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16th EURADOS School: Contribution of dosimetry in the field of nuclear emergency preparedness and radiological accident management

Isabel Bravo, Wolfgang Raskob, Matt Simpson,
Maria Antonia López, Elizabeth A. Ainsbury,
Arturo Vargas, Christelle Huet, Luiz Bertelli,
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Isabel Bravo¹, Wolfgang Raskob², Matt Simpson³, Maria Antonia López⁴, Elizabeth A. Ainsbury³, Arturo Vargas⁵, Christelle Huet⁶, Luiz Bertelli⁷, Jean-Francois Bottollier-Depois⁸, Johan Camps⁹

¹ Portuguese Institute of Oncology, Porto, Portugal

² Retired from Karlsruhe Institute of Technology (KIT), Eggenstein-Leopoldshafen, Germany

³ United Kingdom Health Security Agency (UKHSA), Didcot, United Kingdom

⁴ CIEMAT, Madrid, Spain

⁵ Institute of Energy Technologies (INTE), Technical University of Catalonia, Barcelona, Spain

⁶ Autorité de Sûreté Nucléaire et de Radioprotection (ASNR), Fontenay-aux-Roses, France

⁷ L Bertelli & Associates, Murray, UT, USA

⁸ Retired from Institute for Radiological Protection and Nuclear Safety, Fontenay-aux-Roses, France

⁹ Belgian Nuclear Research Centre (SCK CEN), Mol, Belgium

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Abstract

Ionising radiation is of huge benefit to society, including in medicine and industry. However, there remains a small but very real risk of radiation accidents or malicious events which may lead to small or very large numbers of individuals being exposed to unplanned doses, either patients, workers, or members of the public.

Ionising radiation dosimetry in its various forms, including environmental, computational, individual monitoring, internal and external dose assessment, is a crucial part of any radiation emergency response, not least following a large scale civil nuclear accident. In such scenarios, direct monitoring of doses to individuals and the environment as well as prediction of the pathways for exposure which relies on detailed calculations using a variety of models and approaches, directly support decision making in terms of incident resolution (who should try and fix the problem and how), as well as protective actions for potentially exposed members of the public and the environment. Dosimetry is also key to the recovery phase, including the initial and longer term clean up as well as the long-term consequences for exposed workers, members of the public, non-human biota and the environment.

The EURADOS School 2023 sought to establish the basics of preparedness for nuclear emergency situations, before introducing the audience to the key considerations of dosimetric monitoring and modelling to support real time decision making and prediction of evolution of the accident, individual monitoring and external and internal dosimetry for individuals and the environment. The contribution of dosimetry to communication to the public and stakeholders, which is essential for both the acute and clean up phases of the incident, was also considered, as were the next steps in terms of the longer term. Case studies on real or potential incidents were presented to put the presented theory into practical context.

1 Biological Effects in Nuclear and Radiological Accidents

Isabel Bravo, Medical Physics, Radiobiology and Radiological Protection Group, Research Centre, Portuguese Institute of Oncology, Porto, Portugal

1.1 Introduction

The use of ionising radiation (IR) in medicine, industry and energy production has numerous applications that have wide-ranging benefits for society and individuals. In medicine, the wide and diversified use of ionising radiation for diagnostic (radiology), therapeutic (external radiotherapy and brachytherapy, flash therapy, nuclear medicine), and industrial sterilization (of medical products and devices) are all excellent examples. While ionizing radiation has invaluable applications, it may also present a significant health risk on living organisms and will be an important subject of great interest and research for decades.

Thus, radiobiology emerged as a science, an interdisciplinary field of basic and clinical sciences that aims to identify mechanisms and processes underlying the action of ionizing radiation on biological tissues and their cellular and molecular components.

When radiological or nuclear accidents or intentional catastrophic events occur, albeit being exceedingly rare, they may have major consequences regarding the professionals, public or patients involved. Nuclear and radiological accidents can cause huge harm to individuals, the environment and the economy. Chernobyl (USSR, 1986), Goiania (Brazil, 1987) and Fukushima Daiichi (Japan, 2011) were awful catastrophes demonstrating how devastating these catastrophes can be (Obrador et al, 2022). These accidents or intentional events involve unintended exposure to ionizing radiation with varying biological effects depending on several factors such as type, intensity and duration of exposure. One of the crucial aspects of radiobiology is the variable sensitivity of different tissues to ionizing radiation and it is thus necessary to understand the processes behind the biological response and the implications for radioprotection and risk assessment, as well its use in medicine, industry and energy production.

1.2 Molecular action of ionising radiation

Living tissues are continuously exposed to ionising radiation originating from natural sources ("background" radiation) or from artificial sources, as demonstrated by the discoveries of Roentgen, Becquerel and Curie at the end of the 19th century.

As radiation moves through biological material, it triggers a series of events, interacting with atoms and molecules in the intracellular-environment, with the consequent transfer of energy. All biological molecules can be altered in response to radiation, but the resulting consequences vary according to the relevance of specific molecules. The most consequential target within the cell is DNA because it contains the genetic information required to maintain cell function and survival. DNA is therefore the mark of the biological effects, namely cell death, loss of clonogenic capacity, gene mutations and chromosomal aberrations, with the consequent somatic, hereditary, teratogenic and carcinogenic effects.

Ionising radiation causes damage to cells or tissues through the deposition of energy leading to a sequence of events. Different types of radiation cause different types of biological damage. The effects range from a few seconds to many years, depending on the processes involved. These

biological changes can affect a cell in various ways, resulting in premature death, delayed cell division or genetic modification that is passed on to subsequent generations.

All of these injuries begin at the cellular or molecular level. These biological effects are subsequently manifested due to the varying impact these injuries may have on millions of cells in a specific organ or tissue. Considering that the nature of the damage is molecular or cellular it is imperative to understand the mechanisms involved.

Biological effects of radiation are typically divided into two categories: the first one consists of exposure to high doses of radiation over short periods of time producing acute or short-term effects. The second category represents exposure to low doses of radiation over an extended period of time producing chronic or long-term effects. High doses tend to kill cells while low doses tend to damage or change them. High doses can kill so many cells that tissues and organs are damaged. Effects from low doses spread out over long periods of time and may not be observed for many years.

1.3 Direct and Indirect Effects of ionizing radiation

Ionising radiation (IR) causes a cascade of events which begins immediately after it is emitted (Hill, 2017).

The initial ionisation (physical phase) is followed by immediate damage to vital macromolecules at the cellular level - **direct effect**, or indirectly through interaction with primarily water molecules, resulting in free oxygen radicals, highly reactive at the molecular level (physicochemical phase) - **indirect effect**.

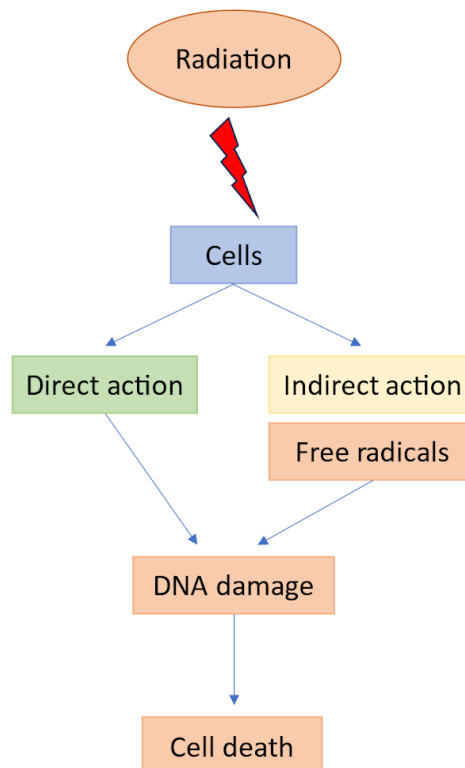


Fig. 1.1: Direct and indirect effect of radiation.

As cells are mainly made up of water, most of the damage caused by X-rays to biological molecules happens through free radicals, particularly hydroxyl radicals (Figure 1.1).

Free radicals resulting from these reactions initiate complex bio-chemical reactions, which can lead to the destruction or inactivation of vital molecules in the cell.

This is followed by the chemical phase when sublethal damage can be repaired.

The biological phase follows, when unrepaired DNA lesions are expressed (Figure 1.2).

Time Scale of Radiation Action	
Physical stage: Ionization, excitation	10^{-18} s
Physical/Chemical stage: Free radical formation	10^{-13} s
Chemical stage: DNA damage	10^{-6} s
Biological stage: Repair, replication, mutation, early and late effects	10^{-6} s Until years

Fig. 1.2: Stages of radiation action.

1.4 Time scale of the effects of ionising radiation

While the initial deposition of energy and the subsequent events occurs very quickly, the chain of biological effects that begins with the induction of cell death (necrosis or apoptosis) or repair of sublethal and potentially lethal damage, leading to tissue repair and remodelling can take much longer to express these effects (Fig 1.3)

There are also various factors that affect the cell's response to radiation:

- Physical factors (dose, dose rate, fractionation, LET (linear energy transfer) and RBE (relative biological effectiveness)): Different types of IR cause distinct biological effects: i.e., alpha particles have a high LET value but a short range of action and are thus less penetrating; in contrast gamma rays have a higher RBE and can affect a wide range of tissues.
- Chemical factors (radiosensitisers, radioprotectors, oxygen concentration)
- Biological factors (proliferative state, phase of the cell cycle, physiological or metabolic state, genotype).

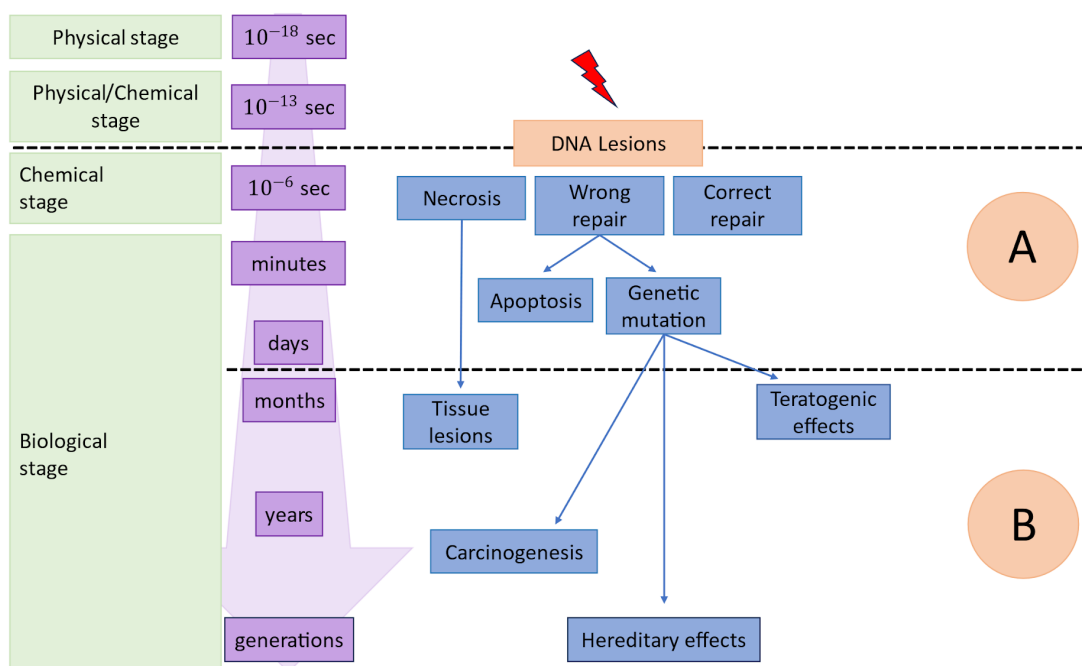


Fig. 1.3: Time scale of the effects of ionising radiation: biological changes manifest themselves after a latency period that can range from minutes to weeks to years after exposure. A - early effects; B - late effects.

Thus, when a cell absorbs radiation, there are several different possibilities:

- > the absorption of radiation may have no adverse effect
- > if injury occurs, it may be repaired without any trace of the exposure
- > if it is not repaired, the damage will be expressed in a time scale that depends on the proliferative status of the cell or
- > if it is not repaired, the apoptosis pathway may be activated.

The most frequent cause of radiation-induced cell death is the inability to correct the double strand breaks in the DNA chain and manifests itself when cells attempt the subsequent cell division.

Thus, the most proliferative cells express damage or die much earlier than cells with longer proliferation times. Consequently, biological changes in cells and tissues with subsequent specific organ and whole-body impact due to ionising radiation occur only after a latency period, which can range from minutes to weeks or even years, depending on a wide-range of factors (total dose, dose rate, cell kinetics, control of genes regulating the cell cycle.).

1.5 Radiation-induced damage

The damage caused by ionising radiation can be classified as follows (Hall and Giaccia, 2019):

- > Lethal damage
- > Potentially lethal damage
- > Sub-lethal damage

Lethal damage causes cell death and it is therefore irreversible. It can lead to tissue damage, organ failure and result in death of the patient. It is important to note that unrepaired DNA double-strand breaks are responsible for this type of damage.

Potentially lethal damage can be repaired if the environmental conditions post-irradiation are modified, *i.e.* if they are sub-optimal for cell growth. This means that under normal conditions this damage is not repairable and are therefore potentially lethal to the irradiated cells.

Sublethal damage involves a lesion in one strand of DNA or far apart in both strands. This type of damage can be repaired under normal conditions, as long as there is enough time for the repair process to take place and when no further damage occurs until full repair has taken place. This process may vary between 30 min and several hours, according to cell type.

In other words, the longer the time interval between two doses of radiation, the greater the number of cells that survive, because there is time to repair the damage before subsequent damage occurs (the surviving cells respond to the second irradiation as if they had never been irradiated).

After the Chernobyl nuclear accident, dosimeters worn by the workers were over-exposed and did not allow an estimate of the doses received. Nevertheless, using biological dosimetry (primarily based on chromosome aberrations) it was possible to estimate individual doses for various groups of exposed population. Biological dosimetry has inherent logistical and measurement limitations which may limit its use in catastrophic nuclear or radiological events *i.e.* chromosomal aberrations analysis has an upper limit of 5 Gy, requires a blood sample and 2 days to 3 days period before final results are available. As a consequence of these limitations, a wide range of planning and response activities must also include additional activities *i.e.* physical dosimetry, public information, development and procurement of medical countermeasures, first responder protections and many others.

1.6 Stochastic / Deterministic / Genetic Effects

Radiation induced effects can be classified according to the absorbed dose and biological response, depending on the time of manifestation and the organ involved. The dose of radiation absorbed by tissue (Gray) represents the amount of energy deposited (in Joules) per kg.

Depending on the dose and pattern of response, radiobiological effects are classified as deterministic or stochastic; in terms of the time of expression, as immediate or delayed; in terms of cells involved, as somatic or genetic (hereditary).

Tissue reactions, previously known as deterministic effects, are directly dose-dependent, increasing as the dose increases - the higher the dose, the greater the severity of the effects. But below specific threshold these deterministic effects do not occur. Once this threshold is exceeded the severity of the effect increases with dose. An example of deterministic effects is acute radiation syndrome.

With stochastic effects, the severity of the effects is independent of the dose, but the probability of occurrence increases with the dose. These effects are also known as probabilistic effects because they do not have a threshold dose.

This does not imply greater severity of effects, because if an effect occurs at a lower dose, its severity may be equally severe.

Stochastic effects result from DNA damage and mutations, they can occur with any dose, even very low doses (hence the widely used concept of ALARA - As Low As Reasonably Achievable – to prevent both unnecessary exposure and overexposure to ionizing radiation).

The probability of manifestation of these effects increases with dose, that is the higher the dose the greater the probability. Examples include radio induced cancer and hereditary mutations.

The Chernobyl nuclear accident (1986) led to major biological effects, with long lasting effects in morbidity and mortality in the affected population. Well known biological effects include thyroid cancer, especially in the children affected due to radioactive iodine-131 accumulating in the thyroid gland and an increased risk of other types of cancer (leukaemia, breast cancer).

The effects of radiation on human populations can be classified as somatic or genetic. Somatic effects represent the damage caused to individuals by different exposures throughout their lives.

In other words, they are the result of damage to individual cells and are only present in the individual that was irradiated, not in subsequent generations. As mentioned earlier, since tissues react in different ways to radiation, the severity of these effects will depend not only on the dose received but also on the organ involved. These effects can be delayed or even immediate, depending on how long they take to progress.

Hereditary or genetic effects are related to radiation that cause mutations that will appear in the offspring of the irradiated individual. Radiation induced mutations in the DNA of germinative (reproductive) cells have the potential to be inherited by future generations thus increasing the risk of genetic diseases in the offspring.

All these effects can take days to years or even generations to develop, and their severity varies according to many different parameters such as total radiation dose, time of exposure, type and dose rate of the source, volume and type of organ involved as well as specific characteristics of the population involved i.e. age, health status.

1.7 Acute Radiation Syndrome (ARS)

Radiological and nuclear accidents involve unintended exposure to radiation and therefore present a wide range of effects depending on radiation characteristics, duration of exposure and individual sensitivity of the different tissues. Some potential biological effects of radiological / nuclear accidents may include the acute radiation syndrome (ARS):

ARS, also known as radiation toxicity or radiation sickness is an acute illness caused by irradiation of the entire body major portion of body by a high dose of penetrating period of time.

In major radiological or nuclear accidents or intentional catastrophic events, a range of symptoms may develop shortly after exposure. The severity of ARS depends on type of radiation, total dose, time of exposure and multiple additional factors some of which are: patients age, chronic medical conditions, or traumatic injuries or burns (radiation and thermal) directly related to the detonation

ARS Syndrome includes:

- **Hematopoietic Syndrome:** occurs with doses 1 Gy to 10 Gy. Patients suffer from infection, haemorrhage and bone marrow aplasia that may lead to death unless a bone marrow transplant is provided.
- **Gastrointestinal Syndrome:** occurs with doses 10 Gy to 15 Gy. Death usually occurs in 2 weeks to 3 weeks due to irreparable damage to the gastrointestinal tract leading to dehydration, infection and electrolyte imbalance.
- **Neurovascular Syndrome;** occurs with doses higher than 20 Gy. Patients survive less than 3 days due to increased intracranial pressure and circulatory system breakdown.

1.8 Factors affecting tissue sensitivity to ionizing radiation: The Rs of Radiobiology

In addition to the molecular and cellular factors that determine intrinsic cellular radiosensitivity, the complexity of the response increases with the characteristics of the surrounding tissues: while cells can respond differently to the same dose, tissue responses are essentially determined by the rate of cellular proliferation, the extent of cell differentiation, metabolic activity and tissue repopulation.

Highly proliferative tissues such as bone marrow, gastrointestinal mucosa and epithelial tissue are particularly sensitive to ionizing radiation. In contrast, less proliferative tissues, expressing greater differentiation are more radioresistant (i.e. muscle and nerve tissue).

Another major factor influencing radiosensitivity is oxygen concentration. Tissues with low content (i.e. connective tissue) are more resistant to radiation due the decisive role of O₂ in the formation of free radicals and subsequent DNA damage.

The 5 Rs of radiobiology, namely repair, redistribution, repopulation, reoxygenation and radiosensitivity provide an insight into the mechanisms through which cells respond to ionizing radiation (Hall and Giaccia, 2019). Their understanding contributes towards improving radiological protection strategies as well as advances in the effectiveness of radiation therapy.

Repair of sublethal damage includes all processes responsible for identifying and correcting radiation induced damage to the DNA molecule.

Redistribution: Cells in the late synthesis-phase (S) are more resistant to IR. Cells in mitosis (M) are the most sensitive. If an asynchronous population of cells in different phases of the cell cycle is exposed to IR, more cells are killed in the sensitive than in the resistant phases of the cell cycle.

Repopulation represents the cell proliferation that occurs to compensate for the destroyed population: This occurs when, after irradiation, the surviving cells divide to counteract for the cell death caused by the previous exposure.

Tissues respond to cell death by increasing their rate of cell proliferation. This homeostatic response to cell loss does not only occur after irradiation. However, as in these conditions cell death occurs when the cells attempt subsequent mitotic division, the onset of this process varies for different tissues and is related to the cell cycle time of the cells that make up the tissue.

An example: cells of the basal layer of the epithelium take 4 days to complete the cell cycle; 8 days after irradiation sufficient cell loss has already occurred to activate accelerated proliferation.

Reoxygenation is a process specific to tumor cells and is one of the main factors determining response to radiotherapy.

Finally, intrinsic **Radiosensitivity** is related to the genotype of individual cells leading to variations in response to IR. Different organs or tissues are composed of different cell types and the response to IR depends on the intrinsic sensitivity of the different cell populations in that organ or tissue and the proliferative features of each population.

1.9 Conclusion

Biological impacts from nuclear and radiological accidents will vary upon the type and dose of radiation, as well as the time and recurrence (single or serial) of exposure. The impacts of radiation on the body may show up rapidly (acute radiation syndrome, ARS or even death) or they may require several years after exposure (genetic effects or cancer). Although our current efforts are necessarily related to capabilities and capacities to prepare and respond to nuclear or radiological accidents,

these basic science and response principles must also be used to prepare for intentional catastrophic nuclear or radiological events. The experiences of Chernobyl and Fukushima nuclear accidents showed that dosimetry was the essential tool in the emergency situation for decision making processes, such as evacuation and application of protective measures. However, at the consequent post-accidental phases, it was crucial also for medical health surveillance and in further adaptation to changed conditions with regard to radiation protection of the affected populations (Barquinero et al, 2021).

Investigating and verifying the health effects of ionizing nuclear radiation accidents is essential to protect public health, as well as to provide information as knowledge and understanding are critical for emergency preparedness (Kobashi et al, 2020).

To know more:

www.iaea.org

www.unscear.org

www.icrp.org

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2 When do we need a sound risk assessment based on monitoring and modelling in a nuclear emergency to assure proper decision making and a balanced long-term health care for the population?

Wolfgang Raskob, Retired from Karlsruhe Institute of Technology (KIT), Germany

2.1 Abstract

The CONFIDENCE project (COping with uNcertainties For Improved modelling and DEcision making in Nuclear emergenCIes) was part of the European Joint Programme CONCERT¹. CONFIDENCE started in January 2017 and included 31 partners from 18 countries. The project was designed to better understand and reduce uncertainties relevant for decision-making in the planning, emergency and transition phases of a consequential nuclear or radiological emergency. Work package 2 was specifically designed to develop approaches and tools integrating external and internal dosimetric monitoring data to further develop a comprehensive picture of the radiological/nuclear situation. This data-driven analysis, supported by close linkage of dose simulations and risk assessment tools, will be critical to support operational and logistical decision making, most importantly medical screening and triage.

A key finding was that combining historical and current monitoring information with simulation results from mathematical models via data assimilation approaches was recognised as the most promising approach to dosimetry-informed emergency decision making. However, the work also showed that an operational approach integrating such readings and simulations in real time is not yet available. Nevertheless, it was strongly recommended that resources be devoted to filling this gap.

A second key outcome was an initial evaluation of a health risk assessment software. This software took into account the uncertainties of the dose results from a decision support system using e.g. ensemble approaches for the source term and the forecast weather. Although the software became available during the project, its usability in decision-making required further discussion.

Although the project clearly improved the understanding of uncertainties in the different phases of an emergency, the reduction of uncertainties could only be highlighted in a few examples. Operational implementation in decision support systems is still a task for the future.

2.2 Introduction

The Chernobyl accident in 1986 demonstrated the need for a coherent, harmonised, and sensitive response to nuclear emergencies to support the effective implementation of measures, and to assure that health impact on the affected population is minimised (e.g. Papamichail & French, 2005 and 2013). To address this need, decision support systems were developed, e.g. ARGOS (Hoe et al., 2009) and JRODOS (Ehrhardt & Weis, 2000; Levdin, Trybushnyi, Zheleznyak, & Raskob, 2010), two systems that are widely used in Europe and worldwide. However, an intrinsic problem for decision making in the different phases of a nuclear or radiological emergency is the uncertainty arising from

¹ <https://cordis.europa.eu/project/id/662287>

missing or incomplete information, especially in the urgent and early response phases of such an emergency (see Fig 2.1).

Incomplete information on the source term and the prevailing weather can result in dose assessments that differ dramatically from reality. Uncertainty is also an intrinsic part of model parameters. In the presence of uncertainty, ineffective decisions can often be taken (e.g. overly conservative or optimistic predictions, inadequate consideration of non-radiological risks), which can lead to more overall harm than good due to secondary causalities, as observed after the Chernobyl and Fukushima accidents (e.g. Tsuboi et al, 2022).

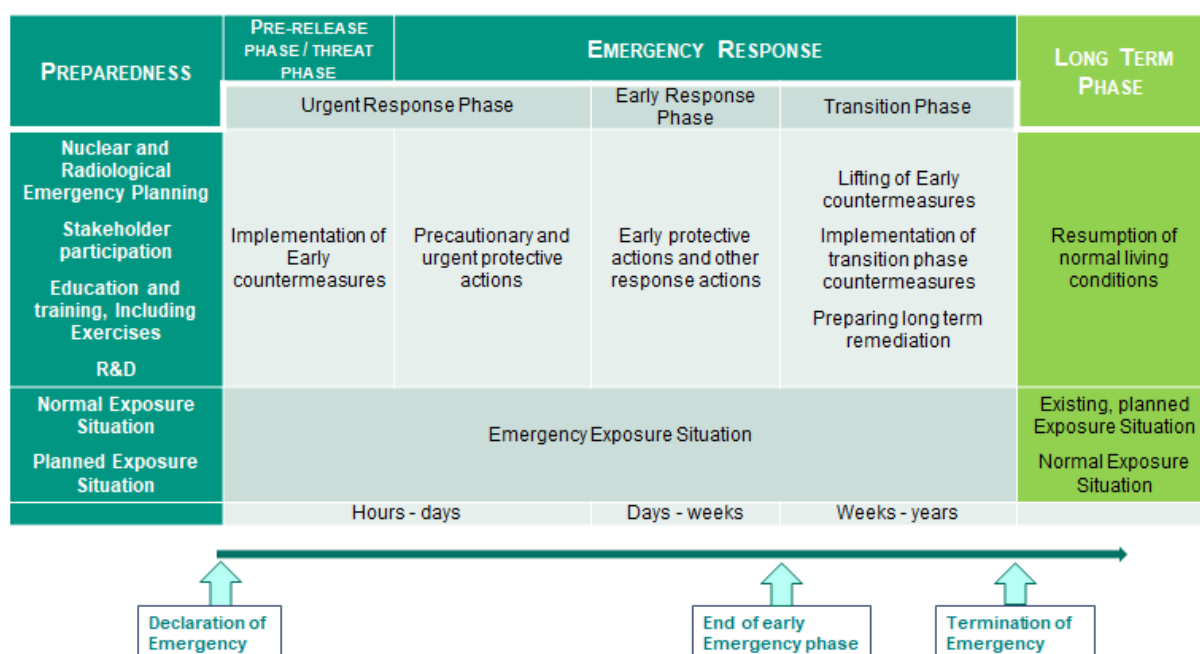


Fig. 2.1: Phases of a nuclear or radiological emergency (adapted from NERIS 2019).

Decision-making in the emergency phase is typically based on deterministic model simulations. Thus, recommendations related to evacuation, sheltering and distribution of iodine tablets are often issued before the release begins. Based on prior assumptions or tabletop exercise results, the effect of evacuation and iodine tablets will not be optimal. At later stages, monitoring is sufficiently advanced to provide a robust contamination and dose map. There are also times when some monitoring information is available together with the simulation results. However, to date there is no operational tool to combine simulated dose estimates with monitoring data to provide a more consolidated picture of the radiological situation. Data assimilation using results from stationary monitoring systems during cloud passage as well as mobile monitoring could fill this gap.

Risk assessment in the early and transitioning phases is completely missing in managing emergencies. However, this is important for decision making about the need for medical surveys and long-term health support for the affected population.

This paper will address both the use of monitoring information for decision-making and an approach allowing performing risk estimations early in an emergency. In order to perform operational risk assessments early in an emergency, the radiological picture, in particular doses to the affected population must be derived in a conclusive manner. This includes simulation, monitoring, assimilation of both and assessment of uncertainties. This was recently studied in the frame of the CONFIDENCE project (2017 – 2019).

For further reading on all aspects mentioned above, please have a look on the results of the CONFIDENCE project (Raskob et al. 2020a). All papers are available under open access at <https://www.radioprotection.org/articles/radiopro/abs/2020/02/contents/contents.html>.

2.3 Using monitoring Information

Monitoring stations, in particular gamma dose rate information, are typically available around nuclear power plants (NPPs) and also at national level. The data exchange platform EURDEP provides means to collect and exchange such information (<https://remap.jrc.ec.europa.eu/Advanced.aspx>). Figure 2.2 shows gamma dose rate information from September 2023. It is clear that in some countries the monitors are dense, while in others they are mainly concentrated on sensitive objects.

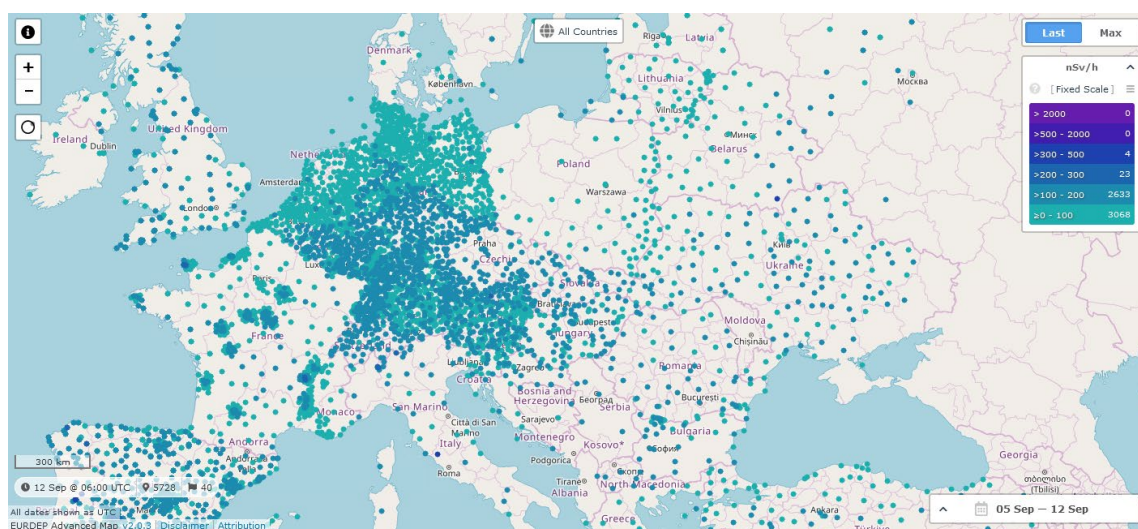


Fig. 2.2: EURDEP data exchange platform - gamma dose rates monitor Information from September 12, 2023.

In addition to gamma doses, monitors providing spectroscopic information are important for estimating the nuclide vector from the release. In addition, some countries operate air concentration monitors that can sample and measure gaseous and particulate fractions separately.

Mobile teams can use helicopters, cars and, increasingly, drones to provide detailed and accurate monitoring information for areas of interest.

The above-mentioned information allows to improve/adapt decision-making. Figure 2.3 shows results from a demonstration with an arbitrary source term and NPP location. The area in blue is proposed for sheltering the population for at least 24 hours. However, possible monitoring information – and this is available, as can be seen from Figure 2.2 - can be used to modify the proposed countermeasure area. In this respect, monitoring information is most important to adjust decisions taken in the early phase of an emergency.

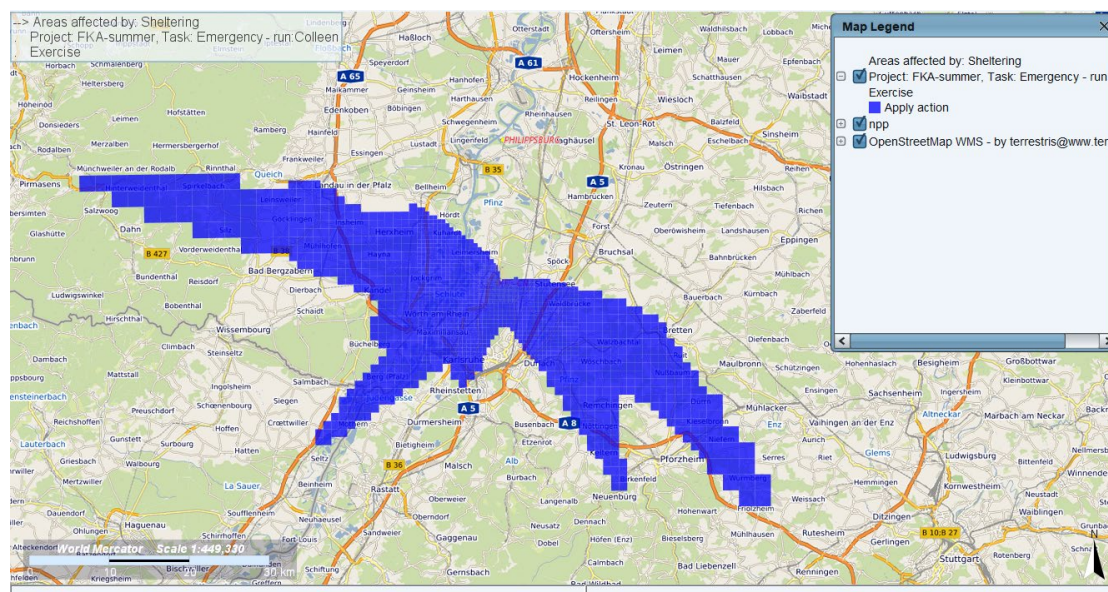


Fig. 2.3: Areas affected by sheltering from prognostic simulations.

Bleher et al. (2019) described a tool that was developed at the German Federal Office for Radiation Protection (BfS) to obtain individual dose assessments for people following a nuclear or radiological emergency.

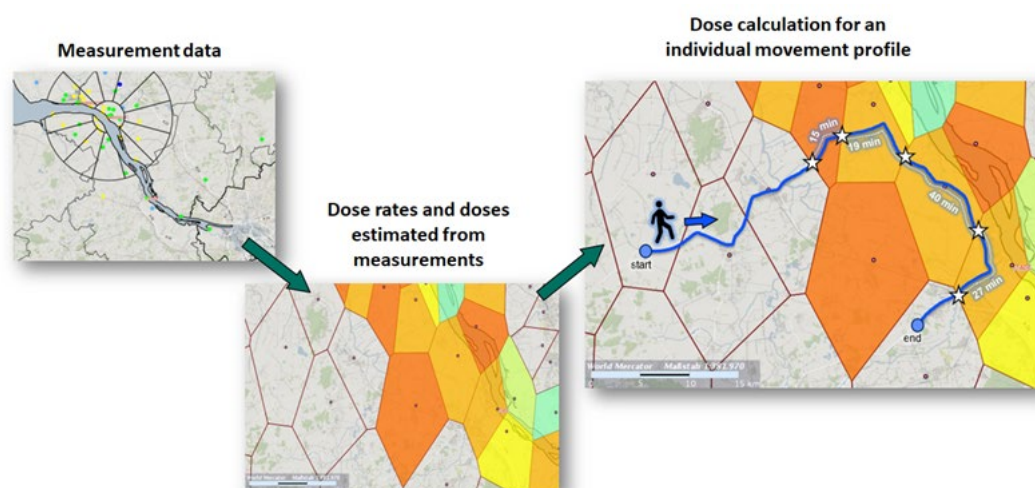


Fig. 2.4: Tool at BfS to estimate individual doses following a release of radionuclides.

Figure 2.4 indicates that larger areas might be affected by potential countermeasures. The area for sheltering amounts to about 1000 km² and affects about 500 000 people. Iodine tablets were distributed to about 330 000 children and 233 000 adults in even larger areas. In this respect, large areas need to be monitored in a fast and reliable manner. In the early and brief time window in which the above-mentioned countermeasures should be rapidly dispensed, very little monitoring information might be available. However, data assimilation (Figure 2.5) – combining measurements with simulation results – might be one possibility to overcome that problem (see e.g. Eleveld et al. 2007, Gering 2007, Raskob et al. 2010, Zhang and Wang 2022).

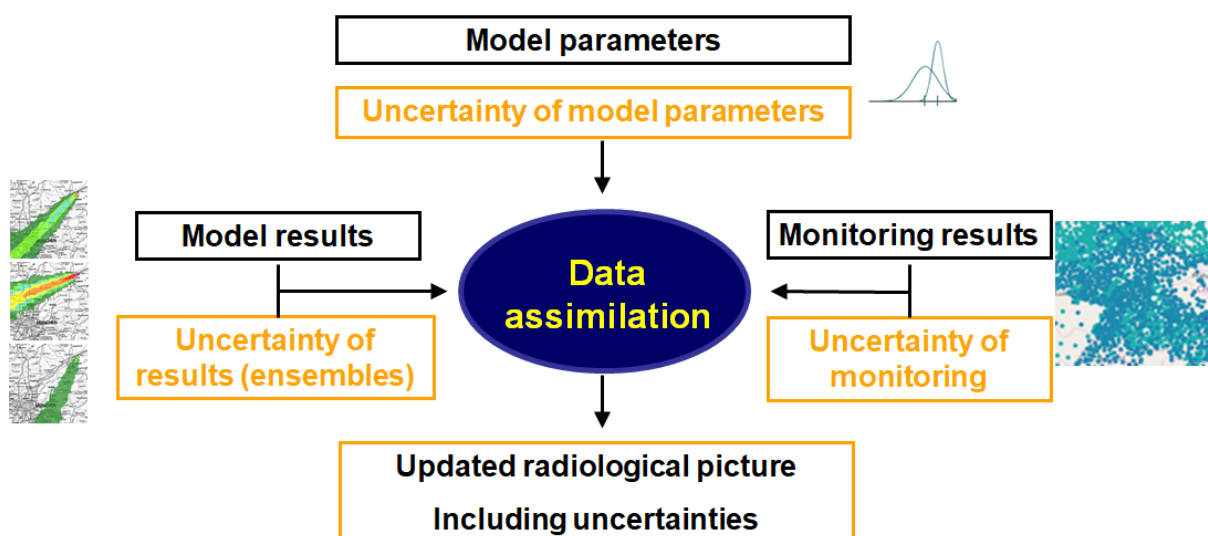


Fig. 2.5: Data assimilation scheme (adapted from Bleher et al. 2020).

2.4 Risk assessment early in the emergency

Accurate risk assessment is only possible if a reliable picture of the doses to the affected population can be obtained. Such an assessment is an integral part of the long-term management following a radiological or nuclear emergency. It is typically used to evaluate the need for health care actions in the contaminated areas. However, with current technologies it takes time to obtain a consolidated picture of the radiological situation taking into account all relevant exposure pathways. The Fukushima accident from 2011 is a good example. The World Health Organisation (WHO) did not publish their health risk assessment report until 2013 (WHO 2013). This report is based on “preliminary” dose estimations. As one can see from Figure 2.6, it took about one year to obtain the preliminary dose estimates and additional nine month before the risk report was published.

2.5 Effective system and processes

Within the CONFIDENCE project, work was carried out to demonstrate how this gap can be closed. Means to perform a sound dose characterisation of the population are available with data assimilation procedures – even if not operational, but the theoretical framework was further improved within CONFIDENCE (Bleher et al. 2019). One pre-condition when developing this tool was that the risk estimation method should be based on the same methodology as used by the WHO (WHO 2013, Walsh et al. 2014). There are two major tasks to be performed. First, effective dose limits had to be converted to the organ equivalent doses as they are mostly relevant for radiation risk assessment. Second, the risks of all solid cancers and leukaemia per unit relevant organ had to be estimated with appropriate models. This is necessary as the use of effective doses for risk estimations is not recommended (ICRP 2007). The methodology, e.g. definitions of necessary quantities such as Lifetime Baseline Risk (LBR) and Lifetime Attributable Risk (LAR) can be found in Walsh et al. (2019).

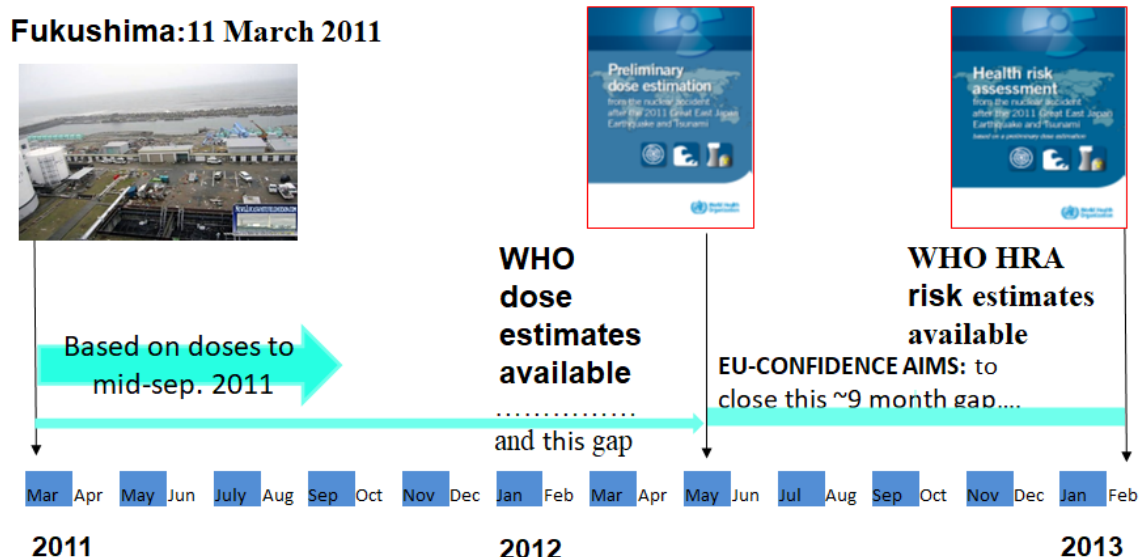


Fig. 2.6: Timeline from the accident to the risk report.

The methodology results in the numbers of cases expected per 10,000 persons, at 100 mGy organ dose, over lifetime exposure. This reference number can be scaled to any dose monitored. Important to note is the fact that also uncertainty in monitored values as well as in the dose estimations are considered in the methodology.

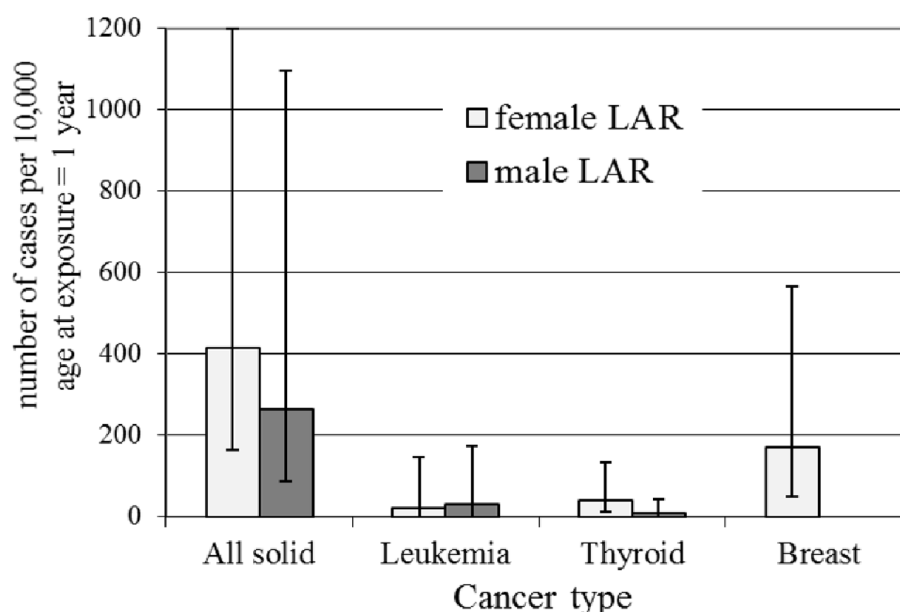


Fig. 2.7: All cancers considered for 100 mSv effective dose.

Data considered for the derivation of the reference value of 100 mSv are the following (references in Walsh et al. 2019):

- > German cancer register (RKI) – incidence and mortality rates for the selected cancer outcomes
- > Scandinavian (Sweden, Denmark, Finland and Norway) cancer statistics (NORDCAN, IARC) – incidence and mortality rates for the selected cancer outcomes
- > Switzerland
- > General survival data from life tables for Germany (Destatis.de) and other European

- Within CONFIDENCE, this methodology was integrated into a computer tool for test and application. However, only the external dose was considered in this first simulation model. To apply the tool, the following information is required:
- The average population size & structure in region of interest is used to calculate the absolute number of expected cases
- Population is subdivided into three age groups (female as conservative assumption)
 - Children (5 to 10) years
 - Adults (20 to 40) years
 - Seniors > 65 years

With that information on the affected population, the doses derived via simulation, monitoring or both – via data assimilation – can be used to estimate the absolute number of expected cases (spontaneous and radiation-attributed). If uncertainty information is available, e.g. by using not only one deterministic simulation but an ensemble of numerical weather and source term data (Leadbetter et al. 2020), these uncertainties can be also considered in the risk estimations.

2.6 Conclusions

Within the CONFIDENCE project, a sophisticated risk estimation software was developed and tested, however only using doses from external exposure pathways as input. Nevertheless, that tool demonstrated the ability to derive health risks for the affected population as soon as dose estimates are consolidated. This would close the nine-month gap between the preliminary dose estimations and the first risk report after the Fukushima accident (WHO 2013). Risk assessment early in an emergency would allow authorities to plan health response activities well in advance and would also be a valuable tool for communicating with the public and enabling informed decision-making.

Effort in CONFIDENCE was also devoted to improving the operational picture based on improved monitoring approaches and by combining both, results from numerical simulation model and monitoring via data assimilation approaches. Several demonstration projects were developed and demonstrated at the final workshop (Raskob et al. 2020a).

What is missing so far is the translation of science into practice and operational use. End users of the DSS and decision-makers attending the final dissemination meet identified this is a priority. (Raskob et al. 2020b). The current European research project PIANOFORTE might be a possible basis for such new developments as it contains open calls with topics that can be defined by all relevant stakeholders over the next few years (<https://pianoforte-partnership.eu/>).

The most significant outcome of the CONFIDENCE project is the realization that operational awareness is an essential component of increasing the planning and operational capabilities described in the CONFIDENCE project. The pre-event modelling and simulations shown are a strong foundation for the tools necessary for planning, response, and recovery activities by designated end users or operational groups. Understanding the scope and requirements to move to this next step is a critical task of the greatest importance and urgency.

Further, the demonstrated collaboration between the countries, alliances, in-country resources, etc. is evident from the results of the CONFIDENCE project. However, moving to the next steps requires additional partners, within the alliances, that are responsible for planning and implementing operational decisions in 'near real time'. It will also be necessary to engage the physical resources available to, among others, operational modelling concepts, hardware, software, communications, and data analysis.

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3 Preparing for people monitoring and using the data to inform the wider monitoring programme

Matthew Simpson, Tony Riddell, Duncan Cox, United Kingdom Health Security Agency

3.1 Abstract

Within the United Kingdom, the National Health Service [NHS] has the responsibility for providing health monitoring in all situations. While they have wide-ranging capabilities for individual specific health requirements, United Kingdom Health Security Agency [UKHSA] maintains the equipment and trained personnel to perform specific large-scale people monitoring for radiological events and support the NHS and other first responders. UKHSA would also set up and run the national emergency response centre and associated monitoring coordination, guiding the short to medium-term monitoring of people and the environment.

Recent developments in easy-to-use data management and collection systems allows the electronic collection of the data from people monitoring. This then allows live accessing of the internal and external contamination levels monitored by both dose assessment teams and monitoring coordination bodies. These two areas can then utilise the data to provide more accurate internal dose assessments for individuals and identify the areas where individuals are getting higher (or lower) levels of contamination than predicted, thus allowing better-targeted monitoring.

3.2 Methodology

3.2.1 Legislative Environment

Radiological emergencies will vary from incident to incident as the nuclide/s, dispersal mechanism, location, or size changes – area or individuals exposed. Preparation for any incident needs to have sufficient flexibility to cover alpha, beta, or gamma contamination and be scalable from 10's to 1000's of people. It also needs to integrate into the other responding services in the area.

In the United Kingdom, the Civil Contingencies Act (Cabinet Office, 2006) provides the broad operating parameters for the agencies that would respond to any major incident of which a Radiological/Nuclear event is one. The incident would be managed by the Police and be supported by Health Services, Fire Service, Local Government, Central Government, and other agencies depending on the nature of the event. The Act requires all the agencies to work towards suitable plans to meet foreseeable incidents effectively driving emergency preparedness in all areas including radiological and nuclear.

During a radiological or nuclear event, people monitoring is provided by the National Health Service [NHS] as part of their Public Health responsibilities (UK Government, 2018). The UK Health Security Agency [UKHSA] is an agency within the Department of Health and works alongside the NHS providing large-scale radiological people monitoring through the Radiation Monitoring Unit [RMU] (Thompson et al., 2011).

In the last five years, additional legislation has been brought in that further strengthens the need for people monitoring and the preparation for it through REPPIR (UK Government, 2018). This directs locations with significant radiological risks to have enhanced planning in place and specifically mentions the need for people monitoring.

3.3 Radiation Monitoring Units

UKHSA have developed the RMU capability over the last 20 years and has benefited from the international work performed in this area (IAEA, 2000) (IAEA, 2005) (Rojas-Palma et al., 2009). Improved instrumentation such as portal monitors and nuclide identifiers supplement the broad array of handheld instruments now on the market. There has also been work coordinated by IAEA, ICRP, EU, and WHO developing methodologies and suggested action limits (IAEA, 2000). From this and the examination of the risk scenarios, a series of decision matrixes have been developed that would guide the monitoring protocols needed.

The UK has split the monitoring process into four stages and each area is discussed below. Where possible, the RMU should be situated in a building providing light, heating power, and welfare capability. The RMU would only be required for large numbers of potentially contaminated people (100's) and hence be in place for days if not weeks after the event and the capability requires main power to maintain this for any duration.

The flow of people through the facility also needs prior consideration as it is sensible to ensure a one-way system and prevent people pre- and post-decontamination from close interaction that could cause recontamination. Most large facilities should be able to provide this flexibility, however, by prearranging possible venues, this setup can be assessed prior to an incident, making the setup and running easier.

The staffing to run the facility is split between the local government, who will manage the facility and logistics of moving the people to and from the facility, and the technical staff. It is estimated that 8 to 10 technical staff are needed for a throughput of 250 people per day, multiplied up if a greater throughput is needed.

3.3.1 Stage 1

Stage 1 is the initial triage designed to immediately identify anyone with significant levels of external contamination. The method for this depends on the available resources and the threat. It is easiest to perform with portal monitors either in walkthrough mode or with a limited dwell time (10 seconds to 15 seconds) which can ensure maximum throughput while still ensuring detection levels can meet the desired outcomes. If no portals are available, this can be done using hand-held instruments by very quickly running the instrument over the front and back of the individual taking about 1 minute. If the nuclide of interest is an alpha emitter, portals are of no use, so hand-held instruments with an alpha contamination capability must be utilised.



Fig. 3.1: Stage 1 Portals

The limits used should be worked out prior to any event, based on national legislation and international guidance. While this may under- or over-estimate the actual threat, not having a trigger limit set will require high-pressure decision-making in the field, which is likely to cause more issues. As the situation develops, advice can be given by the emergency operation centre which will have access to all the information and personnel to provide improved trigger levels, remembering that at this stage it is a decision over the need for immediate decontamination and then further monitoring (stage 2).

This area is likely to be most at risk from contamination being spread and hence suitable protection is required for the area and the staff. Protective flooring can be used to minimise the longer-term

requirement for remediation of the facility used and to also enable rapid replacement if contamination is found. This then needs to be coupled with contamination monitoring of the environment as per normal contamination control measures. PPE such as coveralls, gloves, overshoes, and masks will help further protect the staff. There should also be standard radiation safety training for all staff involved in monitoring. Equipment can be protected using thin plastic films - while it slightly reduces the efficiency, it greatly reduces the chances of it becoming permanently contaminated and hence unusable. Note this is not possible for alpha emitters, as it would prevent the instruments from working.

3.3.2 Stage 2

This stage is designed to find where any contamination might be on an individual to ensure the triage monitoring (stage 1) did not miss anything. It should also work to a lower trigger level for an area, as opposed to the whole-body levels used for the previous stage 1 monitoring.

Consideration should be made to the areas to be monitored. In the early stages, the contamination will be airborne, so the head, hair, and face should be monitored. Contamination pick up while moving through a contaminated area will be on the lower legs and feet. Finally, it is very difficult not to touch surfaces, so the hands, elbows, knees, and buttocks (sitting and leaning), and the bottom of bags or rucksacks should be monitored as well.



Fig. 3.2: Stage 2 monitoring stations

It will take longer to process individuals through this section of the process, compared to stage 1, as monitoring all the appropriate areas will take a few minutes (at least five minutes if it is an alpha-only emitter, as monitoring will need to be slower and more careful). As such, as many stations as possible should be set up to prevent backlogs or flow management earlier in the process will be required to restrict excessive build-up of individuals.

As with stage 1, anyone exceeding a trigger level should be decontaminated and then remonitored at stage 2. An individual should only be subject to decontamination twice, as further decontamination is not beneficial and positive monitoring results at that point are likely due to internal contamination, which will be assessed at stage 3.

If an individual has been monitored at stage 2 and not exceeded any trigger/action levels, they are deemed clean enough to be sent home via stage 4. Anyone who has required decontamination should be sent through to stage 3, as it is probable that they have some internal contamination that will require quantifying.

As this area is receiving people who could still have loose contamination (but at a lower level than in stage 1) there is still a requirement for the control of contamination. Floor covering and regular monitoring is required, and PPE should still be worn by the staff.

3.3.3 Stage 3

Stage 3 monitoring is looking for internal contamination and hence is reliant on all external contamination being removed from the individual. Thus, all people going through stage 3 should already have passed through stages 1 and 2. However, this does not have to be at the same venue or the same time, as it may not be possible to have a venue large enough to house all parts. If this is

the case, individuals can be provided with a urine collection kit, sent home, and ask to attend the stage 3 venue within 48 hours.

Stage 3 can be split into three parts with a rapid gross assessment for gamma emitters, a more detailed gamma emitter assessment, and then urine collection and subsequent analysis. As implied, if the possible contamination is a pure alpha or beta emitter, only the final urine collection and analysis should be used, because the other measurement equipment in use is not sensitive to these emitters.



Fig. 3.3: Stage 3 Whole body triage

A rapid assessment is performed using large NaI instruments connected to a scaler. If it is an Iodine-131 event, measurement of thyroid can be done at the throat and for any other gamma emitter at the centre of mass to give a whole-body assessment. The calibration of the instrument will require either whole-body phantoms or mathematical modelling, as it will be significantly different from any calibration provided by Instrument Calibration facilities that use point or plaque standards. Counting for 10 seconds should allow an estimate down to below 2 mSv, which can then be used as the trigger level for further monitoring.

Using portable HPGe detectors is used to accurately quantify the nuclide and activity of any internal contamination at the levels that would trigger from the NaI detectors. Again, a calibration using a full whole-body phantom or mathematical modelling is needed. The output will be activity related to the nuclide/s which is especially vital if the contamination is a mixed nuclide source. This can then be converted into a dose directly using generic assumptions and can be transmitted to dose assessment experts for a more accurate assessment using the increased information that will be available about the event and the individual.

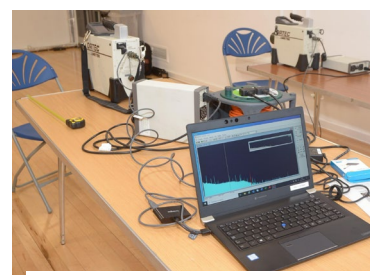


Fig 3.4: Stage 3 Whole body using HPGe

UK legislation and common-sense dictate that a provisional dose should not be provided to the individual at the remote monitoring unit (RMU). It is very unlikely that the individual will have sufficient knowledge to understand a radiation dose and its implications and the teams in the field do not have the time or the training to provide the required advice. As such, the information provided (which trigger levels have been exceeded) will guide the individual to the correct advice and support. If the individual could receive over 200 mSv, either from external or internal contamination, that would trigger a direct visit to a hospital, while anything else would default to a more routine visit to a healthcare professional.

The last part of stage 3 is the collection of urine samples. This will provide a good backup to the field measurements and offer improved accuracy to the dose assessment if the individual was not 100% decontaminated prior to the internal monitoring. Due to the detection limits needed for some nuclides, especially alpha emitters, a large sample is preferred. Also, most dosimetry models usually predict excretion in 24-hour intervals. This then leads to either a full 24-hour sample collection (-to 4 l) or a mechanism for correction from the provided to a 24-hour equivalent sample. This can be done by measuring the creatinine levels. Once collected, the sample can then be sent to a radiochemistry capability for analysis.

As all individuals visiting stage 3 should be free of external contamination, the need for floor covering and PPE is no longer required. Routine monitoring should remain in place as part of the management of the whole facility though.

3.3.4 Stage 4

When all monitoring has been complete, the individual needs to be released with sufficient information and provide enough information that any monitoring information can be passed accurately to their healthcare provider.

While the UK does not provide specific numbers for radiation dose to the individual, they will be provided information about what has occurred and what should happen now against the trigger levels used, i.e. you were not contaminated or you have received a small amount of internal contamination, please visit your healthcare professional. This is fully backed up with the national messaging and information provided by the government in support of the event and specific information for the healthcare professionals.

3.4 Decontamination

The UK has a standing announcement for any wide-scale contamination event, 'Remove, Remove, Remove'. This will be broadcast on media outlets with the idea to reduce the health effects on the individual. Remove yourself from the area, remove outer clothing and when possible, remove the contamination by fully disrobing and showering, usually when returning home. There are obvious disadvantages to this - those affected will be leaving the scene and hence both spreading contamination but also are more difficult to trace. The advantage is that the individual will have a significantly lower chance of a significant dose from the contaminant, and especially with radiation lowering the potential beta/gamma dose through clothing and from surfaces. The advantages to the individual far outweigh the possible disadvantages.

Mass decontamination is usually a large, disrobed showering facility and provision of replacement clothes. The large UK capability is provided by the Fire Service, but smaller facilities are provided by other first responders. It is agreed that the mass decontamination will be available before the RMU is fully operational and hence anyone suspected of being contaminated would likely be put through the decontamination and then sent to the RMU for monitoring. The same level of mass decontamination will also be attached to the RMU to enable the decontamination required there to be enacted.

The facility will need to manage both the solid and liquid waste. People's clothes and belongings will need to be collected and bagged for either disposal or decontamination after storage. The wastewater from the showers will need to be banded and collected during operation.

Another method for decontamination is using wipes in a buddy/buddy method. For instance, while waiting in the initial line for the RMU, wipes can be used to clean outer surfaces and skin with people cleaning those in their family or friend group. This has two advantages of both helping to decontaminate and occupy people who are likely to be in a high state of anxiety. Most individuals now have mobile phones, and they are very important containing payment methods and communication information, most people will not be prepared to give up their mobile device so by offering wipes to clean them significantly reduces any contamination risk and reduces the possible need to remove the devices. However, it is difficult to manage delivery of the wipes and more importantly collection of the potentially contaminated waste, so this will require significant manpower.

3.5 Data Management

The management of data throughout this process is vital. We need to ensure the correct information is collected about the individual and the associated monitoring data and that it is securely managed to the right outcomes. Traditionally, most has and some still will be collected physically as this is the best option to ensure the collection can occur, as electronic collection can be subject to loss due to power or connectivity.

However, with the access to 4G, it is now possible to get internet access for most connectable devices in most locations. The UK has about 92% coverage, however all urban areas have coverage and an RMU is only required in and around large population areas (Hutton & Baker, 2022). It is therefore possible to use electronic collection methods using an app or similar. This then allows the data to be immediately securely stored and then accessible for further analysis away from the field deployment.

This data set offers two advantages - one for the individual and one for a wider situational awareness. If an individual exceeds an internal dose trigger in the stage 3 internal assessment the information can immediately be accessed and used to perform a more in-depth dose assessment. All in-field assessments use generic factors around the breathing rate, gender, age, particle size, chemical form etc. all of which make a difference to the estimated dose. These values can be ascertained, updated and then used by internal dose experts to provide the improved dose estimate. They can also use all the most up to date data sets, such as those provided by ICRP (ICRP, 1996) (ICRP, 1995a) (ICRP, 1989) (ICRP, 1993) (ICRP, 1995b). This can then be transmitted to the healthcare professionals who will have the information prior to the arrival of the individual.

A secondary advantage is obtaining location data with any monitoring result. By knowing where someone was when they got contaminated and their monitoring results, it is possible to get an improved understanding of the areas where people that need monitoring could be or were and hence target those over others who are at lower risk. Standard modelling would provide downwind cone which often cannot initially account for urban changes caused by large structures and topography. In the UK, it is not appropriate to collect and share personal data where it is not required, therefore, any data used at this point should be anonymised prior to use, i.e. a location and value used, not identifying any individuals.

3.6 Summary

In UK legislation, the NHS is responsible for the health of the public and during a radiation emergency this includes radiation monitoring. UKHSA provides the specialist radiation monitoring capability in support of the NHS and other first responding agencies. A facility to provide this monitoring will be set up and the public will pass through the four stages. Stage 1 is a rapid triage from which a decision is made about immediate decontamination. The individual then passes to the stage 2 monitoring for more area specific monitoring and again will go for decontamination if any trigger levels are exceeded. If contamination was found, they are sent to stage 3 for internal contamination assessment. If no contamination has been found, the individual is then sent to stage 4. At stage 3, they are assessed for internal contamination and provided with a urine collection kit for bioassay and then pass to stage 4. At stage 4, they will be provided with welfare and information prior to being sent to home or hospital, depending on the results of the monitoring.

Initially, individuals will be asked to perform self-decontamination and then a more robust decontamination is performed by the Fire Service both at the site of the event and in conjunction with the RMU.

Any data collected should be put to best use both for the individual by getting improved dose assessment but also to help improve the situational awareness of the emergency planners and incident commanders while still ensuring individuals rights are maintained.

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4 EURADOS developments on emergency internal dosimetry

María Antonia López, CIEMAT – Avda. Complutense 40, 28040 Madrid, Spain

4.1 Introduction

Intakes of radionuclides may occur by inhalation, ingestion, injection, through intact skin or a wound. The doses due to internal exposures cannot be measured directly but they need to be assessed by the interpretation of monitoring data of retained activity in the body, activity in biological samples or activity concentration in the air. Biokinetic and dosimetric models are defined to describe the metabolic behaviour of contaminants inside the body and the internal irradiation of organs and tissues.

EURADOS Strategic Research Agenda (SRA) [1] was published as a EURADOS Report in 2020 [2] with the visions for radiations dosimetry over the next two decades. Vision 3 of SRA focused on an efficient dose assessment in case of radiological emergencies, and challenge 1 was addressed to quantify doses from internal emitters after accidents.

EURADOS established a work program to deal with internal dosimetry in case of radiological and nuclear emergencies, in the frame of EURADOS SRA, focusing on the:

- improvement of in vivo monitoring techniques for adults and children;
- calibration, monitoring, dosimetry and clinical management in case of internal contamination through wounds;
- rapid in vitro emergency bioassay methods for alpha emitters (actinides) and ^{90}Sr ;
- linking internal dosimetry, after accidental intakes of radionuclides, with biological dosimetry methods.

Achievements and current developments of these research topics are presented in the following paragraphs.

4.2 Improvement of In Vivo Monitoring Techniques for Adults and Children

The EURADOS survey on the *in vivo* monitoring data obtained from the measurement on foreign visitors, that were in Japan during the Fukushima Daiichi NPP accident [3] and potentially exposed, showed the relevant need of performing proper age dependent calibration of in vivo monitoring systems to determine the retained activity of radioiodine in the thyroid and of gamma emitters (mainly ^{134}Cs , ^{137}Cs and ^{132}Te) in total body. Figure 4.1 shows a plot of in vivo monitoring data of 176 individuals with activity detected in the thyroid or in total body, at a time after 11 March 2011.

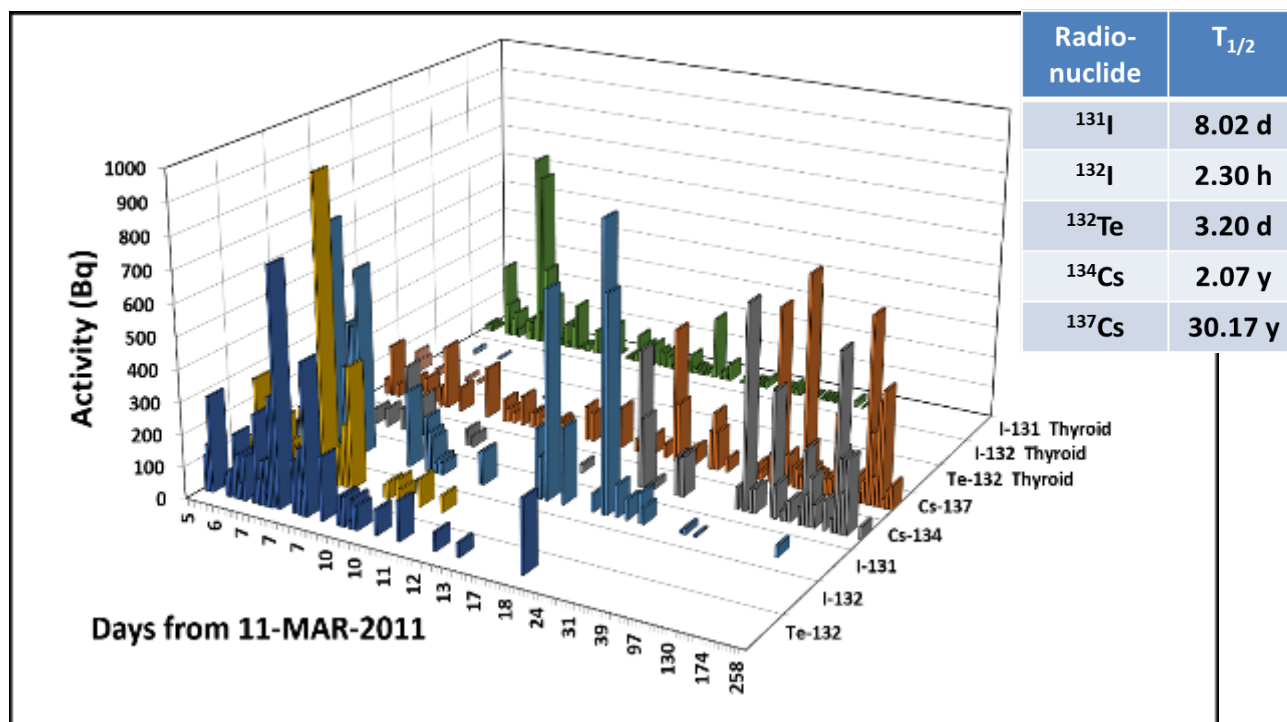


Fig. 4.1: EURADOS survey: in vivo monitoring data of persons with activity > Detection Limit at t (days) after 11 March 2011 [3].

4.3 CATHYMARA Project

The CATHYMARA project “Child and Adult Thyroid Monitoring after Reactor Accident” (2016-2017) was developed by members of EURADOS WG7 on internal dosimetry and WG6 on Computational Dosimetry. An age-dependent thyroid calibration phantom corresponding to the adult and 10 years and 5 years-old children, was fabricated by SCK-CEN in Belgium (see Figure 4.2) and was used in two intercomparison exercises organised for spectroscopic and non-spectroscopic devices. Results of both exercises were successful, showing a good performance of European devices to quantify radioiodine in thyroid.



Fig. 4.2: Cathymara thyroid phantom, simulating contamination of ^{131}I in the thyroid of an adult and 10 years and 5 years-old children, fabricated at SCK-CEN (Belgium).

Technical recommendations [4] were established at the end of the CATHYMARA project for thyroid monitoring in case a nuclear accident, and a summary is presented as follows:

- To perform iodine measurements in thyroid 1 days to 7 days after incorporation via inhalation, during the initial phase post nuclear accident, because earlier measurements can be biased by non-thyroidal content;
- To perform measurements with germanium detectors considering, at least, a representative group of the exposed population, to identify the isotopes incorporated, especially those of short half-life;
- To use age dependent “Dose content” functions $Z(t) \text{ Gy Bq}^{-1}$ and $Z(t) \text{ Sv Bq}^{-1}$ for dose assessments from monitoring data;
- To promote participation in intercomparison exercises and nuclear emergency panels;
- To promote citizen participation in thyroid measurements by providing detection equipment, training, tools and database access for recording results;
- To initiate constructive dialogues with members of civil society.

4.4 CONFIDENCE Project

The CONFIDENCE Project “Coping with uncertainties for improved modelling and decision making in nuclear emergencies” (2017-2019) was developed with the involvement of EURADOS WG3 on Environmental Dosimetry, WG7 on internal dosimetry and WG10 on Retrospective Dosimetry. The main concern in this project was focused on the improvement in individual dose measurements techniques following nuclear emergencies for improving situation awareness and risk estimation [5]. In the early phase of the emergency, environmental monitoring data provide information on the potentially affected population group, and a preliminary individualized dose calculation according with geographical location of individuals allows the identification of the critically exposed subgroup (see Figure 4.3). Individual dose measurements are then carried out.

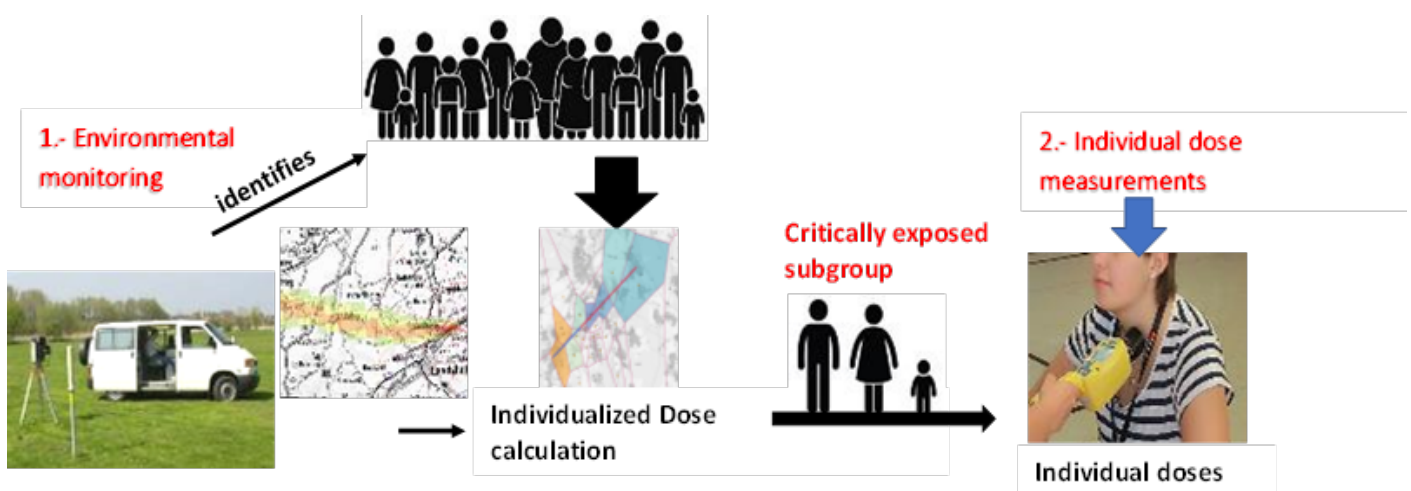


Fig. 4.3: CONFIDENCE WP2: improving the situation awareness in the early phase of a nuclear accident.

During the CONFIDENCE Project age-dependent Dose per Content functions $Z_{I-131}(T,A)$ were developed for each age group “A” at a time “T” after intake. The Dose per content function directly relates the measured ^{131}I activity content in the thyroid $M_{I-131}(\text{Bq})$ at the time T after the intake, with the committed absorbed dose to the thyroid at the 30th day, e.g. from a single acute intake of ^{131}I :

$$D_{I-131}^{Th(30d)}(T, A) = M_{I-131}(T) \cdot Z_{I-131}^{Th(30d)}(T, A)$$

Correction of the ^{131}I dose must be considered in case of intakes of all short-lived iodine isotopes and ^{132}Te .

The IDOSE tool [6] was developed by RPI (Ukraine) during the CONFIDENCE project as Data processing App for Dose Assessment of radioiodine intakes, which can be operated in smartphones and tablets. Input data are the time of intake, the location of individual, the birth date, the sex, the counting date and the activity $M(\text{kBq})$ of ^{131}I and ^{132}I (if available). The database includes age-dependent dose per content functions for infants, 1 y, 5 y, 10 y and 15 y old children, adult male and female. The output data are 30-days thyroid absorbed dose and effective dose.

4.5 EIVIC Project

The EIVIC Project “European In Vivo Counting Intercomparison Exercise” (2020-2022) was funded by European Commission (DG Energy Luxembourg), and was developed by IRSN, BfS and EURADOS WG7 members of CIEMAT (Spain) and KIT (Germany). The EIVIC project [7] counted with the participation of 41 Whole Body Counters (WBC) of 21 countries (18 from European Union). Two brick phantoms were provided to participants with radioactive rods, simulating homogeneous internal contamination of an adult man in different intake scenarios.



Fig. 4.4: EIVIC brick phantom simulating internal contamination of gamma emitters in total body in an adult individual.

The aim of the EIVIC project was comparing the results of European WBCs using different detector systems, calibration phantoms and counting geometries. This is important in preparedness for cross-border collaboration in emergencies. The exercise consisted in 4 tasks:

Task 1 – Identification/quantification of ^{60}Co , ^{133}Ba , ^{137}Cs

Task 2 – Nuclear emergency scenario with ^{134}Cs and ^{137}Cs

Task 3 – Radionuclides employed in nuclear medicine ^{68}Ge and ^{88}Y

Task 4 – Calibration procedures with ^{133}Ba and ^{152}Eu (suitable only with HP Ge detectors)

The result of the intercomparison showed a good agreement in most cases, for the majority accredited and not accredited labs, that employed NaI(Tl) or HP Ge detectors and brick or bottle phantoms for calibration.

Next step to be considered is the EURADOS intercomparison on children monitoring.

4.6 EURADOS Intercomparison of age-dependent thyroid phantoms

The EURADOS Intercomparison of age-dependent thyroid phantoms has been organized at CIEMAT Whole Body Counter as part of the research program of the Task Group “Internal Dosimetry for Emergency” within EURADOS. Age-dependent thyroid phantoms fabricated by 7 institutes from European countries and Japan (see Figure 4.5) were sent to CIEMAT and measurements were performed using the same germanium detector inside a shielded room. Counting efficiencies and detection limits were obtained and compared for different age groups. The analysis of the results is in progress. Important parameters to take into consideration are the tissue equivalence of the materials, the thyroid volume, the overlying tissue thickness, and the thyroid lobe dimension.



Fig. 4.5: EURADOS Intercomparison of Age-dependent Thyroid phantoms.

4.7 Calibration, Monitoring, Dosimetry and Clinical Management of Internal Contamination through Wounds

The EURADOS/IRSN Intercalibration and Intercomparison for wound monitoring has been organized in 2022-2024 counting with the participation of 19 laboratories from Europe and Canada. Future plans may include to expand the project in Asia. Three wound calibration phantoms were fabricated by IRSN (France) simulating a puncture wound (see Figure 4.6). The phantoms are made of 10 tissue equivalent plates with a diameter of 10 cm and a thickness of 0.2 cm. The available depths of the source are 2 mm, 4 mm, 6 mm, 8 mm, 10 mm, 12 mm, 14 mm and 16 mm, from top to bottom. Detector-phantom distance for measurement was set to be ≥ 1 cm. The work is divided in three tasks:

Exercise 1: Calibration using Wound Phantom 1. The radionuclides and activities of the sources are known. Radionuclides for calibration are ^{22}Na , ^{133}Ba , ^{137}Cs and ^{241}Am .

Exercise 2: Measurement using Wound Phantom 2 for identification/quantification of radionuclides. The position of the sources is known, while the activity is unknown. The source is placed at depths 4 mm and 16 mm.

Exercise 3: Measurement using Wound Phantom 3 for identification/quantification of radionuclides. The position of the sources (the depth) and the activity are unknown.

The analysis of the results is in progress.



Fig. 4.6: IRSN Wound Phantoms simulating a puncture wound

The WHO's Radiation Emergency Medical Preparedness and Assistance Network (REMPAN) and EURADOS have established a collaboration to present the state of the art on monitoring, dosimetry and clinical management of internal contamination through wounds, counting with the participation of 21 members from expert groups of 7 countries of Europe and America. The objective is to present current status, a gap analysis and to establish research needs for future developments.

The management of contaminated wounds is complex and case-specific. Rapid decisions are required on wound decontamination, excision of tissues and chelation therapy. ISO 20031:2020[8] standard establishes guidelines on the monitoring and dosimetry for wound contamination. The CDG (Clinical Decision Guidance) quantity was proposed in NCRP Report 161[9] to assist in making treatment decisions for individuals who have had radionuclide intakes.

Monitoring procedures starts with the direct measurements of local activity retained in the wound site. In vivo and in vitro bioassays are performed to determine systemic contamination. Dose assessment requires the application of appropriate biokinetic and dosimetric models.

Injection model may be applied assuming direct uptake into the blood (e.g. in case of highly soluble compounds). NCRP 156 Model was published as biokinetic model for radionuclide contaminated wounds. In case of administration of DTPA, a DTPA Therapy model is required for a proper interpretation of monitoring data. There are available tools to help dosimetrists in dose evaluation in case of contamination through wounds.

4.8 Rapid in vitro Emergency Bioassay Methods for Alpha Emitters (actinides) and ^{90}Sr

A EURADOS study is in progress consisting in the review of methods with sensitivity enough to meet the requirements for emergency bioassay in typical nuclear accidents. This study takes into account the measurements of large number of biological samples, radiochemistry processes, sample turnaround to be shortened and the involvement of in vitro and non-in vitro labs in an emergency network. EURADOS Guidance Report on screening and rapid in vitro emergency bioassay is in preparation considering the topics as follows:

- 1.- Introduction and context
- 2.- Types of Samples
- 3.- Techniques of measurement
- 4.- Screening methods
- 5.- Rapid methods
- 6.- Conclusions

4.9 Application of Biological Dosimetry Methods in Case of Accidental Intakes of Radionuclides

A collaboration has been established between EURADOS WG7 “Internal dosimetry” and WG10 “Retrospective dosimetry” for the evaluation of the usefulness and limitations of biological and EPR (Electron Paramagnetic Resonance) dosimetry methods in cases of internal and mixed internal/external exposures. The main outcome of this study was the publication in 2020 [10] of the “EURADOS review of retrospective dosimetry techniques for internal exposures to ionising radiation and their applications”. The collaboration between WG10 and WG7 in this field continued with a workshop organized during the European Radiation Protection Week in October 2022.

This study allowed to focus on different aspects of the internal dosimetry and the biological dosimetry. Induction of chromosome aberrations has been observed in peripheral blood lymphocytes of subjects internally contaminated, but is it possible to derive a meaningful estimate of radiation dose of the incorporated radionuclides? The challenges are: (1) to identify scenarios of incorporated radionuclides where bio-dosimetry and EPR methods may be applied, (2) to identify calibration methods for generating calibration curves to be used for internal emitters and (3) to develop well defined studies under controlled conditions.

Best case scenarios are radionuclides with rapid absorption into the blood after incorporation and homogeneous distribution of the contaminant inside the body:

^3H (HTO) – case study Lloyd et al. [11],[12]:

Committed Effective Dose $E(50)$ Sv is obtained from concentration of tritium in urine.

Regarding Cytogenesis dosimetry, in vitro calibration curves were obtained mixing tritium with lymphocytes. Good agreement (DC, FISH) with $E(50)$ was found in one contaminated individual.

^{137}Cs – case study Goiania: high influence of external exposures [10]:

There was agreement only in few cases of the cytogenesis dosimetry with the committed internal doses calculated from whole body measurements

^{131}I - Therapy of patients with differentiated thyroid carcinoma (DTC) [10]:

Radiodine treatments were applied here for the ablation of remnant tissues after thyroidectomy. The uptake of iodine in the thyroid is very low and a homogeneous whole body internal irradiation can be assumed.

4.10 Conclusions

EURADOS has been involved in different actions with the aim of improving the measurement and dosimetry of internal emitters after radiological or nuclear emergency situations. Past and ongoing activities have been developed within Working Group 7 and in collaboration with other EURADOS Working Groups and institutions, mainly in a European frame but not only.

The focus was put first on the improvement of age dependent calibration of in vivo monitoring systems and the development of age-dependent dose per content functions, which allows a better reliability on dose assessments in case of public exposures.

An intercomparison of Whole Body Counters using and adult phantom and the intercalibration exercise for wound monitoring resulted in an improvement in in vivo monitoring capabilities of total/partial body counters in Europe.

The review of rapid methods for in vitro emergency bioassay represents a very valuable survey on protocols for indirect measurements of activity in biological samples, which is especially relevant in case of internal exposures to alpha emitters and ^{90}Sr .

Finally, to mention the monitoring, dosimetry and clinical management of intakes of radionuclides via wounds and the use of biological dosimetry methods in case of accidental internal exposures, fields of research open for future actions.

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5 Biological and physical retrospective dosimetry

Elizabeth A. Ainsbury, Jonathan Eakins, UKHSA Centre for Radiation, Chemical and Environmental Hazards, Chilton, Didcot, Oxford, UK; **Sara Della Monaca**, Istituto Superiore di Sanità, Core Facilities, Viale Regina Elena 299, 00161, Rome, Italy; **Juan S. Martinez**, Institut de Radioprotection et de Sûreté Nucléaire (IRSN), PSE-Santé, SERAMED, LRAcc, Fontenay-aux-Roses 92262, France, **Nadica Maltar-Strmečki**, Ruđer Bošković Institute, Bijenička c. 54, 10000 Zagreb, Croatia; **Georgia Terzoudi**, Institute of Nuclear & Radiological Sciences & Technology, Energy & Safety, National Centre for Scientific Research “Demokritos”, Athens, Greece

5.1 Introduction

In a radiation emergency, dosimetry in its various forms is essential to ensure the appropriateness of the response. The 'ideal dosemeter' or suite of dosimetry tools would be radiation specific, with a persistent signal to allow detection some time post-exposure; would be cheap, and easy to calibrate and process quickly; would have low inter-individual variability; and would also give an indication of individual levels of short- or long-term risk. A dosimetry system that fulfils all these criteria does not currently exist. However, over the last 100 years, a large number of complementary techniques have been developed in support of radiation dose estimation.

At the individual level, in occupational settings, radiation workers who might reasonably expect to receive a dose below the annual dose limit during routine operations will generally be wearing a personal dosemeter, such as a thermoluminescence device (Yukihara et al., 2022). However, such dosemeter badges may be worn incorrectly, can be forgotten or lost, and can be exposed to radiation when their wearers are not, or vice versa, and can only be read once, plus the provision and related servicing of such devices can be costly. Another important fact is that generally, passive devices are only read out every few months, depending on the industry and the potential for exposure, resulting in an important time delay in case of a potential overexposure.

For those not wearing dosimetry badges, for example potentially exposed members of the public during an emergency scenario, dose assessment based on clinical signs and symptoms as described in the TMT Handbook (Rojas-Palma et al., 2009) is currently recognised as the gold standard approach. However, such techniques are only really applicable at doses on the order of several Gy as there is a great deal of variation and thus uncertainty associated with such signs and symptoms. They vary between individuals of different ages, with different underlying health conditions, and often the competing causes of symptoms such as vomiting can also be stress-induced (Demidenko et al., 2009).

The techniques of biological and physical retrospective dosimetry can contribute to both triage and estimation of dose and uncertainty, as well as assessment of the exposure scenario, at the individual level (IAEA 2011; Kulka and Wojcik 2017, and references therein). The principles of retrospective dosimetry are very simple: a sample is taken from the suspected exposed individuals or from an object they carried with them during the incident, the radiation-induced signal is measured in that sample, the result is interpreted by comparison to a (usually predefined) calibration curve to estimate dose and uncertainty. Afterwards, the result is used in combination with other available information to assess the need for and/or the most appropriate methods of clinical intervention. The available techniques have been developed over many years, and include the traditional and robust cytogenetic methods such as the dicentric chromosome assay and fluorescence in situ hybridisation (FISH) for detection of stable translocations, as well as the newer methods based on transcriptional

analyses of gene expression, and the physical dosimetry methods based on luminescence or electron paramagnetic resonance.

In this paper, biological and physical retrospective dosimetry are described from the perspective of radiation emergency response. Details of the most common techniques and their relative advantages are discussed, as well as the current challenges within this field. Recent work to address these challenges under the framework of EURADOS Working Group 10 on Retrospective Dosimetry is also presented.

5.2 Methods

5.2.1 Biological dosimetry

Biological dosimetry was initially based on the physical, biochemical, and cellular changes observed in blood lymphocytes in response to radiation exposure (Trowell, 1952; Bender and Gooch, 1962; Bender and Gooch, 1966; Dolphin, 1969; Dolphin et al., 1970; see also IAEA 2011). Dicentric chromosome aberrations, the gold standard in biological dosimetry, are formed when DNA double-strand breaks in two separate chromosomes are mis-repaired, leading to the formation of a chromosome with two centromeres (as well as an acentric fragment composed of the remaining pieces of these chromosomes). Currently, this is the most reliable and well-tested method of biological dosimetry, not least because dicentrics are, with the exception only of a few radiomimetic agents, radiation specific. In principle, a blood sample is collected from the possibly exposed individual; the human peripheral blood lymphocytes can be stimulated and cultured so that aberrant chromosomes, which are the result of the mis-repaired DNA damage caused by radiation, can be directly observed during the metaphase stage of cell division about 48 hours later (missing crucial time in terms of potential provision of supportive care). Lymphocytes are particularly suitable because they are present in circulating blood and it is generally accepted that this mobile population of cells could reflect an average whole-body dose in cases of partial body irradiation. Also, as the half-life of circulating lymphocytes is on the order of three years, the damage can be still seen and evaluated months or sometimes years after an exposure to ionizing radiation. The type of damage analysed is specific to radiation and many data show that comparable yields of aberrations are obtained in vitro and in vivo (Fabry and Lemaire, 1986). By irradiating samples of whole blood in vitro at 37°C, conditions which simulate an exposure to the body, dose-effect (or calibration) curves for most types and qualities of external radiation likely to be encountered in an accident have been produced by the biodosimetry community since the 1960s. Radiation can be more densely or more sparsely ionizing, and the linear energy transfer or LET is a good measure for that. Calibration curves for high LET (or densely-ionizing) radiations tend to exhibit a linear dose dependence with biological effects, and for low LET (or sparsely-ionizing) radiations or protracted exposures tend to be linear-quadratic in nature (Lloyd et al., 1984). Obtaining a sample involves simple venipuncture of approximately 10 ml of blood and results can be produced within 3 to 4 working days of sample receipt. Methods based on the expected Poisson distributed nature of aberrations in lymphocytes exist to detect partial body exposures, as do methods to separate the gamma and neutron components of dose following a criticality accident. In recent years, automation of certain steps of the technique (Garty et al., 2010) has also significantly reduced the amount of person-time needed to carry out this assay. Though the training, equipment, cost, and time required is still significant and presents logistical challenges in the case of a mass exposure event. However, the sensitivity threshold of the cytogenetic method (with a minimum detectable dose on the order of 100 mGy)

and above all, the length of time needed to culture and analyse the aberrations in cells, mean that this assay is chiefly used in support of situations where other dosimetry methods are not available.

Micronuclei are small pieces of nuclear material which have become excluded from the main daughter nuclei during the process of mitosis (Fenech and Morley, 1985). Their frequency of occurrence in circulating lymphocytes can be directly related to the magnitude of exposure to radiation or to a number of other mutagens. Because of their ease of detection and relatively high frequency of induction by radiation, micronuclei have shown potential as a supplement to chromosome aberration analysis as a practical method of biological dosimetry (Fenech, 2009). It is anticipated that, in the event of a major nuclear accident, blood samples could be processed and examined for micronucleus frequency much more rapidly than for conventional chromosome aberrations. This is because the microscope images of micronuclei are far simpler to analyze than chromosome spreads and so the microscopy acquisition and analysis steps are quicker and require less skilled staff. However, blood lymphocytes do require 72 hrs in culture for complete expression of micronuclei. This is 24 hrs more than for the conventional chromosome aberration test and during this time some preliminary chromosome scoring would be done to select those individuals most at risk. In addition, micronuclei are not radiation specific, and as such, there is a large degree of variation between data from different populations, thus population specific background data are strongly recommended for each assessment (Thierens et al., 1996).

In situations where exposure occurred many months or years prior to blood sampling, the proportion of circulating lymphocytes exposed to radiation may have reduced, making dose estimation based on dicentric analysis no longer possible. In such cases, developments in molecular biology have led to the manufacture of specific sequences of DNA which may be labelled by immunological techniques with fluorescence dyes. These DNA sequences, or probes, can be hybridised onto metaphase chromosomes on microscope slides and the probes, with their fluorescence tags, will attach to the complementary sequences in the chromosomes (known as Fluorescence In Situ Hybridisation, FISH). The cells may then be viewed in a fluorescence microscope. Cocktails of such probes may be mixed to paint several pairs of chromosomes in the cell as well as centromeres, thereby facilitating the detection of 'translocations' (i.e., 'swapping') of chromosomal regions between chromosomes. In contrast to dicentrics, these translocations are stable in the sense that they will permit the cell to divide successfully. Since this type of damage is not lost from the lymphocyte pool by the normal turnover of cells, FISH is one of the few techniques, if not the only one that can be used as a retrospective biological dosimeter for investigating subjects many years after exposure (Lucas et al., 1992; Rodriguez et al., 2004). This application has been shown to be effective and is in use in a number of laboratories, but the expense of the probes as well as its lower sensitivity when compared to the dicentric chromosome assay (DCA), limits the number of cases for which FISH is routinely performed. Another important point to consider is that the basal translocation rate in humans increases with age as chromosomal translocations accumulate throughout life. This creates the need to have a case-by-case control population for any given analysis.

Using the well-established cytogenetic methodologies mentioned above, the need for G0 peripheral blood lymphocyte stimulation delays dose estimation, which is a high priority in radiation emergency medicine as time to intervention is crucial in such cases. To overcome this shortcoming, the Premature Chromosome Condensation (PCC) that enables the analysis of radiation-induced chromosomal aberrations in non-stimulated G0-peripheral blood lymphocytes for biodosimetry has been introduced (Pantelias and Maillie, 1984). This method enables rapid quantification of an exposure by means of the analysis of fragments or rings (Lamadrid Boada et al, 2013) and can be

induced either by fusion of human lymphocytes with mitotic cells (fusion-mediated PCC) using Polyethylene glycol (PEG) as the fusogen, or with the use of specific chemicals such as okadaic acid or calyculin A (chemical-induced PCC) (Kanda et al., 1999). Unlike fusion mediated PCC, chemically induced PCC method requires stimulation of lymphocytes for one cell division. To quantify radiation-induced chromosomal aberrations in G0-phase lymphocytes using the fusion PCC-assay, the excess PCC fragments (more than 46 elements) are scored. The dose is then estimated based on the dose response calibration curves that have a linear shape and the residual yield of excess fragments depends on the time elapsed for repair between the irradiation and the cell fusion. Furthermore, when the PCC-assay is combined with the fluorescence in situ hybridization (FISH) technique, inter- and intra-chromosomal rearrangements can be analysed directly in G0 lymphocytes for radiation biodosimetry purposes and retrospective assessment of radiation-induced effects (Karachristou et al., 2015). In addition, a quick, automatable, and minimally invasive micro-PCC assay was proposed for rapid individualized risk assessments in large-scale radiological emergencies (Pantelias and Terzoudi, 2018). The major advantages of PCC over the other methods of biodosimetry are rapid dose estimation and detection threshold to much lower doses. Also, PCC is dose-dependent, with the nature of the response varying between publications, and the technique is effective for whole-body irradiations, high doses, and partial-body exposures. There is also evidence that PCC is more sensitive than the dicentric or MN assays in terms of detection of the proportion of undamaged cells (Terzoudi et al., 2017). However, this assay requires the availability of either fresh or frozen mitotic cells, expertise in cell fusion procedures and analysis of lymphocyte prematurely condensed chromosomes. Due to these limitations, the test is still not widespread and thus, it could be used in a complementary manner to DCA.

One of the earliest stages in the cellular response to radiation-induced DNA double-strand breaks is the phosphorylation of the histone variant H2AX (Rogakou et al., 1998). By using a fluorescent antibody specific for the phosphorylated form of H2AX (gamma-H2AX) and immunofluorescence microscopy, discrete nuclear foci at sites of double-strand breaks can be visualized. The formation and loss of gamma-H2AX foci after X-ray exposure has been measured following exposure to radiation doses as low as 1 mGy, and focus yields have been shown to increase linearly with dose. In addition, the gamma-H2AX assay has been successfully used to detect DNA damage and repair as well as to estimate radiation doses received by peripheral blood leukocytes during diagnostic computed tomography scans and interventional cardiology procedures (Rothkamm et al., 2007; Habibi et al., 2021). These studies demonstrate that gamma-H2AX can serve as a sensitive and quantitative protein biomarker of human low-level radiation exposure. Although the gamma-H2AX assay encompasses some of the features that one would expect from an ideal biodosimetric assay—specifically, rapid analysis within a day, high sensitivity to doses of a few milligrays, linear dose response across a broad dose range, minimally invasive sample collection, and ability to reveal partial body exposures—it has limitations that prevent its general applicability to all types of biodosimetric tasks (Barnard et al., 2013). In particular, the fast signal loss due to innate repair of most DNA damage following exposure restricts its use to the first ~24 hours following exposure and makes it difficult to use this approach for dose determinations after unplanned accidental acute exposures and protracted low-dose exposures. Also, manual scoring is highly skilled (with training to full competency taking several years) and also tedious and subjective, and automated scoring systems for data collection have not yet proved as sensitive. Furthermore, the level of inter-individual variation of baseline yields of foci is highly variable.

5.2.2 Physical dosimetry

In recent years, a number of 'physical retrospective' or 'fortuitous' dosimetry techniques have been developed for use in radiation emergency response. Fortuitous dosimeters are objects that individuals happen to be carrying, or are in close proximity to during the suspected exposure, and are based on materials that happen to have appropriate dosimetric properties. The physical mechanism of retrospective dosimetry is similar to that for conventional occupational dosimeters: energy from the radiation is stored, somehow, in the material during exposure, and that energy is then released at some later time to provide a readable signal that is proportional to the dose deposited. A variety of possible candidate materials and objects have been investigated for dosimetric properties, using different physical techniques. The most common techniques considered include optically stimulated and thermally stimulated luminescence (OSL and TL), and electron spin or paramagnetic resonance (ESR and EPR). A number of reviews, comparisons and considerations of the various possibilities have been published (Trompier et al., 2017; McKeever et al, 2021; Van Hoey et al, 2021; Yukihiro et al., 2022, Fattibene et al, 2023; see also ICRU, 2019). Furthermore, recent studies have shown that a wide range of fortuitous materials that can be found on or around the exposed individuals have shown good dosimetric properties, e.g. salt and salty crackers (Maltar-Strmečki et al, 2021, Alghamdi et al., 2022), earbuds and wireless headphones (Discher et al, 2023), common pharmaceuticals and dietary supplements (Ekendahl et al, 2022), and dental porcelain (Sfampa et al, 2023).

Optically or thermally stimulated luminescence techniques rely on external stimulation followed by deexcitation of the trapped radiation signal, which is stored as electron-hole pairs created in crystalline materials during radiation exposure. Recombination of the pair then leads to the emission of light: this luminescence signal is proportional to the amount of recombined trapped charges, and is thus usually directly proportional to the dose the material has received. Luminescence analyses of the dosimetric properties of different types of electronic components are quite popular and have been reported by a number of different authors (Inrig et al, 2008; Ademola et al, 2017; and summarised in Trompier et al, 2017). Another consolidated application is the one on mineral glass, especially display glass of smartphones (Discher et al, 2013; Kim H et al, 2019). New methods, still poorly investigated but often very promising at least from the technical perspective, involve the study of luminescence properties of new materials as tobacco (Ademola et al, 2017), pharmaceuticals, and food supplements (Ekendahl et al, 2022), salty snacks (Bernhardsson et al, 2009; Maltar-Strmečki et al, 2021), fabrics (Bossin et al, 2020; Pakari et al., 2022), chip cards (Kim H et al, 2020; 2024), circuit boards (Sholom et al, 2016) and headphones (Discher et al, 2023). The related, but different, techniques of photo-transferred luminescence (PTTL), radiophoto luminescence (RPL), and infra-red (IRSL) and violet-stimulated luminescence (VSL) have also been explored by some authors in attempts to improve dosimetry (Chandler et al, 2019; Yasuda et al, 2023; Discher et al, 2021).

ESR / EPR dosimetry is based on quantification by spectroscopy of the changes in the concentrations of free radicals formed in materials following exposure to ionizing radiation (including the trapped electron and hole pairs used in TL and OSL). The induced free radicals are, again, approximately proportional to the dose absorbed. However, use of a calibration curve is recommended (Fattibene and Callens, 2010). EPR has been tested on a variety of different materials but is most operationalised on mineral glass of the type found in watch faces and mobile phone screens (Trompier et al, 2013; Marciniak et al., 2022 and references therein; Port et al. 2023), as well as using low-cost alanine (Gougelet et al, 2010).). For small groups, *in vivo* and *ex vivo* EPR dosimetry based on available tooth,

nail and/or bone samples has been shown to be most useful (Beinke et al., 2022, Swartz and Flood, 2023), especially for assessing long-term risk. These techniques also have limitations including those associated with fading sample preparation time.

The gold standard for EPR dosimetry is tooth enamel, almost exclusively used in long-term epidemiological studies because of the invasiveness of sample collection (Fattibene and Callens, 2017). Development of new techniques like open cavities for *in vivo* measurements (Pisa et al., 2019) and the spread of Q band resonators, which allow the analysis of tiny fragments (e.g., biopsies) (Romanyukha et al, 2014) makes in principle possible the application of the method in small scale emergency situations.

Many studies were conducted in the last years also on the EPR signals of irradiated toe- and fingernails, particularly focusing on finding ways to separate the radiation-induced signal from the endogenous (background) one (Tkatchenko et al., 2022). Radiation-induced radicals in sugars were also investigated as markers of an accidental exposure. Sugars are, in fact, nearly ubiquitous: they are contained in pills as well as in food supplements (Sagstuen et al., 1983).

The physical dosimetry techniques present similar technical challenges to the biological techniques, including the need for appropriate laboratory equipment and deep subject matter expertise, sample- and often laboratory-specific calibration techniques, transport logistics, and sample preparation to remove contaminants that might influence the recorded signals. Signal fading within the samples might also be an issue, requiring the need for time-dependent calibration curves. However, there are some additional drawbacks that are specific to fortuitous dosimeters: for example, individuals may be reluctant to hand over valuable items such as mobile phones for destructive testing, different types of such dosimeters might have very different dosimetric properties, and not all individuals may be in possession of them during the exposure. There are also, though, a number of advantages - for example, the radiation-specific nature of the signal, and specifically for EPR, good signal stability, non-destructive measurements, and the possibility of repeating the measurement, as well as complementarity with the biological assays to facilitate elucidation of the partial body nature of exposures. Minimum detectable doses vary widely according to the techniques employed and the properties of the materials used for dosimetry, including fading, but can be on the order of a few mGy.

One other limitation of physical dosimeters is that they can only ever record the doses that they receive, whereas it is the concurrent dose to the individual that is of concern. Factors that could cause these to diverge include the location of the dosimeter relative to the individual, the orientation of the dosimeter and individual relative to the source, and the tissue-inequivalence of the dosimeter material. Computational techniques are therefore highly complementary to physical retrospective dosimetry. These include Monte Carlo modelling of the exposures to the individual and dosimeters, including the possibility that several might be available and distributed around the body during the irradiation, with a view to harmonize and mitigate their respective results (Eakins et al, 2015; Kim M et al, 2019). Such approaches are most appropriate in small-scale incidents (ICRU, 2019), where the exposure conditions are more likely to be known with sufficient reliability, and permit dose reconstruction analyses to be performed following partial- or full-body irradiation (Clairand et al, 2008; Huet et al, 2008; Discher et al, 2021a; Entine et al, 2022).

5.3 Specific considerations for radiation emergency response

The specific considerations in relation to the use of biological and physical retrospective dosimetry in emergency scenarios have been well addressed by a number of authors in recent years. These include the fact that a high level of expertise is required to perform the techniques in specialised laboratories, for which equipment and staffing is expensive. This, coupled with the fact that the likelihood of a genuinely large-scale radiation accident or incident requiring retrospective dosimetry is seen to be small (despite the consequences being large), means that teams tend to be small and limited to highly-resourced countries. The ‘surge capacity’ of these teams is therefore also limited (Kulka and Wojcik, 2017, and references therein). In recent years several solutions to this problem have been explored, for example in biodosimetry, to score a small number of cells first in ‘triage’ or ‘rapid response’ mode, to identify the most highly exposed individuals for more detailed follow up (ISO 20243), as well as the development of standardisation and automation across the various different techniques for which a large amount of progress has been made (e.g. Wilkins and Beaton-Green, 2023). The largest and perhaps objectively most successful advancements in this field, though, have come from the focus on networking between laboratories at the national, regional, and international levels (Wilkins et al., 2015; Wilkins et al., 2016; Ainsbury et al., 2017, Kurihara et al., 2020, Port et al., 2023).

Similar conclusions were recently drawn in a critical review of the possible future perspectives of research in physical fortuitous dosimetry techniques (Fattibene et al., 2023). Authors hope for an effort, by the labs, to create a network promoting the standardisation of the protocols (both in sample managing and preparation and in measurements) and a common activity for a deeper understanding of the physical mechanisms underlying signal formation.

5.4 Recent contributions of EURADOS Working Group 10: Retrospective Dosimetry

EURADOS Working Group (WG) 10 on Retrospective Dosimetry is one of the world leaders in terms of development of techniques for biological and physical retrospective dosimetry and their practical application for routine and emergency response. The aims of WG 10 are to facilitate networking in this area, with a particular focus on the development and evaluation of new methods, including: characterisation of uncertainties; establishment of multi-assay approaches to dose assessment; consideration of approaches for partial body and internal exposures; and dissemination of knowledge about retrospective dosimetry among authorities, scientific institutions, and stakeholders.

The WG 10 2019 field test and interlaboratory comparison, for example, was organised to address multiple objectives, including: testing use of blood-based techniques and OSL and TL on mobile phones to assess absorbed doses, and to inform the circumstances of exposure between heterogeneous and homogeneous situations; calculation of organ absorbed doses from dose measurements in diverse physical materials; Monte Carlo simulations and calculation of organ absorbed doses in phantoms based on the actual exposure geometries; as well as reconstruction of the dose and exposure conditions (homogeneous exposure, partial shielding, lateral (left or right)) by combining the large number of different biological and physical dosimetry techniques and materials tested as part of this exercise (Waldner et al., 2021). Publications thus far have described the dicentric assay (Endesfelder et al., 2021) and gene expression (Abend et al., 2021), and further publications are in preparation presenting the results of the mobile phone dosimetry, the other exposed materials, and the Monte Carlo simulations.

Another ongoing interlaboratory comparison within EURADOS WG 10 concerns EPR on tooth enamel. The aim of the ILC was to re-check the performances for laboratories that took part in the past ILCs together with an involvement of new laboratories that during the last years have developed an EPR method whose performances should be tested for the first time. The analysis of the ILC results is currently underway. Work on the development of uncertainty analysis techniques is also underway (Ainsbury et al., 2018), as is work on use of biological dosimetry techniques for internal dosimetry (Giussani et al., 2021).

5.5 Conclusions

When small or large numbers of individuals are suspected of being exposed to ionising radiation, biological and physical retrospective dosimetry has been consistently demonstrated to be a hugely valuable tool in support of dose assessment. In certain cases, it is also possible for the tools to be combined to help elucidate the circumstances of exposure, for example whether the exposure was partial body or heterogeneous in nature. Development in biological dosimetry started in the 1960s (IAEA, 2011), and today we have a robust set of tools and techniques of biological and physical retrospective dosimetry in support of routine and emergency dosimetry. However, despite the relatively small number of laboratories worldwide, this is a very active field of research and development, including within EURADOS WG 10.

In addition to the key challenges common to all approaches, such as improving the technical capabilities of specific dosimetry techniques for large-scale emergency response, networking in the development and validation of techniques is absolutely crucial to ensuring that we can respond with the state-of-the-art retrospective dose estimation. EURADOS WG 10, RENEB, and other networks therefore remain essential – this has been, and must remain, an international collaborative effort going forward.

5.6 References

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6 Environmental monitoring and the use of unmanned aerial systems for radiological surveillance

Arturo Vargas Institute of Energy Technologies (INTE)—Technical University of Catalonia, 08028 Barcelona

6.1 Introduction

The protection of first responders and public in general against radioactive contaminations caused by radiological accidents, including terrorist attacks and in war scenarios such as the invasion of Ukraine, where radiological facilities and nuclear power plants were affected by the attacks of Russian troops, is of major importance.

Following a nuclear or radiological event, knowledge of how contamination spreads and the area it covers, is essential for decision makers who must determine the most appropriate response. In the immediate vicinity of a nuclear or radiological source of contamination, as well as in case of large-area ground contaminations, and before first responders enter in the contaminated area, remote detection of alpha and gamma radioactivity by using unmanned aerial systems (UASs) is an excellent solution to protect operators and early responders against contaminations and irradiation.

Since the released radionuclides in an emergency scenario are usually unknown and there is no optimal detector-UAS configuration, the end-user should determine in advance the most probable expected radiological situation in terms of type of radionuclides and dose rates, in order to select the most convenient configuration.

In such context, spectrometric gamma detectors that can identify radionuclides are recommended to be mounted on the UAS. “Classical” dose rate monitors such as Geiger-Muller (GM) tubes can be used for a fast screening to evaluate the dose rate map for further analysis with spectrometric detectors. In the European project “Preparedness” (<http://www.preparedness-empir.eu/>) spectrometric detectors were adapted and mounted on selected UASs based on their flight capabilities. The airborne detectors mounted on the UAVs were tested and calibrated in measurement campaigns.

In case of a radiological scenario involving dispersion of alpha emitting radionuclides in the environment, the situation is more complex because currently a commercial detection system to measure remotely alpha particles is not available. In case of such an emergency scenario, the only option is to evacuate the population from the affected areas and then run diagnostics by hand, thus exposing the emergency teams to considerable risk. Even then, the results of emergency field applications are notoriously ambiguous, time consuming and tedious due to the centimetre range of the alpha particles in air. In the European project 19ENV02 “RemoteALPHA” (<https://remotealpha.drmr.nipne.ro/>) a remote alpha airborne monitor to be mounted in UAS has been developed and tested in experimental campaigns.

In the next sections, a summary of the developments carried out and the experience gained in both projects is described. In addition, future activities in the field of remote radioactive measurements by using UASs will be pointed out.

6.2 Remote detection of gamma emitting radionuclides

6.2.1 Unmanned Aerial System (UAS) and gamma spectrometric detector selection

The selection of the most convenient UAS and the gamma spectrometric detector is one of the most important aspects for a fast and adequate response to a given emergency scenario. For instance, there is no reason to send a huge aircraft to monitor a city street, or deploy the small drone with very limited flight time to a remote area. The detection system size and efficiency determine a range of UAS options that are able to handle such a payload. Consequently, the size of the aircraft determines the flight time and operational/communication range for a device. The choice of a specific configuration depends on the radiological situation and the assessment of the overall scenario. The selection of the UAS-detector configuration should fit the most common scenarios that are previously forecasted by the end-user. Table 1 provides an approximate classification of the UAS by carrier size. The detector materials classified in each category represent the most commonly used in commercial devices but should not be considered as a closed classification. Table 2 gives an attempt to classify the Unmanned Aerial Monitoring System (UAMS) selection depending on a measurement task following some radiological emergency scenarios (Schneider et al, 2015).

Table 6.1: The UAMS carrier size classification and features (Luchkov et al. 2021).

			
	Small	Medium	Big
Payload	< 1 kg	< 6 kg	< 50 kg
Detector	CZT, CsI	Nal, LaBr ₃ , CeBr ₃	HPGe, Nal of high volume
Range	< 2 km	< 5 km	< 20 km
Flight time	< 15 min	< 30 min	< 90 min
Price	≈ 1 k€	≈ 10 k€	≈ 250 k€
Licensing	Easy*	Normal**	Complicated***

* UAVs below 2 kg TOW (Take Off Weight, refers to the total weight of the UAS including the aircraft itself, payload (such as cameras or sensors), fuel, and any other necessary equipment at the moment it takes off from the ground) do not require any licensing

** Under 25 kg TOW, a drone pilot license is required

*** For aircrafts with TOW > 25 kg the full pilot license is required

Table 6.2: The proposed UAMS category choice for a given radiological emergency scenario.

Scenario	Scale	Area	Requirements	UAMS category
Terrorism (dirty bomb)	Small	Populated (stadium, market, festival, etc.)	Quiet and small system to prevent panic, hot spot detection	Small
Illicit trafficking	Small	Not populated (harbour, customs, railway, etc.)	Fast and efficient source localization	Small or medium
Radiation source transportation accident	Small	Localized (city, road, etc.)	Radioactive spill boundary detection; panic prevention if public involved	Small
	Medium	Dispersed (water bodies, air transfer)	Radioactive substance dispersion tracking (e.g., follow river path)	Medium or big
Nuclear accident	Small	NPP territory	Fast assessment, localization and decontamination	Small or medium
	Medium	NPP vicinity (closest town, countryside)	Fast assessment, possible temporary evacuation zones detection	Medium or big
	Big	NPP exclusion zone (30 km radius)	Evacuation stage: Cover tens of km ² as fast as possible Assessment stage: Detailed mapping of exclusion zone	Big, then medium

It should be pointed out that in the “Preparedness” project a medium-sized UAS was mounted with an angular sensitivity detector, i.e., a real-time gamma-ray “source-locator”. While spectrometers installed on UASs need to scan the whole affected area to find radioactive sources, the source-locator detector is able to identify the direction of a hotspot of radioactivity on ground in real-time and optimizes the scanning process by adjusting the flight path towards the direction of the source position as will be described in a further section.

6.2.2 Complementary instruments and equipment to mount on the UAS

In addition to the radiological detector mounted on the UAS, it is essential to install complementary instruments and equipment to gather the necessary information for assessing the level of

radiological contamination in the environment. A real-time 3D geolocation system, providing longitude, latitude, and height above ground level for the UAS, is necessary to analyse the acquired radiological data. To stabilize the acquired spectra in terms of energy, it is common to use ambient and detector temperatures, making the installation of such probes worthwhile. Batteries play a crucial role in ensuring sufficient endurance and a quick system for changing and recharging them is essential. Installing a fast and reliable communication system with the ground station is also necessary. Finally, it is highly recommended to equip the UAS with a 3D visualization system to monitor the actual status of the geometry of buildings and terrain as a result of a nuclear or radiological accident.

In the next sub-sections, brief descriptions of such complementary instruments and equipment are provided.

6.2.2.1 UAS 3D geolocation

The most common method is either to install or use the Global Positioning System (GPS) system already installed in the UAS by the UAS manufacturer. This second solution makes the radiological measurement system dependent on the UAS and, therefore, its use is not recommended. Any change in the UAS manufacturer would necessitate modifications to the radiological measurement system. An important concern of the GPS geolocation is the requirement of a robust satellite coverage. In a degraded scenario, which is quite probable in an emergency situation, the system becomes unreliable. Even with a relatively good satellite communication, the horizontal position uncertainty is approximately 1 meter to 2 meters, and considerably more for the vertical measurement. Consequently, an alternative system for measuring the altitude above ground level (AGL) is commonly installed on the UAS.

There are 3 main altitude indicators that provide information regarding the flight height: GPS altitude, barometric sensor and laser rangefinder. The user should be aware that GPS and barometer do not give information about the local terrain changes, which can be quite significant along the course of the flight. Figure 6.1 demonstrates the terrain profile for a single flight over the former uranium mine in Seelingstädt, Germany: for a nominal altitude of 25 m AGL, the actual height ranges from 7 m to 70 m.

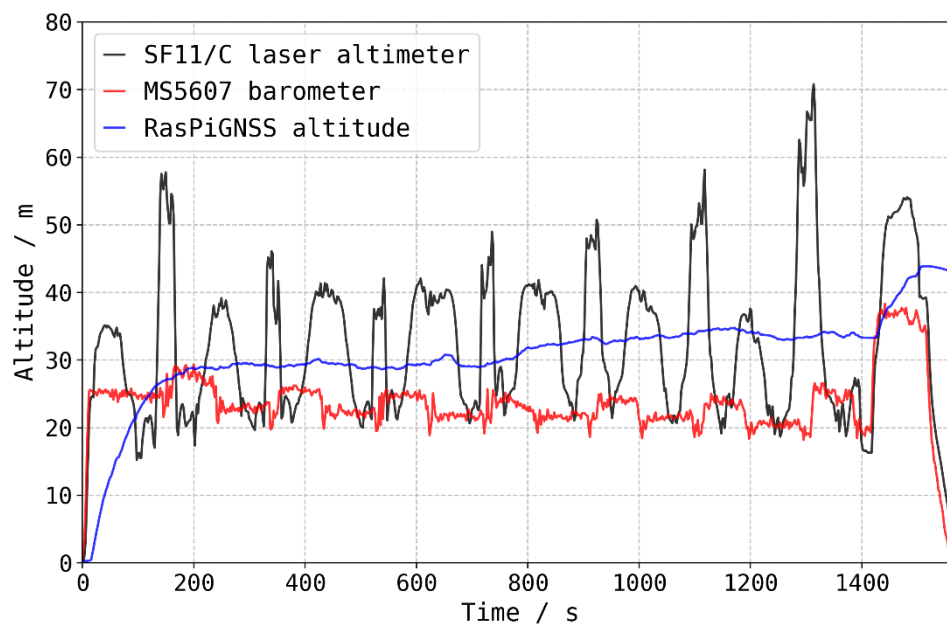


Fig. 6.1: The difference between the altitude measurements provided by the laser altimeter (black), barometer (red) and GNSS (Global Navigation Satellite System) positioning device (blue) for a span of a single flight (take-off to landing).

An alternative strategy that avoids the mentioned problems of GPS is to implement an on-board Real Time Kinematic (RTK) positioning system on the UAS. RTK positioning works by using a base station located at a known position and a mobile receiver located at an unknown position. The base station receives signals from GPS or other navigation satellites and uses these signals to calculate its position. The mobile receiver also receives signals from the same satellites and uses them to calculate its position. This strategy involves installing, in addition to the on-board computer, an RTK board and a communication system for RTK, through which RTK corrections will be sent. This enables precise centimetre-level positioning in both vertical and horizontal profiles.

6.2.2.2 Energy stabilization of the acquired spectrum

One of the most important issues of concern is the detector energy calibration stabilization due to temperature variations during the flight. The fact that the energy calibration changes during the flight complicates the spectra analysis and, therefore, can lead to mistakes in the radionuclide identification and dose rate calculations. The problem of temperature variation is especially prominent for scintillators with photo multiplier tubes (PMT), since both parts (the scintillating crystal and the PMT) are temperature sensitive. Moreover, the speed at which the crystal and the PMT reach thermal equilibrium with the ambient air are different. This makes the correction due to the temperature drift of energy calibration quite a difficult task. Methods to correct the temperature drift should be provided by the manufacturer or implemented by the end-user.

One method to compensate the thermal drift consists in adjusting the energy calibration according to the temperature measured by an external sensor or by internal sensors of the detection material. The user should take care on the thermal inertia of the measurement system and, therefore, it is advisable to carry out a calibration in a climatic chamber. Following, an example of a calibration method, developed in the Preparedness project, by using an external temperature sensor is

described. The modified energy calibration function is calculated as a second order polynomial (Eq. 1) with the slope as a quadratic function of external temperature (Eq. 2), namely:

$$E(k, T) = p k^2 + a(T) k + b \quad (1)$$

$$a(T) = a_2 T^2 + a_1 T + a_0 \quad (2)$$

where k is the channel index and T is the temperature in degrees Celsius. Figure 6.2 demonstrates the raw and stabilized spectra carried out in a climate chamber test, where the constants p , b , a_0 , a_1 and a_2 have been determined and published in (Luchkov et al. 2021).

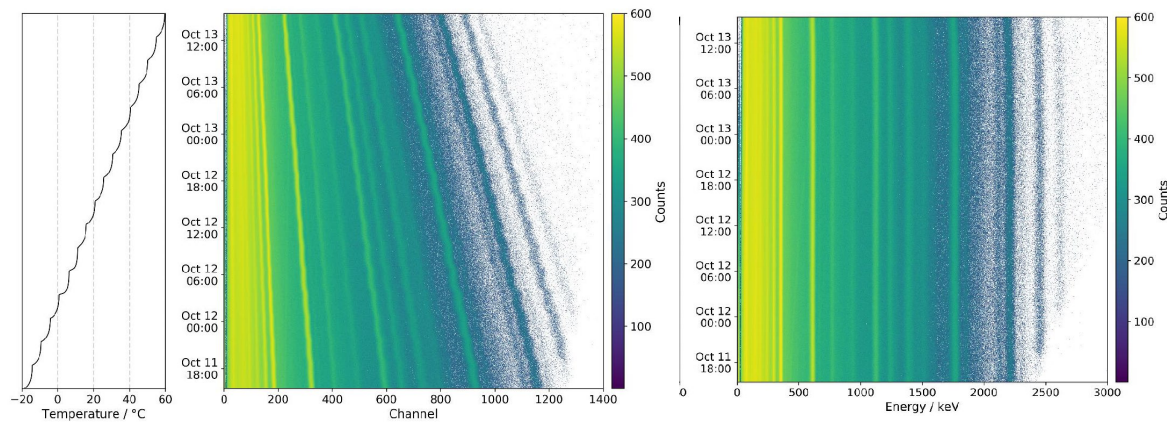


Fig. 6.2: Climate chamber CeBr₃ detector test with ²²⁶Ra source: raw channel spectra (left) and energy stabilized spectra (right).

The user should also be aware that compensation for temperature drift is also commonly achieved by utilizing a permanently fixed peak. For example, peaks attributed to internal ¹³⁸La contamination in the LaBr₃ crystal can be employed to adjust the electronic gain or recalibrate the energy curve (Vargas et al., 2018). Another option is to use natural peaks, such as the 1460 keV gamma line of ⁴⁰K and the 609 keV of ²¹⁴Pb in the acquired spectra. However, due to the short integration time of a few seconds in the acquired spectra, multiple spectra should be combined to accumulate sufficient statistical counts for calculating the coefficients of the energy calibration curve.

6.2.2.3 Batteries

One important issue of the concern is the flight endurance. Based on current UAS technology, a flight time of at least 15 minutes is recommended when investigating a certain area. The end-user should consider the characteristics of the batteries, i.e, their capacity, weight and the influence of meteorological conditions such as temperature and humidity. Furthermore, due to the application of UAS in emergency situations, the batteries have to be easily replaced with a minimum loss of time between flights and fast recharged in-situ. Therefore, the time needed to recharge the batteries and the number of batteries needed for the surveillance, should also be considered when planning the flight in the emergency scenario.

6.2.2.4 Ground communication system

The data transmission between the aircraft and the ground station should be reliable and fast enough to get the correct information in real time.

Currently the 4G/5G seems to be the easiest way for long range communications between the UAS and the ground station. However, the user should take into consideration that the flight surveillance area could have bad 4G/5G coverage, so the use of a point to point radiofrequency communication

should also be considered as an alternative option. 4G/5G can cover huge distances and is an easy method to be implemented. A point to point can cover distances that depends on the emitted radio frequency and power transmission, usually commercial devices can cover few km, which are far enough for a visual flight. However, the velocity of the data transmission is much smaller than for internet communication.

6.2.2.5 Visualization of the terrain scenario.

In the measurement campaign conducted in the former uranium mines at Seelingstädt as part of the "Preparedness" project, the results highlighted the complexity of data analysis in the event of a hypothetical real emergency situation. On one hand, while analysing the measurement campaign data, the difficulty of this task was observed, primarily due to the heterogeneity in the concentration of radionuclides, as reflected in the values of the ambient dose equivalent rate $H^*(10)$ that were measured (Luchkov et al., 2024). Additionally, the uneven terrain is constantly changing due to movements of soil with elevated natural radioactivity which is then covered with shielding soil layers preventing the use of topography databases. A traditional video camera does not provide the necessary information on the 3D terrain model that will be particularly relevant in events where the accident may have altered the terrain relief or the location of objects, such as in a radiological incident involving explosion or fire.

In this context, a potential improvement currently being developed involves creating 3D terrain models in quasi real-time (using photogrammetry or LiDAR) overlaid onto the radiological map model, as illustrated in the example in Figure 6.3 (Matos, 2022). Similarly, the Nuclear Safety Council in Spain has initiated the "gamma-dron" Project (2024-2026) for overlaying radiological data on the 3D terrain model obtained with LiDAR.

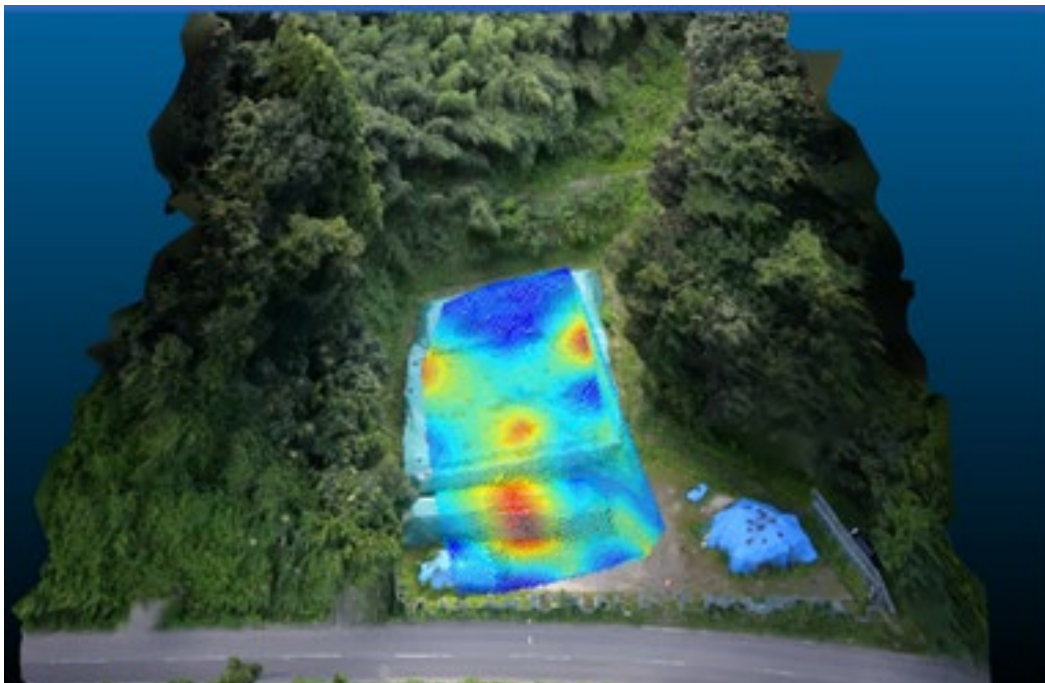


Fig. 6.3: Example of 3D photogrammetry combined with the radiological map (Matos, 2022).

The radiological variables calculated from the acquired gamma spectrum are, basically, the ambient dose equivalent rate, $H^*(10)$ at 1 meter height, and the activity of radionuclides in the terrain. In

addition, the identification and location of artificial sources are also relevant results that can be directly obtained from the analysis of the acquired spectra.

In general, if the spatial distribution of radionuclides has been determined, the other variables can be calculated from them. However, this calculation requires such high computational time, that it is worth to calculate them in the post-processing analysis but not for real time visualization of radiological maps. Therefore, is the most practical is to calculate $H^*(10)$ rates and identify the presence of artificial radionuclides using rapid and robust methods. One of the "Preparedness" outcome recommendations is to present the $H^*(10)$ heat map together with the method for identifying artificial radionuclides known as "Man Made Gross Count" (*MMGC*). The following are summaries of the methodologies that can be used to estimate the mentioned variables: i) Ambient dose equivalent, $H^*(10)$; ii) Identification of the presence of artificial radionuclides (*MMGC*) and iii) Spatial distribution of radionuclide activity.

Ambient dose equivalent, $H^*(10)$

There are different methods to calculate the $H^*(10)$ rates from acquired gamma spectra which are described elsewhere in the literature, such as in Dombrowski (2014). In the investigation in the "Preparedness" project, the conversion coefficients method was selected to be applied to all the developed detectors mainly because it is reliable, robust and easy to implement in the software (Vargas et al, 2021). In the conversion coefficients method, the spectrum is divided into several energy regions and the corresponding conversion coefficients are applied to the aggregate count rates detected in each energy band to convert the spectrum into $H^*(10)$ rates. The conversion coefficients can be calculated either by Monte Carlo (MC) simulations or by experimental irradiations in reference facilities. Then, the absorbed dose rates, D rates, are calculated with the following equation:

$$D = \sum_{i=1}^j w_i n_i \quad (3)$$

where w_i is the conversion coefficient for energy band i ($\text{nGy h}^{-1}/\text{s}^{-1}$), n_i is the count rate in s^{-1} in the band i , and j is the number of energy bands. The absorbed dose coefficients calculated by Monte Carlo simulations can be fitted in different energies energy ranges as it is shown in Figure 6.4 for a NaI 50 mm x 50 mm detector. Then, the $H^*(10)$ rate is calculated by multiplying the absorbed dose rate by the following conversion function that allows going from absorbed dose to ambient dose equivalent for each energy bin (Al Kanti et al., 2020)

$$\frac{H^*(10)}{D} = \frac{x}{ax^2 + ax + c} + d \arctan(gx) \quad (4)$$

where $x = \ln(E/E_0)$, E is the average energy of the energy bin and $E_0 = 9.85 \text{ keV}$.

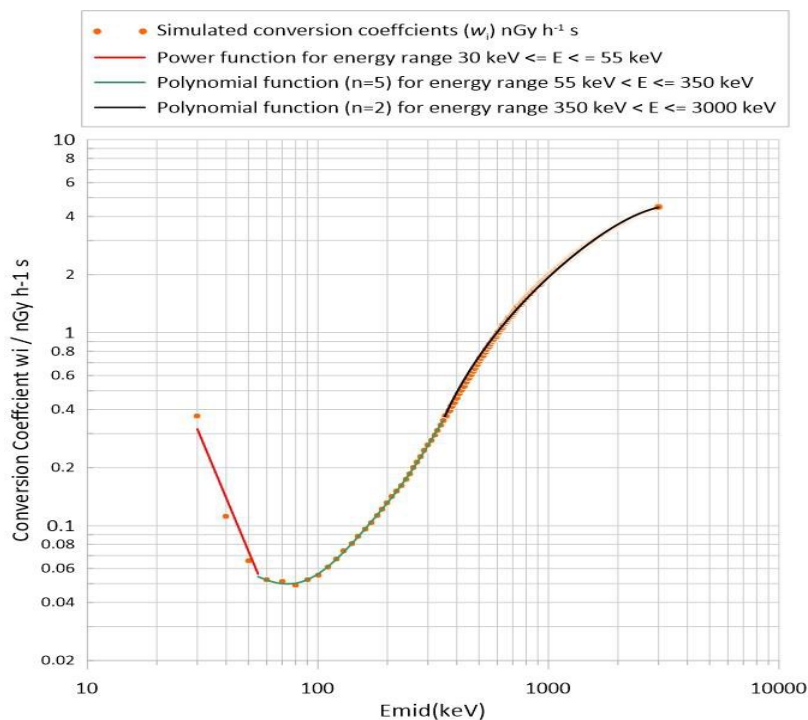


Fig. 6.4: Conversion coefficients calculated by Monte Carlo simulations for a NaI crystal of 50 mm x 50 mm and adjusted with the equations for different energy ranges.

To project the measured dose rate to the reference height (usually 1 m height AGL), the altitude profiles of the background and the artificial (net) dose rate need to be considered. For the case of homogeneous area contamination (natural radioactivity), the dose rate behaviour with altitude follows the exponential integral profile and for a point source it follows the inverse of the square distance as it is illustrated in Figure 6.5 obtained from the measurement carried out by a CeBr_3 detector with a ^{137}Cs point source of the activity of about 345 MBq. Detailed results of the calculated equations can be found in Vargas et al. 2021.

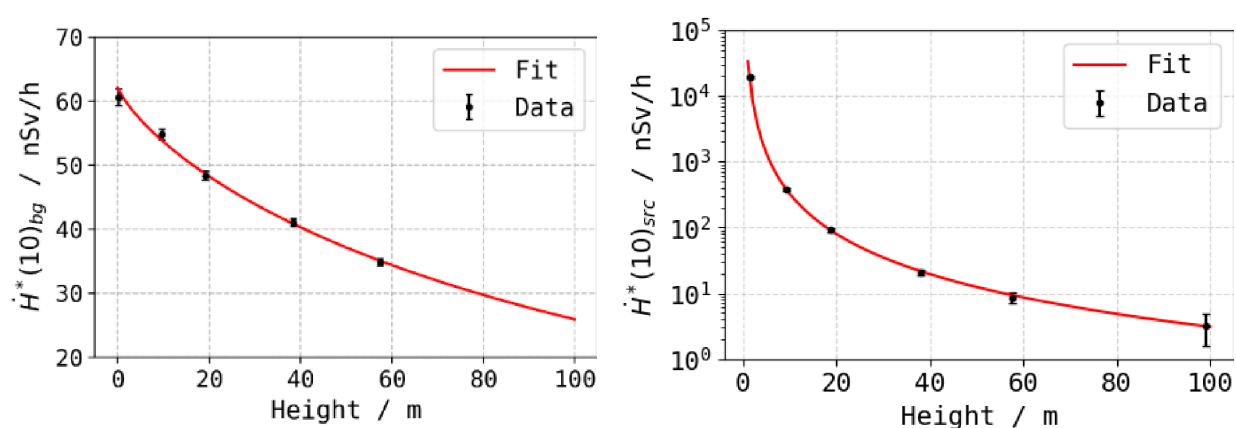


Fig. 6.5: Altitude profiles of the natural background $H^*(10)$ rate (left) and the net dose rate (right) of a ^{137}Cs source located on the ground (right), measured with CeBr_3 1.5" x 1.5" detector at the Mollerussa aerial site in Spain (Vargas et al. 2021). The heights (in meters) refer to the flight altitudes above ground. Uncertainties are displayed with coverage factor $k = 1$.

In the Preparedness project the responses of a 50 mm × 50 mm (diameter × height) NaI, a 38 mm × 38 mm CeBr₃ scintillator and a 1500 mm³ CZT semi-conductor airborne spectrometric detector mounted on the DJI Matrice 600 Pro were compared in a weeklong comparison campaign carried out at the aerial site of Mollerussa (Spain) in September 2019. The decision thresholds that indicated the presence of artificial radioactivity were calculated for the count rates in the low energy region of the spectra, the man-made count rates and the ambient dose equivalent rates from the background flights performed at altitudes ranging between 10 m and 60 m. The capability of the different airborne systems to detect and determine the activity of a ¹³⁷Cs point source (model A3011 from Eckert & Ziegler company) with a certified activity of 346 MBq (referenced on May 27th, 2019) at different flight altitudes were compared. Finally, the airborne systems showed the ability to localize the ¹³⁷Cs point source by flying in parallel lines at 10 m, 20 m and 40 m heights. The results of the comparison have been published (Vargas et al., 2021).

As an example of the results, Figure 6.6 shows the distribution of $\dot{H}^*(10)$ rate for a background flight at a nominal altitude of 20 m. Following the ISO 11929-4 (2018), with the standard deviation of background measurements $\tilde{u}(0)$, the decision threshold a^* for a measurand with a 95 % confidence level can be calculated as:

$$a^* = 1.645 \cdot \tilde{u}(0) \quad (5)$$

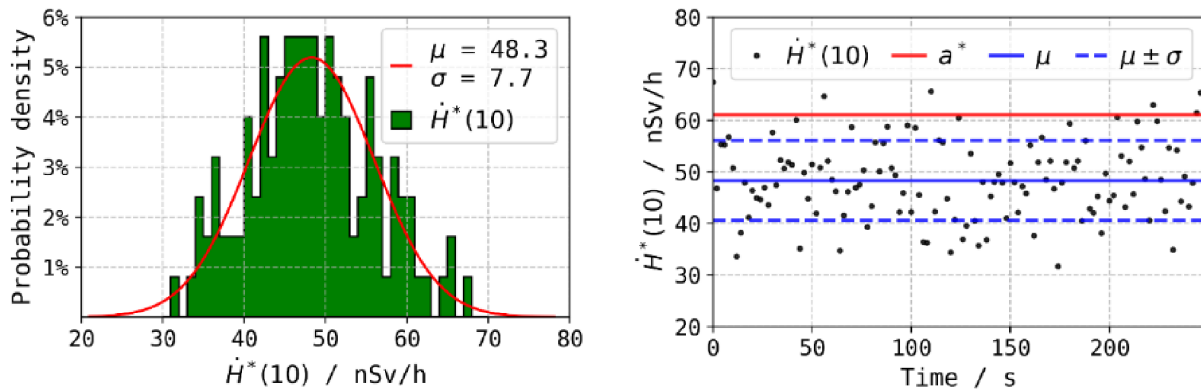


Fig. 6.6: $\dot{H}^*(10)$ rate histogram distribution (left) and time series (right) for a 20 m height AGL background flight measured with a CeBr₃ 1.5" x 1.5" detector at Mollerussa aerial site in Spain.

Figure 6.7 illustrates an example of the net $\dot{H}^*(10)$ rates, i.e., the measured $\dot{H}^*(10)$ rates subtracting the background, during the hovering flights over the ¹³⁷Cs point source measured by the CeBr₃ 1.5" x 1.5" detector. The detector is able to detect the ¹³⁷Cs source for altitudes up to 40 m for the spectrum acquisition time of 2 s. At 60 m, the net $\dot{H}^*(10)$ rates are below the decision threshold but still above 0 nSv h⁻¹. This means that the detector is not able to detect the source with a spectrum integration time of 2 s and require summing up the spectra in order for the source to be detected.

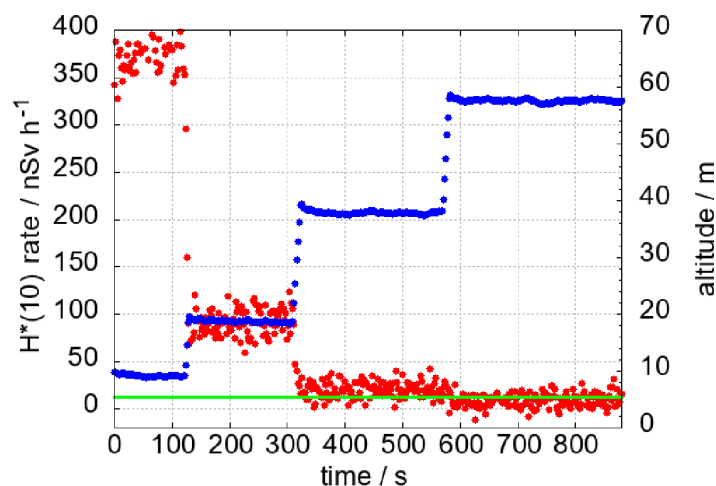


Fig. 6.7: $H^*(10)$ rate time series in hovering flights measured with a CeBr_3 1.5" x 1.5" detector over the ^{137}Cs source at different altitudes. Red dots are the measured values with an integrating time of 2 s, blue dots are the measured height AGL and green line indicates the decision threshold value.

Man-made gross count (MMGC)

The man-made gross count (MMGC) rate, is a simple method to improve the detection of artificial sources by suppressing the influence of the variations in the concentration of the natural radionuclides. This method is based on the fact that even if the natural radioactivity varies, the shape of the produced background spectrum remains almost constant. Therefore, the ratio of the count rate in the natural energy region (n_{high}) to the count rate in the artificial plus natural energy region (n_{low}) is essentially constant in case where there are no artificial radionuclides. The high energy region is defined from the ^{40}K to 3000 keV, with everything above 3000 keV considered to be of cosmic origin. For instance, for a NaI crystal of 50 mm x 50 mm the energy range spans approximately from 1360 keV to 3000 keV. In the low energy range, it's advisable to avoid very low energies for improved accuracy and robustness. Thus, in a common scenario where radionuclides such as ^{137}Cs and ^{131}I are predominant, the starting energy of the low energy range can be around 200 keV. This ratio (r_{art}) is calculated using Eq. (4) by carrying out measurements when no artificial radionuclides are present, i.e. n_{MMGC} is zero. The r_{art} has been calculated for different altitudes from the aggregated 2-second spectra, measured at each altitude, and then fitted by a linear function. The derived altitude dependent r_{art} is then applied to each measured spectrum to determine the man-made count rates:

$$\text{MMGC} = n_{\text{low}} - r_{\text{art}} n_{\text{high}} \quad (4)$$

Figure 6.8 shows the ratio of the low energy region to counts in the high energy region as a function of the altitude above ground level for a CeBr_3 detector of 1.5" x 1.5". This ratio increases due to the scattering in air that lead more transfers of photons from the high to the low energy as the altitude increases.

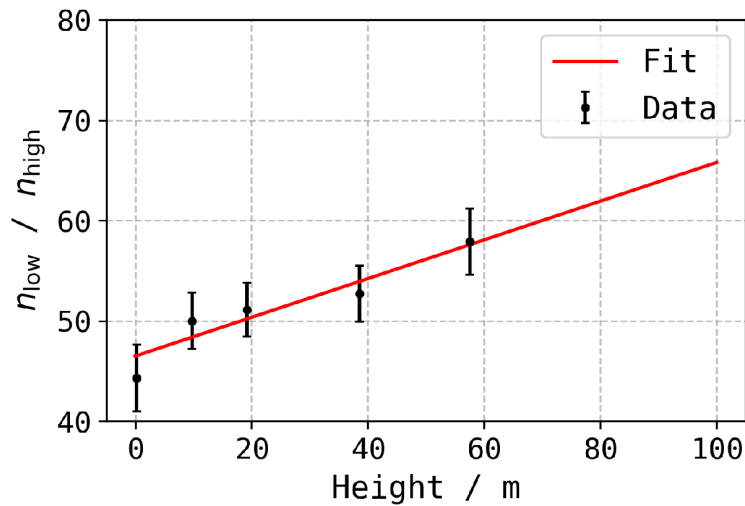


Fig. 6.8: Measured ratio of counts in the low energy region to counts in the high energy region as a function of the altitude AGL for a CeBr_3 detector of 1.5" x 1.5"

In Figure 6.9, the $H^*(10)$ rate and the *MMGC* rate are shown for the measurements carried out in the former uranium mines at Seelingstädt. This figure demonstrates that the *MMGC* can clearly identify the artificial source region (blue line peaks), while the $H^*(10)$ or the total count rates fluctuation is due to both the natural and the artificial source variations but cannot be distinguished.

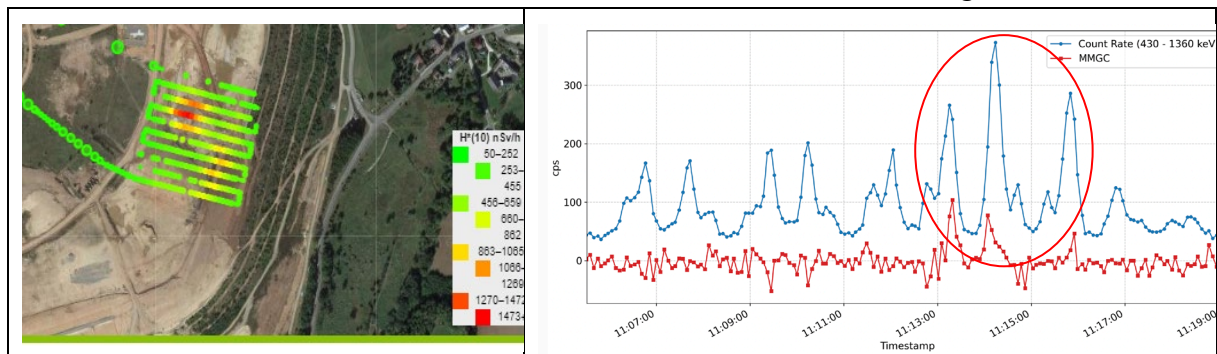


Fig. 6.9: A single flight during the measurement campaign carried out in the former uranium mine in Seelingstädt (Germany). Right: temporal plot of total count rate in the 430 keV to 1360 keV (blue line) and *MMGC* rate (red line). In red circle the measurements when the UAS is flying over the source which is clearly identified by the *MMGC* method. Left: heat map of the $H^*(10)$ at 1 m height AGL

Spatial distribution of radionuclide activity

The determination of concentrations of terrestrial radioisotopes (Bq m^{-2} , Bq m^{-3} , or Bq kg^{-1}) from measurements with a detector positioned in the air at a certain distance from the soil surface is a specific task of gamma spectrometry and is the most comprehensive way to quantify the radiological status of a scenario, as other radiological variables can be derived from these activities, such as $H^*(10)$ rates.

There are three methods for calculating the activities of radionuclides from spectra: a) based on spectrum counting, such as the International Atomic Energy Agency's (IAEA) spectral window method (IAEA, 1991), b) full spectrum analysis (FSA), c) based on fluence (e.g., compartmental model). The drawbacks and advantages of each method are briefly discussed here.

6.2.2.6 IAEA spectral window method

This method has been commonly used in airborne measurements with manned systems for the mapping of NORM radionuclides. Essentially, it calculates net counts in photopeaks in energy spectrum regions corresponding to potassium, uranium, and thorium spectral lines. This method requires prior knowledge of inference ratios to correct the count rate in a photopeak due to the effects of other radionuclides. The method can be extended to other radionuclides such as ^{131}I or ^{137}Cs . The most significant limitation of this method is that the presence of radionuclides must be known "a priori," and additionally, the quantity of radionuclides cannot be high, because of the increase of the uncertainties due to the increase in the number of inference ratios

6.2.2.7 Full Spectrum Analysis (FSA)

The purpose of the full spectrum analysis method is to reconstruct an input gamma spectrum by several so-called reference spectra. The FSA has been developed in different approaches such as in Hendriks *et al.*, 2001 or applied to UAS measurements in Vargas et al. 2021, and was found to be a successful spectrum analysis tool. When applied to scintillation detectors, it enhances their potential for radioisotope quantitative determination by reducing the required statistics of a single measurement and therefore its duration in time.

The spectra acquired in-situ are fitted by a linear combination of the unit activity spectra derived for each isotope from the corresponding Monte Carlo simulation. The events registered in each channel in the measured spectrum N can be expressed as

$$N(i) = \sum_{k=1}^m A_k S_k(i) + B(i) \quad (5)$$

where $N(i)$ is the number of counts in channel i , A_k is the activity of k^{th} source (for m different radionuclide source configurations), S_k is the unit activity spectrum for the k^{th} source configuration, and $B(i)$ is the number of counts in the channel i due to inherent background and secondary cosmic radiation.

The solution of Eq. (5) comes to the minimization of the following cost function:

$$\chi^2 = \frac{1}{n-m} \frac{\sum_{i=1}^n [N(i) - \sum_{k=1}^m (A_k S_k(i) - B(i))]^2}{\left(\sum_{k=1}^m A_k S_k(i) \right)} \quad (6)$$

where n is the number of selected channels in the ROI (Region Of Interest).

The FSA (Full Spectrum Analysis) method has proven to be useful in UAS-based measurements since, due to the short integration time of each spectrum, the number of counts in each energy channel is usually not sufficient to carry out an analysis of each photopeak individually. Therefore, it is more practical to implement a method that analyses the entire spectrum, including the contribution of scattered photons.

Additionally, the FSA method is highly valuable for analysing spectra with a complex mixture of contributions from various radioisotopes, but requires a list of radionuclides present in the spectrum. In an unknown radiation environment, this complicates automated processing and typically necessitates manual input of this information.

6.2.2.8 Compartmental model

In this model, spectrum deconvolution is initially used to obtain the incident fluence at the detector. The scenario is discretized into i compartments. The fluence calculated through deconvolution at

the detector is then applied to the formula for the contribution of flux at a point (detector) produced by a differential volume (compartment):

$$d\phi = \frac{A_m \rho \nu}{4\pi r^2} \exp\left(-\sum_i \mu_i r_i\right) dV, \quad r = \sum_i r_i, \quad (7)$$

Where A_m is the specific activity (Bq g⁻¹), ν is the gamma emission probability per radioactive decay, ρ is the volume density of the emitter (g cm⁻³), μ_i and r_i represent the linear attenuation coefficient (cm⁻¹) and photon transport distance (cm) of the i -th compartment, respectively.

In order to determine the activity in each compartment the following equations system has to be solved:

$$F_{n \times m} \cdot A_{m \times 1} = \dot{H}_{n \times 1} \quad (8)$$

Where $F_{n \times m}$ is the matrix of response factors for each source mesh node $j(1, m)$ and for each measured data point $i(1, n)$. $A_{m \times 1}$ is the activity of the source term and $\dot{H}_{n \times 1}$ rate is the measured ambient equivalent dose rate. A straight-forward solution to Eq. (8) to calculate the $A_{m \times 1}$ does not exist due to the ill-conditioned nature of the response factor matrix F . In the work by Connor et al, 2022), a least-squares solution is employed using an algebraic reconstruction with Tikhonov regularization. However, the solution highly depends on the choice of the regularization parameter and cannot be trusted in general.

Figure 6.10 (Luchov 2024) illustrates the specific activities calculations of ²¹⁴Bi, ²⁰⁸Tl and ⁴⁰K isotopes derived from the 25 m AGL flight at the former uranium mines in Seelingstädt. The density of soil (clayey) was set to 1.24 g cm⁻³ in the compartmental model. The radionuclide concentrations in soil are color-coded to show the contributions of potassium, uranium decay chain (via bismuth) and thorium decay chain (via thallium) to the dose rate produced in the different regions of the map. The general arching shape of the dose rate distribution is followed by all tracked radionuclides.

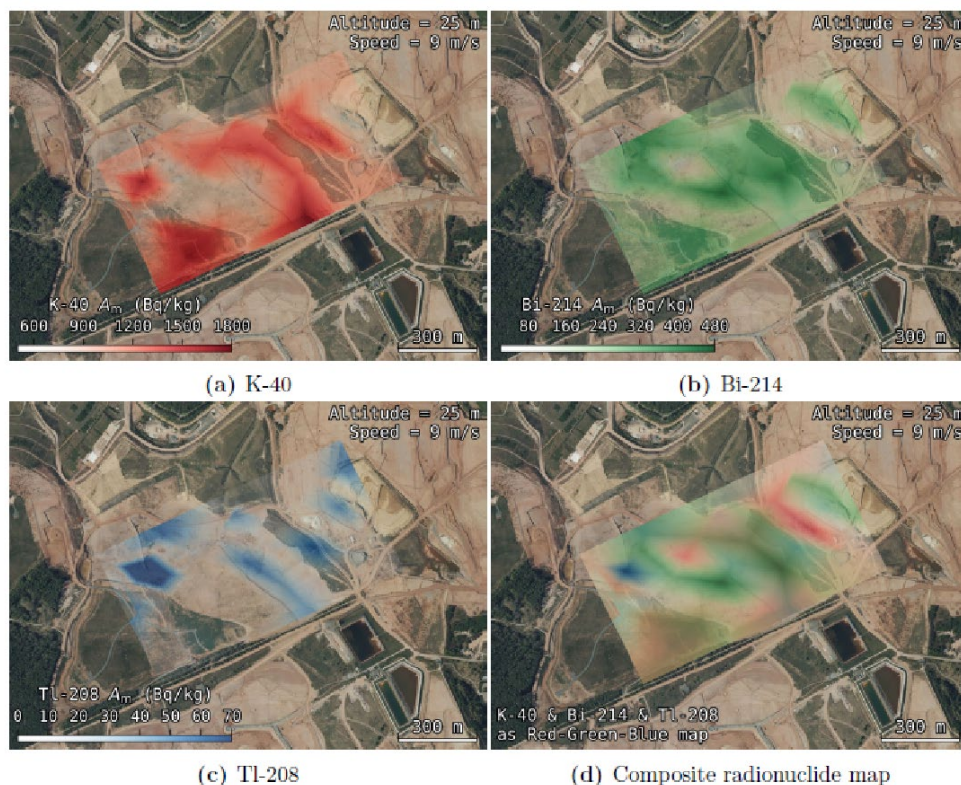


Fig. 6.10: Survey area maps of ^{214}Bi , ^{208}Tl and ^{40}K soil radionuclide concentrations measured at a nominal altitude of 25 m height AGL. Raw 2 s spectra are grouped in 16 entries to acquire the necessary statistics for the flux calculation. The composite radioelement map uses the normalized radionuclide concentration to highlight the dominant regions (Luchkov 2024)

6.2.3 Source location

Besides the visual estimation of the source localization according the highest $H^*(10)$ rates or the MMGC shown in the radiological maps, there are several calculation algorithms to determine the location of a point source. In the paper of Vargas et al. (2021), two algorithms are described: i) based on an Evolution Algorithm (EA) which is a generic population-based heuristic optimization algorithm applied to a NaI 50 mm x 50 mm detector and ii) using the compartmental model described in a previous section applied to a 38 mm x 38 mm CeBr_3 detector. Table 6.3 shows the estimated position of a ^{137}Cs point source of 345 MBq by using the two methods, the uncertainties and the absolute distance to the actual position ($41.612430 \pm 1.6\text{E-}5$ N, $0.854505 \pm 1.6\text{E-}5$ E) for the NaI and CeBr_3 detectors, respectively. The restored positions have satisfactory precision comparable within the accuracy of the GPS modules used.

Table 6.3. Source coordinates and corresponding uncertainties as measured with the NaI detector (top values) and the CeBr₃ detector (bottom values). Actual position is $41.612430 \pm 1.6\text{E-}5$ N, $0.854505 \pm 1.6\text{E-}5$ E.

Nominal altitude (m)	Longitude		Latitude	
	measured	abs. difference (m)	measured	abs. difference (m)
10	$0.854505 \pm 6\text{E-}6$	0.0 ± 1.4	$41.612420 \pm 5\text{E-}6$	1.1 ± 1.9
	$0.854505 \pm 1\text{E-}5$	0.0 ± 1.5	$41.612444 \pm 8\text{E-}6$	1.6 ± 1.6
20	$0.854513 \pm 1.6\text{E-}5$	0.7 ± 1.8	$41.612434 \pm 1.3\text{E-}5$	0.4 ± 2.3
	$0.854503 \pm 3.2\text{E-}5$	0 ± 3	$41.612411 \pm 2.4\text{E-}5$	2 ± 3
40	$0.85451 \pm 9\text{E-}5$	0 ± 8	$41.61245 \pm 6\text{E-}5$	2 ± 7
	$0.85434 \pm 2.0\text{E-}4$	14 ± 17	$41.61237 \pm 9\text{E-}5$	7 ± 10

A more efficient method of locating the radioactive hotspots that uses data of the real-time gamma-ray locator detector, was developed in the Preparedness project. With this method, the UAS computer acquires the angular radiation vector information from the locator at its 10 Hz refresh rate (i.e., in real-time). The system has no moving parts and relies on a multiply segmented, partially shielded, scintillation detector, consisting of almost 300 g of CsI(Tl). The radiation direction is deduced from the relative dose rates sensed by the different segments of the complex sensor cluster and can report radiation direction from all 4pi of the solid angle. At the same pace, the on-board computer executes calculations of the anisotropy of the vector directions to judge the possible presence of a radioactive source in the vicinity and, if the anisotropy criterion is reached, the UAS will automatically stop and start collecting more angular data to better estimate the position of the radioactive source. These results will then be shown to the pilot in the remote graphical interface to decide on further actions, i.e. to send the UAV to a new location possibly closer to the radiation source or to continue the pre-planned path. Within a few such consecutive source position estimations, the UAS should arrive on top of the radioactive source location.

As an illustrative result of the locator detector, a test flight at an altitude of 10 m and a UAS velocity of 2 m s^{-1} is presented in Figure 6.11. The UAS autonomously flew over the planned path and kept calculating in real time the radiation anisotropy estimator to see if the criterion of meaningful anisotropy was reached. When this criterion was met (Figure 6.11, point a), the UAS stopped and started accumulating more data and calculating the direction towards the radioactive source, sending to the pilot the absolute geolocation positions where the path towards the source probably was. These positions are illustrated as "x" markers in green colour in Figure 6.11 under section (a) and a "+" marker in the same colour represents their centre of mass. Then, the UAS pilot flew to the next measuring point (Figure 6.11, point -b). The same procedure was repeated until the UAV reached the position (d) where the UAS descended from the initial 10 m altitude to approximately 2 m to improve the statistics with better measurement.

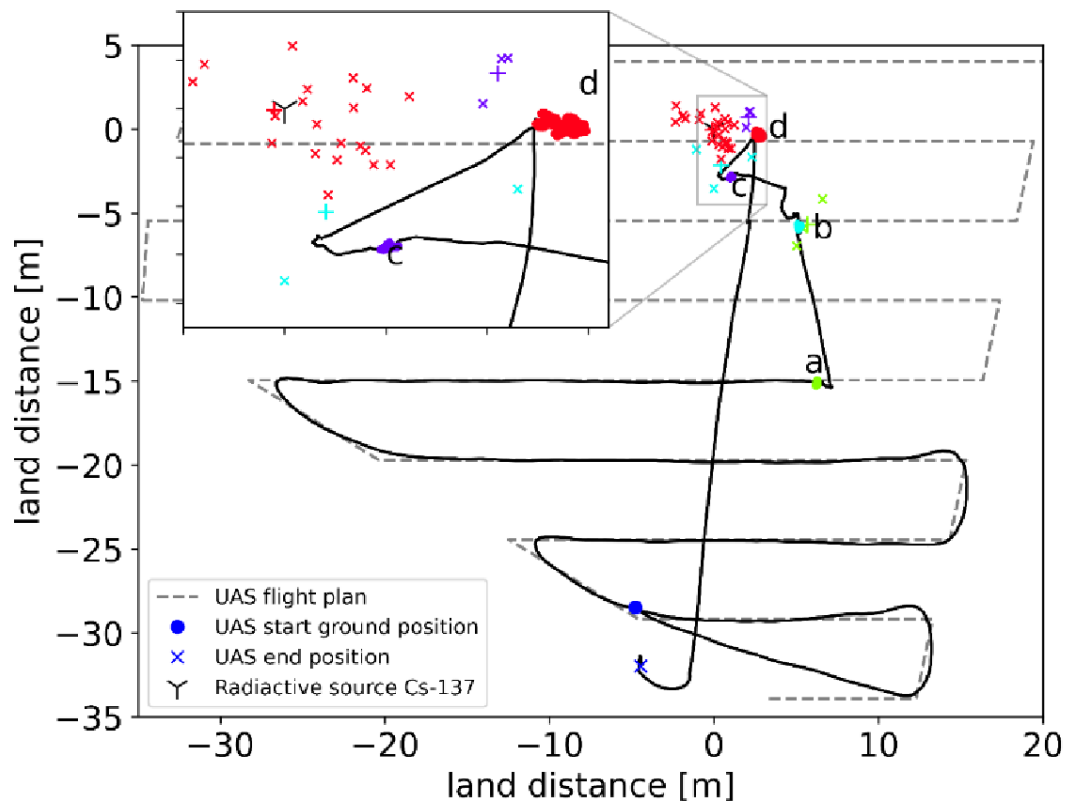


Fig. 6.11: Field flight with the result of source localisation. The dots represent the position of the UAV while calculating the direction towards the radioactive source (marked with x) and their average marked by +. The same colour represents the same group (drone position and calculated source direction), also marked with small letters (a, b, c, d).

A video of the results obtained by a NaI spectrometer and the locator detector mounted on a DJI Matrice pro 600 UAS can be accessed using the following link: <https://www.youtube.com/watch?v=IV45uvionKI&t=46s>. The final report of the Preparedness project and other publications can be found in the link: <https://www.euramet.org/research-innovation/search-research-projects/details/project/metrology-for-mobile-detection-of-ionising-radiation-following-a-nuclear-or-radiological-incident>.

6.3 Remote detection of alpha emitting radionuclides.

A detection system to remotely measure alpha particles is not commercially available and, currently, in order to minimize the health risk in an event with this kind of radionuclides, it is necessary to provide the first responders with appropriate protection equipment and with alpha detectors that should be located very close to the alpha-emitter radionuclide. This leads to a highly time consuming and tedious task. Optical detection of alpha particle emitters is a technique that could be an interesting solution to measure remotely the alpha-emitters radionuclides and avoid to expose the personnel at risk.

The physical principle of the optical technique is based on the alpha radiation detection via radioluminescence of air. An alpha particle induces radioluminescence light when absorbed in air. Radioluminescence in air is generated mostly by the emission of molecular nitrogen (N_2), and to a much lesser extent, by trace amounts of nitric oxide (NO) in air, with wavelength spanning three ultraviolet (UV) bands and about 99% of emission occurring in the 280 nm to 440 nm spectral range

(Sand, 2016). The diurnal background light is composed of approximately 95% UV-A (wavelengths from about 315 nm to 400 nm), 5% UV-B (wavelengths from about 280 nm to 315 nm), and almost no UV-C (wavelengths from about 100 nm to 280 nm). Therefore, the optical detection of alpha-emitting radionuclides is complex because the UV-A emissions are eclipsed by the much greater solar irradiance. Due to the low UV-C background emissions, the idea of acquiring only the UV-C (solar blind) light during daytime is a good alternative. However, the intensity of UV-C radioluminescence is very low (< 1%) which makes the detection quite challenging.

The overall objective of the RemoteALPHA project is to develop novel optical systems for the remote detection and quantification of large-scale contamination with alpha emitters in the outdoor environment for the first time, allowing sound and quick countermeasures in the case of a radiological emergency. The optical detection system integrated in the UAS developed at this project is based on a light weight Poly(methyl methacrylate) (PMMA) Fresnel (Orafol Fresnel Optics) having a diameter of 452.9 mm (SC 2045) and nominal focal lengths of 391.5 mm at 546 nm. The photon filtering is done with 3 Semrock FF01-260/16-25 interference bandpass filters centered at 260 nm with a bandwidth of 16 nm. At this operation wavelength selected by the interference filters, the focal length of the Fresnel lens is 424.5 mm. To achieve the best UV-C band filtering, a set of two corrective fused silica lenses (Thorlabs LC4252 and LA4052) were installed to parallelize the collimated photon beam. An additional filtering performance was achieved by tilting two filters in the set by 3 degrees relative to the optical axis.

Photon counting is performed with a PMT detector that converts incident light into an electrical signal that is analysed by a digital pulse processing unit. The detection system uses the solar-blind Hamamatsu H11870-09 photomultiplier tube with transistor-transistor logic (TTL) output and the GBS Elektronik MCA527 OEM+ multichannel analyzer (MCA) working in the scaler (gating) mode to translate these TTL signals into the timestamped count rate.

A carbon fibre-reinforced cone-shaped polymer frame has been built to hold the lens and the detection system. The frame materials were chosen to provide the necessary stability for the system while keeping it lightweight. The detection system has a height of 790 mm and a diameter of 520 mm at the base. Due to the dimensions of the system, no commercial UAS was tall enough to natively integrate the detector under its belly, since the height of the detector is larger than UAS legs. This made it difficult to integrate the detector into a commercial UAS unless the detector itself acted as a landing gear. Therefore, the structure of the Fresnel lens has been designed such that it can be used as a landing gear of the unmanned aircraft. The alpha radioluminescence detector with the landing legs, the MCA counter, the laser altimeter, and the onboard computer with a 4G/LTE modem were integrated with the DJI Matrice 600 Pro (M600P) UAS as shown in Figure 6.12.



Fig. 6.12: Left: Radioluminescence detection system integrated with the DJI Matrice 600 Pro and ready to fly. Right: Flight performed with the system at the Drone Research Laboratory (DroneLab) of the Technical University of Catalonia (UPC)

In order to characterize the detection system two parameters are crucial, i.e., the field of view (FOV) and the detection efficiency. With a field of view of the order of 10 cm, it is obvious that in scenarios with point alpha sources, the flight tracks should be very close together, the speed of the UAS should be very low, and the integration time should be short to maximize the probability of their detection. Another concern with this type of scenario is that the UAS is not always completely perpendicular to the ground due to variations in wind and UAS speed resulting thus in changes in the roll and pitch of the whole flying system. These variations in the roll and pitch of the UAS reduce the number of counts acquired compared to a vertical position that is completely aligned with the ground. But on the other hand, this variation in roll and pitch increases the probability of detecting the source because the area covered by the detector is larger than when the UAS is in constant vertical alignment.

Figure 6.13 displays a screenshot of a flight performed in December 2022. The color and height of the cylinders indicate the counts measured in 0.1 seconds. The colour range and height of the cylinders can be modified to optimize the visualization of the source detection. The detected counts can be also read numerically in the right-hand data box. In this data box, the UAS telemetry is also displayed: speed, pitch, roll, yaw, longitude, latitude, and altitude. Behind the telemetry box, the commands box is placed which contains the dialog for each possible command that can be sent to the UAS. The visualization software also shows the UAS position and the location on the ground where the detector is pointed (violet line). The software has more extensive functionality beyond the application-related features shown here, such as user management tools, layer visualization, and flight plan editing.

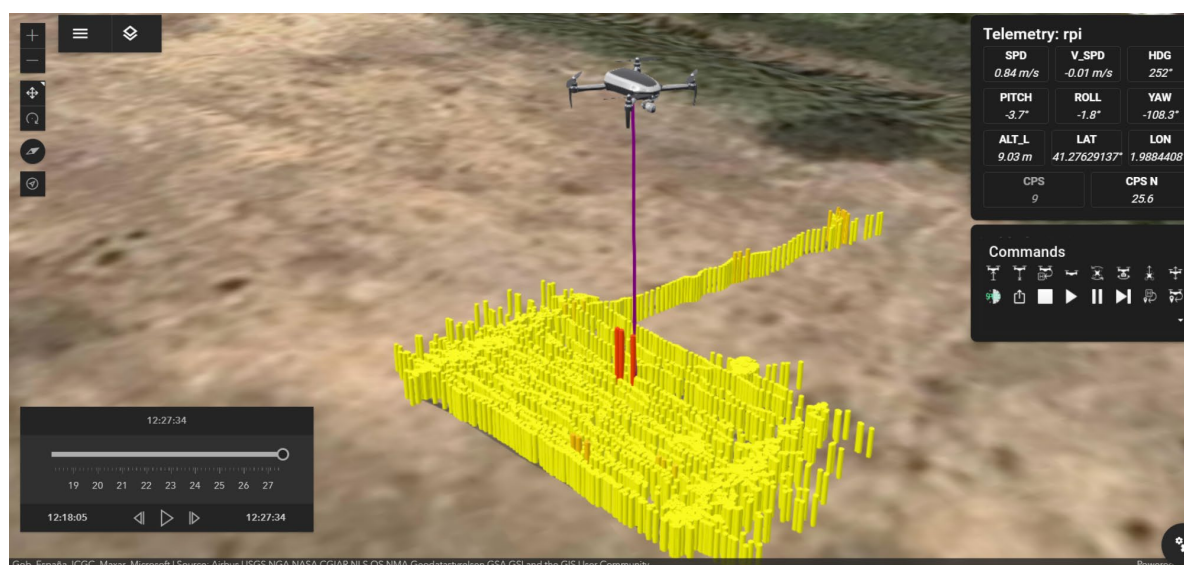


Fig. 6.13: Screenshot of the visualization software during a test flight around an ^{241}Am source. To cover the complete area and considering the small FOV a distance between parallel lines of 10 cm needs to be used. However, the UAS does not permit placing waypoints in such close proximity, therefore the shortest distance allowed by the UAS, approximately 70 cm, was selected. Given the manufacturer-specified horizontal positioning error of 1.5 m, it will be necessary to perform several iterations of the parallel line scan pattern with a small offset to ensure that the UAS passes through areas it omitted in the previous iteration.

A complete description of the system can be found in Royo et al. (2024). The main conclusion from the results presented is that alpha particles can be detected remotely from an UAS by using optical detectors. To the best of authors' knowledge, these results demonstrate for the first time remote detection of alpha particles from an UAS. The described prototype system, with certain operational improvements, can be used to assess contaminated areas without the need to put first responders at risk.

As mentioned before, due to GPS uncertainties and the wind effects (roll and pitch variations), the same flight plan repeated many times does not cause UAS to fly exactly over the same place and, therefore, the measurements are scattered between consecutive runs. For instance, during experiments with five LEDs (Light-emitting diode) placed in random positions, not all of them were detected during the same iteration. During the first flight, three of the five LEDs were detected. Using the same flight plan in the following flights, either no LEDs or all of them were detected. Concluding the flight results, the probability of detecting a point source constitutes approximately 60%. This probability can be boosted by increasing the field of view (FOV) of the detection system. However, increasing the FOV also has some other consequences such as the reduction of the detector efficiency and background light leakage. Extended sources with larger lateral dimensions (e.g., on the order of GPS uncertainty), on the other hand, would not be susceptible to difficulties encountered in the case of point sources.

Another important conclusion of this work is that the operational experience gathered with the novel radioluminescence detection system can be readily applied to the extended alpha source detection (spread over the area), while the point source localization needs further hardware improvements. All test flights carried out during this study were performed using point sources – LEDs or the americium source – due to the difficulty of simulating an extended source. Even though

point sources are difficult to detect given the optical constraints, the prototype system was able to detect them with a reasonable probability of about 60%. Extended sources are far less dependent on alignment stability and are therefore much easier to detect.

The final report of the RemoteALPHA project and other publications can be found in the link: https://www.euramet.org/research-innovation/search-research-projects/details?tx_eurametctp_project%5Bproject%5D=1687&cHash=c8e79ec377e929c4b2f6a6360dbc0968

6.4 Acknowledgements

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7 Contribution of computational dosimetry to the management of radiological accidents

C. Huet, Health Division, Research Dosimetry Department, ASNR

7.1 Context

Since the 20th century, the use of ionizing radiation has experienced a great expansion that was consequently accompanied by a number of radiological accidents. There were approximately 600 accidents between 1980 and 2013, involving, in total, 2390 overexposed persons (Coeytaux *et al.*, 2015). Most of the reported cases occurred in the medical sector with 32% in radiotherapy and 31% in fluoroscopy and in the industrial sector with 27%. Local or partial body irradiations represent the most common type of irradiation condition, which means that exposure are heterogenous. In addition, given the threat of malevolent use of radioactive materials by terrorist groups, to be prepared to mass casualties exposures became crucial.

For the medical management of victims of external exposure, priority is given to diagnosis (Fliedner *et al.*, 2001). Thus, in case of small-scale accidents involving a small number of people (such as a confined industrial or medical accident), the first step of the medical management is to make a diagnosis that will help to define a therapeutic strategy. For medium to large-scale accidents involving a large number of people, the primary purpose of the initial-phase dose assessment is to identify individuals in potential danger of short-term deterministic effects in order to prioritize care for these individuals.

Therefore, dosimetry is a key point for triage and diagnosis purposes. As a matter of fact, absorbed dose is an indicator of tissue or organ damage that helps clinicians to assess radiation-induced damage and define the therapeutic strategy. Thus, for external overexposure, the dosimetry challenge is to obtain an accurate and reliable assessment of the absorbed dose and more specifically of the dose distribution within the body.

7.2 Dose assessment: the dosimetric tripod

Dose can be assessed using data from different complementary fields, namely the dosimetric tripod as described below.

7.2.1 Clinical dosimetry

First, clinical dosimetry is based on observation of the time to onset of symptoms (Fliedner *et al.*, 2001). In the case of localized irradiation, the evolution of skin lesions can give an indication of the dose received by the surface tissues whereas in the case of whole-body irradiation, clinical signs or symptoms of the acute radiation syndrome such as nausea or vomiting are indicators of the mean whole-body dose. The main drawback of clinical dosimetry for whole-body irradiation is that symptoms are not specific to the irradiation. For instance, vomiting may be due to the stress generated by the announcement of a suspected irradiation. It is then difficult to distinguish between the overexposed individuals and the “worried well”.

7.2.2 Biological dosimetry

The second method is biological dosimetry. It is based on the analysis of cytogenetic damage in peripheral blood lymphocytes. The counting of dicentric chromosomes produced by irradiation is

considered as the gold-standard technique (IAEA, 2001). The main drawback of biological dosimetry is that, although it can identify a partial-body radiation exposure, it is not possible to estimate the dose distribution within the body. Moreover, even if for triage purposes the number of metaphases to score can be reduced, it takes at least 48 hours after the sample has reached the laboratory to get the dose estimation.

7.2.3 Physical dosimetry

Finally, one can use physical dosimetry. Physical dosimetry involves three different fields: individual dosimetry, retrospective dosimetry and dose reconstruction.

Regarding individual dosimetry, the feedback from past accidents shows that the majority of the overexposed person were not monitored individuals or did not wear a dosimeter at the time of exposure. That condition makes individual dosimetry inapplicable in most cases.

Physical retrospective dosimetry consists in assessing the dose from samples collected on the victims or in their close environment. These samples can be biological samples such as tooth enamel (Romanyukha *et al.*, 2013), bone or non-biological samples such as glass (Discher and Woda, 2013) or electronic components from mobile phones (Bassinat *et al.*, 2014). Depending on the nature of the sample, different techniques such as electron spin resonance (EPR) or luminescence (TL/OSL) can be used. The advantage of retrospective dosimetry is that no *a priori* knowledge of the exposure scenario is required. In addition, the dose assessment can be achieved within a few hours after sample receipt. However, only the dose at a point, that is to say, the dose at the sample level, can be obtained.

Dose reconstruction consists in reproducing the circumstances of the accident by experimental or numerical means. Among the techniques of the dosimetric tripod, it is the only one that can give information on the dose distribution within the body. Nevertheless, the knowledge of the exposure scenario is necessary. That means that an accurate dose assessment is possible only if information such as the individual's position in relation to the source, exposure time, source characteristics, etc. is available.

7.3 Numerical dose reconstruction

The principle of dosimetric reconstruction of a radiological accident using numerical simulations is firstly, to model the person exposed to radiation using a numerical anthropomorphic phantom in the accident environment and secondly, to calculate the absorbed doses within the body using a Monte Carlo transport code. Concerning this point, a number of general-purpose Monte Carlo particle transport codes such as MCNP (Goorley *et al.*, 2013), Geant4 (Agostinelli *et al.*, 2003) and PHITS (Sato *et al.*, 2013), are available.

A numerical phantom is a computerized representation of human anatomy for use in radiation transport simulations. Three classes of computational phantoms have been developed (Xu *et al.*, 2014): stylized, voxel and hybrid phantoms. The earliest computational phantoms, called stylized phantoms, appeared in the 1960s; the organs are defined from a combination of mathematical equations which describe spheres, cylinders, cones, ellipsoids, etc. (Fisher and Snyder, 1966). The advantages of these phantoms are smooth surfaces and flexibility – they can be modified by appropriate changes in parameters of the equations they are described with. However, the major disadvantage of stylized phantoms is an approximate description of the human anatomy due to the use of simple geometrical volumes to model organs. Voxel phantoms, developed in the late 1980s, are made of a huge number of tiny cubes called voxels (Williams *et al.*, 1986). These phantoms are

created from CT or MR images of patients or cadavers. Thus, the anatomical accuracy and realism are greatly improved compared to stylized phantoms, but voxel phantoms are hardly flexible and therefore difficult to modify in terms of morphology and posture. Finally, hybrid phantoms combine both the flexibility of stylized phantoms and the anatomic realism of voxel phantoms. They are based upon non-uniform rational B-spline (NURBS) and/or polygon mesh surfaces. However, until recently, only a few Monte Carlo codes were able to handle the mesh format. In addition, the computing speed with this kind of phantoms is in most of the codes much slower than with voxel phantoms.

The choice of the phantom used in the reconstruction depends on the medical need. For a whole-body irradiation, the key clinical issue is whether spontaneous secondary resumption of bone marrow activity is possible. Thus, the role of dosimetry is to estimate the heterogeneity of the dose distribution within the body, in particular at the level of the bone marrow areas. For that purpose, generic stylized, voxel or mesh phantoms can be used. Regarding localized irradiation, tissue necrosis occurs when absorbed dose to tissue exceeds 25 Gy. The key clinical issue is whether a surgery is needed to remove these tissues (Benderitter *et al.*, 2010). The objective of dosimetry is to define the dose distribution at the lesion level. Anatomical accuracy is required, thus personalized voxel phantoms created from the images of the victim are constructed.

7.4 Contribution of computational dosimetry to small scale radiological accidents management

For accidents involving few people, numerical dose reconstruction makes a valuable contribution to the medical management of exposed persons. Figure 7.1 lists the biological and physical dosimetric methods used by IRSN for radiological accidents since 2005. As shown in this figure, numerical dose reconstruction has been used each time, whether for local or whole-body irradiation.

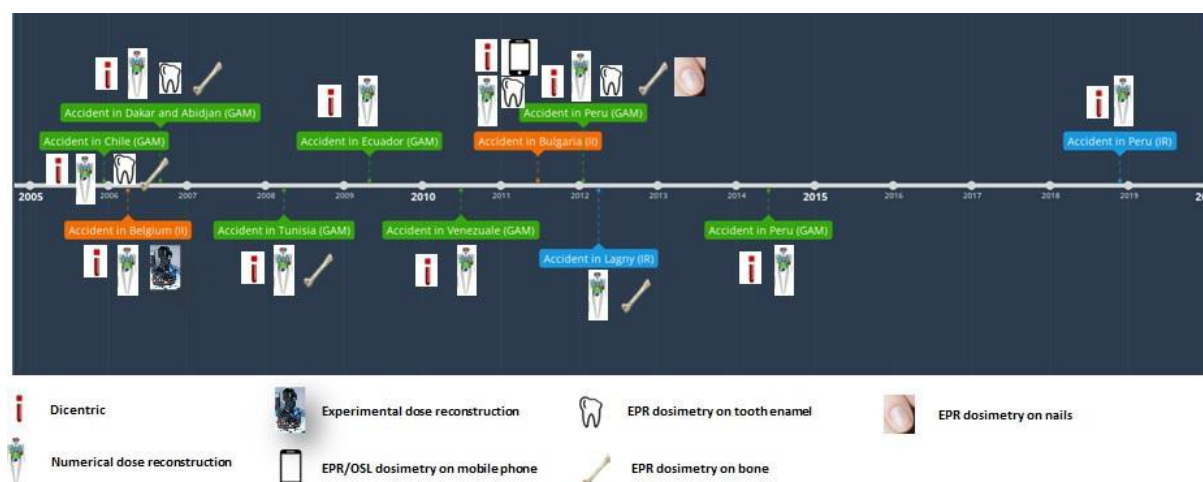


Fig. 7.1: Overview of biological and physical dosimetric methods used by IRSN for the management of radiological accidents since 2005. In green: industrial radiography source, in orange: sterilization facility, in blue: interventional radiology.

7.4.1 Localized irradiation: surgical guidance

For more than 20 years, IRSN has been developing an in-house tool named SESAME (Lemosquet *et al.*, 2004; Huet *et al.*, 2009; Huet *et al.*, 2019) for the numerical dose reconstruction of localized irradiations. This tool allows developing personalized voxel phantoms from computed tomography or MR images of the patient and reconstructing the exposure scenario to be used as input of a MCNP

Monte Carlo simulation. The added value of SESAME is its ability to superimpose the calculated dose distribution on the tomographic images of the patient.

SESAME tool was implemented for the first time in 2005 for an accident which occurred in Chile. At a cellulose plant under construction, a worker picked up a high activity Ir-292 source inadvertently fallen from an industrial radiography equipment and put it in his left back trousers pocket for about 10 min. A few hours after exposure, an erythema of about 4 cm extent appeared on his left buttock and fifteen days later a central necrosis occurred. A personalized voxel phantom created from the CT images of the patient was generated and the numerical dose reconstruction was performed on the basis of the scenario described by the patient. Based on the numerical dose reconstruction only, it was the first time that an anticipatory surgery was performed on tissue that appeared to be healthy but that would have become necrotic. An excision measuring 5 cm in depth by 10 cm in diameter corresponding to the 20 Gy isodose was done on the left buttock at the HIA Percy in Clamart (France) (Lataillade *et al.*, 2007, IAEA, 2009). In combination with mesenchymal stem cells injection, complete healing was observed. A similar approach was adopted in the case of a radiography source accident in South America in 2009 (Courageot *et al.*, 2011).

7.4.2 Whole-body irradiation: guidance of treatments for hematopoietic syndrome

Following a whole-body irradiation, a hematopoietic syndrome occurs for a whole-body dose around 2 Gy to 3 Gy. Some areas of bone marrow may be under-exposed. The presence of residual hematopoiesis can radically affect medical decisions regarding the indication for hematopoietic stem cell transplantation.

In 2006, an operator was irradiated in a sterilization facility. A few hours later, he had symptoms of acute radiation syndrome, but he did not make a link with possible irradiation. Nearly three weeks after the exposure, a hematological syndrome (grade 4) was diagnosed. The numerical dose reconstruction performed with a mathematical anthropomorphic phantom showed that the dose distribution within the body was heterogeneous with a factor of 2 between the front and the back and between the pelvis and the skull (Huet *et al.*, 2008). This information was crucial to guide the treatment of the hematopoietic syndrome: as the dose heterogeneity suggested spontaneous secondary resumption of bone marrow activity in some areas, no transplantation was performed and only growth factors were administered (Gourmelon *et al.*, 2010). Similarly, in 2006, following an accident with a radiography source, one patient suffered a hematopoietic syndrome (grade 3). Heterogeneity of the dose was assessed by numerical dose reconstruction using a mathematical anthropomorphic phantom (Clairand *et al.*, 2008). The dose distribution within the patient's body supported the evidence of a residual hematopoiesis. Consequently, even in this case, only growth factors were administered (Gourmelon *et al.*, 2010).

7.5 Contribution of computational dosimetry to large scale radiological accidents management

A European consensus concerning the medical management of mass radiation exposure was obtained in 2005 (Gorin *et al.*, 2006). It was agreed that the first 48 h after a radiological accident involving masses of people are crucial. In that time period, the victims should be processed by an emergency triage system where the patients are scored on the basis of both clinical and biological criteria. However, as mentioned earlier, these triage criteria are neither sufficiently specific to irradiation nor sensitive to allow a reliable categorization. Therefore, complementary and alternative methods are needed.

To deal with large scale accidents, the Multibiodose project (Wojcik *et al.*, 2014) aimed at analyzing a variety of biodosimetric tools and adapting them to different mass casualty scenarios. Seven dosimetric assays were tested (manual and automated dicentric, automated micronucleus, gamma-H2AX, skin speckle, serum protein, EPR, OSL) and a classification of dosimetric triage in three categories was proposed (below 1 Gy, between 1 Gy and 2 Gy and above 2 Gy). Research and development in biological and physical dosimetry for large scale accidents is still a very active field. It aims to:

- reduce the time of analysis and lower the detection limits of the techniques used for diagnosis,
- investigate less invasive methods, new materials and new biomarkers,
- develop tools deployable on site.

In addition, efforts are being made to increase the capacity and to harmonize the practice. To this end, laboratory networks have been created, and intercomparison and training initiatives launched.

Until now, computational approaches have been little considered by international bodies as a dosimetric tool for large-scale accidents, including in the recent ICRU Report 94 (ICRU, 2019) which deals with the description of the most recent applicable dosimetry methods to be applied to individuals in early-phase response to radiological accidents. Nevertheless, research teams have initiated work in two directions: the derivation of conversion coefficients to be applied to measurements on fortuitous dosimeters, and numerical dose reconstruction.

7.5.1 Derivation of dose conversion coefficients

When considering irradiation from external sources, it is important to note that physical retrospective dosimetry methods rarely provide an estimate of dose to the actual target of interest that is to say the dose to an organ or the whole-body dose. Instead, they measure the dose to the physical dosimeter material and not to the individual. Thus, computational dosimetry can be used to derive dose conversion coefficients from dose measured on a sample. It consists of modelling one or several sample locations, several geometries of exposure and calculating predetermined conversion coefficients with a Monte Carlo code.

Combining Monte Carlo simulations with MCNPX 2.4 code and a modified MIRD phantom, Herve *et al.* (2007) derived whole-body and organ dose coefficients to be employed with the dose estimates obtained through EPR measurements on fortuitous dosimeters (dental enamel, sugar) for both photons and neutrons, in several accidental exposures scenarios (AP, PA, RLAT and LLAT standard exposures, source at 1 m, in a pocket, in a hand, and contaminated floor) The same group developed a practical tool based on that database of coefficients to be used in emergency situation.

More recently, Eakins and Kouroukla (2015) calculated the whole-body dose conversion coefficients to be employed with OSL dose assessments performed on the electronic components of a mobile phone. They used MCNPX code and ICRP 110 male voxel phantom. They modelled the whole mobile phone located at four different positions (chest, leg, back and hip) and performed simulations for seven exposure conditions (AP, PA, LLAT, RLAT, ISO, rotational and contaminated ground) for three radiation sources (^{192}Ir , ^{137}Cs and ^{60}Co).

Similarly, Kim *et al.* (2019) performed simulations using the GEANT4 code and MESH phantoms to derive conversion coefficients to be used with the dose measured on the screen of a mobile phone, with the phone located at four different positions (chest, hip, thigh and hand) for six exposure

conditions (AP, PA, LLAT, RLAT, ISO and ROT) and three different postures (standing, kneeling, and squatting). Three commonly used radiation sources (^{192}Ir , ^{137}Cs and ^{60}Co) were investigated.

Although for some exposures conditions these studies have shown large differences between fortuitous dosimeter dose and organ/whole-body dose, for large-scale events external exposures ICRU Report 94 still recommends using the incidental dosimeter dose in the initial phase dose assessment instead of the dose derived from conversion coefficients. In any case, these computational studies have made it possible to quantify the uncertainties associated with each approach.

7.5.2 Numerical dose reconstruction

As mentioned before, reliable numerical dose reconstruction requires knowledge of the exposure scenario. The exposure scenario is based on victims' or witnesses' testimonies. However, often, there is a lack of reliability of testimonies due to the combined effects of a delay in identifying the accident, a physical distance between the individuals and the teams in charge of reconstruction and an individual's forgetfulness of important details. To overcome these difficulties, in 2016, IRSN in collaboration with the French defense radiation protection department, started the development of a dedicated tool called SEED (Entine *et al.*, 2022). One of the special features of this tool is that it can be rapidly deployed on site, as close as possible to the scene of the accident.

SEED uses Geant4-based GATE version 8 Monte Carlo platform and enables rapid modeling of the irradiation scene thanks to a user-friendly interface and a set of available libraries (phantoms, materials, furniture, ...) (Figure 7.2). Different types of phantoms are available, from simple phantoms to anthropomorphic mathematical phantoms and personalized voxel and MESH phantoms, making it a versatile tool that can be used throughout the different steps of the management of radiological accidents, *i.e* from dosimetric triage to accurate dose distribution assessment within the body to guide medical treatment. Phantom morphology and posture can also be modified. In addition, the user can move around in the scene in all directions thanks to a "first person camera". After an experimental validation in 2019 which consisted in irradiating dummies equipped with dosimeters with a high activity Ir-192 radiography source and reconstructing the exposure scenario with SEED in the laboratory (Entine *et al.*, 2022), on-site deployment of the tool was tested in 2022. The scenario simulated an irradiation by a ^{140}La source of people exposed in two different configurations: one third of the time seated at a table near the source, and two thirds of the time lying on cots further away. Four people played the role of fictitious victims, so that they could be questioned about their conditions of exposure (position, time spent in each position, *etc.*). First dose assessments were available 12 hours after the arrival on site.

One of the ongoing developments involves integrating other dosimetric information (dose rate measurement, dose from fortuitous dosimeter, *etc.*) directly into the irradiation scene. This kind of information could be useful, on one hand to "calibrate" if necessary, the Monte Carlo simulation, and in the other hand to inter-relate the different dosimetric assessments.

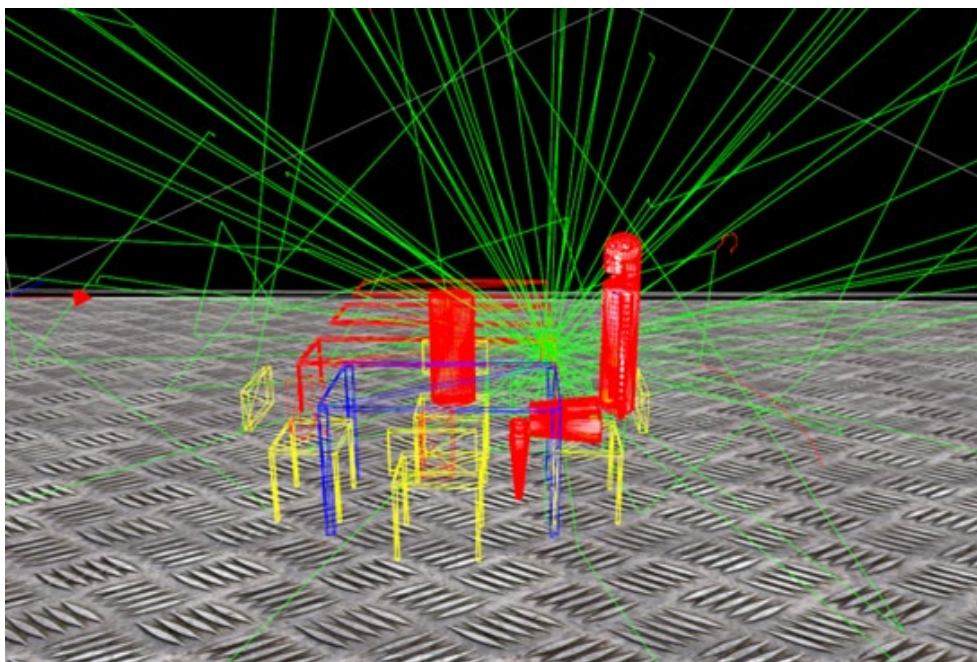


Fig. 7.2: Example of an irradiation scene modelled in SEED

7.6 Conclusion

Over the last twenty years, numerical dose reconstruction has made a valuable contribution to the management of radiological accidents for small scale accident for diagnostic purpose, either alone or in combination with other dosimetric techniques.

Regarding initial-phase dose assessments for large numbers of people, even if it remains a conceptual approach, computational dosimetry may have a valuable contribution. In addition to current developments, new AI techniques and increased computer capacity will certainly help expand interest in computational dosimetry for the management of medium to large-scale accidents.

For both type of events, future directions of research in computational dosimetry have already been identified (Fattibene *et al.*, 2023).

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8 The Goiânia Radiological Accident

Luiz Bertelli, L Bertelli & Associates, Murray, UT, USA ; **Dunstana Melo**, Melohill Technology, Rockville, MD, USA

8.1 Abstract

A ^{137}Cs source assembly from a teletherapy unit was removed from a derelict building belonging to a clinic in Goiânia, Brazil. The source was ruptured and caused external and internal exposure of several persons. Right after the discovery of the accident, a population call was performed requesting the citizens to go to the Olympic Stadium. As a result, 112 000 people were monitored in a short period of time. Individuals who presented with some contamination received medical treatment in various levels depending on the degree of contamination. As a result, 249 individuals presented indicative external and internal doses, being 129 from those with conclusive external and/or internal doses. 49 were admitted to hospitals with 22 under intensive medical care. A long-term individual monitoring protocol was established and it was comprised of in vitro bioassay of faecal and urinary excretion samples, in vivo monitoring through an improvised whole body counter unit and through cytogenetic dosimetry. This procedure allowed obtaining individual retention parameters to be applied in dose assessments and proved crucial in determining the efficacy of the decorporation therapy. It also provided valuable information for epidemiological purposes. Two cases of pregnancies were also followed through long-term monitoring.

8.2 Cause and Discovery

The basic references for understanding the occurrence, mitigation, dosimetric and medical evaluations, and follow-up after the accident can be found in two IAEA references (1)(2).

Around the end of 1985, a private radiotherapy institute moved to new premises taking the ^{60}Co teletherapy unit but leaving in place the ^{137}Cs unit without notifying the licensing authority, the Brazilian National Nuclear Energy Commission (CNEN), as required by law. The old building was partly demolished. As a result, the ^{137}Cs source became insecure.

On September 13th 1987, two people entered the premises, found some scrap materials of potential value, removed the source assembly from the radiation head, took it home, and tried to dismantle it. After they partly dismantled the equipment, they were severely exposed to external gamma radiation with localized burns to their bodies. One later had to have an arm amputated as a result of this exposure.

The equipment was then sold to a junkyard where the caesium chloride source was ruptured. Since it is highly soluble and readily dispersible, it caused environmental contamination and external and internal exposure of several persons. The ^{137}Cs source caught people's attention since it glowed blue in the dark and was passed from hand to hand. Remnants of the source, housing, and the assembly were sold for scrap to a second junkyard where additional source fragments were distributed to more persons attracted by the source's glow. This proceeded for five days, by which time a number of people were showing symptoms of acute hematopoietic syndrome arising from their exposure to radiation from the source.

Since the symptoms were initially not recognized as being due to irradiation, vomiting and diarrhea did not cease with over-the-counter drugs. Some looked for hospitals and were treated as carriers of infectious diseases. As a result, there were no improvements to their conditions.

On September 28th 1987, one of the victims, who was the wife of the first junkyard owner, connected the illness with the source and took part of it to the public health department, locally called “Vigilância Sanitária”, in a public bus. She told the staff there that “it is killing my family”. She had an estimated whole-body dose of 5.7 Gy and an estimated intake of 100 MBq. She died on October 23rd at 38 years of age. Her action started a chain of events which led to the discovery of the accident.

8.3 Response and Mitigation

Right after the source was delivered on September 28th, a local physicist was the first to assess the source at the health department with a scintillometer (a dose rate meter with a scintillation detector highly sensitive to radiation) that had been lent to him. It had a fast response time and a dynamic range of $0.02 \mu\text{Gy h}^{-1}$ to $30 \mu\text{Gy h}^{-1}$, being normally used for geological measurements. After switching on the monitor, it immediately deflected full scale irrespective of the direction in which he pointed it. He assumed the meter was defective and fetched a replacement, which also showed similar measurements. He took action on his own initiative to evacuate the area. At the same time, the local physicist informed the police, the fire brigade and the regulatory authority, which is CNEN (1).

The speed and the scale of the response were impressive. People were screened at the Olympic Stadium (football stadium) and the supposed contaminated people were separated. The more severely irradiated and/or contaminated patients were transferred to Marcilio Dias Naval Hospital in Rio de Janeiro. The less severely irradiated people were located in a local hospital (Goiânia General Hospital). Several other sites of significant contamination were quickly identified and the residents were evacuated and transferred to the Institute for Protection of Minors (IPM). The chronology of the response and recovery efforts can be divided into three distinct phases:

Early phase: characterized by very intense efforts of medical response, shielding, surveys, and evacuation. During this phase, there was a loss of control time interval, which spanned from the source rupture date to the discovery date, namely September 13th to the 28th, 1987. After that, from September 29th to October 3rd, the initial actions by CNEN started until control was gained.

Intermediate phase: from October 3rd to 10th 1987. It was characterized by the administrative re-organization, foci decontamination, container design, logistics, and the establishment of the provisional waste site.

Recovery phase: from October 19th to December 21st 1987. Further administrative re-organization, decontamination, demolition, and restoration of houses.

During the enormous response efforts, very comprehensive area monitoring was performed which also included aerial survey by an equipped helicopter and testing of water samples from creeks and wells. As a result, more than 67 km² of urban areas of Goiânia were monitored, including schools and other public places. The road monitoring was performed using monitors mounted in cars which covered 80% of Goiânia. Over 2,000 km were monitored, including some areas outside Goiânia (3). The extensive monitoring showed that the main foci of contamination were the junkyards and surrounding residences, comprising a total area of about 1 km² (3). There was contamination in eight main foci, 45 public areas, and 142 houses. Forty-six houses had to be decontaminated and seven were demolished. Figure 8.1 was extracted from “The Radiological Accident in Goiânia” book from IAEA (1) and shows the main places where the actions took place: the Goiânia Institute of Radiotherapy (IGR) clinic, where the source was taken; the place at the 57th Street where the source was first exposed; the public health department (Vigilância Sanitária), where remains of the source

were taken; the contaminated junkyards, and the Olympic Stadium, where the population monitoring happened (1). The map scale is about 1:8,000.



Fig. 8.1: Principal sites of contamination

In addition, significant international cooperation occurred. The Brazilian authorities informed IAEA about the accident soon after its discovery, and requested assistance which was comprised by experts and equipment. During the initial phase, several experts also came through international agencies or through existing bilateral agreements.

8.4 The ^{137}Cs Source

According to IAEA (1), the source was thought to have been produced at the Oak Ridge National Laboratory facility in the United States of America in about 1970. The radioactive source was ^{137}Cs in the form of highly soluble caesium chloride salt. It had been compacted and doubly sealed within two stainless steel capsules, as shown in Figure 8.2, which was extracted from "The Radiological Accident in Goiânia" book from IAEA (1). These in turn were sealed within what is known as an international standard capsule having standardized dimensions common to most radiotherapy units. Figure 8.2 shows a cross-sectional view of a source sealed in an international standard capsule.

It was inferred that at the time of accident, in September 1987, the source had an activity of 50.9 TBq, a dose rate at 1 meter of $4.56 \text{ Gy}\cdot\text{h}^{-1}$ and a mass of only 93 grams. The sealed source was inside a rotating assembly of a teletherapy unit with an adjustable shutter. In order to produce a radiation beam, the shutter was rotated electrically to align the source capsule with the radiation aperture. Figure 8.3 shows the remains of the source placed on a chair at the Health Department courtyard in Goiânia. The dose rate at the surface was $10 \text{ Gy}\cdot\text{h}^{-1}$ and $0.4 \text{ Gy}\cdot\text{h}^{-1}$ at a distance of 1 meter (3).

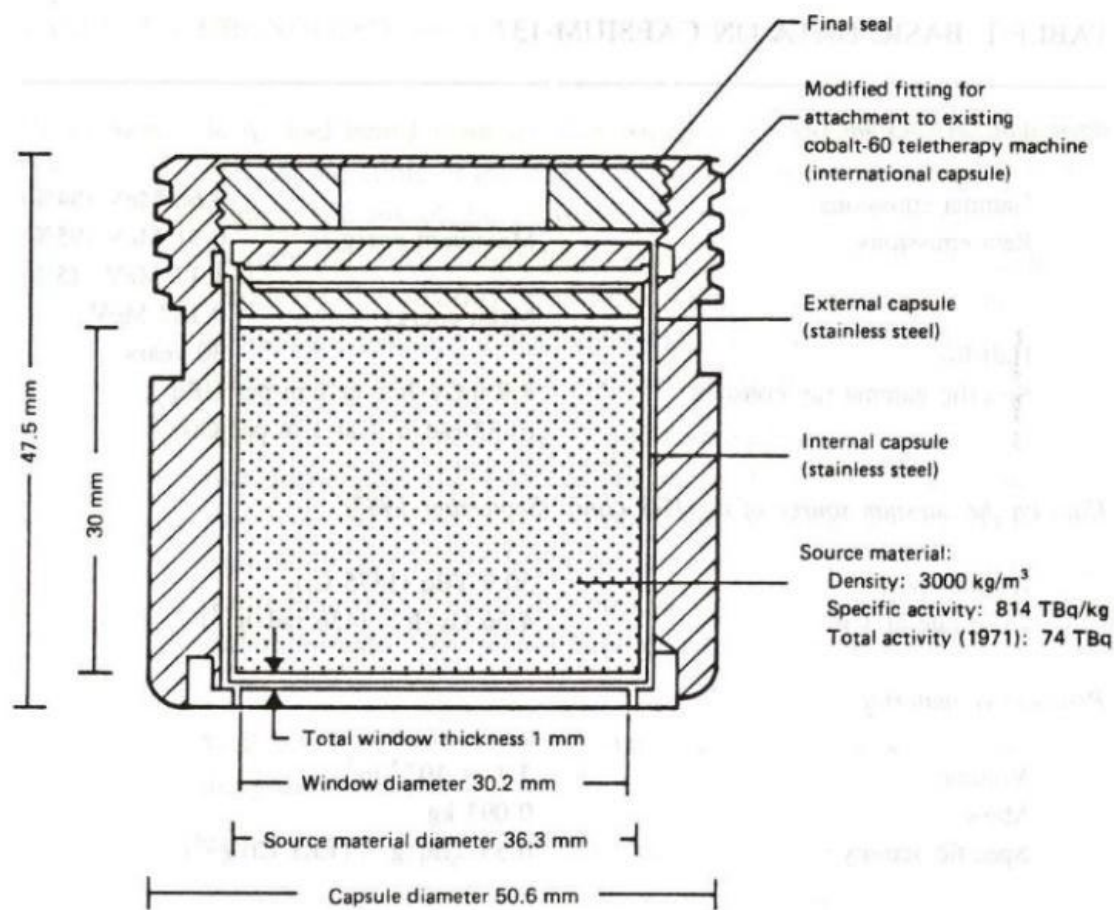


Fig. 8.2: Cross-sectional diagram of an international standard capsule. Such a capsule of radioactive caesium chloride was broken open in the accident in Goiânia (IAEA).



Fig. 8.3: The remains of the source placed on a chair at the Health Department courtyard

8.5 Individual Monitoring and Medical Care

Right after the discovery of the accident, during the early phase of the response, a population call was performed requesting citizens to go to the Olympic Stadium. This was achieved through radio and TV announcements and through cars riding through the city equipped with loudspeakers. Upon their arrival, monitoring was carried out aiming to identify the contaminated sites and the individuals whose health could be in danger due to external exposure and/or due to internal contamination. As a result, 112 000 people were monitored in a short period of time.

Individuals who presented with some contamination received decontamination and medical treatment by a medical team which was inside the Olympic Stadium. Those who required further medical assistance were transferred to hospitals. The levels of patient care were divided into three categories:

Primary care level: Performed at the dispensary of the Institute for Protection of Minors (IPM). It was comprised of patients presenting with external contamination and slight internal contamination, warranting decontamination measures impossible to implement elsewhere. Also attended by those who had their homes and properties interdicted, which added a medico-social character to this level. The importance of providing an adequate system of social and psychological support to those individuals directly and indirectly affected and to the personnel working in response to the emergency following a radiological accident was acknowledged by CNEN and stated on the IAEA publication "The Radiological Accident in Goiânia" (1).

Secondary care level: Performed at the Goiânia General Hospital. Patients with first- and second-degree local radiation injuries, or those who had received doses capable of causing a slight-to-moderate impairment (1 Gy to 2 Gy) of the hematopoietic system but who would not require special isolation measures or replacement therapy (platelet transfusions, for example). Also attended by those with moderate-to-severe internal contamination. They could benefit from ^{137}Cs removal procedures.

Tertiary care level: performed at the Marcilio Dias Naval Hospital in Rio de Janeiro. Patients with severe impairment with the hematopoietic syndrome, as well as those presenting with third-degree local skin radiation injuries.

As a result, 249 individuals presented indicative of external and internal doses, being 129 from those with conclusive external and/or internal doses. 49 were admitted to hospitals with 22 under intensive medical care. Fourteen persons were transferred to the Marcilio Dias Naval Hospital in Rio de Janeiro. Four people died, and one person had the forearm amputated. Forty-six individuals underwent decorporation therapy using Prussian Blue (ferric ferrocyanide, PB), known as Radiogardase, in dosages ranging from 3 to 10 Gy d⁻¹ for adults and adolescents and 1 to 3 Gy d⁻¹ for children. PB reduced the committed absorbed dose by a factor that varied between 1.7 and 3.4 for children and adolescents and between 2.1 and 6.3 for the adults, apparently being more efficient in adults. In reality PB reduced the half-time of Cs in the body to approximately 26 d, independently of the age and weight of the individuals, and of the PB dosages (7).

Figure 8.4 shows the flowchart representing the steps for the individual monitoring for emergency response. According to IAEA,(1) the hormone-like drug, granulocyte macrophage colony stimulating factor (GM-CSF), was used in the treatment of overexposed persons, with questionable results.

During the first two months, the individual monitoring for internally contaminated persons was performed solely by in vitro bioassay which comprised urine and faecal samples which were sent to

the Institute of Radiation Protection and Dosimetry (IRD) in Rio de Janeiro for analysis. It was also the most appropriate bioassay method during the first days because of the external contamination. Cytogenetic dosimetry was also employed. The blood samples were also sent to IRD. An improvised whole-body counter was installed at the Goiânia General Hospital in November 1987. It was comprised of a 20 cm x 10 cm NaI(Tl) detector with a 5 cm Pb shielding. The detector was positioned at 2 m from the floor due to the very high level of contamination of the persons. The system proved to be adequate, having a minimum detectable activity of 9 kBq for 2 minutes of counting. Long-term in vivo and in vitro monitoring were needed in order to evaluate the efficacy of PB in reducing the radiation dose.

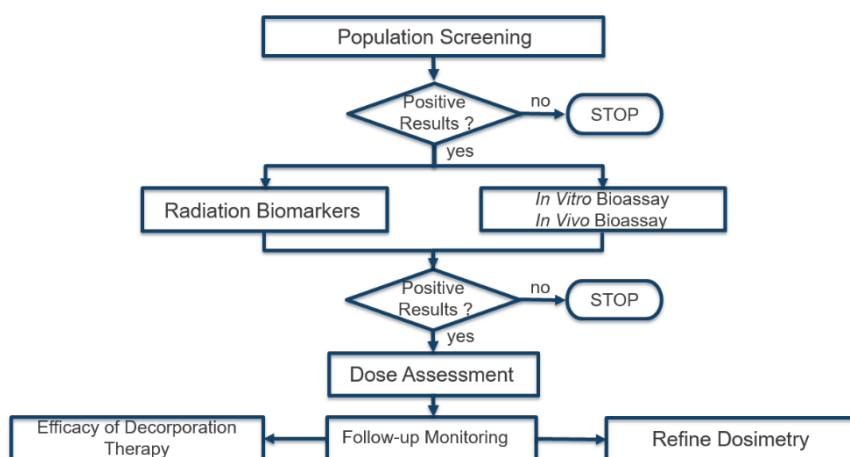


Fig. 8.4: Flowchart of the individual monitoring for emergency response

Figures 8.5, 8.6 and 8.7 show the dose distribution for the patients, for the rest of the monitored citizens, and for the workers who were involved in the cleanup operations, respectively. Amongst the 1 009 workers involved in the cleanup 77% (772 workers) of them received doses below 1 mSv. The highest worker dose was 43 mSv.

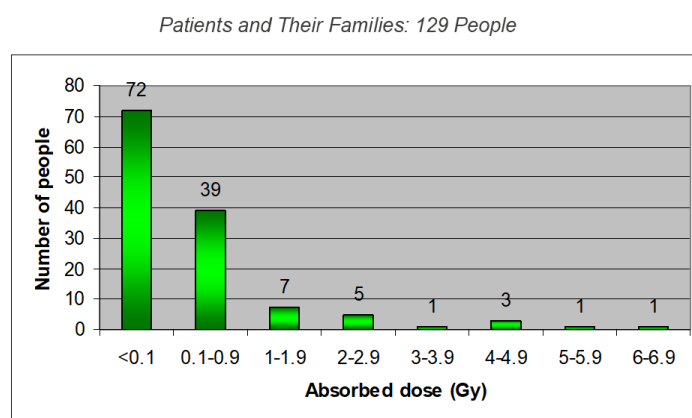


Fig. 8.5: Dose distribution for the patients and their families

Monitored Population: 810 Individuals

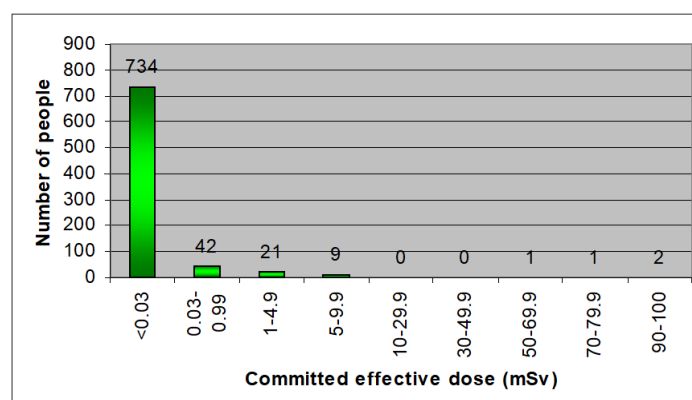


Fig. 8.6: Dose distribution for the monitored population

Workers involved in cleanup operations - 1009 workers

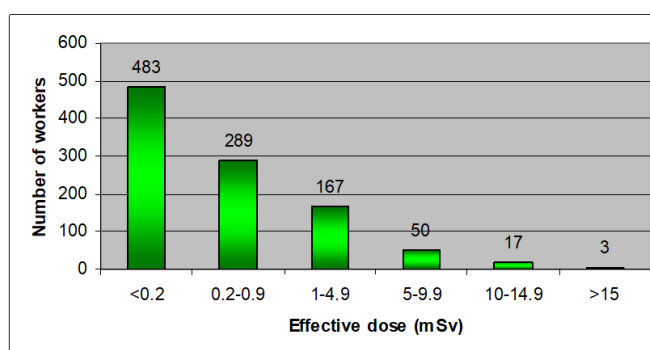


Fig. 8.7: Dose distribution for the workers involved in cleanup operations

Figure 8.8 shows a database scheme used for screening, legal, and epidemiological purposes. It contained five main parts which aimed to provide information to the dosimetric records. The database was organized by the Institute of Radiation Protection and Dosimetry, which is under the umbrella of the Brazilian Regulatory Authority (CNEN). Lately, the database was transferred to the Brazilian Ministry of Health (5), which was responsible for including the health effect outcomes of the Goiania cohort and also responsible for its administration (6). It was comprised of:

- Screening Measurements: measurement methodology, type of detector, results, date of measurement, additional observations.
- Bioassay Measurements: type of bioassay, sample information, measurement information, data of collection for in vitro bioassay, date of measurement, corresponding results and additional observations.
- Dose Assessment Biomarkers: biomarker assay, type of sample, date of collection, results.
- Committed Dose Assessment: type of bioassay, collection date, route of intake, biokinetic model, dosimetric model, committed dose estimate.
- Decorporation Therapy: successive bioassay data, dose calculation applying individual retention parameters, perturbed biokinetic model, committed dose estimate.

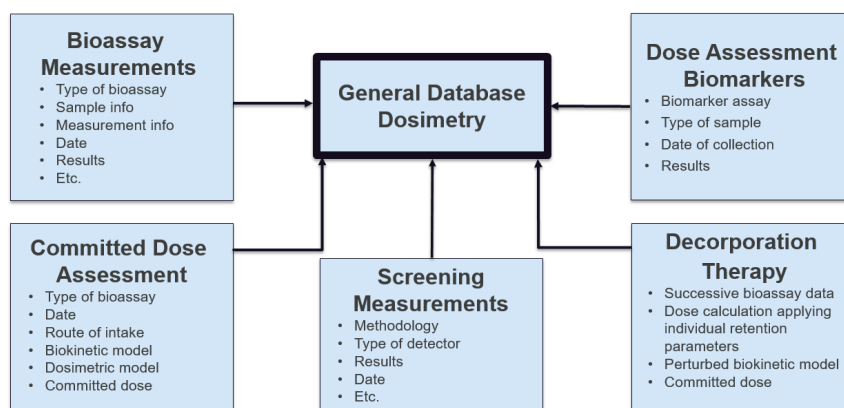


Fig. 8.8: Database scheme used for screening, legal and epidemiological purposes

There were some difficulties regarding the registry, dosimetry, and follow-up aspects. There were persons who didn't have a numbered ID such as a Social Security Number. There were also people with similar names showing different levels of contamination, which caused confusion. However, since the number of people exposed to radiation was small, all these problems were sorted out. Non-adults presented with much smaller anatomical characteristics than those specified in the corresponding reference phantoms used in dose calculations. Long-term monitoring proved to be extremely useful to evaluate the efficacy of Prussian Blue, as well as to evaluate when the treatment should be stopped.

8.6 Importance of Long-term Monitoring

Monitoring should be continued during the period that decorporation therapy is applied. Long-term monitoring will allow individual retention parameters to be obtained and applied on the dose assessment. It will also minimize the uncertainty of the committed dose assessment, especially if decorporation therapy is applied. Figure 8.9 shows the final calculated committed absorbed doses for 12 adult males, 3 adult females, 4 adolescents and 6 children, as shown on the horizontal axis, applying the individual retention parameters, where PB refers to the administered Prussian Blue (7). The dose considering the Prussian Blue treatment tends to be smaller than that without the treatment for the same individual. The same can be observed when comparing the doses using the customized individual model parameters (Actual parameters (PB treatment)) against those using the standard parameters of the biokinetic model (Preliminary Model).

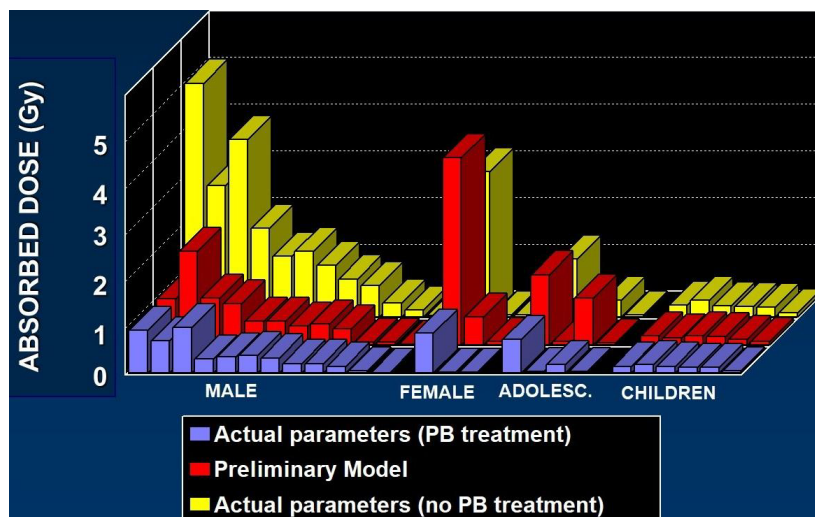


Fig. 8.9: Calculation of committed absorbed dose applying the individual retention parameters

Long-term monitoring will give the opportunity to obtain additional information that should be recorded in the registry for further epidemiological studies. Another very important outcome of long-term monitoring, in the particular case of the Goiânia accident, was the determination of a much longer retention term associated with the retention of caesium in the skeletal muscles, which did not exist in the previous caesium biokinetic model. The biological half-life is around 500 days. This term, together with the inclusion of the retention of caesium by the red blood cells, were incorporated in the updated caesium biokinetic model which was released by ICRP in its publication 137 [4].

8.7 Caesium Retention in Pregnant Woman

Table 8.1 shows two cases of caesium retention in pregnant women (2). In the first case, the caesium intake occurred during the fourth month of pregnancy while in the second case it occurred three years and eight months before the onset of the pregnancy. The individual intake in the first case was smaller than that of the second case. However, it can be seen that the caesium concentration in the infant belonging to the first case is much higher than that of the second.

	Cs Intake – 4 th month of pregnancy		Cs Intake – 3 y and 8 months before pregnancy	
	Body burden (Bq)	Cs concentration (Bq/kg)	Body burden (Bq)	Cs concentration (Bq/kg)
Mother (initial BB)	61,087	912	300,000	4,545
Mother (~ 4 y later)	-	-	9.77	132
Infant	3,885	971	0.037	10
Placenta	377	919	<MDA ^a	-

Table 8.1: Two cases of caesium retention in pregnant women (IAEA-TECDOC 1009, 1998)

It was observed that when the caesium intake occurred during pregnancy the elimination of caesium from the body was accelerated. The biological half-time in the mother's body was estimated as 45 d, which is about 25 % lower compared to that for non-pregnant woman (7). At this time, the woman has elevated levels of oestrogen, progesterone, and aldosterone. In addition, there is a rapid growing tissue mass and an increased metabolic rate.

The data in Table 8.1 allows us to conclude that the transfer factor from mother to foetus is dependent of the caesium available in the blood. For the case of the caesium intake during pregnancy the transfer factor from mother to foetus equals 1. The caesium concentration in the foetus is comparable to that in the mother and in the placenta, indicating an easy and homogeneous transport from mother to foetus.

On the other hand, for the second case where the intake occurred three years and eight months before the onset of the pregnancy the transfer factor equals 0.08. In this case, the pregnancy occurred during the period when the long-term Cs retention was predominant (biological half-life of 475 days). The caesium clearance was influenced by the subcellular fraction retention in the skeletal muscle tissue, in other words, caesium was not available in the blood to be transferred to the foetus.

8.8 Final Remarks

The importance of long-term monitoring was demonstrated in this publication. It would be unimaginable to perform appropriate dose assessments without the necessary amount of data. Moreover, the efficacy of a decontaminating agent can only be determined with a reasonable number of data points. In addition, it would have been a lost opportunity not to be able to obtain important pieces of information such as the discovery of a longer-term in caesium retention by the

body and the behaviour of caesium in the different types of pregnancies. Most importantly is that the dosimetry is the basis for any epidemiological study, the long-term monitoring is crucial to lower the uncertainty of the dose estimate, and subsequently lower the uncertainty of the risk estimate for any health effects.

8.9 References

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9 How useful is the dose assessment for the management of an irradiation accident?

J.F. Bottollier-Depois, I. Clairand, C. Huet and F. Trompier, Health Division, ASNR, France

9.1 Introduction

In an event of an irradiation accident, the dose can be the main indicator of injury of tissues and organs that helps physicians to evaluate the radio-induced damage. The purpose of a dosimetric evaluation in a radiological accident is to estimate, as accurately as possible, the dose received and its distribution throughout the organism so that the medical team can fine-tune the diagnosis and prescribe the most suitable treatment. The medical treatment strategy may be very different depending on the exposure situation, such as for localised or global irradiations at high doses. In most cases, the dosimetric data are complementary to those produced by clinical investigations (ICRU, 2019).

As each irradiation accident is specific, such an evaluation needs to consider various parameters characterizing the exposure situation: type of source and radiation, energy distribution, exposure duration, scenario, *etc.* Furthermore, the information available is (very) partial in general.

So, that is why the strategy adopted to assess the dose and its distribution is based on complementary approaches and means:

- Clinical observations: symptomatology;
- Physical dosimetry: dose reconstruction (experimental & numerical techniques) and retrospective dosimetry on materials collected from the victim or its environment;
- Biological dosimetry based on cytogenetics: study of DNA misrepairs.

9.2 Main dosimetric techniques used

9.2.1 Numerical simulations (Huet et al., 2009)

The principle of dosimetric reconstruction of a radiological accident using numerical simulations is, firstly, to model the person exposed to radiation using a numerical anthropomorphic phantom in the accident environment and, secondly, to calculate the dose absorbed at different points in the organism using a radiation-material interaction Monte Carlo computer code. For instance, the Monte Carlo code MCNPX is widely used in dosimetry. The victims are described using voxel phantoms or mathematical anthropomorphic phantoms. The environment is modelled based on the scenario: global/localised irradiation, shielding, geometry of the source, *etc.* Figure 9.1 is an example of a simulation using a voxel phantom with a ^{60}Co source at one meter at chest level.

The reliability of the dose evaluation depends directly on the accuracy of the data related to different positions of the victims in relation to the source and the duration of the exposure for each of these positions.

More details on dosimetric reconstruction using numerical tools are provided in Chapter “*Contribution of computational dosimetry to the management of radiological accidents*” of this report.

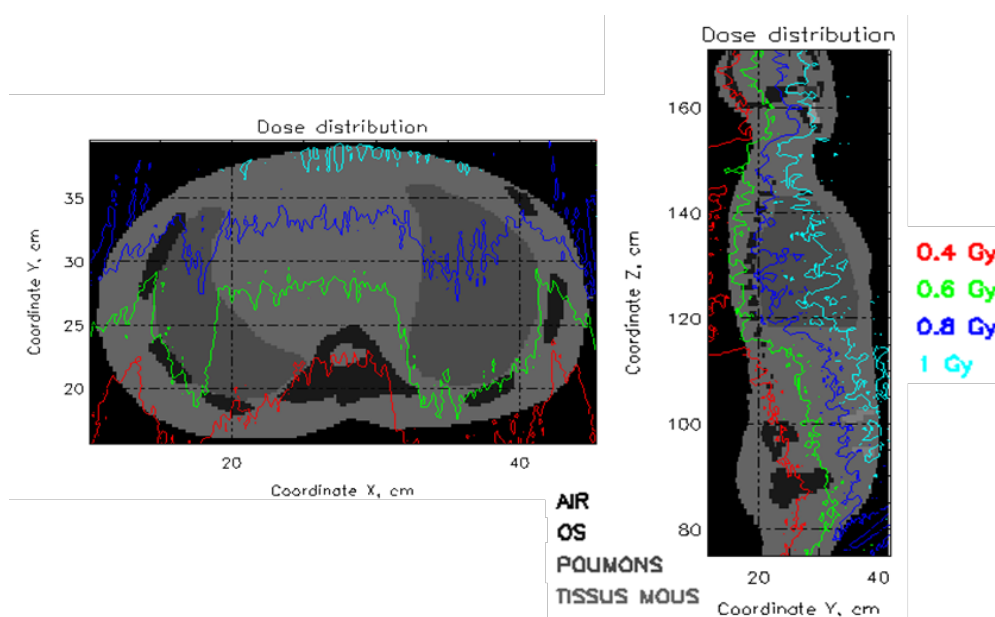


Fig. 9.1: Dose distribution using a voxel phantom with a ^{60}Co source at 1 meter at chest level.

9.2.2 Electron paramagnetic resonance (ICRU, 2019)

Electron paramagnetic resonance (EPR) is a spectroscopy method used for the observation of the absorption of a microwave by unpaired electrons placed in a magnetic field. Interaction between ionising radiation and the material generates excitation and ionisation phenomena of the atoms and molecules from which it is made, leading to the formation of free radicals. The quantity of free radio-induced radicals is proportional to the dose received in the material. There is a wide range of materials that can be measured by EPR, and various materials on the victim or in his environment may be used (tooth enamel, nails, sugars, glasses, polymers, *etc.*).

The main biological materials used in EPR dosimetry are calcified tissues, such as bone tissues, tooth enamel and dentine. Radio-induced radicals in calcified tissues are extremely stable, for example with lives in the order of 10^9 years for tooth enamel.

9.2.3 Biological dosimetry by cytogenetics (ICRU, 2019)

Exposure of a cell to ionizing radiation can cause breaks in the chromosomes present in the nucleus of that cell. In the case of an exposed person, the peripheral blood lymphocytes can be sampled for a dosimetric analysis. Within the nucleus of these cells, part of chromosomes from different radiation-induced breaks can be met by mistake, thus constituting a chromosome rearrangement. The frequency of rearrangements is related to the nature of the radiation source, the intensity of the irradiation and the dose rate. For instance, the IRSN laboratory method utilizes blood samples to cultivate lymphocytes for 48 hours in order to induce their division. The division is then blocked by colchicine in the first stage of metaphase. These metaphases are then spread over glass slides, stained, and observed under the microscope at high magnification. Scoring of chromosomal aberrations is visually done on hundreds of metaphases (usually 500).

The reference dose effect calibration curve used was obtained by counting dicentric and centric rings in blood samples exposed in vitro to homogeneous and acute gamma radiations from a ^{60}Co source with a dose rate of $0.5 \text{ Gy} \cdot \text{min}^{-1}$ (Figure 9.2). However, the direct relationship between

dicentric frequency and the dose received by an individual may vary according to the quality of the radiation and the dose rate.

The uncertainty of the result is a function of the uncertainty of the reference curve (adjusted relative to the experimental points) and the uncertainty of the observed frequency of dicentrics. The uncertainty values of the frequency are plotted on the high and low calibration curves, which gives the lower and upper confidence interval of the dose (Figure 9.3).

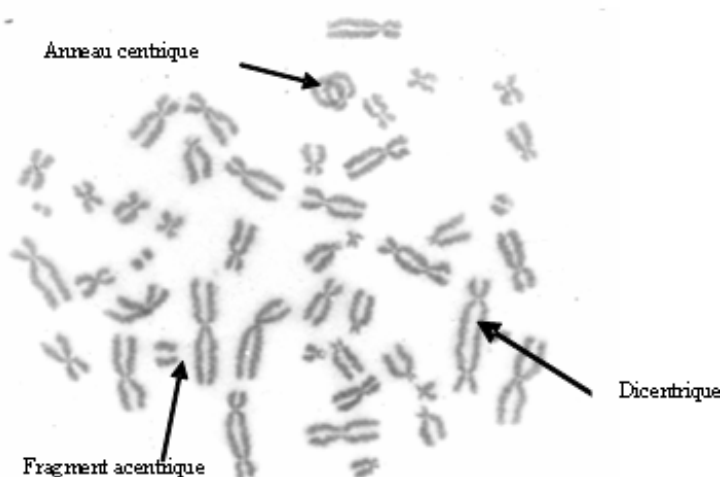


Fig. 9.2: Lymphocyte in metaphase irradiated at 5 Gy (^{60}Co , $0.5 \text{ Gy} \cdot \text{min}^{-1}$).

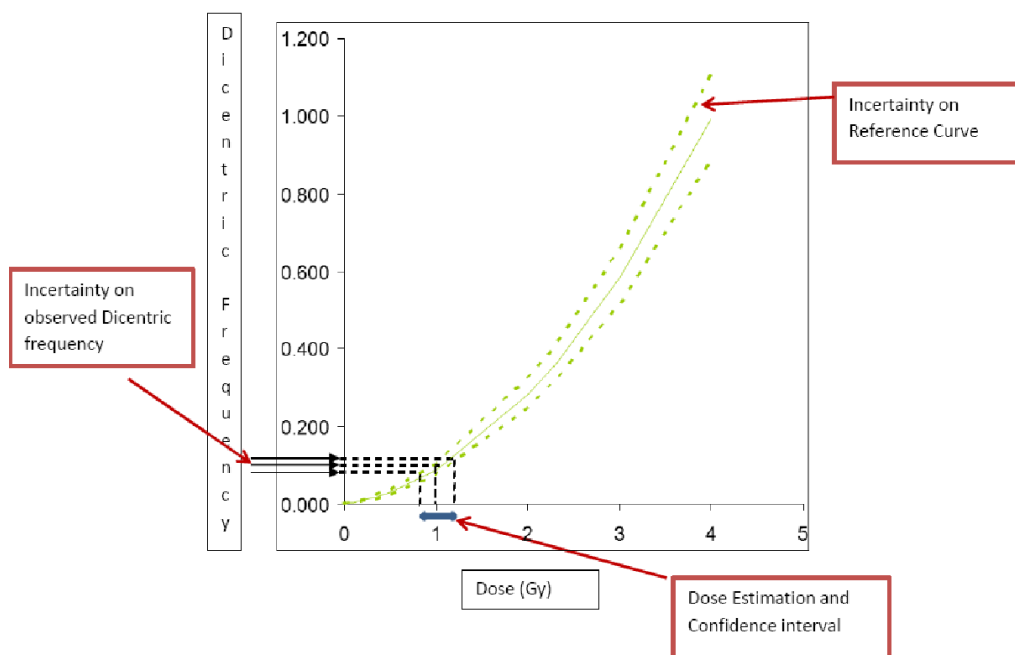


Fig. 9.3: Uncertainty of the dose estimation by biological dosimetry by cytogenetics.

9.3 Case reports

Three case reports are presented: localized irradiation, whole body irradiation, and a report with both irradiation scenarios.

Chile (December 2005) – localized irradiation (IAEA, 2009)

The context of the accident is the following:

- *Place and date:* building site of a manufactory, Nueva Aldea, Chile, December 15, 2005;
- *Context:* a worker found a ^{192}Ir source from a gammagraphy device. He handled it with his bare hands and put it in the back left-hand pocket of his pants before it was detected by someone with an electronic dosimeter;
- *Source characteristics:* ^{192}Ir Iridium, $3.3 \cdot 10^{12}$ Bq (90 Ci);
- *Irradiation characteristics:* 10-minute exposure in the back left-hand pocket of his pants and suspicion of localized exposures at the level of the buttock, hand, head and torso;
- The IAEA appointed IRSN to investigate on site;
- The victim was transferred to France on December 29th, 2005, for treatment at the Percy Military Hospital, France.

The medical situation entailed 1) anticipating a potential necrosis at the buttock level and the need for appropriate surgery (i.e. remove the correct quantity of tissue to avoid the spread of necrosis) and a graft of mesenchymal stem cells (MSC). Indeed, tissue seems healthy in appearance during the first days or weeks but can finally lead to a necrosis if the dose is higher than 25 Gy, the limit for necrosis and, 2) evaluating the doses to other areas of the body.

The first objective of dosimetry was to provide the 25 Gy isodose to guide the surgery. In such a case, a geometry that is as realistic as possible needs to be used. In practice, a personalized voxel phantom of the buttock area was created using CT images of the patient and the source was placed at back left-hand pocket level (Figure 9.4). The dose distribution was calculated, in particular the 25 Gy isodose (Figure 9.5). The size of the piece of tissue to be removed at the buttock level (Figure 9.6) was half of a sphere: 10 cm in diameter and 5 cm in depth (Lataillade *et al.*, 2007).

The second objective was to assess the dose at the head level using the EPR technique on teeth. The dose range was 5.6 Gy (tooth 2-4) to 8.5 Gy (tooth 4-5). Even if such doses are very high, the relevant information was that no necrosis was expected at the head level.

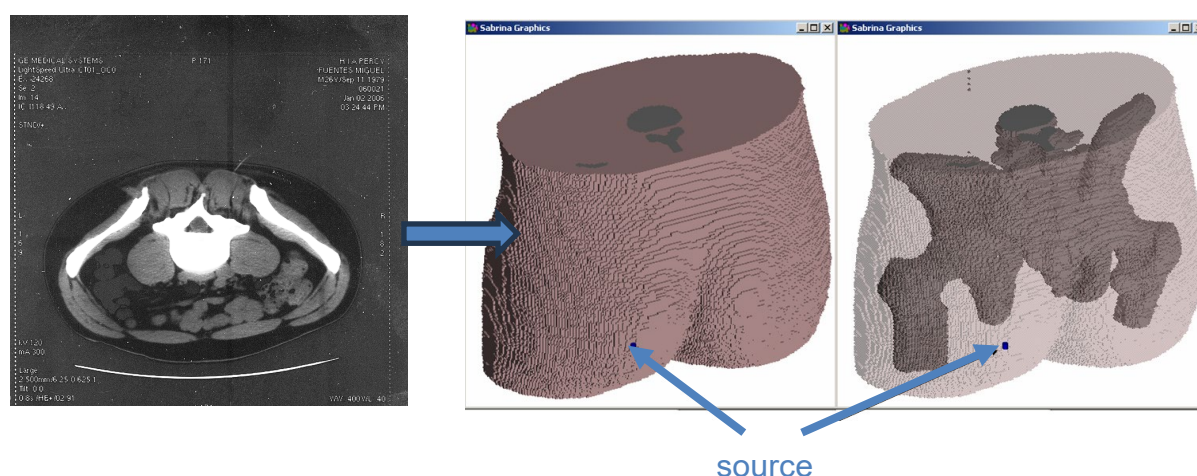


Fig. 9.4: Geometry used for the dose reconstruction using CT images of the patient (left) to build a voxel phantom (right).

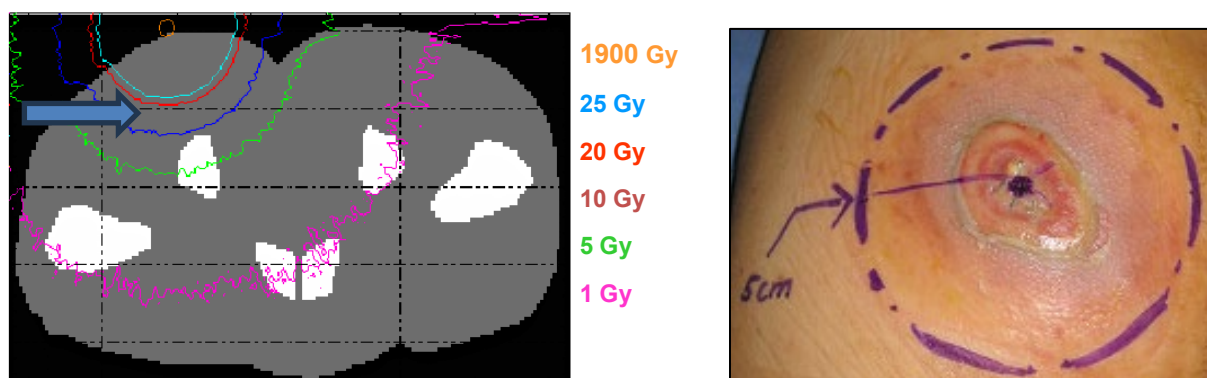


Fig. 9.5: Dose distribution (left) and piece of buttock to be removed (right).



Fig. 9.6: Surgery (left) and 10 months after the MSC and skin grafts (right).

9.3.1 Belgium (March 2006) – whole-body irradiation (Huet et al., 2008)

The context of the accident is the following:

- > *Place and date:* industrial irradiation facility, Fleurus, Belgium, March 11th, 2006
- > *Context:* a worker went inside the irradiation cell when the source was not in the storage position;
- > *Source characteristics:* cobalt-60, 30 000 TBq (800 000 Ci);
- > *Exposure characteristics:* whole body irradiation with a duration of 20 s;
- > The patient was in medullary aplasia and some bone marrow areas received high doses;
- > The patient was transferred on March 31st, 2006, to the Haematological Department of the Percy Military Hospital, France.

As it was a whole-body irradiation with potentially a high dose, it was necessary, from a medical perspective, to assess the dose distribution at the level of the bone marrow. Indeed, the objective was to find out if some parts of the bone marrow were “under-exposed”, typically below 6 Gy, leading to a possible spontaneous secondary resumption of its activity in these areas.

Two approaches were used to assess the dose distribution, in particular in the different bone marrow areas:

- > An on-site dosimetric reconstruction, in conditions as close as possible to those of the accident, using a tissue-equivalent anthropomorphic phantom fitted with dosimeters and positioned in the irradiation room (Figure 9.7);
- > By Monte Carlo calculations using a mathematical phantom (Figure 9.8).

The AP dose gradient is about a factor of 2, both at the pelvis and the skull levels. Furthermore, the fact that some bone marrow areas are clearly below 6 Gy suggests a spontaneous secondary resumption of the bone marrow activity in these areas.

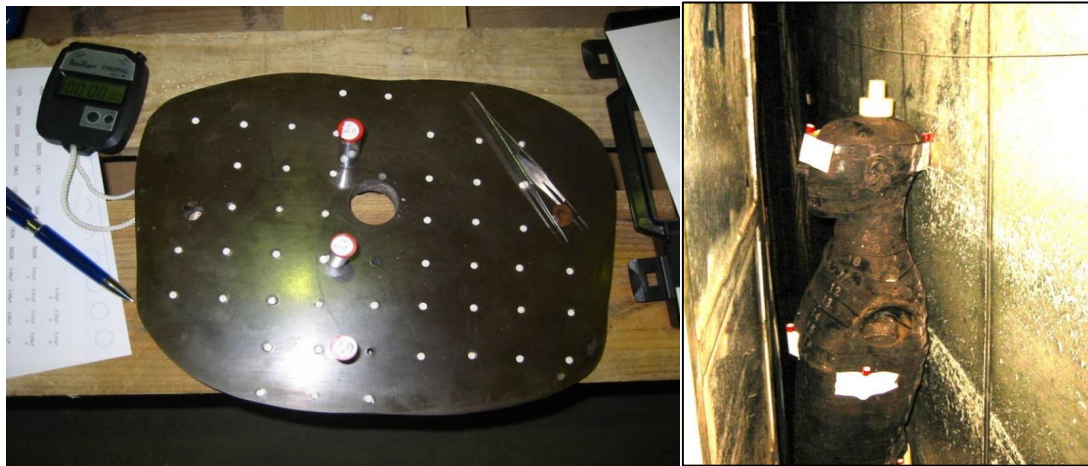


Fig. 9.7: Phantom slide with dosimeters (a) and phantom in the irradiation facility in a position representative of the victim's position (b).

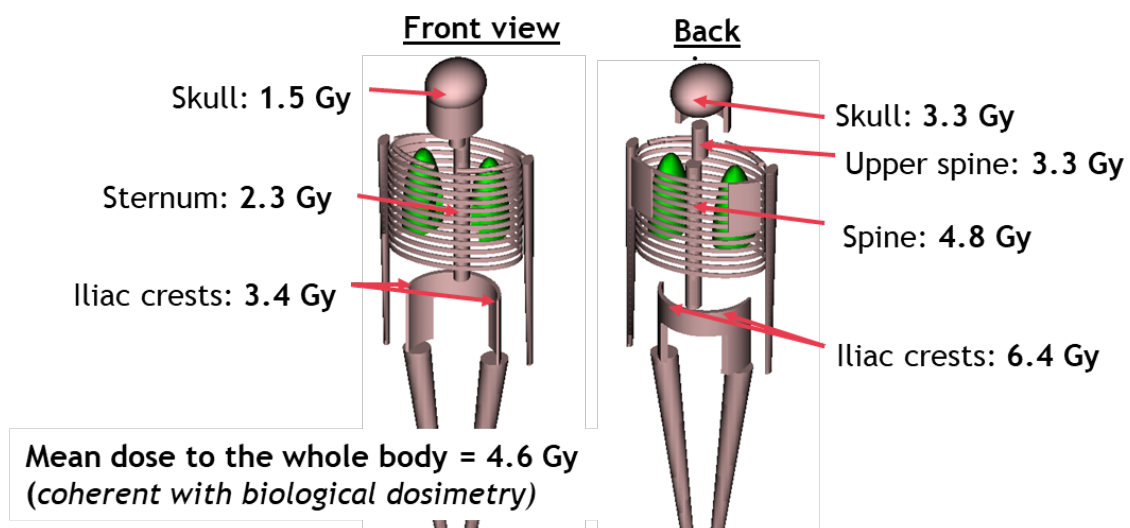


Fig. 9.8: Dose calculation using a mathematical phantom.

9.3.2 Peru (January 2012) – Localized and whole-body irradiations (IAEA, 2018)

The context of the accident is the following:

- *Place and date:* hydroelectric company in Chilca, Peru, January 12, 2012;
- *Context:* 3 workers were conducting in-situ gammagraphy activities in several pipes and the source was not in the lead container but somewhere in the flexible cable;
- *Source characteristics:* Iridium-192, 3.6 TBq (98 Ci);
- *Exposure characteristics:* localised to the hands and whole body for a duration of about 2.5 hours.

Taking into account the testimony of the operators and regarding the clinical observations and the initial dosimetric data from the personal dosimeters, the dose was very high and heterogeneous, both to the whole body and to the hands. Such an exposure can be explained only if we consider a global exposure in the vicinity of the source for several hours and a very localised exposure (finger(s) in contact with the collimator for a very short time).

The medical consequences could be necrosis of the fingers requiring an amputation and a hematopoietic syndrome due to the dose received in the bone marrow areas.

Dosimetry was used in a supporting role to estimate doses at the level of the finger(s) and dose heterogeneity in the body. In addition to the individual passive dosimeter, different complementary techniques were used: calculation, biological dosimetry, EPR dosimetry on nails and teeth.

The calculations showed that the mean distance between the source and the victim was roughly 40 cm during the 2.5 hour irradiation scenario (based on the testimony) with the dosimetric data at whole body level and hands obtained by measurements (Table 9.1). The EPR technique was used for the first time in fingernails and the results are displayed in Figure 9.9, showing that the amputation of one finger could not be avoided.

Table 9.1: Dose estimations with various techniques at different location in the body

<i>Dose in Gy</i>	<i>Mean trunk dose</i>	<i>Tooth</i>	<i>Chest</i>	<i>Hands</i>
<i>Measurements</i>	2.5 to 3.5*	4**	6 to 7***	~35**** (nails)
<i>Calculations</i>	2.8	3.2	6	20 (hands)

* *Biological dosimetry*

** *EPR dosimetry – average between two teeth*

*** *Individual passive dosimeter (from dosimetry lab in Peru)*

**** *EPR dosimetry – average among nails (total dose: global + localised irradiations)*

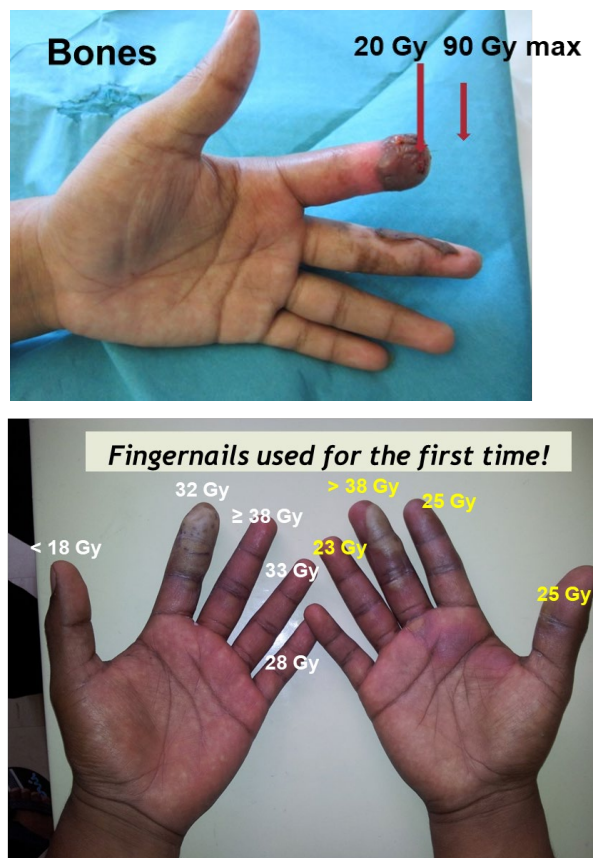


Fig. 9.9: Dose assessment at the finger level using EPR on fingernails.

9.4 Conclusion

The main questions to be answered regarding medical management in the event of irradiation accidents depend strongly on the exposure configuration, localized or whole body:

- > *Localized irradiation*: necrosis; surgery/amputation needed; skin or MSC grafts...?
- > *Whole-body irradiation*: hematopoietic syndrome; bone marrow affected...?

To address such questions, the dose assessment and its distribution in the body is a key issue to get the best possible diagnosis and to define the most appropriate therapeutic strategy.

In practice, complementary techniques are needed depending on the accident configuration, such as biological dosimetry, calculations using phantoms, retrospective dosimetry (EPR on human samples, etc.)

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10 Case studies on the current nuclear risk in Europe

Johan Camps, Pieter De Meutter, Belgian Nuclear Research Centre (SCK CEN) 2400 Mol, Belgium

10.1 Abstract

Nuclear risk is, in general terms, the product of the probability of occurrence of a nuclear event and the consequences or impact of such an event. With the invasion of Ukraine by Russia, the nuclear risk in Europe has increased and studies of the nuclear risk are important in context of nuclear emergency preparedness. To evaluate the consequences of an event happening in Ukraine, at larger distances, at a scale of typically 1000 kilometres, two scenarios are evaluated: a nuclear accident in one of the four active nuclear power plant sites in Ukraine and a nuclear weapon explosion in Ukraine. Results of the Lagrangian stochastic particle model Flexpart are used to evaluate the impact at larger distances. For both cases it is observed that the main consequences are limited to a potential contamination of the food chain in specific areas, depending on characteristics of the events and the meteorological conditions at the time of the event. For the nuclear power plant accident events, probabilities to exceed typical food contamination levels are presented. For a nuclear weapon detonation an example case is used to discuss potential consequences emphasizing impact distant to the impact area and effect of the prompt blast. The detectability of such events in the typical early warning networks installed across Europe is also mapped out and discussed.

10.2 Introduction

With the invasion of Ukraine by Russia on 24 February 2022, in an escalation of the Russian-Ukrainian war which began in 2014, Europe is confronted with the biggest attack on a European country since the Second World War. This has significantly increased the Nuclear and Radiological risk in Europe and beyond.

Before the invasion, nuclear and radiological risks in Europe have been mainly attributed to the civil use of nuclear technology, terrorism targeting existing nuclear installations and the malevolent use of ionising radiation and the nuclear weapon programmes in the post-cold war era. With the invasion of Ukraine, Russia brought its war to the Chernobyl, Zaporizhzhia and other nuclear sites in Ukraine, violating international protocols², and having the potential of affecting more European countries. The nuclear infrastructure of Ukraine consists of 15 operable reactors on 4 sites (Rivne, Khmelnytskyi, South Ukraine and Zaporizhzhia) and the Chornobyl site with the in 1986 destroyed reactor number 4 and the 3 others in permanent shutdown. Efforts by the International Atomic Energy Agency to secure these plants have been initiated and are on-going, however the risk level remains concerning. IAEA formulated seven indispensable pillars for ensuring nuclear safety and security during an armed conflict ('Seven Pillars')³ which are derived from and are aligned with the IAEA safety standards and nuclear security guidance. At all nuclear sites in Ukraine at least part of the seven pillars are impacted. On the Zaporizhzhia nuclear power plant site, the largest plant in Europe with 6 reactors, all 7 pillars are impacted. Probably even more of a concern are Russia's thinly veiled threats to use nuclear weapons in context of the conflict. These threats seem also not to be limited to Ukraine, an increased importance of Russia's nuclear deterrent in for example the Arctic waters of the far north is reported.

² E.g. GC(53)/DEC/13 of the IAEA General Conference of September 2009.

³ <https://www.iaea.org/topics/response/nuclear-safety-security-and-safeguards-in-ukraine>, consulted 09/07/2024

With the war in Ukraine, it is generally perceived that Europe faces an increased nuclear risk as discussed above. However, risk is generally defined as the product of the probability of occurrence of an event and the consequences or impact of the event. This paper focusses on the impact of a nuclear event on Europe for two specific scenarios. The first scenario is the impact of a nuclear accident –triggered directly or indirectly by the Russian invasion- in one of the four nuclear power plants of Ukraine. The second scenario describes a typical nuclear weapon used on the territory of Ukraine. For both scenarios we mainly focus on the potential impact at larger distances, typically on Europe outside Ukraine.

10.3 Case one: The impact of an event at a NPP in Ukraine on Europe

10.3.1 The source term

To calculate the impact of a potential nuclear accident scenario, independent of the initiating event, the amount and time evolution of radioactive material released to the environment, i.e., the source term, should be known. In general, the source term depends on many factors such as the physical radioactive inventory of an installation at the moment of the event, the different barriers and leak factors (e.g., loss of containment) and the presence and efficiency of systems for mitigating releases to the environment (sprinklers, filters, etc.). Due to differences in the physico-chemical properties of the forms in which radionuclides can be encountered, the ratio between the radioactive inventory and the quantity of different radionuclides released is not at all constant. In this context, often groups of radionuclides are considered having similar physico-chemical properties and behaviour in the installation and the environment, such as noble gases (no chemical interactions, e.g., xenon and krypton isotopes), iodine isotopes in molecular or organic form, beta-gamma aerosols (e.g., caesium isotopes but also iodine bound in an aerosol) and alpha-aerosols (e.g.; plutonium isotopes).

Instead of calculating full detailed accident scenarios, we assume in this study the impact as a function of the released source term for three radionuclide groups/specific radionuclides. These groups, described in detail in [1], cover past nuclear events (Three Mile Island, Chernobyl, and Fukushima) and include caesium-137 (^{137}Cs ; half-life of 30 years), elemental 131iodine-131 (^{131}I ; half-life of 8 days), and a passive air tracer representative of radioactive noble gases such as ^{133}Xe (^{133}Xe ; half-life of 5.2 days) and xenon-135 (^{135}Xe ; half-life of 9 hours). The focus on ^{137}Cs is motivated by the fact that this radionuclide is abundantly present in a nuclear reactor, both when the reactor is in operation and long after its shutdown. It is important for the long-term consequences in case of an accident. The focus on iodine and the radioactive noble gases ^{133}Xe and ^{135}Xe is determined by the fact that these radionuclides are abundantly present in case of failure of a nuclear reactor in operation and are more volatile, so a larger fraction of the “material at risk” can be released into the environment. However, when half-lives are much shorter the consequences to diminish over time when reactors have been shut-down for a period of time. The range of released quantities considered in this study are between 0.1 PBq and 100 PBq for ^{137}Cs , 2 PBq and 2000 PBq for ^{131}I and 10 PBq to 10 000 PBq for noble gases (A petabecquerel, PBq, is 10^{15} Bq).

10.3.2 Calculation methodology

As our study focusses on the impact at larger distances, the use of an atmospheric transport and dispersion model and weather data representative for this scale is of utmost importance. The Lagrangian stochastic particle model Flexpart [2], coupled with archived numerical weather data from the European Centre for Medium-Range Weather Forecasts provided by the Royal

Meteorological Institute of Belgium was used. For a given release of radionuclides, the potential impact and the affected area largely depend on the weather conditions at that time. Therefore, to get an idea of the potential impact of a release at any time, it is necessary to consider a range of possible weather conditions. This was achieved by randomly selecting 500 release start times between 24 February 2022 and 24 February 2023, corresponding to the first year of the invasion of Ukraine by Russia. Simulations were started for each of these selected times. An atmospheric release of radionuclides was modelled at each of the four Ukrainian nuclear power plants. Each release was assumed to have a duration of 6 hours. This is a rather short release and therefore conservative assumption in terms of impact, as a more protracted release would result in a wider spread of the source term.

A total of 6000 Flexpart simulations have been performed and analysed (500 release times x 4 facilities x 3 radionuclide species).

10.3.3 Results

While direct countermeasures such as sheltering, stable iodine prophylaxis (for the inhalation exposure pathway related to iodine releases), and/or evacuation can be advisable depending on the source term and meteorological conditions in an area of up to several tens of kilometres, the analyses in this study focused on the larger scale impact. This larger scale impact mainly relates to radioactive deposition and its effect on the food chain. For completeness, results are also reported for the potential detection of the radioactive cloud in the dose rate stations of the early warning networks across Europe [3].

An example result of one Flexpart simulation is shown in Figure 10.1. It shows the total deposition of Cs-137 on 09:00 UTC 16 August 2022 after a 6 h release of 1 PBq of ^{137}Cs at the Rivne nuclear power plant starting at 09:00 UTC on 12 August 2022.

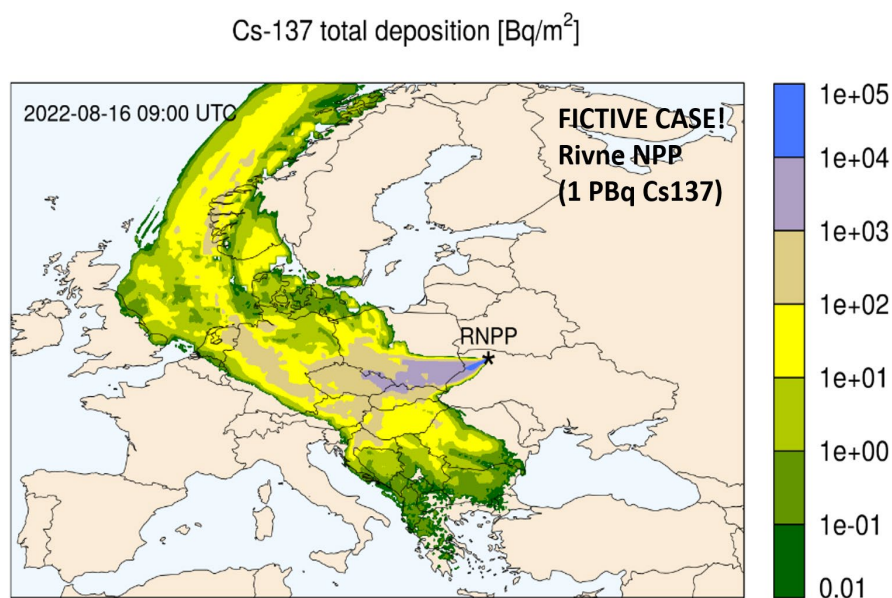


Fig. 10.1: Example of one simulation and one type of output, showing the deposition of ^{137}Cs after a release of 1 PBq at the Rivne NPP. Deposition levels are shown down to 0.01 Bq/m², which are levels far below any health standard, including food and feedstuff (for more details see text).

To assess the potential impact on the food chain, the total deposition is considered, which is the sum of dry and wet deposition. If the total deposition exceeds a chosen threshold of $10\,000\text{ Bq/m}^2$ of ^{137}Cs , it is assumed that countermeasures are required for the food chain. This threshold corresponds to a reasonable derived value for the deposition for which certain food products can be contaminated to a level above the maximum permitted levels of radioactive contamination (for caesium and iodine isotopes) of food and feed following a nuclear accident⁴. It is not completely excluded that certain food products can be already contaminated above the permitted levels with somewhat lower deposition levels and that, during a real event, permitted food contamination levels are further reduced. This happened in the aftermath of the Fukushima accident for food from Japan. However, the 10 kBq/m^2 can be seen as a level at which potential food contamination will be an important concern.

Based on 500 simulations, with start of release randomly selected over the first year of the Russian invasion, Figure 10.2 shows the probability of exceeding this predefined threshold of 10 kBq m^{-2} ^{137}Cs total deposition for the Zaporizhzhia NPP. For ^{137}Cs releases of 1 PBq and below, the probability that the level of 10 kBq m^{-2} will be exceeded outside Ukraine is extremely low. For larger releases the probability that the level will be exceeded increases. For a release of 10 PBq ^{137}Cs from one of the four active NPPs in Ukraine, the probability that the level will be exceeded in a neighbouring country can be up to 10%. Especially for the Zaporizhzhia NPP the analysis shows that due to dominant wind directions in that region, the probability distribution is spatially much more extended to the east compared to the west.

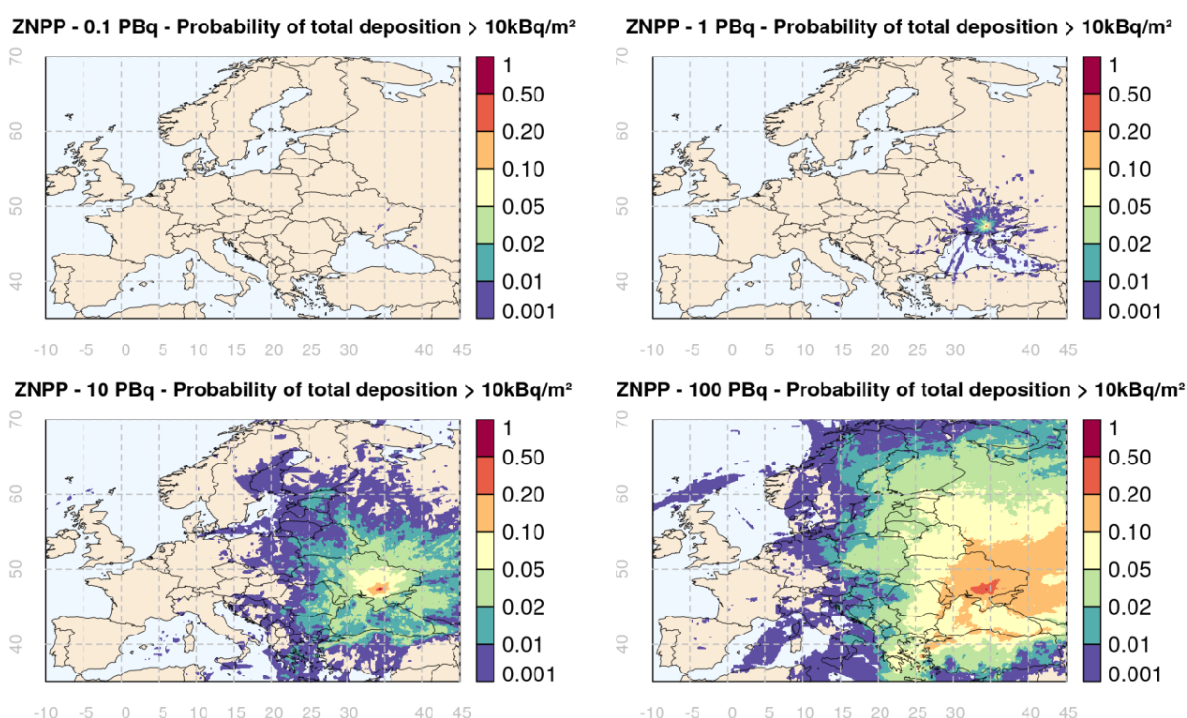


Fig. 10.2: Probability that the total deposited ^{137}Cs exceeds a threshold of 10 kBq m^{-2} following a hypothetical release from the Zaporizhzhia NPP in Ukraine.

Although the inventories of the NPPs contain more than 100 PBq ^{137}Cs , this quantity is larger compared to the amount of ^{137}Cs released during the Chernobyl accident (around 89 PBq). Releases

⁴ <https://eur-lex.europa.eu/EN/legal-content/summary/maximum-levels-of-radioactive-contamination-in-foodstuffs.html>, consulted on 09/07/2024

of iodine isotopes, such as ^{131}I can be higher. Consequently, areas in which the 10 kBq/m^2 value of ^{131}I will be exceeded will be larger, but due its half-life of 8 days this contamination will disappear relatively soon after an accident. Further it has to be noted that for the Zaporizhzhia NPP, all reactors have been shut down for months, making that short- lived radionuclides (such as the iodine isotopes) have decayed. Additional results for the different sites including also hypothetical iodine releases can be found in [1].

Using Rivne NPP as an example, Figure 10.3 shows an example of the probability of a detection of a plume by the dose rate stations of the early warning networks installed in European countries as a function of the released source term of noble gases. Data collected by 39 participating countries are shared in near real time through the EUropean Radiological Data Exchange Platform (EURDEP) [3]. Some 5000 stations reporting ambient dose equivalent rates ($H^*(10)$) on an hourly to daily basis are included. As a threshold for detection the level of 10 nSv h^{-1} above natural background is chosen. Background variations, e.g., when it starts raining, can be much larger. We focussed on the noble gas releases here, because they are due to their inert character the first to be released in case of an accident, potentially in the largest quantities and are not depleted in the atmosphere due to deposition processes. Probabilities become relatively high of detecting it in large areas across Europe, outside Ukraine. The analysis shows that also for the Rivne NPP the probability distribution is more extended towards the East in comparison to the west.

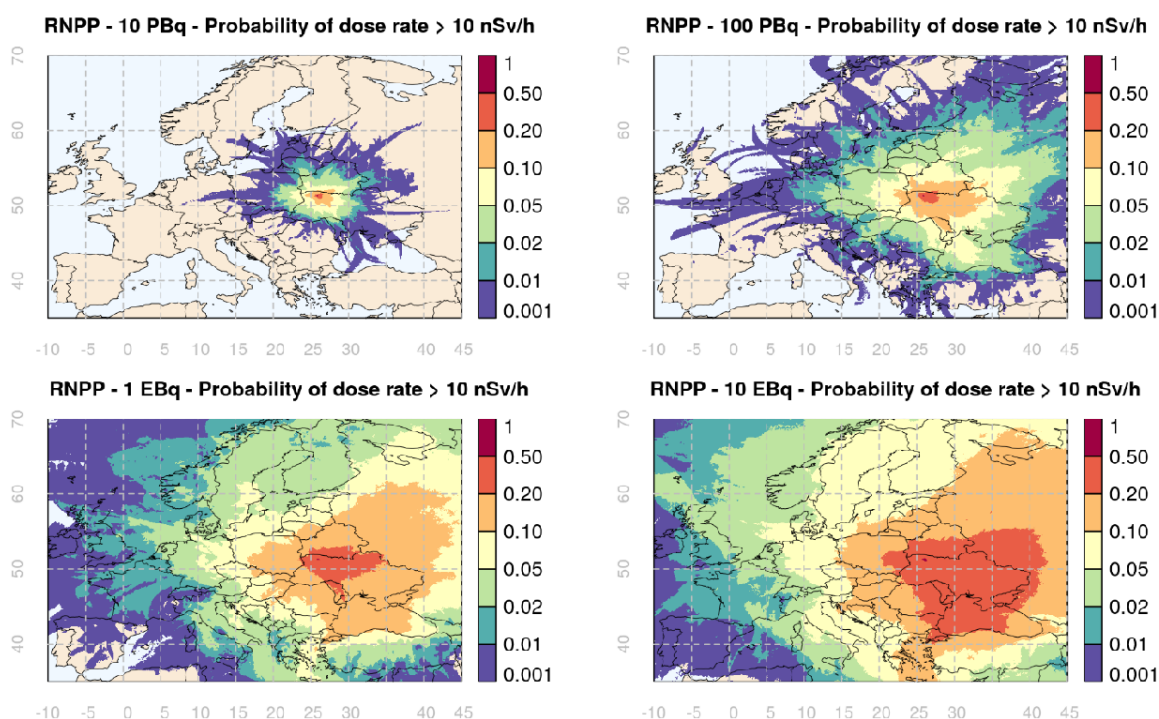


Fig. 10.3: Probability that the gamma dose rate due to noble gases exceeds a threshold of 10 nSv h^{-1} following a hypothetical release from the Rivne NPP in Ukraine.

Although, in case of an accident with a substantial release of radionuclides, direct countermeasures are not very probable to be taken outside Ukraine, food countermeasures could be required in large parts of Europe. Based on the simulations also the mean area affected (over all simulated weather conditions) could be determined. A $10 \text{ PBq } ^{137}\text{Cs}$ release would result in a mean area exceeding 10 kBq m^{-2} of around $65\,000 \text{ km}^2$. For a $200 \text{ PBq } ^{131}\text{I}$ release the mean area would be around $543\,000 \text{ km}^2$. For substantial releases of noble gases also detections in the dose rate stations of the early warning network outside Ukraine are likely. It should be noted that for deposition levels of

10 kBq m⁻², used as a value for discriminating when food countermeasures are likely, ambient dose equivalent rates would be around 30 nSv h⁻¹ above their natural background.

10.4 Case two: The impact of a nuclear explosion event in Ukraine on Europe

Estimates of operational nuclear weapons stockpiled by Russia are available via open literature including reports from the Stockholm International Peace Research Institute [4]. From the 4500 nuclear weapons that Russia stockpiles, approximately 1600 are operational, 2900 are in reserve and another 1500 are awaiting dismantling. The typical yield is estimated to be in the range of 10kT to 800 kT. Russia did the nuclear explosion test with the highest yield, known under the name Tsar Bomba, having an explosive yield of 50 MT. 1 kT is 1000 ton TNT equivalent and corresponds to an energy of $4.2 \cdot 10^{12}$ joule or $2.6 \cdot 10^{25}$ MeV. For comparison, non-nuclear (chemical) explosives have yields up to a maximum of 11 ton TNT. Weapons can be of different nature: fission, boosted fission and fusion or thermonuclear. However, all contain a nuclear fission part. Delivery systems of nuclear weapons bombs is generally referred to as the nuclear triad: from land with short range to intercontinental ballistic missiles (ICBM), from sea with submarine launched ballistic missile (SLBM), and from air with aircrafts with nuclear capability.

The effects of nuclear weapons are well described in the literature. A fireball immediately develops after the detonation, which is many times more brilliant than the sun and can cause blindness up to large distances. A 50 kT device yield, in less than one second will grow to its maximum diameter of 840 m, and after one minute will cool down and lose its glow. A near surface explosion will form a debris-filled mushroom cloud. The cloud top will reach an altitude of around 12 km for a 50 kT explosion. The fission and activation products will, depending on their volatility, condensate in particulates in a range of sizes going from sub micrometers to millimeters. These particulates will be primarily transported by the wind, however especially the larger-size particles will gravitationally settle on the ground in the neighbourhood, while the smaller ones and volatile elements will be transported over longer to even very long distances. Size distribution and consequently fall-out is largely depending on the amount of debris mixed into the cloud and this depends largely on the location of the explosion. The energy liberated during the explosion is shared as follows: 50% goes to blast and ground shock, 35% to thermal radiation and 15% to ionizing radiation, from which 5% prompt radiation (first minute) and 10% delayed in the form of radioactivity created and partially present in the fall-out (decaying with half-lives typically of minutes to years).

While the direct effects, such as potential effects from the blast, thermal effects and prompt radiation are largely determined by the yield, the radioactive fall-out is strongly depending on the size distribution of the radioactive particles. In general, the effects are described for a near-surface explosion, i.e., an explosion at a limited altitude above ground zero. This has the potential for the largest consequences: no shielding for shock and thermal waves and also no (outdoor) or limited shielding (indoors) for the prompt radiation. Due to a large amount of debris mixed with the radioactivity created during the explosion, the particle distribution will contain many larger particles which increase the fallout. As an example of such a scenario Figure 10.4 shows the effects of a near-surface detonation above Brussels. It is clear that such an explosion can make in a densely populated area a huge number of victims.

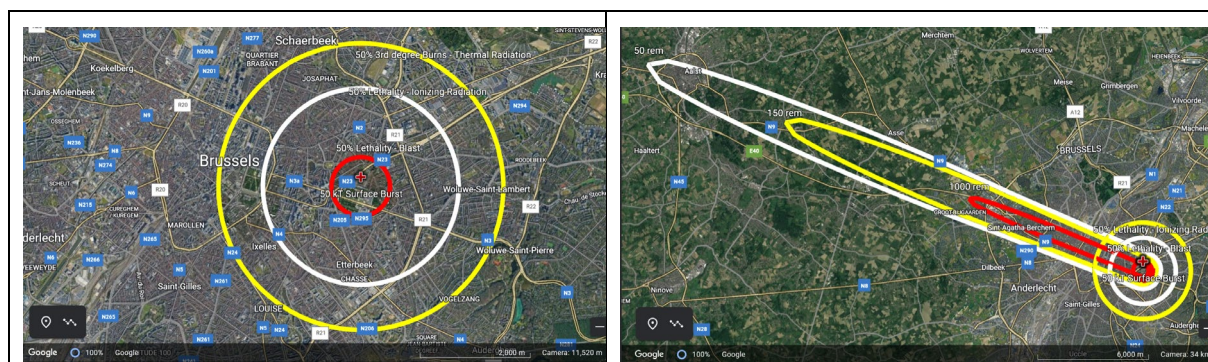


Fig. 10.4: The effects of a near-surface detonation of 50 kT. Left side show some of the direct effects: 50% lethality blast (red circle, radius 0.53 km), 50% lethality prompt ionizing radiation (white circle, radius 1.79 km, no shielding) and 50% 3rd degree burns (yellow circle, radius 2.61 km). Right side includes also the dose from fallout (accumulation for 6 hours) with a south-easterly wind, showing contours for 10 Sv (red, 6.7 km²), 1,5 Sv (yellow, 15 km²) and 0.5 Sv (white, 47 km²). Calculation performed using HOTSPOT [5].

However, as in the first case, the aim of this study is to analyse the impact and potential consequences on the larger scale (typically 1000 kilometres) of a nuclear explosion in Ukraine. For this a Flexpart calculation was set-up to study the transport and dispersion of ¹³¹I. In a nuclear explosion fission of the uranium or plutonium material takes place over a very short time. This is different from a nuclear reactor where fission goes on continuously. This means that the fission products are mainly short-lived, yielding very high dose rates in the nearby fall-out. Iodine isotopes are produced in large quantities in a nuclear explosion and ¹³¹I with a half-life of 8 days can pose important challenges for the food chain, especially for milk products, meat and fresh vegetables. As discussed, particles sizes and consequently deposition and depletion of the radioactive cloud, largely depend on the conditions of the explosion. To study the effect, a simulation was done for 3 particle sizes, the amount of ¹³¹I produced in a 50 kT nuclear explosion (fission), for a specific location (ground zero near Lviv) and selected moment in time (12 August 2022), representing meteorological conditions that bring the radioactive cloud in the direction of Western Europe. The radioactivity was assumed to be distributed along the typical mushroom cloud based on literature data [6], with the cloud top typical for a 50 kT explosion (altitude 12 km). Figure 10.5 shows the results of this calculation for 3 particle –sizes, i.e., 0.6 µm, 6 µm and 60 µm.

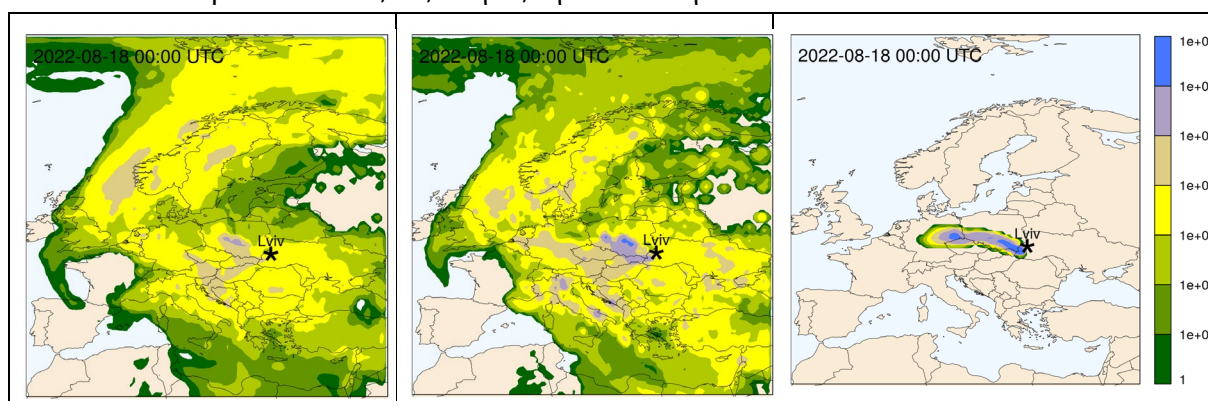


Fig. 10.5: Deposition levels of ¹³¹I calculated for a hypothetical 50 kT detonation near Lviv for different particle sizes (left: 0.6 µm, middle 6 µm and right 60 µm). See text for more explanation.

Deposition seems to spread over large parts of Europe in the scenario under consideration. To evaluate the potential impact on the food chain at larger distances due to iodine deposition, the threshold deposition level of 10 kBq m^{-2} as used in our first case study can be applied here. This value is exceeded in the light brown, purple and blue areas on the maps on Figure 10.5. For particle sizes of $0.6 \mu\text{m}$ and $6 \mu\text{m}$, areas exceeding 10 kBq m^{-2} extend from Ukraine via neighbouring countries up to areas in Norway. For larger particle sizes ($60 \mu\text{m}$) the range to which the level is exceeded reduces significantly, but deposition values in the more nearby area are much higher. As a conclusion from this calculation, it is clear that depending on characteristics of the explosion, the food chain can be affected up to distances of 1000 kilometres and more away from the explosion. This was also observed in some of the atmospheric nuclear explosion tests, for example due to rain-out of radioactivity in Tahiti at a distance of 1200 km from the test-site Mururoa in French Polynesia on 17 July 1974. The population received an estimated dose of 0.8 mSv [7]. Although this is rather limited, contamination levels of certain food products such as milk would require food countermeasures in Europe.

As a remark, another effect, i.e., an electromagnetic pulse initiated by the prompt ionizing radiation in the atmosphere and having the potential to impact electronics and communication lines, could have a range of thousand and more kilometres from the point of explosion, depending on the explosion height. This effect is mainly important at larger distances if the explosion is at high altitude.

10.5 Conclusions

With the invasion of Ukraine by Russia the nuclear risk in Europe is believed to have increased. In this paper we concentrate on the impact of a potential event and have considered two cases. The first is a nuclear power plant accident directly or indirectly associated with the war in one of the four operating nuclear power plants. The second is a nuclear weapon detonation in Ukraine. The consequences of the two cases are in the surroundings of the event totally different and have been studied in detailed and described in literature. Here, however, we focus on the impact at larger distances of the order 1000 kilometres). Similar for both cases is that, depending on the precise characteristics of the two types of events and the meteorological conditions following the event, contamination of the food chain at large distances (order 1000 kilometres and more) with short-lived ^{131}I for which countermeasures are needed cannot be ruled out. At large distances contamination with long-lived radionuclides, such as ^{137}Cs is rather improbable for nuclear weapon events, while possible for nuclear power plant accidents. Also in both events, again depending on conditions, signals in radiation and radioactivity networks can be potentially measured in certain areas at large distances. The study performed is in line with experience from past accidents and atmospheric nuclear testing. Studies like this are important in context of preparedness, allowing to take the necessary steps in case of a real event to follow-up the situation, also at large distances, in a sound and robust way.

10.6 References

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