OVERVIEW OF NATURAL AND ARTIFICIAL RADIOACTIVITY DOSIMETRY AND ACTIVITY MEASUREMENT METHODS, USEFUL FOR FORENSIC PURPOSES

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ABSTRACT

Purpose: The primary purpose of this study is to review the mechanism of exposure to ionizing radiation through different pathways with a focus on both external and internal exposure scenarios. By reviewing the methods for examining the behavior and distribution of natural and artificial radionuclides in different matrices, the research aims to examine the forensic relevance of radiation dose assessment and its application in identifying sources and situations of exposure using radiation detection methods. The use of radiation detectors and dosimetry methods ensures accurate detection and analysis, making the results applicable to radiation protection.

Design/Methods/Approach: This paper presents a review of dosimetry quantities that can be derived from radionuclide concentration data in various matrices. The focus is placed on understanding the relationship between radionuclide activity concentrations and the calculation of key dosimetry quantities, including absorbed dose, equivalent dose, and effective dose. The approach involves reviewing existing methodologies and mathematical models used for dose estimation based on internal and external exposure scenarios and their applicability to the forensic studies. No original measurements were performed; this paper is a methodological review.

Findings: Various dosimetry quantities, such as absorbed dose, equivalent dose, and effective dose, can be estimated based on radionuclide activity concentrations in different media. These quantities are essential for assessing both external and internal exposure to ionizing radiation. This framework enables dose assessments based on contamination data. The findings underline the importance of combining radiological measurements with dose modeling tools in order to support health risk assessment.

Originality/Value: This paper provides a structured overview of how key dosimetry quantities can be determined from radionuclide concentration data, without direct dosimetry measurements, and emphasizes the role of nuclear techniques and dose assessment methodologies as fundamental tools in

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forensic analysis in the context of radiological or nuclear emergencies. This study provides a valuable reference for professionals involved in dose assessment and ionizing radiation protection.

Keywords: exposure to ionizing radiation, dosimetric quantities, radiation detectors, NORM.

About the authors

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INTRODUCTION

The naturally occurring sources of radiation are cosmic radiation and primordial radioactive elements that were created when the Earth was born about 4.5 billion years ago. Concentrations or quantities of naturally occurring radioactive material of sufficient activity to be of interest are known by the acronym NORM (Cember and Johnson, 2009).

The general population can be exposed to the natural levels of radiation that are mainly terrestrial in origin, belonging to the uranium and thorium series and ^{40}K in the soil. Many industrial processes (such as ore and oil mining, the building industry, etc.) contribute to the so-called technologically enhanced natural radioactivity, which means that the radionuclides present in the raw materials are concentrated in the final product or the waste. Discarding these products in the environment can cause, as a consequence, the exposure of the public to the ionizing radiation (Janković et al., 2023).

Cosmic radiation consists of high-energy particles originating from the processes in the Sun and other parts of our galaxy (UNCEAR, 2022). For the most part, these particles are filtered by the magnetic field of the Earth. However, one part reaches the atmosphere, where it can interact with the elements present in the upper layers (N_2, O_2, Ar and others) or even reach the surface of the Earth. Examples of isotopes produced in the interaction of cosmic radiation with the elements present in the atmosphere are ^{7}Be, ^{3}H, and ^{14}C.

Artificial radionuclides are present in our everyday life due to their usage in medicine as a part of radiation therapy or diagnostics (short-lived radionuclides used in a controlled manner and regulated by Rulebooks and Practice Guides) and as a consequence of nuclear accidents and nuclear weapon tests in the past (presence of some long-lived radionuclides such as ^{137}Cs and ^{90}Sr in the environment).

To quantify the potential exposure to individuals in surrounding areas, such contributions to radiation levels are typically assessed through the measurement of ambient dose equivalent rate, which reflects the external radiation field relevant to human exposure at or near site boundaries. Also, by knowing the baseline values through systematic and continuous monitoring, recognizing the events that are out of scope of normal operations is possible. In case of the unplanned release of radionuclides in the environment, methods used for monitoring are the first that can provide the warning results (Dimović et al., 2024). It is then possible to backtrack the obtained results both in time and space, which enables us to reach a conclusion about the nature of the release.

COMPONENTS OF A COMPREHENSIVE RADIATION PROTECTION PROGRAM

A well-structured radiation protection program includes several key elements to ensure regulatory compliance and worker safety in environments involving ionizing radiation:

- Personal dosimetry monitoring – covering both external and, when necessary, internal radiation doses.
- Routine surveys and area monitoring are performed to assess and document ambient radiation levels, potential contamination by radioactive materials, and to evaluate the likelihood of worker exposure.
- Radioactivity control, including control of goods at border crossings during import, export, and transit and the procedure of using monitors and intervention procedures in the case of illicit trafficking of radioactive and nuclear materials across the border (Official Gazette of RS, No. 86/19).
- Worker training on radiation protection, including health effects associated with radiation dose and radiation protection procedures and controls to minimize dose and prevent contamination. Use of ionizing radiation in various fields of application: medicine, veterinary medicine, industry, agriculture, and scientific research; supplementary training for work with sealed and unsealed sources of ionizing radiation and with devices that generate ionizing radiation; training of personnel responsible for the implementation of radiation protection measures. Periodic renewal and updating of knowledge.
- Emergency preparedness procedures are established to ensure timely identification and effective response to radiological emergencies. Advanced knowledge and essential skills necessary for effectively addressing complex challenges and threats in the field of Chemical, Biological, Radiological, and Nuclear (CBRN) safety and security.

Operations involving exposure to ionizing radiation should be designed so any unnecessary exposure should be avoided, and all exposures shall be kept as low as reasonably achievable (ALARA) (Cember and Johnson, 2009).

DEFINITIONS OF DOSIMETRY QUANTITIES

The dose calculation equations are based on two types of exposure to radiation: external and internal exposure. The first type of exposure is resulting from radioactive sources external to the body. Internal radiation exposure occurs when radioactive material is taken into the body. This can happen through inhalation of contaminated air or ingestion of food and water containing radioactive substances. Once inside the body, the radioactive material becomes an internal source of radiation, continuously exposing tissues to radiation until it either decays naturally or is eliminated through biological processes such as excretion or metabolic activity.

The effects of the exposure are classified into categories – stochastic effects and deterministic effects, taking into account the probability of appearance; somatic and genetic effects, according to the part of the body or cells where the effects can be detected; and acute and late effects, taking into account the time passed between the exposure and the effect (Cember and Johnson, 2009).

The appearance of some radiation effects on the biological systems can provide an insight into the timeline of the events leading to the expression of the said effects and thus contribute to the reconstruction of the unwanted event.

Absorbed Dose (D)

Absorbed dose (*D*) is the amount of energy absorbed per unit mass (Official Gazette of RS, Nos. 86/2011 and 50/2018).

$$D = \frac{dE}{dm} \tag{1}$$

where *dE* is the mean energy of ionizing radiation transferred to matter per volume element and dm is the mass of matter contained in that volume element. The unit of absorbed dose is the gray [Gy].

Exposure Dose (X)

It refers to the ionization of air caused by X-rays or gamma radiation. It is defined as the amount of electric charge of ions of one sign produced by X-ray or gamma radiation in air per unit mass of air**.** The unit of exposure dose is [C/kg].

Equivalent Dose, $H_{T,R}$

The equivalent dose $H_{T,R}$ takes into account the fact that the effects of ionizing radiation on a specific tissue or organ depend not only on the amount of energy absorbed per unit mass but also on the type of radiation. It is defined as the product of the average absorbed dose in tissue or organ T, delivered by radiation type R, and the corresponding radiation weighting factor W. The unit of equivalent dose is [Sv] (Official Gazette of RS, Nos. 86/2011 and 50/2018).

$$H_{T,R} = D_{T,R} W_R \tag{2}$$

If the radiation consists of multiple types of radiation with different W_R values, then the total equivalent dose is:

$$H_T = \sum D_{T,R} W_R \tag{3}$$

The radiation weight factor, $W_{R,}$ is a dimensionless factor that expresses the difference in biological effects of different types of ionizing radiation. Radiation weighting factor values for different types and energies of radiation are given in Table 1 (Official Gazette of RS, Nos. 86/2011 and 50/2018).

Table 1: *Radiation weight factors.*

Radiation type and energy range	**Radiation weight factor W_R**
Photons, of all energies	1
Electrons and muons, all energies 1	1
Neutrons, energy below 10 keV	5
Neutron energy from 10 to 100 keV	10
Neutrons, from 100 keV to 2 MeV	20
Neutrons, from 2 MeV to 20 MeV	10
Neutrons, energy higher than 20 MeV	5
Protons, except for recoil protons, with energies higher than 2 MeV	5
Alpha particles, fission fragments, heavy nuclei	20

Effective Dose (E)

Effective dose depends on the type of irradiated tissue or organ. The same equivalent dose will represent a different risk for the possible development of the disease on different organs. Therefore, when assessing the damage caused by ionizing radiation to various organs, the equivalent dose is multiplied by the tissue weighting factor, W_T, resulting in the effective dose. The unit of effective dose is [Sv].

$$E = \sum H_T W_T \quad (4)$$

The total effective dose is obtained by adding the effective dose that comes from exposure to external radiation and from the radiation of radionuclides introduced by ingestion and inhalation (Official Gazette of RS, Nos. 86/2011 and 50/2018):

$$E = E_s + \sum_j e(g)_{j,ing} I_{j.ing} + \sum_j e(g)_{j,inh} I_{j.inh} \quad (5)$$

where: E_s is the contribution of external radiation to the total effective dose; $e(g)_{j,ing}$ and $e(g)_{j,inh}$ are expected effective doses per unit intake of radionuclide j ingested by food or inhalation for an individual in age group g; $I_{j,ing}$ and $I_{j,inh}$ are the corresponding activities of radionuclides introduced through food or inhalation.

The ingestion dose assessment takes into account the concentrations of individual radionuclides in water or food, the amount consumed on an annual basis, and dose coefficients. (Clement 2012). The dose coefficients for ^{226}Ra, ^{90}Sr and their decay products that are involved in dose calculation are given in Table 2.

Table 2: *Dose coefficients for ^{226}Ra, ^{90}Sr and their decay products for different age groups* (Clement 2012).

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Table F.1. (*continued*)

		Infant		f_1	*e* (Sv/Bq)				
Nuclide	$T_{½}$	f_1	*e* (Sv/Bq)	≥1 year	1 year	5 years	10 years	15 years	Adult
Strontium									
Sr-80	100 m	0.6	3.7E−09	0.4*	2.3E−09	1.1E−09	6.5E−10	4.2E−10	3.4E−10
Sr-81	25.5 m	0.6	8.4E−10	0.4*	4.9E−10	2.4E−10	1.4E−10	9.6E−11	7.7E−11
Sr-82	25.0 d	0.6	7.2E−08	0.4*	4.1E−08	2.1E−08	1.3E−08	8.7E−09	6.1E−09
Sr-83	32.4 h	0.6	3.4E−09	0.4*	2.7E−09	1.4E−09	9.1E−10	5.7E−10	4.9E−10
Sr-85	64.84 d	0.6	7.7E−09	0.4*	3.1E−09	1.7E−09	1.5E−09	1.3E−09	5.6E−10
Sr-85m	69.5 m	0.6	4.5E−11	0.4*	3.0E−11	1.7E−11	1.1E−11	7.8E−12	6.1E−12
Sr-87m	2.805 h	0.6	2.4E−10	0.4*	1.7E−10	9.0E−11	5.6E−11	3.6E−11	3.0E−11
Sr-89	50.5 d	0.6	3.6E−08	0.4*	1.8E−08	8.9E−09	5.8E−09	4.0E−09	2.6E−09
Sr-90	29.12 y	0.6	2.3E−07	0.4*	7.3E−08	4.7E−08	6.0E−08	8.0E−08	2.8E−08
Sr-91	9.5 h	0.6	5.2E−09	0.4*	4.0E−09	2.1E−09	1.2E−09	7.4E−10	6.5E−10
Sr-92	2.71 h	0.6	3.4E−09	0.4*	2.7E−09	1.4E−09	8.2E−10	4.8E−10	4.3E−10
Yttrium									
Y-86	14.74 h	0.001	7.6E−09	0.0001	5.2E−09	2.9E−09	1.9E−09	1.2E−09	9.6E−10
Y-86m	48 m	0.001	4.5E−10	0.0001	3.1E−10	1.7E−10	1.1E−10	7.1E−11	5.6E−11
Y-87	80.3 h	0.001	4.6E−09	0.0001	3.2E−09	1.8E−09	1.1E−09	7.0E−10	5.5E−10
Y-88	106.64 d	0.001	8.1E−09	0.0001	6.0E−09	3.5E−09	2.4E−09	1.6E−09	1.3E−09
Y-90	64.0 h	0.001	3.1E−08	0.0001	2.0E−08	1.0E−08	5.9E−09	3.3E−09	2.7E−09
Y-90m	3.19 h	0.001	1.8E−09	0.0001	1.2E−09	6.1E−10	3.7E−10	2.2E−10	1.7E−10
Y-91	58.51 d	0.001	2.8E−08	0.0001	1.8E−08	8.8E−09	5.2E−09	2.9E−09	2.4E−09
Y-91m	49.71 m	0.001	9.2E−11	0.0001	6.0E−11	3.3E−11	2.1E−11	1.4E−11	1.1E−11
Y-92	3.54 h	0.001	5.9E−09	0.0001	3.6E−09	1.8E−09	1.0E−09	6.2E−10	4.9E−10
Y-93	10.1 h	0.001	1.4E−08	0.0001	8.5E−09	4.3E−09	2.5E−09	1.4E−09	1.2E−09
Y-94	19.1 m	0.001	9.9E−10	0.0001	5.5E−10	2.7E−10	1.5E−10	1.0E−10	8.1E−11
Y-95	10.7 m	0.001	5.7E−10	0.0001	3.1E−10	1.5E−10	8.7E−11	5.9E−11	4.6E−11

Polonium									
Po-203	36.7 m	1.0	2.9E−10	0.5	2.4E−10	1.3E−10	8.5E−11	5.8E−11	4.6E−11
Po-205	1.80 h	1.0	3.5E−10	0.5	2.8E−10	1.6E−10	1.1E−10	7.2E−11	5.8E−11
Po-207	350 m	1.0	4.4E−10	0.5	5.7E−10	3.2E−10	2.1E−10	1.4E−10	1.1E−10
Po-210	138.38 d	1.0	2.6E−05	0.5	8.8E−06	4.4E−06	2.6E−06	1.6E−06	1.2E−06
Astatine									
At-207	1.80 h	1.0	2.5E−09	1.0	1.6E−09	8.0E−10	4.8E−10	2.9E−10	2.4E−10
At-211	7.214 h	1.0	1.2E−07	1.0	7.8E−08	3.8E−08	2.3E−08	1.3E−08	1.1E−08
Francium									
Fr-222	14.4 m	1.0	6.2E−09	1.0	3.9E−09	2.0E−09	1.3E−09	8.5E−10	7.2E−10
Fr-223	21.8 m	1.0	2.6E−08	1.0	1.7E−08	8.3E−09	5.0E−09	2.9E−09	2.4E−09
Radium									
Ra-223	11.434 d	0.6	5.3E−06	0.3††	1.1E−06	5.7E−07	4.5E−07	3.7E−07	1.0E−07
Ra-224	3.66 d	0.6	2.7E−06	0.3††	6.6E−07	3.5E−07	2.6E−07	2.0E−07	6.5E−08
Ra-225	14.8 d	0.6	7.1E−06	0.3††	1.2E−06	6.1E−07	5.0E−07	4.4E−07	9.9E−08
Ra-226	1600 y	0.6	4.7E−06	0.3††	9.6E−07	6.2E−07	8.0E−07	1.5E−06	2.8E−07
Ra-227	42.2 m	0.6	1.1E−09	0.3††	4.3E−10	2.5E−10	1.7E−10	1.3E−10	8.1E−11
Ra-228	5.75 y	0.6	3.0E−05	0.3††	5.7E−06	3.4E−06	3.9E−06	5.3E−06	6.9E−07
Lead									
Pb-195m	15.8 m	0.6	2.6E−10	0.4††	1.6E−10	8.4E−11	5.2E−11	3.5E−11	2.9E−11
Pb-198	2.4 h	0.6	5.9E−10	0.4††	4.8E−10	2.7E−10	1.7E−10	1.1E−10	1.0E−10
Pb-199	90 m	0.6	3.5E−10	0.4††	2.6E−10	1.5E−10	9.4E−11	6.3E−11	5.4E−11
Pb-200	21.5 h	0.6	2.5E−09	0.4††	2.0E−09	1.1E−09	7.0E−10	4.4E−10	4.0E−10
Pb-201	9.4 h	0.6	9.4E−10	0.4††	7.8E−10	4.3E−10	2.7E−10	1.8E−10	1.6E−10
Pb-202	3E5 y	0.6	3.4E−08	0.4††	1.6E−08	1.3E−08	1.9E−08	2.7E−08	8.8E−09
Pb-202m	3.62 h	0.6	7.6E−10	0.4††	6.1E−10	3.5E−10	2.3E−10	1.5E−10	1.3E−10
Pb-203	52.05 h	0.6	1.6E−09	0.4††	1.3E−09	6.8E−10	4.3E−10	2.7E−10	2.4E−10
Pb-205	1.43E7 y	0.6	2.1E−09	0.4††	9.9E−10	6.2E−10	6.1E−10	6.5E−10	2.8E−10
Pb-209	3.253 h	0.6	5.7E−10	0.4††	3.8E−10	1.9E−10	1.1E−10	6.6E−11	5.7E−11
Pb-210	22.3 y	0.6	8.4E−06	0.4††	3.6E−06	2.2E−06	1.9E−06	1.9E−06	6.9E−07
Pb-211	36.1 m	0.6	3.1E−09	0.4††	1.4E−09	7.1E−10	4.1E−10	2.7E−10	1.8E−10
Pb-212	10.64 h	0.6	1.5E−07	0.4††	6.3E−08	3.3E−08	2.0E−08	1.3E−08	6.0E−09
Pb-214	26.8 m	0.6	2.7E−09	0.4††	1.0E−09	5.2E−10	3.1E−10	2.0E−10	1.4E−10

DOSE AND ACTIVITY MEASUREMENT METHODS AND FORENSIC APPLICATION

Dosimetric Measurements

Dosimetry values are measured using passive dosimetry (thermoluminescent dosimeters TLD and optically stimulated luminescent dosimeters OSL) in stationary mode, ionizing chambers (mainly in a laboratory environment), or with various portable dosimeters and radiation monitors such as Geiger-Muller (GM) probes or scintillation probes.

Passive dosimeters are mainly used in stationary mode, exposed for a predefined period of time, and as such, can serve to monitor the cumulative dose and have limited possibility to determine the direction of the radiation. For example, TLD as well as other types of passive dosimeters can be used for the measurement of the total absorbed dose received by the walls of a container for detecting the prior presence of a radioactive material. By simultaneously considering both the accumulated dose on the walls of containers and the type and amount of the radioactive material producing that dose, rather unique information regarding the history of an interdicted or unknown sample might be uncovered (Schwantes et al., 2009). More recently, a broad variety of retrospective dosimetry problems were published: the telephone chip cards and electronic components of portable electronic devices such as cell phones, pagers, and digital watches are considered for dose reconstruction using TLD and OSL techniques (Göksu, 2003; Inrig et al., 2008; Beerten et al., 2009). Namely, the surface-mounted components of electronic devices, such as resistors, surface-mounted capacitors and diodes, inductors, integrated circuits, smart chip cards including subscriber identification module (SIM) cards, glasses,

and displays produce material-specific TL and OSL responses, from which the dose absorbed by the given component can be estimated.

Portable instruments have a probe (GM tube or scintillation material) that is able to give a readout of the absorbed dose immediately, and they can be hand-held. The typical range of portable instruments is 0.1 μSv to 1 mSv, and they are sensitive to gamma and beta radiation of various energies, depending on the model and material of the probe. The main advantage is that they are portable, enabling the operator to detect the presence of an ionizing radiation source or contamination in the specific direction or on the specific surface. For forensic purposes, this type of instruments serves for locating the source of ionizing radiation as well as contamination caused by the presence of the source at some point (provided that the source contains unbound radioactive material). Thus, it can be used for searching for radioactive material in transportation vehicles or at borders as well as detecting radioactive contaminants in environmental samples. Also, it can be used to distinguish the contamination resulting from the nuclear accident from the other scenarios, since the contaminants released in the accident (main contribution is ^{137}Cs) will lead to a slower radioactive decay compared to, for example, a normal reactor shutdown, which is dominated by ^{60}Co (Marques, 2021).

Measurement of the Activity Concentration

For the measurement of gross alpha and gross beta radiation, gas-proportional counters are mainly in use. The principle of the detection is the ionization of the gas mixture. This type of detector is useful for the screening measurements, giving fast results without identification of the particular elements present in the measured samples (Janković et al., 2015; Sarap et al., 2024). In cases involving nuclear materials, this method is a standard part of the initial radiological survey to get baseline data on the radioactivity level as well as an estimation of the level of contamination. A high gross alpha and gross beta measurement result prompts using other specific detection methods to identify the exact radionuclides that are present and their quantities. This type of measurement is particularly useful in analyzing drinking water or surface waters since it is capable of producing fast results as an indication of a non-planned nuclear event. Also, by measuring air samples shortly after collection and monitoring the decay rate, typically within hours of sampling, anomalies such as a sharp rise in activity can serve as an indicator of a radiological release or incident. Also, the correlation of measured alpha and beta activities can be informative. A positive correlation indicates a common or related source of alpha- and beta-emitting radionuclides, while any discrepancy from the positive correlation should be investigated.

Liquid scintillation spectrometers (LSCs) contain a liquid substance that is excited in the interaction with the ionizing radiation. In this type of measurement, distinct alpha- or beta-emitting radionuclides can be detected and their concentration quantified (Todorović, Nikolov, and Stojković, 2018). The great sensitivity and precision of LSCs make them the best choice for the detection of trace amounts of radionuclides, especially those that emit low-energy beta particles. LSCs are used, among others things, for the measurement of ^{14}C concentration in the sample, which serves for the determination of the age of a sample – carbon dating, which is readily used in various forensic applications. The low detection limit for ^{3}H is favorable for the determination of the age and status of underground water bodies and water tables, thus enabling us to determine whether the underground water body was exposed to the outside environment. Also, it serves to follow the concentration of ^{3}H and other reactor products in the effluents even when they are consequently diluted in rivers or streams. Also, it is used for radiolabeled material, especially biological material that can give insight into microbial activity beyond the limits of biomolecular analysis (Heuer et al., 2020).

For gamma-emitting radionuclides, gamma spectrometers are used (Krneta Nikolić et al., 2018). These spectrometers are capable of detecting a whole spectrum of gamma photons ranging from below 50 keV up to 3 MeV. The gamma spectrometry enables us to detect and quantify all gamma emitting radionuclides present in the sample in a single nondestructive measurement, which is its main advantage over other methods. The detectors in use can be semiconductor detectors (Krneta Nikolić et al., 2018), such as high purity germanium (HPGe) detectors and detectors containing doped semiconductor material such as Cadmium Telluride (CdTe), Cadmium Zinc Telluride (CZT), detector etc. Also, scintillation detectors such as NaI, CsI, BGO, or LYSO are very commonly used when high energy resolution is not required. By comparing the results obtained by gamma spectrometry to the baseline data, it can be concluded if the contamination has occurred. Every detection of artificial radionuclides in an environmental sample (excluding ^{137}Cs) can be an indication of an unplanned release.

Using state-of-the-art measurement methods such as mass spectrometry (Veljković et al., 2024; Veljković et al., 2025), it is possible to determine the isotopic ratio and thus obtain information regarding the origin of the detected radionuclides. Standard measurement methods such as gamma spectrometry and measurement of ^{3}H and ^{90}Sr, as well as ambient dose equivalent measurements, can give information on the local unplanned events, provided that the continuous monitoring of the environment is conducted.

The presence of ^{233}U can be a sensitive indicator that anthropogenic uranium has been blended with natural uranium. Furthermore, since it has a unique production pathway, ^{233}U can be used to distinguish different sources of anthropogenic uranium. In a previous study (Tumey et al., 2009), samples that were expected to contain only natural uranium were collected from a site with known anthropogenic contamination. The major uranium isotope ratios showed a deviation from natural values, indicating the presence of recycled uranium. The relative abundance of neutron activation products in the debris produced by a nuclear detonation can provide important clues as to the design and the materials used in a nuclear device. This information can be used for source attribution, which is essential to deterring nuclear weapon states from providing assistance to terrorist organizations (Straume et al., 2003).

The results of the examination support the response to the unauthorized use of these materials and help investigators make informed decisions regarding nuclear security. Using isotopic, chemical, and physical characteristics of nuclear and other material, together with related forensic evidence to include DNA, hair, fingerprints, tool marks, and explosive residues, nuclear forensics can potentially link samples of interest to people, places, and events (https://www.iaea.org/topics/nuclear-forensics).

CONCLUSION

This paper emphasizes the critical role of dose assessment in understanding the impact of ionizing radiation on human health. By analyzing how absorbed, equivalent, and effective doses can be derived from radionuclide concentration data in different media, the paper provides a theoretical foundation for evaluating both internal and external exposure scenarios. This study contributes to radiological risk evaluation, especially in cases where nuclear techniques must be applied. By combining the calculation and the measurement, we can estimate the pathways of the contamination or exposure of the individual. The discrepancy between the calculation and the measurement of the dosimetric values (where such measurements are possible) that cannot be explained by the normal propagation of the measurement uncertainty has to be taken into account as a possible indication of malpractice or deviation from the safety and security protocols. No original measurements were performed; this paper is a methodological review.

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