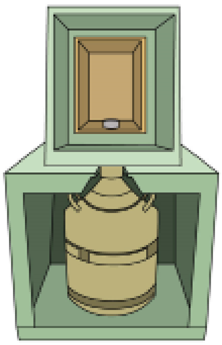
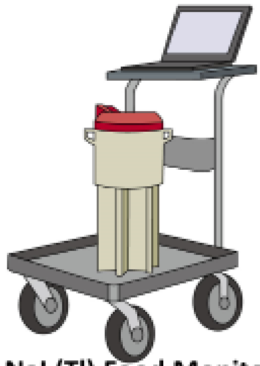
Dose Measurement and Calculation

Various Measuring Instruments

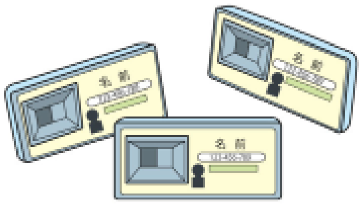


Ge Semiconductor Detector
Used to measure radioactivity in foods or soil; Effective in measuring low levels of radioactivity concentrations

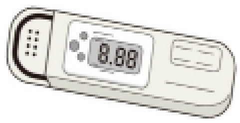


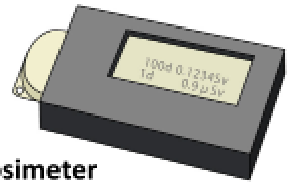
NaI (Tl) Food Monitor
Suitable for efficient radioactivity measurement of foods, etc.

Whole-body Counter
Assess accumulation of γ-ray nuclides in the body using numerous scintillation counters or the like



Integrating Personal Dosimeter
Worn on the trunk of the body for 1-3 months to measure cumulative exposure doses during that period





Electronic Personal Dosimeter
Equipped with a device to display dose rates or cumulative doses during a certain period of time and thus convenient for measuring and managing exposure doses of temporary visitors to radiation handling facilities

While radiation is not visible to the human eye, it is known to cause ionization and excitation (p.45 of Vol. 1, “Principles of Radiation Measurement”), and a variety of measuring instruments using these effects have been invented for different purposes and applications. The measuring instruments shown above all utilize the excitation effect.

To measure radioactivity concentrations in foods and soil, measuring instruments wherein a germanium detector (Ge detector) or a NaI (Tl) detector that can measure γ-ray spectra is installed in a lead shield are used. Ge detectors are excellent in γ-ray energy resolution and suitable for determining traces of radioactive materials. NaI (Tl) detectors are not as excellent as Ge detectors in terms of energy resolution but are easy to handle and have relatively high detection efficiency, so they are widely used in food inspection.

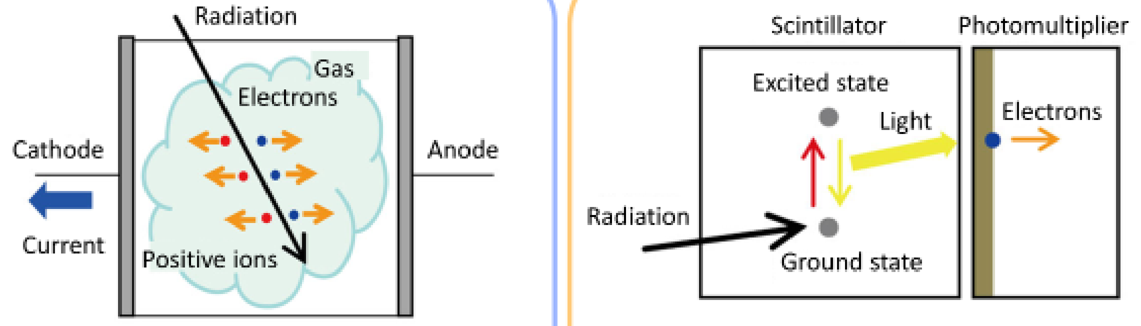
Also commercially available are whole-body counters that use numerous scintillation counters or Ge detectors worn on the body to assess accumulation of γ-ray nuclides in the body, as well as integrating personal dosimeters and electronic personal dosimeters for managing personal exposure. In particular, after the accident at Tokyo Electric Power Company (TEPCO)’s Fukushima Daiichi NPS, a variety of electronic personal dosimeters have been invented to allow easy monitoring of information on exposure at certain time intervals.

(Related to p.60 of Vol. 1, “Instruments for Measuring Internal Exposure”)

Included in this reference material on March 31, 2013
Updated on March 31, 2017

Principles of Radiation Measurement

Measurements are carried out utilizing the interaction between radiation and substances.



Ionization (with gas atoms)

- Detectors are filled with gases such as inert gases or air.
- When radiation passes through gas, molecules are ionized, creating positive ions and electrons.
- Positive ions and electrons are drawn toward the electrodes and are converted into electric signals for measurement.

GM counter survey meters, ionization chambers, etc.

Excitation

- When radiation passes through a scintillator, molecules are excited, but they return to their original state (ground state).
- Light emitted in the process is amplified and converted into a current for measurement.

NaI (Tl) scintillation survey meter, etc.

Radiation is known to interact with substances when passing through them. The amount of radiation can be measured utilizing the interaction between radiation and substances.

Geiger Muller (GM) counter survey meters and ionization chambers utilize the ionization between radiation and gas atoms. Ionization effect refers to the process in which radiation ejects electrons from nuclei in a substance. Detectors of GM counter survey meters and ionization chambers are filled with gases. When radiation passes inside a detector, it causes ionization of gas atoms, separating atoms into positive ions and electrons. Separated electrons and positive ions are attracted to the electrodes, causing a current to flow. This is converted into electric signals, which are then measured as the amount of radiation.

NaI (Tl) scintillation survey meters utilize excitation with substances. Radiation gives energy to electrons of nuclei, and when an electron jumps to an outer orbit, this phenomenon is called excitation. An atom in that state is unstable (excited), and when it returns to a stable state (ground state), it gives off energy in the form of light. This is called the excitation effect. A scintillator is a substance that emits light in response to incident radiation. Weak light emitted from a scintillator is amplified using a photomultiplier and is converted into an electric signal to measure radiation. Aside from NaI (Tl) scintillation survey meters, germanium semiconductor detectors also utilize the excitation effect for radiation measurement.

(Related to p.18 of Vol. 1, "Ionization of Radiation - Property of Ionizing Radiation")

Included in this reference material on March 31, 2017

Dose Measurement and Calculation

Detection Limit

ND: Abbreviation of "Not Detected"

ND = The measured value was less than the detection limit.

✖ **The measured value was zero.**

The measurement result "ND" means that the measured value was less than the detection limit.

Detection limits vary depending on measurement time and the sample amount.
Detection limits are set by each measurement laboratory in accordance with the purpose of the measurement.

- **The longer the measurement time is, the lower the detection limit becomes.**

The measurement time is increased by X times.
→ The detection limit becomes $\frac{1}{\sqrt{X}}$ times.

Example 1: When the measurement time is doubled, the detection limit becomes $\frac{1}{\sqrt{2}}$ times.

Example 2: In order to lower the detection limit from 60 Bq/kg to 30 Bq/kg, the measurement time needs to be increased by four times.

- **The larger the sample amount is, the lower the detection limit becomes.**

Example: If the detection limit is 200 Bq/kg when the sample amount is 0.2 kg, increasing the sample amount to 1 kg leads to lowering the detection limit to 40 Bq/kg.

Prepared based on the "Analysis of Radioactive Materials" (December 2011) by Ministry of Agriculture, Forestry and Fisheries
https://www.maff.go.jp/j/syouan/seisaku/data_reliance/maff_torikumi/pdf/rad_kensyu.pdf (in Japanese)

Results of the measurement of radioactivity or dose rates are sometimes indicated as “Not Detected (ND).”

This does not mean that there is no radioactive material but means that the measured concentration of radioactive materials is below the measurable detection limit.

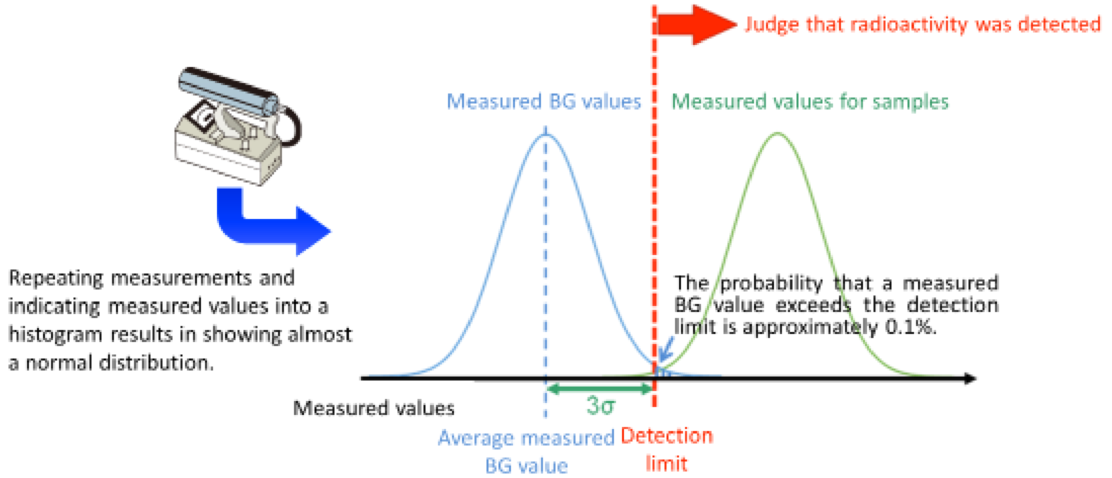
Detection limits vary depending on the measurement time and the sample amount, and in general, the longer the measurement time is and the larger the sample amount is, the lower the detection limit becomes. When setting a detection limit lower, even a small amount of radioactive materials can be detected, but required time and cost are larger and this may lead to a decrease in the number of samples to be tested. Accordingly, detection limits are set by individual analytical laboratories in accordance with the purpose of the measurement.

Included in this reference material on March 31, 2019
Updated on March 31, 2021

Dose Measurement and Calculation

Ideas on Detection Limits (3σ Method)

- Even a minor change in measurement conditions can influence measurement results and there is also background (BG) radioactivity derived not from samples themselves. Therefore, due consideration is required when setting a detection limit in order to assure statistical reliability.
- One of the representative ideas on detection limits is the 3σ method. Under the 3σ method, a detection limit is defined as a value obtained by adding three times sigma of the measured background values to their average. When a measurement result exceeds this value, it is judged that signals (radioactivity, a dose rate, etc.) from a sample are detected.



When measuring background radioactivity or dose rates using a survey meter or other equipment, even a minor change in measurement conditions can influence measurement results. Therefore, it is necessary to repeat measurements in order to obtain reliable measurement results.

Indicating values obtained through repeated measurements into a histogram results in showing a normal distribution. The minimum amount of radioactivity that can be detected as a statistically significant value under the condition of fluctuating background dose rates is referred to as a detection limit (or lower limit).

Under the 3σ method, one of the representative ideas on detection limits, a detection limit is defined as a value obtained by adding three times sigma to the average of the measured background values. This is because when the measured value is larger than 3σ, the probability of BG measurements that exceed the detection limit by fluctuation is approximately 0.1%.

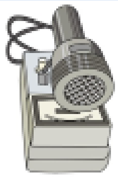
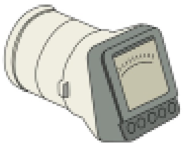
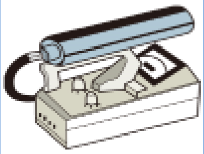
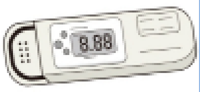
In addition to the 3σ method, there is the Currie method. Under this method, a lower detection limit is set in consideration of the fluctuation of sample measurements so as to reduce the probability of a "false negative," where measurements close to but above the detection limit are judged as Not Detected (ND).

Reference

- "Practical handbook for γ-ray measurement," authored by Gordon Gilmore and John D. Hemingway, translated into Japanese by Yonezawa Nakashiro, et al., NIKKAN KOGYO SHIMBUN, LTD. (2002)
- "Ideas on detection limits and minimum limits of determination," by Uemoto Michihisa, Bunseki 2010 5, 216-221 (2010)

Included in this reference material on March 31, 2019
Updated on March 31, 2022

Instruments for Measuring External Exposure

Type		Purpose	
GM counter survey meter (ionization)		Contamination detection	Has a thin entrance window and can detect β-particles efficiently; Suitable for detecting surface contamination
Ionization chamber survey meter (ionization)		γ-ray ambient dose rate	Accurate but unable to measure low dose rates like a scintillation type can
NaI (Tl) scintillation survey meter (excitation)		γ-ray ambient dose rate	Accurate and very sensitive; Suitable for measuring γ-ray ambient dose rates from the environment level up to around 10μSv/h
Personal dosimeter (light-stimulated luminescence dosimeter, luminescent glass dosimeter, electronic dosimeter, etc.) (excitation)		Personal dose Cumulative dose	Worn on the trunk of the body to measure personal dose equivalent of the relevant person's exposure while it is worn; A direct-reading type and types with alarm functions are also available.

Survey meters are either for inspecting body surface contamination or for measuring ambient dose rates. Geiger Muller (GM) tube-type survey meters are highly sensitive to β-particles and are thus suitable for inspecting body surface contamination. They are relatively affordable and useful in locating contamination and confirming the effects of decontamination.

Ionization chambers are most suited for measuring high-level ambient dose rates but cannot measure very low dose rates. Therefore, a scintillation type is most suited for measuring ambient dose rates in the general environment.

NaI (Tl) scintillation survey meters can also measure the radioactivity intensity, but measurement results vary depending on the level of radiation at the measuring location and the way of measurement. Since calibration at a facility with a radioactive source that serves as a reference is required before converting the measurement results into becquerels, expert assistance is required to implement the measurements.

Personal dosimeters provide cumulative exposure dose readings. An electronic direct-reading type allows a person to confirm the degree of exposure at certain time intervals or after every operation.

Included in this reference material on March 31, 2013
Updated on March 31, 2019

Dose Measurement and Calculation

Methods of Measuring Doses

Example: NaI (Tl) scintillation survey meter (TCS-171)

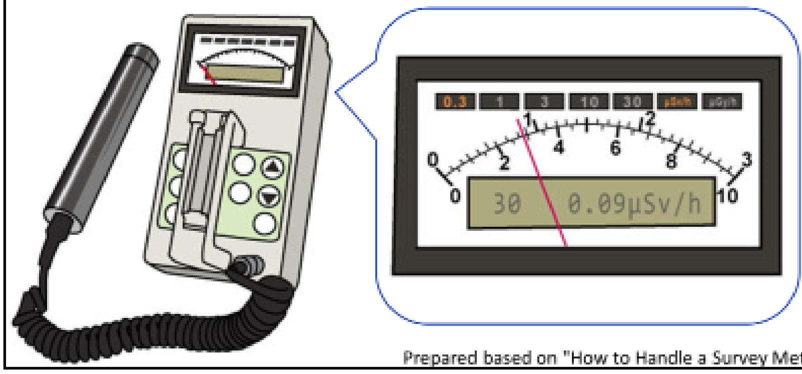
(i) Background measurement

(ii) Field measurement

- Range (the reading is indicated near the center of the scale)
- Adjustment of time constant (the value is to be read when a period of time three times the time constant elapses)

(iii) Dose calculation

- Reading × Calibration constant = Dose (µSv/h)



How to interpret the readings

0.3, 3, 30 µSv/h in the upper row
1, 10 µSv/h in the lower row

- The photo shows a range of 0.3 µSv/h.
- Read the value in the upper row
- The needle pointing at 0.92

The reading at 0.092 µSv/h

For example, when the calibration constant is 0.95
Dose = 0.092×0.95 = 0.087 µSv/h

Prepared based on "How to Handle a Survey Meter" on the website of the Prime Minister's Office

A method of measuring γ-ray ambient dose rate using a NaI scintillation survey meter is shown as an example of a method of measuring doses.

Before measurement, the device is checked for soundness (appearance, power supply, high voltage) and then background is measured (set a range at 0.3µSv/h and a time constant at 30 sec). Normally, the background value is around 0.1µSv/h.

Field measurements are normally carried out at a height of about 1 m above the ground. The counting range is adjusted so that the meter readings come near the center of the scale. The time constant is adjusted according to the purpose of measurement. For measurements in a rough, wide range or of high doses, the time constant is lowered. To make accurate measurements or to measure low doses, the time constant is increased. After a period of time about three times the time constant has elapsed since the start of a field measurement, the average of the readings is read (for example, the value is read after the lapse of 90 seconds when the time constant is 30 sec.).

The dose equivalent rate (µSv/h) can be obtained by multiplying the reading by the calibration constant that is preset for each measurement condition.

When using measuring instruments, precautions should be taken such as checking whether they operate properly before use, handling them carefully because they are precision instruments, covering measuring instruments with polyethylene sheets during rain or when making measurements in highly contaminated areas, etc.

Included in this reference material on March 31, 2017
Updated on March 31, 2019

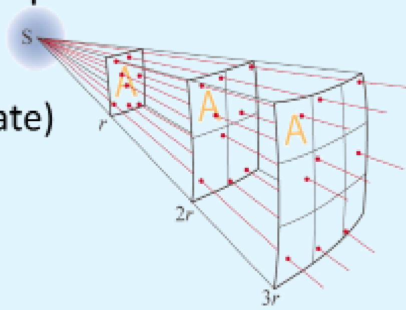
Dose Measurement and Calculation

Characteristics of External Exposure Doses

1) **Distance**: Dose rates are inversely proportional to the distance squared.

$$I = \frac{k}{r^2}$$

I : Radiation intensity (dose rate)
r : Distance
k : Constant



2) **Time**: Doses are proportional to the time of exposure provided the dose rates are the same.

(Total) dose (microsieverts) = Dose rate (microsieverts/h) × Time

The intensity of radiation (dose rate) is strong (large) when the source of radiation (radiation source) is close, and it gets weaker (smaller) as the distance increases, even if the amount of radioactive materials remains the same. When the radioactive materials are located only in one place (point source), the dose rate becomes smaller in inverse proportion to the distance squared. Dose rates also decrease due to atmospheric influence, etc.

When radioactive materials are evenly distributed on a broad plain surface, the formula to express the relationship between the distance and the dose rate is rather complicated, but as in the case of a point source, the higher it is from the ground surface, the lower the dose rate is. However, radioactive materials are not evenly distributed in reality and a plain surface is not necessarily smooth, and also owing to attenuation of radiation in the air or other reasons, the dose rate does not always match the value obtained from the relational expression.

Calculation of external exposure doses is based not on the radioactivity intensity (becquerels) but on the amount of radiation (grays or sieverts) the human body is exposed to.

If the dose rate is constant, the total exposure dose can be calculated by multiplying the dose rate by the time of exposure to radiation.

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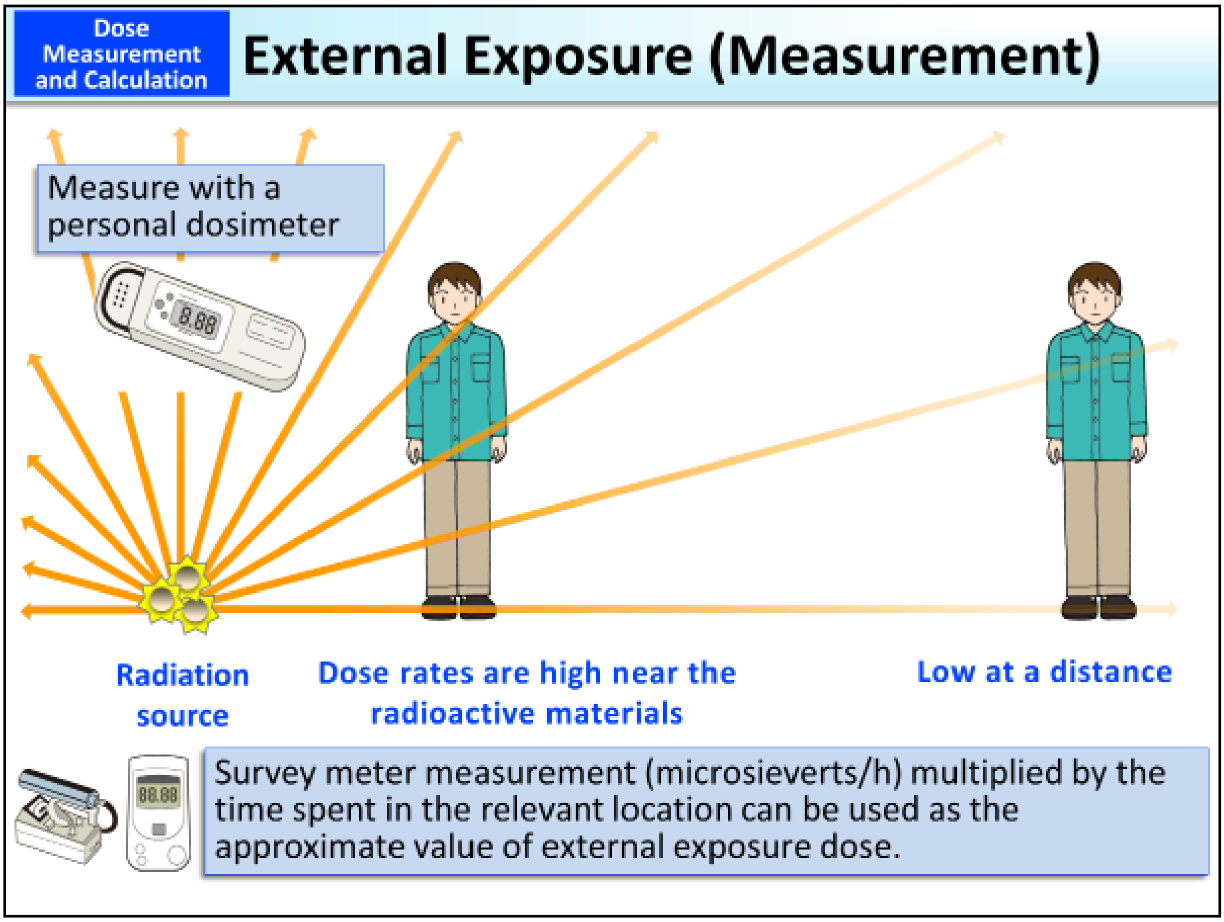


One of the means to measure doses due to external exposure is to wear a personal dosimeter on the body. Personal dosimeters can measure cumulative amounts of radiation exposure for a certain period of time, and provide dose rate readings.

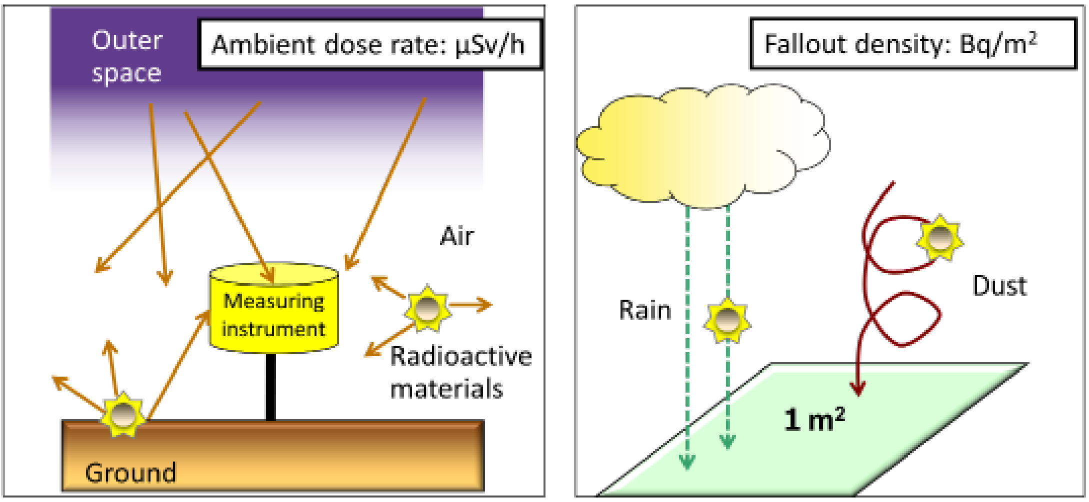
Another means is to measure radiation dose rates in a workplace with a survey meter to estimate the level of exposure supposing that a person stays in that place. Since α-particles and β-particles from outside the body do not reach into the body (p.22 of Vol. 1, "Penetrating Power and Range of Effects on the Human Body"), γ-rays are measured to obtain doses due to external exposure. Many recent instruments provide readings in microsieverts per hour (μSv/h), so such readings are multiplied by the time a person spent in a certain location to roughly calculate his/her external exposure dose. However, these measurements must be made with an instrument, such as a NaI (Tl) scintillation survey meter, that has proper performance and is well calibrated.

Included in this reference material on March 31, 2013
Updated on March 31, 2021

Dose Measurement and Calculation

Measurement of Environmental Radiation and Radioactivity

- Ambient dose rate shows measured amount of γ-rays in the air.
  Indicated in microsieverts per hour (μSv/h)
- Fallout density is the amount of radioactive materials that have deposited (or descended) per unit area in a certain period of time.
  e.g., becquerels per squared meter (Bq/m^2)



The ambient dose rate is obtained by measuring γ-ray doses in the air, and is indicated in microsieverts per hour. γ-rays from radioactive materials suspended in the air and γ-rays from radioactive materials fallen on the ground are both detected. The measured value is not limited to the amount of radiation derived from accidents. Major natural radiation is that from the ground and cosmic rays.

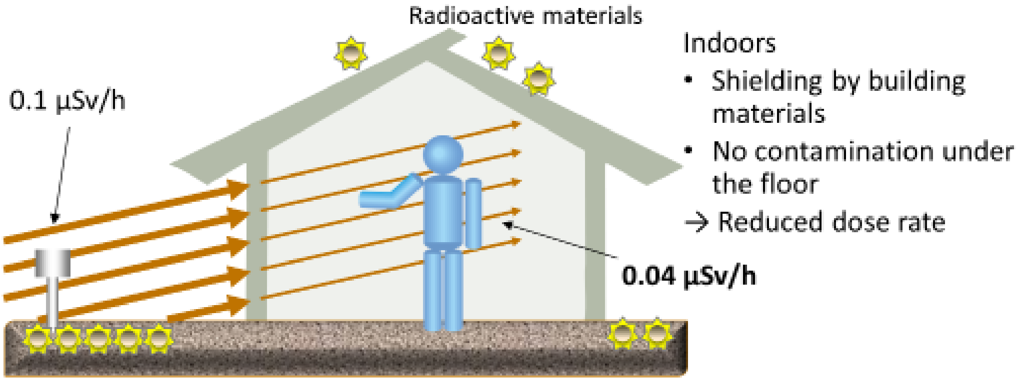
Normally, a measuring instrument is placed at a height of about 1 m from the ground, because most important internal organs are located at this height in the case of an adult. There are cases where a measurement instrument is placed at a height of 50 cm from the ground in places where mainly children spend time, such as schools and kindergartens.

The amount of radioactivity in fallout is expressed as the amount of radioactive materials fallen per unit area. Generally, such amount is expressed as a numerical value per day or month for each kind of radioactive material.

Included in this reference material on March 31, 2013
Updated on March 31, 2019

Dose Measurement and Calculation

Shielding and Reduction Coefficient



Location	Reduction coefficient*
Wooden house (one or two stories)	0.4
Block or brick house (one or two stories)	0.2
The first and second floors of a building (three or four stories) with each floor 450-900m^2 wide	0.05
Upper floors of a building with each floor 900m^2 or wider	0.01

* The ratio of doses in a building when assuming that a dose outdoors at a sufficient distance from the building is 1

Source: Prepared based on the "Disaster Prevention Countermeasures for Nuclear Facilities, etc." (June 1980 (partly revised in August 2010)), Nuclear Safety Commission

In the absence of an appropriate survey meter for measuring ambient dose rates (p.48 of Vol. 1, "Instruments for Measuring External Exposure"), calculations can be made based on the ambient dose rates that the government or local municipalities issued. For the amount of exposure outdoors, measurement results obtained near the relevant building are used. To calculate doses indoors, the indoor ambient dose rate is estimated by multiplying the value of nearby outdoor dose rate by a reduction coefficient.

Reduction coefficients, which take into consideration the effect of shielding by the building and the fact that there is no contamination under the floor, vary depending on the types of buildings and whether radioactive materials are suspended or deposited. When radioactive materials are deposited on soil or a building, in the case of a wooden house, for example, radiation from outside is blocked and the total amount of radiation indoors is reduced to around 40% of the initial amount outdoors. Houses made of blocks, bricks or reinforced concrete have higher shielding effects and radiation levels inside are lower than in wooden houses.

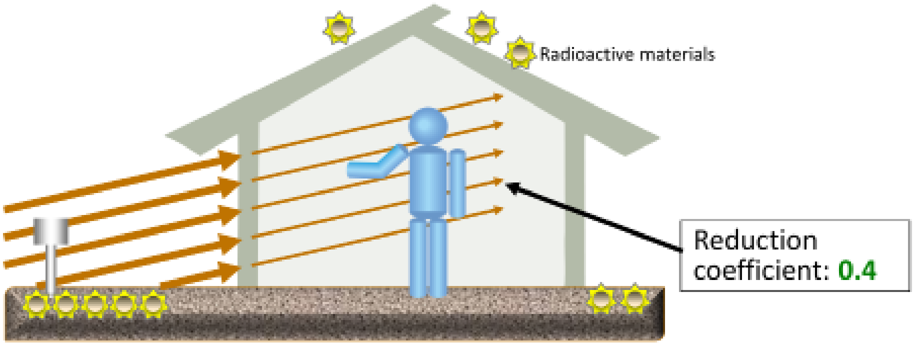
Included in this reference material on March 31, 2013
Updated on March 31, 2019

Dose Measurement and Calculation

Additional Exposure Doses after an Accident (Example of Calculation)

It is important to subtract values in normal times.

Dose rate (increase due to an accident: µSv/h)
Actual measurement - Value in normal times
= Exposure rate at the time of the accident



Exposure rate at the time of the accident
× Number of hours spent outdoors in a day
+
Exposure rate at the time of the accident × 0.4
× Number of hours spent indoors in a day

Daily exposure dose

× 365 days = Annual additional exposure dose

The ambient dose rate measured with a survey meter includes γ-rays from natural environment. To calculate the amount of radiation released due to the accident at Tokyo Electric Power Company (TEPCO)'s Fukushima Daiichi NPS alone, the values measured before the accident (background values) must be subtracted from the currently measured ambient dose rates to ascertain the increase caused by the accident. The values before the accident are available on the website, "Environmental Radioactivity and Radiation in Japan" (https://www.kankyo-hoshano.go.jp/data/database/, in Japanese).

The value obtained by multiplying the increased outdoor and indoor dose rates by the time spent indoors and outdoors is an approximate increase in exposure dose compared with normal times (additional exposure dose).

The calculation example for obtaining a daily additional exposure dose after the accident is made under the assumption that a person stays outdoors for eight hours and stays in a typical Japanese house with a reduction coefficient of 0.4 for 16 hours. Furthermore, an annual additional exposure dose is estimated by multiplying the daily additional exposure dose by 365, the number of days in a year.

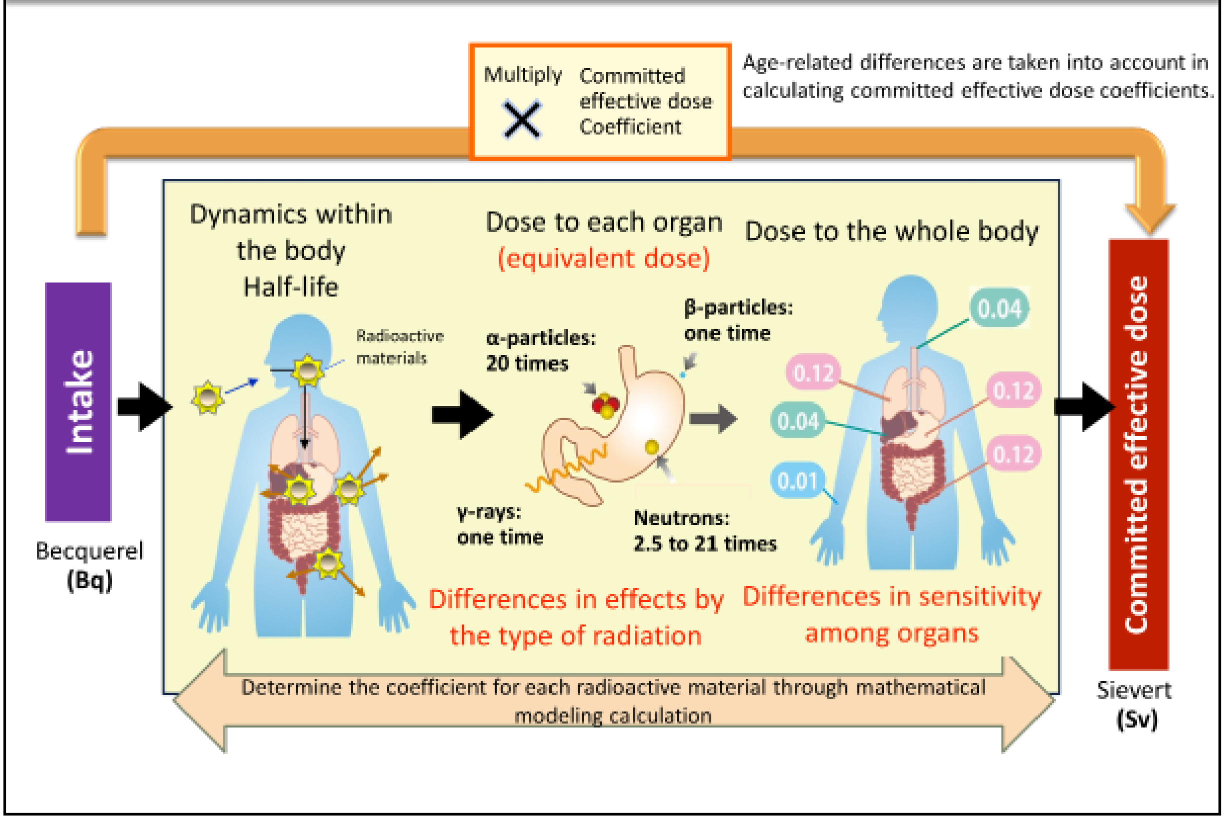
The value of 0.23 µSv/h, which was adopted as the reference value in designating Intensive Contamination Survey Areas where mainly municipalities conduct decontamination after the accident, is derived from the annual additional exposure dose of 1 mSv (hourly exposure dose of 0.19 µSv, which becomes 1 mSv in annualized terms under the same assumption on the safe side as applied in the above calculation example, plus the exposure dose due to natural radiation of 0.04 µSv).

This calculation example is a simplified estimation method provided under the conservative assumption in the response to the accident at TEPCO's Fukushima Daiichi NPS. Therefore, it is considered that the actual external exposure dose of an individual in real life may be lower than the calculation result.

Included in this reference material on March 31, 2013
Updated on March 31, 2022

Dose Measurement and Calculation

Calculation of Internal Exposure Doses



Methods of obtaining effective doses due to internal exposure are essentially the same as for external exposure. However, how to calculate absorbed doses for respective organs and tissues is different.

The part of the body where radioactive materials accumulate varies by their types. Even the same type of radioactive material differs in the behavior within the body, such as metabolism and accumulation, depending on whether they enter the body via the respiratory organs through inhalation or via the digestive tract together with foods and drinks. Moreover, how long radioactive materials will remain in the body varies depending on whether the person is an adult, a child, or an infant.

Mathematical model calculation is performed for each of these different conditions to determine the relationship between the intake of radioactive materials and the absorbed dose of each organ and tissue. Then, differences in sensitivity by types of radiation and among different organs are taken into account in the same manner as for calculation of external exposure doses. An internal exposure dose calculated in this way is called a committed effective dose (in sieverts) (p.56 of Vol. 1, “Committed Effective Doses”).

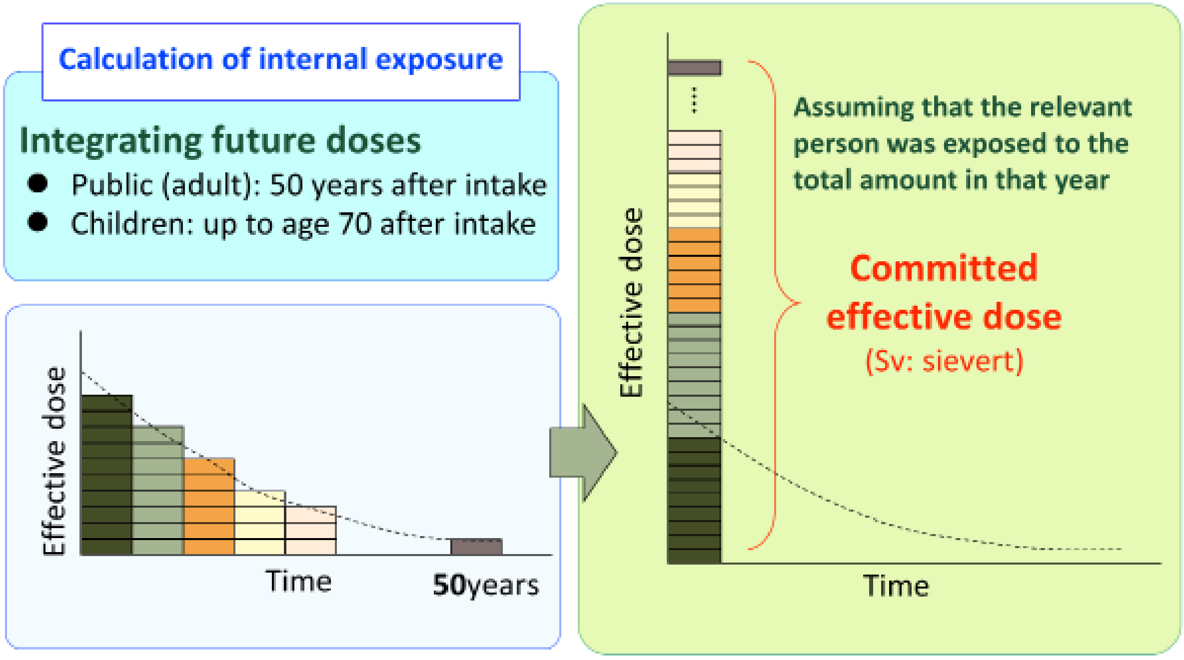
Specifically, internal exposure doses can be obtained by multiplying intake (in becquerels) by a committed effective dose coefficient. Committed effective dose coefficients are defined in detail for each type of radionuclide and age group (p.57 of Vol. 1, “Conversion Factors to Effective Doses”).

Included in this reference material on March 31, 2013
Updated on March 31, 2019

Dose Measurement and Calculation

Committed Effective Doses

Exposure dose estimating how much radiation a person will be exposed to in lifetime from a single intake of radioactive materials



Radioactive materials remain in the body for a certain period of time after being taken into the body. In the meantime, the body will be continuously exposed to radiation. Thus, the total amount of radiation that a person will be exposed to into the future is calculated as dose due to internal exposure based on a single intake of radioactive materials. This is called a committed dose (in sieverts).

Any radioactive materials taken into the body will decrease over time. One contributing factor is the decay of the radioactive materials. Another is excretion as urine and feces. The rate of excretion from the body varies according to the types of elements, their chemical forms, and the age of the person. With these differences taken into account, the cumulative amount of radiation that the human body will receive in a lifetime from radioactive materials is assumed as the amount received in the year of the intake, and a committed dose is calculated.

In particular, the lifetime cumulative dose based on effective dose is called "committed effective dose." The lifetime here is 50 years for adults, and for children it is the number of years up to reaching age 70. In the case of radioactive cesium, which is discharged out of the body at a fast rate (Cesium-134 and Cesium-137 have effective half-lives of 64 days and 70 days, respectively) (p.31 of Vol. 1, "Radioactive Materials Derived from Nuclear Accidents"), most of the committed dose is considered to be received within 2 to 3 years after its intake.

Included in this reference material on March 31, 2013
Updated on March 31, 2015

Conversion Factors to Effective Doses

Committed effective dose coefficients (μSv/Bq) (ingestion)

	Strontium-90	Iodine-131	Cesium-134	Cesium-137	Plutonium-239	Tritium*
Three months old	0.23	0.18	0.026	0.021	4.2	0.000064
One year old	0.073	0.18	0.016	0.012	0.42	0.000048
Five years old	0.047	0.10	0.013	0.0096	0.33	0.000031
Ten years old	0.06	0.052	0.014	0.01	0.27	0.000023
Fifteen years old	0.08	0.034	0.019	0.013	0.24	0.000018
Adult	0.028	0.022	0.019	0.013	0.25	0.000018

μSv/Bq: microsieverts/becquerel

*Tissue free water tritium

Source: Prepared based on the ICRP Publication 119, Compendium of Dose Coefficients based on ICRP Publication 60, 2012, International Commission on Radiological Protection (ICRP)

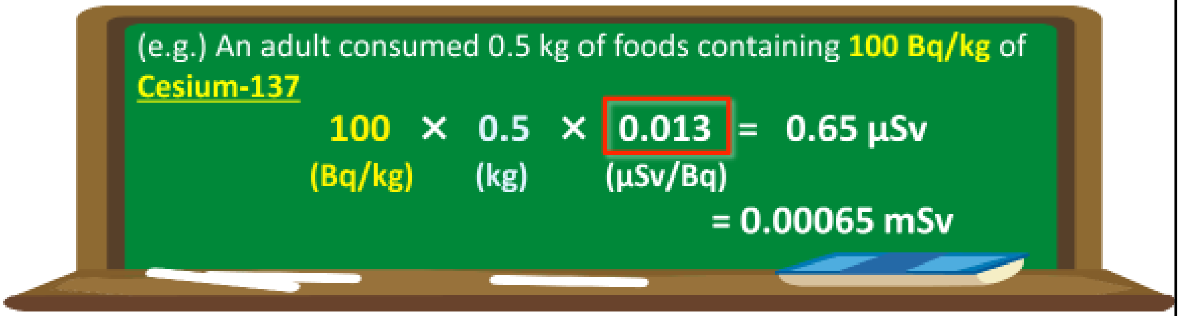
For dose assessment for internal exposure, doses are calculated by estimating an intake for each nuclide and chemical form and multiplying estimated intakes by dose coefficients. Dose coefficients are committed equivalent doses or committed effective doses for an intake of 1 Bq and a specific value has been given for each nuclide, chemical form, intake route (ingestion or inhalation), and for each age group by the ICRP.

The commitment period, i.e., the period during which doses are accumulated, is 50 years for adults and the number of years up to reaching age 70 after intake for children.

Included in this reference material on March 31, 2013
Updated on February 28, 2018

Dose Measurement and Calculation

Exposure Doses from Foods (Example of Calculation)



Committed effective dose coefficients (µSv/Bq)

	Iodine-131	Cesium-137
Three months old	0.18	0.021
One year old	0.18	0.012
Five years old	0.10	0.0096
Adult	0.022	0.013

Bq: becquerels; µSv: microsieverts; mSv: millisieverts

Source: Prepared based on ICRP Publication 119, Compendium of Dose Coefficients based on ICRP Publication 60, 2012, International Commission on Radiological Protection (ICRP)

For example, the dose that an adult who consumed foods containing Cesium-137 will receive is calculated here.

Suppose the person has consumed 0.5 kg of foods containing 100 Bq of Cesium-137 per 1 kg.

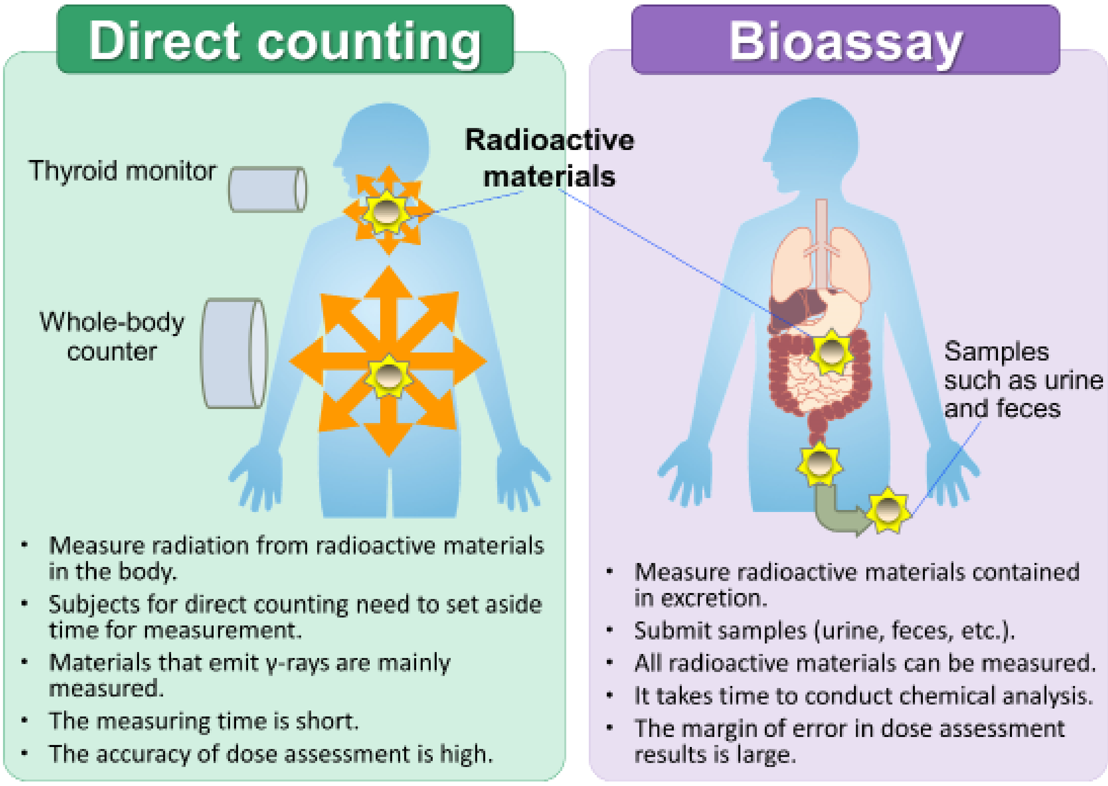
The amount of Cesium-137 actually consumed is 50 Bq. This value is multiplied by an effective dose coefficient to calculate committed effective dose (p.56 of Vol. 1, “Committed Effective Doses”).

Committed effective dose coefficients are defined in detail for each type of radioactive material, each intake route (inhalation or ingestion), and each age group (p.57 of Vol. 1, “Conversion Factors to Effective Doses”).

Included in this reference material on March 31, 2013
Updated on March 31, 2015

Dose Measurement and Calculation

Measuring Methods for Estimating the Intake of Radioactivity



Direct counting methods that directly measure γ-rays coming from within the body or bioassay methods that measure the amount of radioactive materials in samples such as urine and feces are used to estimate the intake of radioactive materials, which is required for calculating internal exposure doses.

In direct counting, the longer the measuring time, the more accurate values can be obtained. However, external measuring instruments also measure radiation from the environment while measuring radiation from the human body. Therefore, if measurements are carried out in locations at high ambient dose rates, sufficient shielding against environmental radiation is required. These instruments cannot measure radioactive materials that do not emit γ-rays or X-rays.

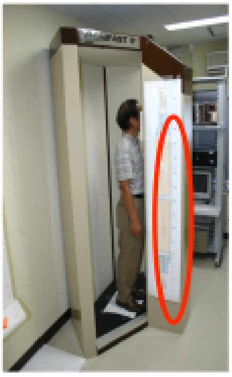
Bioassays can measure all kinds of radioactive materials but cannot provide accurate numerical values based on a single sampling. Therefore, it is necessary to collect samples for several days (urine, feces, etc.). Given that the amounts of radioactive materials excreted varies depending on individuals, their health conditions and amounts of food consumption, the margin of error is considered to be larger than that by direct counting.

Based on the results obtained using these methods, intake scenario (i.e., such as date of intake, acute or chronic intake, chemical form or particulate size, route of intake etc.) is taken into consideration and mathematical models (p.55 of Vol. 1, “Calculation of Internal Exposure Doses”) are used to calculate the percentages of radioactive materials remaining in the body or excreting into the samples measured to determine the intake of radionuclides. In both methods, if an exposure scenario is not certain, calculation results will have a larger margin of error.

Included in this reference material on March 31, 2013
Updated on March 31, 2021

Dose Measurement and Calculation

Instruments for Measuring Internal Exposure

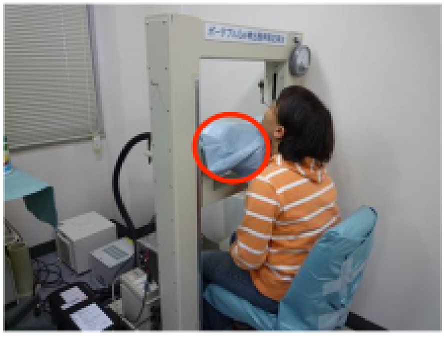


Stand-up whole-body counter

Scanning bed whole-body counter

Chair whole-body counter

Thyroid monitor



○ Detector

An instrument for measuring γ-rays emitted from the whole body, called a whole-body counter, is used to directly measure internal radioactivity. Whole-body counters have several types, including a stand-up type, bed type, and chair type.

Since radioactive cesium is distributed throughout the body, a whole-body counter is used to measure its amount within the body. If internal exposure by radioactive iodine is suspected, a thyroid monitor is used, as iodine accumulates in the thyroid (p.127 of Vol. 1, “Thyroid”). A radiation detector is applied to the part of the neck where the thyroid gland is situated to measure γ-rays emitted from there.

The time required for measurement is 1 to 5 minutes for simplified whole-body counters, 10 to 30 minutes for precision whole-body counters, and 2 to 5 minutes for thyroid monitors.

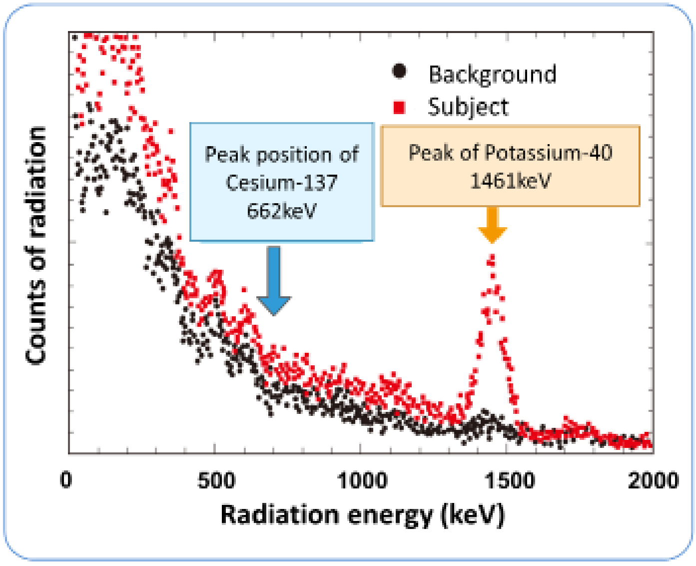
(Related to p.166 of Vol. 2, “Internal Exposure Measurement Using a Whole-body Counter”)

Included in this reference material on March 31, 2013
Updated on February 28, 2018

Dose Measurement and Calculation

Data on Internal Exposure Measured by Direct Counting

Whole-body counter



Measure radiation emitted from within the body ⇒ Measure internal radioactivity for each radioactive material

The amount of potassium in the body is around 2 g per 1 kg of body weight, and approx. 0.01% of that amount is radioactive potassium (Potassium-40)

keV: kilo electron volts

Radioactivity of each nuclide can be quantitatively assessed by measuring radiation emitted from within the body using a whole-body counter.

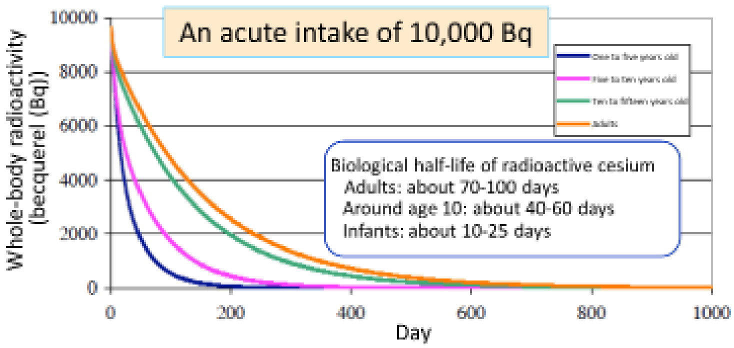
The black round dots in the graph represent values measured while no one is on the bed (background state). When the subject is on the bed, radiation peaks appear, as indicated by the red square dots. The energy of γ-rays is unique for each radioisotope. For example, radioactive potassium, K-40, emits γ-rays with energy of 1,461 keV. Therefore, if such amount of energy is detected, this reveals the existence of K-40 within the body. The gamma-ray energy of Cesium-137 is 662 keV.

While potassium is an element essential to life, approx. 0.01% of all potassium is radioactive. Radioactive potassium is mainly contained in water in cells and is present in muscles but is seldom present in fat cells that contain little water (p.8 of Vol. 1, “Naturally Occurring or Artificial”).

Included in this reference material on March 31, 2013
Updated on March 31, 2015

Dose Measurement and Calculation

Radioactivity in the Body and Dose Assessment

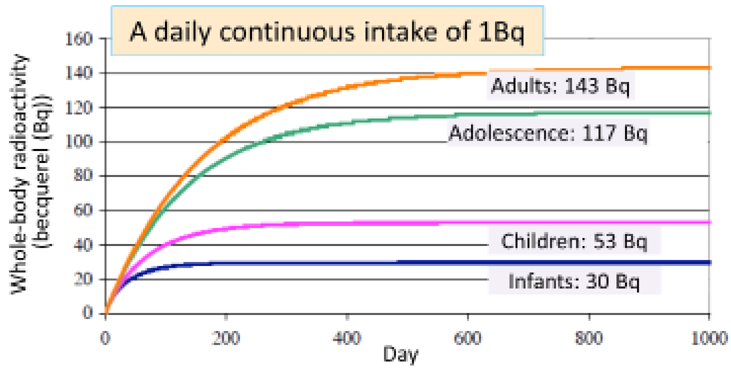


The younger a person is, the faster the metabolism.

↓

Estimation of initial exposure

- will be effective for no longer than around a year even for adults.
- will be effective for up to around half a year for children.



The younger a person is, the smaller the amount of radioactive materials remaining in the body.

↓

In estimating additional exposure through ingestion,

- significant values are unlikely to be obtained in children.
- it is more reasonable to examine adults in order to detect trace intake.

Source: Prepared based on a material released for the Japan Society of Radiation Safety Management Symposium in Miyazaki (June 29, 2012)

Whole-body counters (WBCs) can measure the radioactivity content in a body on the day of measurement. Similar to other radiation measuring devices, WBCs have a detection limit depending on their performance and counting time.

Given that radioactive cesium has a biological half-life of 70-100 days for adults (p.11 of Vol. 1, “Half-lives and Radioactive Decay”), around one year after the accident would be the time limit for estimation of the initial body burden (in the case of a single intake event at the beginning). As shown in the upper figure, the radioactivity of cesium incorporated into the body decreases in around a year to nearly zero, namely the level before the intake. Subsequent whole-body counting is performed for the purpose of estimating chronic exposure, mainly from foods (p.61 of Vol. 1, “Data on Internal Exposure Measured by Direct Counting”).

In contrast, whole-body counting for children is likely to yield values lower than the detection limit because trace amounts of the initial intake can be observed for a period of about half a year, and the residual radioactivity accumulated in the body by chronic intake is also minimal in children. In such cases, it would be more reasonable to examine adults and estimate their internal doses in terms of understanding the internal exposure situation in details, taking into account the fact that the committed effective dose coefficients are similar for both children and adults, despite the notable difference in their metabolism rates.

In order to estimate the committed effective dose from the measurement result for the radioactivity in the body, it is necessary to use an appropriate intake scenario and an appropriate model aligned with the exposure circumstances, such as acute or chronic intake, inhalation or ingestion as a dominant route of intake, the time when the intake started, and so on.

Regarding radionuclides with short effective half-lives, such as I-131, the radioactivity in the body diminishes rapidly, making it difficult to detect such radionuclides as time progresses. Additionally, pure beta-emitters lacking γ-ray emission, such as Sr-90, also cannot be detected by a whole-body counter (WBC).

Included in this reference material on March 31, 2013
Updated on March 31, 2023