Guideline for Radioactivity Measurements in the Environment and Individual Dose Assessment Following a Nuclear or Radiological Emergency

RCARO/ASEANTOM Project

2021

RCA Regional Office (RCARO)

Korea Institute of Radiological & Medical Sciences (KIRAMS)

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Contents

Introduction :1

This guideline document is developed as part of activities under the partnership project between the RCA Regional Office and the ASEAN Network of Regulatory Bodies on Atomic Energy (ASEANTOM), to improve the emergency monitoring capabilities of ASEANTOM member countries following a nuclear or radiological emergency by facilitating the development of detailed standard operating procedures regarding radioactivity measurements (of individuals and the environment).

In a nuclear or radiological emergency, radiation measurements provide the essential data needed by authorities and other decision-makers to evaluate the radiological emergency and implement timely protective actions. All measurements must be performed efficiently under various conditions, which means that pre-prepared emergency monitoring strategies and measurement plans must be in place. These strategies define and describe the main factors, principles, procedures, and methods related to measuring and sampling activities in different radiation situations. Radiation monitoring and dose assessment measurements obtained during emergency situations play a crucial role in the following:

- classifying the accident;
- decision-making for protective actions and subsequent measures, both in the field and the laboratory;
- preventing the wider spread of radioactivity; and
- protecting workers, the public, and the environment.

From preparedness planning, and at every stage of the developing situation (continuing through to after the emergency has been resolved), radiation monitoring and dose assessment provide a firm basis upon which decisions regarding protective actions can be confirmed or revised.

In general, a good emergency monitoring strategy starts from the advance identification

of potential hazard situations and extends to the environmental sampling performed during the late phases of an accident or event. It combines the arrangements and systems applied in routine monitoring with the special requirements set by emergency monitoring protocols, and the use of fixed monitoring stations with that of mobile measurement teams. It contains elements for analyzing, transmitting, and presenting measurement data, as well as for linking the data with the inputs/outputs of different forecasting and decision-support systems. It also considers the intrinsic characteristics of potential threat scenarios and includes options for adapting all measuring activities to suit the prevailing environmental conditions. Moreover, the various relationships to practical constraints set by societal and economic issues must also be considered.

Creating a monitoring strategy is a complicated task and requires asystematic approach. In general, the factors that must be considered in such a strategy can be roughly categorized as belonging to one of two groups:

- "Static"factors (ICRP, 1997), whose status or related contents are known or available before any nuclearor radiological emergency arises. Examples include population distribution, geography and topography, land use, legislation and official agreements, fixed potential sources (e.g., nuclear facilities etc.) and other recognized threat scenarios, routine monitoring arrangements, and resources allocated to emergency monitoring (e.g., measuring equipment, capacity of laboratory measurements, extra manpower, decision-assistance systems with auxiliary support material). These factors cannot typically be changed quickly, at least not during the early phases of an accident.

- "Dynamic"factors (ICRP, 1997), whose contents become clear only at the beginning (or during the course) of an accident or a specific event. The scenario (source term, location) and prevailing environmental conditions (e.g., weather) are the most evident items in this group. In addition, the extent and nature of resources actually available at the time may differ significantly from those listed in connection with the static factors; for instance, an event may take place during a holiday season in which some key personnel are absent, some of the equipment may be broken, or serious hardware

or software problems — including problems of data-communication systems — may occur.

A monitoring strategy should be versatile and realistic, and it should include provisions for the various back-up arrangements needed in case problems arise with the primary measuring equipment or expert personnel. However, a strategy must not seek to be universal and to cover all possible situations. Accidents or events can occur that are not included in a well-defined, comprehensive strategy plan. Thus, a strategy should not stifle the creativity of the emergency authorities or the deployment of common sense.

The applicability of any emergency monitoring strategy must be tested regularly in different exercise scenarios (e.g., table-top exercises, drills, and field exercises). Strategies should also be updated whenever appropriate. For example, updates should be implemented following the identification of a new threat scenario or the acquisition of a new type of measuring equipment.

Purpose : **2**

The purpose of this guideline is to make recommendations regarding the emergency monitoring processes — including the monitoring of environments and individuals — used to respond to a nuclear or radiological emergency, to thereby mitigate and minimize the consequences of such emergencies and to protect emergency workers, members of public, and the environment against the harmful effects of ionizing radiation in emergency exposure situations.

These guidelines are intended for use by the Association of Southeast Asian Nations Network of Regulatory Bodies on Atomic Energy (ASEANTOM), to facilitate the development of detailed standard operating proceduresregarding radioactivity measurements (of individuals and the environment), taking account of the emergency monitoring resources available in each ASEAN country. In addition, this guideline is designed to improve the emergency monitoring capabilities of ASEANTOM member states following a nuclear or radiological emergency.

Nuclear and Radiological Hazard Assessment in the Asean Region

: 3

An emergency management system should be designed to be commensurate with the hazard assessment results, and it should facilitate an effective emergency response to reasonably foreseeable events (including those of very low probability). The hazard assessment provides the basis for a graded approach regarding the preparedness for and response to a nuclear or radiological emergency (IAEA, 2004). This chapter discusses the pertinent nuclear and radiological hazards in the ASEAN region.

The hazard assessments for the ASEAN region provide the basis for our review and assessment of the needs and adequacies of radiation-monitoring capabilities in the region, which seeks to integrate these monitoring capabilities into a regional system. Hazards in the ASEAN region have been identified and assessed, to design arrangements commensurate with the potential consequences of a nuclear or radiological emergency (Kurihara et al., 2018). This hazard assessment included considerations of the following:

(a) Events that could affect a facility or activity, including events of very low probability and events not considered in the design;

(b) Events involvinga combination of a nuclear/radiological emergency and a conventional emergency (e.g., earthquake, tropical cyclone, or severe weather) that could affect wide areas and/or impair capabilities to provide support during the response;

(c) Events that could affect several facilities and activities concurrently, including their mutual interactions; and

(d) Events at facilities across the ASEAN region that may affect the region as a whole.

To establish the required arrangements, the assessed hazards were grouped into emergency preparedness categories (Table 1)(IAEA, 2004). These categories form the basis of the graded approach toward the application of justified and optimized nuclear or radiological emergency response arrangements.

Table 1: Emergency preparedness categories.

Category	Description
I	Facilities (e.g., nuclear power plants) for which on-site events[a,b] (including those not considered in the design[c]) are postulated to have severe deterministic off-site effects[d] warranting precautionary urgent protective actions, urgent protective actions, or early protective actions, as well as other response actions designed to facilitate an emergency response in accordance with international standards[e], or for which such events have occurred in similar facilities.
II	Facilities [e.g., certain types of research reactors and nuclear reactors used to provide power for the propulsion of vessels (e.g., ships and submarines)] for which on-site events[a,b] are postulated to impart doses to people off-site that would warrant urgent or early protective actions, as well as other response actions designed to facilitate an emergency response in accordance with international standards[e], or for which such events have occurred in similar facilities. Category II (as opposed to Category I) does not include facilities for which on-site events (including those not considered in the design) are postulated to have severe deterministic off-site effects, or for which such events have occurred in similar facilities.
III	Facilities (e.g., industrial irradiation facilities or certain hospitals) for which on-site events[b] are postulated to warrant protective actions and other on-site response actions designed to facilitate an emergency response in accordance with international standards[e], or for which such events have occurred in similar facilities. Category III (as opposed to Category II) does not include facilities for which events are postulated that could warrant urgent protective actions or early protective actions off-site, or for which such events have occurred in similar facilities.
IV	Activities and actions that could give rise to a nuclear or radiological emergency that would warrant protective actions and other response actions designed to facilitate an emergency response in accordance with international standards[e] in an unforeseen location. These activities and actions include the following: (a) transport of nuclear or radioactive materials and other authorized activities involving mobile dangerous sources (e.g., industrial radiography sources, nuclear-powered satellites,

	and radioisotope thermoelectric generators), and (b) theft of a dangerous source and use of a radiological dispersal device or radiological exposure device[f]. This category also includes: (i) detection of elevated radiation levels of unknown origin or of contaminated commodities; (ii) identification of clinical symptoms due to exposure to radiation; and (iii) a transnational emergency (not in Category V) arising from a nuclear or radiological emergency in another state. Category IV represents a hazard level that applies for all states and jurisdictions.
V	Areas within emergency planning zones and distances[g] in a state for a Category I or II event located in another state.

[a] That is, on-site events involving an atmospheric or aquatic release of radioactive material, or external exposure (e.g., due to a loss of shielding or a criticality event) originating from an on-site location.

[b] Such events include nuclear security events.

[c] This includes events outside of the design-basis accidents and — as appropriate — conditions outside the design-extension conditions.

[d] See "Deterministic Effect" under Definitions in the references (IAEA, 2015).

[e] See the goals of emergency responses in Para. 3.2 and the generic criteria in Appendix II of the references (IAEA, 2015).

[f] A radiological dispersal device spreads radioactive material using conventional explosives or other means. A radiation exposure device features radioactive material designed to intentionally expose members of the public to radiation. These could be fabricated, modified, or improvised devices.

[g] See Para. 5.38 of the references (IAEA, 2015).

EPC I refers to facilities for which on-site events (including very low probability events) involving an atmospheric or aquatic release of radioactive material or external exposure from an on-site location are postulated to have severe detrimental health effects (doses exceeding those for which intervention is expected under all circumstances) off-site. At present, no EPC I facilities operate within the ASEAN region. All commissioned EPC I facilities in the broader South and East Asian geographical area lie outside ASEAN (Kurihara et al., 2018).

EPC II refers to facilities (e.g., certain types of research reactors) for which on-site events are postulated to expose people off-site to doses that warrant urgent protective actions in accordance with international standards, or for which such events have occurred in similar facilities. There are a number of research and training reactors in the ASEAN region (ICRU, 2001). However, the maximum distance of planning [i.e., the Ingestion and Commodities

Planning Distance (ICPD)] for EPC II facilities is 50 km (IAEA, 2004); therefore,

- the planning of emergency preparedness and response measures is limited to the country operating the given facility, and
- it is assumed that no other ASEAN country would be affected by any EPC II facility emergency.

Whilst nuclear power plants (NPPs) are not yet operational in ASEAN, the radiation sources of different activities can be involved in emergency situations. These emergencies are referred to as "radiological emergencies," and the emergency preparedness categories associated with these sources/facilities are categorized as EPC III, IV, and V. EPC III hazards are associated with facilities (e.g., industrial irradiators) for which on-site events are postulated to produce doses or contamination that warrants urgent protective on-site actions, or for which such events have occurred in similar facilities. EPC III, IV, and V facilities and activities arelimited to hazards within the various ASEAN member states and should be listed in the member state National Radiation Emergency Response Plans (Kurihara et al., 2018). The present hazard assessment identifies the requirements that must be considered in these plans. Such nuclear and radiological hazards are very unlikely to have regional impacts. Nevertheless, they are included in planning arrangements, to provide a basis for uniform responses to such events across the ASEAN region and to facilitate technical support to a member state that requests it.

In the present assessment of radiological hazards for the ASEAN region, the risk categories are identified as predominately EPC III and EPC IV (Kurihara et al., 2018). This determines the risk level for the overall region as low-medium. Zombori identified that airborne contamination from outside the ASEAN region could only be expected from the west (Indian sub-continent) and North (China, Korea, and Japan) (Kurihara et al., 2018). The hazards in the region remain liable to change according to the adoption of NPP technologies [CABALLERO-ANTHONY]. These factors must be considered when setting any national or regional monitoring program with the correct radiation detection levels for airborne or water monitoring this necessitates detectors that are effective in low dose-rate ranges.

Design of a Monitoring and Sampling Program

<div style="text-align: right">: **4**</div>

4.1. Introduction

The type, scale, and extent of a monitoring program should be commensurate with risk and determined by considering the source characteristics, expected/current radionuclide release rates, radionuclide composition, and the likelihood and magnitude of potential exposures to the public, workers, and the environment. In this chapter, the various aspects of sampling and measurement programs are introduced in detail.

4.2. Characterization and Baseline Monitoring

Baseline characterizations may be undertaken as part of the hazard, safety, or environmental impact assessments. They provide information and data to support the design of monitoring practices related to a facility or activity. Baseline characterizations should be undertaken to obtain a better understanding of the prevailing circumstances and conditions relevant to exposure (e.g., typical diets of local residents, habits that could result in exposure, and time spent in different areas and environments or conducting various types of activities), to identify relevant exposure pathways and establish the baseline radiological conditions at sites or areas of interest. Radioactivity and radiation levels should then be compared to the baseline radiological condition across time and space during later stages of the facility lifetime or activity, to evaluate the impacts of the facility or activity on the public and environment and to ensure protection and safety.

In emergency exposure situations, such baseline characterization data can be used for the prioritization of hazards, risks, and impacts, to help establish a protection or remediation strategy; they can also be applied in decision-making processes, to identify appropriate protective and/or remedial actions. Subsequently, they can provide a

baseline against which to compare the efficacies of the protective and remedial actions undertaken, to manage the situation and reduce doses.

Baseline characterization is typically of limited duration and can be terminated once adequate characterization data have been collected (e.g., to support planning and decision-making). The data should be used to develop a relevant monitoring strategy and program for locations at which baseline measurements were taken and retained in operational monitoring programs, to facilitate comparison over time.

When planning for emergencies, the baseline characterization data and monitoring data collected during operations can be used to understand the sources, radiation levels, and exposure, to thereby support the development of the protection strategy, evaluate the effectiveness of protective actions, and derive an initial baseline. When transitioning to an existing or planned exposure scenario after the emergency, this radiological baseline data can be used to inform the delineation of affected areas and the establishment of recovery objectives, or to verify that remediation efforts have met the end-state criterion, respectively.

4.3. During an Emergency

The purpose of monitoring in an emergency exposure situation (emergency monitoring) is to facilitate decisions regarding emergency classification, appropriate protective actions, and other response actions, as well asto assess doses and health risks. Emergency monitoring includes source, environmental, and individual monitoring in accordance with the monitoring strategy developed from the hazard assessment.

The overall strategy for emergency monitoring should be developed at the preparedness stage. It forms an essential component of a protection strategy that considers hazards (and threats) and the types of protective actions and other response actions potentially required to protect workers, emergency workers, members of the public, and — as relevant — patients and helpers during emergency, or to mitigate the consequences of a nuclear or radiological emergency.

The specific objectives of emergency monitoring are as follows:

(a) Provide information for emergency classification on the basis of emergency action levels (EALs);

(b) Inform decision makers regarding the need to take protective actions and other response actions, primarily on the basis of operational intervention levels (OILs);

(c) Help prevent the spread of contamination;

(d) Provide information to protect emergency workers;

(e) Provide reliable and timely data regarding the level and degree of hazards resulting from an emergency;

(f) Determine the extent and duration of the hazard;

(g) Provide details regarding the physical and chemical characteristics of the hazard;

(h) Confirm the efficiency of remedial measures such as decontamination; and

(i) Provide the information required to identify individuals in need of specialized urgent/emergent medical care or long-term medical screening and follow-ups.

The emergency monitoring strategy should include arrangements for the following:

(a) Source monitoring;

(b) Environmental monitoring;

(c) Individual monitoring of emergency workers; and

(d) Screening of members of the public from the affected area.

The monitoring strategyshould facilitate the use of pre-established operational criteria (i.e., conditions on the site, EALs, and OILs) in accordance with the protection strategy and provision of instruments displaying/measuring those parameters readily comparable to operational criteria. In these arrangements, the expected responses of instrumentation, structures, systems, and components at the facility under emergency conditions should be considered. The routine monitoring program may also form the basis for emergency monitoring programs at these types of facilities, as part of emergency preparedness measures.

Design of a Monitoring and Sampling Program

The nature and extent of the actual emergency may be uncertain; however, it is important that advanced arrangements be made to prepare for a range of postulated emergencies. The design of an emergency monitoring program will depend on the scale of the emergencies envisaged. Arrangements should be made regarding measurement instruments, sample collection, sample analysis, dose assessment, uncertainty analysis, the interpretation of results, and the communication and receipt of results to and from other organizations, as appropriate.

Environmental monitoring practices during the urgent emergency response phase should focus on understanding the changing situation and making timely decisions regarding the adjustment of protective and other response actions. It should focus on essential measurements (e.g., dose rate measurements), and it might not be comprehensive with respect to the radionuclides and environmental media measured. During this phase, all available meteorological information and dispersion-modelling predictions should be implemented to determine the areas in which people may be affected and where monitoring resources should be deployed. The combined information obtained from monitoring and modelling should be used to make informed decisions regarding the focusing of monitoring resources in the potentially most contaminated areas.

Once the urgent emergency situation is over and the necessary urgent protective actions (and other response actions) have been taken, a sampling program — employing preselected sampling locations — should be established to determine whether early protective and other response actions (e.g., temporary relocations) should be implemented, or whetherrestrictions on the consumption of local foodstuffs can be lifted.

4.4. After the Emergency and the Transition to an Existing Exposuresituation

The practical objective of environmental and individual monitoring during the transition phase from an emergency toan existing or planned exposure situation is to facilitate the reassessment of (i) hazards in the affected areas and (ii) residual doses to the public, to

facilitate comparisons with generic criteria defining the transition or termination of an emergency.

The monitoring strategy in the transition phase from an emergency exposure situation to either an existing or a planned one should facilitate the following:

a) Safe and effective management of radioactive waste produced in a nuclear or radiological emergency, including radioactive waste generated from associated protective actions and other response actions;

b) The safe and effective transition from an emergency exposure situation to an existing one, in accordance with the protection strategy.

Environmental monitoring during the transition phase should be comprehensive and more detailed than during the emergency response one. Such monitoring will be less constrained by the need for timely action, and it may begin to resemble the monitoring for an existing exposure situation, with greater focus on tracking trends in environmental contamination and supporting future decision-making regarding remedial actions or the termination of an emergency.

Rapid Techniques for Radioactivity Measurements

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5.1. Introduction

The detection of radiological accidents and the monitoring of contamination spread are of great importance. Thus, the rapid and effective selection and deployment of measurement procedures is required when responding to the initial stages of an emergency. This chapter discusses the various measurement techniques that can be used to provide timely information for diagnosing an evolving exposure situation.

5.2. Ambient Gamma Dose Rates

The real-time availability of gamma dose-rate measurements during and after an accident improves the abilities of decision makers to accurately respond to a radiological emergency. Measurement stations containing Geiger–Mueller (G–M) detectors are used to determine external radiation exposure, by continuously measuring the ambient gamma dose rate in the environment. This facilitates the rapid and reliable detection of small changes in environmental radioactivity over a large area, as well as the identification of long-ranging trends. Figure 1 depicts the basic components of a gamma dose-rate measurement system.

Figure 1: Remote ambient gamma dose-rate monitoring system

Gamma dose rate detector   Measurement software   ArcGIS mapping product
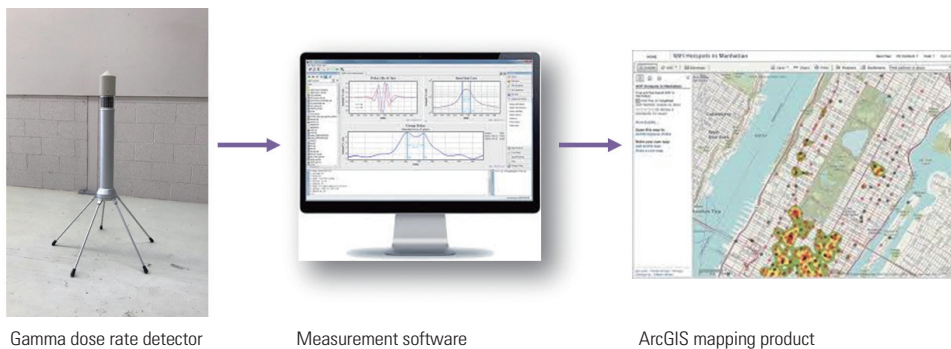
When used as part of a network, these measurement stations can determine the natural radiation exposure of the public, as well as any increase resulting from nuclear activities. The fraction of the background dose rate imparted to humans from environmentalgamma radiation is highly variable and depends on factors such as the radioactivity of the local rock and soil, building material characteristics, and the designs of buildings in which people live and work. Dose rates imparted to humans from natural environmental gamma radiation therefore exhibit considerable geographical variability. To ensure that measurements can be made with sufficient accuracy, it is necessary to determine the response of the monitoring equipment to cosmic and terrestrial radiation, as well as to consider the inherent background signal produced by electrical noise and radioactivity within the monitoring equipment itself.

When evaluating dose-rate variations over time, it is important to appreciate that the total dose rate arising fromenvironmental radiation does not remain constant over the course of a day or from one day to the next. Several natural temporal variations are associated with diurnal temperature changes and the accompanying turbulence. Precipitation also plays a major role in natural variations of the background. For example, rain or snow can scavenge airborne radon progeny, causing an increase in radiation levels lasting several hours. Subsequently, wet ground attenuates the terrestrial component, causing radiation levels to drop below the previous baseline once the precipitation stops; the level gradually increases as the ground dries. Other possible natural variations relate to seasonal influences in the exhalation rate of radon from the ground and changes in the cosmic ray component, which is associated with atmospheric pressures and solar cycles.

The recommended operational quantity for area monitoring is ambient dose equivalent (sieverts, Sv). G–M detectors can measure over a wide range of 0.05-10 Sv/h. Gamma dose-rate stations are typically installed 1 m above flat grassland; they continually measure the ambient gamma dose rate at 10 minute intervals before transmitting these data to a central data station. The quality of the data can be monitored using a list of parameters (e.g., system humidity, high voltage error, server status, etc.) to assess whether the detectors are operating within required tolerances. Quality control activities

are required to ensure data integrity, correctness, and completeness; to identify and address any errors or omissions; to document and archive dose-rate data; and to record all quality control activities. These activities should focus on detector calibration using a traceable sealed gamma source (e.g., ^{152}Eu) and data availability across the network (e.g., exceeding 95%).

In the case of an emergency, detector networks facilitate the rapid recognition of increased levels of radioactivity in the air, and they can monitor the dispersion of a radioactive plume almost in real-time (Figure 2).Affected areas exhibiting increased ambient dose-rate levels can be rapidly identified, and the measured data facilitate an expediated dose assessment for these areas.

Figure 2: Early-warning detection system, showing the system architecture and data flow. FTP: file transfer protocol; I/O: input/output; PDC: primary domain controller; QA: quality assurance; ARM: advanced reduced-instruction-set-computer machine
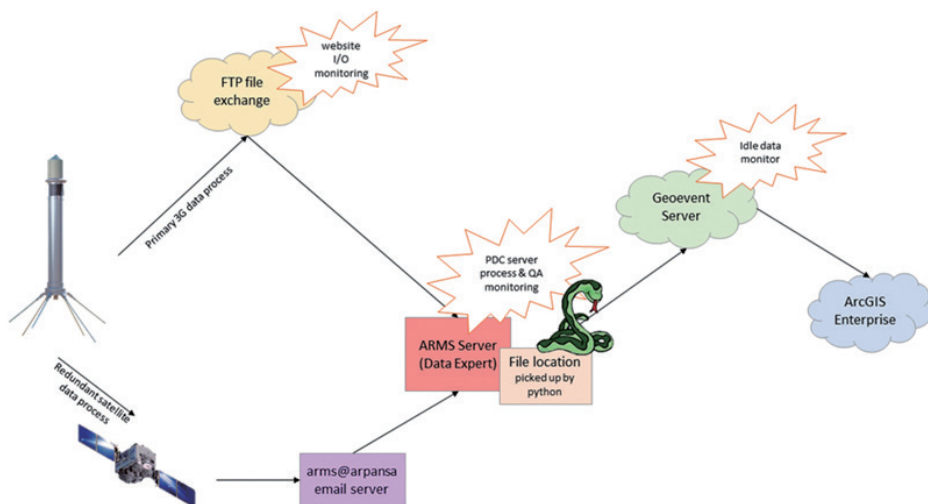
5.3. Sample Collection and Preparation

Sampling locations should provide an overview of the immediate vicinity of the contamination source, as well as distal areas. In the early stages following an event,

sampling and measurements should be performed in all directions but primarily in the dominant wind direction for an airborne release or downstream for an aquatic one. Locations will depend on the spatial distribution of the gamma dose rate in air with respect to the release of gamma-emitting radionuclides. Monitoring is therefore focused on areas exhibiting the highest contamination risk, whilst also taking land use into account. The external gamma dose rate can be measured directly, without soil and vegetation sampling. However, samples are required to establish intervention levels on the basis of the irradiation dose values over various stages (e.g., evacuation, resettlement, and temporary relocation), as well as to introduce restrictions upon water and food consumption.

The objective of optimization is to estimate the distribution of environmental contamination — within a given error margin — at minimum cost and time expense. The optimization of the sampling plan therefore considers the personnel resources available for sample collection, the time and cost of measurement, the quantity and mass of samples, the size of the study area, the depth of sampling, and the vertical and spatial resolutions required to fulfil the monitoring objectives. In practice, the inhomogeneous distribution of contaminants is often the largest contributor to data uncertainty and is typically not quantified. Accuracy, precision, and other data-quality indicators that characterize the robustness of the analytical data are sensitive to the sample preservation, transport, and laboratory analysis procedures; however, they neglect the spatial variability of the contaminant at the site. It is therefore important that samples are collected that can provide the confidence levels required for effective environmental management.

5.4. Sample Pre-Treatement

After sampling, pre-treatment is required to ensure homogeneity and appropriate conditions for quantification. Drying, grinding, sieving, and ashing are often sequentially performed for solid samples, including soil/sediment, vegetation, foodstuffs, animal/ human organs, bone, and tissues. It should be noted that drying and ashing should only

be used for non-volatile radionuclides. Fresh samples of volatile radionuclides such as ^{3}H and ^{14}C should be processed without drying or ashing, and high-temperature ashing should be avoided for semi-volatile radionuclides such as ^{210}Po. Most non-volatile radionuclides can be extracted from soil and sediment using acid digestion. This can be performed using a mixture of mineral acids (HCl, HF, HNO_3, $HClO_4$, and H_3PO_4) in open systems or pressure vessels, or with the assistance of microwaves. For biological samples, H_2O_2 is often added to acid digestion to prompt the decomposition of organic matter in the samples. For large-volume water samples, pre-concentration is performed either in-situ or in the laboratory. Typically, evaporation or co-precipitation are used. The evaporation involves reducing the sample volume by careful heating. The co-precipitation approach selected depends on the chemical properties of the targeted radionuclide; coprecipitates such as ion hydroxides, manganese oxide, and calcium phosphate/carbonate/oxalate are often employed.

5.4.1. Air Particulates

Ambient air sampling is performed to determine environmental impacts and ensure compliance with public-health and environmental standards or protection guidelines, rather than to provide detailed radiological characterization. Air-particulate and ambient-air sampling is performed using an air sampler; this is essentially a pump that operates at a known or specified flow rate for a timed sampling period and which draws in air through a suitable filter medium (see Figure 3(a)). A variety of sample holders can be fitted to the air pump, allowing filter discs or cartridges to be used. The filter is subsequently analyzed for the contaminant in question. The activity on the filter is assessed in Bq; thus, when the volume of air sampled is known, average activity concentration results can be given in Bq/m^3. The type of filter used depends on the contaminant to be measured. Charcoal filters are used for radioiodines, glass-fiber or paper filters are used for gross beta and gamma particulates, and water bubblers are employed for tritiated water/vapor.

Figure 3: Air particulate samplers

(a) Portable air sampler and charcoal cup

(b) High volume air sampler

Air sampling methods should be designed to consider the environmental characteristics of the site, the radiological releases from site operations, and potential receptors. Transportable air samplers (e.g., high-volume air samplers) operated by portable electrical generators can be installed at locations of interest (see Figure 3(b)). Portable air samplers with an operating voltage of 12 V are useful in field-sampling situations. The flow rate should be pre-calibrated. Care should be taken if a heavy dust load is present on the filter, because this may restrict the flow rate.

Filter-sample analysis can be performed using gamma spectrometry, liquid scintillation counting, and/or gas-flow proportional counting. Whilst most filters are surface collectors and can be readily analyzed, the analyst should determine whether it is necessary to dissolve the filter for composite analysis or further specific isotopic analyses. After analysis, and once the data quality has been reviewed, suitable correction factors can be applied; these ensure that results are not underreported and that a conservative approach to emissions estimates is maintained. Depending on the sample method, a variety of correction factors can be applied, including self-absorption, sampler efficiency, decay correction factors, and more. The radioactive decay factor reflects the interval between the midpoint of the sample collection period and the sample analysis time. In most cases, this factor can be set to 1 because the time lapse between collection and analysis is much shorter than the half-lives of the radioisotopes of concern. The self-

absorption factor corrects for the bias produced by radiation emitted from the collected particles, dust/particulates, and filter media itself. In filters, this factor depends on the quantity of material collected. The sampler efficiency factor accounts for biases produced by problems with the sampler operation. If the sampler operates without interruption during the sampling period, the efficiency is 100% (or 1); however, when the operation is incomplete or interrupted, the sampler efficiency factor is determined by dividing the sample collection time by the entire sample period. If the sampler efficiency factor is too low, an invalid sample may be produced (Babu et al., 2010).

All data results should be evaluated with respect to established criteria, to evaluate potential changes over time. Repeat measurements at fixed sampling locations can be used to indicate measurements outside the normal operating range, as well as the expected statistical deviations. Data trending can also indicate increasing or decreasing emissions over various cycle times or events.

The sampling frequency and length of time selected to obtain representative samples should also consider environmental factors that may affect collection efficiencies and the accuracy of the results. Multiple composite samples or longer measurement times may be required to achieve lower detection limits (ICRU, 2002).

5.4.2. Soil and Vegetation

Terrestrial sampling programs for emergency response monitoring facilitate decision making and should be performed within the minimum time required to ensure representative soil and vegetation samples. Typical objectives include the following:

- Assessing radionuclide distribution in terrestrial environments;
- Validating predictions of contamination from environmental models, thereby facilitating their development and uncertainty reduction;
- Mapping provisional contamination level estimates to identify areas where contamination of the soil or vegetation exceeds intervention levels; and
- Assessing public exposure.

The sampling priorities depend on the land use (e.g., residential, agricultural, rural, or commercial) and the types of industrial activity, public services, and infrastructure. Additional protections for people, livestock, crops, and water supplies may be necessary. Embargoes on the use of water and food and the maintenance or restoration of vital infrastructure should then be based on the OILs.

Understanding the behaviour of radionuclides with respect to depth becomes increasingly important over time following an incident. Hence, sample-gathering procedures should employ incremental sampling (with respect to depth) at representative locations, to evaluate trends in the reduction of the external dose rate under the long-term vertical migration of radionuclides within the soil.

Radionuclide fallout can penetrate deeply into the ground, and the intensity of this penetration is determined by the chemical properties of the element, the physical and chemical properties of the fallout, the landscape, and the soil and climate characteristics. Radionuclides are uniformly mixed in the arable (i.e., ploughed or tilled) stratum of the soil; over time, these can migrate into the subsoil horizon. Neglecting this vertical migration could lead to significant errors when evaluating the activities and areal distributions of radionuclides. Therefore, to obtain a representative sample from a field site, it is necessary to understand the following:

(a) the source of radioactive contamination;

(b) the physical and chemical characteristics of the radioactive material; and

(c) its depth migration into soils.

Soil and vegetation sampling is performed according to dose-rate measurements only after the end of the release or plume passage. The measurements of radionuclide concentrations provide the ground-deposition values and data required to create contamination maps. Radionuclide concentrations in vegetation provide important information regarding the form of fallout. Samples of grass, lichen, and mosses are important indicators of fallout radionuclides. For emergency food restrictions, leafy vegetables (a good indicator for plant food) should be collected on a daily basis. Samples

from vegetables, fruit, grains, and mushrooms should be collected at time of harvest.

Plants are primarily contaminated during routine and emergency releases, through direct deposition of aerosol-bound and gaseous radionuclides or by direct contamination (by wind or rain splash) from resuspended radionuclides. Root uptake can also represent a significant route, especially for medium- to long-lived radionuclides. The heterogeneity of radioactive contamination is lower in plant samples compared with soil ones, because plant samples are collected from a greater area and the distribution of plant root systems across a larger area effectively averages out the soil contamination.

Soils can generally be divided into topsoils, surface soils, and subsurface soils. Topsoils are directly exposed to contamination and play an important role in the resuspension and transport of particles. Surface soils and subsurface soils cover the zone where plant root systems are located; as such, they represent a reservoir of radionuclides for plant uptake, an important source of external irradiation. When the soil surface is radioactively contaminated, depth sampling must capture the complete profile of the radionuclide under investigation. The deeper the sampling, the greater the sample mass collected and the higher the costs of transportation, preparation, and analysis. The optimal practice is to sample the soil to the depth of the root system (ca. 10 cm) when predicting contamination in the pasture vegetation of unploughed fields. Sampling at stony sites and sites featuring heavy sod is difficult; in these instances, sampling is commonly performed to a shallower depth of 1-2 cm.

Soil samples can be single samples or composite (combined) samples. The area of a sample is either single and non-separable (i.e., one-core extraction) or a combination of areas adjacent to each other (i.e., extraction of adjacent cores). For composite samples, the distance between the individual samples must be large enough that the radionuclide contents are mutually independent. The representativeness of a composite sample is higher than for a single sample, and the variance is therefore lower. Soil and vegetation samples should be collected on open ground unlikely to have experienced any disturbance to the deposition pattern. The sampling of depth profiles can generate a full soil characterization. Samples of agricultural soil can be used to study the ingestion

pathways; here, it is also appropriate to take conjugate samples of crops or vegetables at the same location. Typically, local species are preferable. Care should be taken not to include adhering soil particles in the sample, because such particles would most likely be removed during food preparation procedures. Pasture is important because of the rapid uptake of important radionuclides (e.g. radioisotopesof iodine and cesium) by animals — particularly cattle — and the subsequent transfer to milk. Pastures should be sampled where wet or dry deposition is expected to be maximal. Samples of milk and undisturbed soil should also be collected at these locations.

The sample area should be horizontal and flat, in an open area far from large trees or buildings. Plants should be of uniform height. However, it is not always possible to choose an open area (e.g., in a forest or urban environment). Increasingthe sample area, mass, or volume reduces the measurement uncertainty for the contaminant concentrations in soil or plants, and it potentially provides more sample material, thereby reducing the measurement time; however, it increases the transport costs and preparation time. The selection of these parameters represents a component of the sampling program's optimization. It is assumed that the activity concentration of the target radionuclides exceeds the minimum detectable activity (MDA) for the detection method employed.

The detection limits for environmental sample analysis depend upon the objectives of the sampling campaign and the requirements of the monitoring program. To achieve these levels of detection, the quantities of soil and plant samples are typically 0.1-1 kg for gamma- and beta-emitting radionuclides, and only a few grams for alpha-emitting ones. Sampling can be performed using manual techniques (e.g., hand excavation, hand auger, and corer samples) or via power-driven [e.g., all types of drilling, including small diameter drilling (e.g., ram core soundings) and cone penetration testing] or mechanical excavations. To sample uniform surface layers of soil, ISO 18589-2:2007 recommends a 20 cm square, 5 cm deep frame or a 5 cm wide, 5 cm deep ring (more suitable for incremental composite sampling). The sample consistency often depends on water content, which affects the storage and transport requirements. The selection of proper sampling equipment and relevant sampling procedures should ensure that the sample is

representative of the sample type, provides a sufficient sample quantity for the selected laboratory measurement method, and is of the required sensitivity to ensure compliance with the set data. To mitigate the bias introduced by sample processing, preparation, and measurements, it is essential to synchronize the procedures prior to measuring. Only validated or verified procedures should be used, and written procedure instructions should be available to all laboratories assisting in sample preparation and measurement. This will ensure that all data used for quality-requirement characterization are comparable, have the same quality, and can be combined with each other.

5.4.3. Surface Water

Following the accidental release of radioactive substances, decisions regarding the extent of contamination in surface and groundwater supplies must be made in a timely manner (ISO, 2019); to this end, numerous measurements must be collected as quickly as possible, to rapidly obtain an overview of contamination in bodies of surface water.

Surface water monitoring focuses on the following areas:

- Areas of water bodies with current or potential utility (e.g., collection of drinking water through bank filtration/direct collection, irrigation of agriculturally used areas, etc.);
- Sections of rivers in cross-border regions;
- Mouths and estuaries of rivers;
- Coastal waters; and
- Uncontaminated sections of rivers that serve as reference sites.

Small volumes (< 20 L) of surface water can be collected directly into a polyethylene bottle or barrel; meanwhile, for large volume samples, a submersible pump is often used. When the depth distribution of radionuclides is studied, water depth profiles must be collected; for this, a Nansen bottle is often used, especially for seawater collections to depths of up to 4000 m. Water samples are nearly always acidified to pH 1-2 with HNO3, to reduce the adsorption of radionuclides on the container walls; the only

exception to this arises in analyses of certain radionuclides such as iodine.

In the case of flowing water, mixed water samples should be continually collected at the riverbed profile, at a depth not exceeding 1 m. The collection period should be commensurate to the half-life of the radionuclides of interest. In standing waters, the periodic collection of representative samples (with an adequate periodicity) is sufficient. In the case of an accidental release, daily mixed samples or spot samples should be collected.

In the case of an accidental release of radioactive substances, it must be expected that the measurement and sampling containers are also contaminated. This makes verification checks obligatory. To minimize such contamination, it is recommended that measurement containers (Marinelli beakers) be stored in a manner that prevents their contamination through airborne activity.

Measurements should be performed on unfiltered water samples. This ensures that the activity attached to suspended matter is also quantified. By quantifying both the content of suspended matter andits specific activity, a distinction can be drawn between nuclides present as dissolved and particulate states in the sample, respectively. At moderate concentrations (e.g., 25 g/m^3), a significant proportion of the radionuclides may be found attached to the suspended matter, depending on the radionuclide involved. Elevated concentrations of suspended matter (of more than 100 g/m^3) may occur under particulate nuclide increases of 90% or more; these will therefore need to be quantified separately. Measurements on filtered water samples should be avoided, because the separation of dissolved and particulate nuclide portions is problematic and the results thereby obtained will produce an overly optimistic assessment when exposure is evaluated (e.g., for the exposure path "Irrigation of Agriculturally Used Land").

5.5. Sample Measurement Using Gross Measurement Techniques Instrumentation

Gross alpha and beta analyses are used as a rapid method to estimate the activities of

alpha and beta radionuclides in water, air filters, soils, sludge, and wastewater. Gross screening techniques are designed to quickly provide information regarding a particular action level, with minimal chemical preparation. The purpose of gross alpha and beta measurements is to provide adequate information concerning the activity within samples, and to determine whether further detailed analyses are required. Both gas-flow proportional counters and liquid-scintillation counters can be used for these measurements.

5.5.1 Gas-flow Proportional Counter

This system consists of a gas-flow detector, supporting electronics, and an optional guard detector for reducing the background count rate. A thin window can be placed between the gas-flow detector and sample, to protect the detector from contamination; alternatively, the sample can be placed directly into the detector. This system does not typically provide useful data for identifying radionuclides unless it is preceded by nuclide-specific chemical separations. Even the deposition of the sample material on the planchette is critical to the analysis process. In some analyses, ringed planchettes may assist the even deposition of sample material. An uneven deposition may produce an incorrect mass-attenuation correction, and it may introduce a position-dependent bias into the analysis. The latter case arises from the fact that gas-flow proportional counters are not radially symmetric; thus, rotating an unevenly deposited sample by 45° may drastically alter the instrument response.

5.5.2 Liquid Scintillation Spectrometry

Typically, samples are subjected to chemical separations, and the resulting materials are placed in a vial with a scintillation cocktail. When the alpha or beta particle energy is absorbed by the cocktail, light pulses are emitted; these are detected by photomultiplier tubes. One pulse of light is emitted for each particle absorbed. The emitted light intensity is related to the energy of the alpha or beta decay. This system can provide useful data for identifying radionuclides when the system is coupled to a multi-channel analyzer. For gross counting, samples (e.g., smears and filters) can be placed directly into a liquid scintillation counter (LSC) vial with a liquid scintillation cocktail and counted without

preparation. Certain samples contain more complicated matrices that require chemical separation prior to being placed and counted in LSC vials. Calibration sources are also kept and counted in these vials; hence, the geometry of the source and the sample with respect to the detector are similar.

5.6. Sample Measurement Using Gamma Spectroscopy

5.6.1. In-Situ Gamma Spectroscopy

In-situ gamma spectroscopy (ISGS) systems combine the peak resolution capabilities of laboratory methods with instrumentation that is portable and robust enough to be used under field conditions. These solid state systems can perform quantitative, multi-channel analysis of gamma-emitting isotopes in both solid and liquid media over areas as large as 100 m^2, facilitating spectrographic analyses that allow the user to identify constituentradionuclides and differentiate them from background radiation. ISGS system measurements can also provide thorough coverage within broad survey areas, minimizing the risk of failures to detect isolated areas of elevated radioactivity, which could potentially be missed when collecting discrete samples.

ISGS semiconductor systems can require a full day to be installed. These systems often require 1 h for physical setup, 6-8 h for the semiconductor to reach the appropriate temperature operating range, and another hour for quality control measurements. Count times obtained using ISGS semiconductor systems tend to exceed those associated with simpler detector systems for static measurements; however, this may be offset by enlarging the field-of-view. A measurement time of several minutes is common, depending on the intensity of the targeted gamma energies and the presence of attenuating materials.

ISGS semiconductor systems require calibration for their intended use. Whilst ISGS semiconductor systems can be calibrated using traditionally prepared radioactive sources, several ISGS systems offer software that allows the user to calculate efficiencies by entering parameters such as elemental composition, density, stand-off

distance, and physical dimensions. The supplied geometry templates assist in generating calibration curves that can be applied to multiple collected spectra. The high resolution of these systems, coupled with their advanced electronic parameter controls, allow them to overcome issues related to source-to-detector geometry and produce quantitative concentrations of multiple radionuclides in a variety of media (e.g., soil, water, air filters). Because ISGS systems integrate all radioactivity within their field of view, lead shielding and collimation may be required to "focus" the field-of-view upon a specified target for certain applications.

5.6.2. High Purity Germanium (HPGE) Detection

High purity germanium (HPGE) detection systems are laboratory-based and consist of a germanium detector connected to a cryostat (either mechanical or a Dewar of liquid nitrogen), a high-voltage power supply, spectroscopy grade amplifier, analog-to-digital converter, and multi-channel analyzer. This system offers a high resolution for peak energies and is capable of identifying and quantifying individual gamma peaks in complex spectra. It is particularly useful when a sample contains multiple gamma-emitting radionuclides which must all be identified and quantified.

Geometry considerations are of upmost importance for spectroscopic gamma analyses. The sample's position on the detector may significantly affect the analysis results, depending on the size and shape of the germanium crystal. Moreover, the instrument must be calibrated with a source of the same physical size, shape, and weight as the samples to be analyzed. Discrepancies between the volume or density of the sample and the source may introduce additional uncertainties to the analysis results. Sample homogeneity is a critical factor in gamma spectroscopy analyses, particularly for relatively large samples. For example, the settling of sediment during the analysis of a turbid aqueous sample will generate a high bias from any activity contained in the solid fraction. Likewise, the positioning of areas containing elevated activity in a solid sample will produce a bias in the overall sample activity (i.e., the activity will be disproportionately high if the particle is located at the bottom of the sample, and disproportionately low if located at the top of the sample).

5.7. Sample Measurement Using Alpha Spectrometry

The alpha spectroscopy with multi-channel analyzer system consists of an alpha detector housed in an evacuated counting chamber, a bias supply, amplifier, analog-to-digital converter, multi-channel analyzer, and computer. Samples are placed at a fixed distance from the solid-state partially implanted silica (SIPS) alpha detector for measurement. The multi-channel analyzer yields an energy spectrum that can be used to both identify and quantify radionuclides. The overall properties of the instrumentation permit excellent peak resolutions, although this technique often requires a complex chemical separation to obtain optimal results. The radiochemical separations have been simplified in recent years through the application of chromatographic and resin-specific techniques. Once separated, the radionuclide fractions are prepared in thin geometries suitable for measurement by the SIPS detectors, using either micro-precipitation or electrodeposition techniques.

The sample geometry (e.g., the lateral positioning on a detector shelf) in certain detectors may represent a small source of additional uncertainty. Uncertainty in the preparation of the actual calibration standards, as well as the applicability of the calibration standards to sample analysis, should also be considered.

5.8. Sample Measurement Using Inductivley Coupled Plasma Mass Spectrometry (ICP-MS)

Non-radiometric techniques such as mass spectrometry have also been used to determine low-level concentrations of radionuclides. Inductively coupled plasma mass spectrometry (ICP-MS) has become the most important of the various mass-spectrometric techniques available (Meck et al., 2009). Owing to the low mass abundance of certain radioisotopes, mass spectrometric techniques are limited to radioisotopes with half-lives exceeding 70 years (MOE, 2019). The sample processing techniques follow those of alpha spectrometry, in which spectral effects are eliminated using the same careful radiochemical separations, to leave the radionuclide fractions in a form suitable for mass

spectrometry.

5.9. Analytical Quality Assurance and Control

Quality assurance and quality control are integral components of good quality management and should be integrated into all radionuclide monitoring programs. For quality assurance and control in radio-analytical procedures, an effective laboratory quality management system should incorporate several components, including proficiency tests and inter-laboratory comparisons; blank samples, to prevent unexpected sources of activity in reagents or cross-contamination of samples processed in parallel analysis of reference materials; analysis of duplicate samples; and accreditation by an appropriate accreditation body.

Data Quality and Control : **6**

The laboratory handling of emergency monitoring samples should be based upon direction from the incident command. The direction to be followed may be similar to the following:

- Analyze the highest-activity concentration samples first, or
- Analyze the lowest-activity concentration samples first, or
- First analyze those samples whose alpha activity concentrations (obtained by screening) exceed a chosen action level, or through use of some other characteristic that the laboratory can measure with their screening instrumentation.

Measurement Quality Objectives (MQO) represent statements of performance objectives or requirements for the selected method's performance characteristics. The performance characteristics of a method can include the following:

- uncertainty,
- detection capability,
- quantification capability,
- applicable concentration range,
- specificity, and
- robustness.

The MQOs, alongside other analytical requirements, serve as the basis for the laboratory's choice of method under a performance-based approach. The laboratory should possess performance data to demonstrate the method's ability to achieve the project-specific MQOs.

Provision should be made for the establishment of a radiation monitoring and assessment center at which the efforts of all teams conducting emergency monitoring and assessment are coordinated in accordance with the protection strategy. The

arrangements should include a system to efficiently collect and integrate monitoring data from different organizations. The effectiveness of these arrangements should be evaluated in exercises that simulate response conditions.

Population Monitoring and Individual Dose Assessment :7

7.1. Introduction

Measuring the received doses of staff, emergency workers, and the general public after a nuclear or radiological emergency is crucial to realizing an effective response. This chapter discusses the processes of population monitoring and individual dose assessment.
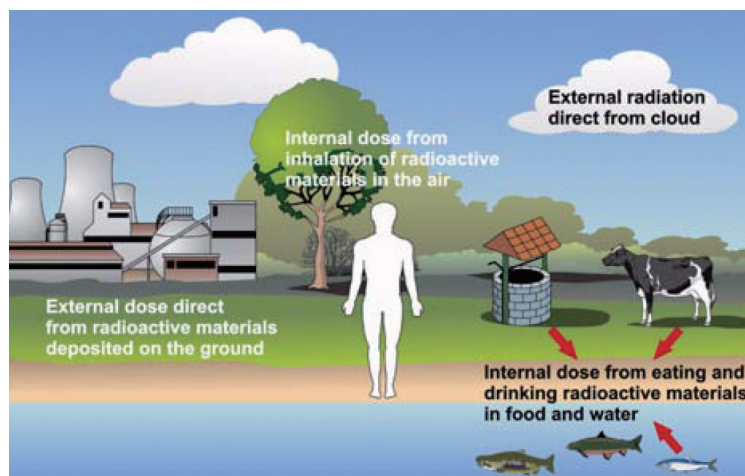
7.2. Exposure Pathways

Exposure pathways are the routes of exposure of the radionuclides as a source to the populations of concern. Radiation exposure can occur from radionuclides outside the body (external exposure) and within the body (internal exposure) via various exposure pathways (ICRU, 2015). In the case of a major nuclear accident, the radionuclides are released into the environment from the site of the accident as airborne materials and effluent discharge. The radionuclides in air are eventually deposited on the ground and remain there or are immigrated with time by weathering effects. The major exposure pathways are as follows: (1) external irradiation from the radioactive plume (cloudshine), (2) internal contamination via inhalation during exposure to the radioactive plume, (3) external irradiation from radionuclides in the ground (groundshine) and (4) internal contamination via ingestion of the contaminated food and drink items. These exposure pathways are illustrated in Fig. 4 (IAEA, 2006). The first two exposure pathways are dominant at the early phase following an accident, whereas the last two exposure pathways are important in the following phases. To implement effective radiation protection measures for the public, the exposure conditions must be understood in a timely manner based on dose prediction models using results of the environmental monitoring. The dose predictions can be validated by comparing with results of individual monitoring, such as personal dosimeters worn by persons and direct in-

vivo measurements. One important feature of these exposure pathways is that the contributing radionuclides to the doses are time-dependent in the case of a nuclear accident: the internal thyroid dose due to the intake of radioiodine (e.g., ^{131}I) at the early phase and the effective dose due to external radiations from radiocesium (e.g., [134]Cs, ^{137}Cs) on the ground at the late phase. Further information is described elsewhere (ICRU, 2015).

Figure 4: Main exposure pathways following a major nuclear accident (IAEA, 2006).

(taken from 2.3.4, "Considerations of Exposure Pathways" in the ICRU Report 92)

The monitoring data usually do not directly provide the radiation doses received by members of the exposed groups of the public; mathematical models and calculations are necessary to convert results from monitoring programs into doses. The models used to calculate doses depend on the exposure conditions, the available results of the monitoring, the purpose of the assessment, and the magnitude of the doses. The model should simulate the major pathways contributing to the exposure of the population groups under consideration.

In the emergency exposure situation, exposure conditions caused by the releaseof airborne radionuclides may change rapidly under changing facility conditions, meteorological conditions, and release rates; the decay of short-lived radionuclides;

and more. Computer-based ecological and dosimetric models can help elucidate these changing exposure conditions. Exposure from both airborne and deposited radionuclides should be considered in the dose-assessment process. In existing exposure situations, exposure pathways are typically well defined and unlikely to vary rapidly. External exposure and the ingestion of agricultural and/or natural foodstuffs containing radionuclides may contribute substantially to the doses received by the public. Owing to the gradual penetration of long-lived radionuclides into the soil, the importance of resuspension — and subsequently of the inhalation pathway — decreases with time.

(taken from 2.3.5. "Atmospheric Transport and Ground Deposition" in the ICRU Report 92)

Most nuclear or radiological emergencies that affect large areas involve a release of radioactive materials into the atmosphere. The atmosphere readily transports and disperses these radionuclides, creating many pathways through which human exposure may occur. The presence of people in a radioactive plume results in exposure by inhalation and submersion; meanwhile, in certain circumstances (e.g., when the plume does not touch the ground), the cloudshine from remote plumes can dominate. Inhalation results in internal exposure to all types of radiation emitted by incorporated radionuclides; submersion results in external exposure to beta and gamma radiation.

In the earliest phases of nuclear or radiological emergencies, the airborne transfer of radionuclides represents the primary exposure pathway. Once the release and subsequent atmospheric transport have been completed, the deposition-based pathways become the primary concern and may remain a concern for many years. The physicochemical transformations include the condensation of volatile compounds on natural aerosols and released particles, the coagulation thereof, the radioactive decay of released radionuclides, and the ingrowth of progeny radionuclides. Radioactive materials are removed from the atmosphere by downward components of the atmospheric transport. The dry deposition of aerosols and gasses on the ground surface is determined by the gravitational settling, turbulent diffusion, and surface adhesion and sorption; it depends on properties of the surface (e.g., the landscape) and upon the presence and types of vegetation, buildings, and water bodies. Wet deposition is caused by the

scavenging of aerosols and gases; it involves various processes, including nucleation, collision, dissolution, and evaporation. Types of wet deposition include below-cloud scavenging (washout) and in-cloud scavenging (rainout) for rain and ice phases, as well as cloud/fog deposition.

Typically, the airborne exposure pathways are of short duration; that is, unless continuous or multiple releases occur, the air contamination persists for only a short period of time (i.e.,days to weeks). The material deposited on the ground can become airborne again through a process called resuspension. The inhalation of resuspended material is not typically a dominant pathway of exposure for the majority of members of the public during the early phase of a nuclear emergency. However, resuspension should be considered for individuals involved in activities associated with intensive dust formation, such as plowing the contaminated soil; this pathway can become more important during the intermediate and late phases of an emergency.

(taken from 2.5. "Monitoring Data for Radiological Assessments" in the ICRU Report 92)

Information regarding the concentrations of radionuclides in the environment is required for calculating dose projections, implementing protective actions, identifying and evaluating areas for remediation, and refining dispersion models. The radionuclides may be the result of an initial release into the environment (e.g., radioactive cloud, deposition of the cloud onto the ground, or the release of liquids); alternatively, they may originate from secondary processes (e.g., resuspension, washout, or decontamination).

(taken from 2.5.1. "Data for Assessments of External Exposure" in the ICRU Report 92)

External exposure assessments in an emergency exposure situation will generally include assessments of exposure originating from the plume and ground depositions; in certain circumstances, these assessments can also include direct exposure from the source/facility where the emergency occurred. In the emergency situation, the external exposure arising from radionuclides present in the plume can be derived from dose-rate measurements or airborne-radionuclide-concentration measurements. Where possible, it is important to sample and measure the radiologically significant radionuclides in

the plume. The external exposure from radionuclides deposited on the ground can be derived from dose-rate measurements taken after the plume's passage; it can also be derived from activity concentrations measured in environmental samples (e.g., soil and grass). External doses arising from the deposition of radioactive materials are typically calculated for a limited time period (several hours to several days); this is consistent with the implementation of urgent protective actions (sheltering or evacuation). The effect of shielding from building structures may be taken into account, provided that data are available and that sheltering has been effective.

In the existing exposure situation, a set of dose-rate measurements taken at various inhabited locations ⸺ both outdoors and indoors ⸺ can be directly used to assess the external doses. Asan alternative source of monitoring data, the soil deposition levels of particular radionuclides in the assessment area can be used to estimate external doses. With the use of radionuclide-specific coefficients, these data can be converted into dose-rate values above undisturbed ground (e.g., lawns), plowed soil, or solid surfaces (e.g., asphalt or concrete).

(taken from 2.5.2. "Data for Assessments of Internal Exposure" in the ICRU Report 92)

In the emergency exposure situation, the doses arising from internal exposure should be primarily derived from inhalation measurements, because this pathway is important for the implementation of urgent protective actions (e.g., shielding, evacuation, and prophylaxis with stable iodine). The internal dose arising from inhalation or radionuclides present in the plume can be derived from the results of environmental monitoring (i.e., measurements of radionuclide activity concentrations in the air) or from model results, by using source monitoring. Both the radionuclide composition and the physical and chemical forms of airborne radionuclides can strongly determine the assessed inhalation dose. Early intake of radionuclides via food and drinking water can also substantially contribute to the dose received by the public in emergency exposure situations, particularly the thyroid dose resulting from the intake of radioiodine. Because randomized food sampling in the first few weeks may be complicated by emergency disruptions, thyroid dose assessments based upon food and drinking water monitoring

should be complemented using measurements of ^{131}I in human thyroids. For the thyroid dose assessment, special attention should be afforded to the most vulnerable population group: children and adolescents.

In the existing exposure situation, the internal doses imparted to population groups from the ingestion of contaminated food and/or drinking water should be determined on the basis of environmental monitoring data, using a simple radiological assessment model that accounts for the origin and consumption rate of particular food products and the seasonal variations in relevant parameters. Regularly obtained data regarding activity concentrations in locally produced agricultural foodstuffs can be directly applied to assess the annual intake and associated committed dose. To validate internal dose assessments based on food intake, whole-body count data can be used to substantially reduce dose uncertainty. The contribution of inhalation to the internal doses received by a representative person in the existing exposure situation may be substantial for radionuclides with low solubility and low food-chain mobility (e.g., actinides and transuranics), especially for persons working in the open air and in dusty conditions. The set of periodically obtained data regarding activity concentrations in air can be directly used to assess the annual intake and associated committed dose. If measurement data are unavailable or insufficient, activity concentrations in the air can be roughly estimated from soil deposition rates using a resuspension model.

7.3. Population Monitoring

In the event of a significant radiation emergency, population monitoring — also referred to as "public monitoring" — is essential. People in the affected area maybecome externally or internally contaminated by radioactive materials. Population monitoring refers to the process of monitoring the contamination of people in the affected area; it commences soon after the start of the emergency, and it continues until all potentially affected people have been monitored and evaluated.

Population monitoring aims to identify individuals who require immediate medical

attention or medical treatment for contamination or exposure — to thereby facilitate decision making regarding urgent protective actions or other response actions — and to provide reliable information regarding the potential radiation risk through contamination/exposure (Roberts, 2009; Tsuruta et al., 2014).

Generally, initial population monitoring can be performed using appropriate hand-held radiation instruments. For population monitoring, the monitoring of individuals for external and internal contamination provides more reliable information. It should be noted that normal radiation monitoring procedures cannot determine whether a person has been exposed to external radiation sources. In this section, the triage of external and internal contamination for population monitoring is introduced.

7.4. Triage of External Contamination

Triage of external contamination is performed to assess the quantity of radioactive materials on the surface of human body (i.e., on skin, clothing, etc.). External contamination on the body can be spread to other people, places, or items. Furthermore, the radioactive materials remaining on skin or clothing can result in internal contamination through release into the air.

The monitoring results for external contamination allow decisions to be made regarding the need for the external decontamination of people. The rapid triage of external contamination allows decisions to be made regarding which type of further monitoring program, if any, will be required. Emergency response personnel entering an accident area must be checked for skin and protective clothing contamination upon leaving the contaminated area. This should be conducted at a designated contamination monitoring area or public reception center. The monitoring team should identify an appropriate location where the background radiation level is not significantly high. Sports centers, schools, warehouses, stadiums, and community centers can be used for this purpose.

The medical care and treatment of conventional injuries takes precedence over decontamination (IAEA, 2005; Tsuruta et al., 2014). The medical treatment or transport

of injured people should not be delayed by monitoring. Next, priority should be given to those considered to be the most potentially contaminated. Monitoring results (including personal information) should be recorded for each person monitored. If the number of potentially contaminated people greatly exceeds the monitoring resources available, the monitoring approach must be adapted to screen people and identify those most contaminated. External contamination monitoring is performed using either portal monitors or hand-held instruments.

7.4.1 Triage using portal monitors

Portal monitors are specialist monitoring equipment designed to be easily assembled, disassembled, and transported to locations, where they are used to screen individuals for the presence of radioactive materials, as shown in Figure. 5 (IAEA, 2018).

**Figure 5:** Portal monitor [Canberra Inc. (Left) and IAEA (Right)]

Portal monitors facilitate the rapid monitoring of large numbers of people. They can be operated in a walk-though mode for high throughput; however, greater sensitivity is achieved if the person remains stationary for a fixed time (typically 10 s). Rapid throughput is achievable (up to 300 people per hour) (ICRU, 2003). Conventional portal monitors detect gamma and beta radiation using plastic scintillation detectors, and the measurements obtained are typically of count rate (in cps or cpm). They are unable to identify

radionuclides. Therefore, portal monitors are most suitable for the initial triage of external contamination.

7.4.2 Triage using hand-held instruments

Hand-held instruments may be used to scan for external contamination on skin and clothing across the entire body. Monitoring personnel assess people coming in from contaminated areas, by slowly passing a probe over the entire body surface; this is known as the "frisking method." Figure 6 illustrates the process of external contamination monitoring using hand-held instruments.

Figure 6: Triage of external contamination using hand-held instruments (courtesy of KIRAMS)
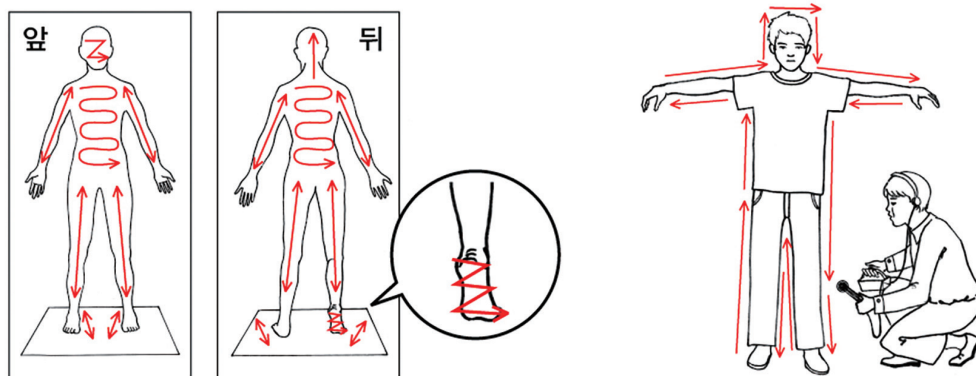
The detailed steps for triage using hand-held instruments are as follows (IAEA, 2005; US NCRP, 2008a):

(a) Place the probe ~1 cm from the person's body, being careful not to touch the body;

(b) Slowly move the probe at a speed of 3-5 cm/s over the areas to be monitored;

(c) Monitor the entire surface of the body, from the top of the head to the bottom of the shoes;

(d) Monitor wounds using a specialized wound probe (if necessary);

(e) Pay particular attention to the orifices of the face (e.g., the eyes, nose, mouth, and ears);

(f) Pause the probe at locations most likely to be — or identified as — contaminated; and

(g) Record the monitoring results on the pre-established recording sheet.

The background levels of the radiation instruments employed should be recorded both prior to and periodically during the monitoring process. If external contamination is confirmed or suspected, further evaluations should be conducted via follow-up monitoring after the removal of outer clothing. Monitoring personnelshould periodically perform operational checks on the monitoring instruments, including battery checks. The instruments should be protected using a plastic bag, to prevent their becoming contaminated.

7.5. Triage of Internal Contamination

In case of suspected internal contamination, screening should be performed as early as possible. The purpose of the initial triage of internal contamination (using portable instruments) is to classify people according to their levels of internal contamination; this is used to guide decisions regarding further actions. In particular, the monitoring results allow decisions to be made regarding the need for treatments to reduce internal exposure (e.g., decorporation therapy) (US NCRP, 2008a).

The monitoring of internal contamination levels should be performed using portable, transportable, or hand-held instruments. Internal contamination monitoring is typically performed for anyone initially identified as being externally contaminated, or for individuals identified as being at riskowing to their proximity to a radiation emergency involving airborne radionuclides. Before internal contamination measurements, the external contaminations of monitored individuals should be removed, to prevent further internal contamination of the publicand to reduce uncertainty in the internal contamination results. Whole-body or thyroid measurements should be applied with hand-held instruments for the in-vivo monitoring of internal contamination.

7.5.1 Whole-body measurement with portable devices

Whole-body measurement, as a direct measurement procedure for internal contamination, is the most accurate bioassay method for assessing internal contaminations from radionuclides that emit strongly penetrating gamma radiations. However, such counting systemsare not always available, because the majority of them are fixed-type and located at specialist facilities or laboratories. Rapid screening using portable devices such as survey-meters, dose-rate meters, and gamma-spectrometers should be made available atthe designated monitoring area. If the measured results exceed the pre-determined screening level, additional measurements should be performed, because uncertainties in portable-device-based rapid screening generally exceed those of fixed-type counting systems equipped with highly sensitive detectors. Figure 7 shows the whole-body measurement process for triage of internal contamination, realized using a portable gamma detector.

Figure 7: Whole-body measurement (courtesy of KIRAMS)

7.5.2 Thyroid measurement with hand-held instruments

Thyroid measurements should be made to identify internal radioiodine contamination. Portable gamma-spectrometers or dose-rate meters can be used for thyroid measurements. The International Atomic Energy Agency (IAEA) recommends an OIL

of 0.5 μSv/h — that is, the ambient dose equivalent rate measured by a device in contact with the skin in front of the thyroid — as the thyroid screening criterion (IAEA, 2013; IAEA, 2017a, 2017b). This monitoring method can be used to identify individuals warranting registration and medical follow-up due to the intake of radioiodine. The thyroid must be monitored within the first week, to determine whether an individual has inhaled or ingested sufficient radioiodine to warrant medical follow-ups (IAEA, 2013; IAEA, 2017a, 2017b). It is difficult to identify these individuals using dose-rate meters during the first week after radioiodine intake.

Portable gamma spectrometers [including scintillation (e.g., NaI or CsI) and semi-conductor (e.g., germanium) detectors] can be used to quantify the activity of radioiodine in the thyroid. For such quantification, the gamma-spectrometer calibration process must take account of the measurement geometry prior to thyroid measurement. In this step, the measurement geometry (i.e., the distance of the detector from the neck surface) should be consistent, to obtain accurate thyroid measurements. Figure 8 shows an example of thyroid measurement using a portable gamma spectrometer.

Figure 8: Thyroid measurement (courtesy of KIRAMS)

A suitable location for monitoring should be determined, with as low a level of background radiation as possible. The detection limit (i.e., the MDA) of the measurement system should be determined from the background radiation levels and counting efficiency, depending on the measurement geometry and configuration.

External Dose Assessment :**8**

8.1. Introduction

The primary objective of dosimetry during the early stages of a radiological accident is to evaluate the radiation doses of individuals and help determine the initial medical treatment by providing exposure information to medical staff (ICRU, 2015). There are multiple approaches for retrospective dosimetry for patients with acute exposure to ionizing radiation, including biological dosimetry, physical dosimetry, bioassays, and neutron activation; the method employed depends on the exposure condition and type.

8.2. Physical Dosimetry

Among the approaches described above, some are used to evaluate the absorbed dose by patients using personal dosimeters worn during radiological accidents; opportunistic dosimetry; exposure-situation reconstructions; and in-vivo and in-vitro bioassays.

Electron paramagnetic resonance (EPR), thermoluminescence (TL) andoptically stimulated luminescence (OSL), are various physical dosimetry methods as opportunistic dosimetry employed when existing dosimetry data are unavailable or inadequate to evaluate the exposure information in a radiological accident. Furthermore, alternative objects are used to retrospectively assess exposures in humans. Under these dosimetry methods, the dose delivered internally and externally to the human body is evaluated using biologically derived materials (e.g., tooth enamel) and objects or devices possessed by the exposed patients (e.g., clothing and cell phones) (Roberts, 2009). Therefore, additional calculations may be required to convert measurements to the dose of a specific organ/tissue of the body.

8.2.1 Electron Paramagnetic Resonance (EPR) dosimetry

EPR is a spectroscopic technique that has been used to evaluate unpaired electrons

in materials since the 1950s. Unpaired electrons are present in incompletely filled atoms, atoms with orbitals, molecules, and molecule ions. They can be produced by an endogenous factor or via processes such as oxidation.

Ionizing radiation can produce radicals in soft tissues, belongings, and household substances, including glass, plastic, and keratinous tissues. Except calcite-containing tissues, these substances do not have sensitive and stable EPR signals. However, they can be very useful for dosimetry during the early stages of radiological accidents, because the corresponding samples can be readily collected. Figure 9 illustrates the energy resonance absorption peak commonly obtained using EPR equipment. The difference between the EPR signals at the negative and the positive peaks is called the amplitude of the derivative signal (ICRU, 2015).

Figure 9: A single-line electron paramagnetic resonance (EPR) signal in the X-band, as well as the descriptive parameters for amplitude, line width, and base line
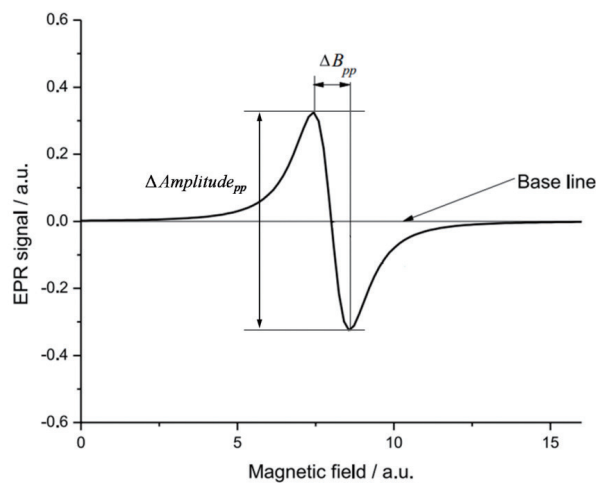
Calcified Tissue

Hydroxyapatite (HAP) is found in calcite-containing tissues in the human body. As typical HAP-containing tissues, tooth enamel, dentine, and bone contain 97%, 70%, and 50% HAP, respectively. Ionizing radiation provides sufficient energy to generate radicals in the

Co_3^{2-} molecules in HAP, and these Co_3^{2-} molecules serve as an effective radiation marker. Tooth enamel is a calcified tissue which exhibits the greatest HAP content across the human body, followed by dentin and bone. Compact bones are less dense and contain more organic substances than tooth enamel crystal. Tooth enamel is the gold standard specimen in EPR-based retrospective dosimetry. Detailed information is provided in the ICRU Report 68 (ICRU, 2001).

The method using tooth enamel for retrospective dosimetry has a disadvantage because it is often difficult to collect a tooth sample immediately after the accident occurs. Therefore, this method has been used mainly for epidemiological studies in which sufficient time was available to collect tooth samples. To overcome this limitation, recent research has sought to evaluate accident doses without tooth extraction, by developing an in-vivo evaluation method using L-band measurements (Yamaguchi et al., 2021).

Keratinous Tissue

Nails and hair can be easily collected for retrospective dosimetry. The primary ingredient of nails is a hard, keratinous structure consisting of crystalline fibers and amorphous protein matrices (Dalgarno et al., 1989; Sato, 2016). The fiber structure consists of alpha (α) — helical peptide chains wrapped in stable microfibers, and the matrix phase is stabilized with cysteine — cysteine disulfide bridges. Thus, the keratinous tissues in nails are connected via secondary bonding (Reyes et al., 2008). Hair, similar to nails, is mainly composed of alpha keratin, but also contains melanin [52]. Melanin is an amorphous insoluble, heterogeneous, dark, biological polymer containing a population of intrinsic, semiquione-like radicals (Arnaud et al., 1983).

Nails were first considered as a specimen for retrospective dosimetry in 1950. Recent efforts have been directed to apply the fingernail-EPR dosimetry method to the treatment of radiotherapy patients who can be potentially exposed to high doses (Thompson, 2000). Fingernail specimens are more suitable than the other biological specimens (e.g., teeth and bones) used in EPR retrospective dosimetry, because they can be collected painlessly; furthermore, they are useful for accidents involving local exposure on hands. However, additional research is required to address the technical limitations of the

fingernail-EPR dosimetry method compared to other EPR dosimetry methods, including the effects of moisture and the fading of radiation-induced signals. Hair presents complications as an EPR dosimetry material. The intrinsic semiquinone-like EPR signal of melanin is extremely intense and overlaps with the RIS (ICRU, 2015). Owing to its advantages of easy sample collection, several studies are currently underway on the application of hair as a potential specimen in retrospective dosimetry. The presence of melanin pigments in hair makes the radiation-induced signals (RISs) short-lived and difficult to distinguish. Further research is required for application of hair-based retrospective dosimetry.

Sugar

A common household substance, sugar — which comes in various types, including glucose, fructose, lactose, galactose, and mannose — can be found in numerous foods and medicinal products. All sugars can be used as EPR specimens in the event of a radiation emergency. The characteristics of sugar include low background signals, linearity over a wide range of doses (up to 10 kGy), and relatively stable and durable RISs (US NCRP, 2008b). The main limitations in the use of sugar for retrospective dose assessment for individuals are the availability of sugar samples at the site of the accident and the assignment of individual doses to the victims based on the measured absorbed dose in the samples (ICRU, 2015).

Glass

Glass is a non-crystalline amorphous silicon dioxide solid (silica). The basic structural unit of silicon is a tetrahedron in which each silicon atom forms four bonds with oxygen. Silicon dioxide solids are typically manufactured using alkali-oxides and dopants such as lead, boron, barium, cerium, and manganese (ICRU, 2015). Glass is readily available in everyday life and found in various personal items. The most commonplace glass is soda-lime silicate glass, which is a mixture of silicon dioxide, sodium, and calcium oxide. Representative glasses used in retrospective dosimetry include liquid-crystal displays (LCDs) and cell-phone touchscreens. The characteristics of EPR signals can vary between specimens made of different types of glass (Juniewicz et al., 2020).

Glass can be useful for initial retrospective dosimetry evaluations following an acute radiological accident, owing to its ubiquity. However, the glass must be broken prior to EPR measurement (Bassinet et al., 2010). Additionally, the detection limit may be evaluated too high,owing to the variability in extraneous background signals associated with light exposure (Fattibene et al., 2014). Therefore, research should be continued for developing methods of (i) extracting EPR signals whilst eliminating unnecessary signals and (ii) evaluating signals without breaking the glass specimens.

Cotton

Unlike other specimens, fibers can be easily collected anywhere and are useful for mapping the absorbed dose distribution across the entire human body. Among the specimens used for EPR signalevaluation, cotton is the most widely studied in retrospective dosimetry (Symon et al., 1995). Cotton fibers are composed of polysaccharide chains arranged in crystalline structures surrounded by amorphous porous substances. Several studies have reported positive results regarding the use of cotton fibers in retrospective dosimetry; however, existing studies are insufficient to achieve accurate results when using cotton in emergency dosimetry (Jasinska et al., 1970; US NCRP, 2008b). For example, solar exposure, residual detergent molecules produced by laundering, moisture content and the contamination with dust can complicate EPR fiber analysis. Therefore, proactive research should be conducted to apply cotton fibers in dosimetry for radiation emergency situations (Viscomi, 2011).

Plastic

Plastic is a generic term describing a broad range of synthetic, semi-synthetic, organic, and amorphous solids; these typically contain other substances to improve performance or reduce production costs. Plastic isreadily found in everyday objects, including cell phones, credit cards, buttons, watches, and glasses (Sholom et al., 2010). It can be used as a retrospective dosimetry specimen in radiation emergencies, because a small piece of plastic can be cut without requiring a particular specimen collection and preparation procedure (Trompier et al., 2010). However, the diverse types and compositions of plastic

make it difficult to predict whether a particular type of specimen is suitable for radiation emergency situations.

8.2.2 Luminescence Dosimetry

TL and OSL are physical phenomena in which crystals featuring internal defects are altered under exposure to radiation or high-energy particle radiation; the electronic charges excited by radiation are trapped in the defect site and are released in the form of light under thermal or optical stimulation. Analysis of the emitted light at given stimulation conditions reveals the state of the crystals and their exposure history, which can be used in various fields of dosimetry, including environmental impact assessment, cosmic radiation monitoring in aviation, and radiation management of medical workers as well as patients.

Under thermal or optical stimulations, trapped charges induced by radiation exposure can be excited to the conduction band, and they are recombined at recombination centers, thus releasing luminescence proportional to the accumulated dose. The principles of TL and OSL are schematically illustrated in Figure 10 (ICRU, 2015).

Figure 10: Energy banddiagram in a (crystalline) insulator, to illustrate the TL/OSL phenomenon. T stands for (electron) trap and R for (hole) recombination center. Both metastable energy states are placed in the forbidden zone between the valence band (VB) and the conduction band (CB). The black and open circles indicate the trapped electron and trapped hole, respectively. Note that the excitation and recombination pathway can also proceed in the opposite direction, with holes being released from the hole trap into the VB and recombining at the electron trap. The direction of the process depends on the materials and defects involved. In this sense, the assignment of T and R to the electron and hole trap is arbitrary (ICRU, 2015).
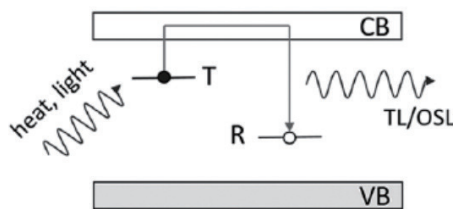
In recent years, intensive research has been conducted into various types of materials for TL and OSL techniques; however, no standardized retrospective dosimetry protocol has been suggested for emergency applications (McKeever et al., 2020). The luminescence materials used as commercial dosimeters are highly sensitive to radiation and can provide precise measurements over a wide range of dose. Furthermore, because radiation exposure situations can vary widely, non-biological objects carried by individuals should be considered for radiation emergency dosimetry alongside biological materials such as teeth and fingernails. In the following section, the luminescence properties of several personal items and general materials proposed as emergency dosimetry specimens are described.

Chip Cards

In many countries, electronic chips are used in various SIM cards, including credit cards, electronic passports, and electronic identification cards (ICRU, 2015). On these chip cards, contact-based modules are located on the front of the card covered by a metal contact; they allow electronic information to be read. The plastic-laminate-covered UV-curable epoxy inside the chip can be extracted and used for OSL dosimetry.

Current research suggests the possibility of using UV-curable chip card modules in radiation emergency dosimetry (Kim et al., 2020; Woda et al., 2009). Further research is required on filler substances and contactless modules in chips, to develop practical dosimetry techniques.

Surface-mount Devices and Integrated Circuits in Mobile Phones

Mobile phones have become popular amongst the general public since their introduction in the late 1990s. This has made mobile phones useful as personal dosimeter devices. Most people own and carry a mobile phone near to their body. Moreover, mobile phones are owned for several years before replacement. Surface-mount devices (SMDs) on circuit boards contain highly standardized dosimetric materials in the form of resistors and inductors, and their luminescence properties in dosimetry are nearly completely standardized (Woda et al., 2011). In SMDs, the resistors and inductors are composed of

alumina materials (Al_2O_3) more than 95% that exhibit TL and OSL signals when they are exposed to radiation (Trompier et al., 2010).

Integrated circuits offer a significant potential for application in retrospective dosimetry [Sholom et al., 2014). Mobile phones offer many characteristics of an ideal personal dosimeter, such as a very high utilization rate compared to other materials used in retrospective dosimetry (Sholom et al., 2016). However, a non-destructive approach is required to increase the acceptance of the material obtained from patients because the extraction of the material inevitably accompanies damage to a mobile phone.

LCD Display and Touch-screen Glass of Mobile Phones

Because glass is generally a radiation-sensitive material, the mobile-phone LCD display composed of various glass layers is expected to work effectively for emergency dosimetry (Discher et al., 2020; Kim, 2019). Compared to electronic components on circuit boards (which become smaller over time), glass displays tend to become larger, thereby providing abundant specimens. Additionally, touch-screen displays can be sampled at a lower cost than the cost of replacing an entire phone to extract the circuit board. Touch-screens in existing smartphones typically consist of an external protective glass layer and either an LCD or active-matrix organic light-emitting diode display on a glass substrate. In this report, the latter is referred to as "display glass," and the external protective glass is referred to as "touch-screen glass."

Display glass is one of the most efficient samples in the luminescence method because of their high radiation sensitivity and low detection limits of less than several tens of m Gy (Discher et al., 2013; Mrozik et al., 2014). At present, touch screen glass is also one of the highly studied materials using different protocols such as EPR and TL techniques. However, the largest budget on the detection limit is the zero dose (a dose converted from non-radiation signals), which contributes variably to the evaluated dose. Advanced protocols are required to minimize or completely eliminate these variable signals.

Biologically-derived and -related Materials (Teeth and Dental Repair Ceramics)

Tooth enamel can be used in an EPR-based dosimetry method that utilizes in-vivo and in-

vitro evaluation specimens. The EPR properties of these materials have been extensively and continuously studied (Christodoulides et al., 1971; Jang et al., 2018). Additionally, attempts are underway to make the materials required for luminescence-based dosimetry methods. These attempts may offer a simple and easy radiation evaluation technique for application in real-world situations. The ultimate goal is todevelop a small high-sensitivity portable system that can evaluate the dose in in-vivo teeth. Although dental ceramics are not biological materials, they are included in this section because they are important in dentistry and exhibit promising results inthe in-vivo and in-vitro dosimetry fields. Dental ceramics are available in various chemical forms and can be grouped into four basic composition categories: glass-based (mainly silica), glass-based with fillers, crystalline-based with glass fillers (mainly alumina), and polycrystalline solids (Ekendahl et al., 2017). The properties of TL and OSL from some materials have been investigated. Since OSL studies on tooth enamel were first published, significant advances have been made in device sensitivity andmaterial understanding. Given its high sensitivity, easy discoloration correction, and resilience to moisture, tooth-based OSL is a potentially attractive method for in-vivo radiation emergency dosimetry techniques (DeWitt et al., 2010).

8.3. Biological Dosimetry

When unplanned ionizing radiation exposure occurs, biological dosimetry based on cytogenetic assays can be conducted to estimate the dose absorbed by the individual to obtain useful information for medical management (IAEA, 2011). Among human specimens, peripheral blood lymphocytes are the most validated materials for biological dosimetry. For the assessment of radiation-induced damage to cells, the frequency of morphologically detectable chromosomal aberrations, i.e., dicentric chromosomes, translocations, premature chromosome condensation fragments and micronuclei, has been used as the measure in biological dosimetry in combination with an appropriate dose response curve (Fig. 1) (IAEA, 2011; Suto et al., 2013). Generally, the lower limit for dose estimation of whole-body exposure by cytogenetic assays as employed in many

laboratories is restricted to approximately 0.2 Gy (IAEA, 2011). Chromosome aberrations observed by cytogenetic assays can be caused by not only one's recent acute radiation exposure but also one's history of external, internal, whole-body, and partial-body radiation exposure in one's life by the time of peripheral blood sampling. Types and frequencies of chromosomal aberrations also reflect the total effect on the cells from all typesof radioactive sources one has been exposed to. Mathematic procedures exist to provide dose estimation after partial-body exposure or protracted exposure (IAEA, 2011). For accurate dose assessment in a radiation accident/incident, all efforts have been made in the past to collect and integrate information on state of radiation accidents/incidents and other data including physical dosimetry and medical examination as much as possible. In this section, cytogenetic assays whose protocols have been established by the International Organization of Standardization (ISO) are introduced.

Figure 11: An example of a dose response curve: dose–effect relationship between exposed dose with ^{60}Co gamma-ray irradiation and dicentric chromosome aberration yield (Suto et al., 2013, modified). The 95% confidence intervals are shown as dotted curves. Exposed doses can be estimated by applying a patient's chromosomal aberration yield to an appropriate dose–response curve prepared in advance from experimentally irradiated human peripheral blood lymphocytes
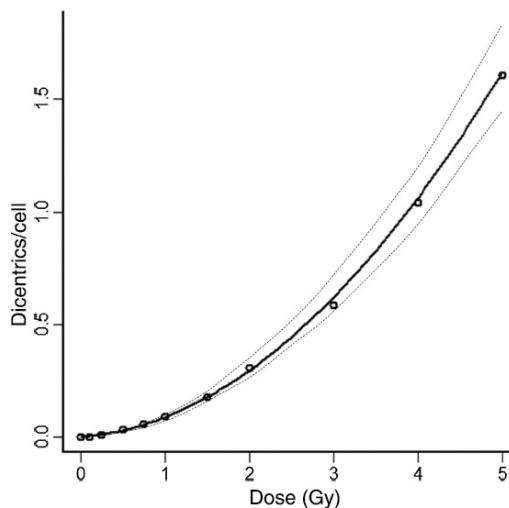
Dicentric Chromosome Assay (DCA)

A dicentric chromosome is an aberrant chromosome with two centromeres derived from the joining of parts from two broken chromosomes which are generally accompanied by one acentric fragment (Fig. 2). In the dicentric chromosome assay (DCA), dicentric chromosomes are detected in microscopic examinations using chromosome preparations stained with Giemsa stain solution (IAEA 2011). A tricentric chromosome is equivalent to two dicentrics and should have two accompanying fragments, while quadricentric will have three fragments, and so on. Likewise, a multicentric chromosome with N centromeres is scored as (N-1) dicentric chromosomes. The spontaneous frequency of dicentric equivalents is very low in the healthy general population (approximately zero to one dicentric per 1000 cells), which is one of the reasons why DCA is considered to be the 'gold standard'of biological dosimetry for radiation emergency medicine. Because dicentrics are unstable chromosome aberrations that disappear with time after exposure, this assay is particularly useful within one month after radiation exposure. Protocols of DCA and DCA for mass-casualty events have been internationally standardized and shared among lead laboratories of biological dosimetry (ISO 2014a, ISO 2008). This method wasactually used in the Fukushima Daiichi Nuclear Power Station (NPS) accident in 2011 (Suto et al., 2013; Kulka et al., 2018) using a microscopic image acquisition system equipped with automatic metaphase finding and capturing functions for rapid performance. The questionnaire sheet in the protocol (Annex B in Suto et al., 2013) was helpful to collect information regarding factors that affect the increase in the dicentric yield other than radiation exposure in the accident and thus are necessary for accurate assessment.

Figure 12: An example of a metaphase cell stained with Giemsa for Dicentric Chromosome Assay (DCA) (Suto et al., 2021). Two dicentric chromosomes and one centric ring are accompanied by three acentric fragments.
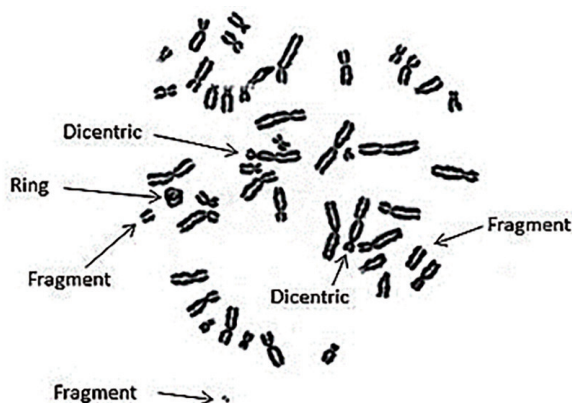
To prepare for large-scaleradiation accidents/incidents with mass casualty, domestic, regional and international networks for collaboration and technical harmonization are being established in the world (Kulka et al., 2018) including Asian countries (Jang et al., 2018). Advanced techniques are also being developed to obtain results more rapidly: i) automated dicentric detection software (Gruel et al., 2013; Li et al., 2019; Romm et al., 2013), ii) fluorescence in situ hybridization (FISH) technique using centromeric and telomeric probes for accurate detection of dicentrics (Shi et al., 2012; Suto et al., 2012), and iii) dicentric chromosome detection by imaging flow cytometry (Beaton et al., 2013).

Cytokinesis Block Micronucleus (CBMN) Assay

The cytokinesis block micronucleus (CBMN) assay is another established method for biological dosimetry (IAEA 2011). Micronuclei (MN) arise from acentric fragments or whole chromosomes that are not incorporated into daughter nuclei during cell division. They are observed as small spherical objects within the cytoplasm of the binucleated daughter cell under a microscope using Giemsa stain (Fig. 3) or fluorescent dyes. Because MN are unstable chromosome aberrations, this assay is particularly useful within one month after radiation exposure.

Although MN is not radiation-specific and its spontaneous yield is relatively high and increasing with age, MN scoring is much easier and less time-consuming than that of dicentrics. The CBMN assay is used in some networks (ISO, 2014b). Advanced

techniques are also being developed using imaging flow cytometry to detect MN rapidly (Wang et al., 2019).

Figure 13: An example of binucleated cells with two micronuclei stained with Giemsa. The figure is modified from Fig. 19 (IAEA, 2011)
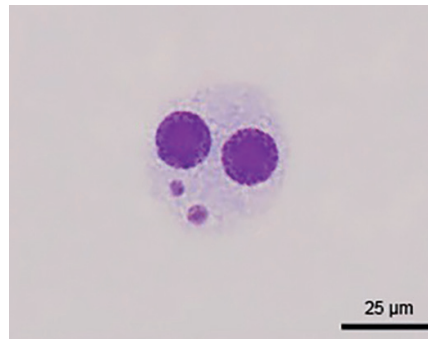

FISH Translocation Assay

Multicolor FISH using differentially colored chromosome-specific painting probes enables the detection of inter-chromosomal exchanges, i.e., aberrations among different chromosomes such as dicentric chromosomes, translocations, insertions, centric ring chromosomes, etc. (Fig. 4) (IAEA, 2011; Suto et al., 2015). Among inter-chromosomal exchanges, translocations are stable aberrations and are useful for retrospective dosimetry of past radiological accidents/incidents. The experimental protocol of the FISH translocation assay has been internationally standardized (ISO, 2019). Background frequencies increase significantly with age and can even vary between individuals of similar age and dose history (IAEA, 2011). ISO has recommended several methods of adjustment for age (ISO, 2019). Using the multicolor FISH method, dicentrics can also be detected. The detection efficiency by FISH is almost 100%, and dicentric detection by this method is useful to confirm conventional DCA results for low-dose exposure (Suto et al., 2015).

One of the actual applications of the FISH translocation assay is the retrospective dosimetry research project of restoration workers at the Fukushima NPS in Japan, who

engaged in working there for some time between March 14 and December 16, 2011, when the emergency exposure limit had been raised from 100 mSv/year to 250 mSv/year in effective dose (Suto, 2016; UNSCEAR, 2020). During that period, approximately 20,000 workers engaged in emergency work at the NPS, out of which 174 workers were considered to be exposed to radiation exceeding 100 mSv as effective dose. The predicted number of the subjects including those 174 workers is 787 at maximum in this project based on the FISH translocation assay.

Figure 14: Translocation between chromosomes 1 and 4 [tr (1; 4)] detected by three color FISH. Chromosome 1: red; chromosome 2: green; chromosome 4: yellow; other chromosomes: blue. This figure is modified from Supplementary Fig. 2 of Ref. 15. Counterparts of translocated chromosomes 1 and 4 and two sets of chromosome 2 show no detectable aberrations
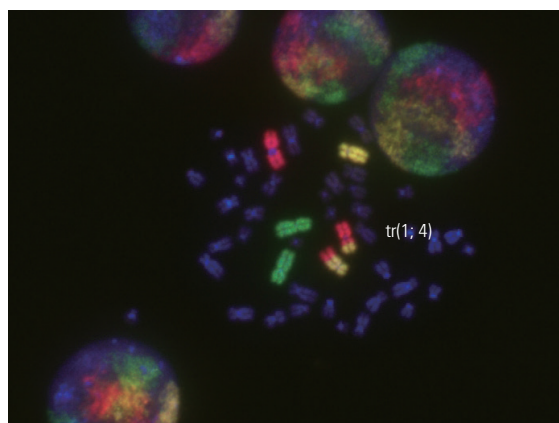
In summary, cytogenetic dosimetry has been useful because it provides estimated doses to support radiation emergency medicine thus far. Further progress in advanced technology for detecting chromosome aberrations and development of inter-laboratory collaboration by networking is still awaited.

Internal Dose Assessment

Unexpected internal contamination may occur because of radioactive material leakage or accidents in nuclear facilities. The internal contamination typically arises through inhalation, ingestion, wounds, or the skin. After an accident, internal contamination can occur in various indoor and outdoor environments, owing to the movement/spread of contaminants. Internal contamination can occur in various places where radionuclides or radioactive drugs are typically used (e.g., medical facilities, industrial sites, laboratories, and nuclear reactors).

One method of measuring internal contamination is via the in-vivo/in-vitro bioassay monitoring of samples obtained from human bodies (US EPA, 2012, 2019). In-vitro bioassay monitoring includes radioactivity measurements obtained from excretions, nasal smears, and blood and tissue samples. In the event of a radiation emergency, the analytical capacity, incident situation, and human/physical support scale must be identified to develop a permissible special protocol suitable for each organization. Furthermore, the collection of biological samples from patients — to facilitate treatment planning — should be decided upon after consultation with medical staff, and sample collection for other research purposes should be decided upon after consulting with the respective research institutions.

The radioactivity measurement of a biological sample is defined by a bioassay, and the types of bioassays include in-vivo and in-vitro measurement methods. The in-vivo measurement methods include whole-body counting (WBC) and organ counting (e.g., thyroid or chest counting); the in-vitro measurement methods include radioactivity measurements of the biological sample. The top priority in the treatment of an injured person is medical treatment. The medical treatment should be performed quickly and should ensure that the vital signs of the patient are clear. Health physics treatments (including the bioassay) can be performed once medical stability has been secured. Bioassays for radioactivity monitoring should be repeated over an extended period, because the retention/excretion trend should be examined according to the internal

contamination condition.

9.1. In-Vivo Bioassays

Bioassays are any procedures used to determine the nature, activity, location, or retention of radionuclides in the body by direct (in-vivo) measurements or by indirect (in-vitro) analyses of materials excreted or otherwise removed from the body (IAEA, 2004). In this section, in-vivo bioassays are mainly described.

In-vivo bioassays or direct measurements are a method to determine the activity of the radionuclide(s) of concern in the entire body or a specific organ (e.g., thyroid, lungs, liver) using a photon detector placed near the subject. The target radionuclides should be those emitting photons with a detectable energy range and sufficient yields. In principle, the radionuclides emitting only alpha or beta particles are inapplicable to in-vivo bioassays with a very few exceptions (e.g., bremsstrahlung measurements for beta emitters) (ICRU, 2003). Fig. 15 demonstrates the instruments used for in-vivo bioassays: whole-body counter (WBC) and thyroid monitor. WBCs are designed to measure gamma rays from the radionuclides distributed throughout the body or some specific organs in the trunk (e.g., the lungs, the gastro-intestine tract). Thyroid monitors are designed to measure radioiodine localized in the thyroid. The photon detector(s) equipped with these instruments is (are) arranged considering the sensitivity and/or the view of measurement for the subject, namely the measurement geometry. There are several types of WBCs depending on the measurement geometry, as shown in Fig. 15. The detectors mostly used for WBCs and thyroid monitors are NaI(Tl) scintillation detectors and hype-pure germanium (HPGe) semiconductor detectors. NaI(Tl) detectors are often used for WBCs with a screening purpose for a large number of subjects, whereasHPGe detectors are often used for WBCs for detailed measurements of a small number of subjects. Typical counting times for these two types of WBCs are a few minutes and 10-30 minutes, respectively. Some of the latter-type WBCs are installed in low-background shielding chambers.

Figure 15: Instruments used for in-vivo bioassays (taken from MOE, Japan)

Stand-up type WBC

Bed type WBC

Chair type WBC

Thyroid monitor

⭕ Detector
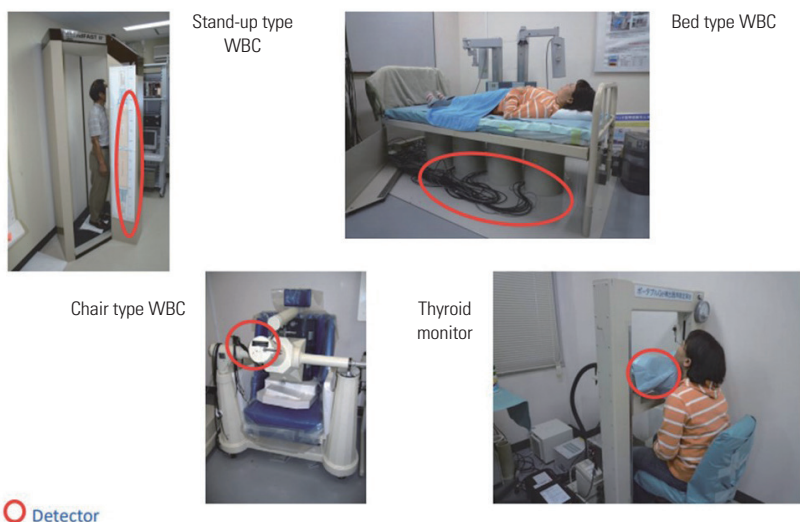
The instruments for in-vivo bioassays must be calibrated with anthropometric phantoms that imitate actual distributions of the radionuclides of concern in the human body (ICRU, 2003). One of the calibration phantoms for WBCs that have been widely employed is a bottle manikin absorption (BOMAB) phantom (ANSI, 1999) (Fig. 16). Different-shaped containers of this phantom represent each part of the human body, and the known-radioactivity is installed to have a uniform destruction throughout the body. This configuration is the most appropriate for radiocesium (e.g., ^{134}Cs, ^{137}Cs). Representative calibration phantoms for thyroid monitors are also presented in Fig. 17. These phantoms imitate radioiodine localized in the human thyroid in the neck (ANSI, 1973; ANSI, 2014; ORINS, 1959). The calibration phantom is supposed to be measured with the same measurement geometry. Thus, the counting efficiency for the target radionuclide is obtained. In general, full-energy absorption peaks (FAPs) observed in a pulse height spectrum perform measurements with in-vivo counters. Thus, the counting efficiency is determined by dividing a peak net count rate in the region of interest (ROI) for the target radionuclide by the radioactivity or the photon emission rate of the phantom; the unit should be count per second (cps) per Bq or cps per photon/s.

Figure 16: External view of a BOMAB phantom

Figure 17: External views for neck phantoms (left: ORINS phantom, right: ANSI phantom)
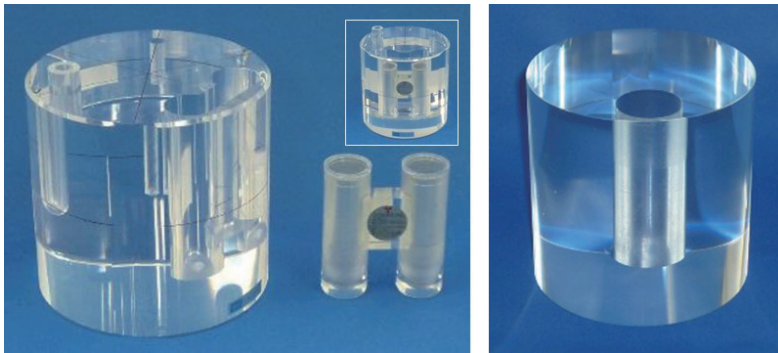
Finally, one caution is described when using the instruments for in-vivo bioassays. Surface contamination on the body of the subject can significantly influence the measurement, resulting in overestimation of the internal dose. It is thus essential to perform surveys for the body surface contamination and completely remove the contaminant before measurements. In-vivo bioassays have significant advantages over in-vitro bioassays (see 7.4.2) in terms of a relatively short counting time, the direct determination of the body content at the time of measurement without biokinetic models, the accuracy of measurements, and so on. Because of these advantages, in-vivo bioassays are a primary method for population monitoring following a major nuclear accident. Further considerations are, however, desired in particular on measurements of young children along with calibration for the measurement.

9.2. In-Vtiro Bioassays

Bioassays are an effective screening method for examining the internal contamination of ordinary people, workers, and first responders exposed to radioactive particles that can cause internal contamination in human bodies. Performing internal contamination assessments and dosimetry through analysis of human biological materials (e.g., urine) is defined as an in-vitro bioassay method. The biological sample analysis methods vary widely, depending on the characteristics of the contaminated radionuclides, the analysis target concentration, and the sample conditions. Furthermore, additional considerations should be given to the emission radionuclides'radiation types, sample types, and dose limits, as well as the complexity of the measurement method. In all bioassay processes, attention should be paid to the contamination of experimenters and analytical equipment during analysis.

On-site applicability and promptness should be ensured for the rapid application of biological-sample screening at the incident site. In particular, the biological sampling time determines the total time required for rapid sample analysis. Meanwhile, the characteristics of the contaminants must be identified when analyzing radioactivity, because the chemical and physical properties of each radionuclide differ. These can represent important indicators for the proper use of the analysis results. The analysis procedures that can be applied to each radionuclide and sample are described below (ANSI/HPS, 1999; IAEA 2004; Kurihara et al., 2018; Li et al., 2010). Compared with the in-vivo bioassay, the advantage of the in-vitro bioassay is that, once the sample has been obtained, it can be analyzed repeatedly. Furthermore, the measurement time can be adjusted as required.

Laboratory-based or field-deployable gamma detectors can be used in emergencies, by adapting the measurement sensitivity and analysis time to the measurement of gamma-emitting radionuclides, such as cesium (^{137}Cs), cobalt (^{60}Co), and iridium (^{102}Ir). A liquid scintillation counter (LSC) or alpha spectrometer can be used for strontium (^{90}Sr) and polonium (^{210}Po) analyses. Recently, ICP-MS equipment, which offers high sensitivity and rapid analysis, has been employed; this is typically applied to refractory

nuclides such as ^{239}Pu and ^{235}U. Actinide radionuclides have a very low initial excretion radioactivity compared with other gamma-emitting radionuclides. The low measured radioactivity indicates that the analysis struggles in terms of the accuracy and MDA. On the other hand, lightweight and soluble elements exhibit a relatively high excretion rate. Therefore, the analysis methods should be varied according to the radionuclides, radiation type, and element properties (Table 2).

Table 2. Principal methods used in indirect assessment.

Available emissions	Radiometric Methods							
	Alpha			Beta			Gamma	
Sample	Urine	Faeces or physical sample	Breath	Urine	Faeces or physical sample	Breath	Urine	Faeces or physical sample
Source preparation requirements	Isolation of radioelements from matrix, chemical separation of elements		None	None or concentration	Separation from matrix	None	None	
Detection methods	Gross alpha counting, alpha spectroscopy			LSC (limited energy discrimination)	Gross beta counting		Gross gamma counting, gamma spectroscopy	

Selection of Samples and Methods of Analysis

In general, analysing radioactivity concentrations by sampling human biological materials is defined as an in-vitro bioassay. Such in-vitro bioassays are suitable for analyzing radionuclides with low penetration abilities (e.g., tritium) and alpha-emitting radionuclides. This is because the in-vivo bioassay results for radionuclides that emit

low-energy photons suffer from high uncertainty and low sensitivity.

On the other hand, the results of an in-vitro bioassay — in which human biological materials are directly monitored — offer high sensitivity for estimating ingestion. However, uncertainty factors (e.g., ingestion time and elapsed time) must be assessed separately for each individual when analyzing the measurement results in terms of doses. Depending on the incident situation, an in-vivo bioassay using WBC equipment can be implemented. However, in some cases, the in-vitro bioassay is more suitable when considering the accommodation capacity of the institution. Table 2 classifies the pre-treatment and measurement methods according to the sample types, radiations, and radionuclides.

Methods of Analysis

The radioactivity analysis procedure for biological samples is determined by several factors. When relatively low ingestion and committed effective doses are expected, the analysis can be adequately performed using physicalsamples (e.g., air sampling, etc.) collected from the site. However, if a relatively high dose is expected (1 mSv or higher), it is appropriate to perform biological sample analysis; this facilitates direct internal contamination assessment and dosimetry.Biological sample analysis refers to the detection and quantitative analysis of radiation emitted from the radionuclides present in the sample. In general, the radioactivity can be measured only when the radionuclides to be analysed are extracted from the sample matrix. This ensures the reproducibility of the radioactivity analysis'sensitivity and results. Actinide-series radionuclides must be radio-chemically separated before the radioactivity is measured, because of their similar chemical properties.

The radioactivity analysis of biological samples varies depending on the responding institution's accommodation capacity and technology level; hence, specific methods are not set as international standards. However, appropriate analysis procedures can be established from previous research records. The pre-analysis capabilities, including correction and analysis-result validation, should be enhanced by selecting procedures appropriate for each laboratory. Whether the analysis criteria defined by the regulatory

authority are satisfied represents an important factor that must be considered when selecting the sample quantity and analysis procedure.

In a sample-specific analysis method, the entire procedure can be primarily determined according to the type of detected radiation. When measuring radiation with low penetration abilities (e.g., alpha or beta), a pre-treatment process must be performed to extract the target radionuclides. However, in the case of a urine or liquid sample, methods of mixing the samplewith a liquid scintillator and counting the emitted scintillations are also used. In the case of gamma-emitting radionuclides (which emit highly penetrating photons), the measurement can be performed directly on bulk samples without a separate extraction process. These radionuclides allow direct counting in the human body; however, the measurements obtained from excretions can reflect the actual contamination level better and may be more effective for screening purposes. If several radionuclides are present in the actual sample (e.g., one in which various materials are mixed), an additional analysis method must be undertaken. Gamma-emitting radionuclides can be assessed quantitatively using gamma spectroscopy according to the inherent energy emission region. However, it is difficult to distinguish between beta-emitting radionuclides because the emitted energy of each radionuclide exhibits a continuous spectrum. In such cases, energy discrimination may be performed using a detector, typically an LSC.

Alpha-emitting radionuclides are mono-energetic and feature a narrow emission energy. However, the penetration ability of the radiation is very weak and the alpha-ray shielding of other materials is high, making it difficult to monitor using a raw sample. Therefore, the radioactivity should be monitored after extracting the target radionuclides, by pre-treating each element. Monitoring should be performed by classifying the measurement procedures according to the purpose. Expressed otherwise, the measurement time, detection limit, and other parameters should be set by classifying the normal and emergency situations. Because routine or task-related monitoring is associated with occupational radiation exposure, a monitoring system is required for contaminated radionuclides that can typically arise in the workplace. Furthermore, a preparatory system should be established according to the chemical, health, or physical characteristics of

the radionuclides to be analyzed. On the other hand, in the case of special monitoring applied during an emergency, it is necessary to identify the radiation exposure type and conduct the analysis procedure in response.

Biological Samples

The primary sources of bioassay data are urine, feces, breath, and blood; in special circumstances, teeth and hair may also be used. Furthermore, nose-blow and nasal-smear methods are used in screening tests of the initial internal contamination, though these are only used as initial data in a limited screening. For the selection of bioassay samples, themajor paths for excreting contaminated radionuclides and the ease of sample collection must both be considered. In general, urine samples are most effective in terms of the ease of collection and the assessment of body absorption and contamination levels.

In the case of certain radionuclides (e.g., plutonium) entering the human body through inhalation and ingestion, fecal samples should be collected and analyzed. In this case, dosimetry can be performed using the data obtained after collecting and analyzing all fecal samples. However, such samples are time-consuming and difficult to collect. Furthermore, the analysis is difficult, and the measurement data are difficult to interpret. In the case of special monitoring conducted to identify certain radionuclides, various sampling methods are sometimes required for routine or task-related monitoring.

Urine Samples

Urine consists primarily of water, though it also contains wastes and other substances that are excreted after passing through the blood and being filtered by the kidneys. These substances are retained in the bladder for several hours before excretion. The daily urine excretion of the reference man, as suggested by the International Commission on Radiological Protection (ICRP, 1997) is 1.4 L/d. However,in reality, the amount excreted by an individual varies considerably from person to person. Although 24-h sample collection is difficult, strong results can be obtained if 24-h samples are collected and analyzed in routine monitoring. In this case, the first urine excreted in the morning must be used in

the analysis, because it contains the highest quantity of in-vivo substances.

The assessment methods that use a spot urine sample to correct the 24-h sample monitoring data include sample-volume and creatinine-correction methods. The normalization correction method that employs the daily creatinine excretion exhibits a smaller variability than that which uses the sample volume to correct the measurement values. In general, when evaluating the routine monitoring of a certain radionuclide, the spot urine sample data are corrected using the aforementioned normalization method. Tritium is excluded from the 24-h sample collection. In the human body, tritium is saturated in water because of its chemical characteristics; hence, the water component of urine represents the concentration in the body; consequently, 24-h samples are not required. However, if the sampling frequency is increased, then the contamination level trend in the body can be identified, and the data resolution can be improved.

Faeces

A fecal sample contains waste products transferred as cellular debris sloughed from the intestinal walls through the gastrointestinal (GI) tract. It contains materials cleared from the lungs and systemic materials excreted into the GI tract, though it primarily contains wastes excreted through the digestive system after digestion. It takes approximately two days for the wastes to be discharged from the body after passing through the GI tract. However, this differs, predominately depending on personal eating habits and physical body conditions. Furthermore, the composition and quantity vary considerably from person to person. Therefore, sampling is typically conducted over 3-4 days. This reduces the uncertainty caused by individual variability, and it produces a reliable value for the daily excretion rate.

Fecal samples are particularly characterized by biodegradation. Therefore, the samples are ashed and frozen immediately after collection, to prevent the decomposition and spread of contamination. The experimenter should consider the prevention of biological contaminants and radioactive contamination spread. A wet/dry ashing process can be applied during the radionuclide separation of a fecal sample. Later, the radionuclides are analyzed using the remaining ashed or concentrated sample.

Sample Collection and Transport

Contamination can be minimized by using clean disposable containers to collect urine samples. Furthermore, the lid must be firmly closed to prevent sample leakage and the subsequent spread of contamination. The excretion and sampling times must be recorded for ingestion assessment and dosimetry, and the sample identification factors should be properly indicated. Because they contain a considerable quantity of organic materials, the biological samples should be processed as rapidly as possible, to prevent decomposition. If the samples cannot be processed immediately, they should be refrigerated at or below 4℃, frozen, or preservative-treated.

The sample-collection schedule should be established by taking into account the planned administering of medical treatment drugs (e.g., chelation therapy and administration), to factor in the excretion rate and excretion path variables involved when using such protective drugs. For the fecal samples (typically collected over 1-3 days), a toilet-mounted container should be selected, because the entire spot fecal excretion must be collected. Furthermore, the samples must be ashed in a high-temperature ashing furnace before radionuclide analysis; hence, use of an appropriate wrapper (e.g., paper foil) is recommended.

9.3. Internal Dosimetry from Bioassay Data

Internal dosimetry or internal dose assessment from bioassay data is illustrated in Fig. 18. One important note is that the internal dose cannot be directly measured like the external dose from a personal dosimeter. To assess the internal dose, one should estimate the initial intake (amount) of the radionuclide of concern. The intake defined here is the activity of a radionuclideincorporated into the body in a given time period or a result of a given event (IAEA, 2004). Both in-vivo and in-vitro bioassay results are only a portion of the intake, which depends on the elapsed time after the internal contamination event.

The concept of internal dosimetry is based on the knowledge on biokinetic behaviors

of radionuclides and anatomical models to calculate the dose imparted to the target organs and tissues; however, this is out of the scope of this guideline. In terms of internal dose calculations from bioassay data, one should use appropriate dosimetric data and make a reasonable assumption for the intake scenario. Dosimetric data used for calculations include retention/excretion rates as a function of time after the event and dosecoefficients for the internal dose. The former includes results of biokinetic model prediction for radionuclides and is provided for typical bioassay items for each major radionuclide on the unit of Bq per Bq intake. The latter includes data to convert the intake to the effective dose or the tissue equivalent doses in the case of inhalation or ingestion. The internal dose is normally assessed in terms of the committed dose that is an integrated dose over 50 years post the event for radiation workers or up to 70 years old. These datasets have continued to be updated by the ICRP based on the latest scientific knowledge. The currently-used datasets (based on the concept of ICRP Publ. 60 (Simion et al., 1991) are summarized in Table 3. A software package for internal dose calculations is also available (Ishigure et al., 2004).

The intake scenario is a series of assumptions on specific conditions of the intake event, such as the route of intake (inhalation or ingestion or both), the physicochemical parameters of the radionuclide incorporated into the body (e.g., the solubility, the aerosol size in the case of inhalation), and the physiological parameters of individuals (e.g., the age and sex of the subjects of concern, the daily breathing volume, and thedaily food and drink consumption). All these parameters are difficult to be obtained in most cases, and it is thus necessary to make suitable assumptions depending on the purpose of dose assessment (e.g., a conservative assessment for implementing radiation protection measures, or a realistic dose assessment for an epidemiological study of the affected populations).

One remarkable issue on the intake scenario was observed in the dose assessment of members of the public from their WBC measurements in the Fukushima Daiichi NPP accident. The targets of these WBC measurements were [134]Cs and [137]Cs. The intake scenario was decided to be an acute inhalation intake scenario on the next day following the accident (i.e., 12 March 2011) for the most conserWvative dose assessment.

This scenario was considered reasonable at the early time; however, it often leads to unreasonable estimations of the intake in particular for young children. This is due to their relatively rapid metabolism for Cs, as illustrated in Fig. 19. For example, the whole-body retention rate of ^{137}Cs for 10-y-old children at the 300th day post intake event is 0.04 (4%) in the case of inhalation of aerosols with default physicochemical parameters determined by ICRP (e.g. Type F as the absorption parameter in the respiratory tract and 1 m in activity median aerodynamic diameter (AMAD)). This means that the initial intake of ^{137}Cs is 250 times as large as the residual body content measured by a WBC measurement on the 300th day. In contrast, the daily breathing volume should be larger in adults than in children; however, the Fukushima caseoccasionally demonstrated opposite estimations of the early intake from late WBC measurements. This suggests possibilities of accidental intake events during the period between the accident and the measurement or trivial body surface contamination at the time of measurement. Therefore, population monitoring following a nuclear accident should be implemented in a timely manner to reduce uncertainties in the intake scenario. Further information is described elsewhere (Kurihara et al., 2018).

Figure 18 : Internal dose assessment from bioassay data
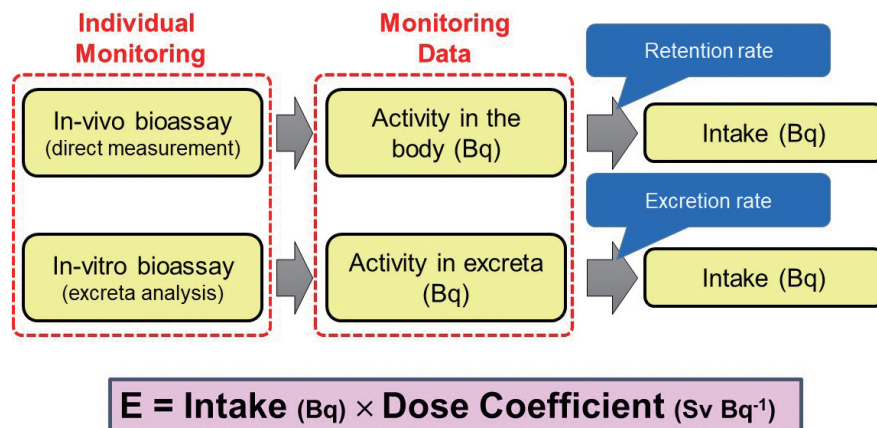
Table 3. Datasets relevant to internal dose assessments.

Publication	Items
ICRP Publication 68	Dose coefficients for workers - Inhalation (1 and 5 micron), Ingestion
ICRP Publication 78	Dose coefficients for workers - Inhalation (5 micron), Ingestion Retention/Excretion rates up to 10 days post intake
ICRP Publication 71, 72	Dose coefficients for public - Inhalation (1 micron), Ingestion
ICRP CD-ROM(CD1)	Dose coefficients for workers and members of the public - Inhalation (0.001-10 micron), Ingestion
IAEA Safety Series No.37	Dose coefficients for workers - Inhalation (1, 5 micron), Ingestion, Injection Retention/Excretion rates
IAEA EPR-Medical 2005	Dose coefficients for workers and members of the public - Inhalation (1, 5 micron), Ingestion Retention/Excretion rates up to 10 days post intake
ICRP Publication 119	Covering ICRP CD-ROM, External dose coeffcients

Figure 19: [137]Cs whole-body retention rates for various age groups in the case of acute inhalation intake
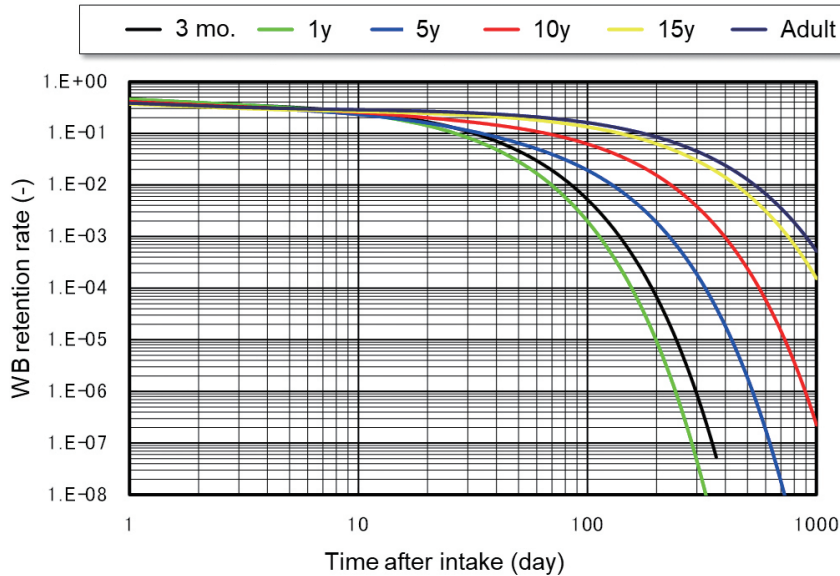
Radiation Doses in Non-Human Biota

Environmental protection should be considered when planning for emergency exposure situations. If significant impacts on the populations of certain species (e.g., threatened or endangered species) are anticipated in the event of an emergency, alternative siting options or the implementation of procedures to protect these populations may need to be considered in emergency situations.

The assessment of non-human biota in emergency planning is particularly important in unpopulated areas. Environmental impact assessments should consider the likely consequences of exposure as a result of different possible emergency exposure scenarios (ICRP, 2008). In these situations, it should be noted that assessments must consider the dynamic conditions of an emergency. Steady-state models are not always relevant for these types of releases.

During emergencies, the protection of the environment will be optimized by normal emergency practices. It is clear that human protection will take precedence during this time, because resources are typically spent on humans; however, thorough planning will mean that clearly defined procedures are in place that can be applied during the emergency phase. These include decisions regarding the protection of the environment, weighed up against protection of the food chain.

Once the emergency has been stabilized and the scenario is transitioning to an existing exposure one (ARPANSA, 2017), the need for remedial measures (remediation) should be weighed up against the immediate and long-term impacts upon potentially displaced non-human biota populations. Particular attention should be afforded to the effects upon threatened or endangered species. At this point, traditional (steady-state) assessment models can be used to determine the long-term impacts of exposure.

For existing exposure situations involving environmental contamination, an initial assessment should be conducted to characterize the existing radiological conditions

of the contaminated area, including baseline background data. This should include identifying the sources and pathways of exposure for key receptor organisms, estimating the dose rates to those organisms, and comparing these with relevant environmental reference values (ARPANSA, 2015). A decision should then be made regardingthe required management or remedial measures, as well as the motivations thereof, taking full account of the costs and benefits of the action. The outcome of the initial assessment should help guide the decision-making process in the following way:

- If the assessed dose rates to key receptor organisms (or keystone species) exceed the relevant environmental reference values, then the optimization should seek to reduce exposures to levels at or below the relevant environmental reference value, assuming that the costs and benefits of doing so are justified.

- If the assessed dose rates to key receptor organisms are at or below the relevant environmental reference values, then the principle of protection optimization should continue to be implemented, assuming thatthe costs and benefits are such that further efforts to reduce exposure are justified.

- In either case, the justifiable effort should be to reduce the exposure to levels as low as reasonably achievable, rather than to simply achieve a value lower than the screening or reference levels.

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CONTRIBUTORS TO DRAFTING AND REVIEW

Seokwon Yoon	Korea Institute of Radiological and Medical Sciences (KIRAMS), Korea
Jae Seok Kim	Korea Institute of Radiological and Medical Sciences (KIRAMS), Korea
Wi-Ho HA	Korea Atomic Energy Research Institute (KAERI), Korea
Osamu Kurihara	National Institutes for Quantum Science and Technology (QST), Japan
Yumiko Suto	National Institutes for Quantum Science and Technology (QST), Japan
Liesel Green	Australian Radiation Protection and Nuclear Safety Agency (ARPANSA), Australia
Hiroshi Yasuda	Hiroshima University, Japan
Tani Kotaro	National Institutes for Quantum Science and Technology (QST), Japan
Thawatchai Itthipoonthanakorn	Office of Atoms for Peace (OAP), Thailand
Hyung Taek Kim	Korea Atomic Energy Research Institute (KAERI), Korea

RCA Regional Office (RCARO)

Korea Institute of Radiological & Medical Sciences (KIRAMS)