flux density. The organ dose rate, the effective dose rate, etc. can be calculated by multiplying the result by the coefficient determined from the energy absorption coefficient of a material.

SUM OF DATA, FLUX, DOSE

Individual results are integrated from 3 MeV to the specified channel in the low energy region.

Note 1) It is possible to calculate up to five sets of RESULTS for the output, including a comparison with other measured values such as ionization chambers whose characteristics are known as a result of correcting the absorption for a scintillator case, and the values such as the genital gland dose. In this case, the values other than DATA will change.

Note 2) The underlined content is used in the graphical display of the results.
That is, the data number; representation of C, D, B; cosmic ray dose rate C =; and the gamma dose rate G =.

Note 3) Items marked as * are also graphically displayed. Each graph is shown as semilogarithmic, and SUM OF DOSE is shown as a linear graph in percentage. Among these numerical values, the underlined data is made into a successive graph.

Appendix 3 Example of unfolding of NaI (Tl) scintillation spectrum by the method of successive approximation[*1]

There are various methods for converting the pulse-height distribution obtained by a NaI (Tl) scintillation spectrometer into a photon energy spectrum (the response correction). In this document, we describe an example of analyzing the data measured by a 3 in. φ spherical NaI (Tl) scintillator using the method of successive approximation, based on a response matrix.

Appendix 3.1 Outline of the analysis method

(1) Energy calibration

Energy calibration is performed with the aid of a computer to determine the number of channels of the total absorption peaks of ^{208}Tl (2.61 MeV) and ^{40}K (1.46 MeV) among the environmental gamma-ray pulse-height distributions, and by using the obtained two peak channels and their corresponding energies, and the energy typically associated with the "zero point."

(2) Adjustment of the input data

The pulse-height distribution is divided every 20 keV for values below 100 keV, and every 100 keV for values above 100 keV to conform to the response matrix division to obtain a corrected pulse-height distribution.

(3) Subtraction of background components

Cosmic ray components and the detector-specific background are subtracted. Cosmic ray components are obtained by the coincidence method with a plastic scintillation spectrometer. In addition, the detector-specific background is obtained from the measurements taken in an iron chamber.

(4) Response correction

The pulse-height distribution is corrected using a response matrix and converted to an incident gamma-ray spectrum. (The response matrix used at this time is shown in Table 3.1. The details of the method are also given in Appendix 3.2.)

(5) Calculation of dose rates

The energy spectrum is expressed in the roentgen unit, by multiplying the incident gamma-ray spectrum by the energy absorption coefficient of air that corresponds to each energy. In addition, the exposure dose rate at the measurement location can be obtained by integration across the entire energy range (details of this method are given in Appendix 3.3).

Appendix 3.2 Energy analysis by the method of successive approximation

The probability of energy E being absorbed when one photon of energy E_0 is incident on the detector is R (E_0, E). When the energy distribution of the incident photon is X (E_0), the distribution Y (E) of the absorbed energy E can be expressed as follows:

$$\mathrm{Y(E)} = \int_0^\infty R(E_0,\ E) \bullet\ X(E_0)dE \qquad \text{(Attachment 3-1)}$$

[*1] As this is quoted from the Radioactivity Measurement Series No. 20 "Environmental Gamma-ray Spectrometry" (1990), the former unit of “R” is used.

Here, when the incident energy and the absorbed energy are equally divided into n equal parts, E_j (j = 1, 2,..., n) is incident from the equation (Appendix 3-1), and the distribution Y (E_i) of the absorbed energy E_i (i = 1, 2,..., n) can be expressed:

$$\mathrm{Y}(\mathrm{E}_i) = R_{ij} \bullet X(E_j) \bullet \Delta E \qquad \text{(Attachment 3-2)}$$

Further, for a wide range of incident energies E_j (j = 1, 2,..., n), we have:

$$\mathrm{Y}(\mathrm{E}_i) = \sum_{j=i}^{n} R_{ij} \bullet X(E_j) \bullet \Delta E \qquad \text{(Attachment 3-3)}$$

The preceding equations can be used. In the equation (Attachment 3-3), if n = ∞, the equation of (Appendix 3-1) is applied.
X (E_j) (j = 1, 2,..., n) represents the energy spectrum of the radiation incident on the detector, and Y (E_i) (i = 1, 2,..., n) is the pulse-height distribution observed on the detector. R_{ij} represents the response function of the detector relative to the incident radiation of energy E_j.
The equation (Attachment 3-3) can be expressed as follows, using the square matrix R composed of n rows and n columns, and the n-dimensional column vectors X and Y:

$$\mathrm{Y} = \mathrm{R} \bullet \mathrm{X} \qquad \text{(Attachment 3-4)}$$

Next, a pulse-height distribution Y_0 is taken as a first approximation of the spectrum to be obtained.

$$X_1 = Y_0 \qquad \text{(Attachment 3-5)}$$

Y_1 is obtained by multiplying this term by the response matrix R.

$$Y_1 = \mathrm{R} \bullet X_1 \qquad \text{(Attachment 3-6)}$$

Next, for X_2 of the second approximation, the ratio of the corresponding elements of X_1 and Y_1 is multiplied by the corresponding element of Y_0. Thus, the first element of X_2 is given by:

$$(X_2)_1 = \frac{(X_1)_1}{(Y_1)_1} \bullet (Y_0)_1 \qquad \text{(Attachment 3-7)}$$

is obtained from the preceding equation. Y_2 is then obtained by further multiplying X_2 by R.

$$Y_2 = \mathrm{R} \bullet X_2 \qquad \text{(Attachment 3-8)}$$

Similarly:

$$(X_3)_1 = \frac{(X_2)_1}{(Y_2)_1} \bullet (Y_0)_1 \qquad \text{(Attachment 3-9)}$$

If such procedures are repeated 50 times, a value converged within the range of up to 10^{-1} is obtained.

Appendix 3.3 Dose conversion

Response correction is performed by the successive approximation method, and the dose conversion from the incident gamma-ray spectrum is performed using the following equation:

$$\mathrm{D} = 1.724 \times \frac{\Sigma_i (N_i \bullet E_i \bullet \mu_i) \bullet \Delta E}{\pi r^2 \bullet W \bullet T} (\mu R/h) \quad \text{(Attachment 3-10)}$$

where,

E_i : the center energy of the i^{th} histogram (eV)

W : the W value of air (33.85 eV/ion pair)

r : the radius of the scintillator (3.81 cm)

μ_i : the linear attenuation coefficient of air (cm^{-1}) relative to the photon of energy E_i

N_i : the number of incident photons/□ E (items/(100 keV • s)) entering the i^{th} histogram

T : measurement time (s)

Table 3.1 Response matrix of a 3 in. φ spherical NaI (Tl) scintillation detector

the central energy of a photon (MeV)

the central energy of a photon (MeV)

	0.01	0.03	0.05	0.07	0.09	0.15	0.25	0.35	0.45	0.55	0.65	0.75	0.85	0.95	1.05	1.15	1.25	1.35	1.45	1.55	1.65	1.75	1.85	1.95	2.05	2.15	2.25	2.35	2.45	2.55	2.65	2.75	2.85	2.95
0.01	1.000																																	
0.03	0.029	0.768	0.012																															
0.05	0.008	0.112	0.787	0.018																														
0.07	0.008	0.110	0.106	0.765	0.013																													
0.09	0.002	0.013	0.107	0.11	0.760	0.017																												
0.15	0.022	0.016	0.012	0.009	0.041	0.838																												
0.25	0.041	0.031	0.027	0.020	0.023	0.058	0.754																											
0.35	0.036	0.033	0.031	0.028	0.026	0.126	0.018	0.578																										
0.45	0.028	0.027	0.027	0.026	[illegible]	0.136	0.105	0.005	0.463																									
0.55	0.019	0.019	0.018	0.018	0.018	0.091	0.039	0.083	0.006	0.397																								
0.65	0.016	0.015	0.015	0.015	0.015	0.076	0.076	0.030	0.072	0.038	0.041	0.032																						
0.75	0.013	0.013	0.012	0.012	0.012	0.062	0.062	0.069	0.081	0.059	0.006	0.233	0.033																					
0.85	0.011	0.011	0.011	0.010	0.016	0.053	0.053	0.057	0.061	0.072	0.050	0.010	0.255	0.005																				
0.95	0.010	0.010	0.009	0.009	0.009	0.017	0.017	0.017	0.019	0.050	0.065	0.045	0.012	0.221	0.008																			
1.05	0.008	0.003	0.008	0.008	0.008	0.010	0.011	0.012	0.013	0.015	0.064	0.038	0.037	0.012	0.200	0.007																		
1.15	0.007	0.007	0.007	0.007	0.008	0.036	0.035	0.037	0.039	0.010	0.011	0.061	0.063	0.031	0.013	0.183	0.008																	
1.25	0.005	0.005	0.006	0.006	0.006	0.028	0.031	0.031	0.031	0.031	0.036	0.059	0.013	0.062	0.027	0.037	0.163	0.010																
1.35	0.005	0.005	0.005	0.005	0.005	0.025	0.026	0.027	0.029	0.010	0.030	0.035	0.039	0.045	0.051	0.024	0.035	[illegible]	0.019															
1.45	0.004	0.004	0.005	0.005	0.005	0.023	0.024	0.024	0.024	0.027	0.028	0.029	0.032	0.037	0.041	0.050	0.022	0.031	0.120	0.017														
1.55	0.004	0.004	0.004	0.005	0.005	0.022	0.022	0.023	0.023	0.024	0.024	0.028	0.028	0.032	0.036	0.040	0.048	0.022	0.027	0.118	0.015													
1.65	0.004	0.004	0.004	0.004	0.004	0.020	0.020	0.020	0.020	0.021	0.024	0.021	0.025	0.026	0.030	0.038	0.040	0.017	0.024	0.023	0.113	0.014												
1.75	0.003	0.003	0.003	0.004	0.004	0.017	0.018	0.018	0.018	0.018	0.019	0.023	0.020	0.021	0.025	0.028	0.037	0.030	0.041	0.021	0.022	0.106	0.013											
1.85	0.003	0.003	0.003	0.003	0.004	0.016	0.016	0.016	0.016	0.016	0.016	0.018	0.022	0.021	0.022	0.023	0.027	0.038	0.031	0.014	0.025	0.023	0.098	0.014										
1.95	0.002	0.002	0.002	0.003	0.003	0.012	0.012	0.013	0.013	0.014	0.016	0.016	0.019	0.025	0.022	0.023	0.024	0.026	0.039	0.036	0.041	0.024	0.026	0.090	0.014									
2.05	0.002	0.002	0.002	0.002	0.003	0.011	0.011	0.012	0.012	0.013	0.013	0.014	0.016	0.018	0.023	0.022	0.022	0.023	0.027	0.040	0.036	0.038	0.023	0.029	0.083	0.016								
2.15	0.002	0.002	0.002	0.002	0.003	0.011	0.011	0.012	0.012	0.012	0.013	0.013	0.014	0.014	0.018	0.022	0.021	0.021	0.022	0.026	0.039	0.031	0.034	0.021	0.034	0.082	0.014							
2.25	0.002	0.002	0.002	0.002	0.002	0.011	0.011	0.012	0.012	0.012	0.012	0.013	0.013	0.013	0.014	0.017	0.022	0.020	0.020	0.022	0.026	0.038	0.033	0.032	0.021	0.027	0.077	0.014						
2.35	0.002	0.002	0.002	0.002	0.002	0.011	0.011	0.012	0.012	0.012	0.012	0.012	0.013	0.013	0.013	0.013	0.017	0.023	0.020	0.020	0.021	0.026	0.087	0.032	0.030	0.020	0.025	0.078	0.012					
2.45	0.002	0.002	0.002	0.002	0.002	0.010	0.010	0.010	0.011	0.011	0.011	0.011	0.011	0.013	0.013	0.013	0.013	0.015	0.02	0.020	0.018	0.021	0.025	0.036	0.030	0.029	0.020	0.026	0.171	0.012				
2.55	0.002	0.002	0.002	0.002	0.002	0.010	0.010	0.010	0.010	0.010	0.010	0.010	0.011	0.011	0.011	0.011	0.011	0.012	0.01	0.023	0.019	0.019	0.022	0.026	0.037	0.031	0.029	0.021	0.025	0.069	0.017			
2.65	0.002	0.002	0.002	0.002	0.002	0.010	0.010	0.011	0.011	0.011	0.011	0.011	0.011	0.011	0.011	0.011	0.011	0.011	0.01	0.016	0.023	0.019	0.018	0.020	0.026	0.037	0.031	0.028	0.020	0.021	0.015	0.021		
2.75	0.001	0.002	0.002	0.002	0.002	0.009	0.009	0.009	0.009	0.009	0.009	0.009	0.009	0.009	0.009	0.009	0.009	0.010	0.01	0.015	0.024	0.019	0.019	0.021	0.021	0.026	0.039	0.031	0.129	0.020	0.022	0.041	0.022	
2.85	0.001	0.002	0.002	0.002	0.002	0.009	0.009	0.009	0.009	0.009	0.009	0.009	0.009	0.009	0.009	0.009	0.009	0.009	0.011	0.011	0.015	0.024	0.019	0.017	0.017	0.021	0.026	0.038	0.131	0.027	0.020	0.023	0.039	0.021
2.95	0.001	0.001	0.002	0.002	0.002	0.008	0.008	0.008	0.008	0.008	0.008	0.008	0.008	0.008	0.010	0.010	0.010	0.011	0.01	0.011	0.011	0.016	0.025	0.019	0.019	0.019	0.023	0.028	0.142	0.032	0.030	0.020	0.022	0.038

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