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Radiation protection – science, safety and security

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SSM – Swedish Radiation Safety Authority
NRPA – Norwegian Radiation Protection Authority
SIS – National Institute of Radiation Protection, Denmark
GR – Icelandic Radiation Protection Institute
NKS – Nordic Nuclear Safety Research
Radiological Society of Finland
EC – European Commission
IAEA – International Atomic Energy Agency
OECD/NEA – Organisation for Economic Co-operation and Development/Nuclear Energy Agency
WHO – World Health Organization
UNSCEAR – United Nations Scientific Committee on the Effects of Atomic Radiation
ICRP – International Commission on Radiological Protection
ICNIRP – International Commission on Non-Ionizing Radiation Protection
ICRU – International Commission on Radiation Units and Measurements
ICRM – International Committee for Radionuclide Metrology
EFOMP – European Federation of Organisations for Medical Physics
ILO – International Labour Organization

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European low dose risk research strategy

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Abstract

Although much is known about the quantitative effects of exposure to ionising radiation, considerable uncertainties and divergent views remain about the health effects at low doses. Many of the European states have national research programmes in this area, however, beyond the Euratom research programme, little has been done to integrate these programmes in the past. The European High Level and Expert Group (HLEG) recently made a motion for the establishment of a Multidisciplinary European Low Dose Research Initiative (MELODI) that would create a platform for low dose research under a jointly agreed Strategic Research Agenda (www.hleg.de). The research agenda would focus on the key policy questions to be addressed and provide a road map for such research in the coming years and decades. The Letter of Intent expressing the will for the stepwise integration of low dose research activities was initially signed by five national organisations, BfS, CEA, IRSN, ISS and STUK. Since 2009, several other organisations have expressed their commitment on long-term integration of research and their interest to join MELODI. In 2010, a Network of Excellence called DoReMi was launched by the Euratom FP7 programme. DoReMi will act as an operational tool

for the development of the MELODI platform during the next six years. The joint programme for research focuses on the areas identified by the HLEG as the most promising in terms of addressing/resolving the key policy questions, namely: the shape of dose response curve for cancer, individual susceptibilities and non-cancer effects. Radiation quality, tissue sensitivities and internal exposures will be addressed as cross cutting themes within the three main research areas.

High Level and Expert Group on European Low Dose Risk Research

Both natural and man-made sources of ionising radiation contribute to human exposure and constitute a hazard for human health. The exposure of workers, and to a smaller extent of the public, to low levels of ionising radiation from nuclear energy production and other industrial uses of radiation have become an integral part of industrialised society. Radiation protection standards rely on current knowledge of the risks from ionising radiation exposure. Any over-, or under- estimation of these risks could lead either to unnecessary restriction or to a lower level of protection than intended. There has been a gradual decline in scientific and regulatory expertise in radiobiology and radiotoxicology during the past decades and the individual groups tend to be small. Moreover, several experimental infrastructures have been closed down during the years. For example, there is currently no place in Europe where larger numbers of animals could be exposed to radiation at low dose rate. At the same time, biomedical sciences are developing very fast and the new understanding on the molecular and systems biology should be transferred to radiation research.

To address these issues, six national funding (or regulatory) bodies with significant programmes/activities or with a policy interest in low dose risk research came together in the beginning of 2008 and Commission, created a group called “High Level and Expert Group on European Low Dose Risk Research” (HLEG). The group consisted of the representatives of the mentioned bodies, the EC as well as experts covering a wide range of disciplines in radiobiology, epidemiology, dosimetry and modelling. The European Commission provided support for the technical secretariat.

The HLEG drafted a Report that was posted in the HLEG website (www.hleg.de), opened to public consultation and published in the final version on the website at the beginning of 2009. A printed version was also published later by the European Commission (High Level and Expert Group, 2009). The Report has three parts, reflecting three objectives: a) Policy Issues, b) State of Science and research challenges, c) Proposed European research strategy. In the third part, a *trans-national organisation* is proposed considering that the complex and multidisciplinary nature of these issues is such that their resolution can be achieved only through the integration of research at a European, (or even international) level. To this aim a platform was proposed called “Multidisciplinary European LOw Dose Initiative” (MELODI), as an instrument capable to guarantee a sustainable, long-term integration of European national R&D programmes.

The HLEG Report identified topics that are important for the radiation protection policy point of view (Figure 1).

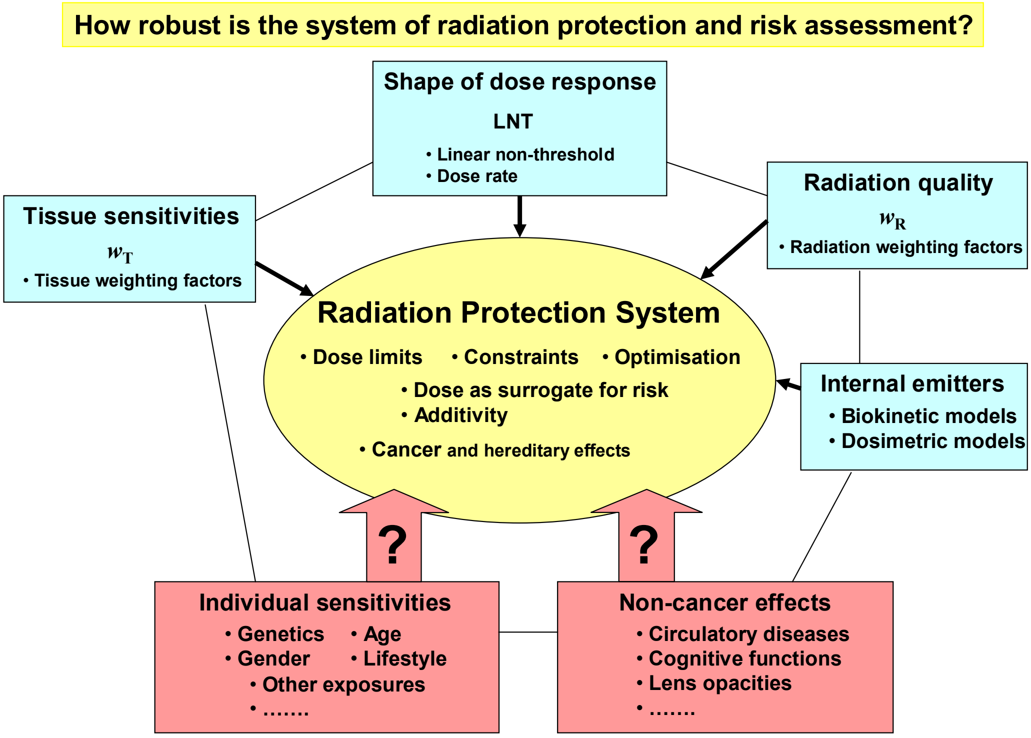

Fig. 1. The main issues where judgements are made in the current system of radiation protection. The four upper boxes denote judgements that fall directly within the main ICRP dosimetric system, while the two lower boxes include issues that are at present included only to a relatively minor degree (from HLEG Report, 2009).

MELODI Initiative

In April 2009 five of the national organisations present in the HLEG, with the support of the EC, signed a Letter of Intent (LoI) in order to create the initial core of the platform, with a view to integrate in a step by step approach EU institutions with significant programmes in the field and to be open to other scientific organizations and stakeholders. These organizations were Bundesamt für Strahlenschutz (Federal Office for Radiation Protection, BfS) - Germany, Commissariat a l'Energie Atomique (Atomic Energy Commission, CEA) - France, Institut de Radioprotection et de Surété Nûcleaire (Institute for Radiation Protection and Nuclear Safety, IRSN) - France, Istituto Superiore di Sanità (National Institute of Health, ISS) - Italy, and Radiation and Nuclear Safety Authority, STUK - Finland.

MELODI is intended as a mean to establish trans-national governance structure to integrate national, bi- and multi- lateral research programmes within an agreed SRA. Therefore, a key role of MELODI is to develop and maintain over time a strategic research agenda (SRA) and a roadmap of scientific priorities within a multidisciplinary approach, and to transfer the results for the radiation protection system. Other aims are to interface with broad range of stakeholders, to oversee investments in key infrastructures, to coordinate knowledge management and education and training and to ensure consistent research methodologies.

The MELODI signatory organisations have taken a number of initial steps in 2009 to set up a provisional operational governance and organisational structure for MELODI. The first ones were to create a MELODI Governing Board, constituted of the designated representatives of the signatory bodies (presently chaired by W. Weiss, BfS, Germany), and to produce some documents defining the process of setting up the final MELODI research platform. An introductory document setting out the key steps and the expected deliverables is available at the MELODI website http://www.melodi-online.eu/. In the beginning of 2010, four more organisations from four additional EU countries had joined the MELODI platform and several others have expressed their willingness to do so.

DoReMi Network of Excellence

The aim of DoReMi is to promote the sustainable integration of low dose risk research in Europe in order to aid the effective resolution of the key policy questions identified by the High Level Expert Group (HLEG) on Low Dose Risk Research (www.hleg.de). DoReMi provides an operational tool for the development of the proposed MELODI platform (Multidisciplinary European Low Dose Risk Research Initiative) consisting of major national bodies and research programmes that have long term commitment in low dose risk research in Europe.

DoReMi
Integrating Low Dose Research

The Joint Programme of Activities (JPA) of DoReMi includes: (i) a Joint Programme of Research (JPR) covering the issues outlined above and providing an overview of the needs for research infrastructures of pan-European interest and facilitating multilateral initiatives leading to better use and development of research infrastructures; (ii) a Joint Programme of Integration (JPI) to develop a coordinated European roadmap for the long term needs of the key players in Europe; and (iii) a Joint Programme for the Spreading of Excellence (JPSE), covering knowledge management, training and mobility and its implementation. The JPR focuses on the areas identified by the HLEG as the most promising in terms of addressing/resolving the key policy questions, namely: the shape of dose response curve for cancer, individual susceptibilities and non-cancer effects. Radiation quality, tissue sensitivity and internal exposures will be addressed as cross cutting themes within the three main research areas. A substantial proportion of the activities will be dedicated to the joint programme of research. The programme describes a multidisciplinary approach including interfaces with the broader biological toxicological and epidemiological communities. Strategic planning will be carried out in close collaboration with MELODI. The long term Strategic Research Agenda (SRA) will be developed by MELODI, whereas DoReMi research priorities are based on a shorter term Transitional Research Agenda (TRA), focusing on goals that are feasible to achieve within the 6 year project and areas where barriers need to be removed in order to proceed with the longer term strategic objectives. The work package structure of DoReMi is shown in Figure 2.

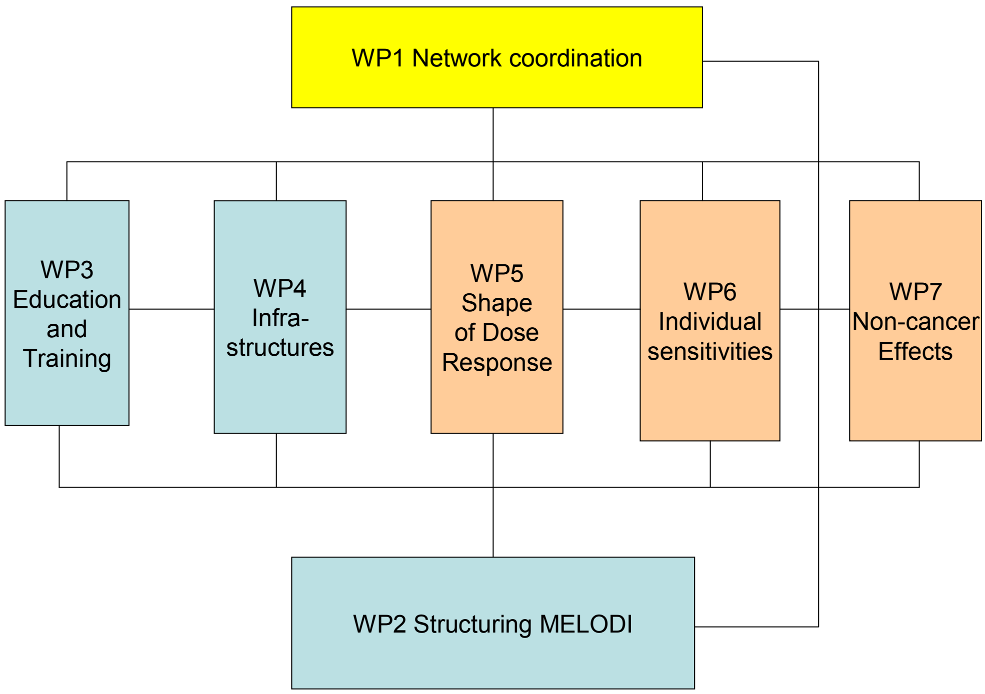

Fig. 2. The Work package structure of DoReMi Network of Excellence. Network coordination is carried out by STUK - Radiation and Nuclear Safety Authority (Sisko Salomaa). DoReMi has three main scientific work packages: WP 5 (Shape of Dose Response) is led by HPA (Simon Bouffler), WP 6 (Individual Sensitivities) is lead by Helmholz Zentrum Munich (Mike Atkinson) and WP 7 (Non-cancer Effects) is lead by IRSN (Patrick Gourmelon and Jean-René Jourdain). WP 3 (Education and Training) is led by University of Pavia (Andrea Ottolenghi) and WP4 (Infrastructures) by CEA (Laure Sabatier). WP2 (Structuring of MELODI) deals with integration activities and strategic planning and is led by IRSN (Dietrich Averbeck).

The network of Excellence started with 12 partners (beneficiaries). New partners are expected to join via open calls, searching for new competencies as well as necessary infrastructures and other platforms for research, such as suitable animal and tissue model systems for low dose risk research. These areas will be defined in more detail in the Transitional Research Agenda and Call Plan. The first open call of DoReMi is planned to take place in autumn 2010. DoReMi web site will be opened in September 2010: www.doremi-noe.net

DoReMi is the necessary complement to MELODI, aiming to achieve fairly short term results in order to prove the validity of the HLEG approach, and to contribute to the development of MELODI. Such objectives and timeframe are typically consistent with what can be expected from a NoE, which is why the EC included the possibility of such a scheme in its 2008 call. The resources invested by the EU and by member organisations in the NoE will therefore aim to:

1) develop and implement a "transition research agenda" (TRA) with a short to medium term time scale, which would attempt to capture the essence of the HLEG strategy,
2) implement the TRA within the limited time frame and organisational perimeters of the NoE,
3) contribute to the wider development of MELODI, by bringing its scientific input into the open dialogue of the Platform and by supporting it through its dedicated resources,
4) take into account the results of MELODI work on the SRA for the successive calls for R&T actions within the NoE. These open calls are expected to progressively bring in new partners into the working structures of both organisations.

The NoE will also be able to take advantage of the MELODI Scientific Committee structure in order to optimize its own independent scientific evaluation process. In practice, DoReMi Work Packages WP2, WP3, and WP4 will develop support structures for MELODI, on research infrastructure in Europe and on education and training operating in such a way that their deliverables feed directly as "draft contributions" into the working groups of MELODI. DoReMi WP5, WP6, and WP7, which are concerned with the scientific issues of the shape of the dose effect relationship, the individual radiation sensitivity and non-cancer effects of radiation, respectively, will produce scientific deliverables from their research and training operational activities, which should constitute the first operational stepping stones on the SRA roadmap. These results as well as the experience feedback from such should also contribute to the furthering of MELODI goals, particularly at the time when the NoE will be nearing its completion. For this purpose, scientists in DoReMi WP's should become also actively involved in MELODI working groups, where the ongoing Low Dose Risk research strategies will continue to be developed year after year. The complementarity of DoReMi and MELODI is illustrated in Figure 3.

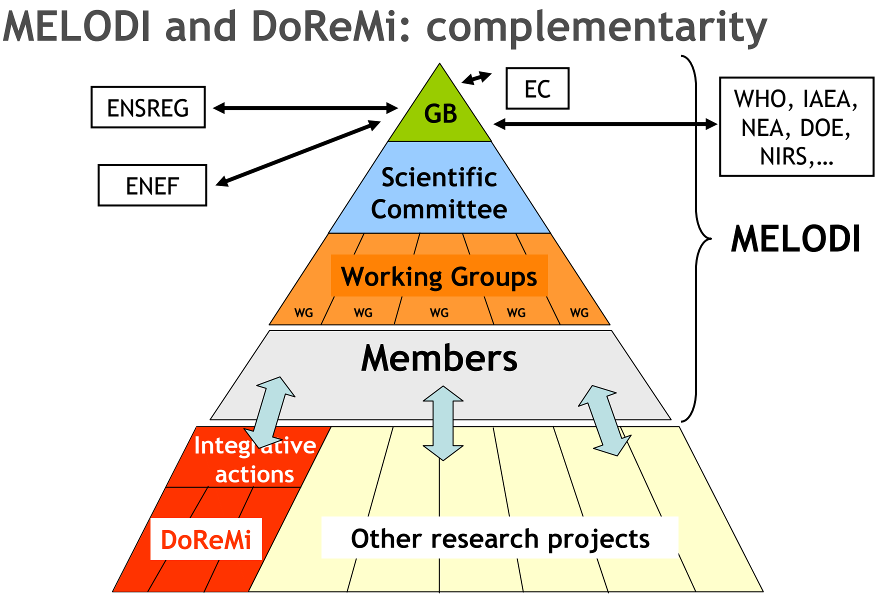

Fig. 3. DoReMi is the necessary complement to MELODI, aiming to achieve fairly short term results in order to prove the validity of the HLEG approach, and to contribute to the development of MELODI. Other research projects based on the joint low dose risk strategy are foreseen. These projects would be funded by EC or by national funding bodies.

Important fora for the scientific discussions and development of the strategic research agendas are the MELODI workshops and DoReMi exploratory workshops that engage the scientific community in low dose risk research. The first MELODI workshop was organised by BfS in Stuttgart on 28-30 September 2010, and the second MELODI workshop will be arranged by IRSN in Paris on 18-20 October, 2010.

Acknowledgements

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Public health perspective in radiological protection

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Abstract
Recent new insights about radon risk and increasing patient exposure due to medical imaging, as well as unexpected cardiovascular effects of ionizing radiation are challenging our classical radiological protection approaches and should be viewed in a larger public health perspective. Concepts as global indoor air quality, long term sustainability, risk awareness or precaution ask for a wider approach encompassing scientific vigilance, value judgements, risk communication and sociological participation in risk governance. Progress in radiological protection needs a new transdisciplinary framing of complexity where social sciences and humanities are more actively involved to complete scientific insight.

Text
Recent new insights about radon risk and the increasing patient exposure due to digitalisation of medical imaging are challenging our classical radiological protection approaches. Unexpected cardiovascular effects of ionizing radiation (at least for relatively low cumulative doses) and some other present science foci are challenging our expert culture and are questioning why the radiological protection community clarifies and embarks such health topics so late. In these important areas for *public health*, the current dose limitation system has been till recently almost not operational while justification and optimisation were not given the priority they deserve.

These – and other- health concerns regarding ionizing radiation occur in a *changing societal context* where scientific paradigms such as LNT are questioned (bystander effects), where members of the public and patients interact actively www based, where the attention is drawn on ambiguities in values and where controversial communication is fed by media polarisation. Amazing bio-monitoring opportunities confront expert beliefs, regulatory concepts and responsibilities in health care and are opening new perspectives for multi-factorial epidemiology. Dose as unique risk indicator is no longer self-evident.

Based on the new evidence, radon risk requires stricter standards. However, hygiene should be reformulated to integrate *global indoor air quality* at home as well as at work. The *collective* dose, currently threatened to be abandoned, was and is still useful for characterising the impact on public health of radon (as well as for radiology).

In the NORM area, where standard settings are currently driven by utilitarian ethics, *sustainability* checks for long term (radon) health impact should be introduced.

A more *effective* implementation of operational justification of medical exposures and the integration of optimised radiological protection in medical Quality Assurance could allow controlling the detrimental health impact. Paediatrics and pregnancy ask in this respect for particular attention. However, *risk awareness* is the major challenge for the future and gives an ethical dimension to the new digital imaging resolution opportunities.

Radiotherapy as well as other oncology practices are facing late multi-factorial secondary effects (like circulatory diseases). They require better scientific insight but also science based communication skills.

Radiological protection in the future cannot continue to refer to the virtual average individual when *susceptibility* is demonstrated.

More generally, a major issue is the capacity of taking into account, without undue delay, new relevant data emerging from ongoing research and having a potential impact on public health in situations involving a risk from exposure to ionizing radiation – as for example the new data regarding the risk of radiation induced circulatory diseases. In this context, the European Commission organises every year, in cooperation with the Group of Experts referred to in Article 31 of the Euratom Treaty, a scientific seminar (called RIHSS seminar, for "*Research Implications o*n Health Safety Standards") to discuss in depth relevant issues emerging from ongoing research. The aim is to identify the potential implications on the European Basic Safety Standards of recent research results and to take this into account *in due time*, while debating on the need of a precautionary approach, if uncertainties are present (which is often the case).

Risk communication, as difficult search for transparency (RISCOM), should not only disclose technical black boxes and messages but demonstrate a respectful approach of the different perceptions. Perception as impression of a health risk reality is not only a characteristic of lay people. The mental construct of experts needs attention as well. Ethical guidance studied by think tanks (such as PISA in Belgium) and elaborated by IRPA could help to better distinguish the common good and to manage conflicts of interests as experienced in Health Councils. IRPA recently launched stakeholder engagement in radiological protection and successful experiences with local communities open new perspectives. They are based on sociological partnership innovations, mutual respect and considerations of distributive justice in decision making. Such new risk governance processes join prudent precaution strategies as put forward by EEA and the Dutch Health Council. They could facilitate to transform public health opportunities into real perspectives. Precaution requires the systematic specification of uncertainties. This is not as evident in expert culture as scientific method requires. Moreover it takes into account ambiguities related to *value judgements*. This extends the scope of risk assessment and management. For example, in the radiological protection system, where ALARA can be considered as a precursor of precaution, an ecosystem approach is still missing.

Our scientific progress in radiological protection needs a new *transdisciplinary* framing of complexity where social sciences and humanities are more actively involved to complete scientific insight.

RPA Radiation Protection Culture initiative: An opportunity for the nuclear industry

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Abstract

Considerable progress has been made in all the fields of application of radiation protection over the past 40 years, supported by an enthusiastic generation of professionals. We are observing now significant development in the nuclear industry throughout the world and the profession is faced with renewal of the generation who thought up today's radiation protection. We must remain vigilant as all the conditions are fulfilled for radiation protection to make no further progress and in this case even to go backwards. One way of preventing or at least limiting the risk is to root radiation protection in culture, to develop a radiation protection culture. Sharing of our know-how and simultaneous optimisation initiatives set up for constant improvement of our practices are common radiation protection values. IRPA is proposing a brainstorming with other radiation protection societies with a view to possibly drawing up the IRPA's "Guiding Principles" for Radiation Protection Culture. This presentation will present the first results of the First international IRPA workshop in Paris (December 2009).

Medical management of radiation accidents

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Novel measurement and analysis technology for safety, security and safeguards

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Abstract
Nuclear interactions have been measured through molecular phenomena since the early days of radiation research. Rutherford used ZnS(Ag) to convert the kinetic energy of alpha particles into visible light in his famous experiments some 100 years ago. Recent technical advances have again made molecular phenomena attractive as a basis of nuclear instrumentation. Current detection technology has provided single photon counters, and there is no need to rely on human eyes. The present paper reviews scientific advances that may ultimately lead to operational methods for safety, security and safeguards. Examples of novel approaches and technologies are given.

1 Introduction
Radiation protection aims at the safety of individuals, society and the environment. Radiation measurements are critical tools to ensure safety and security. Recent technical advances have provided new types of instruments for daily work. One such example is the $LaBr_3$ detector, which provides sufficient energy resolution for continuous real-time radiation surveillance in the environment with a hundred times better sensitivity than the old approach based on Geiger counters (Toivonen et al. 2008). On the other hand, the ruggedized instruments make mobile spectroscopy possible for security applications directly operated by law enforcement.

The radiation risks to society are not limited to the consequences of accidents. The malevolent use of nuclear or other radioactive material is a threat to society that may be realized any day. The proliferation of nuclear weapons is another international security challenge. Key material control, based on nuclear safeguards, is the best way to prevent such an undesirable development. For these tasks, authorities need novel technology. The new safety and security threats have to be eliminated, or if realized, their consequences need to be mitigated.

Nuclear safeguards are measures to verify that states comply with their international obligations not to use nuclear materials for explosives. The objective of nuclear security is the prevention of theft, sabotage, unauthorised access, illegal transfer or other malicious acts involving nuclear material, other radioactive substances or their associated facilities. Crucial elements of the nuclear security regime are effective detection and response measures. For this, novel technology is required.

The term “novel technology” refers to physical phenomena, instruments or systems which have not yet been applied or that are immature for in-field applications. Typical examples are optical methods to detect radionuclide contamination or to identify and reveal indicators and signatures (I&S) specific to the nuclear fuel cycle or weapons programmes.

The term “novel approach” refers to an application that attacks an old problem in a non-conventional manner. For example, alpha spectrometry without sample treatment, or completely without sampling, is an example of a novel approach. Other examples include algorithms or analysis systems that are able to treat measurement results much more thoroughly than conventional methods.

The borderline between new and novel is not sharp. Conventional technologies need to be improved, sometimes in a novel manner, to achieve better analytical performance of instruments. We apply these concepts in a loose manner.

The following review is intended to discuss a few promising novel methods for safety, security and safeguards. The text is by no means exhaustive or complete. It deals with the needs of international organisations and associations, such as the International Atomic Energy Agency (IAEA) and the European Safeguards Research and Development Association (ESARDA). The purpose of the discussion is to introduce novel approaches and technologies to:

- improve destructive analyses (DA);
- improve non-destructive analyses (NDA);
- develop new methods to support the non-proliferation of nuclear weapons;
- support treaty verification;
- combat nuclear terrorism; and
- provide powerful tools for consequence management of nuclear accidents.

2 Sampling – a basis for improvements to spectroscopy

Samples have traditionally been analyzed in laboratories, where several destructive analytical techniques are available. However, if sample treatment could be significantly improved, similar results could be achieved by non-destructive analyses.

NDA of samples has three general advantages (Ihantola 2009):

1. The productivity and timeliness of field missions can be improved. If the measurements are too complicated to be performed on-site, preparation of the samples in the field would nevertheless speed up the process in the laboratory, since it would eliminate the need for complex chemical treatment. For nuclear security applications, rapid methods are of particular importance.
2. On-site analysis of samples enables direct feedback. Therefore, inspections can be focused on those areas appearing most suspicious.
3. The production of high-quality or matrix-free samples would simplify the whole chain-of-custody as compared to traditional swipe sampling and later laboratory analysis. When the samples are prepared for counting and dispatched directly to the laboratory involved, there is no risk of cross-contamination, since the sample treatment is carried out far from the laboratory.

NDA measurement methods have not been fully exploited, either in field conditions or in laboratories. Currently, the detection of penetrating gamma radiation is widely used, whereas direct spectrometry with non-penetrating radiation, i.e., with alpha and beta particles, low-energy X-rays and conversion electrons, is considered to be too complicated. With a careful design of sampling, these measurements are possible (see section 4.1 for alpha measurements), including analysis, particularly when remote expert support (reachback) is available.

3 Novel approaches to spectroscopy

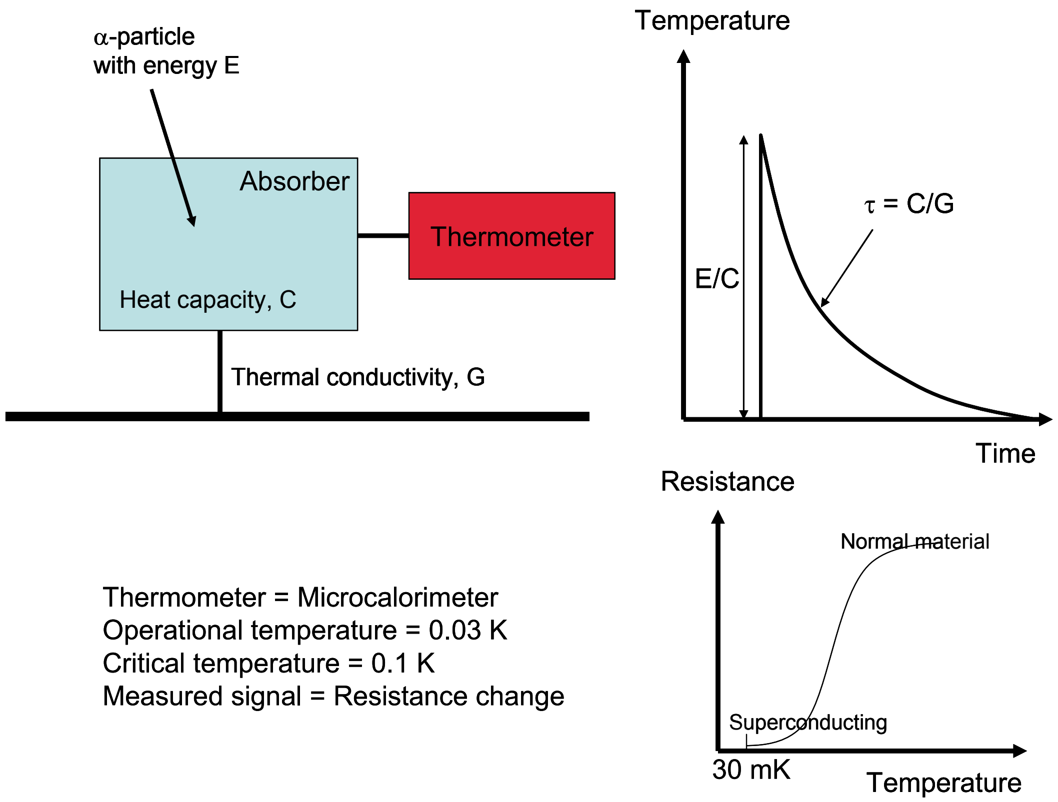
Spectroscopy is the basic work horse of every analytical laboratory. The quality of the measurements and related analyses has continuously improved. Recently, however, scientific and technical innovations have taken place in several domains. Some breakthroughs are elaborated below.

3.1 Microcalorimeters – spectrometry with ultra-high resolution

Microcalorimeters respond to the deposited energy and not to charge (Figure 1). Therefore, they do not have any dead layer characteristic of semiconductors. Transition-edge-sensor microcalorimeters consist of molybdenum and copper. The element ratio defines the critical temperature of the device. The best energy resolution is achieved at a critical temperature of around 0.1 K. The energy signal is derived from the change in the resistance in the circuit after the absorption of radiation quanta. While detecting penetrating radiation, a bulk absorber needs to be attached to the thermometer (= microcalorimeter). These are typically made of tin. There may be several contacts between the absorber and the thermometer. This arrangement eliminates the variability in the energy resolution, which would otherwise depend on the position of the interactions. The measured energy resolution, *FWHM*, of the microcalorimeter detectors is 25 eV for 103 keV photons and 1.06 keV for 5.3 MeV alpha particles (Doriese et al. 2007, Horansky et al. 2008). The corresponding resolutions with a planar HPGe and Si detector are ~450 eV and ~8.5 keV, respectively (Ullom 2008).

In the future, microcalorimeters may significantly impact on alpha spectroscopic procedures, since the achieved resolution of 1 keV is so good that separation of the elements is no longer required, i.e., only rapid chemistry is needed to get rid of the sample matrix. Presently challenging cases such as the analysis of ^{239}Pu (5.157 MeV) and ^{240}Pu (5.168 MeV) in samples will become much easier, since the relevant alpha peaks are completely separated. This also loosens the requirements related to the counting statistics. Therefore, microcalorimeters may in the future take part of the workload away from mass spectrometers. Note also that the routine use of microcalorimeters would reduce the time needed for the chemistry and counting, thus improving the throughput of the laboratory.

The nature of gamma and X-ray spectrometry would drastically change if microcalorimeters entered into routine use. For example, a currently major problem in nuclear security applications is the interference of ^{226}Ra with ^{235}U (Figure 2). However, a microcalorimeter provides a clear separation between the transitions of ^{235}U (185.71 keV) and ^{226}Ra (186.21 keV) (Hoover et al. 2009).



Thermometer = Microcalorimeter
Operational temperature = 0.03 K
Critical temperature = 0.1 K
Measured signal = Resistance change

Figure 1. Operation principle of a microcalorimeter.

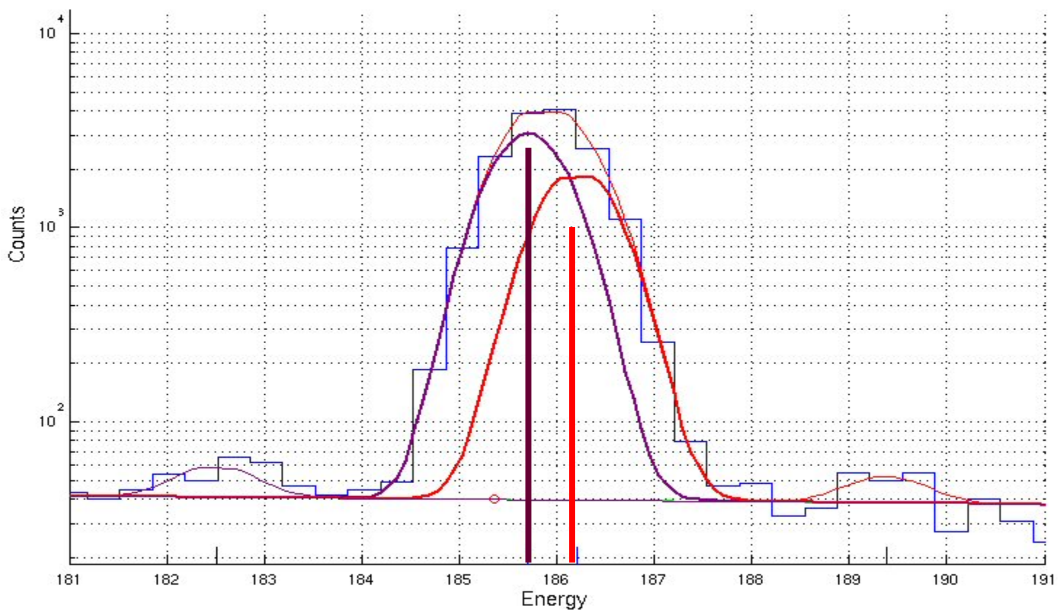


Figure 2. ^{235}U and ^{226}Ra doublet in a high-resolution gamma spectrum. The doublet is difficult to deconvolute, even with a state-of-the-art unfolding algorithm. If the measurement had been performed with a microcalorimeter, the peaks would have been fully resolved. The *FWHM* is smaller than the bar width shown in the figure.

3.2 Scintillation detectors and pulse shape analysis

Scintillation detectors that operate at room temperature can be used for measurements on particles and photons. Much work has been devoted to improving the neutron-gamma discrimination capabilities of scintillators. Such discrimination is achieved by introducing lithium gadolinium borate crystals (particle size ~1 mm) into a plastic scintillator matrix (Menaa et al. 2009). Neutron events are separated from gamma-ray events via the use of pulse shape analysis. Namely, the characteristic (n,α) reaction generally occurs within 10 μs after the recoil pulse. The resulting double peak structure provides an excellent means to disentangle neutrons from gammas.

Another development that we wish to highlight is based on pulse shape analysis and phoswich detectors. Phoswich detectors have layers of different scintillation materials with different properties. The layers are optically connected, and the entire packet is connected to a single photomultiplier tube. The energy signals from such a detector can be elegantly analyzed if the response of the different types of radiation is known. For example, when a beta particle and a gamma ray simultaneously hit the detector, their individual energies can still be determined (de la Fuente et al. 2008). Thus, these detectors, in addition to conventional singles spectrometry, are also capable of beta-gated and alpha-gated gamma-ray spectrometry. This is very significant for their possible use *in situ*, since coincidence spectrometry may remove the need for shielding around the detector. The resulting measuring system could be very compact.

Novel signal processing and data analysis techniques improve the performance and applicability of scintillation detectors. It is a technological, rather than a scientific challenge to bring these techniques into routine use. Furthermore, research and development of new scintillation materials will provide even better detectors for future in-field applications (Bourret-Courchesne et al. 2010).

3.3 Event-mode data acquisition and position-sensitive detectors

The simultaneous use of several state-of-the-art semiconductor detectors provides powerful tools for the analysis of samples (Keillor et al. 2009, Turunen et al. 2010). Multi-detector setups are often meant to be operated in laboratories. In addition to the quality in terms of resolution, the technical maturity of the semiconductors makes these detectors an attractive choice. For example, imaging silicon detectors capable of spectrometry are already commercially available. Position information is useful to investigate the distribution of alpha-particle-emitting radionuclides in various samples or to search for “hot” spots or radioactive particles. For safeguards and nuclear security, the characteristics of the individual particles are of interest; they reveal the history of the radioactive materials handled at the site.

Data acquisition is performed in event mode, also known as list mode. This means that any event, for example the absorption of alpha energy in a certain pixel of the alpha detector, triggers a gate signal for the energy depositions of the other detectors to be registered during a small time slot, typically for one microsecond. Later, the data storage allows the generation of various types of spectra, depending on the query requirements.

Of particular interest are the software-based coincidence spectra, such as position-gated and alpha-gated gamma-ray spectra. The knowledge that the gamma-rays in question have been detected in coincidence with alpha decay of a certain energy and impact location in a silicon detector makes the spectrum analysis and interpretation much more reliable and detailed. Due to the coincidence analysis, the spectrum background is lowered by orders of magnitude, and very low minimum detectable activities are consequently achieved (Turunen et al. 2010, Peräjärvi et al. 2010).

Techniques such as alpha-gamma coincidence spectrometry do not require chemical manipulation of the sample before measurement. For example, $^{240}Pu/^{239}Pu$ atom ratio can be determined directly from the resulting data (Peräjärvi et al. 2010).

4 Alpha radiation – difficult to detect in field conditions, or is it?

Alpha spectrometry is often considered as a mature technology for sample analysis in laboratory conditions, but there is still room for development. The method is based on radiochemical sample processing with radioelement separation and subsequent data acquisition in a vacuum chamber with an alpha spectrometer. In some applications, such as continuous aerosol monitoring, the processing is simplified by sampling the radionuclides on the surface of the sample material and performing the data acquisition in NTP air. In these cases, spectrum unfolding may be complicated because of the omission of radioelement separation. In addition, the alpha-particle energy loss in air between the sample and the detector, and thus the varying path lengths of the alpha particles, may lead to an energy resolution that is far from optimal.

In-field alpha spectrometry is challenging from the technical point of view. Scientifically, there is not much new. A step forward would be if alpha spectrometry could be conducted in the same way as *in-situ* gamma spectrometry. At present, the most common field application is surface contamination measurement, i.e., the number of alpha counts is registered per time interval. However, high-resolution alpha spectrometry using handheld equipment would provide much more detailed information on the nuclides involved. In this case, the overall approach is as follows:

- Radiochemical sample processing should be completely omitted or drastically simplified. The samples should be as thin as possible, but thick samples should not be excluded if they are homogeneous. This means the development of novel sampling procedures. Minor processing, such as water evaporation under an infrared lamp or sample cutting, is acceptable. Alpha spectra should also be registered from smooth surfaces, allowing radionuclide identification.
- Measurements should be performed without a vacuum and the energy resolution should still remain good.
- The size of the equipment for data acquisition should be similar to that of present hand-held monitors.
- Efficient and validated software must be used for data analysis. If this is not possible, an ability to send the spectra to a remote location should exist if the analysis cannot be performed on-site. Reachback is of utmost importance in security applications.

4.1 Novel sampling and measurement techniques facilitate in-field applications

The development of appropriate sampling and sample manipulation procedures for rapid measurements has been conducted for years in different organizations. Particularly for swipe samples and liquid samples, there is room for development (Figure 3a), although some progress has already been achieved in this field (Ihantola 2009, Semkow et al. 2009).

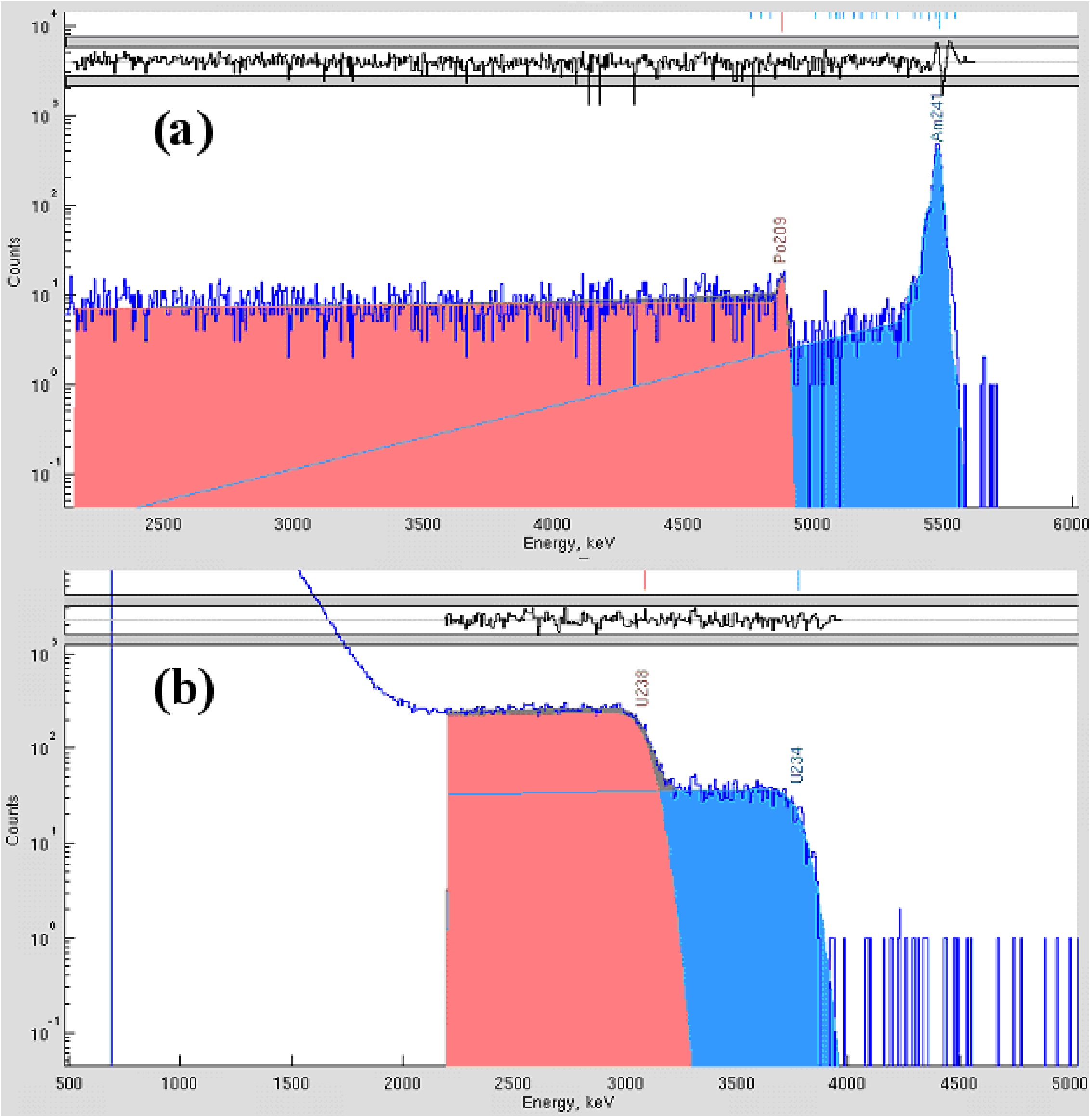


Figure 3. (a) Alpha particle energy spectrum from filtered liquid (3 mL beer) doped with ^{209}Po and ^{241}Am. The spectrum was measured in a vacuum chamber from the filter. (b) Alpha spectrum from the surface of a depleted uranium penetrator. The spectrum was measured in NTP air at a source-to-detector distance of 9 mm. The energy region used in the fitting was 2.2–4 MeV (the energy scale refers to the energy of detected alphas). The large number of counts at small energies is due to the high-energy betas originating especially from ^{234m}Pa.

Nuclide-specific activities cannot necessarily be obtained accurately if the sample is too thick and the peak shapes are different (reflecting differences of radionuclide distribution in the source, see Fig. 3). Reliable estimates for the nuclide ratios may also be impossible in this case. However, the identification of radionuclides is possible. As an example, an alpha spectrum was registered from a depleted uranium penetrator (Figure 3b) in air, i.e. no vacuum was used in the data acquisition. ^{238}U and ^{234}U can be clearly identified from the spectrum, but their activities cannot be measured. Because the shapes of ^{238}U and ^{234}U "peaks" are equal, the activity ratio ^{238}U/^{234}U (in reality ^{238}U / ^{234}U + ^{235}U + ^{236}U) can easily be determined. Here, the value of 6.32 was obtained, whereas 7.05

was reported in Pöllänen et al. (2003). Thus, alpha spectrometry of surfaces of thick sources is possible, even though the measurements are performed in NTP conditions.

4.2 Vacuum is not necessary to obtain alpha spectra with good energy resolution

High-resolution alpha spectra can be acquired in NTP conditions (Pöllänen et al. 2010). A thin source containing ^{239}Pu, ^{241}Am and ^{244}Cm, normally used in the energy calibration of the alpha spectrometers, was used for testing different counting configurations. Using the conventional approach, the energy resolution is far from optimal. The reason is the varying path length of the alpha particles in air. Much better resolution can be achieved by optimizing the measurement system.

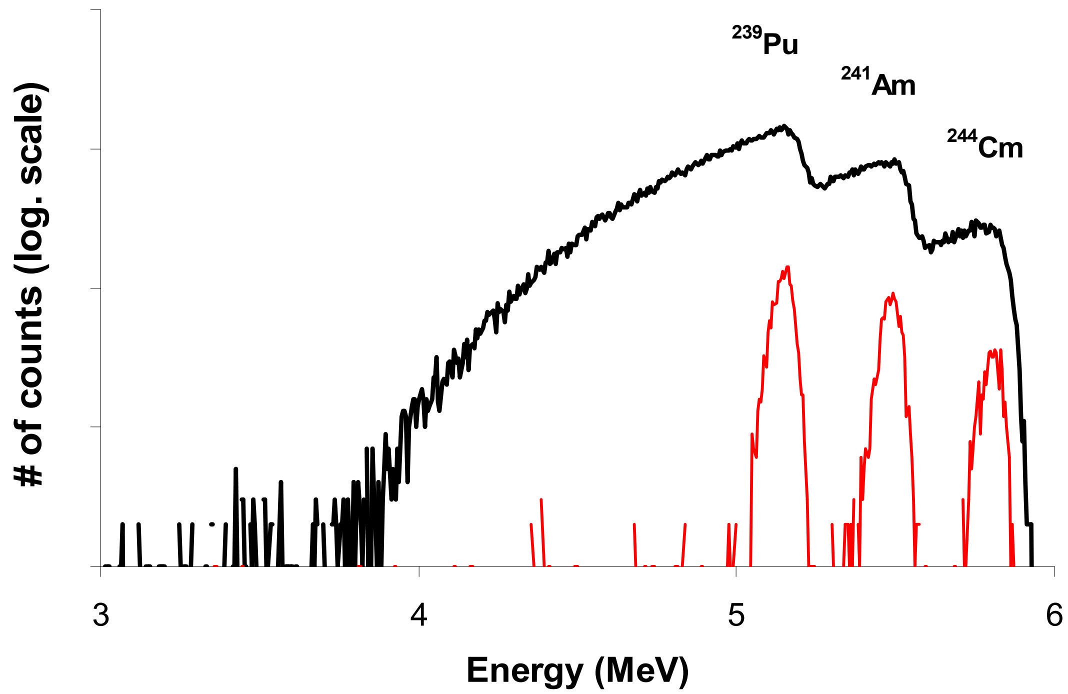


Figure 4. Alpha particle energy spectrum from a calibration source measured at NTP (black line) and with a modified system (red line). In both cases the source-to-detector distance was 9 mm. The number of counts is arbitrarily normalized.

4.3 Efficient data management and spectrum analysis are needed for in-field applications

Alpha spectra from three-dimensional sources or those measured in NTP conditions are different from those obtained from radiochemically processed samples. Multiple overlapping peak groups appear in the spectrum if several radioelements are present in the source. In addition, the shape of the alpha peaks is completely different and depends on the measurement geometry and the radioelement in question. Energy calibration should also be performed with great care because concepts such as peak “location”, “centroid” or “energy” do not characterize the peaks optimally. A prerequisite for a viable method for spectrum unfolding is that the phenomena mentioned above are properly taken into account in the analysis software.

A novel computer code known as ADAM (Toivonen et al. 2009) has recently been developed to cope with the requirements mentioned above. The program is able to unfold complex alpha spectra measured in a vacuum or in NTP conditions. As an example, an alpha spectrum from a radiochemically processed source, and containing ^{238}Pu, ^{239}Pu, ^{240}Pu and ^{242}Pu, was analyzed (Fig. 5). In contrast to several other programs, the activities of ^{239}Pu and ^{240}Pu can easily be reliably determined. This is because the uncertainties of the fit are in good statistical control (covariance calculus). The spectra presented in Figure 3 were also analyzed by ADAM.

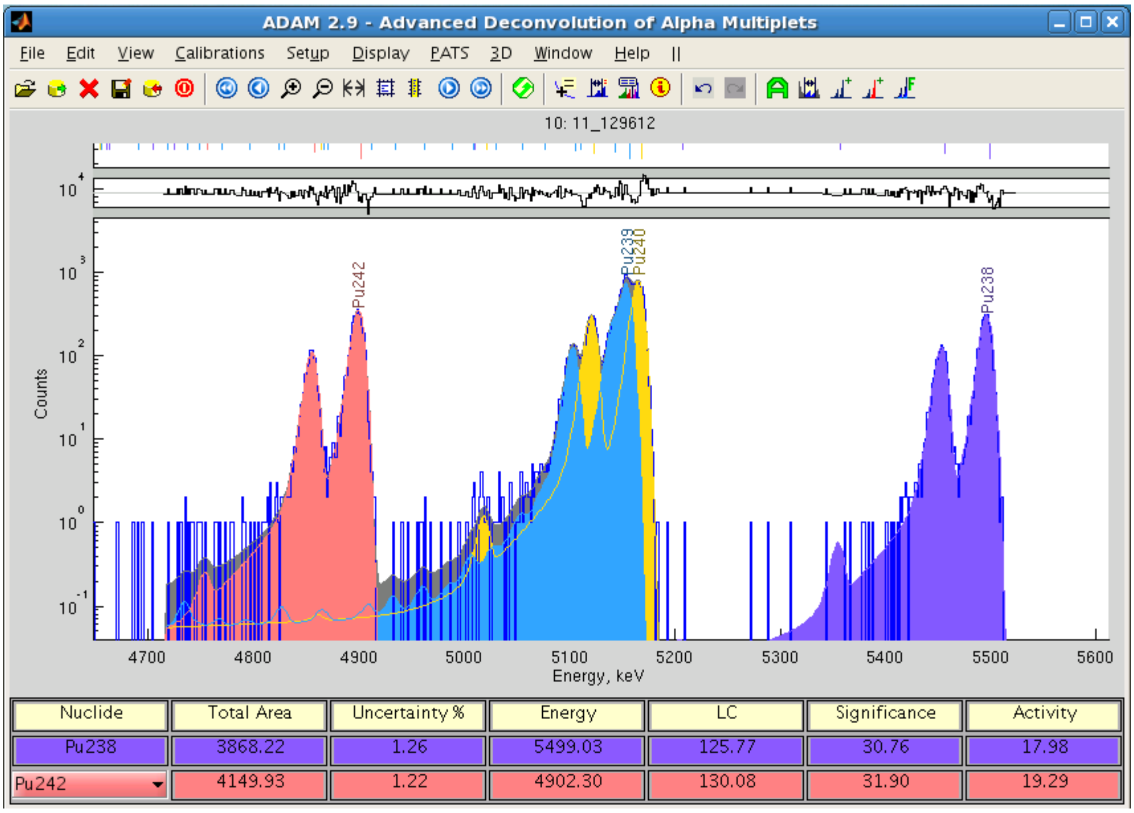


Figure 5. Unfolding an alpha particle energy spectrum from a radiochemically processed sample containing ^{238}Pu, ^{239}Pu, ^{240}Pu and ^{242}Pu. A semicoductor detector of 50 mm^2 in area and 5 cm source-to-detector distance was used in the measurement. The measurement time was 1.5 d.

Spectrum unfolding in the field may be too challenging, especially when the measurement has been carried out by a person who is not familiar with alpha spectrometry and the analysis tools. Efficient reachback is then needed. Such data processing is very demanding. However, if successfully implemented, the operational capability is drastically improved. Timely and high-quality nuclear spectroscopy is available for police, fire fighters and other first responders.

5 Optical sensing

In revealing the I&S of nuclear processes, optical measurements provide advantages that cannot be achieved with traditional nuclear instrumentation. For example, optically stimulated luminescence (OSL) could reveal crucial information on a suspected location that may previously have been a storage site for radioactive materials.

The review below is a short introduction to optical methods useful for safety, security and safeguards. Only some key applications are covered.

5.1 Light Detection And Ranging

Light Detection And Ranging (LIDAR) is an optical remote sensing system that measures scattered light to find information on a distant target. Laser pulses are sent to excite the target molecules. The operating principle of LIDAR is similar to radar, which uses radio waves and measures the time delay between the transmission of a pulse and the detection of the reflected signal. LIDAR uses short wavelengths in the ultraviolet, visible or infrared range. Thus, LIDAR is highly sensitive to aerosols and cloud particles and has many applications in atmospheric research and meteorology.

A laser has a very narrow beam, which allows the mapping of physical features at a very high resolution. In addition, many chemical compounds interact strongly at visible wavelengths, resulting in a good image of these materials. Remote mapping of atmospheric contents is possible by using specific lasers and then studying wavelength-dependent changes in the intensity of the returned signal.

LIDAR has been extensively used for atmospheric research and meteorology. For safeguards, it is envisaged that a mobile laboratory, in the vicinity of a suspected location, could use laser at a certain wavelength stimulating airborne molecules that emanate from nuclear processes. A light sensitive telescope could then scan the stimulated atmosphere, detecting the presence or absence of the signature molecules.

5.2 Laser-Induced Breakdown Spectroscopy

Laser-Induced Breakdown Spectroscopy (LIBS) is based on the analysis of the optical emission of the induced plasma. A LIBS system has a high-power laser to ablate the surface of the material. This creates a micro-vapour that emits optical radiation. A spectrometer is then used to analyze the constituent components. The resolution of LIBS depends on the design of the spectrometer. Instruments can be built to provide elemental results and even isotopic analysis has been envisaged.

LIBS has the power to identify I&S (Kosierb et al. 2009). High-quality gas centrifuges require special construction materials, such as maraging steel. LIBS was able to detect the unique chemical composition of maraging steel, and it could be distinguished from other iron or steel alloys. The origin (site of production) of U_3O_8 (yellowcake) could also be demonstrated.

LIBS is already widely used by research groups and industry. Technology miniaturization has been successful and handheld devices have been developed. LIBS provides many advantages for safeguards. For example, LIBS requires no sample preparation and the results are available in real-time.

The Novel Technology Unit of the IAEA has a portable LIBS. It has been developed by the Canadian Member State Support Programme to the IAEA safeguards.

The future in-field missions will test the power of LIBS to identify I&S related to clandestine nuclear activities.

5.3 Remote sensing of alpha emitters

Alpha emitters are difficult-to-detect nuclides, especially in field conditions. This is because alpha particles have a short range in air (< 4 cm). Remote alpha detection could be possible if there were a suitable energy converter, i.e., a secondary mechanism that carries part of the energy far away from the source. The best known converter is ZnS(Ag), which was used by Rutherford some 100 years ago in his famous experiments to reveal the characteristic properties of alpha radiation. In those days, a pair of human eyes with a microscope formed the optical detector. Air contains natural energy converters - nitrogen and oxygen. Nitrogen molecules (N_2), in particular, serve well for remote sensing of alpha radiation.

Delta electrons are created near the track of the alpha particles in air. These electrons induce the fluorescence of N_2 molecules. Most of the light is at the wavelengths of UV, i.e., between 300 nm and 400 nm. The main challenge in optical detection is to discriminate the weak fluorescence signal from the background. Outside, in sunlight, this is difficult or impossible, but in darkness or in dim light very promising results have been achieved (Bashenko 2004, Lamadie et al. 2005, Hannuksela 2009).

The following goals, inter alia, are fully achievable and merit intensive research:

1. A handheld surface contamination meter with fast response. The device should detect alpha radiation at a distance of 20–50 cm. Efficient and convenient screening of nuclear workers would then be possible (safety). The search for contaminated spots would be much easier than in the past (nuclear security, cases such as the aftermath of Mr Litvinenko). Best sampling sites would be identified (safeguards, nuclear forensics). A prototype of such a device has already been built (Hannuksela et al. 2010).
2. Scanning of people and goods at border crossing points. The optics should be tailored for a distance of 2–5 m. Passengers and their luggage could be silently screened for any spot containing alpha emitters.
3. Imaging room, factory hall or laboratory. An alpha camera could be built for long exposure times in complete darkness. Superimposing the normal visible image and the alpha image would show the overall contamination at the site. A prototype has already been built using CCD technology (Lamadie et al. 2005).
4. Scanning or imaging glove boxes. The device should operate through plexi glass and identify the most interesting locations for further studies.
5. Reconnaissance of a target of interest at very large distances. For special applications, suitable optics could be designed taking into account that the photons arrive at the detector from a very small angle.

6 Challenges to technology

Many laboratories worldwide are developing active and passive detection methods that have great potential for safety, security and safeguards. Examples of active detection methods include the use of neutron beams or high-energy photons to induce characteristic radiation in the target material for detection through coincidence techniques. These systems produce a high radiation dose and cannot be used to screen human beings. An example of the low-dose devices is a person scanner, i.e., a device that sees through the clothes using either low-energy photon scattering or non-ionising radiation at teraherz frequencies.

It is not possible within the framework of the present paper to review the safety and security domain in detail. Therefore, only a few striking examples are given in relation to passive detection methods.

6.1 Compton imaging

Remote location of gamma-emitting sources has traditionally been realized by collimating the gamma detector. These collimators are heavy and the screening may take a long time due to the limited spatial view of the detector. Such a collimator is not needed if the detector is replaced with a stack of position-sensitive silicon and germanium detectors (Vetter et al. 2007). Detailed analysis of the multiple interactions of the incoming gamma radiation, in addition to conventional spectrometry, provides a means to locate the source. To facilitate the interpretation of the results, the radiation information can be combined with visual images.

6.2 Screening of cargo with muons

Cosmic muons are continuously bombarding the Earth with a flux of about 10^4 /(min×m^2). The muons easily pass through matter, i.e., they have long ranges in materials. However, they scatter more easily from heavy and high density materials than from light materials. The object to be scanned is surrounded with appropriate detectors for registering the trajectories of the individual muons. This reveals heavy materials inside the object (Pesente et al. 2009).

Interesting devices have been proposed for future use to complement regular X-ray systems for payloads that cannot be imaged with X-ray devices, for example in the search for stowaways or smugglers. Note that a 300 kV (9 MV) X-ray source is required to make an image through 5 cm (35 cm) of steel. The only drawback related to this technique is the relatively low muon flux on Earth. Scattering information can also be used to classify the inspected materials (density measurements).

6.3 Neutrinos reveal nuclear chain reactions

Antineutrino detectors relying on the $\overline{\nu} p \rightarrow e^+ n$ reaction can be used to monitor nuclear reactor power and fuel composition. The most interesting developments on this field are related to the use of plastic scintillators instead of liquid ones (Battaglieri et al. 2009). Namely, plastic scintillators are preferred since they do not have problems with flammability or toxicity. This is obviously very important if the detector is placed in the vicinity of a nuclear reactor. Solid state antineutrino detectors are composed of scintillator bars that have photomultiplier tubes attached to both ends. Additionally, plastic detector bars are wrapped with gadolinium foils to trigger (n,γ) reactions in the foils. Signals related to the stopping of positrons and 511 keV annihilation radiation together with the gamma-rays following neutron capture by gadolinium form a unique fingerprint for the identification of antineutrino events. The active volume of the detector is 1 m^3. If such a detector is placed at a distance of 25 m from a 3 GW reactor core, it is expected to detect about 5000 antineutrino events per day. This is enough for independent monitoring of the reactor operation.

7 Discussion

Novel approaches and technologies for safety, security and safeguards are a challenge to the scientific community. In-field operations in particular can be much improved using technology that may fall outside the traditional nuclear instrumentation or that is developed from a new perspective.

Basic research has long been using methods that would be very useful in applied research or monitoring. These methods will find their way to applied radiation sciences and to field operations. A typical example of such methods is the recording of radiation interactions as events and then utilizing coincidences for spectrum generation through software. This kind of data acquisition is slowly creeping into standard nuclear instrumentation, but no mature application is widely available. There is room for the development of equipment and for application software that would make the system user-friendly.

Extensive research and development will inevitably lead to better radiological and nuclear detection capabilities. In the aftermath of September 11, major new resources have been allocated to nuclear security, but the achievements in this field will ultimately also be useful for safety and safeguards. In the future, the detection of I&S - regardless of the domain - will be more sensitive, rapid and reliable. Science is for safety, security and safeguards.

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Good regulators and good operators – in search of mutual trust

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Abstract

Everybody ‘knows’ what is expected of a regulatory body and how a licensee establishment is supposed to behave. This presentation aims to show that these concepts can be more elusive than first impressions may indicate. Well-known facts are frequently forgotten by new generations of regulators and operators, and ground once conquered and regarded as secured must be re-captured, time and again.

Obviously, a regulatory body needs to trust each operator: a license (for instance, to use x-ray equipment) constitutes proof of that trust, and indicates that the licensee is perceived as having sufficient competence, responsibility, and resources in view of the risks involved in the licensed activity. However, all too often, inexperienced regulators focus on prescriptive instructions and follow-up of minute details, thus ‘stealing’ responsibility from the licensee. And all too often, inexperienced operators focus on compliance with prescriptive instructions instead of finding out the reasons behind the instructions and negotiating when required the best way of ensuring that goals are met and deviations are spotted before they turn into incidents.

Similarly, an operator needs to trust the regulator: licenses must be awarded, denied, or retracted on relevant grounds only. However, the operator must also be convinced that the regulator’s primary aim is safety rather than punishments. Incident reporting is a vital safety tool, but it can only work if the operator is reasonably sure that an incident report will not generate undue punishments.

Performance-based regulation does not mean lax control compared to prescriptive regulatory systems, nor does it permit careless or non-uniform operating procedures. It does call, however, for genuine mutual trust, and collaboration towards a common goal of safety. Good examples are available in other safe industries, such as aviation.

These concepts are discussed in the 2007 ICRP Recommendations, but only briefly and actually the 1990 ICRP Recommendations provided more advice in this respect. They are also discussed in various IAEA publications, but unfortunately there is some scope for un-necessarily prescriptive solutions, for instance if ‘immaturity’ is invoked.

A plausible radiobiological model of cardiovascular disease at low or fractionated doses

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Abstract

Atherosclerosis is the main cause of coronary heart disease and stroke, the two major causes of death in developed society. There is emerging evidence of excess risk of cardiovascular disease at low radiation doses in various occupationally-exposed groups receiving small daily radiation doses. Assuming that they are causal, the mechanisms for effects of chronic fractionated radiation exposures on cardiovascular disease are unclear. We outline a spatial reaction-diffusion model for atherosclerosis, and perform stability analysis, based wherever possible on human data. We show that a predicted consequence of multiple small radiation doses is to cause mean chemo-attractant (MCP-1) concentration to increase linearly with cumulative dose. The main driver for the increase in MCP-1 is monocyte death, and consequent reduction in MCP-1 degradation. The radiation-induced risks predicted by the model are quantitatively consistent with those observed in a number of occupationally-exposed groups. The changes in equilibrium MCP-1 concentrations with low density lipoprotein cholesterol concentration are also consistent with experimental and epidemiologic data. This proposed mechanism would be experimentally testable. If true, it also has substantive implications for radiological protection, which at present does not take cardiovascular disease into account. The Japanese A-bomb survivor data implies that cardiovascular disease and cancer mortality contribute similarly to radiogenic risk. The major uncertainty in assessing the low-dose risk of cardiovascular disease is the shape of the dose response relationship, which is unclear in the Japanese data. The analysis of the present paper suggests that linear extrapolation would be appropriate for this endpoint.

Quantification of γ-H2AX foci in Jurkat cells following irradiation with α-particles and γ-rays

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Abstract

Objectives: Phosphorylation of histone H2AX occurs at sites flanking DNA double-strand breaks (DSBs) and can provide a measure of the number of DSBs within a cell. We investigated whether the mean intensity and the mean number of radiation-induced γ-H2AX foci per cell vary as a function of radiation quality and dose.

Materials and methods: Jurkat cells were irradiated with different doses of either low linear energy transfer (LET) ^{137}Cs γ-rays or high LET ^{241}Am α-particles. The γ-H2AX foci were detected using immunocytochemistry and quantified by measuring the mean signal intensity of γ-H2AX foci per cell using flow cytometry and by counting the number of γ-H2AX foci with a fluorescence microscope.

Results: The mean numbers of γ-H2AX foci per cell increase with dose and they are fairly identical at 1 Gy for both investigated radiation qualities. The mean intensity of γ-H2AX foci per cell after α-irradiation is significantly increased when compared to γ-irradiation at the same radiation dose. A more advanced flow cytometry analysis reveals that the percentage of γ-H2AX-negative cells as well as the distribution of single γ-H2AX fluorescence signals depend on the radiation quality.

Conclusion: The mean signal intensity, but not the mean number, of γ-H2AX foci per cell depends on the LET in Jurkat cells. When comparing the induction of γ-H2AX foci in Jurkat cells after exposure to γ-rays and α-particles, the analysis by flow cytometry is more appropriate than the microscopical quantification by eye, considering the LET-dependence of foci-size and -intensity, the cell cycle dependence of γ-H2AX frequencies and the distributions of single γ-H2AX fluorescence signals.

Radiation-induced transgenerational instability and genetic risk

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Abstract

Mutation induction in the directly exposed cells is currently regarded as the main component of the genetic risk of ionizing radiation. However, recent data on the delayed effects of exposure to ionizing radiation represent a new challenge to the existing paradigm. The results of numerous studies show that ionizing radiation can not only induce mutations in the directly exposed cells, but can also lead to delayed effects, with new mutations arising many cell divisions after irradiation.

Apart from the studies on mutation rates in somatic cells, considerable progress has been made in the analysis of radiation-induced instability in the mammalian germline, where the effects of radiation exposure were investigated among the offspring of irradiated parents. Our results show that mutation rates at tandem repeat DNA loci and protein-coding genes are substantially elevated in the germline and somatic tissues of non-exposed offspring of irradiated male mice. According to our data, the transgenerational destabilization can be attributed to the presence of a subset of endogenous DNA lesions. We have recently shown that paternal treatment by the alkylating agent ethylnitrosourea also results in the transgenerational effects, thus implying that this phenomenon is most probably triggered by a stress-like response to a generalized DNA damage.

Our data imply that instability detected in the non-exposed offspring is caused by some DNA-dependent signal transmitted from the irradiated father and implicate an epigenetic mechanism for the transgenerational instability. The potential implication of these results for the estimates of genetic risks for humans will be discussed.

Low doses of irradiation on nervous cells impairs neurite outgrowth and causes neuronal degeneration

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Abstract

Brain damage induced by prenatal irradiation is a major concern and an important issue in radiation protection. The embryonic stage of brain development is known to be highly sensitive to radiations, since most of the concerned cells are still mitotic. Thus, an acute irradiation during pregnancy could selectively damage cells, which at that particular time of exposure, are proliferating or migrating. Data collected from the epidemiological studies on the children that were irradiated in utero during Hiroshima and Nagasaki's A-Bombing, showed an increase of some mental and/or cognitive disorders. Depending on the received radiation dose, the severity of the damage occurs at crucial embryonic stages for neuronal survival, proliferation, migration, and/or neural network formation. In this study, we investigate the main cellular and molecular mechanisms involved in radiation-induced neurite outgrowth and apoptosis. Various time points and irradiation doses were compared at different neuronal maturation stages. Image analysis of immuno stained control compared to irradiated neuron cells with low and moderates doses (0.1, 0.2 and 0.5 Gy), showed a clear negative radiation effect on neurite outgrowth depending on the maturation stage of neurons. In particular, the main radiation-induced morphological effects were: shortened neurite length, decreased number of neurite per neuron and reduced synapse branching. These effects could lead to a defect in neural network formation and consequently to possible cognitive disorders at the adult age. The molecular and biochemical events will be further analysed using transcriptomic, proteomic and metabolomic technologies in order to identify the compromised molecular and signaling pathways in neuron cells after exposure to moderate and low doses of X-rays. Molecular strategies (overexpression or silencing) would help to validate the identified pathways.

The European Radiobiological Archives: online access to data from radiobiological experiments

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Abstract

Today's research is providing us more and more with the opportunity to quantify radiation risks at the individual level. New approaches allow the re-analysis of old data using new techniques. Thus, the retrospective analysis of earlier studies is an important resource for modelling and evaluating risk parameters. The European Radiobiology Archives (ERA), together with corresponding Japanese and American databases, hold data from nearly all experimental animal radiation studies carried out between the 1950s and the 1990s, involving more than 400,000 animals. These experiments were performed on different species with the aim of understanding the effects of irradiation. This mass of information has led to the requirement for a computerized database to store, organize and index this data. The previously existing ERA archive has now been transferred to a web-based database to maximise its usefulness to the scientific community and bring data coding and structure of this legacy database into congruence with currently accepted semantic standards for anatomy and pathology. The accuracy of the primary data input was assessed and improved. The original rodent pathology nomenclature was recoded to Mouse Pathology (MPATH) and Mouse Anatomy (MA) ontology terms. A pathology panel sampled histopathological slide material and compared the original diagnoses with currently accepted diagnostic criteria. The mean non-systematic error rate was low with only 1.7%. Detected errors were corrected. The majority of the original pathology terms have been translated into a combination of MPATH and MA ontology terms. The database is accessible online at *http://era.bfs.de*. It has the potential of becoming a world-wide radiobiological research tool for numerous applications, such as the re-analysis of existing data and using the database as an information resource for planning future animal studies.

A large-scale gene expression database on the effect of low doses of radiation: a systems biology approach

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Abstract
Current radiation protection regulatory limits are based on the linear non-threshold theory using epidemiological data from atomic bombing survivors. Recent studies sparked debate on the validity of the theory, especially at low doses. Recent advances in molecular biology have shown that different genes and pathways might be triggered by low doses and by high doses of ionising radiation. Identifying the genes and pathways involved in the response to low doses is therefore critical for a better prediction of a possible clinical outcome following radiation exposure.

The increasing interest in the effect of low doses of radiation coincides with the breakthrough in the development of new high-throughput technologies in molecular biology. The microarray technology allows simultaneous measurement of the expression of thousands of genes. As gene expression is the predominant level controlling cell functioning, this technology has become a powerful tool to measure the impact of low doses of radiation.

A major challenge is how to integrate all data resulting from a wide range of experiments and use them to improve our global understanding of the general molecular response of a cell after radiation exposure to low doses. Therefore, we organized into one microarray compendium all radiation expression studies performed within our laboratory. In total, our compendium comprises 233 arrays representing 89 different biological conditions. Applying a systems biology approach on these data will shed light on the relationship between gene expression and different parameters like dose, tissue and organism. Additionally, identification of pathways triggered upon low doses of radiation might open new opportunities towards biodosimetry and appropriate regulation regarding medical exposure and intrinsic radiation sensitivity.

Introduction
The last decade, molecular biology has been revolutionized by the development of many high-throughput technologies *e.g.* in DNA sequencing, gene expression analysis, electrophoresis, and mass spectrometry. A major advancement herein was the invention

of microarrays allowing the transcriptional analysis of thousands of genes in a single experiment. This technological progress now allows analyzing simultaneously the expression of genes at a genome-wide scale. The usage of this technology is certainly not new, as different research groups – including ours - already used this technology to study the influence of radiation on different tissues like blood, brain, liver, kidney, thyroid and others (Amundson et al. 2000, Franco et al. 2005, Paul and Amundson 2008, Pawlik et al. 2009, Tachiiri et al. 2006, Taki et al. 2009, Zhao et al. 2006).

However, to our knowledge it is the first time that a systematic approach is developed to analyse a large and diverse compendium of microarray data using a systems biology approach. Over the last three years, a large number of microarray experiments using Affymetrix technology were performed within the radiobiology group of the Belgian Nuclear Research Center (SCK•CEN), exposing different tissues from human and mouse origins to various doses of ionising radiation (ranging from 0.025 Gy to 8 Gy of X-rays). This resulted in a total of 233 arrays, representing 89 different conditions. The aim of this study is to combine the knowledge present in all these data sets and explore the influence of different parameters (species, tissue, dose) on the genome-wide transcription profiles. In a first stage, a high-level analysis of the data is performed, applying traditional data mining techniques, and exploring the clustering of the different microarray data in the light of the here above mentioned parameters. In a second stage, the data are analysed at the functional level by classifying them based on their annotation, and mapping the gene expression data on existing pathways in order to identify new pathways triggered upon ionizing radiation.

Material and methods

Data sets

All samples from various organisms (mice or men) and various tissues (blood, thyroid, brain and embryo) were irradiated using X-rays ranging from 0.025 Gy to 8 Gy with a Pantak HF420 RX machine operating at 250 keV, 1.5 or 15 mA, 1 mm Cu filtration and a dose rate of 0.375 Gy/min. Dosimetry was performed on a regular basis with a 0.6 cm^3 ionisation chamber (NE 2571), which was connected to a dosemeter (Farmer dosemeter 2570). The chamber was placed in parallel to the irradiated mouse cages. Dose homogeneity was evaluated as being <1.5%. From all samples, RNA was extracted and converted to biotin-labelled cDNA, which was hybridized to GeneChip® Human Gene 1.0 ST Array and Mouse Gene 1.0 ST Array (Affymetrix, Santa Clara, USA) for human and mouse samples respectively (Table 1).

Table 1. Overview of microarray data.

Species	Tissue	Doses (Gy)	# Arrays	# Conditions
Human	Blood	0.1, 1	60	30
Human	Thyroid	0.0625, 0.5, 4	12	4
Mouse	Thyroid	0.025, 0.05, 0.0625, 0.1, 0.5, 1, 4, 8	86	30
Mouse	Embryonic brain	0.1, 0.2, 0.5, 1	37	10
Mouse	Whole embryo (early gastrula)	0.2, 0.4	38	15
		Total	**233**	**89**

Data preprocessing

Raw Affymetrix data were preprocessed using the "Affy" package (version 1.22.0) in BioConductor (version 2.4) as follows: 1) background correction based on the Robust Multichip Average (RMA) convolution model (Irizarry et al. 2003), 2) quantile normalization to make expression values from different arrays more comparable (Bolstad et al. 2003), 3) summarization of multiple probe intensities for each probeset to one expression value per gene using RMA (Irizarry et al. 2003). To test for differential expression between the different irradiated conditions and the reference conditions (no irradiation) we used the Bayesian adjusted *t*-statistics as implemented in the "LIMMA" package (version 2.18.0) (Smyth 2004). P-values were corrected for multiple testing using the Benjamini and Hochberg's method to control the false discovery rate (Benjamini and Hochberg 1995).

Statistical and data mining applications

All statistical and data mining analyses like principal component analysis, correlation analysis and hierarchical clustering were performed with "Stats" package within the statistical software R (version 2.9.0).

Differentially expressed genes and overlapping gene sets

In order to identify differentially expressed genes, a fixed cutoff on the fold change of 1.5 was used. Statistical significance of the overlap of differentially expressed genes between different experiments was calculated using the hypergeometric distribution. This distribution allows calculating the overlap of genes between two conditions, taking into account the number of differentially expressed genes in both conditions separately.

Functional enrichment

A gene set enrichment analysis (GSEA) determines to which extent a specific gene set is associated with a particular pathway or biological function. In order to calculate the statistically significant enrichment of a set of differentially expressed genes towards a specific pathway or biological process, the KEGG (Kanehisa et al. 2010) and Gene Ontology (Ashburner et al. 2000) databases were used respectively. This analysis was performed using the "GOstats" package (version 2.10.0) (Falcon and Gentleman 2007) of BioConductor (version 2.5)

Results

Data preprocessing

All data were preprocessed as explained in the "Material and methods" section. The relative expression of genes was measured by comparing the gene expression in irradiated tissues with gene expression in control samples of the same tissue. This prevented that we would only find tissue-specific genes when comparing different experiments, which would not be related to irradiation but rather to the intrinsic function of the cells in specific tissues. By recalculating the expression levels to fold changes (*i.e.* the ratio of the expression level in irradiated tissues over the expression level in control non-irradiated samples), the number of conditions was reduced from 89 to 63 experiments. By applying a variation filter, all genes showing minimal variation across different experiments were excluded.

Classification of radiation expression profiles

Using Principal Component Analysis (PCA), a first approach focused on the general similarities or dissimilarities between transcription profiles of different experiments. PCA is an algorithm that reduces the dimensionality of the data while retaining most of the variation in the data set. Experiments grouping closely together in a PCA biplot can be considered as sharing a similar transcriptional response after irradiation.

The extreme position of two thyroid irradiation experiments is a first emerging pattern (see Figure 1). The position of these experiments on the PCA plot could be more easily explained by the experimental setup, than by an effect of dose or tissue. Indeed, in contrast to other irradiation experiments where messenger RNA (mRNA) was isolated at most a few hours after irradiation, the mice in those experiments were irradiated at a specific time point, and the mRNA of the thyroid was isolated after 9 or 18 months respectively.

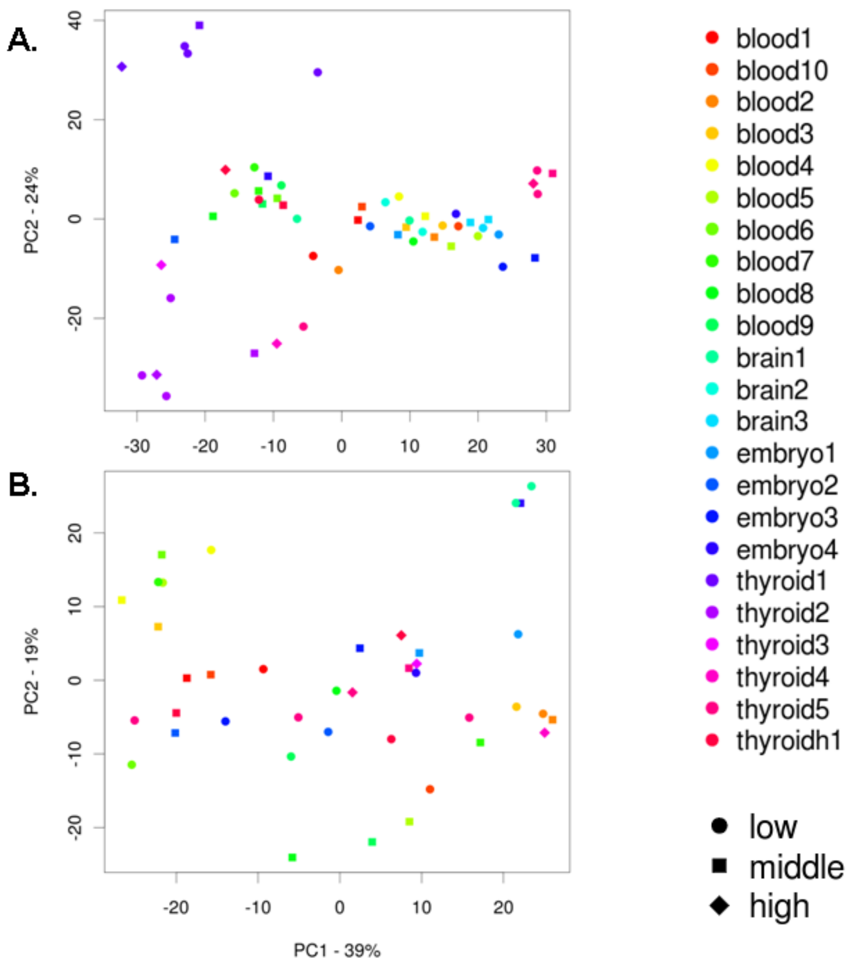

Fig. 1. PCA biplot of A) all irradiation experiments. B) all irradiation experiments with exception of long-term thyroid experiment (mRNA extraction after 9 [thyroid1] and 18 months [thyroid2]). Colour codes represent the different experiments, symbols indicate the radiation dose: low (<0.2Gy), middle (0.2Gy – 1Gy) and high (> 1Gy)

As a second observation, no tight clustering was found for the parameter of dose. This is remarkable as the data set included samples irradiated over a wide dose range, varying from low doses as 0.025 Gy up to doses as high as 8 Gy. The same conclusion could be drawn concerning the species-effect: no clear distinction could be made between the gene expression profiles originating from human or mouse samples. On the contrary, all microarray data seemed to cluster together rather on the type of experiment, even when irradiated with largely different radiation doses. For example considering the arrays measuring the gene expression after irradiation of the thyroid followed by immediate extraction of mRNA, all arrays cluster tightly together regardless irrespective of the irradiation dose (0.1 or 4Gy). Such effect could hardly be explained by an artefact resulting from a batch effect, since experiments performed with the same settings but at time points lying at least one year from each other, tend to cluster together as observed in the irradiated human blood samples.

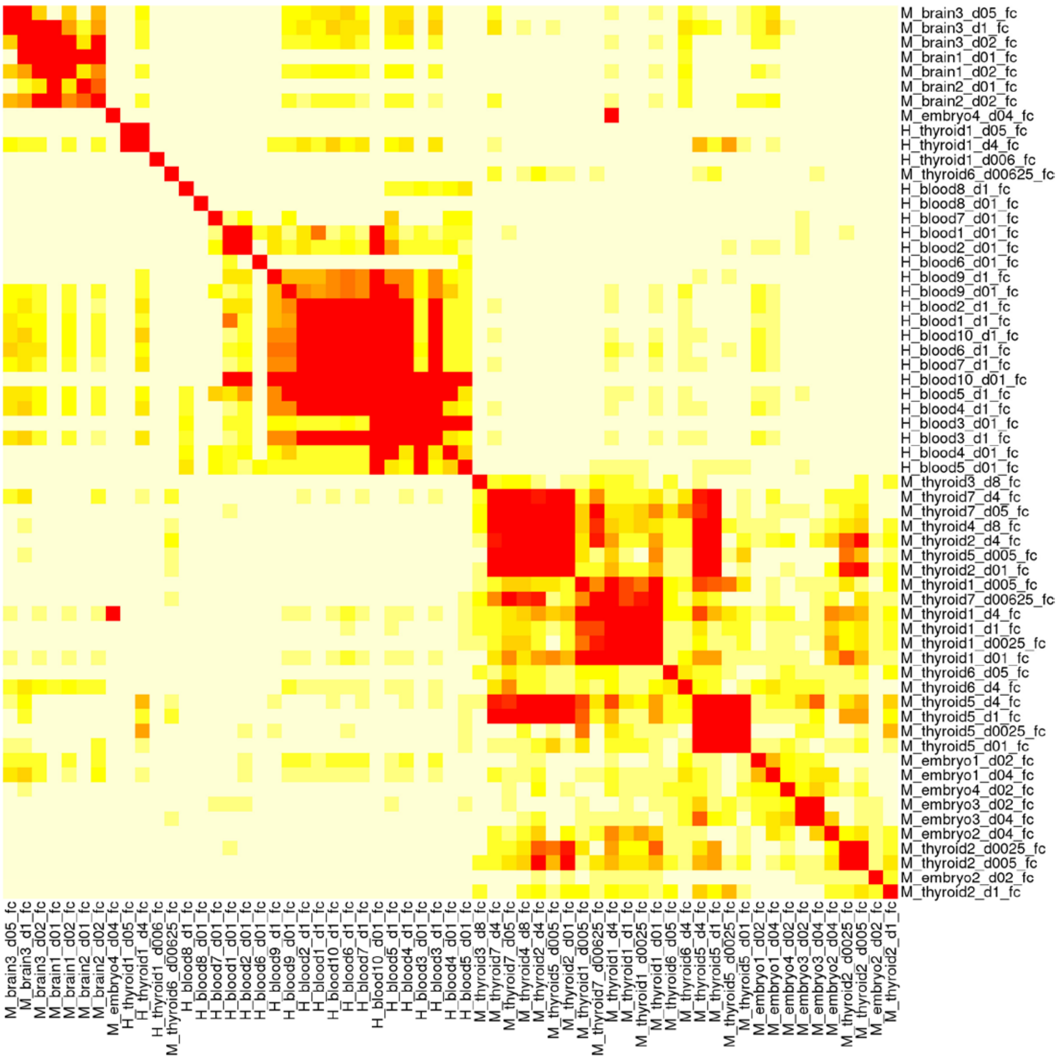

Fig. 2. Clustering of 63 experiments based on the statistical significance (P-value) of the number of overlapping genes shared between two experiments. The red colour indicates a highly significant overlap, the white colour means no significant overlap.

Differentially expressed genes

Whereas previous high-level analysis was based on simultaneous analysis of thousands of genes with the highest variance over the different conditions, analysis of the differentially expressed genes within the various conditions allows interpretation from an alternative angle. Taking into account the number of upregulated genes with a cutoff on the fold change of 1.5, the highest number of differentially expressed genes was observed for the thyroid irradiated samples with late mRNA extraction (after 9 and 18 months respectively) and for a subset of irradiated embryo (gastrula stage) samples. The former observation is in agreement with the PCA results reported in the previous paragraph. When taking into account the overlap of differentially expressed genes between various conditions, clustering based on the statistical significance of the overlap was done according to the tissue type (blood, brain, thyroid, and embryo) (Figure 2). These results suggest the presence of a tissue-specific transcriptional response towards radiation. From a species-specific point of view, the number of overlapping genes between human and mouse irradiated thyroid is very limited.

Focussing on the blood samples, a significant increase in the number of differentially expressed genes is observed when increasing the radiation dose from low (0.1 Gy) to high (1 Gy). Moreover, taking into account the overlap of differentially expressed genes between all blood samples, a tight grouping of the 1 Gy samples is observed, while a less robust overlapping set of genes is observed for the lower dose (0.1 Gy).

Functional interpretation of radiation response

Using a gene set enrichment strategy, functional enrichment of the differentially expressed genes in all irradiation experiments was performed using the KEGG pathway database (Kanehisa et al. 2010) and the Gene Ontology classification system.

No KEGG pathway or GO biological process could be found to be commonly enriched in all (or a majority of) experiments. This corresponds with the observation of differentially expressed genes being tissue specific, as discussed in previous paragraph. As such, clustering based on functional enriched categories results in grouping of the experiments based on tissue.

The most significantly enriched KEGG pathways (cut-off on p-value on 0.05) were the p53 signalling (KEGG:04115), the nucleotide excision repair (KEGG:03420) and the cell cycle (KEGG:04110) pathways, enriched respectively in 28, 22 and 13 out of the 63 experiments. The first pathway was significantly enriched in roughly all irradiated blood samples (0.1 Gy and 1 Gy) and almost all irradiated embryonic brain samples (0.2 Gy and 0.4 Gy) (Figure 3.A). With regard to the blood samples, a clear dose dependency could be observed. The other two pathways could not be assigned to a specific subset of experiments. Remarkably, the apoptosis pathway (KEGG: 04210) was only enriched in four experiments. However, zooming in onto the individual expression values of the genes belonging to this pathway revealed an interesting pattern: a consistent set of genes was found to be upregulated within the thyroid experiments where extraction of mRNA was performed after several months compared with the non-irradiated sample at the same time (Figure 3.B).

Similarly, the most enriched GO biological processes were related to DNA damage and repair (e.g. GO:0006974, GO:0006281) and cell cycle (e.g. GO:0051726, GO:0007067), all enriched in at least 14 out of the 63 experiments, with a maximum of 24 for response to DNA damage stimulus (GO:0006974). The response to DNA damage stimulus was mostly enriched in the irradiated blood samples, as well as in a subset of the embryonic experiments.

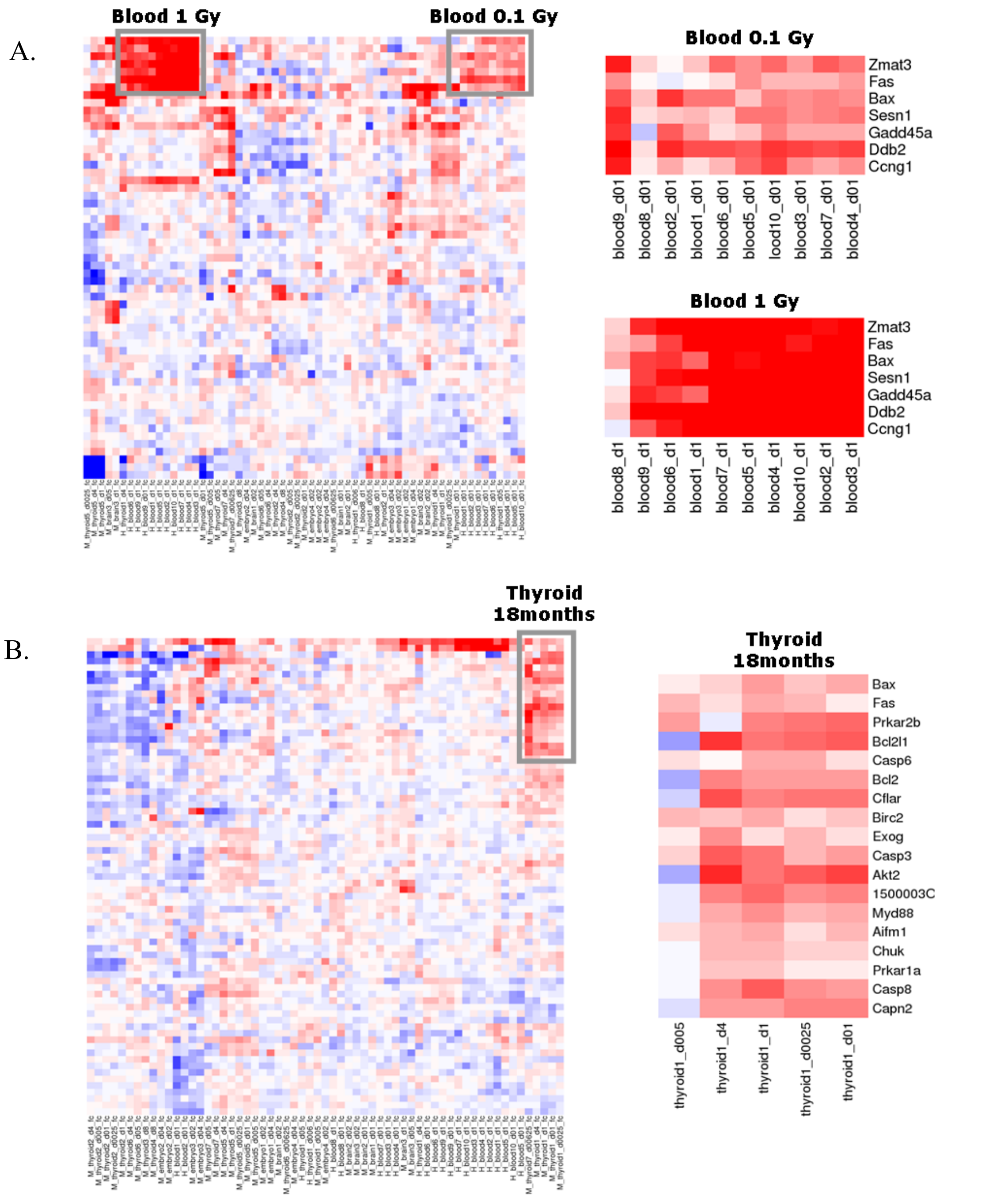

Fig. 3. Heatmap of expression of all genes belonging to A) the p53 signalling (KEGG:04115) and B) the apoptosis (KEGG:04210) pathways. For the p53 signalling pathway, a subset of genes appears to be upregulated in a dose dependent manner in the blood samples. For the apoptosis pathway, a set of genes is upregulated in the long-term experiment with mRNA extraction 18 months after irradiation with low and high doses.

Discussion

The advances in high-throughput technologies like microarrays have revolutionized the field of molecular biology. This microarray technology has already been used extensively to study the effect of low and high doses of ionising radiation at the transcriptional level. However, until now most studies have focussed on the impact of radiation on single specific tissues. In this paper, a more systematic approach was applied to study the radiation effect over various parameters.

A first remarkable observation in the PCA analysis was the aberrant transcriptional profile of two long-term experiments where mRNA was extracted from mouse thyroid tissue several months after low and high doses of irradiation. This deflected transcriptional behaviour was confirmed by the increasing number of upregulated genes in this long-term experiment compared to the acute stress experiment. These data obtained on mouse thyroid tissues are in line with the observation of Fält *et al.* (2003) who noticed that the number of differentially expressed genes increased dramatically with increasing culture time. In their experiment, primary human lymphocytes were irradiated, and mRNA was extracted at different time points (7, 17 and 55 days), showing a clear increase with the time in the number of modulated genes. Moreover, while the changes in the pattern of gene expression reported by Fält *et al.* (2003) occurred after a high radiation dose (3 Gy), the observations in the present study allow to extend their conclusions to doses as low as 0.05 and 0.025 Gy. In contrast with their study in which a limited overlap of genes was found between the immediate and long-term data, our results point towards a significant overlap with the acute response based on the number of upregulated genes shared between short- and long-term experiments. Interestingly, one of these long-term experiments (18 months) showed a significant enrichment of apoptosis related genes when compared to gene expression in a non-irradiated mouse of the same age.

Based on analysis by PCA, irradiation experiments are grouped together according to the experiments rather than by the irradiation dose. Accordingly, clustering based on the number of overlapping upregulated genes shows clear links with the tissue origin. As such, the gene expression in irradiated biological material seems to be tissue-specific rather than dose-specific. A similar, small scale analysis comparing two sets of differentially expressed genes after irradiation of kidney and brain with a high dose of 10 Gy, basically resulted in the same conclusion: only a very limited number of overlapping genes could be identified (Zhao et al. 2006), also pointing towards a tissue-specific response. Similarly, Pawlik *et al*. (2009) found that the expression profile of irradiated liver tissue correlated only to a small extent to the one in other tissues.

As a consequence of the tissue specificity of the transcriptional response after irradiation, no common pathway could be identified yet over different experiments or doses. However, certain pathways e.g. related to p53 signalling, cell cycle, DNA damage and repair were activated in a wide range of experiments, but mainly in the irradiated blood samples. This manifest activation of pathways related to DNA repair and cell cycle is probably related to the increased sensitivity of haematopoietic cells to radiation. Moreover, for the p53 signalling pathway, a dose specific transcriptional response was observed within irradiated blood samples, which is in agreement with similar results by other research groups aiming at identifying a potential set of radiation responsive biomarkers in blood (Dressman et al. 2007, Paul and Amundson 2008).

However, a new trend in disease classification also incorporates pathway information in order to classify diseases based on the activity of entire pathways instead of expression levels of individual genes (Lee et al. 2008, Svensson et al. 2006). Therefore, based on this and previous studies, the p53 signalling cascade might be a good candidate to be used as pathway-based biodosimetry classifier. However, the absence of a non-perturbed basal expression profile of this pathway in biodosimetry application is still a challenging issue (Paul and Amundson 2008).

Conclusions

First, results presented here point towards an important tissue-specific aspect in response to ionzing radiation. Secondly, gene expression results confirm the importance of performing studies on the long-term effect of radiation, as differential expression increased with longer post-irradiation times.

In this paper, the data are only interpreted from a bird's-eye view. However, a more thorough analysis is necessary and currently under progress. Generally, our studies point towards subtle but undeniable changes in gene expression. Therefore, an important future challenge is extracting these subtle changes in a statistically significant way by exploiting additional systems biology tools.

Acknowledgements

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A combined analysis of three European studies on the joint effects of radon exposure and smoking on lung cancer risk among uranium miners

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Abstract

Objectives: Three case-control studies nested in the French, German and Czech cohorts of uranium miners were conducted in the frame of Alpha-Risk, a European research project on the quantification of health risks associated with multiple radiation exposures. These studies aimed at investigating the joint effects of radon and smoking on lung cancer risk. A combined analysis of individual data of the three studies is presented.

Methods: The combined data set includes 1476 cases and 3389 matched controls. Cumulated radon exposure during employment was obtained from measurements or a job exposure matrix. Smoking habits were determined from medical archives and questionnaires. Analysis was performed by conditional logistic regression using a linear excess relative risk model.

Results: Smoking status was established for 1046 cases and 2492 controls. Ninety four percent of cases and 76% of controls were ex- or current smokers. Mean five-year lagged cumulated radon exposure was 335 Working Level Months (WLM) for cases and 211 WLM for controls. The excess relative risk per WLM adjusted for smoking was 0.79% (95% confidence-interval: 0.44 –1.41%). Lung cancer excess relative risk per unit exposure was about two times higher among non-smokers than among smokers and the results suggest a sub-multiplicative interaction between smoking and radon exposure on lung cancer mortality risk.

Conclusions: This collaborative study is the largest uranium miners case-control study on lung cancer with smoking information in Europe. It confirms the persistence of radon effect on lung cancer risk when smoking is taken into account and allows investigating the interaction between radon and smoking.

Cerebrovascular diseases in the cohort of Mayak PA nuclear workers

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Abstract

Incidence of and mortality from cerebrovascular diseases (CVD) have been studied in a cohort of 12210 workers first employed at one of the main plants of the Mayak nuclear facility during 1948–1958 and followed up to 31 December 2000. Information on external gamma doses was available for virtually all of these workers (99.9%); the mean (± standard deviation, SD) total external gamma dose was 0.91±0.95 Gy (99% percentile 3.9 Gy) for men and 0.65±0.75 Gy (99% percentile 2.99 Gy) for women. In contrast, plutonium body burden was measured only for 30% of workers; amongst those monitored, the mean (± SD) absorbed cumulative liver dose was 0.40±1.15 Gy (99% percentile 5.88 Gy) for men and 0.81±4.60 Gy (99% percentile 15.95 Gy) for women. 4418 disease cases and 753 deaths from CVD were identified in the study cohort. Having adjusted for non-radiation factors and internal alpha exposure from incorporated plutonium-239, there was statistically significant increasing trend in CVD incidence with total external gamma dose; the excess relative risk per Gy (ERR/Gy) was 0.464 (95% CI 0.360, 0.567). Much of the evidence for the raised incidence in relation to external dose arose for workers with cumulative doses above 1 Gy, although the data are consistent with a linear trend in risk with external dose. There was no statistically significant trend in CVD mortality risk with external dose or differences in mortality rates between categories for external dose. Having adjusted for non-radiation factors and external gamma exposure, there was statistically significant increasing trend in CVD incidence with total absorbed internal liver dose; the ERR/Gy was 0.155 (95% CI 0.075, 0.235). CVD incidence was statistically significantly higher among workers with a plutonium liver dose above 0.1 Gy, although the trend estimates differed between workers at different plants. There was a statistically significant increased risk of CVD mortality among workers exposed to internal alpha exposure with total liver dose of 0.1–0.5 Gy as compared with workers exposed to lower doses. There was no statistically significant trend in CVD mortality risk with absorbed internal liver dose. The risk estimates for external radiation are generally compatible with those from other large occupational studies, although the incidence data point to higher risk estimates compared to those from the Japanese A-bomb survivors.

Introduction

Ischemic heart disease and cerebrovascular diseases (CVD) are currently among the main death causes in developed countries. CVD are multifactorial diseases, and different endogenous (genetic predisposition, gender, age, hypertension etc.) and exogenous (smoking, emotional stress etc.) factors contribute to their development. Over the last two decades several studies have examined the possible effects of ionizing radiation exposure on circulatory diseases, including CVD. Most of the studies (McGeoghegan et al 2008; Kreuzer et al 2006; Muirhead et al 2009; Shimizu et al 2010; Vrijheid et al 2007) focused on CVD mortality, whereas several studies (Yamada et al 2004; De Bruin et al 2009; Ivanov et al 2006) focused on CVD incidence.

This study was aimed at estimating risks of both CVD incidence and mortality in the cohort of Mayak PA nuclear workers first employed at one of the main plants (reactors, radiochemical, or plutonium) during 1948-1958 in relation to external gamma and internal alpha exposures, taking non-radiation factors into account.

Material and methods

Mayak PA began operations in 1948 as the first and largest nuclear weapons facility in the former Soviet Union and included all the plants necessary for weapon-grade plutonium production, namely, reactors, radiochemical plant, plutonium production plant and auxiliary plants.

From the first days of Mayak PA operation, the special system of personnel medical observation included an obligatory pre-employment medical examination and routine medical examinations of all the workers based on a common standard program. After quitting their job at Mayak, if the former worker stayed in Ozyorsk, he/she was examined at the same specialized medical hospital based on the same standard program. This system of medical observation of Mayak PA personnel health allowed a unique archive of primary medical data to be accumulated and formed the basis for establishment of the unique "Clinic" medical-dosimetry database for the Mayak PA workers cohort (Azizova et al 2008).

The study cohort included 12210 Mayak PA workers, 3552 (29.1%) of whom were women, first employed at one of the main plants from the start of operations through the end of 1958 independently of gender, age, nationality, occupation, and other characteristics. The method of identifying this cohort has been described previously elsewhere (Koshurnikova et al 1988, 1998a, 1998b, 1999).

Vital status as of 31 December 2000 was known for 88.4% of cohort members; of these workers, 52.7% were known to have died and 47.3% were known to be alive as of that time. 53.7% of the 12210 members of the study cohort were known to have left Ozyorsk by 31 December 2000. For persons who continued to be residents of Ozyorsk, vital status was known for all but one person (99.98%). Cause of death was known for 93.5% of deceased cohort members. Morbidity data for the whole period of follow-up were collected for 11597 (95.0%) workers in the study cohort. Only for 5.0% of workers could information not be collected, owing to the loss of their medical documentation. It should be noted that the CVD incidence analysis was restricted to the period of residence in Ozyorsk.

Data on vital status and on dates and causes of death for those workers who migrated from Ozyorsk were provided by the SUBI Epidemiology Laboratory; data on date and causes of death for those Ozyorsk residents whose medical cards and/or case histories had been lost, were provided by the SUBI Occupational Health Laboratory.

Individual dosimetric control of external exposure was performed at Mayak PA from the beginning of operations there. Regular monitoring of internal exposure among those who might have been exposed to plutonium-239 began later, during the 1960s (Vasilenko et al 2007a; Khokhryakov et al 2000). The results of individual monitoring of external and internal exposure were recorded in individual dosimetric cards and journals. The data contained in these documents formed the basis for the establishment of the dosimetric database of Mayak PA workers (Vasilenko et al 2007a, 2007b; Bess et al 2007; Smetanin et al 2007a, 2007b; etc).

Dose estimates from the *Mayak-Doses 2005* dosimetric system were used in this study. Annual doses of external gamma exposure were available practically for all persons in the study cohort (99.9%). The mean (± SD) total gamma dose for the whole period of work at Mayak was 0.91±0.95 Gy (99% percentile 3.9 Gy) for men and 0.65±0.75 Gy (99% percentile 2.99 Gy) for women. The range of total gamma doses was very wide, with 32.6% having a total gamma dose greater than 1 Gy.

Plutonium body burden was measured only for 30.0% of workers. Among workers monitored for plutonium exposure, absorbed dose to liver from alpha radiation was used as a surrogate for the dose to muscle; this latter dose is likely to be similar to the dose to blood vessels and the chambers of the heart. Although doses to the liver and muscle would differ, they should be highly correlated with each other. Consequently, the liver dose can be used to look for any *dose–response* relationship between plutonium exposure and circulatory disease. Amongst those who were monitored, the mean (± SD) total liver dose was 0.40±1.15 Gy (99% percentile 5.88 Gy) for men and 0.81±4.60 Gy (99% percentile 15.95 Gy) for women.

Data on occupational histories and external gamma exposures for the study cohort were provided by the Mayak PA Radiation Safety Department; data on internal alpha exposure from incorporated plutonium-239 were provided by the SUBI Internal Dosimetry Department.

This study focuses on incidence of and mortality from CVD (ICD-9 codes: 430–438). Follow-up started on the date of first employment at one of the main plants and continued until the earliest of: date of the first diagnosis of CVD (for the incidence analysis) or the date of death from whatever cause (for both the mortality analysis and the incidence analysis); 31 December 2000 for those known to be alive at that time; the recorded date of departure from Ozyorsk (for the incidence analysis); and the date of "last medical information" in the case of unknown vital status. Comparisons were performed within the Mayak PA workers cohort first employed during 1948–1958.

The analyses included the calculation of relative risks (RRs) for categories of one or more factors, having adjusted for other variables. The relative risks for these categorical analyses were calculated by maximum likelihood, using the AMFIT module of EPICURE (Preston et al 1993). 95% confidence intervals for the RRs and p-values from tests of statistical significance were obtained via likelihood-based methods, using AMFIT. Attention was initially directed to non-radiation factors, following which measures of radiation exposure were analysed with adjustment (through stratification) for non-radiation factors. Analyses of internal radiation exposures were restricted to workers known to have been monitored for possible plutonium exposure.

In addition to the categorical analyses, models for trends in disease rates with level of radiation exposures were also fitted to the data, using Poisson regression methods. These models again were fitted using the AMFIT module in EPICURE. In particular, the excess relative risk (ERR) (ie. the relative risk minus 1) was modeled by a linear trend with external or internal dose, with adjustment for non-radiation factors.

In these main analyses, adjustments were made – through stratification – for gender, attained age, calendar period, period of first employment at the main plants of Mayak PA, plant, smoking and alcohol consumption.

Sensitivity analyses were conducted to examine the impact of: a) modifying the set of non-radiation factors (extra adjustment for hypertension, body mass index, employment duration) for which adjustment was made in the analyses of radiation factors; b) restricting the mortality follow-up (like that of incidence) to Ozyorsk, because some migrants were lost to follow-up and because of lower autopsy rates among those who left the city; c) adjusting for internal dose in analyses of external dose and vice versa; d) using various lag periods for external and internal doses. Furthermore, examination was made of how radiation risks might vary by gender, or between plants at Mayak or by attained age.

To allow for the possibility that radiation might affect CVD risk by modifying levels of blood pressure (Preston et al 2003; Ivanov et al 2001) and body mass index (Telnov 1985), the level of these factors at the time of preliminarily medical examination (before employment at Mayak PA) was considered, in order to avoid systematic errors that might arise through adjusting for values of these factors at later times. In contrast, smoking and alcohol consumption were classified at the time of last information (for the mortality analysis) or at the time of last information prior to the first diagnosis of CVD (for the incidence analysis).

Results

By 31 December 2000, 4418 disease cases of CVD were diagnosed during 197344 person-years of follow-up and 753 death cases from CVD were identified during 443350 person-years of follow-up.

Non-radiation factors

It is known that CVD are multifactorial diseases, therefore analyses of incidence and mortality risks in relation to non-radiation factors were performed first. Our analyses revealed statistically significant effects of well-known factors such as gender, age, hypertension, body mass index and smoking, which were taken into account in the analyses of radiation risks, either in the main analysis or in sensitivity analyses.

Radiation factors

External gamma exposure: Table 1 shows that CVD incidence was statistically significantly higher among workers chronically exposed to external gamma exposure with a total dose above 0.5 Gy as compared with workers externally exposed to lower doses.

Table 1. RRs and ERR (95% CI) for CVD incidence and mortality in relation to total dose of external gamma exposure (main analysis based on a zero year lag period).

	RRs (vs. <0.5 Gy)		ERR/Gy
	0.5-1.0 Gy	>1.0 Gy	
Incidence	1.137 (1.036, 1.248)	1.599 (1.465, 1.746)	0.464 (0.360, 0.567)
Mortality	1.150 (0.923, 1.434)	0.994 (0.799, 1.237)	-0.022 (-0.117, 0.073)

Sensitivity analyses (results not shown) revealed that when comparing gamma doses of 0.5-1.0 Gy with lower doses, statistically significantly raised risks of CVD incidence were seen among radiochemical plant workers and among females; in other sub-groups, the findings were very close to statistical significance. CVD incidence was statistically significantly higher among workers with a gamma dose above 1.0 Gy relative to workers with a gamma dose below 0.5 Gy, irrespective of the lag period used, whether additional adjustment was made for other non-radiation factors and internal exposure as well as whether analysis was restricted to workers at different plants or by gender.

There was a statistically significant increasing trend in CVD incidence with increasing external dose (Table 1). Sensitivity analyses (results not shown) revealed that adjustments for other non-radiation factors and internal exposure as well as restriction to workers at different plants or by gender had little impact on this finding. There was also no evidence of variation in the ERR per Gy by attained age.

There was no statistically significant trend in CVD mortality risk with external dose, and for the most part mortality rates did not differ to a statistically significant extent between categories for external dose.

Internal alpha exposure: Table 2 shows that CVD incidence was statistically significantly higher among workers exposed to total absorbed dose to liver from internal alpha exposure above 0.1 Gy, as compared with workers exposed to lower doses. This finding held irrespective of the lag period used, whether adjustment was made for other non-radiation factors and external exposure as well as whether analysis was restricted to workers at different plants or by gender (results not shown).

Table 2. RRs and ERR (95% CI) for CVD incidence and mortality in relation to total absorbed liver dose from internal alpha exposure (main analysis based on a zero year lag period).

	RRs (vs. <0.1 Gy)		ERR/Gy
	0.1-0.5 Gy	>0.5 Gy	
Incidence	1.233 (1.126, 1.350)	1.576 (1.346, 1.844)	0.155 (0.075, 0.235)
Mortality	1.401 (1.020, 1.924)	1.047 (0.609, 1.800)	0.120 (-0.116, 0.356)

There was a statistically significant increasing trend in CVD incidence with increasing internal liver dose from alpha exposure. The ERR/Gy for CVD incidence increased with increasing lag period, with values of 0.155 (95% CI 0.075, 0.235), 0.218 (95% CI 0.111, 0.326) and 0.330 (95% CI 0.174, 0.487), 0.508 (95% CI 0.269, 0.747) and 0.853 (95% CI 0.443, 1.262) based on 0, 5, 10, 15 and 20 year lags respectively.

There was a non-linearity in the trend based on zero lag, although the evidence for non-linearity appeared to be weaker when a 5-year lag was used. Adjustments for arterial hypertension, body mass index, duration of work and external exposure had little impact on the estimated trend (results not shown). Much of the evidence for this trend appeared to arise at attained ages of 50 years or more, but the variation in the ERR/Gy by attained age was not statistically significant. Furthermore, the evidence for a dose trend came mainly from males rather than females, as well as from workers at the radiochemical plant rather than plutonium production plant workers.

CVD mortality was statistically significantly raised among workers exposed to internal alpha exposure in total liver doses of 0.1–0.5 Gy as compared with workers exposed to lower doses irrespective of lag period and whether adjustment was made for other non-radiation factors as well as whether follow-up was restricted to Ozyorsk and whether adjustment was made for external exposure. However, when comparing internal doses of >0.5 Gy with doses <0.1 Gy, statistically significantly raised risks were not seen for CVD mortality, possibly reflecting the relatively small number of deaths in this subgroup.

There was no statistically significant trend in CVD mortality in relation to total absorbed liver dose from internal alpha exposure.

Discussion

Our analyses of CVD incidence and mortality revealed statistically significant effects of well-known factors such as gender, age, hypertension, body mass index and smoking, which are consistent with findings of other studies. In contrast, our analyses did not reveal any statistically significant effect of alcohol consumption on CVD incidence or mortality, either for males and females, whereas some studies (Donahue et al 1986; etc) indicated an increase in stroke mortality associated with excess alcohol consumption.

The evidence for associations with radiation in this study relates mainly to CVD incidence. For CVD mortality, the data did not show a statistically significantly increasing trend with either external or internal dose, and the estimates of these trends were lower than those for CVD incidence. There were far fewer deaths from CVD than cases in this study. A complication to interpretation is the lack of knowledge as to those tissues or organs for which radiation exposure might increase the risk of CVD, which is particularly problematic in the case of plutonium intakes. For this analysis, liver dose has been used as a surrogate for the dose to muscle, which is likely to be similar to the dose to blood vessels and the chambers of the heart. Furthermore, the liver and muscle doses should be highly correlated with each other. However, there is uncertainty about which tissue or organ dose is appropriate for this type of analysis. A further complication relates to uncertainties in estimates for internal doses for Mayak workers. In addition, whilst the results for external doses appeared to be broadly consistent with a linear *dose-response* (albeit with uncertainties in estimates at relatively low doses), the results for internal doses appeared to show some degree of non-linearity. Also, whilst the trend estimates in relation to external exposure were similar across sub-groups of workers, the trend in CVD incidence with internal dose differed not only between males and females but also between workers at the radiochemical plant and the plutonium production plant. It should be noted that – because the dose to the liver from intakes of plutonium would be greater than that to the circulatory system – the ERR/Gy estimated here based on liver dose would be lower than that based on dose to blood vessels and the chambers of the heart. At present information is not available from other studies of populations exposed to plutonium that would allow comparison of risk estimates for CVD in relation to such exposures. For these reasons, the findings in relation to internal exposure need to be interpreted with caution.

Comparison with other studies

Table 3 presents estimates of the ERR per Gy from the current study and from other studies of groups exposed to external low-LET radiation in which this trend was estimated.

Table 3. Estimates of ERR per Gy (95% CI) for CVD following exposure to external low-LET radiation.

Cohort	Mean total dose (Gy)	Mortality or incidence?	Lag period (years)	No. of deaths or cases	ERR/Gy
Japanese A-bomb survivors: Life Span Study (Shimizu et al 2010)	0.20	Mortality	5	9622	0.09 (95% CI 0.01, 0.17)
Japanese A-bomb survivors: Adult Health Study (Yamada et al 2004)	0.57	Incidence	13	729	0.07 (95% CI -0.08, 0.24)
Mayak PA workers (this study)	0.84	Mortality	10	744	-0.02 (95% CI -0.12, 0.08)
Mayak PA workers (this study)	0.84	Incidence	10	3840	0.45 (95% CI 0.34, 0.56)
Nuclear workers (international) (Vrijheid et al 2007)	0.018	Mortality	10	1224	0.88 (95% CI -0.67, 3.16)
BNFL workers (UK) (McGeoghegan et al 2008)	0.053	Mortality	15	1018	0.43 (90% CI -0.10, 1.12)
UK National Registry for Radiation Workers (Muirhead et al 2009)	0.025	Mortality	10	1817	0.16 (95% CI -0.42, 0.99)
German uranium miners (Kreuzer et al 2006)	0.041	Mortality	5	1297	0.09 (95% CI -0.6, 0.8)
Chernobyl recovery operations workers (Ivanov et al 2006)	0.109	Incidence	–	12832	0.45 (95% CI 0.11, 0.80)

Whilst the estimated ERR/Gy for mortality among Mayak workers is consistent with the A-bomb survivor findings, some of the occupational studies point towards higher risk estimates. The statistically precise estimate for CVD incidence among Mayak workers is higher than that for the A-bomb survivors but is consistent with the incidence estimate for Chernobyl recovery operation workers and the mortality estimates from the international and BNFL worker studies. The central estimates for the ERR/Gy from the UK National Registry for Radiation Workers and the German uranium miner study are not statistically significantly different from zero and are close to the estimates from the Mayak mortality data, but the confidence intervals also encompass the Mayak incidence findings.

Conclusions

Having adjusted for non-radiation factors and internal alpha exposure, there was an increasing trend in CVD incidence – but not mortality – among Mayak PA workers with total external gamma dose. Much of the evidence for a raised risk of CVD incidence in relation to external dose arises for workers with cumulative doses above 1 Gy. Although the dose response is consistent with linearity, the statistical power to detect non-linearity at external doses below 1 Gy was low. Having adjusted for non-radiation factors and external gamma exposure, there was an increasing trend in CVD incidence among Mayak PA workers with total internal alpha dose to liver. CVD incidence was statistically significantly higher among workers with a plutonium liver dose above 0.1 Gy, although the trend estimates differed between workers at different plants. There was statistically significant increased risk of CVD mortality among workers exposed to internal alpha exposure with total liver dose of 0.1–0.5 Gy as compared with workers exposed to lower doses. There was no a statistically significant trend of CVD mortality risk with absorbed internal liver dose or difference in mortality rates between categories for internal dose. The risk estimates for external radiation are generally compatible with those from other large occupational studies, although the incidence data point to higher risk estimates compared to those from the Japanese A-bomb survivors.

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Ischemic heart disease in the cohort of Mayak PA nuclear workers

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Abstract

Incidence of and mortality from ischemic heart disease (IHD) have been studied in a cohort of 12210 workers first employed at one of the main plants of the Mayak nuclear facility during 1948–1958 and followed up to 31 December 2000. Information on external gamma doses was available for virtually all of these workers (99.9%); the mean (± standard deviation, SD) total external gamma dose was 0.91±0.95 Gy (99% percentile 3.9 Gy) for men and 0.65±0.75 Gy (99% percentile 2.99 Gy) for women. In contrast, plutonium body burden was measured only for 30% of workers; amongst those monitored, the mean (± SD) absorbed cumulative liver dose was 0.40±1.15 Gy (99% percentile 5.88 Gy) for men and 0.81±4.60 Gy (99% percentile 15.95 Gy) for women. 3751 disease cases and 1495 deaths from IHD were identified in the study cohort. Having adjusted for non-radiation factors, there was statistically significant increasing trend in IHD incidence with total external gamma dose; the excess relative risk per Gy (ERR/Gy) was 0.109 (95% CI 0.049, 0.168). This trend with external dose was little changed after adjusting for internal dose. Much of the evidence for the raised incidence in relation to external gamma dose arose for workers with cumulative doses above 1 Gy, although the data are consistent with a linear trend in risk with external dose. The trend with external dose in IHD mortality was not statistically significantly greater than zero, but was consistent with the corresponding trend in IHD incidence even once adjustment for internal dose was made. There was stronger evidence for an association with internal alpha dose to liver in the IHD mortality data than in the corresponding incidence data, although the associated confidence intervals overlap. Furthermore, the estimated trend in IHD mortality with internal dose was lower and not statistically significant once adjustment was made for external dose. The risk estimates for IHD in relation to external radiation are generally compatible with those from other large occupational studies and the Japanese A-bomb survivors.

Introduction

The first evidence of excess mortality from heart disease following radiation exposure came from the study of Japanese A-bomb survivors, who received single whole-body doses in the range 0-4 Gy; the most recent findings are described by Shimizu et al (2010). However, the corresponding data on ischemic heart disease (IHD) incidence in a subset of these survivors provided less evidence of an association (Yamada et al 2004). An increased risk of cardiovascular diseases has also been observed in patients who underwent radiotherapy for breast cancer or for Hodgkin lymphoma with doses of 40 Gy or more to mantle or limited fields (UNSCEAR 2006). In contrast, studies of radiation workers have yielded mixed findings. McGeoghegan et al (2008) reported an association between mortality from non-cancer causes of death, particularly circulatory disease, and external exposure to ionizing radiation in a cohort of nuclear industry workers with a mean cumulative external dose of 0.053 Sv and a 99% percentile of 0.59 Sv. However, a study of radiation workers from 15 countries did not find a statistically significant association between circulatory disease mortality and radiation dose (mean cumulative radiation dose of 0.019 Sv, with some workers receiving more than 0.5 Sv), although – in common with other studies of nuclear workers – the power of this analysis was low (Vrijheid et al 2007).

This study was aimed at estimating risks of both IHD incidence and mortality in the cohort of Mayak PA nuclear workers first employed at one of the main plants (reactors, radiochemical, or plutonium) during 1948-1958 in relation to external gamma and internal alpha exposures, taking non-radiation factors into account.

Material and methods

Mayak PA began operations in 1948 as the first and largest nuclear weapons facility in the former Soviet Union and included all the plants necessary for weapon-grade plutonium production, namely, reactors, radiochemical plant, plutonium production plant and auxiliary plants.

From the first days of Mayak PA operation, the special system of personnel medical observation included an obligatory pre-employment medical examination and routine medical examinations of all the workers based on a common standard program. After quitting their job at Mayak, if the former worker stayed in Ozyorsk, he/she was examined at the same specialized medical hospital based on the same standard program. This system of medical observation of Mayak PA personnel health allowed a unique archive of primary medical data to be accumulated and formed the basis for establishment of the unique “Clinic” medical-dosimetry database for the Mayak PA workers cohort (Azizova et al 2008).

The study cohort included 12210 Mayak PA workers, 3552 (29.1%) of whom were women, first employed at one of the main plants from the start of operations through the end of 1958 independently of gender, age, nationality, occupation, and other characteristics. The method of identifying this cohort has been described previously elsewhere (Koshurnikova et al 1988, 1998a, 1998b, 1999).

Vital status as of 31 December 2000 was known for 88.4% of cohort members; of these workers, 52.7% were known to have died and 47.3% were known to be alive as of that time. 53.7% of the 12210 members of the study cohort were known to have left Ozyorsk by 31 December 2000. For persons who continued to be residents of Ozyorsk, vital status was known for all but one person (99.98%). Cause of death was known for 93.5% of deceased cohort members. Morbidity data for the whole period of follow-up were collected for 11597 (95.0%) workers in the study cohort. Only for 5.0% of workers could information not be collected, owing to the loss of their medical documentation. It should be noted that the CVD incidence analysis was restricted to the period of residence in Ozyorsk.

Data on vital status and on dates and causes of death for those workers who migrated from Ozyorsk were provided by the SUBI Epidemiology Laboratory; data on date and causes of death for those Ozyorsk residents whose medical cards and/or case histories had been lost, were provided by the SUBI Occupational Health Laboratory.

Individual dosimetric control of external exposure was performed at Mayak PA from the beginning of operations there. Regular monitoring of internal exposure among those who might have been exposed to plutonium-239 began later, during the 1960s (Vasilenko et al 2007a; Khokhryakov et al 2000). The results of individual monitoring of external and internal exposure were recorded in individual dosimetric cards and journals. The data contained in these documents formed the basis for the establishment of the dosimetric database of Mayak PA workers (Vasilenko et al 2007a, 2007b; Bess et al 2007; Smetanin et al 2007a, 2007b; etc).

Dose estimates from the *Mayak-Doses 2005* dosimetric system were used in this study. Annual doses of external gamma exposure were available practically for all persons in the study cohort (99.9%). The mean (± SD) total gamma dose for the whole period of work at Mayak was 0.91±0.95 Gy (99% percentile 3.9 Gy) for men and 0.65±0.75 Gy (99% percentile 2.99 Gy) for women. The range of total gamma doses was very wide, with 32.6% having a total gamma dose greater than 1 Gy.

Plutonium body burden was measured only for 30.0% of workers. Among workers monitored for plutonium exposure, absorbed dose to liver from alpha radiation was used as a surrogate for the dose to muscle; this latter dose is likely to be similar to the dose to blood vessels and the chambers of the heart. Although doses to the liver and muscle would differ, they should be highly correlated with each other. Consequently, the liver dose can be used to look for any *dose–response* relationship between plutonium exposure and circulatory disease. Amongst those who were monitored, the mean (± SD) total liver dose was 0.40±1.15 Gy (99% percentile 5.88 Gy) for men and 0.81±4.60 Gy (99% percentile 15.95 Gy) for women.

Data on occupational histories and external gamma exposures for the study cohort were provided by the Mayak PA Radiation Safety Department; data on internal alpha exposure from incorporated plutonium-239 were provided by the SUBI Internal Dosimetry Department.

This study focuses on incidence of and mortality from IHD (ICD-9 codes: 410–414). Follow-up started on the date of first employment at one of the main plants and continued until the earliest of: date of the first diagnosis of CVD (for the incidence analysis) or the date of death from whatever cause (for both the mortality analysis and the incidence analysis); 31 December 2000 for those known to be alive at that time; the recorded date of departure from Ozyorsk (for the incidence analysis); and the date of "last medical information" in the case of unknown vital status. Comparisons were performed within the Mayak PA workers cohort first employed during 1948–1958.

The analyses included the calculation of relative risks (RRs) for categories of one or more factors, having adjusted for other variables. The relative risks for these categorical analyses were calculated by maximum likelihood, using the AMFIT module of EPICURE (Preston et al 1993). 95% confidence intervals for the RRs and p-values from tests of statistical significance were obtained via likelihood-based methods, using AMFIT. Attention was initially directed to non-radiation factors, following which measures of radiation exposure were analysed with adjustment (through stratification) for non-radiation factors. Analyses of internal radiation exposures were restricted to workers known to have been monitored for possible plutonium exposure.

In addition to the categorical analyses, models for trends in disease rates with level of radiation exposures were also fitted to the data, using Poisson regression methods. These models again were fitted using the AMFIT module in EPICURE. In particular, the excess relative risk (ERR) (ie. the relative risk minus 1) was modeled by a linear trend with external or internal dose, with adjustment for non-radiation factors.

In these main analyses, adjustments were made – through stratification – for gender, attained age, calendar period, period of first employment at the main plants of Mayak PA, plant, smoking and alcohol consumption.

Sensitivity analyses were conducted to examine the impact of: a) modifying the set of non-radiation factors (extra adjustment for hypertension, body mass index, employment duration) for which adjustment was made in the analyses of radiation factors; b) restricting the mortality follow-up (like that of incidence) to Ozyorsk, because some migrants were lost to follow-up and because of lower autopsy rates among those who left the city; c) adjusting for internal dose in analyses of external dose and vice versa; d) using various lag periods for external and internal doses. Furthermore, examination was made of how radiation risks might vary by gender, or between plants at Mayak or by attained age.

To allow for the possibility that radiation might affect IHD risk by modifying levels of blood pressure (Preston et al 2003; Ivanov et al 2001) and body mass index (Telnov 1985), the level of these factors at the time of preliminarily medical examination (before employment at Mayak PA) was considered, in order to avoid systematic errors that might arise through adjusting for values of these factors at later times. In contrast, smoking and alcohol consumption were classified at the time of last information (for the mortality analysis) or at the time of last information prior to the first diagnosis of IHD (for the incidence analysis).

Results

By 31 December 2000, 3751 disease cases of IHD were diagnosed during 205249 person-years of follow-up and 1495 death cases from IHD were identified during 443350 person-years of follow-up.

Non-radiation factors

Our analyses revealed statistically significant effects of well-known factors such as gender, age, hypertension, body mass index and smoking, which were taken into account in the analyses of radiation risks, either in the main analysis or in sensitivity analyses.

Radiation factors

<u>External gamma exposure</u>: Table 1 shows that IHD incidence was statistically significantly higher among workers chronically exposed to external gamma exposure

with a total dose above 1.0 Gy as compared with workers externally exposed to doses below 0.5 Gy.

Table 1. RRs and ERR (95% CI) for IHD incidence and mortality in relation to total dose of external gamma exposure (main analysis based on a zero year lag period).

	RRs (vs. <0.5 Gy)		ERR/Gy
	0.5-1.0 Gy	>1.0 Gy	
Incidence	1.017 (0.919, 1.126)	1.198 (1.087, 1.320)	0.109 (0.049, 0.168)
Mortality	0.918 (0.778, 1.084)	1.115 (0.959, 1.295)	0.065 (-0.017, 0.148)

Sensitivity analyses (results not shown) revealed that IHD incidence was statistically significantly higher among workers with a gamma dose above 1.0 Gy relative to workers with a gamma dose below 0.5 Gy, irrespective of the lag period used, whether additional adjustment was made for other non-radiation factors and internal exposure. The RRs above 1 Gy for IHD incidence were statistically significant only among men as well as reactor or plutonium plant workers.

There was a statistically significant increasing trend in IHD incidence with increasing external gamma dose (Table 1). Sensitivity analyses (results not shown) revealed that using different lag periods, adjustments for other non-radiation factors and internal exposure had little impact on this finding. Much of the evidence of the raised risk related to men as well as reactor and plutonium plant workers. The findings were too imprecise to judge whether the relative risk in relation to external dose varied by attained age.

There was no statistically significant trend in IHD mortality risk with external dose, and for the most part mortality rates did not differ to a statistically significant extent between categories for external dose.

Internal alpha exposure: Table 2 shows that IHD incidence and mortality were statistically significantly higher among workers with a total absorbed dose to liver from internal alpha exposure of either 0.1-0.5 Gy or above 0.5 Gy, as compared with workers exposed to doses below 0.1 Gy.

Table 2. RRs and ERR (95% CI) for IHD incidence and mortality in relation to total absorbed liver dose from internal alpha exposure (main analysis based on a zero year lag period).

	RRs (vs. <0.1 Gy)		ERR/Gy
	0.1-0.5 Gy	>0.5 Gy	
Incidence	1.171 (1.060, 1.295)	1.225 (1.036, 1.447)	0.032 (-0.014, 0.079)
Mortality	1.332 (1.082, 1.640)	1.591 (1.158, 2.185)	0.276 (0.050, 0.501)

Table 2 shows that the estimates of the RRs and of the ERR/Gy were higher in the mortality data than in the corresponding incidence data. These findings were maintained when different lag periods were used (with a tendency for the estimated ERR/Gy to increase with increase lag), when (for the mortality analysis) follow-up was restricted to Ozyorsk, and – for the most part – when additional adjustment was made for other non-radiation factors (results not shown). For IHD mortality, the trend in risk with internal liver dose was statistically significantly greater than zero, but the estimated ERR/Gy was lower and was not statistically significant once adjustment was made for external dose. Overall, there was no statistically significant trend in IHD incidence with internal liver dose, either with or without adjustment for external dose, although the ERR estimate was greater for males than for females and the difference between the gender-specific values was statistically significant. The findings were too imprecise to judge whether the relative risk in relation to internal dose varied by attained age.

Discussion

Our analyses of IHD incidence and mortality revealed statistically significant effects of well-known factors such as gender, age, hypertension, body mass index and smoking, which are consistent with findings of other studies. In contrast, our analyses did not reveal any statistically significant effect of alcohol consumption on IHD incidence or mortality, either for males and females.

There was a statistically significant increasing trend in IHD incidence with total external gamma dose. Much of the evidence for this trend in IHD incidence appeared to relate to external doses in excess of 1 Gy. Adjustments to other non-radiation factors as well as internal exposure did not have much effect on this trend. The trend with external dose in IHD mortality, whilst not statistically significantly greater than zero, was consistent with the statistically significantly trend estimate in IHD incidence. Among workers with internal liver doses there was stronger evidence for an association with these doses in the IHD mortality data than in the corresponding incidence data, although the associated confidence intervals overlap. Furthermore, the estimated trend in IHD mortality with internal dose was maintained when adjustment was made for non-radiation factors, but was lower and not statistically significant once adjustment was made for external dose.

A complication to interpretation is the lack of knowledge as to those tissues or organs for which radiation exposure might increase the risk of IHD, which is particularly problematic in the case of plutonium intakes. For this analysis, liver dose has been used as a surrogate for the dose to muscle, which is likely to be similar to the dose to blood vessels and the chambers of the heart. Furthermore, the liver and muscle doses should be highly correlated with each other. However, there is uncertainty about which tissue or organ dose is appropriate for this type of analysis. A further complication relates to uncertainties in estimates for internal doses for Mayak workers. It should be noted that – because the dose to the liver from intakes of plutonium would be greater than that to the circulatory system – the ERR/Gy estimated here based on liver dose would be lower than that based on dose to blood vessels and the chambers of the heart. At present information is not available from other studies of populations exposed to plutonium that would allow comparison of risk estimates for IHD in relation to such exposures. For these reasons, the findings in relation to internal exposure need to be interpreted with caution.

Comparison with other studies

Table 3 presents estimates of the ERR per Gy from the current study and from other studies of groups exposed to external low-LET radiation in which this trend was estimated. It can be seen that the estimates from the various studies are mostly consistent. In particular, the incidence findings for Mayak PA workers are consistent with incidence and mortality results for the Japanese atomic studies. The wide confidence interval from the 15-country nuclear worker study means that these results are not greatly informative, whilst the morbidity data reported by Chernobyl clean-up workers – although yielding a trend estimate higher than that found here – are consistent with the Mayak findings. The only finding that is inconsistent with the Mayak results arises for workers at BNFL in the UK, whereas a larger study of UK radiation workers that included BNFL workers yielded a lower ERR estimate.

Table 3. Estimates of ERR per Gy (95% CI) for IHD following exposure to external low-LET radiation.

Cohort	Mean total dose (Gy)	Mortality or incidence?	Lag period (years)	No. of deaths or cases	ERR/Gy
Japanese A-bomb survivors: Life Span Study (Shimizu et al 2010)	0.20	Mortality	5	3252	0.02 (95% CI 0.10, 0.15)
Japanese A-bomb survivors: Adult Health Study (Yamada et al 2004)	0.57	Incidence	13	1546	0.05 (95% CI 0.05, 0.16)
Mayak PA workers (this study)	0.84	Mortality	10	1461	0.07 (95% CI 0.02, 0.15)
Mayak PA workers (this study)	0.84	Incidence	10	3133	0.12 (95% CI 0.05, 0.19)
Nuclear workers (international) (Vrijheid et al 2007)	0.018	Mortality	10	5821	-0.01 (95% CI 0.59, 0.69)
BNFL workers (UK) (McGeoghegan et al 2008)	0.053	Mortality	15	3567	0.70 (95% CI 0.33, 1.11)
UK National Registry for Radiation Workers (Muirhead et al 2009)	0.025	Mortality	10	7168	0.26 (95% CI 0.05, 0.61)
Chernobyl recovery operations workers (Ivanov et al 2006)	0.109	Incidence	–	10942	0.41 (95% CI 0.05, 0.78)

Conclusions

Having adjusted for non-radiation factors, there was statistically significant increasing trend in IHD incidence with total external gamma dose. This trend with external dose was little changed after adjusting for internal dose. Much of the evidence for the raised incidence in relation to external gamma dose arose for workers with cumulative doses above 1 Gy, although the data are consistent with a linear trend in risk with external dose. The trend with external dose in IHD mortality was not statistically significantly greater than zero, but was consistent with the corresponding trend in IHD incidence even once adjustment for internal dose was made. There was stronger evidence for an association with internal alpha dose to liver in the IHD mortality data than in the corresponding incidence data, although the associated confidence intervals overlap.

Furthermore, the estimated trend in IHD mortality with internal dose was lower and not statistically significant once adjustment was made for external dose. The risk estimates for IHD in relation to external radiation are generally compatible with those from other large occupational studies and the Japanese A-bomb survivors.

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Risk of thyroid cancer among Chernobyl liquidators

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Abstract

Aims: While the increased risk of thyroid cancer is well demonstrated in children exposed to radioactive iodines in the most contaminated areas around the Chernobyl power plant, the effect of exposure on adults remains unclear. The objective of the study was to evaluate the radiation-induced risk of this disease among Chernobyl liquidators.

Methods: A collaborative case-control study of thyroid cancer was conducted, nested within cohorts of Belarussian, Russian and Baltic liquidators of the Chernobyl accident. The liquidators were mainly exposed to external radiation, although substantial dose to the thyroid from iodine isotopes may have been received by those who worked in May – June 1986 and who resided in the most contaminated territories of Belarus. Individual doses to the thyroid from external radiation and from iodine-131 were estimated for each subject.

Findings: 107 case patients and 423 matched controls were included in the study. Median age at the moment of first exposure was 37 years. Most subjects received low

doses (median 69 mGy). The doses were much higher for women (median 196 mGy) than for men (median 64 mGy).

A significantly elevated Odds Ratio was seen at doses of 300 mGy and above. The overall Excess Relative Risk (ERR) per 100 mGy was 0.38 (95% confidence interval (CI): 0.10, 1.09). Risk estimates were similar for iodine-131 and external exposure – ERR per 100 mGy was 0.45 (95% CI: 0.10, 1.61) and 0.38 (95% CI: -0.11, 2.07), respectively.

Conclusions: Although higher than risk estimated from a-bomb survivors exposed as adults, the significantly elevated risk observed in the present study is similar to that obtained in the recent studies of thyroid cancer following exposure to iodine-131 in childhood in the areas contaminated after the Chernobyl accident. The increased risk appears to be related to iodine-131 exposure, as well as to doses from external radiation.

Lens opacities among physicians occupationally exposed to radiation

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Abstract
We estimated the prevalence of lens opacities in physicians occupationally exposed to radiation.

Based on a nationwide registry of 1312 physicians, mostly radiologists with occupational exposure to ionising radiation, 120 subjects were invited to participate and 59 (49%) consented. The inclusion criteria included age 45-70 years, cumulative recorded radiation dose >10 mSv, and duration of work with dose monitoring >15 years. The participants filled in a questionnaire regarding occupational history and other risk factors for lens opacities. A full ophthalmological examination was performed. Lenticular changes were graded using the Lens Opacities Classification System, version II (LOCS II).

Any lens opacities were detected in 42% (95% confidence interval [CI] 29-56) of the 57 physicians without prior cataract surgery. Nuclear opacities were found in 11% (95% CI 4-22), cortical 8% (95% CI 2-19) and posterior subcapsular in 5% (95% CI 1-15) of the subjects. The prevalence of lens opacities increased with age, smoking and cumulative recorded radiation dose. After controlling for age, sex and smoking, cumulative radiation dose was positively associated with any lens opacities (p=0.06). Intervention radiologists and cardiologists had higher rates of cortical and posterior lens opacities than the other physicians.

Our preliminary results show posterior subcapsular lens opacities in physicians exposed to occupational radiation, consistent with recent studies on low dose radiation exposure. A full study with an unexposed reference group for risk estimation is warranted.

Introduction
The prevalence of lens opacities varies by age and diagnostic criteria used. The aetiology of cortical cataract is related to ultraviolet radiation and diabetes, whereas established risk factors for nuclear cataract include nutritional factors and smoking (Taylor et al. 1988, West et al. 1989, Klein et al. 1999). Other factors associated with

cataracts include diabetes, steroid use, ocular injury, myopia and cardiovascular disease (Taylor et al. 1988, Leske et al. 1991, Delcourt et al. 2000).

Radiation cataract is a well established effect of high doses shown in experimental studies and also in humans among survivors of the Hiroshima and Nagasaki atomic bombs (Otake v1992, 1996, Minamoto et al. 2004, Nakashima et al. 2006). Radiation induces changes in the lens epithelium, typically in the posterior subcapsular part of the lens. Radiation-related cataract has been regarded as a deterministic effect, which occurs only after high doses, with a threshold of approximately 1 Gy.

However, this view has been challenged by recent studies, which have shown increased risk of lens opacities even after low dose exposure (Ainsbury et al. 2009). Re-analyses of atomic bomb survivor data have demonstrated that the findings are compatible both with threshold (about 1.5 Sv) and linear models (Otake et al. 1992, 1996, Minamoto et al. 2004, Nakashima v2006). Besides posterior subcapsular opacities, dose-dependence in frequency of cortical cataracts has been shown among A-bomb survivors. Posterior subcapsular opacities were also more frequent among Ukrainian children affected by the fallout from the Chernobyl nuclear power plant accident (with a prevalence of 1%) than in controls (Day et al. 1995). More than 50% of Icelandic airline pilots exposed to cosmic radiation had some lens opacities, and 7-8% of them had posterior subcapsular cataracts (Rafnsson et al. 2005).

We conducted ophthalmological examinations among radiologists and other physicians exposed to occupational radiation in order to determine the frequency of lens opacities, especially posterior subcapsular ones, and evaluated the relation of lens changes to occupational factors.

Material and methods

The population at risk was identified from the national occupational radiation exposure registry maintained by STUK - Radiation and Nuclear Safety Authority, a governmental institution responsible for radiation protection in Finland. At the time of the study, the registry covered a total of 1312 physicians monitored for radiation exposure.

Eligible subjects were physicians included in the occupational exposure registry aged 45-70 years, history of dose monitoring for 15 years or more, and a recorded cumulative effective dose exceeding 10 mSv. The information for assessment of eligibility was obtained from the dose registry. No separate dose estimates for the lens were available. For logistic reasons, the subjects had to be current residents of the Uusimaa region. All 120 subjects who fulfilled these criteria were invited to participate.

The year of start and duration of dose monitoring, as well as the recorded cumulative dose were obtained from the registry. The recorded doses are based on film dosimeters worn outside the lead apron at work and therefore they overestimate the effective dose by at least a factor of 10. The relation of the recorded dose to the exposure to the eye is complex and it is unclear if measurements above or below the lead apron are better indicators of ocular doses. Furthermore, the registry covers only doses exceeding the recording threshold, which was decreased over time from 3.0 mSv to 0.1 mSv in a three-month monitoring period (3.0 mSv in 1964 until 1974, 1.5 mSv in 1975-1979, 0.5 mSv in 1980-1988, 0.3 mSv in 1989-1997 and finally 0.1 mSv from 1998 onward). Recorded doses were not available before 1969 and earlier doses were missing. The frequency of missing dosimeter readings was very low (<0.1%).

The study was approved by the ethical committee of the Helsinki University Central Hospital (tracking no. 352/E9/06) and followed the tenets of the Helsinki Declaration.

All the 120 eligible subjects were approached by mail and invited to participate in the study. The radiological society was contacted prior to the study and it informed its members in advance. The letter explained the purpose and procedures of the study and included also the study questionnaire for collection of information on medical and work history, use of radiation shielding, and smoking. The 59 subjects who gave their consent were asked to fill in the questionnaire. Based on questionnaire data and ophthalmological examinations, two subjects with prior cataract surgery were excluded from the study.

The study subjects underwent a comprehensive eye examination at Department of Ophthalmology, Helsinki University Hospital after pupillary dilatation. The Lens Opacities Classification System, version II (LOCS II), was used to grade lens changes (Chylack et al. 1989). LOCS II utilises a set of standard slit-lamp and retroillumination color transparencies for grading different degrees of nuclear, cortical, and subcapsular cataract. The system has four grades for nuclear opalescence and colour, five grades for cortical, and four subcapsular opacities. If the two eyes differed in grade, the worse grade was recorded.

The 95% confidence intervals for prevalences were calculated using the binomial distribution. Frequencies of categorical variables were compared using the chi square test and for small frequencies the Fisher's exact test. Differences in mean age, cumulative dose and duration of work career between subjects with and without lens opacities were evaluated with the non-parametric Kolmogorov-Smirnov test (due to the skewed distribution). Logistic regression analysis was conduced with presence versus absence of lens opacities as the binary response variable. The main results are based on any lens opacities (LOCS grade I-II nuclear, cortical and posterior subcapsular opacities), but a separate analysis was also conducted for posterior and cortical changes combined (excluding nuclear opacities).

Results

The mean age at examination of the 57 physicians in the final analysis (28 men and 29 women) was 58 years (median 60, range 46-70). Of them, 42 were radiologists (including 9 interventional radiologists), 14 cardiologists and a surgeon. The mean duration of radiation work was 24 years (median 25, range 4-45). Eleven subjects (19%) reported having used protective eyewear (lead glass spectacles), but only six (11%) had used them regularly.

Thirty-three (58%) of the 57 examined physicians were free of any lens changes, while a lens opacity (LOCS II grade 1-2) was found in 24 (42%, 95% CI 29-56%; Table 1). Nuclear opacities were most common (six subjects; prevalence 11%, 95% CI 4-22%), followed by cortical changes (4 cases with LOCS grade 1-2, prevalence 8% 95% CI 2-19; with 7 additional subjects showing cortical traces 19%, 95% CI 10-32%). The physicians with any lens opacities were older (mean age 64 versus 54 years among subjects without any lens changes, $p<0.001$) and more commonly smokers than those without such eye changes. They also had had a longer career (30 versus 20 years, $p>0.001$) and a higher cumulative radiation dose (84 vs. 42 mSv, $p=0.02$).

Posterior subcapsular opacities were found in two subjects (4%, 95% CI % 1-15). One had bilateral changes and the other a grade II opacity. The two physicians with posterior subcapsular opacities were aged 68 and 70 years, and had radiation work histories of 31 and 36 years with cumulative doses of 11 mSv and 22 mSv. One was a radiologist and the other an intervention radiologist. In addition, an intervention radiologist with a previously operated cataract in one eye had a posterior subcapsular opacity in the other, non-operated eye.

All lens opacities combined, as well as posterior and cortical changes combined were associated with increasing age. Smoking was significantly associated with all lens opacities combined, but not with cortical and posterior lens changes. Starting radiation work prior to 1975 was associated with all types of lens changes, as well as cortical and posterior changes combined. Having been monitored for radiation exposure for at least 20 years was associated with all lens opacities and working with radiation on a daily basis with cortical and posterior changes combined.

Age and smoking status were associated with significantly increased prevalence of lens opacities in the logistic regression analysis even after mutual adjustment. Duration of radiation work was associated with the presence of any lens opacities in the bivariate analysis, but the association was no longer significant after adjusting for age.

When nuclear changes were excluded, the cumulative radiation dose was not associated with cortical and posterior opacities. Intervention radiologists and cardiologists had higher risk of cortical and posterior lens changes compared with other specialists (mainly other radiologists).

Discussion

Lens opacities were common among Finnish physicians exposed to ionising radiation, but the majority of them were nuclear opacities, which generally are not thought to be related to ionising radiation. Posterior subcapsular opacities, which classically are associated with ionising radiation, were found in 4% of the participants. Cortical opacities were found in 7% subjects, but another 12% had cortical traces, and if these findings are included the proportion of subjects with cortical opacities is as high as 19%, exceeding the number of nuclear findings. None of the subjects had a cataract that would require surgery. However, two had already undergone cataract surgery.

There was some indication for an association of lens opacities with the cumulative radiation dose, though the relation was no longer significant after adjustment for age. Further, this association was not observed when the analysis was restricted to cortical and posterior opacities only (the types of changes associated with radiation in earlier studies). The main limitations of our study are the small size and lack of an unexposed reference group.

Overall, the exposure levels were well below those traditionally thought as the threshold for cataract induction in the entire study group (mean 60 mSv, maximum 300 mSv), as well as those with the posterior subcapsular and cortical opacities (mean 65 mSv, maximum 72 mSv). However, recent studies have suggested increased cataract rates among subjects with occupational, environmental and medical radiation exposures with similar dose levels.

Radiation exposure varies substantially between radiologists. Those who perform interventional procedures such as catheterisations may receive doses approaching the annual dose limits (currently, 20 mSv), whereas others are subjected to almost no radiation during their entire career. We selected the study subjects among those with the highest recorded doses, long work history and tasks with potential for radiation exposure. Yet, detailed information on frequency of performing various procedures was not collected. Nevertheless, those working with fluoroscopic intervention procedures (intervention radiologists and cardiologists) had a higher frequency of cortical and posterior opacities combined than the other physicians.

We had access to results of dose monitoring based on personal dosimeters. However, their readings are not a direct indicator doses for the eye. The recorded doses are likely to substantially overestimate ocular doses, because the dosimeters were worn at the chest level outside the lead apron. The radiation dose is primarily due to scatter from the patients and is highly non-uniform (Martin 2009). The dose to the eyes depends heavily on the amount of radiation delivered to the patient, proximity of the physician to the x-ray tube and use of protective devices such as lead spectacles or ceiling-mounted shields.

Infrequent use of eye protection may reflect a prevailing perception that risk for cataract is minimal or irrelevant. In our study, only one in five exposed physicians used any eye protection at all and only one in ten used them regularly.

Few previous studies have evaluated occurrence of lens opacities among medical personnel. Case reports indicate lens injuries following exposures substantially exceeding the dose limits (Vano et al. 1998). A recent study, without dosimetric information, showed a significantly increased frequency of cataracts among 35,705 U.S. radiologic technologists (Chodick et al. 2008). Similarly, an Italian showed increased prevalence of lens changes among radiologists and radiological technicians compared with unexposed medical workers (Milacic 2009). Further, posterior subcapsular cataracts in 8% and smaller dot-like lens opacities in 37% of the 59 participants of the annual meeting of the Society of Interventional Radiology (RSNA 2004).

Conclusions

Radiologists with any signs of cataract had a longer history of radiation work and a larger cumulative radiation dose than those not affected. Yet, risk of cataracts is strongly age-dependent and also cumulative dose and career length increase with age. After adjustment for age, sex and smoking the effect of dose was of borderline significance and duration of radiation work was no longer significant. Yet, a higher frequency of cortical and posterior lens changes was associated with interventional procedures even after adjustment for other factors. The study indicates the need for a more comprehensive study on radiologists to assess the effect of radiation on the lens of the eye.

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Occupational cataracts and lens opacities in interventional cardiology: The O'Cloc study

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Abstract

Interventional cardiologists are repeatedly and acutely exposed to scattered ionizing radiation (X-rays) during their diagnostic or therapeutic procedures. These exposures may cause damages to the eye lens and induce early cataracts known as radiation-induced cataracts. The O'CLOC study is a French epidemiological study designed to test the hypothesis of an increased risk of cataracts among interventional cardiologists as compared with unexposed cardiologists. This multicenter cross-sectional study will include a total of 300 cardiologists aged > 40 years: a group of interventional cardiologists (approx. 2/3 of coronary interventionists and 1/3 of electrophysiologists) and a group of unexposed cardiologists (clinicians or echocardiographists), matched for age, sex and place of work. Individual information, including risk factors of cataracts (age, diabetes, myopia, etc. ...), will be collected during a telephone interview. For the exposed group, a specific section of the questionnaire is focusing on their occupational history, the procedures description (type, frequency, use of radiation protection tools) and will be used to classify "comparable exposure level" groups according to their estimated cumulative dose. For all participants, clinical eye examinations will be performed to specifically detect cataracts even at the early stages (lens opacities, LOCS according to the international standard classification). The overall analysis will provide an estimation of the risk of cataract in interventional cardiology comparatively to not-exposed reference group, taking into account other risk factors. A complementary comparative analysis of risks according to the level of exposure is also planned. This epidemiological study will provide further knowledge on the potential risk of occupational radiation-induced cataracts in interventional cardiology and will contribute to the awareness of cardiologists in radiation protection.

Cardiovascular disease and radiation – review and meta-analysis of epidemiological evidence at low doses

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Abstract

Although the link between high doses of ionizing radiation and damage to the heart and coronary arteries has been well established for some time, the association between lower dose exposures and late occurring cardiovascular disease has only recently begun to emerge, and is still controversial. In this paper, we extend an earlier systematic review by Little and colleagues on the epidemiological evidence for associations between low and moderate doses of ionizing radiation exposure and late occurring blood circulatory system disease. Excess relative risks per unit dose in epidemiological studies vary over at least two orders of magnitude, possibly a result of confounding and effect modification by well known (but unobserved) risk factors, and there is statistically significant ($p < 0.00001$) heterogeneity between the risks. This heterogeneity is reduced, but remains significant, if adjustments are made for the effects of fractionated delivery or if there is stratification by endpoint (cardiovascular disease vs stroke, morbidity vs mortality). One possible biological mechanism is damage to endothelial cells and subsequent induction of an inflammatory response, although it seems unlikely that this would extend to low dose and low dose-rate exposure. A recent paper of Little et al. proposed an arguably more plausible mechanism for fractionated low-dose effects, based on monocyte cell killing in the intima. Although the predictions of the model are consistent with the epidemiological data, the experimental predictions made have yet to be tested. Further epidemiological and biological evidence will allow a firmer conclusion to be drawn.

Co-exposure to radiation and methyl mercury during critical phase of neonatal brain development in mice enhances developmental neurobehavioral defects

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Abstract

Organisms, including man, are continuously exposed to low doses of ionizing radiation as well as persistent and nonpersistent chemicals in the environment. Hence, in the process of developing numerical limits for environmental protection, there is a strong need to consider interactive effects between radiation and other environmental stressors. It is known that ionizing radiation, as well as methyl mercury, can give rise to neurotoxicological and neurobehavioural effects in mammals and that developmental neurotoxic effects can be seen after exposure during gestation. However, there is a lack of knowledge concerning effects and consequences from low-dose exposure during critical phases of perinatal and/or neonatal brain development and the combination of ionizing radiation and environmental chemicals. Epidemiological studies of patients with haemangioma have indicated that radiation exposures to the brain during infancy might deteriorate cognitive ability in adulthood. Ten-day old neonatal NMRI male mice were exposed to a single oral dose of MeHg (0.40 or 4.0 mg/kg bw). Four hours after the MeHg exposure the mice were irradiated with 60Co gamma radiation at doses of 0,2 and 0,5 Gy. The animals were subjected to a spontaneous behaviour test at the ages of 2- and 4-months, and the water maze test at the age of 5 months. Neither the single dose of MeHg (0.4 mg/kg bw) nor the radiation dose of 0.2 Gy affected the spontaneous behavior, but the co-exposure to radiation and MeHg caused developmental neurotoxic effects. These effects were manifested as disrupted spontaneous behavior, lack of habituation, and impaired learning and memory functions. Studies are continuing to verify the effects ant to elucidate possible underlying mechanisms.

DNA-Triplex-forming-oligonucleotides as a tool to target specific DNA sequences

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Abstract

Purpose: Triplex-forming-oligonucleotides (TFOs) are able to bind complementary DNA sequences in a sequence specific manner and are therefore a promising tool to manipulate genes or gene regulatory units in a cellular environment. TFOs might have also therapeutic potential e.g. as a carrier for Auger-Electron-Emitter (AEE) to target DNA of tumour cells. A main obstacle is the access of the TFOs to their targets in the cell nucleus. Thus we studied the intracellular biokinetics of TFOs with the focus in the intracellular transfer from the cytoplasm into the nucleus.

Methods: TFOs specific for the genes *cdkn2a*, *bcl2*, *brca1*, *chk2*, *cdk4* were designed using TFO Target Sequence Search (Univ. of Texas). DNA-Triplex-formation was confirmed by electrophoretic-mobility-shift assay (EMSA) in vitro. For biokinetic studies SCL-II cells were transfected by electroporation with Alexa488-labelled TFOs. Transfected cells were subsequently cultured for 1 h, 6 h, 12 h, 18 h, 24 h, 30 h, 48 h and 72 h and TFO signal intensity was determined in single cells and in isolated cell nuclei by flow cytometry (FACS-Canto II, BD) at each time point.

Results: Sequence design of TFOs by TFO Target Sequence Search (Univ. of Texas) for the desired genes is generally not suitable to predict DNA-Triplex-formation in vitro as could be demonstrated by EMSA. The desired Triplex-DNA-formation could be confirmed for 57 % of all tested TFOs by EMSA. The biogenetic studies showed that TFO-Alexa488 positive cells were detectable as soon as 1 h after transfection and the signal intensity remained constant for at least 30 h. 72 h after transfection the signal was less intense but still detectable. A substantial loss of Alexa488-labelled TFO positive cell nuclei was observed within the first 6 h post-transfection followed by a significant increase up to 18 h post-transfection.

Conclusions: Stable Triplex-DNA-formation in vitro can not be predicted by the sequence of TFOs only. TFOs initially located in the cytoplasm are re-located to the cell nucleus within 12 h after delivery of the TFOs probably during cell division.

Introduction

Triplex-forming-oligonucleotides (TFO) are short oligodeoxyribonucleotides, able to bind complementary DNA sequences in a sequence specific manner. They are major groove binding ligands which are associated to the DNA duplex via hydrogen bonds known as Hoogsteen-Bonds. Triplexes can only be formed with homopurine-homopyrimidine regions of the DNA (Kamenetskii 1995). The orientation to the target duplex can be parallel and antiparellel depending on the base composition of the TFO. While the bases thymine, cytosine and guanine are able to bind in both orientations, triplexes with adenine as the third base can only form a triplex in antiparallel conformation. TFOs containing cytosine form triplexes only under acidic conditions as the cytosine has to be protonated (Praseuth et al. 1988).

Regions containing homopurine-homopyrimidine sequences are quite common in the human genome as could be demonstrated by Goni et al. 2003. The presence of a third strand in the major groove can influence the behaviour of the DNA, changing its ability to recognize specific proteins and consequently altering mechanisms controlling DNA function. Despite the direct interaction of TFO and DNA it can also serve as a strong tool, for example as a carrier system in biotechnological and biomedical applications (Vekhoff et al. 2008).

During the decay of 125I a shower of low-energy Auger-electrons is emitted leading to the deposition of very high energies in a rather small volume (Pomplun et al. 1987). Panyutin et al. 1994 showed that 125I-labelled TFOs binding to DNA lead to a significant increase of double strand breaks. Though the energy deposition of the Auger electrons is limited to a 10 to 100 nm sphere at the decay site, the damage to the cellular environment will be only minor, even at high activity concentrations of 125I (Sedelnikova et al. 1998, Kriehuber et al. 2004). On the contrary, the decay of 125I close to its target will induce significant damage which is difficult to repair (Sedelnikova et al. 1998). Referring to these characteristics the TFO can serve as a tool to silence genes very specifically without necessarily killing the cell itself. This application could be useful for Antigene Radiotherapy (Panyutin 2003), which is the generic term for damaging of selected genes by a high dose of radiation from radionuclides delivered to this gene by specific DNA-binding molecules.

One of the main problems, despite the binding of the TFO to its target sequence, is the delivery of the TFO to the point of action, usually the cell nucleus. Alexa488-labelled functional TFOs were used to examine the dispersion and distribution of TFOs in cells and isolated cell nuclei after transfection of tumour cells by electroporation employing flow cytometry at different time points post-transfection. Viability and apoptosis induction was analyzed in parallel.

Material and methods

Oligonucleotides and TFO Labeling

TFOs and biotinylated oligonucleotides were designed employing TFO Target Sequence Search (Univ. of Texas) and synthesized by Metabion (Martinsried) in standard quality. Alexa488-labelled TFOs were purchased by TIB Molbiol (Berlin). Labeling of TFOs with ^{125}I was performed employing the Primer Extension Method. Therefore the preTFOs (TFOs used for the Primer Extension reaction) and template

(200 pmol each) were annealed in 1 x Klenow buffer (500 mM Tris-HCl, pH 8.0, 50 mM $MgCl_2$, 10 mM DTT) for 10 min incubation at 90 °C followed by slow cooling (1 °C/min) to room temperature. Primer extension reaction was carried out in the presence of 10 µCi ^{125}IdCTP (Perkin Elmer, Rodgau) by 2.5 units of exonuclease-free Klenow fragment (Fermentas, Leon-Rot) in a total volume of 10 µl. After 15 min at 37 °C the reaction was stopped by heating to 70 °C for 10 min. The duplex-DNA product was purified by ethanol precipitation and resuspended in TEN_{100}-Binding Buffer (10 mM Tris-HCl, pH 7.5, 1 mM EDTA, 100 mM NaCl). The labelled duplex-DNA was bound to Streptavidin Magnetic Particles (Roche, Mannheim) by incubation for 10 min at room temperature followed by two washing steps with TEN_{1000} Wash Buffer (10 mM Tris-HCl, pH 7.5, 1 mM EDTA, 1 M NaCl). The magnetic particles were resuspended in TES Buffer (50 mM Tris-HCl, pH 7.5, 1 mM EDTA, 150 mM NaCl). The duplex-DNA was denatured by addition of NaOH to a final concentration of 200 mM for 5 min. The magnetic particles were then removed with a magnet and the supernatant containing the ^{125}I-TFOs was collected, ethanol precipitated and resuspended in nuclease free water (Qiagen, Hilden). At this step the amount of incorporated ^{125}I-dC was calculated by activity measurement on a PerkinElmer 1480 Automatic Gamma-Counter. Labelled TFOs were stored at -70 °C.

Amplification of Target DNA-Fragments from DNA isolates of SCL-II cells

PCR Primer sets were designed for the amplification of DNA fragments containing the specific target-sequence for the labelled TFOs. The primers were synthesized by Metabion in standard quality. The sequence lengths varied from 200 bp to 2000 bp and the amplification was carried out on a TProfessional Gradient Cycler (Biometra, Goettingen). After the amplification the DNA fragments were ethanol precipitated, resuspended in water and stored at -20 °C.

In Vitro Triplex Formation and Binding Assay

^{125}I-labelled TFO (~ 0.6 µCi) were mixed with the amplificated DNA fragment (~ 10 µg) containing the specific TFO target sequence and TFO-binding-buffer (10 mM Tris-HCl, pH 5.8 and pH 7.5, 10 mM $MgCl_2$, 1 mM Spermidine) in a total volume of 24 µl. The mixture was incubated at 37 °C for 24 h. After the incubation the triplex was frozen for at least 30 days (0.5 half-life of ^{125}I) at -70 °C for decay accumulation. For *in vitro* verification of triplex formation an electrophoretic mobility shift assay (EMSA) was performed. Aliquots of the samples were loaded on a 12% native polyacrylamid gel containing 10 mM $MgCl_2$ and ran at 150 V for 20 h in TBE buffer (Tris-Base 89 mM, pH 8.0, boric acid 89 mM, EDTA 2 mM). Afterwards the gel was silver stained for visualization. Triplex formation was detected by band shift in silver stained gels and additionally verified by exposition to Fujifilm MS Imaging Plates BAS-MS3543 and visualization with a FLA-5000 Imaging System (Fujifilm, Düsseldorf).

Cell Line and Culture Conditions

SCL-II (squamous carcinoma cell line II) cells were grown in minimum essential medium Eagle (MEM, PAA Laboratories GmbH, Cölbe) with L-Glutamine, supplemented with 16 % fetal bovine serum (FBS , PAA Laboratories GmbH, Cölbe) in a water-saturated atmosphere of 5 % CO_2 at 37° C as previously described (Kriehuber et al. 2004) and cultivated at standard conditions.

Transfection experiments

Cells were grown in cell culture flasks (PAA Laboratories GmbH, Cölbe) to 70-80 % confluency, trypsinized and resuspended in MEM. The required number of cells ($2x10^6$) was centrifuged for 10 min at 90 x g and resuspended in 100 µl Nucleofector Solution (Lonza AG, Cologne). 5 µl of Alexa488-labelled TFO [20 pmol/µl] (TIB Molbiol, Berlin) was added and the electroporation reaction was performed on the Amaxa Nucleofector Device I (Lonza AG, Cologne) employing transfection program L-013. Immediately after the reaction 500 µl of MEM was added and the cell solution was transferred into a 6-well plate at a cell concentration of $3x10^5$ cells/well and, after adding 2 ml MEM, incubated at 37 °C.

Flow cytometric Analysis of Transfected Cells

SCL-II cells transfected with Alexa488-labelled TFO were incubated at 37 °C. After 1 h, 6 h, 12 h, 18 h, 24 h, 30 h, 48 h and 72 h at each time samples were trypsinized and washed two times in PBS buffer. For analysis of cell nuclei the cells were further processed following a nucleic isolation protocol. Cells were incubated with Cell-Extraction buffer (Sucrose 320 mM, Hepes 10 mM, $MgCl_2$ 5 mM, Triton X-100 1%, pH 7.4) for 10 min on ice followed by two wash steps with Nucleic-Wash buffer (Sucrose 320 mM, Hepes 10 mM, $MgCl_2$ 5 mM, pH 7.4). Finally, cell nuclei were resuspended in Nucleic-Wash buffer. According to the samples, controls transfected with non-labelled TFOs and without any TFO were performed according to the above protocol. Transfected cells and isolated cell nuclei were analyzed in a flow cytometer (BD, Canto II) for presence of Alexa488 signal. Apoptosis induction was quantified employing non-labelled TFOs using the Annexin-V-FITC/PI-assay (BD, Heidelberg) according to the manufacturer's recommendation.

Results

Triplex formation *in vitro*

A variety of oligonucleotides (Table 1.) were analyzed for their ability of triplex formation with their specific DNA-target *in vitro*, employing electrophoretic mobility shift assay (EMSA). About 43 % of the tested oligonucleotides (not all displayed), all having a suitable sequence to form a Triplex-DNA with their specific DNA-targets, failed to form a DNA-triplex in the EMSA *in vitro* (Fig.1, Lanes 2, 17 and Table 1.). Triplex formation *in vitro* can therefore not be predicted on TFO and DNA-target sequence only. Target sequences for the functional TFOs were located in the genes *cdkn2a*, *bcl2*, *brca1*, *chk2* and *cdk4* (Table 1.). Non-complementary TFOs (Non-sense TFOs) failed in all cases, as expected, to form DNA-triplexes (Fig.1, lanes 3, 6, 9, 12, 15, 18, 22, 25).

Triplex formation of ^{125}I-labelled TFO with its specific target sequence could be demonstrated and verified using an autoradiographic assay (Fig.2 a; red arrow). The lower band (green arrow) represents an enrichment of fragments, resulting from the special gel running conditions and characteristics of a low pH electrophoresis (pH 5.8)

(Osafune 2004). After verification of triplex formation, samples were frozen at -70°C for decay accumulation.

Biokinetic of TFOs in human cells *in vitro*

The flow cytometric analysis showed that immediately after transfection Alexa488-labelled TFOs were distributed throughout the cells as TFOs could be detected 1 h post-transfection in whole cell samples and in isolated nuclei (Fig.3). A transfection efficiency of ~ 80 % in whole cells and ~ 60 % in isolated nuclei was observed. Within the first 6 h post-transfection a substantial loss of TFO positive cells and cell nuclei (~ 65 % and ~ 15 % respectively) occurred, which corresponded to an increase of apoptotic cells (Fig.3, black line), followed by a significant increase of TFO-positive cells and nuclei up to 18 h post-transfection (~ 85 % and ~ 25 % respectively) where it remained constant for further 6 h. Between 30 h and 72 h after transfection a smooth decline of TFO positive cells and cell nuclei was detected. The percentage of apoptotic cells reached its maximum 12 h post-transfection and constantly declined to control level at 72 h post-transfection.

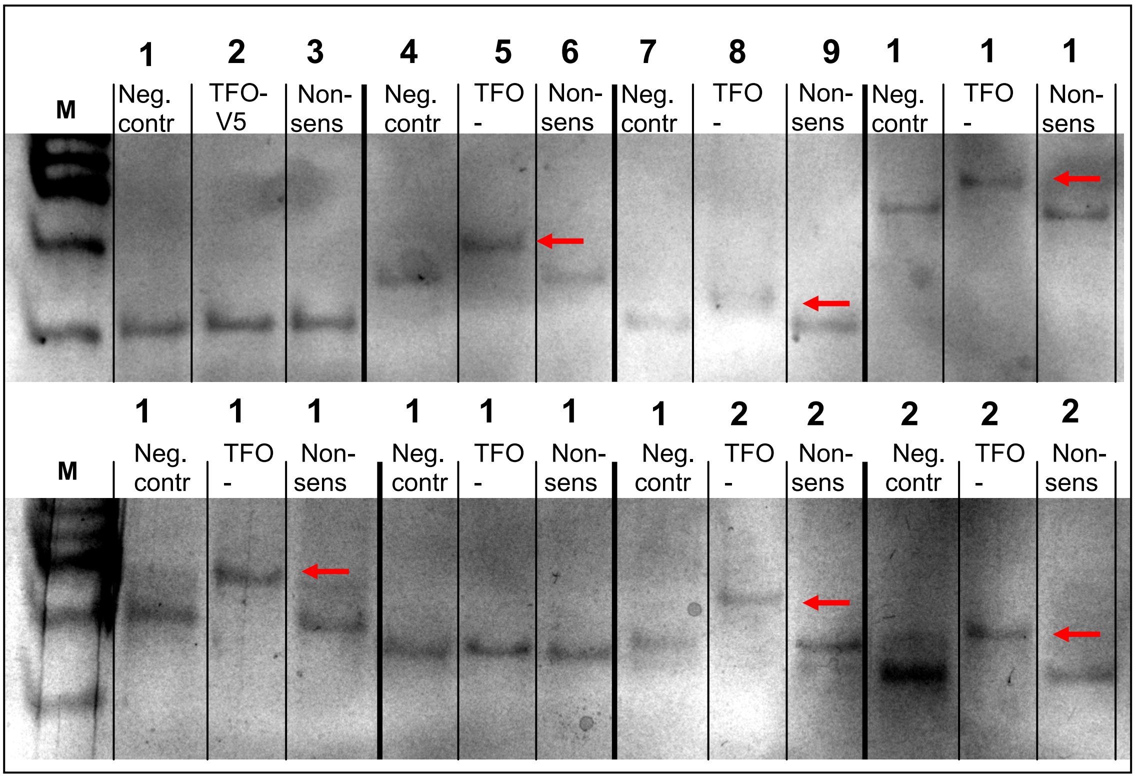

Fig. 1. Triplex formation *in vitro* visualized with electrophoretic mobility shift assay. A band shift shows successful triplex formation (red arrow; lane 5, 8, 11, 14, 20 and 24); TFO-V5 and TFO-V11 failed to form a DNA-triplex (lane 2 and lane 17). Native polyacrylamid gel; silver staining; Neg. contr.: Controls containing the dsDNA target fragment only (lane 1, 4, 7, 10, 13, 16, 19 and 23). TFO: Triplex forming oligonucleotides with their specific target sequence (lane 2, 5, 8, 11, 14, 17, 20 and 24). Non-sense.; Non-sense samples containing the target sequence with a non-specific (non-sense) TFO; TFO-V4 (lane 3, 6, 9, 12, 15, 18, 22 and 25).

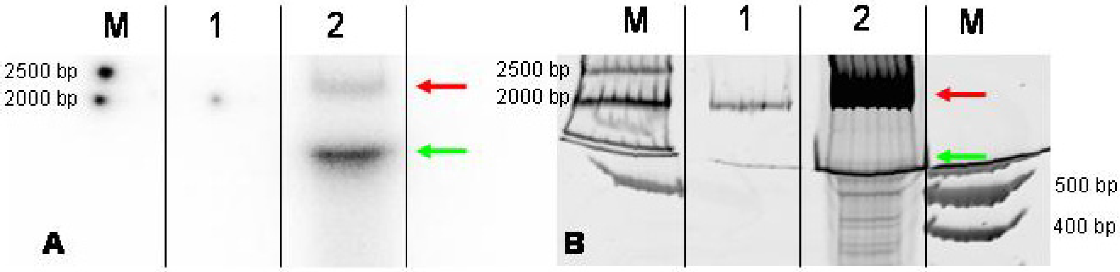

Fig. 2. Triplex formation of ^{125}I-labelled TFO-V13 with its specific target on a 1910 bp dsDNA fragment *in-vitro*. Electrophoresis in 10 % native Polyacrylamide gel (pH 5.8). (A) Autoradiographic scan. M; marker. Visualization of labelled TFO bound to its target (red arrow). Concentration of shorter fragments (green arrow) in lane 2. (B) Same gel stained with ethidium bromide. M: marker. Lane 1: dsDNA target fragment. Lane 2: TFO with high concentrated target fragments (red arrow). Concentration of shorter fragments (green arrow).

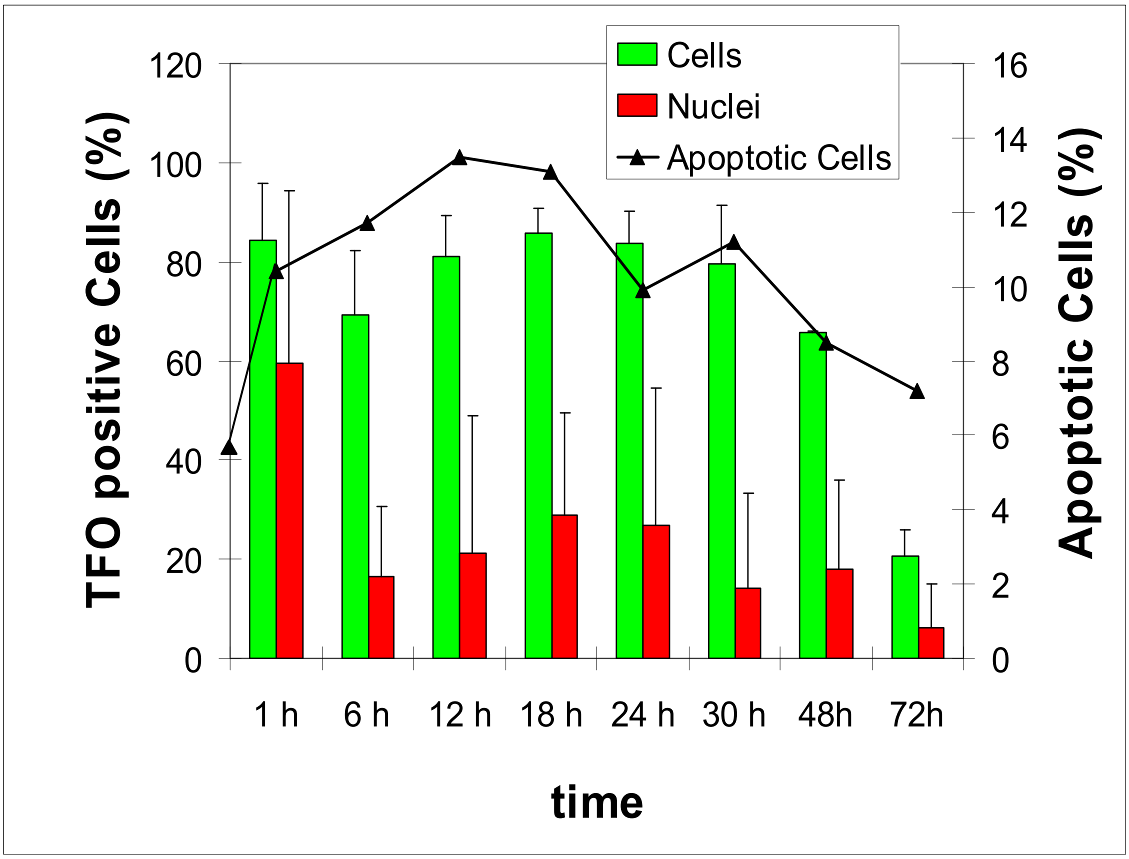

Fig. 3. Flow cytometric analysis of SCL-II cells after transfection with Alexa488-labelled TFO-V2 (N=4). Cells were transfected by electroporation with 100 pmol of Alexa488-labelled TFO-V2 and incubated at 37 °C. Cell and nucleic samples were prepared for flow cytometric analysis after different time points (1 h – 72 h). Percent of TFO positive cells (green bars, left axis) and cell nuclei (red bars, left axis) plotted against time after transfection. Cell samples transfected with non-labelled TFO-V2 were additionally analyzed for apoptotic cell death (black line); percent of apoptotic cells (right axis) plotted against time after transfection.

Table 1. Nucleotide sequence of binding TFOs and corresponding target sequence. Marked in red and bold-typed the nucleotide which were used for ^{125}I-labelling. Asterisks (*) indicates no triplex formation could be detected for these TFOs.

Name	TFO-Sequence (5´- 3´)	Target-Sequence (5´- 3´)	Gene (locus)
TFO-V2	**ttt gtt ggg tgg tgg gtt ggt tgt gtt**	**aag aga agg aag gga gga ggg aag aaa**	*cdkn2a* (21969523)
TFO-V5*	**gtt gtg tgt gtg tgt gtg**	**ctt ctc tct ctc tct ctc**	*survivin* (76219872)
TFO-V6	**ttg tgg gtg tgg tgg gct tt**	**aaa ggg gag gag agg gag aa**	*cdk4* (58148139)
TFO-V7	**gtt gtg gtt gtg gtt ggg**	**ggg aag gag aag gag aag**	*cdk4* (58148122)
TFO-V8	**ggt gtt ggt tgg ttg ttt tgg tgg tgg gc**	**ggg gag gag gaa aag aag gaa gga aga gg**	*bcl2* (60874077)
TFO-V9	**gtg ttt gtt ttt gtt ggg tgg tgt ggg gc**	**ggg gga gag gag gga aga aaa aga aag ag**	*bcl2* (60796002)
TFO-V10	**tgg gtg tgt gtt ggt gtg ttg ttc c**	**gga aga aga gag gaa gag aga ggg a**	*brca1* (41203138)
TFO-V11*	**tgg gtg gtg ggt gtt tgt tgt g**	**g aga aga aag agg gat gag gga**	*brca2* (32932129)
TFO-V12	**ggg tgg tgt ggg tgg gtt ttt ttt**	**aaa aaa aag gga ggg aga gga ggg**	*chk2* (29133020)
TFO-V13	**ggg ttt ttt gtg ggg gtg ttc**	**gaa gag ggg gag aaa aaa ggg**	*chk2* (29099560)

Discussion

The TFO Target Sequence Search (Univ. of Texas) proposed functional TFO sequences, which did not form DNA triplexes *in vitro* with their specific target sequence (Fig.1 and Table 1). Although all TFOs were designed according to the following special criteria: - minimum length of 15 bp, - guanine content minimum 50 %, - no pyrimidine interrupts allowed, - default settings for TFO-target binding at near-physiological conditions (pH 7.2, 37°C, 10 mM $MgCl_2$), it could be clearly demonstrated that every single TFO has to be confirmed in EMSA.

To obtain sufficient Triplex formation with ^{125}I-labelled cytosines the pH of the binding buffer had to be adjusted to pH 5.8 to protonate the cytosine in order to get proper Hoogsteen hydrogen bonds. However, pre-testing confirmed triplex formation for non-labelled TFOs with a maximum of two cytosines under physiological conditions, which is consistent to the findings of Duca et al. 2008.

The biokinetic studies confirmed that the applied Amaxa transfection protocol caused an initial efficient transfer of TFOs into the cell nucleus (Fig.3.). The substantial loss of TFO-positive nuclei (- 40 %) within the first 6 hours is probably caused by the chosen transfection procedure, by which a direct initial transfer of TFO into the cell nucleus occurs (Hamm et al. 2002). After transfection, a pronounced increase of apoptotic cells was recorded. Though this effect is not described in the literature for TFO after electroporation, it might be assumed that the high TFO concentration led to the observed increased apoptotic cell death by a nucleotoxic effect, as described by Shen et al. 2003 in human lung carcinoma cells transfected with TFOs by a liposome-based transfection method.

The increase of TFO positive cell nuclei 6 – 24 h post-transfection reflects either an active transfer of TFOs into the cell nucleus by still unknown processes or a passive relocation of TFOs into the cell nucleus during mitosis. The observed transfer to the cell nucleus is in good agreement with the data published by Sedelnikova et al. 1998 and 1999, showing a rapid relocation of radionuclide-labelled TFOs into the nucleus within 3 h after delivery to the cytoplasm using DMRIE-C liposomes.

The observed significant decrease of TFO-positive cells 48 h post-transfection is probably due to dilution caused by ongoing cell division. Degradation processes cannot be generally excluded but are unlikely since Sedelnikova et al. 1999 could confirm triplex stability in HeLa cells for at least 48 h and Kadenbach et al. 2005 showed non-degradation of TFO by FRET in SCL-II cells for at least 24 h post-delivery.

Conclusions

TFOs, either non-labelled, fluorochrome-labelled or ^{125}I-labelled, are suitable to form stable DNA-triplexes with their specific target sequences *in vitro*. It can be concluded from our results that the prediction of DNA-triplex-formation shall not be based on the TFO sequence alone, but has to be approved by additional experimental methods, like the electrophoretic mobility shift assay. Within the first 18 h after transfection a fraction of TFOs was relocated from the cytoplasm into the nucleus, probably by the break-down of the nuclear envelope during cell division. TFOs can be detected in considerable amounts in the cytoplasm and in the cell nucleus at least 48 h post-transfection. Electroporation is a suitable method to deliver TFOs in cells and cell-lines, although the used transfection procedure induced enhanced apoptotic cell death in the transfected cells, especially in cells, where TFOs were initially directly transfected into the cell nucleus.

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Molecular, metabolic and genomic response of TPC-1 cells to external X-irradiation

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Abstract

The release of radioactive iodine isotopes into the atmosphere in the wake of the Chernobyl disaster and the subsequent increase in the cases of thyroid cancer amongst children provided us with solid evidence of the link between irradiation and thyroid cancer. The link between the radiation dose and the adverse effects on human health has been established and a linear correlation drawn. However, that correlation only holds for doses higher than a certain threshold, below which there is lack of unanimity in the scientific community. We attempted to subject TPC-1, a cell line of human papillary thyroid carcinoma origin to a range of X-ray doses from low to high (0.0625 to 4 Gy) and test their effect on cellular proliferation, apoptosis and molecular signaling to uncover the effect of low doses of radiation on them. We observed a decrease in proliferation concomitant with a decrease in cells in the S phase of the cell cycle even at the lowest dose. On the other hand, there was no significant increase in apoptotic or necrotic cells at the same time point. On the protein level, an increase in p53 and some of its targets (e.g. p21 and mdm-2) was observed whereas an absence of p73 was apparent. An up-regulation of anti-apoptotic proteins such as Bcl-2 and Akt was also noted. A microarray run was performed and revealed an up-regulation in the p53 pathway whereas one gene, RFLP-1, appeared to be consistently up-regulated at all doses tested. Further research into this gene are underway. In addition, analysis of the metabolic profile of control and irradiated cells will be done on the supernatants of these cells. Metabolomics is a relatively new high throughput technique and is sensitive enough to hopefully measure the effect of low doses of radiation. Finally, analysis of the changes in microRNAs in irradiated versus control cells is currently underway.

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Irradiation of mouse zygotes and genomic instability: influence of genes involved in DNA repair, cell cycle regulation and apoptosis

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Abstract
Studies under way in our laboratory aim at investigating how heterozygous mutations in important genes might influence the radiation sensitivity of early embryos. We mainly focus on the zygote (first day of gestation in mammals), which occurs while women cannot be aware of pregnancy. Moreover, x-irradiation of wild-type zygotes has been reported to result in the development of a genomic instability in two different mouse strains. The results presented here concern the development of such effect after x-irradiation of zygotes of other strains, carrying a mutation in either the p53 gene or the PARP gene. P53+/- embryos were obtained by mating CF1 p53+/+ females with CF1 p53-/- males, while PARP+/- embryos were obtained by mating PARP-/- females/males (C57BL genetic background) with C57BL males/females. Females showing a vaginal plug were x-irradiated 2 h after fertilization (0.2 or 0.4 Gy) and sacrificed on day 8 of gestation. Their gastrulas were collected and cultured for 7 h in rat serum supplemented with colchicine and their cells were fixed for chromosome analysis. The frequencies and types of chromosome anomalies did not differ between control p53+/- and p53+/+ cells or between control PARP+/- and PARP+/+ cells. Anomalies were essentially of the chromatid-type, with a majority of chromatid gaps. No increase of chromosome damage was found after irradiation of p53 +/- or PARP +/- zygotes and, again, chromatid-type anomalies largely predominated over chromosome-type anomalies. These results differ from those obtained by others in wild-type embryos irradiated with higher doses. Moreover, they suggest that the presence of one mutated allele for two important genes would not result into an increased risk of developing a chromosomal instability after irradiation with moderate doses at the zygote stage. Similar studies are under way with other important DNA responses genes (supported by a contract between the SCK•CEN and the Federal Agency for Nuclear Control).

Individual sensitivity in targeted and non-targeted effects of radiation – ATM as a model for characterizing individual susceptibility

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Abstract
The aim of this study is to investigate the role of genetic heterogeneity in the *ATM* gene with respect to the individual variation of response to ionizing radiation. The issues of targeted and non-targeted effects particularly at low doses will be addressed. ATM is a major activator of cellular response to DNA-double strand breaks. *ATM* germ line mutations cause ataxia-telangiectasia (AT), a rare resessive autosomal disorder. Female carriers are phenotypically normal but there is evidence of increased risk of breast cancer among these women. We hypothesize that cells from *ATM* mutations carriers are more susceptible to bystander signals than ATM-proficient cells. The experimental system is based on co-culture technique where irradiated cells communicate through media with unirradiated cells. The co-culture system will be established by using human lymphoblastoid cell line TK6 cells that are irradiated by X-rays with doses of 0.01, 0.1, 1, and 2 Gy and co-cultured with unirradiated bystander cells for 1 hour and 18 hours. The induction of direct and bystander effects of radiation will be analyzed using MTT [3-(4,5-dimethylthiazol-2-yl)-2,5-diphenyl tetrazolium bromide] cell viability assay, apoptosis and chromosomal aberration assays. In addition to results from TK6, corresponding data from experiments using cell lines with different *ATM* mutations will be evaluated against the wildtype *ATM* gene.

Influence of genetic polymorphisms on the yield of chromosomal aberrations among Estonian Chernobyl cleanup workers

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Abstract

The frequency of chromosomal aberrations (CAs) in peripheral blood lymphocytes has been applied as a biomarker for biodosimetry of radiation. Individual variability in CA frequency is known to be caused by several factors such as age and various kinds of exposures as well as genetic factors. The aim of the present study is to evaluate the effect of individual susceptibility to the amount of CAs in Estonian Chernobyl cleanup workers. The data- and biobank of a study cohort of 4832 men contains detailed information of, e.g., recorded doses, smoking habits, medical radiation exposure, and health history. From this cohort, 300 men with doses of 48–300 mSv were chosen to the present study. In addition, 100 unexposed Estonian men were used as a control group. CAs were analyzed in peripheral blood lymphocytes using the fluorescence in situ hybridization (FISH) technique. In order to investigate the generic susceptibility, we studied single nucleotide polymorphisms (SNPs) in DNA repair genes, i.e., ATM (Phe858Leu, Pro1054Arg, Asp1853Asn, intron 5, rs228599 and intron 61, rs664143), LIG4 (Thr9Ile, Asp568Asp), XRCC1 (Arg194Trp, Arg280His, and Arg399Gln), XRCC3 (Thr241Met), XRCC5 (intron, rs3835 and 5' near gene, rs11685387), and XRCC6 (Gly593Gly and 5' near gene, rs2267437) and in an apoptosis gene, i.e., CASP8 (Asp302His and 5' near gene, rs6747918). The individual yield of CAs (particularly translocations) will be statistically analysed with respect to the SNPs, recorded dose, and other confounders. The results will bring more knowledge to the underlying mechanisms of individual variation in the induction of CAs. The information is also essential in the context of the well-established association between high CA frequency and increased cancer incidence.

The frequency of the chromatid breaks in G2 peripheral blood lymphocytes as an *in vitro* indicator of human individual sensitivity to ionizing radiation

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Abstract

Individuals differ in their response to ionizing radiation related to human individual radiosensitivity. This radiosensitivity has been linked with the sensitivity of human cells to radiation and other DNA-damaging agents. The radiosensitivity of human lymphocytes is commonly assessed by the frequency of micronuclei or chromatid breaks after *in vitro* irradiation of peripheral blood lymphocytes in the G_0- or G_2-phase of the cell cycle.

This study was performed to test the ability of these predictive indicators of intrinsic radiosensitivity for the identification in the normal population of individuals with increased radiosensitivity. The study was carried out with blood samples taken from 29 healthy donors and irradiated with 1 Gy of X rays to induce DNA damage converted into micronuclei or chromatid breaks scored in cultured peripheral blood lymphocytes at interphase (G_o assay) or metaphase (G_2 assay).

The results obtained indicate that chromatid breaks in peripheral blood lymphocyte can be the indicator of choice for the assessment of individual radiosensitivity because they are sensitive and specific to radiation and have low background frequency.

Introduction

Individuals differ in their response to ionizing radiation related to human individual radiosensitivity. There are radiosensitive sub-groups with hypersensitivity to carcinogenic risks (stochastic effects) and hypersensitivity to deterministic effects (Hoeller *et al.*,2003). Identification of individuals with increased inherent radiosensitivity is of relevance for their protection from the adverse effects of radiation after occupational or environmental exposure and after radiotherapy in oncology.Both situations require, however, a good indicator of individual radiosensitivity.

Radiation can induce a large spectrum of DNA damage in human cells. In general, damage located on one DNA strand is more effectively repaired than that affecting both strands. Misrepaired or non-repaired double strand breaks (DSB) are considered as the main lesions involved in cellular radiosensitivity and predisposition to cancer (Pichierri *et al*.,2000). In this regard, radiosensitivity would result from mutations or polymorphism of cancer predisposition genes, which are involved, in the repair of radiation-induced DSBs or their conversion in chromosome damage, which is linked to the signaling of the DSB to the checkpoint mechanism (Terzoudi *et al.,*2009). If the chromosomal radiosensitivity in human cells has been attributed to low DSB repair activity or to an impaired cell cycle control, increased yield of chromosomal damage after in vitro cell irradiation, can be a predictive indicator of radiosensitive individuals and patients. To date two such indicators are commonly used for predicting individual radiosensitivity. These are: micronuclei scored at interphase after the in vitro irradiation of human peripheral blood lymphocytes in the G_o-phase of the cell cycle (Champion *et al.,* 1995) and chromatid breaks scored at first metaphase following in vitro G_2-phase lymphocyte irradiation (Terzoudi *et al.,* 2006, Terzoudi *et al.,* 2009). Micronuclei and chromatid breaks are considered to result from non- or misrepaired DSBs (Natarajan, 1993). They are known to be repaired by the two processes non-homologous end joining (NHEJ) and homologous recombination (HR) that depends on the phase of the cell cycle. While NHEJ was shown to be active in all cell phases, HR is involved only in late S and G2 phase (Bentomane, 2004; Jeggo, 1990). In cultured peripheral blood lymphocytes DSBs induced in G_o-phase are processed during G_1 to S-phase transition so that the yield of residual lesions visualized and quantified at interphase cells as micronuclei reflects the G_1 checkpoint potential to facilitate recognition and repair of radiation induced DSBs before entering the S phase. Consequently, the yield of G_2 chromatid breaks represents the effect of the G_2 checkpoint during G_2 to M- phase transition of cultured peripheral blood lymphocytes. A variation in the yield of micronuclei or chromatid breaks between individuals has been correlated to variations in individual radiosensitivity (IR). Since inter-individual variation in radiosensitivity obtained for examined individuals are described by normal distribution with a mean value (MV) and a standard deviation of the mean (SD), individuals may be classified as normal when IR=MV±SD, resistant when IR< MV-SD and sensitive when IR> MV+SD (Terzoudi *et al*., 2006).

Both predictive assays for individual radiosensitivity were adapted at the CLOR for monitoring of occupation health and safety. The aim of this study was to test the ability of these assays for the identification of individuals with increased radiosensitivity in order to direct to more relevant screening for cancer susceptibility.

Material and methods

Subjects and blood donors

The subjects of this study comprised 29 healthy female and male donors employed at CLOR. Informed consent was obtained from each of the donor after an explanation of the objective of the study.

Whole blood samples (about 5 ml) were collected into heparinized tubes. Half of blood was irradiated with 1 Gy of X-rays for inter-individual analysis of variations in radiosensitivity measured using G_0- and G_2 -assays, while the other half was kept as a non-irradiated control. Peripheral blood lymphocytes were cultured adding 0.5 ml of whole blood to 4.5 ml PB-MAX Karyotyping Medium (Gibco). This medium was composed of liquid PRMI-1640 basal medium supplemented with 10% Fetal Bovine Serum, 1% phytohemagglutinin (PHA), 1% L-glutamine, gentamicin sulfate at 35 mcg/ml and 1% heparin (Sigma).

Irradiation

Irradiation of whole blood or proliferating peripheral blood lymphocytes was performed at room temperature using a PANTAK X-ray machine operating at 243,0 kV, 10 mA with additional filtration by 1,62 mm Cu and 4 mm Al. The average energy of X-rays was 122 kV. The measured dose rate at the distance of 50 cm from a focus, which was a point of irradiation, was 0,35 Gy/min. Before every irradiation the standard value of dose was measured by PTW-UNIDOS electrometer with 0,6 cc volume ionization chamber.

G_o-assay

G_o-assay was performed according to the cytochalasin-B-blocked micronucleus assay described by Fenech *et al.,* 1986. Briefly, immediately after irradiation duplicate lymphocyte cultures were incubated for 72 hours at 37^{o}C in a humidified incubator in an atmosphere of 5% $C0_2$ and 95% air. After 48 h of stimulation with PHA, cells were treated for 24 h with 6µl/ml of cytochalasin-B (Sigma) in order to block cytokinesis. Thereafter, cells were submitted to hypotonic stress (0.075M KCl) for 20 minutes and fixed three times in 3:1(v/v) methanol/acetic acid. Fixed cells were pooled and dropped on preclined wet slides, air-dried for at least 24 h and stained with 2 % Giemsa solutions for 10 minutes. Micronuclei were scored in binucleated cells with well-preserved cytoplasm with a light microscope (Nikon) using a 1000x magnification. 1000 binucleated cells from various slides were scored for each donor and at a given dose. Micronuclei were considered if their dimension did not exceed one third of the diameter of the main nucleus, with the exception micronuclei touching the main nuclei as well as free-lying nuclei.

G_2-assay

G_2-assay was carried out as described by Terzoudi and Pantelias, (2006). Duplicate lymphocyte cultures were incubated for 69-70 hours at 37^{o}C in a humidified incubator in an atmosphere of 5% $C0_2$ and 95% air before their use for radiosensitivity measurements. Half of proliferating cells were then irradiated with 1 Gy and incubated for 30 minutes at 37^{o}C to allow division of cells irradiated at mitosis. Lymphocytes were than blocked in metaphase by the addition of 0.15µl/ml Colcemid (Gibco) to the cell cultures (Gibco) for 60-90 min. Further preparations of metaphase arrested cell was similar to that described for the G_o-assay. However, for better scoring of chromatide beaks an additional wash with 5% acetic acid was added after the hypotonic stress. The aim of this modification was to produce metaphases with well spread chromosomes as free as possible from the surrounding cytoplasm or nucleoplasm. For each donor and dose, 250 metaphase cells were scored for chromatid breaks. Chromatid gaps were scored as breaks only when they were longer than a chromatid width (Fig.2.).

Results

To classify healthy individuals according to their radiosensitivity, we analysed the frequency of chromatid breaks in peripheral blood lymphocytes from 19 donors. The mean age in this group was equal to 41 years and the range of age was between 25 and 64 years. Only for one donor we found 2 spontaneous chromosome breaks in 250 scored cells that was substracted to obtain the radiation-induced yield of aberrations. The mean, standard deviation of the mean, median, lowest and highest values of chromatid breaks frequency in control and irradiated cells are sumarized in Table 1. When data were analysed using a normal distribution, the mean value of the frequency of radiation-induced chromatid breaks was determined to be 0.237 ± 0.073 per lymphocytes. The ratio of highset to lowest values is also given and indicates the range of individual variability. The ratio is 2,61. In Table 1 are also presented the mean, lowest and highest values of the micronuclei frequency for both spontaneous and radiation induced micronuclei. Results were obtained in our preliminary experiments with the chromosomal sensitivitiy of human lymphocytes. Blood samples were taken from 10 healthy donors in the age between 22 and 61 years. The mean micronucleus frequency was determined to be 0.019±0.013 (ratio 0,004-0,048) per cell for spontaneous micronuclei and 0.061±0.011(ratio 0,042-0,076) for radiation-induced ones. The ratios of highest/lowest values for spontaneous and induced micronuclei are 12 and 1,81, respectively. This indicates that a significant variation exists between individuals in the frequency of spontaneous micronuclei, which reflects the level of cumulative damage occuring during the live time of lymphocytes. Radiation-induced micronuclei that reflect chromosomal radiosensitivity of human limphocytes also vary between individuals. However, our results indicate that the variability of radiation-induced micronuclei among the10 donors appears to be about one and half fold lower that found for radiation-induced chromatid breaks (Table 1).

The inter-individual variation in radiosensitivity obtained using chromosomal radiosensitivity determined in G_2- phase lymphocytes shows Figure 1. The theoretical clasification (Terzoudi *et al.*, 2006) in resistant (< MV-SD), normal (MV±SD) and sensitive (> MV+SD) results in this case in 16 normal and 3 sensitive donors. Figure 3 shows the inter-individual variation in chromosomal radiosensitivity in G_0-phase lymphocytes. According to above criteria 1 of 10 donors was classified as resistant, 7 as normal and 2 as sensitive.

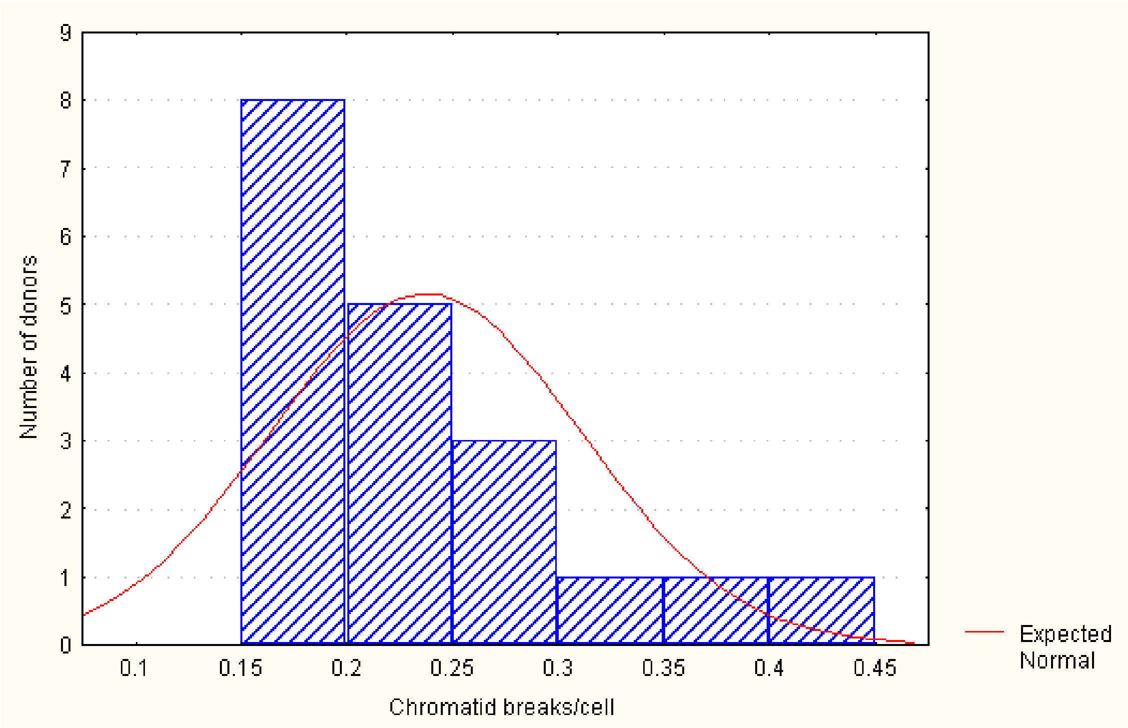

Fig. 1. Inter-individual variation in radiosensitivity obtained from 19 healthy blood donors using the G_2 assay.

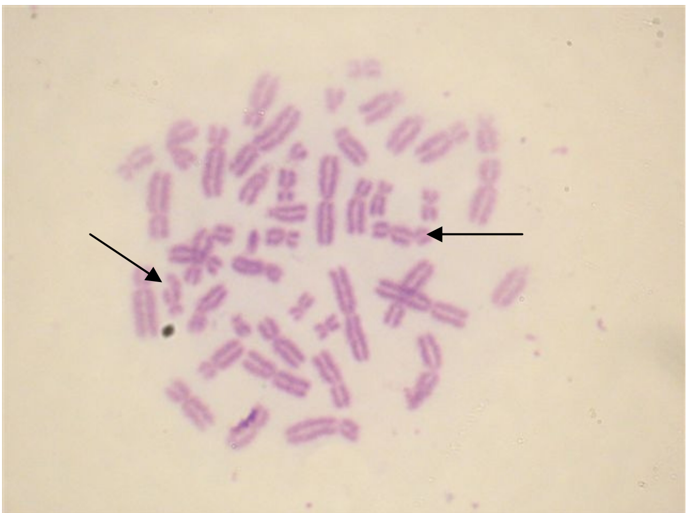

Fig. 2. Chromatid brekas after 1 Gy of X-ray irradiation as visualized at methaphase pheripheral blood lymphocyte from a healthy donor using G_2 assay. Notice two chromatid breaks.

Table 1. Mean, standard deviation of the mean, median, lowest and highest values of micronucleus (MN) frequency in 10 healthy donors and chromatid breaks(CB) frequency in 19 healthy donors.

Variables	Chromatid breaks/cell spontaneous	Chromatid breaks/cell induced	Micronuclei/cell sponatneous	Micronuclei/cell induced
No. of donors	19	19	10	10
Mean	0.0004	0.237	0.019	0.061
SD	0.0018	0.073	0.013	0.011
Median	0,000	0.209	0.017	0.061
Lowest	0.000	0.164	0.004	0.042
Highest	0.008	0.428	0.048	0.076
Highest/Lowest	-	2.61	12	1.81
Mean-SD	-	0.164	-	0.050
Mean+SD	-	0.310	-	0.072

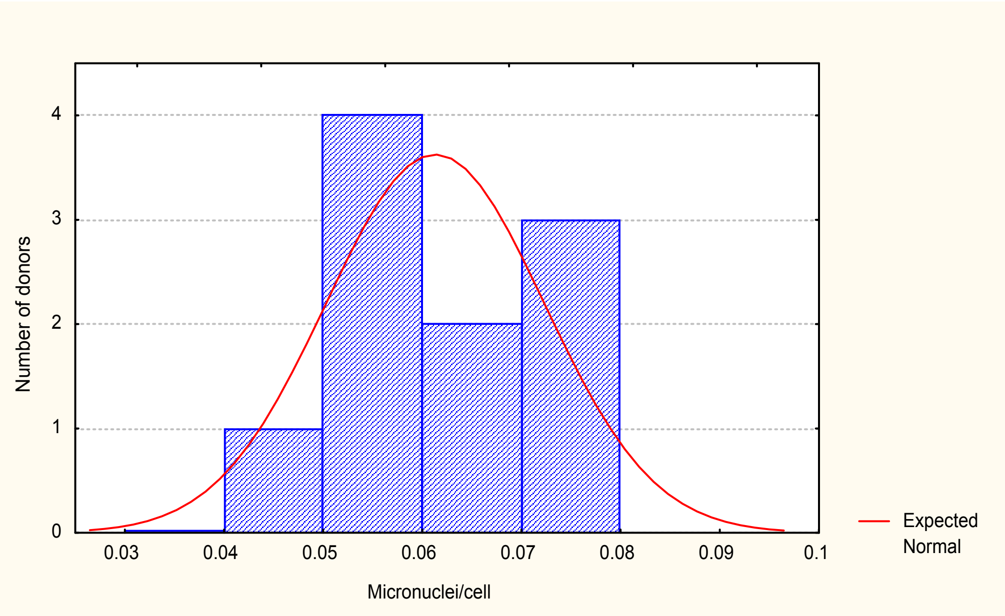

Fig. 3. Inter-individual variation in radiosensitivity obtained from10 healthy blood donors using.

Conclusions

In conclusion: micronuclei are less specific to ionising radiation than chromatid breaks because they detect not only radiation-induced chromosome breaks but also chromosome loss induced by chemical agents (Fenech *et al.,* 1986). They are also less sensitive than chromatid breaks (MV=0,061 vs. MV=0. 0.237), and have high and variable background frequency (range 4 - 48 per 1000 cells). Therefore, the increased frequency of radiation-induced chromatid breaks in peripheral blood lymphocyte can be the indicator of choice for screening a large number of individuals for hypersensitivity or for the assessment of individual radiosensitivity in radiation therapy.

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Comparative cytogenetic analysis of chromosomal aberrations and premature centromere division in persons exposed to radionuclides

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Abstract

The aim of this research is determining correlation between the frequency of premature centromere division (PCD) in metaphases in persons professionally exposed to radionuclides and noted chromosome aberrations (CA). Biological dosimetry was performed by conventional cytogenetic technique. The presence of PCD was confirmed by fluorescent in situ hybridization (FISH). The assay of pL1.84a repetitive DNA for chromosome 18 was used for detection of centromeric region. The analysis included 50 subjects from the Clinical Centre of Serbia (C) (the average age of 45.24±1.18 and the average exposition time 17.96±1.15), and 40 subjects from the control group (K) (the average age of 44.40±0.98 and the average years of 19.67±0.98), which were not exposed to genotoxic agents in their workplaces. The results showed that frequencies of CA and PCD are statistically significantly higher in subject exposed to radionuclides than in the control group (Mann-Whitney U test, P<001). The number of isochromatid breaks was statistically significantly higher in group of exposed to radionuclides (smokers compared to non-smokers) (Spearman's test, r=0.32; P<0.05). Analysis showed that there was a positive correlation between PCD and dicentrics, acentric fragments and chromatide breaks (Spearman's test, r=0.49, 0.54, 0.29, P<0.05). Applying the FISH method showed the presence of PCD in metaphases and interphase nuclei in subjects exposed to radionuclides. Considering PCD as manifestation of genomic instability, our research suggest that this phenomenon could be used as a possible cytogenetic biomarker for proffesional exposure to ionizing radiation.

Key words: chromosome aberrations, ionizing radiations, lymphocytes of peripheral blood, premature centromere division

Introduction

In the occupational pathology, discovery of radiation damages is extremely important, especially in providing the proof that certain changes in health status of an patient may be the consequence of chronicle exposure to ionizing radiation. A number of researchers have examined the impact of small radiation doses among the people employed by health institutions, and it was concluded that there was an increase in incidence chromosome aberrations in comparison to the control group (Ghiassi-Nejad et al 2002, Jovićić et al 2009, Milačić 2005, Garaj-Vrhovac et al 2006).

Beside the classical researches related to the impact of ionizing radiation to lymphocytes in peripheral blood in the exposed persons, attention has been recently paid to new approaches and manners of analyzing necessary for definition of cytogenetic markers, all aimed at recognition of harmful effects of ionizing radiation. In addition to classical analysis of chromosome aberrations (CA), the attention is focused to the detection of premature centromere division (PCD).

This phenomenon, found in lymphocytes of peripheral blood, is characterized by divided chromatides of chromosomes already in metaphase of cell cycle. However, if premature division of constitute heterochromatin occurs at centromere level (PCD), then we may observe the phenomenon of aneuploidy, which has been confirmed in large number of papers (Vig et al 1974).

Disturbance of centromere in the form of premature centromere splitting (Eng-centromere splitting, centromere spreading), centromere puffing, actually presents manifestation of disturbance in spatial and time mechanisms of mitosis process with various types of neoplasia, which may further lead to genetic instability (Litmanović et al 1998). The answer to the question whether PCD causes aneuploidia in carcinogenic cells is not known, but it is sure that there are indications today which support the thesis which pertains to correlation between PCD and aneuploidia. The aim of our research was focused on determination of PCD prevalence, as well as on determination of correlation between PCD and chromosome aberrations among people who are occupationally exposed to radionuclides. In addition, the idea whether PCD could be observed as a possible cytogenetic biomarker among people occupationally exposed to radiation was also taken into consideration. In order to confirm presence of PCD at certain chromosome in metaphases and interphase nuclei and in order to answer the question at which stage of cell cycle centromere regions split, we have applied fluorescent *in situ* hybridization (FISH).

Material and methods

Our research included a group of 50 patients employed by Clinical Centre of Serbia, who are occupationally exposed to radionuclides (C) and 40 patients from the control group (K), who have never been exposed to physical or chemical agents in their workplaces (Table 1). The effective radiation doses were measured by thermoluminiscent dosimeter (TLD) once a month during their occupational exposure. Biological dosimetry was carried out by the means of modified micromethod for lymphocytes of peripheral blood and conventional cytogenetic technique for the analysis of chromosome aberration (Moorhead et al 1983, IAEA: Biological dosimetry 1986). The analysis of CA and PCD was examined in 200 metaphase cells. In order to confirm the results of premature segregation of centromere regions, FISH was applied.

The results in this paper were processed by applying Statistics 5 (StatSoft, Inc) and SAS 6.12 software for PS.

Table 1. General characteristics of the exposed and control patients groups

		Control (N=40)	Exposed (N=50)	*P*-value
Age	Mean±SE	44.40±0.98	45.24±1.18	0.599
	Min-Max	34-60	27-59	
Sex				
Male	N	16	23	0.568
Female	N	24	27	
Smoking status				
Smokers	N	12	22	0.173
Non-smokers	N	28	28	
LT-labour time	Mean±SE	19.67±0.98	21.94±1.13	0.144
	Min-Max	13-36	5-35	
ERS-exposition	Mean±SE	/	17.96±1.15	
	Min-Max	/	5-35	
Effective dose (2008) (mSv)	Mean±SE	/	2.19±0.27	
	Min-Max	/	0-10.10	
Effective dose (2004-2008) (mSv)	Mean±SE	/	9.87±1.31	
	Min-Max	/	0.16-47.38	

Results and discussion

The results obtained in the researches of CA and PCD in persons occupationally exposed to the low doses of ionizing radiation are provided in Table 1. On the basis of statistic data (Mann-Whitney U test), we have reached the conclusion that frequency of CA and PCD is more significant in lymphocytes of peripheral blood of people from the exposed group than of those from the control one ($p>0.05$), Table 2. We may pose a question whether frequency of PCD may be related to artifacts, or presence of PCD really may be seen as biological response of a cell to the effects of ionizing radiation. Some authors suggest that presence of PCD in human kariotype is not accidental (Bühler et al 1987).

If the presence of PCD was accidental, than such presence should be recorded in the control group as well. However, frequency of PCD is far lower in the control than in the exposed group. Presence of PCD recorded to a smaller extent in the control group implies a conclusion that PCD may have occurred there as a consequence of the effects of outer environmental factors.

Table 2. Frequency of CA and PCD in limphocytes of the exposed groups and control ones (ACF - acentric fragments, HB – chromatide breaks, iHB – isochromatide breaks, tPCD – metaphases with more than 10 PCD)

		Control	Exposed	*P*-value
No of aberrated cells	Mean±SE	0.85±0.16	4.06±0.15	0.000
	Min-Max	0-4	2-7	
Dicentrics	Mean±SE	0.13±0.05	0.66±0.11	0.001
	Min-Max	0-1	0-2	
Ring	Mean±SE	/	0.12±0.05	0.334
	Min-Max	/	0-1	
ACF	Mean±SE	0.37±0.09	2.64±0.17	0.000
	Min-Max	0-2	0-5	
HB	Mean±SE	0.33±0.07	0.68±0.07	0.004
	Min-Max	0-1	0-1	
iHB	Mean±SE	0.17±0.06	0.5±0.07	0.008
	Min-Max	0-1	0-1	
tPCD	Mean±SE	0.25±0.09	2.12±0.25	0.000
	Min-Max	0-2	0-7	
PCD 1-5	Mean±SE	4.05±0.25	8.5±0.41	0.000
	Min-Max	1-7	2-14	
PCD 5-10	Mean±SE	1.72±0.22	4.64±0.23	0.000
	Min-Max	0-5	2-9	

Statistic data have shown that there was no difference in frequency of CA and PCD between smokers and non-smokers in the exposed group. Our researches are partially in accordance with the authors' researches (Chung et al 1996, Lovreglio et al 2006) who have established that smoking does not affect chromosome aberrations, although there are studies which proove opposite (Sierra-Torres et al 2004). However, further statistic analysis has shown that there is a significant correlation between smoking and frequency of isochromatide breaks in the exposed group of patients (Spearman's test, r=0.32; $P<0.05$). Among the patients from the control group, statistic data indicate that frequency of chromatide breaks is significantly higher among smokers than among non-smokers (means±SE: smokers 0.67±0.14 vs. non-smokers 0.18±0.07). In addition, it was established that frequency of acentric fragments was significantly higher among women than among men in the control patients group (women 0.54±0.12 vs. men 0.12±0.09).

It is important to emphasize that frequency of tPCD among the occupationally exposed people was positively correlated with the dicentric frequency, acentric fragments and chromatide abruptions (Spearman's test, r=0.49, 0.54, 0.29, $P<0.05$).

Further analysis has established that there was no positive correlation between the measured effective doses among patients and frequency of chromosome breaks and PCD (Andreassi et al 2009).

Applying the FISH method we have established that centromere division occurs already in the cell cycle interphase, i.e. in G2 phase of the cell cycle. In addition, we have confirmed the presence of PCD in metaphases and interphase nuclei of patients exposed to radionuclei (Table 3).

Table 3. Distribution of fluorescent signals for centromere region of chromosome 18 in metaphases and in interphase nuclei by applying DNA repetitive probe, both to the exposed and control groups

	INTERPHASE NUCLEI		
Patients	**Number of analyzed nuclei**	**Without PCD**	**With PCD**
1.	83	70 (84.337%)	13 (15.663%)
2.	102	95 (93.137%)	7 (6.863%)
		METAPHASES	
	Number of analyzed metaphases	**Without PCD**	**With PCD**
1.	28	14 (50%)	14 (50%)
2.	24	24 (100%)	0 (0%)

Conclusions

Our researches suggest the possibility to use PCD as possible cytogenetic biomarker in persons occupationally exposed to radionuclides. It is necessary to emphasize that these results include preliminary researches in relatively small sample. Our future researches will include larger number of patients in combination with *in vitro* researches, which will surely contribute to better understanding of this phenomenon.

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Population study of chromosomal aberrations in aircrew members compared to a control group

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Abstract

Aircrew was included as an occupational exposed radiation group in the last report of the International Commission for Radiation Protection (ICRP 90).

Chromosome translocations are a biomarker of cumulative exposure to external ionising radiations better than chromosome dicentrics, we analysed the frequency of translocations from peripheral blood of a group of aircrew members and controls matched for the age, smoking habits and gender.

Metaphases spreads obtained after incubation of peripheral blood lymphocytes in a PHA enriched culture medium for 48 hours at 37ºC (laboratory routine procedures). Chromosomes 1 and 2 painted red and a pancentromeric probe used simultaneously to coloured all centromers green; to distinguish translocations and dicentrics, this procedure gives an efficiency of 28% when extrapolating to the full genome. There was analysed around 2000 metaphases per person, which means 560 cell equivalents.

Our results do not show statistically significance differences in translocations frequency between both populations.

Introduction

The presence of chromosomal aberrations in peripheral blood lymphocytes provides a good biomarker of exposure to ionizing radiation. The analysis of dicentric chromosomes is the "gold standard" for estimating doses in cases of accidental overexposure to ionizing radiation. The main limitation of this type of dosimeter is its instability, because of altered morphological involved, cells carrying dicentric fail its mitosis and these alterations are not transmitted to daughter cells, so the post exposure frequency of dicentric decreases over time.

The translocations are another type of chromosomal aberration induced by ionizing radiation by the same mechanism as dicentric chromosomes, but do not lead to morphological changes that hinder the cell mitosis, so its frequency will remain constant throughout the time and also because of its potential accumulation, allow the identification of chronic exposures.

The development of hybridization techniques with specific DNA probes has facilitated "paint" chromosome translocations to identify quickly and easily, allowing the analysis of a large number of cells and thus calculating the frequency of translocations routinely in biological dosimetry studies.

To study the biological impact of ionizing radiation on chronically exposed populations, the most appropriate method is fluorescence in situ hybridization (FISH) as it has proven that it can detect an increase in the frequency of translocations due to such exposure.

With increasing altitude increases exposure to ionizing radiation from solar and cosmic origin, so many countries has included aircrew as occupationally exposed, as recommended by the UNSCEAR 2000 (United Nations Scientific Committee on the effects of atomic radiation). Ionizing radiations at altitudes used in long haul flights are mainly from neutron and gamma radiation but are also detected protons, alpha particles and heavy nuclei. It has estimated that 60% of crew exposure to cosmic radiation is due to radiation of high linear energy transfer (LET), especially from neutrons generated by its interaction with the atmosphere.

Because of the proven relationship between the frequency of chromosome aberrations and increased cancer risk, is necessary to perform frequency studies in populations exposed to ionizing radiation, among which are the air crews.

This report describes the work done at the Biological Dosimetry laboratory in Hospital General Universitario Gregorio Marañón, on the collaboration agreement between the Foundation for Biomedical Research at University General Hospital Gregorio Marañón, Iberia Airlines of Spain, and Co-Muprespa Fraternity.

Material and methods

Population Description

To determine the increased risk of a given population associated with a factor, it is important to compare it with another population with similar characteristics but not exposed to that factor. Therefore, since the objective of this work is to determine whether there is an increased risk of disease associated with exposure to ionizing radiation of cosmic origin, between the flight crew in Iberia, this work includes the comparison of two group's persons:

- Aircrew members for intercontinental flights above 9000 meters, which must meet the following requirements:
  Being active
  At least 10 years of continuous service with full schedules.
  At least 5 years of continuous service in long-haul fleets, preferably A-340
- Iberia ground staff.

Matching populations for age, gender and smoking habits
The grounds for exclusion to be part of the study are:

- Being younger than 45 years at the time of blood collection
- Radiological studies in the last year TGI type, barium enema, urography descending, thoracoabdominal CT, and interventional radiology.
- Previous diagnosis of cancer or other disease treated with chemotherapy and / or radiotherapy
- Work history in professions associated with cancer risk
- Substance abuse

All individuals who are part of the work were selected by The Iberia Medical Service, which has also obtain and complete the necessary information and informed consent in addition to blood collection and shipment to Biological Dosimetry Laboratory at HGUGM.

LABORATORY METHODOLOGY

Sample consists of three tubes of blood per person, sent to the laboratory together with the informed consent document and the name. Manipulation of samples performed as appropriate to the laboratory culture and processing protocols.

Fluorescence in situ hybridization FISH

Hybridization is performed on 3 consecutive days following the standardized protocol in the laboratory which allows the hybridization of chromosomes 1 and 2 with rhodamine labelled probes, a FITC pancentromeric labelled probe, and a DAPI counterstained.

Analysis of metaphases

The analysis done blind, ie the slides coded with a number assigned by the laboratory so that the analyst does not know the origin and characteristics of the individual studied.

The analysis performed by fluorescence microscope (Leica DM5000B) with appropriate filters for visualization of rhodamine, FITC and DAPI. We analyzed about 2000 metaphases per individual, discarding people who could not be analyzed more than 600 metaphases.

There have photographed all chromosome abnormalities found but for the study only account translocations in stable cells, ie those without dicentric and acentric fragments or rings, and selected only cells where there are close to 46 chromosomes and in which all material paint is present.

Results

The study done in a total number of 80 people, 38 controls and 42 aircrews, analysed 67.297 metaphases from control samples and 69.690 from aircrew samples.

We analysed translocations involving chromosomes 1 and 2 which constitute about 16% (16.32 in males or 16.05 in women) of the cellular genome, we calculate the genomic frequency of translocations using the formula described by Lucas Group 1992, for our technique we obtain an approximate efficiency of 28% (27.99% to 27.62% for men and women)

The following tables show the data obtained for each population: 38 controls and 42 crew (tables 1 and 2 respectively), as were sent to the Health Protection Agency (HPA, Chilton Didcot Oxford, UK) for statistical analysis.

Table 1. Control population data. 38 people analysed, genome equivalent = metaphases analysed x f. % translocations to genome = translocations / genome equivalent x 100. Smoking habits: S = smokers. NS = non smokers, included people who gave up smoking at least 10 years ago.

Nº	Gender	f	analysed met.	genome equivalent	GE (gender)	Translo-cations	%trans. to genome	Age	smoking habit
13	M	0,27996	1424	398,72	398,66	1	0,251	57	NS
44	F	0,27622	2069	579,32	571,49	0	0,000	50	NS
45	M	0,27996	2103	588,84	588,76	3	0,509	58	NS
46	M	0,27996	2009	562,52	562,44	2	0,356	52	S
47	M	0,27996	2001	560,28	560,20	1	0,178	52	NS
51	M	0,27996	2003	560,84	560,76	2	0,357	45	S
52	F	0,27622	2171	607,88	599,67	1	0,165	50	S
53	F	0,27622	2005	561,4	553,81	0	0,000	53	NS
54	F	0,27622	1998	559,44	551,88	0	0,000	59	S
55	M	0,27996	2010	562,8	562,72	0	0,000	60	S
56	M	0,27996	2011	563,08	563,00	3	0,533	49	NS
57	M	0,27996	2015	564,2	564,12	4	0,709	59	NS
58	M	0,27996	2007	561,96	561,88	2	0,356	53	S
61	F	0,27622	2086	584,08	576,19	0	0,000	53	NS
63	M	0,27996	2043	572,04	571,96	2	0,350	56	S
64	M	0,27996	2001	560,28	560,20	1	0,178	59	NS
65	F	0,27622	1603	448,84	442,78	0	0,000	58	NS
66	M	0,27996	1624	454,72	454,65	2	0,440	56	NS
67	M	0,27996	1995	558,6	558,52	5	0,895	51	S
68	M	0,27996	2153	602,84	602,75	3	0,498	47	NS
72	F	0,27622	1851	518,28	511,28	2	0,386	54	NS
73	F	0,27622	2049	573,72	565,97	5	0,872	54	NS
74	M	0,27996	2002	560,56	560,48	3	0,535	45	NS
75	M	0,27996	1942	543,76	543,68	1	0,184	62	NS
76	M	0,27996	2050	574	573,92	1	0,174	55	NS
78	M	0,27996	2017	564,76	564,68	2	0,354	48	NS
113	M	0,27996	1094	306,32	306,28	1	0,326	49	NS
116	M	0,27996	1185	331,8	331,75	3	0,904	57	S
118	M	0,27996	982	274,96	274,92	0	0,000	55	S
119	M	0,27996	1529	428,12	428,06	2	0,467	49	S
120	M	0,27996	2010	562,8	562,72	2	0,355	58	NS
121	M	0,27996	2023	566,44	566,36	2	0,353	48	S
122	F	0,27622	822	230,16	227,05	0	0,000	48	NS
124	M	0,27996	1065	298,2	298,16	3	1,006	48	NS
131	F	0,27622	654	183,12	180,65	2	1,092	48	NS
132	F	0,27622	1335	373,8	368,75	0	0,000	45	NS
133	F	0,27622	1338	374,64	369,58	3	0,801	54	NS
134	F	0,27622	2018	565,04	557,40	1	0,177	52	NS

Table 2. Aircrew data. 42 people analysed, genome equivalent = metaphases analysed x f. % translocations to genome = translocations / genome equivalent x 100. Smoking habits: S = smokers. NS = non smokers, included people who gave up smoking at least 10 years ago.

n	Gender	f	analysed metaphases	genome equivalent	GE (gender)	translocations	% transl to genome	Age	smoking habit
17	F	0,27622	1144	320,32	315,99	0	0,000	54	NS
18	M	0,27996	2049	573,72	573,64	7	1,220	54	NS
24	F	0,27622	1430	400,4	394,99	1	0,250	47	NS
25	F	0,27622	2342	655,76	646,90	0	0,000	47	NS
26	M	0,27996	1438	402,64	402,58	2	0,497	58	S
27	M	0,27996	2024	566,72	566,64	3	0,529	54	S
28	M	0,27996	625	175	174,97	2	1,143	54	S
29	M	0,27996	1677	469,56	469,49	3	0,639	54	S
30	F	0,27622	2023	566,44	558,79	2	0,353	54	NS
31	M	0,27996	2012	563,36	563,28	1	0,178	59	NS
37	F	0,27622	2000	560	552,43	1	0,179	59	NS
42	F	0,27622	2239	626,92	618,45	1	0,160	51	S
60	M	0,27996	2041	571,48	571,40	0	0,000	54	NS
62	F	0,27622	2048	573,44	565,69	5	0,872	59	NS
82	M	0,27996	1215	340,2	340,15	0	0,000	59	NS
84	M	0,27996	769	215,32	215,29	0	0,000	53	NS
85	F	0,27622	588	164,64	162,42	0	0,000	57	NS
86	M	0,27996	863	241,64	241,61	1	0,414	58	S
87	M	0,27996	2017	564,76	564,68	0	0,000	51	NS
88	F	0,27622	2070	579,6	571,77	2	0,345	59	NS
90	M	0,27996	708	198,24	198,21	0	0,000	56	NS
91	F	0,27622	2040	571,2	563,48	6	1,050	54	NS
92	F	0,27622	2086	584,08	576,19	1	0,171	45	S
96	M	0,27996	2014	563,92	563,84	3	0,532	52	NS
97	F	0,27622	2024	566,72	559,06	4	0,706	48	NS
98	F	0,27622	2038	570,64	562,93	2	0,350	45	S
100	M	0,27996	1997	559,16	559,08	3	0,537	45	NS
101	M	0,27996	2000	560	559,92	2	0,357	54	S
102	F	0,27622	1562	437,36	431,45	1	0,229	50	S
103	M	0,27996	2001	560,28	560,20	3	0,535	52	NS
104	M	0,27996	1821	509,88	509,81	1	0,196	57	S
105	F	0,27622	479	134,12	132,31	1	0,746	51	NS
107	M	0,27996	2008	562,24	562,16	2	0,356	57	S
108	F	0,27622	891	249,48	246,11	1	0,401	54	S
109	F	0,27622	1005	281,4	277,60	0	0,000	52	NS
110	F	0,27622	2002	560,56	552,99	3	0,535	48	S
111	F	0,27622	1935	541,8	534,48	1	0,185	49	NS
112	M	0,27996	2012	563,36	563,28	3	0,533	53	S
117	F	0,27622	1369	383,32	378,14	1	0,261	49	S
123	F	0,27622	1785	499,8	493,05	3	0,600	52	S
125	M	0,27996	2066	578,48	578,40	3	0,519	46	NS
135	M	0,27996	1233	345,24	345,19	0	0,000	46	NS

The following tables 3 and 4 show the statistics of the results obtained.

First two columns correspond to the expected frequencies due to age. These data are used to calculate excess of translocations observed, shown in the next column. Excess of translocations is less than expected; except for those marked with yellow, and both populations have the same number of cases with excess. Apparently, there is a difference between both populations but, when use t test we obtain a p value of 0.065 (when using only baseline due to sex and age) or 0.063 (when using risk rates estimated by the group of Sigurdson). These data imply that, although there is a difference between the two populations, is not statistically significant.

Table 3. Control population statistics.

	Background for age			Adjusted rate ratio				
IB-n	Per 100	Per GE	Excess translocations	age	smoking	gender	RR	Excess
13	0,93	3,71	-2,71	1,37	1,00	1,00	3,22	-2,22
44	1,06	6,06	-6,06	1,25	1,00	0,92	4,27	-4,27
45	0,93	5,48	-2,48	1,37	1,00	1,00	4,76	-1,76
46	0,83	4,67	-2,67	1,25	1,19	1,00	4,94	-2,94
47	0,83	4,65	-3,65	1,25	1,00	1,00	4,13	-3,13
51	0,92	5,16	-3,16	1,27	1,19	1,00	5,00	-3,00
52	1,06	6,36	-5,36	1,25	1,19	0,92	5,33	-4,33
53	1,06	5,87	-5,87	1,25	1,00	0,92	4,14	-4,14
54	0,87	4,80	-4,80	1,37	1,19	0,92	5,38	-5,38
55	1,14	6,41	-6,41	1,78	1,19	1,00	7,03	-7,03
56	0,92	5,18	-2,18	1,27	1,00	1,00	4,22	-1,22
57	0,93	5,25	-1,25	1,37	1,00	1,00	4,56	-0,56
58	0,83	4,66	-2,66	1,25	1,19	1,00	4,93	-2,93
61	1,06	6,11	-6,11	1,25	1,00	0,92	4,31	-4,31
63	0,93	5,32	-3,32	1,37	1,19	1,00	5,50	-3,50
64	0,93	5,21	-4,21	1,37	1,00	1,00	4,53	-3,53
65	0,87	3,85	-3,85	1,37	1,00	0,92	3,63	-3,63
66	0,93	4,23	-2,23	1,37	1,00	1,00	3,68	-1,68
67	0,83	4,64	0,36	1,25	1,19	1,00	4,90	0,10
68	0,92	5,55	-2,55	1,27	1,00	1,00	4,52	-1,52
72	1,06	5,42	-3,42	1,25	1,00	0,92	3,82	-1,82
73	1,06	6,00	-1,00	1,25	1,00	0,92	4,23	0,77
74	0,92	5,16	-2,16	1,27	1,00	1,00	4,20	-1,20
75	1,14	6,20	-5,20	1,78	1,00	1,00	5,71	-4,71
76	0,93	5,34	-4,34	1,37	1,00	1,00	4,64	-3,64
78	0,92	5,20	-3,20	1,27	1,00	1,00	4,23	-2,23
113	0,92	2,82	-1,82	1,27	1,00	1,00	2,30	-1,30
116	0,93	3,09	-0,09	1,37	1,19	1,00	3,19	-0,19
118	0,93	2,56	-2,56	1,37	1,19	1,00	2,64	-2,64
119	0,92	3,94	-1,94	1,27	1,19	1,00	3,82	-1,82
120	0,93	5,23	-3,23	1,37	1,00	1,00	4,55	-2,55
121	0,92	5,21	-3,21	1,27	1,19	1,00	5,05	-3,05
122	0,87	1,98	-1,98	1,27	1,00	0,92	1,72	-1,72
124	0,92	2,74	0,26	1,27	1,00	1,00	2,23	0,77
131	0,87	1,57	0,43	1,27	1,00	0,92	1,37	0,63
132	0,87	3,21	-3,21	1,27	1,00	0,92	2,80	-2,80
133	1,06	3,92	-0,92	1,25	1,00	0,92	2,76	0,24
134	1,06	5,91	-4,91	1,25	1,00	0,92	4,17	-3,17
		Average	-2,99					-2,41
		SD	1,81					1,77
		Var	3,28					3,12

Table 4. Air crew population statistics.

	Background for age			Adjusted rate ratio				
IB-n	Per 100	Per GE	Excess translocations	age	smoking	gender	RR	Excess
17	1,06	3,35	-3,35	1,25	1,00	0,92	2,17	-2,17
18	0,83	4,76	2,24	1,25	1,00	1,00	4,23	2,77
24	0,87	3,44	-2,44	1,27	1,00	0,92	2,76	-1,76
25	0,87	5,63	-5,63	1,27	1,00	0,92	4,52	-4,52
26	0,93	3,74	-1,74	1,37	1,19	1,00	3,87	-1,87
27	0,83	4,70	-1,70	1,25	1,19	1,00	4,97	-1,97
28	0,83	1,45	0,55	1,25	1,19	1,00	1,54	0,46
29	0,83	3,90	-0,90	1,25	1,19	1,00	4,12	-1,12
30	1,06	5,92	-3,92	1,25	1,00	0,92	3,84	-1,84
31	0,93	5,24	-4,24	1,37	1,00	1,00	4,55	-3,55
37	0,87	4,81	-3,81	1,37	1,00	0,92	4,16	-3,16
42	1,06	6,56	-5,56	1,25	1,19	0,92	5,06	-4,06
60	0,83	4,74	-4,74	1,25	1,00	1,00	4,21	-4,21
62	0,87	4,92	0,08	1,37	1,00	0,92	4,26	0,74
82	0,93	3,16	-3,16	1,37	1,00	1,00	2,75	-2,75
84	0,83	1,79	-1,79	1,25	1,00	1,00	1,59	-1,59
85	0,87	1,41	-1,41	1,37	1,00	0,92	1,22	-1,22
86	0,93	2,25	-1,25	1,37	1,19	1,00	2,32	-1,32
87	0,83	4,69	-4,69	1,25	1,00	1,00	4,17	-4,17
88	0,87	4,97	-2,97	1,37	1,00	0,92	4,31	-2,31
90	0,93	1,84	-1,84	1,37	1,00	1,00	1,60	-1,60
91	1,06	5,97	0,03	1,25	1,00	0,92	3,88	2,12
92	0,87	5,01	-4,01	1,27	1,19	0,92	4,79	-3,79
96	0,83	4,68	-1,68	1,25	1,00	1,00	4,16	-1,16
97	0,87	4,86	-0,86	1,27	1,00	0,92	3,91	0,09
98	0,87	4,90	-2,90	1,27	1,19	0,92	4,68	-2,68
100	0,92	5,14	-2,14	1,27	1,00	1,00	4,19	-1,19
101	0,83	4,65	-2,65	1,25	1,19	1,00	4,91	-2,91
102	1,06	4,57	-3,57	1,25	1,19	0,92	3,53	-2,53
103	0,83	4,65	-1,65	1,25	1,00	1,00	4,13	-1,13
104	0,93	4,74	-3,74	1,37	1,19	1,00	4,90	-3,90
105	1,06	1,40	-0,40	1,25	1,00	0,92	0,91	0,09
107	0,93	5,23	-3,23	1,00	1,19	1,00	3,95	-1,95
108	1,06	2,61	-1,61	1,37	1,19	0,92	2,21	-1,21
109	1,06	2,94	-2,94	1,25	1,00	0,92	1,91	-1,91
110	0,87	4,81	-1,81	1,27	1,19	0,92	4,60	-1,60
111	0,87	4,65	-3,65	1,27	1,00	0,92	3,73	-2,73
112	0,83	4,68	-1,68	1,25	1,19	1,00	4,94	-1,94
117	0,87	3,29	-2,29	1,27	1,19	0,92	3,14	-2,14
123	1,06	5,23	-2,23	1,25	1,19	0,92	4,04	-1,04
125	0,92	5,32	-2,32	1,27	1,00	1,00	4,33	-1,33
135	0,92	3,18	-3,18	1,27	1,00	1,00	2,59	-2,59
		Average	-2,40					-1,83
		SD	1,62					1,57
		Var	2,62					2,47
control vs aircraft:		T-test p	0,065					0,063

Conclusions

The frequency of translocations has not statistically significant differences, although there was a slight increase in the frequency of translocations in the crew population that does not exceed observed frequencies in other populations described in other publications.

It would be interesting to extend the study with a larger number of people, as this would allow us to clarify if the trend continues and / or wide, although noted that the frequencies observed in our study are consistent with published for different population groups.

In the individual analysis, it was found that 6 people have a higher frequency of translocations than expected, but if you look at the data shows, that the number of analyzed genomic equivalent in these cases is low, so even if they can take into account the population study, individual study would require testing more metaphases to allow any conclusion.

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Assessment of dose response for chromosome aberrations due to internal alpha-radiation

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Abstract

Chromosome slides with Chromosome 5 painted using mBAND were analyzed for 40 employees of the Mayak Production Association. Mean frequency of stable intra-chromosomal aberrations per total cells analyzed was 0.005±0.007. Correlation analysis indicated the lack of a significant correlation between external dose and the yield of intra-chromosomal aberrations (r = -0.02). A linear dependence of the yield of intra-chromosomal aberrations on absorbed dose of internal exposure to incorporated ^{239}Pu was revealed: $Y = (0.0303 \pm 0.0065)D_{\alpha}$, where: Y is the yield of intra-chromosomal aberrations (in % per 100 cells); D_{α} is the absorbed dose to bone marrow due to internal alpha-radiation (cGy).

Introduction

The most sensitive and well-known bioindicator of radiation is the yield of chromosome aberrations in lymphocytes from peripheral blood in human (IAEA, 2001). The peripheral blood lymphocytes are a simple and unique model to study radiation-induced mutagenesis. A low spontaneous level of chromosome aberrations in lymphocytes from peripheral blood of healthy donors and high radiosensitivity allow detecting an excess in the yield of radiation-induced chromosome aberrations over spontaneous rate even at low doses (IAEA, 2001; Tucker JD, 2001; Balakrishnan S and Rao SB, 1999).

Many studies have demonstrated that radiation-induced chromosome aberrations are long-lasting in human and can be used as a biological indicator of radiation exposure Aberrations imply the DNA double-strand breaks and repair defects. Stable intra-chromosomal aberrations (e.g. para- and pericentric inversions) are produced by misrepair of pairs of chromosome breaks (DNA double-strand breaks) within a single chromosome; intra-chromosomal aberrations may be contrasted with inter-chromosomal aberrations, produced by misrepair of pairs of chromosome breaks in different chromosomes (Brenner DJ, Sachs RK, 1994).

The theoretical background relates to the observation that densely-ionizing radiations, such as alpha-particles or neutrons, produce uniquely elevated intra-chromosomal aberrations in contrast with almost all other mutagens, such as gamma-rays or general aging processes.

This implies that alpha-particles or neutrons produce multiple chromosome breaks in each of a small number of chromosomes, as compared to X-rays or chemicals, or endogenous aging processes, where a comparable exposure (i.e. yielding the same total number of chromosome breaks) results in fewer breaks in larger numbers of chromosomes, predominantly inter-chromosomal aberrations.

The objective is assessment of dose response for the yield of intra-chromosomal aberrations and internal dose from alpha-radiation.

Material and methods

A cytogenetic analysis of the peripheral blood lymphocytes was performed for 40 workers of the Mayak PA, who were exposed to external and internal radiation. Most individuals analyzed were males (77.5%). External doses from gamma-rays ranged from 0.021 up to 5.726 Gy; ^{239}Pu body burden was 0-10.5 kBq. External doses to red bone marrow (RBM) from gamma-rays ranged within 0.130-4.262 Gy; absorbed dose from alpha-radiation due to incorporated ^{239}Pu was 0.003-0.538 Gy.

For the purposes of cultivation, whole blood samples were drawn from ulnar vein by a heparinized syringe, 5 ml. The lymphocytic pellet and 10 ml of PB-MAX medium were placed in a culture flask. The flask was then incubated in the CO_2 incubator at 37°C for 48 to 68 hrs. Four hours prior to fixation, 0.01 μg/ml colchicine was added in the flask. The pellet was resuspended in 0.075 M KCl and placed in a 37°C water bath for 15 minutes. The pellet was then resuspended in fixative (3:1 ethanol: glacial acetic acid, cooled to 0^0C). The cell suspension was placed on slides and allowed to air dry.

Hybridization of chromosome slides using mBAND was performed in compliance with the XCyte Lab Manual (Metasystems, Germany). The slides were denatured in 0.07 N NaOH for 1 minute at room temperature. mBAND probes were denatured in parallel with denaturation of the slides. The slides were placed in a humidified chamber in a 37°C incubator for 48 hrs. For detection of hybridization signals, a mixture of blocking buffer and detection solution was used. DAPI/antifade was applied as a counterstain.

The slides were analyzed using a fluorescence imaging microscope (Leica DM RA2) with filter sets for DAPI, FITC, Texas Red, Spectrum Orange, DEAC, Cy5. Images were captured using the MetaSystems camera. Although using mBAND only two chromosomes were colored, whole metaphases were also captured to observe any translocations. Karyotyping was performed using Isis 4 software (MetaSystems, Germany). Statistical analysis was performed by standard linear regression methods using Statistica software (Borovikov V, 2003).

Results

Chromosome slides from 40 Mayak workers were analyzed using mBAND fluorescence *in situ* hybridization for chromosome 5. The slides were analyzed to account for stable intra-chromosomal aberrations, such as para- and pericentric inversions, terminal and interstitial deletions, and insertions. Aberrations were scored per 100-150 metaphases. Total 4,825 cells were analyzed. The mean yield of stable intra-chromosomal aberrations per total number of analyzed cells was 0.005 ± 0.007.

A high yield of intra-chromosomal aberrations (4.1 per 100 cells) was found for one worker. Analysis of medical records indicated malignant neoplasm for this individual. Thus, results of cytogenetic analysis based on chromosome slides from this worker were excluded from the statistical analysis.

As demonstrated earlier, the yield of intra-chromosomal aberrations in a group of workers at the reactor complex did not actually differ from that in the control group (Hande MP et al, 2003; Mitchell CR et al, 2004). In addition, no correlation of the yield of intra-chromosomal aberrations with age was found. The correlation analysis revealed no significant correlation between the yield of intra-chromosomal aberrations and external dose (r = -0.02). The analysis indicated a linear dependence of the yield of stable intra-chromosomal aberrations in the peripheral blood lymphocytes on absorbed dose to red bone marrow (RBM) from internal exposure to alpha-radiation from incorporated ^{239}Pu:

$$Y = (0.0303 \pm 0.0065)D_{\alpha}, \qquad (1)$$

where:

Y is the yield of intra-chromosomal aberrations (in percentage per 100 cells);

D_{α} is absorbed dose from internal dose to RBM from alpha-radiation (Gy).

A coefficient, b = 0.0303, specifying the growth rate of aberrations per unit dose (%/cGy), showed high statistical reliability with increased absorbed dose ($p < 0.05$). The regression equation in general was significant by Fisher's ratio test ($p < 0.05$). The correlation coefficient ($r = 0.43$) was also significant ($p < 0.05$).

Fig. 1 illustrates Dependence (1) with the 95% CI and empirical points.

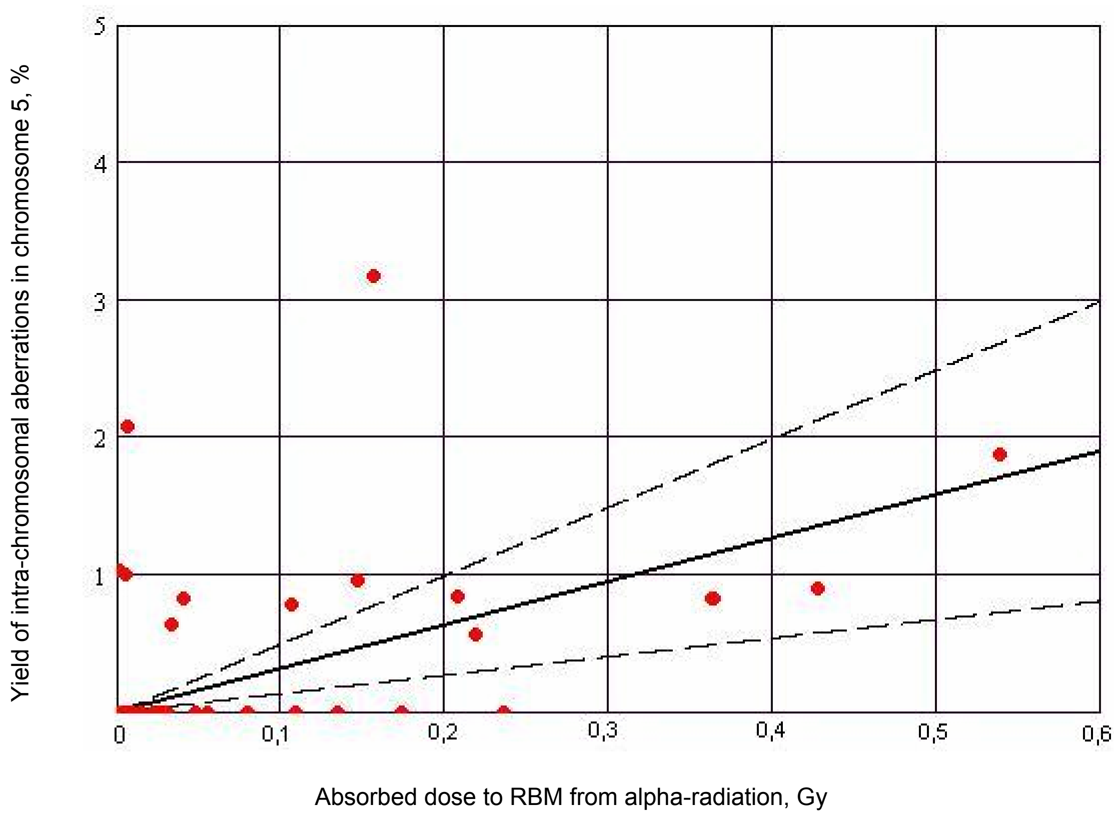

Fig. 1.1. Dependence of the yield of stable intra-chromosomal aberrations on absorbed dose to RBM from alpha-radiation.

Discussion

Our choice of chromosome 5 for this analysis was based on the findings by Mitchell et al., who had analyzed the yield of intra-chromosomal aberrations in three large chromosomes (1, 2, and 5) and demonstrated a higher intra-chromosomal yield per unit DNA length for chromosome 5 ($12.7 \pm 2.1 /10^5$ Mb) than either chromosome 1 ($5.6 \pm 1.4 /10^5$ Mb) or chromosome 2 ($4.5 \pm 2.1 /10^5$ Mb).

The derived linear dependence allows for the approximated assessment of absorbed dose from internal exposure based on the yield of stable intra-chromosomal aberrations. However, a small sample size (40 individuals) did not allow obtaining any statistically significant linear correlation coefficient. In the future, the derived dependence will be refined with the increased statistical data size.

Conclusions

The preformed analysis revealed a linear dependence of the yield of stable intra-chromosomal aberrations in the peripheral blood lymphocytes in human on absorbed dose to red bone marrow from internal exposure to alpha-radiation from ^{239}Pu in workers of the Mayak PA.

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Cytogenetic damage in cells exposed to ionizing radiation under conditions of a changing dose-rate

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Abstract
The current international paradigm on radiation effects is mainly based on the effects of dose with some consideration for the dose rate. No allowance has been made for the influence of a changing dose rate (second derivative of dose) and biological effects of exposing cells to changing dose-rates have never been analyzed. Here, we provide evidence that radiation effects on cells may depend on temporal changes in the dose-rate. In the experiments cells were moved to and from a source of radiation. The speed of movement, the time of irradiation and temperature during exposure were controlled. Here we report the results of first experiments with TK6 cells which were exposed to a constant, to an increasing and to a decreasing dose-rate. The average dose rate and the total dose were same for all samples. Micronuclei were scored as the endpoint. The results show that the level of cytogenetic damage was higher in cells exposed to a decreasing dose-rate as compared both to an increasing and a constant dose-rate. This finding may suggest that the second derivative of dose may influence radiation risk estimates and it is expected to trigger further studies on this issue.

The effect of dimethyl sulfoxide on radiation induced chromosome aberrations in cultured CHO cells

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Abstract

The radioprotector properties of dimethyl sulfoxide (DMSO) are rather attractive for fundamental and applied implication in radiation biology and medicine. The cultured CHO cell line is a suitable model that allows estimation cell survival and chromosome aberration yield in mammalian cells. The goal of the current study was to investigate the redaction of radiation induced chromosome aberration yield in mammalian cells treated by DMSO. The CHO-K1 wild type cells were used as in vitro experimental model. The cells were grown in DMEM media containing 10% bovine serum albumine and antibiotics in monolayer culture. The cell samples with or without DMSO were irradiated to 1.0, 2.0, 3.0 and 4.0 Gy at a dose rate of 0.3 Gy/min, using 60-Co source. The DMSO was added to samples five min before irradiation in concentration of 0.5 and 1 M. The synchronized (G-1 phase) and logarithmic growing (cycling) cells were investigated. After exposure, cell were washed and fresh 10% bovine serum albumine was added to culture. The cells were fixed at 5, 7 and 9 h after irradiation. Approximately, 600 metaphases for chromosome-type aberrations were scored from each dose sample. It was found that DMSO treatment results in a 40–50% reduction of chromosomal aberrant cells as compared with non-treated cells irradiated to the same dose. The radiation-induced frequency of chromosomal aberrations decreased 2-fold having clear dose dependence. This reduction in chromosome aberration frequency is thought to be the result of a free radical scavenging and, perhaps, a suppressing radiation induced bystander effects in CHO culture. The experimental study was accompanied by theoretical Monte Carlo computer simulation of indirect action to DNA of cells by free water radical including OH scavenging in DMSO. The match of experimental and theoretical results is discussed.

The relative biological effectiveness of accelerated 73 MeV protons determined by CHO cell assay

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Abstract

The purpose of the study was to estimate relative biological effectiveness (RBE) of 73 MeV protons using CHO-K1 cells assay. The cells were grown at 37 0Ñ in DMEM media containing 10% bovine serum albumine and antibiotics in suspension. The synchronized (G-1 phase) cells were investigated. The cell samples were irradiated with single doses: 0.5, 1.0, 2.0, 3.0, 4.0, 5.0 and 4.0 Gy, using pulsed proton beam. Gamma-radiation of 60-Co at a dose rate of 0.5 Gy/min was used as a reference source. After exposure cell were washed and fresh 10% bovine serum albumine was added to culture. The cells were fixed at 18, 19 and 21 h after irradiation. Approximately, 500 metaphases for chromosome-type aberrations were scored from each dose sample using traditional Giemsa stain and light microscope. The experiment was carried out twice in a period of a year. The RBE values were determined by comparing the doses needed to obtain the same yield of chromosomal aberrations as with the reference 60-Co gamma source. It was found that the yield of dicentrics per 100 cells follows linear-quadratic model: Y=1.56D+2.48D**2 and Y=6.88D+2.11D**2 (where D – absorbed dose in Gy) for 60-Co rays and 73 MeV proton irradiation, respectively. Therefore, protons were somewhat more biologically effective than 60-Co gamma-rays according to criterion of chromosomal aberrations. The proton RBE values obtained by regression fitting ranged 1.1-2.6 for absorbed doses from 6.0 to 0.5 Gy respectively. The data obtained at other proton acceleration centers agree with our results. Although the RBE value for protons was found to be rather small, it could be of radiotherapy importance. Accurate determinations of the RBE are necessary when extra therapeutic dose is inadmissible.

Effect of low dose gamma-radiation upon antioxidant parameters in heart and skeletal muscle of chick embryo

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Abstract
This study was performed to investigate the effect of low-dose gamma-irradiation upon activity of glutathione peroxidase (GSH-Px), superoxide dismutase (SOD), catalase (CAT) and level of glutathione (GSH) and lipid peroxidation (TBARS) in heart and skeletal muscle of chick embryo and newly hatched chicks. Fertilised chicken eggs were irradiated with the dose of 0.3 Gy gamma radiation (source ^{60}Co) on the 19th day of incubation. The antioxidant parameters were measured in breast muscle (*m.pectoralis superficialis*), thigh muscle (*m. biceps femoris*) and heart of chick embryos on 1, 3, 6, 24 and 72 h after egg irradiation. All parameters were measured spectrophotometrically. On the 1st hour after irradiation lipid peroxidation significantly decreased in all type of muscles. At the same time GSH level and CAT activity were significantly decreased in the breast and thigh muscle of chick embryo while SOD activity was significantly decreased in thigh muscle. On the other side on the 1st hour after irradiation GSH-Px activity was significantly increased in thigh muscle. On the 3rd, 6th and 24th hour of experiment there were no significantly changes in the investigated parameters between the groups. On the 72nd hour after irradiation results showed decreased activity of SOD in thigh muscle; decreased activity of CAT in the breast and heart muscle; decreased TBARS level in thigh muscle; increased level of GSH in the breast and thigh muscle; and increased GSH-Px activity in the breast muscle. The obtained results suggest that acute irradiation of chicken eggs on the 19th day of incubation with the dose of 0.3 Gy gamma radiation causes an oxidative stress in all types of muscles immediately after irradiation. However, the results of GSH-Px activity and GSH level on the one-day old chicks (72nd hour after irradiation) in skeletal muscle suggest probably a stimulating effect of low dose irradiation.

Influence of the ionizing radiation on the individual variability of the antioxicant status indices

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Abstract
The variability of the different parameters of the physicochemical regulatory system of the lipid peroxidation (LPO) in tissues of the laboratory mice (white outbreed and SHK) is studied in norm and under the action of radiation at the low doses and within the wide range of dose rate. It is shown that the different variability and the ability to normalization of the antioxidant activity and composition of lipids, intensity of LPO, the peroxide content of lipids and their antiperoxide activity in tissues are due to the antioxidant status of tissues and the radiosensitivity of mice. It was revealed that the radiation dose and dose rate result to the indefinite changes in the variability of the studied parameters. The data obtained allow us to conclude that the high variability of the LPO regulatory system parameters in the murine tissues plays the important role for the development of the biological consequences after the radiation action depending on the dose and dose rate.

Introduction
At present it is established that the lipid peroxidation (LPO) processes play the important role in the regulation of the cell metabolism [10]. The steady-state of the LPO in tissues of intact animals is provided by the physicochemical regulatory system, which is functioning not only at the membrane level [3], but the cellular and organ levels [13]. Among parameters of this regulatory system in the last case are the antioxidant (AOA) and antiperoxide (APA) activities and composition of lipids; the initial content of the peroxide in lipids, stipulated by their nonsaturation degree; the lipid oxidizability, which may be estimated as the ratio of sums of the more easily (EO) to the more poorly oxidizable (PO) fractions of phospholipids (∑EOPL/∑POPL); the LPO intensity and other. The biological consequences of the radiation action are due to the level of the AOA of the tissue lipids, the activities of the antioxidant defense enzymes and the LPO intensity in organs and tissues of animals and also their ability to repair and normalization after the radiation injury [4, 5, 12, 18].

The aim of this work is to reveal the variability of the different parameters of the antioxidant status in tissues of mice possessing different radiosensitivity both in norm and under the action of radiation at the low doses and within the wide range of dose rate.

Materials and methods

The experiments were carried out on 140 mice SHK (males, 17 – 23 g), 54 white outbreed mice (females, 23 – 30 g) and 20 mice SHK (females, 20 – 25 g). The γ-irradiation of mice SHK (males) was performed by a GUT-Co-400 apparatus (a source of γ -60Co). Mice SHK (males) were divided in 4 groups as follows:

1. The γ-irradiation at the dose of 15 cGy, the dose rate was 0,01 cGy/min (group 1; 30 mice); the irradiation interval was 25 h;
2. The γ-irradiation at the dose of 15 cGy, the dose rate was 0.25 cGy/min (group 2; 30 mice); the irradiation interval was 1 h;
3. The γ-irradiation at the dose of 15 cGy, the dose rate was 9 cGy/min (group 3; 30 mice); the irradiation interval is 100 s.
4. Intact mice from the same party (group 4, 20 mice) serve as biological control.

Mice were irradiated by storage of the food and the drinking-water (group 1) or only the drinking-water (group 2) in cells for animals per 10 mice. Mice of the group 3 were irradiated in the special container where the each mouse could easily drive.

The single acute X-irradiation of mice (females, 10 mice) at the dose of 16 cGy was performed by a RUT-200-20-3 apparatus in the same special container at the following conditions: current strength – 15 mA; filter 0.5 mm Cu; the dose rate is 44 cGy/min; the total irradiation interval 22 s.

Decapitation of mice after γ-irradiation was done within 1, 7 and 30 days and 1 month after X-irradiation. Decapitation of intact mice was performed simultaneously with the experimental group per 6 – 10 animals. As known, the lipid AOA level and the LPO intensity have the season variability [4, 9, 12]. It is need to note that experiments are performed on the mice SHK (males) in April and May, on the white outbreed mice in September and October, on the mice SHK (female) in October and November. All parameters of intact mice were determined for each animal. Mice in experimental groups were divided in subgroups per 1 – 4 animals. After decapitation of mice, blood was collected in test tubes treated by 5% solution of sodium citrate. The erythrocytes from the blood plasma were separated by centrifugation. The murine liver and brain was placed in ice-cooled weighting bottles immediately after decapitation.

Lipids were isolated by the method of Blay and Dyer in the Kates modification [8]. The AOA of lipids was determined by using the methyl oleate oxidation model [4] as the ratio of the differences in the induction period of methyl oleate oxidation in presence and absence of lipids to the concentration of the added lipid. Preliminary methyl oleate was purified by vacuum distillation. The lipid concentration in methyl oleate was 20 mg/ml. Oxidation was carried out in a temperature-controlled chamber at a constant temperature 37±0.1ºC by blowing air through at a rate providing oxidation in a kinetic range. In detail the lipid AOA analysis was described in [14]. The peroxide content in lipids was determined iodometrically. The antiperoxide activity (APA) of lipids, that is their ability to decompose peroxides, was evaluated as the ratio of the difference in the peroxide concentrations of methyl oleate in absence and presence of lipids to the amount of the added lipid [16].

The qualitative and quantitative composition of phospholipids (PL) was determined by thin-layer chromatography [2]. It was used type G or H silica gel (Sigma, USA), glass plates 9×12 cm and mixture of solvents chloroform : methanol : glacial acetic acid : water (25:15:4:2) as a mobile phase. All the solvents were of specially pure or chemically pure grade. The development of chromatograms was performed by iodine vapour. For the colour reaction for phosphorus we used ammonium molybdate and ascorbic acid (Serva, FRG) and perchloric acid of chemically pure grade. The amount of inorganic phosphorus was judged from the optical density of the solution at the wavelength 810 nm with Beckman Du-50 (Austria) or KFK-3 (Russia) instruments. The sterol content was determined spectrophotometrically by the method cited in [17]. In addition to quantitative analysis of the different fractions of PL and sterols the following parameters of lipid composition were also evaluated: the molar ratio of [sterols]/[PL]; the PL proportion in the total lipid composition (% PL); the ratio of the phosphatidyl choline to phosphatidyl ethanolamine content in PL (PC/PE) and the ratio of the sums of the more easily to the more poorly oxidizable fractions of PL (∑EOPL/∑POPL). The last value was calculated by the formula: (PI + PS + PE +CL + PA)/(LPC + SM + PC), where PI is phosphatidylinositol, PS is phosphatidylserine, CL is cardiolipin, PA is phosphatidic acid, LPC are lysoforms of PL, SM is sphingomielin.

The content of LPO products which have reacted with 2-thiobarbituric acid (TBA-reactive substances, TBA-RS) was determined spectrophotometrically by the method described in [1] with adding 10 µl of 0.01% 4-methyl-2,6-di*tert*.butylphenol (BHT) solution in ethanol. The diene conjugates and ketodiene amount in the liver and brain lipids were judged from the optical density of the lipid solutions in hexane (the range of concentration from 0.02 to 0.2 mg/ml) as their ratio at the wavelengths 230±2 nm and 270±2 nm to 205±3 nm, correspondingly [7] used UV-3101 PC (Japan) instrument. Protein was determined according [6].

The statistic treatment of the results were performed using standard method of the variational statistics. The variability of indices was evaluated as the ratio of mean square error of average mean to average mean for group expressed as a percentage.

Results

Earlier it is shown the existence of the reverse correlation between the lipid AOA value and the initial peroxide amount in the spleen lipids of the CBA mice [18]. The high ability to the peroxyl radical formation is also revealed by γ-irradiation of mice SHK at the dose of 15 cGy with the different dose rate for the spleen and erythrocyte lipids [11]. It is need to note that the lipids from these tissues characterize the prooxidant activity in the autooxidation reactions [16]. Besides, the highest heterogeneity of the peroxide content in the murine lipids within the experimental groups were namely established for the spleen and erythrocytes lipids of mice SHK after γ-irradiation at the low doses.

Usually the organ of the intact mice in accordance with their lipid AOA may be presented by following consequences: liver > brain > spleen [4, 12, 16]. The different initial level of these parameters is due to the differences in the sensitivity to the radiation injury another indices of the LPO regulatory system. Thus, the least changes of the LPO intensity which is usually evaluated by the TBA-reactive substances content in a complex biological system [7] were revealed in the liver lipids of mice SHK under their γ-irradiation at the dose of 15 cGy, especially at the 0.25 cGy/min dose rate [11]. Besides, the greatest changes of this parameter were obtained by the radiation of mice SHK at the low intensity dose rate (0.01 cGy/min) in the blood erythrocytes. The substantial dependence of the change scale on the dose rate was obtained for the lipid composition of liver, despite the fact that lipids of this organ characterize the most high level of AOA. The relative changes of generalized parameters in the liver lipids under γ-irradiation of mice with the different dose rate are presented in Fig. 1. It is seen the absence of the normalization practically for the all studied indices value within 1 month after irradiation. Besides, the more substantial changes are revealed for the molar [sterols]/[PL] ratio at the all dose rate. Although the PC/PE ratio is a more stable as compared with the another parameters there are the reliable changes in the relative content of both PC and PE in the PL liver of the irradiated mice especially at the dose rate of 0.01 cGy/min.

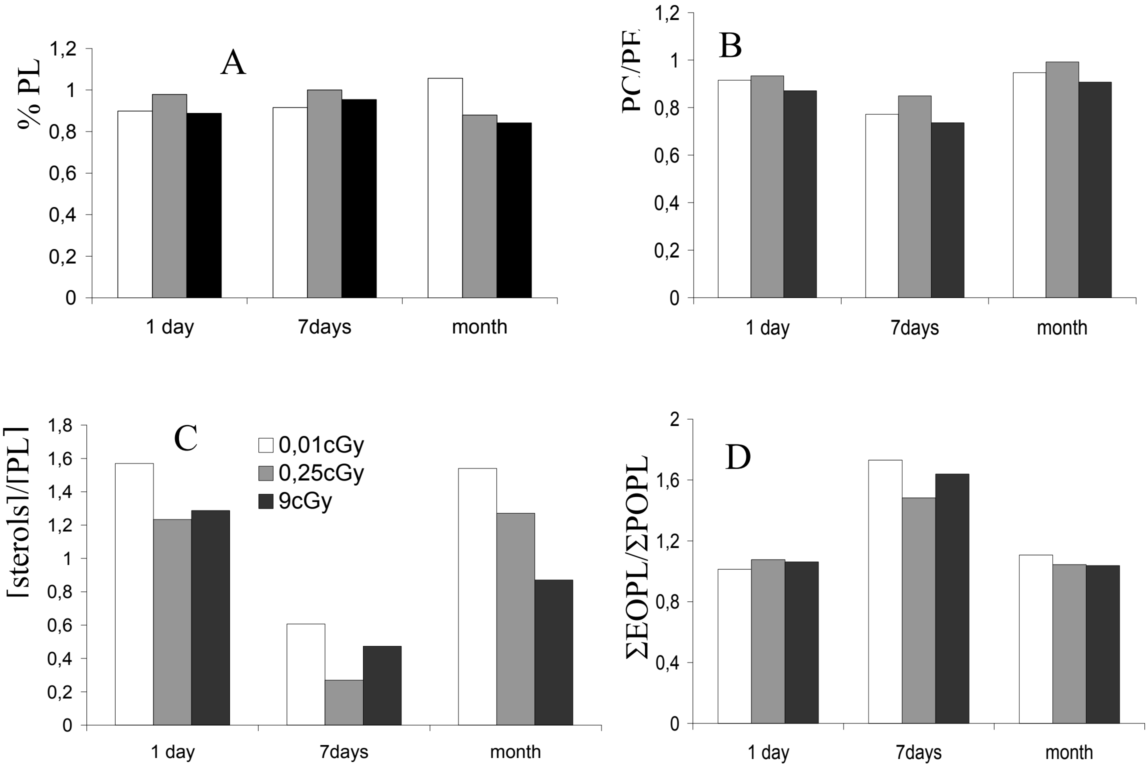

Fig. 1. Influence of the dose rate on the relative changes of the generalized parameters of the liver lipid composition under γ-irradiation of mice at the dose of 15 cGy.

Within 1 month after the acute X-radiation of mice SHK at the dose 16 cGy the reliable difference of the PC/PE and ΣEOPL/ΣPOPL ratio in the liver PL are obtained between the average values for the control and experimental groups. While the more high variability there is within the irradiated mice as compared with control group for the [sterols]/[PL] ratio, the another generalized parameters have a more limit of variance for the control mice. The data which are presented in Table are confirmed this conclusion.

Table. Limits of variations of the generalized parameters of the lipid composition in liver of mice SHK (females) within 1 month after X-radiation at the dose of 16 cGy.

Variant of experience	Parameters of the lipid composition			
	% PL	PC/PE	ΣEOPL/ΣPOPL	[sterols]/[PL]
Intact biological control X-irradiation at the dose of 16 cGy	10,7 – 60,8 21.3 – 83.6	1.19 – 2.28 2.01 – 2.39	0.30 – 1.06 0.43 – 0.53	0.54 – 1.09 0.58 – 0.53

It is necessary to note that the LPO intensity, the diene conjugate (DC) and ketodiene (KD) amount in the liver and brain lipids changes at mice of the different age (Fig. 2 and Fig. 3). As seen, the individual variability of these parameters differs both in the AOA of lipids and the analyzed parameter. So, the greatest variability for the TBA-reactive substance content in brain is revealed for the young mice (11.8 %), but in liver for the mice 18 weeks age (12.7 %). The DC content also has the highest individual variability in the brain lipids of the young mice (21.6 %) and in the liver lipids for mice 18 weeks age (35.5 %). The individual variability of the KD content decreases in the brain lipids from 13.6 % to 8.3 % at the aging of mice. Besides, this parameters has the most variability in liver lipids of the 18 week mice (29.0 %).

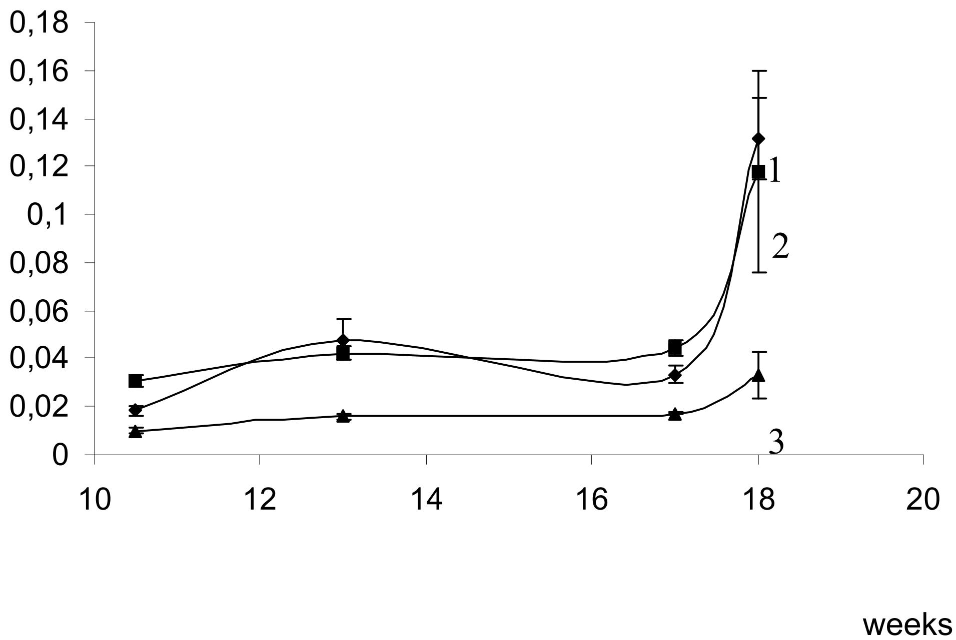

Fig. 2. The aging changes of the TBA-reactive substances (1, nmol/mg of protein), diene conjugate (2, a.u.) and ketodiene (3, a.u.) content in the liver lipids.

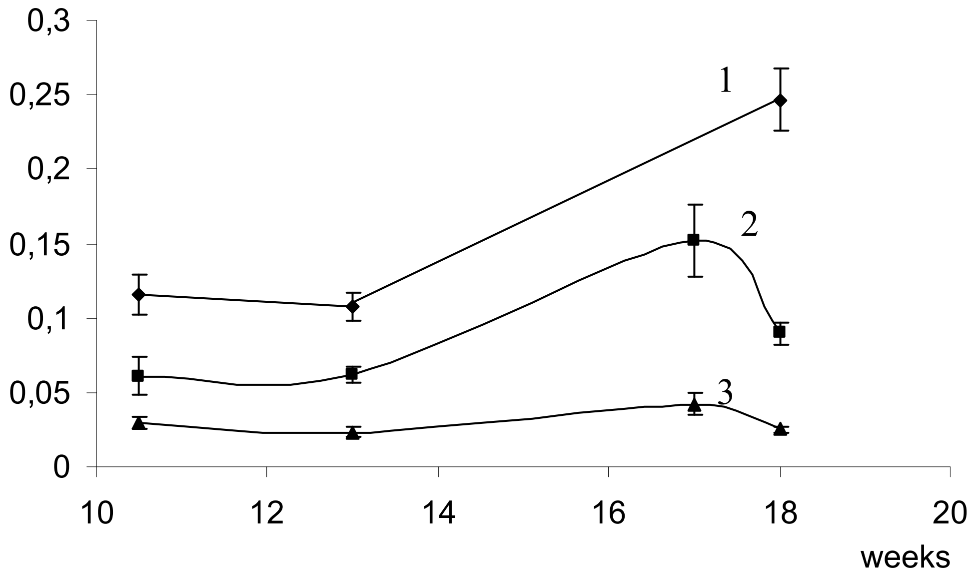

Fig. 3. The aging changes of the TBA-reactive substances (1, nmol/mg of protein), diene conjugate (2, a.u.) and ketodiene (3, a.u.) content in the brain lipids.

Conclusions

As early established, the oxidative processes in lipids have to influence on forming of the biological effects under the acute X-radiation of the laboratory animals which characterize the different radiosensitivity [18]. As also shown, the relationship of membrane and DNA damage with lipid peroxidation under the low doses of some technogenic factors [15]. Besides, the lipid AOA play the important role for the development of the biological consequences under the radiation action at the low doses [13]. The high variability of the physicochemical characteristics and the composition of lipids in norm has in influence on the forming of the biological effects in organs of the irradiated mice. The scale of changes are due to the radiation dose and dose rate and the antioxidant status of tissues. The data obtained allow us to conclude that the individual variability of the antioxidant activity and composition of lipids, intensity of LPO, the peroxide content of lipids and their antiperoxide activity in tissues takes into account for the development of the biological consequences after the radiation action on organism.

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STORE – Sustaining access to Tissues and data frOm Radiobiological Experiments

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Abstract

Sharing of data and biomaterials from publicly-funded experimental radiation science will yield substantial scientific rewards through re-analysis and new investigations. To that end, the STORE Consortium will create a platform for the storage and dissemination of both data and biological materials from past, present and future radiobiological research, a so-called Data Warehouse. STORE will provide a single online portal to radiobiological information that is presently distributed over scientific centres worldwide, and it will provide the necessary standard operating procedures (SOP) for the evaluation of archived tissue usability. The final goal of the project will be to propose viable financial models for the long-term sustainability of both material and data emerging from radiobiological research.

Fig. 1. Logo of STORE (http://www.fp7store.eu).

The strategy to achieve these goals is multi-level: 1) to provide a "one-stop-shop" portal integrating international databases and other repositories currently active, such that the user can find material and data held remotely; 2) to archive primary (raw) data or pointers to data in public databases, from radiobiological experiments and studies, while this resource will be open to individual investigators and to funding agencies as a potential central repository for data sharing; 3) to physically archive threatened material resources which are considered to be a valuable resource to the Community, and whose state of preservation is consistent with STORE benchmarks; 4) to provide a single point of access to the integrated biomaterial resources through standardised request procedures.

The provision of a central portal is expected to help in the dissemination of awareness of the existence of these resources, many of which are, anecdotally, underused because their availability and existence is unknown. The generation of benchmarks for sample preservation and usability by preparation of SOPs will also help to disseminate formal standards by which the usefulness of archive material of this type can be assessed.

Introduction

Since the late 1950s, valuable data have been collected on the effects of experimental, accidental and medical irradiation of humans and animals. Sharing and re-use of data and materials from publicly funded experimental radiation science and clinical research adds huge value to the original investment and can yield substantial scientific rewards through re-analysis and investigation, but such activity requires an infrastructural resource in order to be realised.

High-throughput screening techniques, so called "omics", have been developing in recent years with a breathtaking speed. The question arises which modern techniques are suitable to be used with the FFPE tissue (Tapio et al 2008a). Finding novel methods to re-evaluate the archive material will provide us with a vast amount of research data, the concept of data banking becoming increasingly important. Combined with information on individual exposure, disease diagnostics, and available biomaterial this data become extremely valuable.

The principal aim of data sharing is to maximise the return from the scientific endeavour and make the best use of research and funding. Although opportunities and developments in technology allow rapid, efficient and independent dissemination of vast amounts of information and despite the existence of guidelines on access to data, good practice is not widespread (Schofield et al). Open access to quality-controlled data in interoperable formats would allow better use of the original investment in collecting the data. Re-analysis of experimental results could be used to validate and adjust previous results and, in an interdisciplinary setting, also open up new research avenues well beyond the initial context in which the data were collected. Consequently, current hypothesis could be tested and new hypothesis could be advanced.

STORE, a 3-year project running from 2009 to 2012 and supported by the European Commission, currently works to realise the vision of an open access to data gained from radiobiological studies on animals. The main challenge here is finding financial support for maintenance and development of data resources to best serve the scientific community (Chandras et al).

STORE wants to a) improve the quality of and accessibility to radiobiological data and to promote the awareness of the importance of data sharing and b) create a platform for the storage and dissemination of both data and biological materials from past, present and future radiobiological research. The STORE platform will consist of a combined "Data Warehouse" and physical repository that will enable the sharing of experimental data sets and materials. STORE will provide a single portal to radiobiological information that is presently distributed over scientific centres worldwide. STORE will provide the necessary Standard Operating Procedures (SOPs) for the evaluation of archived tissue usability. The project will put forward feasible financing models for long-term support of data and bio resources in the radiobiology.

Methods

Within STORE two databases will be developed and maintained. The primary database (Data Warehouse) will form a public repository for radiobiological data and will provide links to other relevant databases. The second one will be a smaller bioresource database containing information about the biological samples in the STORE. Both data bases will be linked to each other.

Data Warehouse

The Data Warehouse will contain a) primary (raw) datasets from legacy experiments and new investigations, input by users or curators, b) summary information and links to datasets from individual projects already submitted to other public databases and c) a portal to other existing databases, such as ERA, the Chernobyl Tissue Bank, and the Northwestern Janus database, mediated through the generation of a BIOMART. BIOMART is an open-source application which takes data tables from distant databases and transforms them into a standard structure to allow queries of the same form to be made across all the databases.

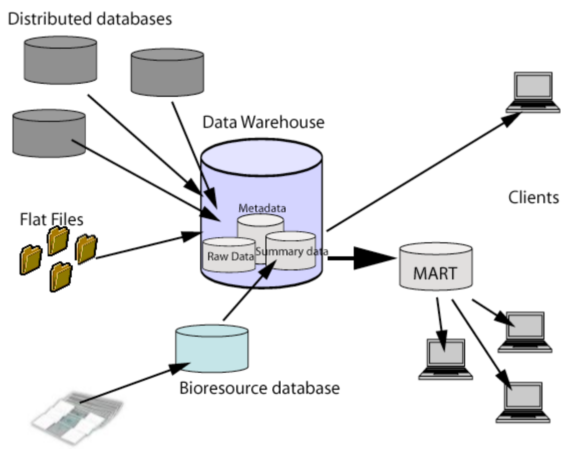

Fig. 2. Scheme of the database.

The Data Warehouse will be a database with the primary data object being the "Study". Key metadata, such as SPECIES, RADIATION QUALITY, ENDPOINT, will be attached to the data and study. Where appropriate, the structure of primary data will be transformed into a standardised form. The data organisation and metadata will be based on the growing group of accepted data structure standards and ontologies for functional genomics and biological and biomedical investigations to enhance interoperability with other databases.

Bioresource database

A second database and curation facility will be established to deal with the physical archiving issues. Current good practice guidelines for biological resource databases have already been described by several agencies, most recently by the NCI (http://biospecimens.cancer.gov/practices/) and within the caBIG project in the USA (http://biospecimens.cancer.gov/practices/cabig_tools/). Such recommendations are a gold standard as they are designed for human clinical material; the degree to which they are implemented in STORE will depend on the results of the investigations on the practice and regulatory issues surrounding tissue retention and international transfer. We suggest that the system developed here will be sample-centric, each sample possessing its own unique identifier corresponding to the individual animal and the study. Interactions with the Data Warehouse to source detailed experimental information on the samples will be accomplished through the individual codes. Where samples have been processed, for example into RNA and DNA, the database will support quality control, aliquoting, bar-coded containers and processes. Tracing and internal audit will be possible so as to identify end users and obtain feedback, and to comply with regulatory requirements.

Regulatory Issues

The regulations concerning archiving and transfer of animal and human tissue across international borders will be a major focus of the project. As we expect that biomaterial will be transported across European but also international boarders during the lifetime of the project, it is one of the main goals to facilitate exchange of samples between remote archives offering researchers advice and support. Based on international regulations on transfer of human and animal biomedical data and biological material a respective set of recommendations will be developed. A particular question important for the radiobiology community is the transport of radioactive material and will be given special attention (see below). To facilitate the comparison of international regulations links to relevant institutions in the US and in Japan have been made.

Sustainability and accessibility of the biological archives

STORE wants to develop operational parameters for a sustainable and accessible radiobiological tissue archive. For that purpose, questions concerning the current status and ownership of biological samples and their documentations have already been investigated within an FP6 project Promotion and Update of the European Radiobiological Archives (ERA-PRO; Tapio et al 2008b) A further task is to prepare a concept for sustainability and accessibility to the biological archive material and documentation in consultation with other international archives. Finally, evaluation of the radiological issues involved in the storage, transport and use of the archive materials is of great importance. Specifically, STORE will give recommendations on how to handle material arising from experiments using long-lived internal emitters. In this case, some radioactivity coming from radionuclides in the tissue has to be taken into consideration when handling the sample.

Establishment of Standard Operation Procedures

The biological material in physical repositories has to be used with greatest care to avoid damage due to its irreplaceable nature.

It has to be borne in mind that most techniques available at the moment were optimised for fresh/frozen tissue. FFPE tissue has recently gained considerable interest as an alternative to fresh/frozen tissue in retrospective biomarker discovery. However, macromolecules such as DNA, RNA and proteins undergo degradation and in the case of proteins also cross-linking during the fixation and aging process, making the applicability of conventional analysis problematic. There is an obvious need for methodological optimisation for FFPE material. Furthermore, as different storing times and conditions and the length of fixation influence the quality of various macromolecules in a different manner, these parameters will be tested using a common tissue model.

Taken together, STORE aims to establish Standard Operating Procedures (SOP) for evaluating quality of radiobiological archive tissue and defining test systems describing the usefulness of such material. Given that successful analysis of biological material isolated from archived tissue is feasible, the procedures performing best in terms of different endpoints compared to the fresh/frozen control tissue, will be defined as SOPs. At the end of the project, the SOPs will be validated by a STORE member not involved in their development.

Discussion

Sharing research data avoids its recollection by unnecessarily repeating animal experiments and enables their reanalysis. The vast amounts of data generated by previous and modern research will be valuable long after the experiments have been finished. It is less clear how data and databases will be maintained in order to guarantee the future access to these resources. The maintenance of databases is particularly at risk of falling victim to the constraints of traditional grant-based funding. We suggest that entirely different criteria have to be applied to judge the value of a database. For example, is it a unique and essential community resource? How are the data curated and validated? How comprehensive, detailed and interconnected are the results?

In many cases, researchers collect ancillary data that extends the primary research outputs: survey data, lab notebooks, spreadsheets, and annotations just to name a few. Much of this ancillary data (and some of the primary data as well) is not currently digitised, e.g. lab notebooks or handwritten specimen labels. Some of this additional data is maintained; much of it is lost, either because it is difficult or impossible to locate, or because it is simply deleted or destroyed. Research data (from old and newly performed experiments) in a digitalised form can be easily transported, worked on with new tools, merged with other data, categorised in new ways and stored in vast volumes. Research data are valuable long-term resources and the way to ensure their potential value is *to share* and *make them accessible*.

There is a need to collect, store, curate and distribute radiobiological data in a standardised form to ensure interoperability within a central database and databases linked to it. Action is urgently required for funding of data and biomaterial repositories on a stable, long-term basis as uncertainty about the future funding and hosting of these resources will be a disadvantage for the whole radiobiological research community.

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Malignant blood diseases and tumours in acute radiation sickness survivors following the Chernobyl accident

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Abstract
In the result of Chernobyl accident acute radiation sickness (ARS) developed in 134 patients, including 28 ones who had died in 1986 due to severe radiation damages. At the end of 1986 we started follow-up study on oncological pathology development in 91 ARS survivors (doses up to 7.1 Gy) and 99 patients with doses below 1 Gy. All patients were residents of Ukraine. During 24 post-accidental years malignant diseases of hematopoiesis were observed in 5 patients and solid tumours in 15 ones. Oncohematological pathology consisted of hypoplasia of hematopoiesis (1 case; 1986), acute myelomonoblastic leukaemia (1 case; 1998) and myelodisplastic syndrome (3 cases; 1993, 1995, 1996). All diseases brought patients to death. Patients' mean age was 46.3±7.5 at the moment of irradiation, 54.0±8.1 on disease onset and 55.5±6.8 years at person's death. Leukaemia and myelodisplastic syndromes arose only in ARS survivors. Their latent period was 9.6±2.1 years. The first malignant tumour, sarcoma of hip soft tissues, was diagnosed in 1992. This patient died in 1993. Before 2009 it was revealed as far back as 14 malignant neoplasm of different localization: cancers of colon (3 cases; 1997, 1999, 2001), stomach (2 cases; both in 2004), thyroid gland (2 cases; both in 2000), throat (1 case; 2000), kidney (1 case; 2000), lung (1 case; 2001), prostate gland (1 case 2001), basal cell carcinoma of head (1 case; 2006), urinary bladder (1 case; 2008), and neurinoma of lower jaw that had a malignant transformation (2003). Amongst persons suffered from cancer, 7 patients were successfully treated but the rest ones died. Totally, malignant tumours were diagnosed in 6 ARS survivors (6.6%) and 9 persons from group of comparison (9.1%). At the moment of death from malignant tumors the patients' age was 66.2±9.8 years. The period from irradiation to neoplasm development averaged 15.9±3.1 years.

Introduction
Malignant tumours and leukaemia are considered the stochastic effects of radiation. The majority of radiogenic tumours have no dose threshold (Beebe 1984; WHO 2005) and a risk of cancer development increases with radiation dose. Studying consequences of the Chernobyl accident some researches came to the conclusion of oncological morbidity

rate increase in victims during post-accidental period (Demina 2002; Bespalchuk et al. 2003; Ivanov et al. 2001; Kirillov et al. 2001) and experimental animals as well, who was keeping in Chernobyl zone (Serkiz 1991). Other authors have the opinion that the accident at the Chernobyl nuclear power plant (CNPP) didn't result in essential growth of cancer and mortality from malignant tumours in exposed persons in comparison with the rest population (Buldakov 2002; Ganul et al. 1991; Ivanov et al. 2001; Moysich et al. 2002; Tukov et al. 1998).

Such contradictory results of numerous investigations possibly connected with the fact that CNPP clean-up staff are characterised by wide radiation dose range. The opinion is existed that doses 0.1 Sv and higher are able to cause statistically significant excess risk of cancer incidence (Beebe 1984; Filyushkin et al. 2000; UNSCEAR 2000). This situation makes the study of oncological and oncohematological pathology in persons exposed to irradiation in doses upper 0.1 Sv, especially ARS survivors, are very actual.

Material and methods

After the Chernobyl accident 91 ARS survivors were under follow-up in the hospital of Research Centre for Radiation Medicine (RCRM). From this number 38 patients survived ARS grade 1 (mild), 41 ARS grade 2 (moderate) and 12 ARS grade 3 (severe). As a control group 99 patients with whole-body radiation doses lower 1 Gy (ARS 0 group) were examining during entire post-accidental period (table 1). Patients' groups didn't differ significantly by age

Table 1. Age-specific and dosimetric characteristic of the examined patients

Variables	ARS 0	ARS grade 1	ARS grade 2	ARS grade 3
Number of persons	99	38	41	12
Gender: M / F	89 / 10	36 / 2	40 / 1	12 / 0
Age at exposure, years mean ± SD min - max. 95% confidence interval	 35.9±10.3 18.4-60.3 33.8-38.2	 33.2±8.2 17.6-56.3 30.5-35.8	 40.9±16.5 17.9-79.3 34.1-42.9	 36.6±12.5 20.4-72.6 30.4-51.3
Persons with known dose	15	30	37	11
Dose of irradiation, Gy mean ± SD min – max 95% confidence interval	 0.4±0.3 0.1-1.0 0.2-0.5	 1.0±0.6 0.1-3.3 0.8-1.3	 2.4±0.9 0.5-4.9 2.1-2.7	 4.5±1.4 2.9-7.1 3.6-5.5

All patients had a hospital examination and treatment at the average once per 2 years.

The check-up protocols included full blood count, biochemical tests (basic metabolic panel, chemical pathology and liver function tests), immunological profile (humoral and cellular indicators), thyroid tests (ultrasound and TSH, FT_4), urinalysis, full physical examination, and various instrumental tests as necessary (electrocardiogram (ECG), electroencephalogram, ophthalmological tests, gastro- or colonoscopy, chest x-ray, functional lung test etc. When necessary, patients were consulted by specialists (ophthalmologist, endocrinologist, pulmonologist, cardiologist, gastroenterologist, haematologist, dermatologist, neuropathologist, psychiatrist, urologist or gynaecologist). Further, more complex diagnostic methods were applied as necessary, including CT scans, biopsy, cardio-stress-ECG, serological tests for specific causes, e.g. hepatitis A, B or C viruses, bone marrow tests etc.

Results

The first case of malignant hypoplasia of haematopoiesis was revealed in August 1986 in a patient, who worked several days in the CNPP zone in May and July 1986. His haematological status is characterised by essential leukocytopenia and slight anaemia. The onset of oncohematologic pathology was mistakably estimated as ARS grade 1 manifestation. However the following disease course dispelled any doubts against blood system pathology. The bone marrow aspiration confirmed the oncohematological diagnosis.

Subsequently two ARS grade 3 and one ARS grade 1 survivors developed myelodisplastic syndrome, which had the lethal outcome for all patients (table 2). One more patient, who had survived ARS grade 2, fell ill with acute myelomonoblastic leukaemia after 11.8 years passed from irradiation and died in spite of treatment.

Table 2. Cases of malignant blood disease in ARS survivors and control group

No	Diagnosis	ARS grade	First diagnosed	Age at (years)		Outcome
				diagnosis	death	
1	Hypoplasia of hemopoiesis	0	Aug 1986	48.6	49.3	Died on Apr 1987
2	Myelodisplastic syndrome	1	Sept 1996	45.0	50.7	Died on May 2002
3	Myelodisplastic syndrome	3	Mar 1993	52.1	52.2	Died on Apr 1993
4	Myelodisplastic syndrome	3	Oct 1995	64.6	64.7	Died on Dec 1995
5	Acute myelomonoblastic leukaemia	2	Feb 1998	59.9	60.7	Died on Dec 1998

All patients with malignant blood diseases were male. Their mean age at the exposure was 46.3±7.5 years, at the diagnosis 54.0±8.1 and at death 55.5±6.8 years. The latent period from exposure to first signs of disease was 9.6±2.1 years. Today the relative number of patients, who suffered from oncohematological pathology and died, is 2.9% of 190 persons under follow-up or 4.4% of ARS survivors.

The first case of malignant tumour was revealed in patients of ARS 0 group in 1992 (table 3). During examination of 34 years old man there was found subcutaneous, solid and tender formation on the right hip with dimensions as a pigeon egg. The patient was sent in a city oncological hospital but he refused from examination without clear explanations. When he was admitted in the RCRM hospital next year, it was found several solid subcutaneous nodes. The patients refused from oncological examination again and at the end of 1993 he died. Autopsy proved the supposed diagnosis sarcoma of hip soft tissues.

The second case of oncological pathology as cancer of sigmoid colon was diagnosed in 1997. A patient (ARS grade 1 survivor) only complained on bloody discharges from anus during defecation. The adenocarcinoma in situ was diagnosed after colonoscopy and proved by histological examination.

Table 3. Cases of oncological pathology in ARS survivors and control group

No	Diagnosis	ARS grade	First diagnosed	Age at (years)		Outcome
				diagnosis	death	
1	Sarcoma of hip soft tissues	0	Apr 1992	34.1	34.8	Died in 1993
2	Cancer of colon	1	Mar 1997	54.5		Operated on Jun 1997
3	Leiomyosarcoma of shin Cancer of colon	0	Aug 1998 Jun 1999	72.5 73.4		Operated on Sep 1998 Operated on Jun 1999
4	Cancer of larynx	0	Jul 2000	59.3	59.9	Died on Feb 2001
5	Cancer of thyroid gland	2	Nov 2000	37.6		Operated on Nov 2000
6	Cancer of thyroid gland	2	Dec 2000	42.9		Operated on Jan 2001
7	Cancer of kidney	0	Dec 2000	44.4		Operated on Jan 2001
8	Cancer of colon	0	Nov 2001	63.6	67.5	Died on Oct 2005
9	Cancer of prostate	0	May 2001	71.7	73.3	Died in 2003
10	Cancer of lung	0	Feb 2001	46.1		Operated on Jun 2003
11	Neurinoma of lower jaw	2	Oct 2003	52.7	53.9	Died on Dec 2004
12	Cancer of stomach	0	May 2004	77.5	78.1	Died on Jan 2005
13	Cancer of stomach	0	Jul 2004	65.7	65.9	Died on Sep 2004
14	Basal cell carcinoma of head	1	Nov 2006	51.9		Irradiation
15	Cancer of urinary bladder	2	Jun 2008	66.8	66.9	Died on Aug 2008

It is necessary to note that all subsequent cases of solid tumour either had no clinical symptoms, being a result of causal findings during the routine examination (endoscopy, ultrasound scanning), or coursed with minimum non-specific complaints (so called "the syndrome of minor signs"). In the majority of cases the tumours were revealed owning to physicians having oncologic alarm. However the events took place when patients ignored doctor's opinion and refused informative diagnostic procedure and treatment as well. An operable stomach cancer with clear clinical symptoms and evident endoscopic picture was diagnosed in ARS 0 patient. The patient refused a surgical treatment and died over 3 months.

During 18 years after the first sarcoma diagnosis 14 cancers of different location were revealed: 3 cancers of colon, 2 ones of stomach, 2 thyroid gland, and a cancer of larynx, lung, kidney, urinary bladder, prostate, as well as basal cells carcinoma of head and neurinoma of lower jaw that was undergone malignant transformation. In a patient of ARS 0 group cancer of colon followed leiomyosarcoma of left shin. He was operated with subsequent fractionated X-ray irradiation of shin in total dose of 52 Gy.

Two patients developed cancer of stomach and larynx that had rapid course. The period from diagnosis to lethal outcome was 3 and 6 months, respectively. The first patient was undergone endoscopic procedure in Feb 2004 without any changes of stomach mucous membrane but in July 2004 he was admitted in RCRM hospital with clinical and endoscopic signs of stomach cancer. Cancer of larynx $T_3N_1M_0$ was diagnosed in the second patient during routine follow-up but a year before the patient had no any complaints. Surgical treatment didn't stop metastatic disease with following lethal outcome.

All malignant tumours were diagnosed in male patients. Their mean age at the moment of exposure was 40.9±12.1 years, at the tumour diagnosis 57.9±17.1 and at the moment of death 66.2±9.8 years. The tumours latent period was 15.9±3.9 years. The overall number of patients with oncological pathology is 7.9%, amongst ARS survivors 7.6%, and in ARS 0 group 9.1%.

Statistical analysis showed that tumour incidence and their frequency rate didn't depend on ARS grade and ARS presence or absence (table 4).

Table 2. Relation between cancer incidence and radiation factor

Radiation factor	Spearman's rank correlation		χ2-test	
	r	p	F	p
ARS grade	-0.07	0.36	2.69	0.44
Presence or absence of ARS	-0.083	0.28	1.2	0.27

Discussion

As it was showed above the malignant blood diseases in ARS survivors preceded solid tumours and are not characterised by wide diversity of nosological form (3 myelodisplastic syndromes, acute leukaemia and hypoplasia of haematopoiesis) whereas in A-bombing victims of Hiroshima and Nagasaki from 1950 to 1987 there was diagnosed various form of acute and chronic leukaemia, Hodgkin's and non-Hodgkin's lymphomas, and multiple myelomas (Mabuchi et al. 1995; Preston et al. 1994). The peak of leukaemia incidence fell on 5^{th}-8^{th} year after irradiation (Mabuchi et al. 1995).

Analysing leukaemia incidence in hibakusha and persons, who was exposed to medical irradiation, E. Hall (1989) concluded that leukaemia latent period is 5 year. Other authors considered it shorter, 2-3 years (Rudnev 1992), or longer, 7-10 years (Nenot et al. 1995). The last term is better corresponded our data: all 4 cases of oncohematological pathology, except the first, appeared 7 years later the exposure. Hypoplasia of hematopoiesis in the patient No. 1 (table 2) developed in improbably short term after the initial contact with ionizing radiation. It doesn't enable to consider this case caused by radiation. Most likely the disease set in early but primary clinical symptoms appeared during his work in the CNPP zone.

From 1992 to 2000 myelodisplastic syndrome developed in 112 from 240800 CNPP clean-up workers, including 3 cases in ARS survivors. According to I. Sekine (2003), amongst 93700 A-bomb victims during the 50-year period only 13 cases of myelodisplastic syndrome developed that is absolutely and relatively less than in Chernobyl victims. Both high frequency rate and the latent period of 6-14 years enable to suspect radiation in this disease incidence. To prove this suggestion the additional study has to be set up.

Cancers that developed in CNPP clean-up staff during the first years after the accident, according A. Romanenko et al. (2000), might be the result of existing cancer promotion but not of the new cancer induction. The peak of radioinduced cancer should expect in 15-25 years after irradiation. The first cancer was revealed on 11th year after the exposure in ARS grade 1 survivor. This fact as well as following cases of malignant tumours onset proves the data that minimal latent period for solid tumours has to be 10 years and more (Kato et al. 1995). After this their incidence will increase constantly.

Cancers is a non-threshold radiation effect. After the whole-body short-term radiation exposure with low linear-energy transfer in dose 0.1 Sv the risk of cancer mortality increases on 0.5-1.4% from basal level (Fabrikant 1981) but if the dose 1 Sv it elevates to 9% in men and 13% in women (UNSCEAR 2000). Basing on this data it should expect higher cancer incidence in ARS survivors in comparison with persons exposed to radiation in doses lower 1 Gy. However our study showed that frequency of cancer was higher in control group (9.1%) than in ARS survivors (7.6%) with insignificant difference. There was not found significant difference of cancer incidence between patients with various ARS grades.

Amongst all oncological cases in ARS survivors and persons exposed in doses lower 1 Gy it is impossible to discriminate what tumours are caused by irradiation or developed spontaneously. These cohorts are too small for epidemiological investigations. The data of Japanese survivors showed that amongst 86572 individuals in the Life Span Study cohort there were 7578 deaths from solid tumours during 1950-1990. Of those cancer death, 334 (4.4%) could be attributed to radiation exposure. However this data cannot be extrapolated on relatively small group of ARS survivors or less irradiated persons.

ARS survivors and control group patients more frequently fell ill with cancer of colon (3 cases), stomach (2 cases) and thyroid gland (2 cases). According to results of hibakusha follow-up (1958-1987) their overall tumours included the first two types of cancer for whom the significant excess risks of morbidity and mortality was determined (Ron et al. 1994). The cancer of lung and urinary bladder, which were diagnosed in a patient of control group and ARS grade 2 survivor respectively, also belong to the tumours with the high mortality risk. There was revealed the significant increment of nonmelanoma skin cancer in the incidence data, but not in the mortality series. No significant radiation effect was seen for cancers of the pharynx, rectum, gallbladder, pancreas, nose, larynx, uterus, prostate or kidney in either series.

Sarcomas are the malignant tumours derived from mesenchyme whereas cancers descended from endothelium. No one investigation dedicated to A-bombing Japanese survivors contained data of sarcomas morbidity increase connected with irradiation. Our investigation revealed only 2 sarcoma cases in patients without ARS in anamnesis. This cohort size is not enough for statistical analysis.

WHO concluded (2005, 2008) that several last decades various organs cancer ranks as second having ahead cardiovascular diseases as mortality reasons in European countries including Ukraine. During 24 post-accidental years amongst restricted group of ARS survivors and persons exposed in doses lower 1 Gy 13 persons died due to oncological and oncohematological pathology, 12 ones of cardiovascular diseases and 11 patients of other reasons including traumas and accidents.

The patients' mean age was 56.4±12.4 years when malignant blood diseases and solid tumours were diagnosed. Thus 4 patients belonged to persons of young and mature age (20-44 years), 8 ones to middle age (45-59 years) and 8 patients to elderly and old age (60 and more years). The life span of 13 patients, who died from oncohematological pathology and tumours, was 59.8±11.5 years and didn't reach the average population level that according to WHO data (WHO 2005, 2008) was 62.3 year for male in Ukraine in 2001-2003 but decreased to 61.6 years in 2005. Therefore more than a half of the patients, who died, didn't live till very low average population life span in Ukraine (e.g. in Finland this index was 75.8 years in 2005). This fact enables to consider that the oncohematologic situation in ARS survivors and control group patients is disturbed. The way out of this is the early diagnostic and a radical cure. The localization of neoplasm (blood, digestive tract, thyroid gland, larynx, kidney, lung, prostate, skin) in followed up patients enables to reveal it on the early stage of development by routine and non-invasive diagnostic methods as blood analysis, x-ray examination, ultrasound, endoscopy, examining by otolaryngologist, gynaecologist, urologist, endocrinologist, pulmonologist, haematologist etc.

The second problem, which associates with successful diagnostic, is early patient's visit to physician for medical care. Due to low sanitary and general culture some patients didn't go to physician even if on the background of old complaints a new ones appeared and health changed for the worse. Some patients visited "folk healer" instead of physician or ignored medical recommendations. Therefore the obligatory annual examination of ARS survivors and other Chernobyl victims has to be combined with community health and medical staff oncologic alarm.

Our investigations had showed that malignant tumours appeared in the Chernobyl victims on the background of chronic non-specific diseases of internal organ systems, which didn't threaten patients' life themselves but as any pathological process increased the probability of cell mutagenesis. So a treatment of these chronic diseases is the one more way to decrease the risk of malignant tumours incidence.

Conclusions

During 24 years after the Chernobyl accident malignant blood diseases and tumours ranked as first by mortality reasons in ARS survivors and other exposed persons in doses lower 1Gy. They left behind even cardiovascular diseases. More than 50% of fallen ill with oncological pathology and died of them didn't reach the level of average population life span in Ukraine, which is 20% less the analogous index in economically developed European countries.

Effective treatment of patients depends on both early diagnostic and patients' readiness to follow doctors' recommendations.

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Radiation-epidemiological estimation of thyroid pathology risk

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Abstract

The main source of irradiation of the population after the Chernobyl accident was ^{131}I and its other short-lived isotopes that effected practically the whole population of Belarus. More than 30 % of children aged up to 2 years old received doses over 1 Gy.

The aim of study is the retrospective analysis of clinical data depending on thyroid exposure level among people aged 0–3 at the moment of the Chernobyl accident.

Out of people observed in State registry, we formed 2 groups which are annually examined on thyroid pathology.

1. Basic group including 1004 residents of Gomel oblast aged 0–3 at the moment of the Chernobyl accident who were exposed to short-lived iodine radioisotopes.
2. Control group including 2020 persons born in 1987–1988 in Gomel region and who are observed in State Registry.

As one would expect, the highest estimates of the relative risk were received on thyroid cancer. Even at rather low absorbed thyroid doses, the attributive fraction was more than 94%. In group with high doses, practically all the cases were radiation-induced (AF = 98,5%).

At the same time, the analysis results showed that radiation component was the predominant one at realization of practically all the spectrum of thyroid pathology among subjects irradiated in early childhood. Evident growth of OR was observed in patients with all nodular forms of goiter with increase of thyroid dose. In subjects with thyroid dose above 1 Gy, the attributive fraction of different forms of nodular pathology varied from 60 to 98%.

Statistically significant estimates of relative risk were calculated on all forms of thyroid nodular pathology.

Relative risk of any goiter nodular form in persons affected in early childhood was 3,7 (AF= 71,1%).

These results suggest that considerable part (up to 71%) of goiter nodular form diseases among residents of Gomel region, affected in early childhood, can be referred to thyroid exposure.

Acute myocardial infarction in the cohort of Mayak PA nuclear workers

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Abstract
Incidence of and mortality from acute myocardial infarction (AMI) have been studied in a cohort of 12210 workers first employed at one of the main plants of the Mayak nuclear facility during 1948–1958 and followed up to 31 December 2000. Information on external gamma doses was available for virtually all of these workers (99.9%); the mean (± standard deviation, SD) total external gamma dose was 0.91±0.95 Gy (99% percentile 3.9 Gy) for men and 0.65±0.75 Gy (99% percentile 2.99 Gy) for women. In contrast, plutonium body burden was measured only for 30% of workers; amongst those monitored, the mean (± SD) absorbed cumulative liver dose was 0.40±1.15 Gy (99% percentile 5.88 Gy) for men and 0.81±4.60 Gy (99% percentile 15.95 Gy) for women. 683 disease cases and 338 deaths from AMI were identified in the study cohort. Having adjusted for non-radiation factors, AMI mortality data showed a statistically significantly increasing trend with external dose, excess relative risk per Gy (ERR/Gy) was equal to 0.265 (95% CI 0.004, 0.526), but the evidence for this trend was greatly reduced if adjustment was made for internal liver plutonium dose. In contrast to the mortality findings, there was very little evidence of an association between AMI incidence and external dose, ERR/Gy was equal to 0.029 (95% CI -0.076, 0.134). Among workers with internal liver doses from plutonium exposures, AMI incidence and mortality were raised at 0.1–0.5 Gy relative to lower doses; there was still borderline evidence of such differences after adjusting for external dose. The trends in AMI incidence or mortality with internal dose were not statistically significant. Whilst the data on AMI incidence and mortality were consistent when analysing internal doses, the same was not true when analysing external doses.

Introduction
Acute myocardial infarction (AMI) is one of the most severe forms of the ischemic heart disease with the highest lethality and high percent of complications resulting in long disability, which reduce the quality of life.

The cause of AMI in the overwhelming majority (93-98.5%) of cases is the atherosclerosis of coronary vessels, which is a multifactorial disease, and different endogenous (genetic predisposition, gender, age, hypertension etc.) and exogenous (smoking, emotional stress etc.) factors contribute to its development. Over the last two decades several studies have examined the possible effects of ionizing radiation exposure on circulatory diseases, including AMI. Most of the studies (McGeoghegan et al 2008; Kreuzer et al 2006; Muirhead et al 2009; Shimizu et al 2010; Vrijheid et al 2007) focused on CVD mortality, whereas several studies (Yamada et al 2004; De Bruin et al 2009; Ivanov et al 2006) focused on CVD incidence.

This study was aimed at estimating risks of both AMI incidence and mortality in the cohort of Mayak PA nuclear workers first employed at one of the main plants (reactors, radiochemical, or plutonium) during 1948-1958 in relation to external gamma and internal alpha exposures, taking non-radiation factors into account.

Material and methods

Mayak PA began operations in 1948 as the first and largest nuclear weapons facility in the former Soviet Union and included all the plants necessary for weapon-grade plutonium production, namely, reactors, radiochemical plant, plutonium production plant and auxiliary plants.

From the first days of Mayak PA operation, the special system of personnel medical observation included an obligatory pre-employment medical examination and routine medical examinations of all the workers based on a common standard program. After quitting their job at Mayak, if the former worker stayed in Ozyorsk, he/she was examined at the same specialized medical hospital based on the same standard program. This system of medical observation of Mayak PA personnel health allowed a unique archive of primary medical data to be accumulated and formed the basis for establishment of the unique “Clinic” medical-dosimetry database for the Mayak PA workers cohort (Azizova et al 2008).

The study cohort included 12210 Mayak PA workers, 3552 (29.1%) of whom were women, first employed at one of the main plants from the start of operations through the end of 1958 independently of gender, age, nationality, occupation, and other characteristics. The method of identifying this cohort has been described previously elsewhere (Koshurnikova et al 1988, 1998a, 1998b, 1999).

Vital status as of 31 December 2000 was known for 88.4% of cohort members; of these workers, 52.7% were known to have died and 47.3% were known to be alive as of that time. 53.7% of the 12210 members of the study cohort were known to have left Ozyorsk by 31 December 2000. For persons who continued to be residents of Ozyorsk, vital status was known for all but one person (99.98%). Cause of death was known for 93.5% of deceased cohort members. Morbidity data for the whole period of follow-up were collected for 11597 (95.0%) workers in the study cohort. Only for 5.0% of workers could information not be collected, owing to the loss of their medical documentation. It should be noted that the CVD incidence analysis was restricted to the period of residence in Ozyorsk.

Data on vital status and on dates and causes of death for those workers who migrated from Ozyorsk were provided by the SUBI Epidemiology Laboratory; data on date and causes of death for those Ozyorsk residents whose medical cards and/or case histories had been lost, were provided by the SUBI Occupational Health Laboratory.

Individual dosimetric control of external exposure was performed at Mayak PA from the beginning of operations there. Regular monitoring of internal exposure among those who might have been exposed to plutonium-239 began later, during the 1960s (Vasilenko et al 2007a; Khokhryakov et al 2000). The results of individual monitoring of external and internal exposure were recorded in individual dosimetric cards and journals. The data contained in these documents formed the basis for the establishment of the dosimetric database of Mayak PA workers (Vasilenko et al 2007a, 2007b; Bess et al 2007; Smetanin et al 2007a, 2007b; etc).

Dose estimates from the *Mayak-Doses 2005* dosimetric system were used in this study. Annual doses of external gamma exposure were available practically for all persons in the study cohort (99.9%). The mean (± SD) total gamma dose for the whole period of work at Mayak was 0.91±0.95 Gy (99% percentile 3.9 Gy) for men and 0.65±0.75 Gy (99% percentile 2.99 Gy) for women. The range of total gamma doses was very wide, with 32.6% having a total gamma dose greater than 1 Gy.

Plutonium body burden was measured only for 30.0% of workers. Among workers monitored for plutonium exposure, absorbed dose to liver from alpha radiation was used as a surrogate for the dose to muscle; this latter dose is likely to be similar to the dose to blood vessels and the chambers of the heart. Although doses to the liver and muscle would differ, they should be highly correlated with each other. Consequently, the liver dose can be used to look for any *dose–response* relationship between plutonium exposure and circulatory disease. Amongst those who were monitored, the mean (± SD) total liver dose was 0.40±1.15 Gy (99% percentile 5.88 Gy) for men and 0.81±4.60 Gy (99% percentile 15.95 Gy) for women.

Data on occupational histories and external gamma exposures for the study cohort were provided by the Mayak PA Radiation Safety Department; data on internal alpha exposure from incorporated plutonium-239 were provided by the SUBI Internal Dosimetry Department.

This study focuses on incidence of and mortality from AMI (ICD-9 codes: 410). Follow-up started on the date of first employment at one of the main plants and continued until the earliest of: date of the first diagnosis of AMI (for the incidence analysis) or the date of death from whatever cause (for both the mortality analysis and the incidence analysis); 31 December 2000 for those known to be alive at that time; the recorded date of departure from Ozyorsk (for the incidence analysis); and the date of "last medical information" in the case of unknown vital status. Comparisons were performed within the Mayak PA workers cohort first employed during 1948–1958.

The analyses included the calculation of relative risks (RRs) for categories of one or more factors, having adjusted for other variables. The relative risks for these categorical analyses were calculated by maximum likelihood, using the AMFIT module of EPICURE (Preston et al 1993). 95% confidence intervals for the RRs and p-values from tests of statistical significance were obtained via likelihood-based methods, using AMFIT. Attention was initially directed to non-radiation factors, following which measures of radiation exposure were analysed with adjustment (through stratification) for non-radiation factors. Analyses of internal radiation exposures were restricted to workers known to have been monitored for possible plutonium exposure.

In addition to the categorical analyses, models for trends in disease rates with level of radiation exposures were also fitted to the data, using Poisson regression methods. These models again were fitted using the AMFIT module in EPICURE. In particular, the excess relative risk (ERR) (ie. the relative risk minus 1) was modeled by a linear trend with external or internal dose, with adjustment for non-radiation factors.

In these main analyses, adjustments were made – through stratification – for gender, attained age, calendar period, period of first employment at the main plants of Mayak PA, plant, smoking and alcohol consumption.

Sensitivity analyses were conducted to examine the impact of: a) modifying the set of non-radiation factors (extra adjustment for hypertension, body mass index, employment duration) for which adjustment was made in the analyses of radiation factors; b) restricting the mortality follow-up (like that of incidence) to Ozyorsk, because some migrants were lost to follow-up and because of lower autopsy rates among those who left the city; c) adjusting for internal dose in analyses of external dose and vice versa; d) using various lag periods for external and internal doses. Furthermore, examination was made of how radiation risks might vary by gender, or between plants at Mayak or by attained age.

To allow for the possibility that radiation might affect stroke risk by modifying levels of blood pressure (Preston et al 2003; Ivanov et al 2001) and body mass index (Telnov 1985), the level of these factors at the time of preliminarily medical examination (before employment at Mayak PA) was considered, in order to avoid systematic errors that might arise through adjusting for values of these factors at later times. In contrast, smoking and alcohol consumption were classified at the time of last information (for the mortality analysis) or at the time of last information prior to the first diagnosis of stroke (for the incidence analysis).

Results

By 31 December 2000, 683 disease cases of AMI were diagnosed during 248030 person-years of follow-up and 338 death cases from stroke were identified during 443350 person-years of follow-up.

Non-radiation factors

Analyses of non-radiation factors revealed statistically significant effects of well-known factors such as gender, age, smoking, hypertension, which were taken into account in the analyses of radiation risks, either in the main analysis or in sensitivity analyses.

Radiation factors

External gamma exposure: Table 1 shows that AMI mortality was statistically significantly higher among workers exposed to external gamma rays in total dose above 1 Gy as compared with workers exposed in doses below 0.5 Gy.

Sensitivity analysis (results not shown) revealed that AMI mortality was statistically significantly higher among workers with a gamma dose above 1.0 Gy relative to workers with a gamma dose below 0.5 Gy, irrespective of lag period used, whether additional adjustment was made for other non-radiation factors and internal exposure. AMI mortality was raised among males and radiochemical plant workers.

There was a statistically significant increasing trend of AMI mortality with increasing external gamma dose (Table 1), but the evidence for this trend was greatly reduced if adjustment was made for internal liver plutonium dose.

In contrast to the mortality findings, there was very little evidence of an association between AMI incidence and external dose.

Table 1. RRs and ERR (95% CI) for AMI incidence and mortality in relation to total dose of external gamma exposure (main analysis based on a zero year lag period).

	RRs (vs. <0.5 Gy)		ERR/Gy
	0.5-1.0 Gy	>1.0 Gy	
Incidence	0.912 (0.710, 1.172)	1.034 (0.823, 1.298)	0.029 (-0.076, 0.134)
Mortality	1.104 (0.766, 1.592)	1.693 (1.229, 2.334)	0.265 (0.004, 0.526)

Internal alpha exposure: Table 2 shows that Both AMI incidence and mortality were statistically significantly raised among workers with cumulative absorbed dose to the liver from plutonium of less than 0.1-0.5 Gy as compared with workers exposed to lower internal liver doses. However, sensitivity analysis (results not shown) revealed that these findings were sensitive once different lag periods were used, whether additional adjustment was made for other non-radiation factors. However, there was still borderline evidence of such differences after adjusting for external dose.

Neither for AMI incidence nor for mortality was there a statistically significant trend in risk with internal dose with or without adjustment for external exposure.

Table 2. RRs and ERR (95% CI) for AMI incidence and mortality in relation to total absorbed liver dose from internal alpha exposure (main analysis based on a zero year lag period).

	RRs (vs. <0.1 Gy)		ERR/Gy
	0.1-0.5 Gy	>0.5 Gy	
Incidence	1.279 (1.026, 1.595)	1.255 (0.878, 1.794)	0.078 (-0.084, 0.241)
Mortality	1.854 (1.216, 2.827)	1.488 (0.743, 2.981)	0.211 (-0.171, 0.594)

Discussion

Our analyses of AMI incidence and mortality revealed, as expected, statistically significant effects of well-known factors such as gender, age, hypertension, smoking, which are consistent with findings of other studies. In contrast, our analyses did not reveal any statistically significant effect of alcohol consumption or body mass index on AMI incidence or mortality, either for males and females.

Having adjusted to non-radiation factors there was statistically significant trend in AMI mortality with increasing external gamma exposure, but the evidence for this trend was greatly reduced if adjustment was made for internal liver plutonium dose. In contrast to the mortality findings, there was very little evidence of an association between AMI incidence and external dose

Among workers with internal liver doses from plutonium exposures, AMI incidence and mortality were raised at 0.1–0.5 Gy relative to lower doses; there was still borderline evidence of such differences after adjusting for external dose. However, trend in AMI incidence and mortality with internal liver dose were not statistically significant.

A complication to interpretation is the lack of knowledge as to those tissues or organs for which radiation exposure might increase the risk of stroke, which is particularly problematic in the case of plutonium intakes. For this analysis, liver dose has been used as a surrogate for the dose to muscle, which is likely to be similar to the dose to blood vessels and the chambers of the heart. Furthermore, the liver and muscle doses should be highly correlated with each other. However, there is uncertainty about which tissue or organ dose is appropriate for this type of analysis. A further complication relates to uncertainties in estimates for internal doses for Mayak workers. It should be noted that – because the dose to the liver from intakes of plutonium would be greater than that to the circulatory system – the ERR/Gy estimated here based on liver dose would be lower than that based on dose to blood vessels and the chambers of the heart. For these reasons, the findings in relation to internal exposure need to be interpreted with caution.

In addition, information is not currently available from other studies of populations exposed to external low-LET radiation or plutonium that would allow comparison of risk estimates for AMI taking non-radiation factors such as smoking or alcohol into account in relation to such exposures.

Conclusions

There was a statistically significantly increasing trend in AMI mortality with external dose, having adjusted for non-radiation factors. Much of the evidence for a raised risk concerned workers with cumulative external gamma doses above 1 Gy. The evidence for this trend was greatly reduced if adjustment was made for internal liver plutonium dose. Among workers with internal liver doses from plutonium exposures, AMI incidence and mortality were raised at 0.1–0.5 Gy relative to lower doses; there was still borderline evidence of such differences after adjusting for external dose. The trends in AMI incidence or mortality with internal dose were not statistically significant. Whilst the data on AMI incidence and mortality were consistent when analysing internal doses, the same was not true when analysing external doses.

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Stroke in the cohort of Mayak PA nuclear workers

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Abstract

Incidence of and mortality from stroke have been studied in a cohort of 12210 workers first employed at one of the main plants of the Mayak nuclear facility during 1948–1958 and followed up to 31 December 2000. Information on external gamma doses was available for virtually all of these workers (99.9%); the mean (± standard deviation, SD) total external gamma dose was 0.91±0.95 Gy (99% percentile 3.9 Gy) for men and 0.65±0.75 Gy (99% percentile 2.99 Gy) for women. In contrast, plutonium body burden was measured only for 30% of workers; amongst those monitored, the mean (± SD) absorbed cumulative liver dose was 0.40±1.15 Gy (99% percentile 5.88 Gy) for men and 0.81±4.60 Gy (99% percentile 15.95 Gy) for women. 665 disease cases and 404 deaths from stroke were identified in the study cohort. Having adjusted for non-radiation factors there were no statistically significant trends in either incidence or mortality from stroke with either total external gamma dose or total absorbed internal liver dose; and stroke incidence or mortality rates did not differ significantly between categories for external or internal dose.

Introduction

Cerebrovascular diseases are multifactorial diseases, and different endogenous (genetic predisposition, gender, age, hypertension etc.) and exogenous (smoking, emotional stress etc.) factors contribute to their development. Over the last two decades several studies have examined the possible effects of ionizing radiation exposure on circulatory diseases, including CVD. Most of the studies (McGeoghegan et al 2008; Kreuzer et al 2006; Muirhead et al 2009; Shimizu et al 2010; Vrijheid et al 2007) focused on CVD mortality, whereas several studies (Yamada et al 2004; De Bruin et al 2009; Ivanov et al 2006) focused on CVD incidence.

This study was aimed at estimating risks of both stroke incidence and mortality in the cohort of Mayak PA nuclear workers first employed at one of the main plants (reactors, radiochemical, or plutonium) during 1948-1958 in relation to external gamma and internal alpha exposures, taking non-radiation factors into account.

Material and methods

Mayak PA began operations in 1948 as the first and largest nuclear weapons facility in the former Soviet Union and included all the plants necessary for weapon-grade plutonium production, namely, reactors, radiochemical plant, plutonium production plant and auxiliary plants.

From the first days of Mayak PA operation, the special system of personnel medical observation included an obligatory pre-employment medical examination and routine medical examinations of all the workers based on a common standard program. After quitting their job at Mayak, if the former worker stayed in Ozyorsk, he/she was examined at the same specialized medical hospital based on the same standard program. This system of medical observation of Mayak PA personnel health allowed a unique archive of primary medical data to be accumulated and formed the basis for establishment of the unique “Clinic” medical-dosimetry database for the Mayak PA workers cohort (Azizova et al 2008).

The study cohort included 12210 Mayak PA workers, 3552 (29.1%) of whom were women, first employed at one of the main plants from the start of operations through the end of 1958 independently of gender, age, nationality, occupation, and other characteristics. The method of identifying this cohort has been described previously elsewhere (Koshurnikova et al 1988, 1998a, 1998b, 1999).

Vital status as of 31 December 2000 was known for 88.4% of cohort members; of these workers, 52.7% were known to have died and 47.3% were known to be alive as of that time. 53.7% of the 12210 members of the study cohort were known to have left Ozyorsk by 31 December 2000. For persons who continued to be residents of Ozyorsk, vital status was known for all but one person (99.98%). Cause of death was known for 93.5% of deceased cohort members. Morbidity data for the whole period of follow-up were collected for 11597 (95.0%) workers in the study cohort. Only for 5.0% of workers could information not be collected, owing to the loss of their medical documentation. It should be noted that the CVD incidence analysis was restricted to the period of residence in Ozyorsk.

Data on vital status and on dates and causes of death for those workers who migrated from Ozyorsk were provided by the SUBI Epidemiology Laboratory; data on date and causes of death for those Ozyorsk residents whose medical cards and/or case histories had been lost, were provided by the SUBI Occupational Health Laboratory.

Individual dosimetric control of external exposure was performed at Mayak PA from the beginning of operations there. Regular monitoring of internal exposure among those who might have been exposed to plutonium-239 began later, during the 1960s (Vasilenko et al 2007a; Khokhryakov et al 2000). The results of individual monitoring of external and internal exposure were recorded in individual dosimetric cards and journals. The data contained in these documents formed the basis for the establishment of the dosimetric database of Mayak PA workers (Vasilenko et al 2007a, 2007b; Bess et al 2007; Smetanin et al 2007a, 2007b; etc).

Dose estimates from the *Mayak-Doses 2005* dosimetric system were used in this study. Annual doses of external gamma exposure were available practically for all persons in the study cohort (99.9%). The mean (± SD) total gamma dose for the whole period of work at Mayak was 0.91±0.95 Gy (99% percentile 3.9 Gy) for men and

0.65±0.75 Gy (99% percentile 2.99 Gy) for women. The range of total gamma doses was very wide, with 32.6% having a total gamma dose greater than 1 Gy.

Plutonium body burden was measured only for 30.0% of workers. Among workers monitored for plutonium exposure, absorbed dose to liver from alpha radiation was used as a surrogate for the dose to muscle; this latter dose is likely to be similar to the dose to blood vessels and the chambers of the heart. Although doses to the liver and muscle would differ, they should be highly correlated with each other. Consequently, the liver dose can be used to look for any *dose–response* relationship between plutonium exposure and circulatory disease. Amongst those who were monitored, the mean (± SD) total liver dose was 0.40±1.15 Gy (99% percentile 5.88 Gy) for men and 0.81±4.60 Gy (99% percentile 15.95 Gy) for women.

Data on occupational histories and external gamma exposures for the study cohort were provided by the Mayak PA Radiation Safety Department; data on internal alpha exposure from incorporated plutonium-239 were provided by the SUBI Internal Dosimetry Department.

This study focuses on incidence of and mortality from stroke (ICD-9 codes: 430–432, 434, 436). In the literature, the term "stroke" is sometimes used as a synonym of a wider category of cerebrovascular diseases. In this study the term "stroke" relates to the following group of diseases: subarachnoidal hemorrhage (ICD-9 codes: 430), other and non-specified intracranial hemorrhage (ICD-9 codes: 432), intracerebral hemorrhage (ICD-9 codes: 431), cerebral infarction (ICD-9 codes: 434) and stroke not specified as either cerebral hemorrhage or cerebral infarction (ICD-9 codes: 436). ICD-9 code 435 corresponds to transient cerebral ischemic attack and is not stroke *per se*.

Follow-up started on the date of first employment at one of the main plants and continued until the earliest of: date of the first diagnosis of stroke (for the incidence analysis) or the date of death from whatever cause (for both the mortality analysis and the incidence analysis); 31 December 2000 for those known to be alive at that time; the recorded date of departure from Ozyorsk (for the incidence analysis); and the date of "last medical information" in the case of unknown vital status. Comparisons were performed within the Mayak PA workers cohort first employed during 1948–1958.

The analyses included the calculation of relative risks (RRs) for categories of one or more factors, having adjusted for other variables. The relative risks for these categorical analyses were calculated by maximum likelihood, using the AMFIT module of EPICURE (Preston et al 1993). 95% confidence intervals for the RRs and p-values from tests of statistical significance were obtained via likelihood-based methods, using AMFIT. Attention was initially directed to non-radiation factors, following which measures of radiation exposure were analysed with adjustment (through stratification) for non-radiation factors. Analyses of internal radiation exposures were restricted to workers known to have been monitored for possible plutonium exposure.

In addition to the categorical analyses, models for trends in disease rates with level of radiation exposures were also fitted to the data, using Poisson regression methods. These models again were fitted using the AMFIT module in EPICURE. In particular, the excess relative risk (ERR) (ie. the relative risk minus 1) was modeled by a linear trend with external or internal dose, with adjustment for non-radiation factors.

In these main analyses, adjustments were made – through stratification – for gender, attained age, calendar period, period of first employment at the main plants of Mayak PA, plant, smoking and alcohol consumption.

Sensitivity analyses were conducted to examine the impact of: a) modifying the set of non-radiation factors (extra adjustment for hypertension, body mass index, employment duration) for which adjustment was made in the analyses of radiation factors; b) restricting the mortality follow-up (like that of incidence) to Ozyorsk, because some migrants were lost to follow-up and because of lower autopsy rates among those who left the city; c) adjusting for internal dose in analyses of external dose and vice versa; d) using various lag periods for external and internal doses. Furthermore, examination was made of how radiation risks might vary by gender, or between plants at Mayak or by attained age.

To allow for the possibility that radiation might affect stroke risk by modifying levels of blood pressure (Preston et al 2003; Ivanov et al 2001) and body mass index (Telnov 1985), the level of these factors at the time of preliminarily medical examination (before employment at Mayak PA) was considered, in order to avoid systematic errors that might arise through adjusting for values of these factors at later times. In contrast, smoking and alcohol consumption were classified at the time of last information (for the mortality analysis) or at the time of last information prior to the first diagnosis of stroke (for the incidence analysis).

Results

By 31 December 2000, 665 disease cases of stroke were diagnosed during 249530 person-years of follow-up and 404 death cases from stroke were identified during 443350 person-years of follow-up.

Non-radiation factors

It is known that stroke is a multifactorial disease, therefore analyses of incidence and mortality risks in relation to non-radiation factors were performed first. Our analyses revealed statistically significant effects of well-known factors such as gender, age, hypertension, body mass index, which were taken into account in the analyses of radiation risks, either in the main analysis or in sensitivity analyses.

Radiation factors

<u>External gamma exposure</u>: Table 1 shows that neither for incidence nor for mortality from stroke was there a statistically significant trend in risk with total external gamma dose, nor did incidence or mortality rates differ to a statistically significant extent between categories for external dose. Sensitivity analyses (results not shown) revealed that this finding held irrespective of the lag period used, whether additional adjustment was made for other non-radiation factors and internal exposure, as well as whether analysis was restricted to workers at different plants, or according to gender or attained age.

<u>Internal alpha exposure</u>: Table 2 shows that neither for incidence nor for mortality from stroke was there a statistically significant trend in risk with total internal alpha dose to liver, nor did incidence or mortality rates differ to a statistically significant extent between categories for internal dose. Sensitivity analyses (results not shown) revealed

that this finding held irrespective of the lag period used, whether additional adjustment was made for other non-radiation factors and internal exposure as well as whether analysis was restricted to workers at different plants, or according to gender or attained age.

Table 1. RRs and ERR (95% CI) for stroke incidence and mortality in relation to total dose of external gamma exposure (main analysis based on a zero year lag period).

	RRs (vs. <0.5 Gy)		ERR/Gy
	0.5-1.0 Gy	>1.0 Gy	
Incidence	1.278 (0.999, 1.636)	1.237 (0.977, 1.567)	0.054 (-0.063, 0.170)
Mortality	1.246 (0.911, 1.703)	1.106 (0.816, 1.498)	0.016 (-0.128, 0.160)

Table 2. RRs and ERR (95% CI) for stroke incidence and mortality in relation to total absorbed liver dose from internal alpha exposure (main analysis based on a zero year lag period).

	RRs (vs. <0.1 Gy)		ERR/Gy
	0.1-0.5 Gy	>0.5 Gy	
Incidence	1.001 (0.800, 1.252)	1.080 (0.751, 1.554)	0.017 (-0.097, 0.131)
Mortality	1.401 (1.020, 1.924)	1.047 (0.609, 1.800)	0.120 -0.116, 0.356)

Discussion

Our analyses of stroke incidence and mortality revealed statistically significant effects of well-known factors such as gender, age, hypertension, body mass index, which are consistent with findings of other studies. In contrast, our analyses did not reveal any statistically significant effect of smoking and alcohol consumption on stroke incidence or mortality, either for males and females.

For both incidence and mortality from stroke, there was no evidence for associations with either external or internal radiation in this study.

A complication to interpretation is the lack of knowledge as to those tissues or organs for which radiation exposure might increase the risk of stroke, which is particularly problematic in the case of plutonium intakes. For this analysis, liver dose has been used as a surrogate for the dose to muscle, which is likely to be similar to the dose to blood vessels and the chambers of the heart. Furthermore, the liver and muscle doses should be highly correlated with each other. However, there is uncertainty about which tissue or organ dose is appropriate for this type of analysis. A further complication relates to uncertainties in estimates for internal doses for Mayak workers. It should be noted that – because the dose to the liver from intakes of plutonium would be greater than that to the circulatory system – the ERR/Gy estimated here based on liver dose would be lower than that based on dose to blood vessels and the chambers of the heart. For these reasons, the findings in relation to internal exposure need to be interpreted with caution.

In addition, information is not currently available from other studies of populations exposed to external low-LET radiation or plutonium that would allow comparison of risk estimates for stroke (defined by the ICD-9 codes 430-432, 434, 436) in relation to such exposures.

Conclusions

Having adjusted for non-radiation factors there was no statistically significant trends in both stroke incidence and mortality risk with either total external gamma dose or total absorbed internal liver dose; and stroke incidence or mortality rates did not differ between categories for external or internal dose.

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Uranium health effects: results from the French cohort of uranium processing workers

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Abstract

A cohort of 2709 male workers (72786 person-years) employed at the AREVA uranium processing plant (1960–2005) in France was constructed to investigate the risk of cancer and non-cancer mortality in relation to occupational exposure to uranium. Cause-specific mortality in the cohort was compared to the national and regional mortality rates (1968–2005) by computing standardized mortality ratios. Exposure to natural and reprocessed uranium compounds, classified by their solubility, was assessed trough the plant-specific job-exposure matrix. Internal comparison between exposed and unexposed subjects was then done to estimate relative risk of mortality in regards of each type of exposure. Cox regression models adjusted for age, calendar period and socioeconomic status were used for analyses. At the end of the follow-up, 15% of the cohort was deceased. From 193 cancer deaths, 48 lung cancers and 18 lympho-hematopoietic cancers, considered as uranium target organs, were observed, as well as 101 deaths from cardiovascular diseases. For none of these causes, mortality among the cohort was significantly increased comparatively to the reference mortality rates. Internal comparison showed an elevated risk of mortality from considered causes among workers exposed to reprocessed uranium-bearing compounds. Mortality risk tended to increase with decreasing solubility of uranium compounds. This cohort study provides original results on cancer and non-cancer mortality related to protracted low-dose uranium exposure. Results suggest that effects on mortality differ by type and solubility of uranium compounds. Availability of data on associated exposure to carcinogenic chemical is another interest of this cohort. The cohort is still young but its further follow-up with extension to other AREVA plants will increase statistical power of the future analyses. Further investigation will focus on cardiovascular risk in dose-response analyses including biological parameters into the models.

Leukaemia risk among European uranium miners in dependence on doses from radon, external gamma, and long lived radionuclides

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Abstract

The study presents recent findings based on leukaemia mortality (69 cases) in three European cohort studies of uranium miners, including 9979 Czech, 3271 French, and 34 994 German miners. The risk is analyzed in relation to cumulated equivalent dose from radon and its progeny, from external gamma radiation, and from inhaled long lived alpha radionuclides. Exposures were estimated from measurements of radon, external gamma, and gross alpha activity in the aerosol. The earlier exposure estimates were based on uranium content in the ore and aerosol measurements in mines. The annual absorbed doses to the red bone marrow from exposure to radon gas, its progeny, long-lived radionuclides, and from exposure to external gamma radiation have been calculated for each miner from the first year of employment to the end of follow-up using the ICRP dosimetric and biokinetic models. The mean cumulated absorbed doses in the entire study are 38 mGy from external gamma, 2.1 mGy from radon and its progeny, and 0.9 mGy from long lived radionuclides. In terms of equivalent dose (using a radiation weighting factor 20 for alpha radiations), about 42% is from radon and its progeny, 39% is due to gamma radiation, and 19% is due to inhalation of uranium and its decay products. The risk coefficients (excess relative risk per sievert) were estimated using Poisson regression analysis. For each separate component, the risk coefficient was significant with p-values: 0.010 (gamma), 0.008 (radon), and 0.020 (long lived radionuclides). The estimated risk coefficient for the combined dose was 3.7 (90%CI: 1.1–8.8). Although the estimated risk is subject to some uncertainty due to small numbers and the dose uncertainty, its magnitude is consistent with estimates from other studies.

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Ionizing radiation and mortality from stomach cancer – Results of the German uranium miners cohort study, 1946–2003

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Abstract

Introduction: In the German uranium miners cohort study a 1.15fold statistically significantly increased risk of mortality from stomach cancer compared to the general male population was observed. The aim of the present analyses is to investigate the influence of ionizing radiation.

Methods: The cohort includes 58,987 men who had been employed for at least six months between 1946 and 1990 at the former Wismut uranium mining company in East Germany. By 2003 a total of 20,920 deaths occurred, among them 595 deaths from stomach cancer. Information on exposure to radon progeny in Working Level Months (WLM), external gamma radiation (mSv) and long-lived radionuclides (kBqh/m^3) is based on a detailed job-exposure matrix. Internal Poisson regression models were used to estimate the excess relative risk (ERR) per unit of cumulative exposure and its 95% confidence limits (CI).

Results: There was a statistically significant increase of death from stomach cancer in relation to either cumulative exposure to radon (ERR/100 WLM = 0.022; 95% CI: 0.001–0.043), external gamma radiation (ERR/Sv=1.83; 95% CI: 0.41–3.25) or long-lived radionuclides (ERR/kBqh/m^3=0.018; 95% CI: 0.0044–0.032), respectively. A statistically significantly increased relative risk was observed at exposure categories above 1,500 WLM (RR=1.74; 95% CI: 1.06–1.43) and above 300 mSv external gamma radiation (RR=2.22: 95% CI: 1.07–3.37).

Conclusion: The present preliminary analyses suggest that occupational exposure to ionizing radiation may increase the risk for stomach cancer among uranium miners. In a next step stomach doses based on dosimetric models will be calculated and the corresponding risk estimates presented. Moreover, potential confounder like exposure to arsenic, fine or quartz fine dust will be considered.

Case-control study of lung cancer incidence under combine occupational and domestic radon exposure

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Abstract
Case-control study of lung cancer incidence for the population of Lermontov city (Caucasus, Russia) was performed. Near the territory of Lermontov city the uranium mining was conducted from 1947 to 1990. The considerable part of male population of the city worked in uranium mining. In the Lermontov city three main districts with different kinds of houses exist: multi-storey buildings constructed after 1970; one – three storey buildings constructed before 1970; one-storey rural type houses. The average indoor radon concentrations in these districts are 170, 540 and 910 Bq/m^3 respectively. The lung cancer rate considerably above the average regional level was observed for the population of Lermontov city. The indoor radon concentrations and occupation radon exposure levels in the group of 121 lung cancer cases and in control group composed of 196 peoples were analyzed. Both archive and contemporary indoor radon measurements in combination with the retrospective detectors were used for reconstruction of domestic exposure. The temperature normalization approach was used for assessment of average annual radon concentration in dwellings from the results of radon measurements by charcoal absorbers. The average radon exposure reconstruction period was 38 years for case group and 34 years for control group. The occupation expose information was obtained from medical-sanitary department of former uranium mining facility. It was shown that in average 32% of total radon exposure for uranium miners was due to domestic indoor exposure. It means that in epidemiological studies of uranium miners the domestic radon exposure also should be taken into account. The excess relative risk of lung cancer incidence in case of combine occupational and domestic radon exposure was ERR=0.011 (– 0.003 ÷ 0.029) WLM^{-1}. After the excluding of uranium miners from analysis (46 men from case group and 9 from control group) the excess relative risk value was estimated as ERR=0.0040 (– 0.003 ÷ 0.034) WLM^{-1}.

Introduction

To estimate the influence of radon exposure on lung cancer incidence the case-control studies on uranium miner cohorts were conducted in various countries (Lubin et.al. 1994, Tomasek 2002, Grosche et.al. 2006). The results of these studies were used for estimation radon exposure hazards both for occupational and indoor exposure (BEIR VI 1999). Unfortunately in the analysis of dose-effect response in uranium miners studies the contribution of domestic indoor exposure was not taken into account. As a result the considerable uncertainties in radiation risk assessment can take place because of the indoor radon exposure in uranium rich regions can be comparable with the occupational exposure.

To estimate the radiation risk under the combine occupation and domestic indoor radon exposure the case-control study for the population of Lermontov city (Caucasus, Russia) was performed. Near the territory of Lermontov city the uranium mining was conducted from 1947 to 1991. The considerable part of male population of the city worked on uranium mining facility.

Material and methods

The main cohort in the study was formed from the peoples diagnosed as lung cancer patients from 1995 to 2004. The diagnoses were verified by instrumental and morphological methods. Practically all diagnosed lung cancer patients were included in the case group. The control group was formed from the Lermontov city inhabitants by random approach. The control group corresponded to the gender, age and professional distribution of the adult population of the Lermontov city. The size of case group was 121 and control group – 196 peoples.

The radon exposure estimation was made by the combination of the archive results of the measurements conducted by Regional Department of Federal Medical Biological Agency from 1992 and the measurements conducted by the nuclear track detectors during the our studies. The measurements made by the Regional Department in general were performed by charcoal absorbers sometimes in the combination with the nuclear track detectors. The average charcoal absorber exposure duration has been 4.7 days. More than 3200 archive results of radon measurements were used during the analysis. In the Lermontov city three main districts with different kinds of houses exist: multi-storey buildings constructed after 1970; one – three storey buildings constructed before 1970; one-storey rural type houses. The average indoor radon concentrations in these districts are 170, 540 and 910 Bq/m^3 respectively.

To estimate the average annual radon concentrations in the dwellings of the case and control groups the archive measurement data were analyzed using the temperature normalization approach. The archive meteorological data on the outdoor temperature on the period 1992 – 2004 were used for analysis. For the different kinds of dwellings in which the multiply measurements were carried out the typical dependence of radon concentration relative change on outdoor temperature was obtained. The example of such dependence for one-storey rural type houses is presented on Fig. 1.

For some flats the direct measurements of radon concentrations were not conducted. In this case the results of the measurements performed in the neighbouring flat in the same house were used. In some cases the coefficient connecting the radon concentration between the ground and upper floors was applied (Fig. 2).

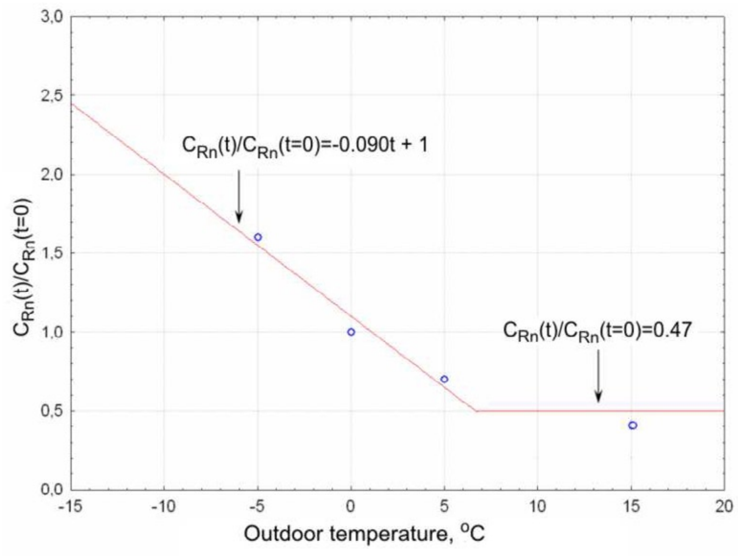

Fig. 1. The dependence of radon concentration on outdoor temperature for one-storey rural type houses in Lermontov city.

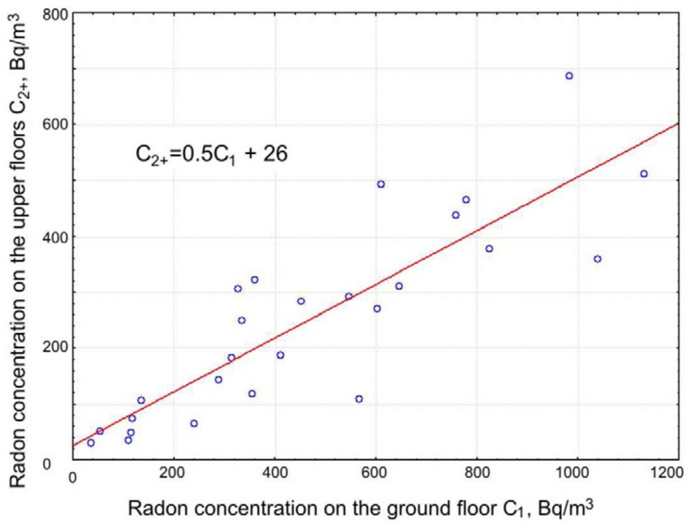

Fig. 2. The dependence of radon concentration between the ground and upper floors for multi-storey houses in Lermontov city.

The correctness of temperature normalization approach for the radon exposure assessment was verified by the use of retrospective surface trap radon detectors. The multilayer LR-115 track detectors designed in our laboratory (Bastrikov, Zhukovsky 2004, Bastrikov et.al. 2006) were placed in the dwellings on the mirrors or furniture glasses. Such comparative measurements have been conducted in 27 dwellings. The age of the objects used for retrospective measurements of radon concentrations was in the range from 13 to 68 years (31 year in average). The good correlation has been found between the values of the average radon concentrations measured by retrospective surface trap detectors and calculated by the temperature normalization approach with the use of archive results of charcoal absorber radon measurements (Fig. 3).

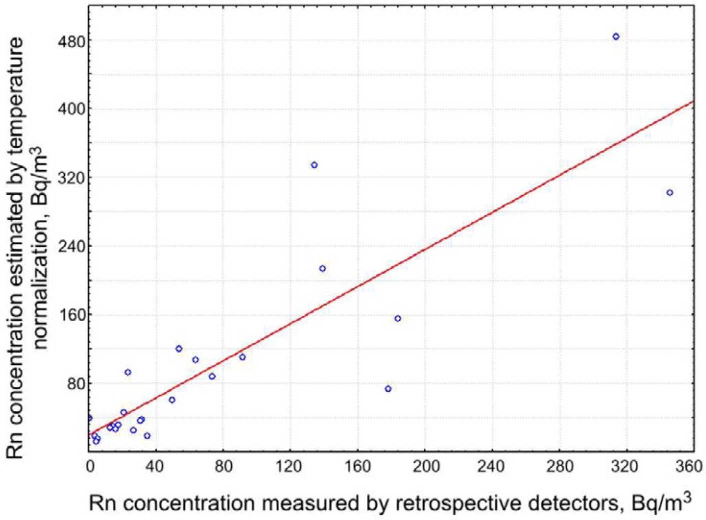

Fig. 3. Correlation between the values of the average radon concentrations measured by retrospective surface trap detectors and calculated by the temperature normalization approach.

Results

The radon concentration distribution parameters for the case and control groups in Lermontov city are presented in the Table 1. For each member of investigated groups the levels of domestic radon exposure have been calculated taking into account the radon levels in the dwellings, duration of the living in the last and in the previous dwellings and the average value of equilibrium factor F=0.44 estimated for the Lermontov city by the direct measurements of equilibrium shift between radon decay products. The average radon exposure reconstruction period was 38 years for case group and 34 years for control group. The last five years before the lung cancer diagnosing or including into control group were not considered during cumulative radon exposure calculations. The parameters of domestic radon exposure distributions are presented in the Table 2.

Table 1. Radon concentration distribution parameters for case and control groups in Lermontov city.

Group	Arithmetic mean, Bq/m^3	Geometric mean, Bq/m^3	σ_{LN}
Case	277	64	1.12
Control	243	73	0.89
All groups	257	70	0.99

Table 2. Parameters of domestic cumulative radon exposure distributions.

Group	Geometric mean, WLM	Minimum, WLM	Maximum, WLM	σ_{LN}
Case	23	0.7	499	1.19
Control	21	0.3	281	1.19
All groups	22	0.3	499	1.19

On the next step of our work the levels of occupational radon exposure were reconstructed. In one of uranium mine near Lermontov city the average radon concentrations decreased from 278 in 1947 to 3.0 kBq/m^3 in 1975. In other uranium mine radon levels decreased from 7.0 to 1.5 kBq/m^3 in period from 1960 to 1991. After 1991 the uranium mining in Lermontov city has been ceased. The levels of individual occupational exposure were reconstructed on the base of monitoring data and information about duration of work in the mines or on the territory of facility logged in workbooks. The occupational exposure was calculated for 60 members of case group and 31 members of control group. Among the professionally exposed peoples 48 miners worked directly in underground mines. The parameters of occupational radon cumulative exposure distributions are presented in the Table 3. The parameters of total radon exposure distribution are given in the Table 4. It was found that for uranium miners in average 32% of total radon exposure was due to domestic indoor exposure. It means that in epidemiological studies for uranium miners the domestic indoor exposure should be taken into account as well as occupation radon exposure.

Table 3. Parameters of occupational cumulative radon exposure distributions.

Group	Geometric mean, WLM	Minimum, WLM	Maximum, WLM	σ_{LN}
Case	42	3	603	1.56
Control	26	3	1352	1.78
All groups	35	3	1352	1.65

Table 4. Parameters of total cumulative radon exposure distributions.

Group	Geometric mean, WLM	Minimum, WLM	Maximum, WLM	σ_{LN}
Case	41	0.7	628	1.34
Control	24	0.3	1388	1.34
All groups	30	0.3	1388	1.36

For radiation risk assessment the odds ratios of lung cancer incidence for combined occupational and domestic radon exposure were calculated. The results are presented in the Table 5.

Table 5. Odds ratio of lung cancer incidence for combined occupational and domestic radon exposure.

Range of Rn exposure, WLM	Peoples in case group	Peoples in control group	Odds ratio	90% confidence interval
0 – 5	8	23	1.00	-
5 – 40	47	102	1.32	0.64 – 2.76
40-140	43	56	2.21	1.04 – 4.68
> 140	23	15	4.41	1.85 – 10.5

The dose-effect dependence can be described by the equation:

$$OR(P_{tot}) = 0.011(P_{tot} - P_0) + 1 , \qquad (1)$$

where P_{tot} – cumulative total (occupational and domestic) radon progeny exposure, WLM; P_0=2 WLM – radon progeny exposure for which the OR=1 assumed. The 90 % confidential interval of the slope factor is in the range from – 0.003 to 0.029 WLM^{-1}.

Odds ratios were also calculated only for situation of domestic indoor exposure. The uranium miners were excluding from analysis (46 men from case group and 9 from control group). The odds ratios for reduced groups exposed only indoors presented in the Table 6.

Table 5. Odds ratio of lung cancer incidence for domestic radon exposure.

Range of Rn exposure, WLM	Peoples in case group	Peoples in control group	Odds ratio	90% confidence interval
0 – 3	4	13	1.00	-
3 – 30	39	100	1.27	0.47 – 3.40
30 – 140	29	68	1.39	0.51 – 3.79
> 140	3	6	1.63	0.37 – 7.22

The dose-effect dependence for domestic radon progeny exposure can be described by the equation:

$$OR(P_{ind}) = 0.004(P_{ind} - P_0) + 1 , \qquad (2)$$

where P_{ind} – cumulative domestic radon progeny exposure, WLM; P_0=5 WLM – radon progeny exposure for which the OR=1 assumed. The 90 % confidential interval of the slope factor is in the range from – 0.003 to 0.034 WLM^{-1}.

Discussion

The most detailed studies of radon-induced lung cancer for uranium miners were presented in the works (BEIR VI 1999, Grosche et.al. 2006). The excess relative risk of lung cancer estimated in these works was 0.0076 (0.0041 – 0.014) WLM^{-1} (BEIR VI 1999) and 0.0021 (0.0018 – 0.0024) WLM^{-1} (Grosche et.al. 2006) respectively. The values of the slope factors obtained in our paper 0.011 (– 0.003 ÷ 0.029) WLM^{-1} for combine exposure and 0.004 (– 0.003 ÷ 0.034) WLM^{-1} for domestic exposure in general are in good agreement with the published data. The differences in the risk assessment for combine occupational and domestic exposure and for indoor exposure in dwellings can be explained by different kinds of uncertainties in occupational and indoor exposure assessments and not very high statistical power of this epidemiological study.

Conclusions

The case-control study is conducted in the Lermontov city (Caucasus, Russia) to establish the connection between lung cancer incidence and combine occupational and domestic radon exposure. The temperature normalization approach has been developed to estimate the cumulative indoor radon exposure on the base of archive radon measurements by charcoal absorbers and track detectors. The good correlation has been found between the values of the average radon concentrations measured by retrospective surface trap detectors and calculated by the temperature normalization approach. The levels of individual occupational exposure were reconstructed on the base of monitoring data and information about duration of work in the mines or on the territory of facility. It was found that for uranium miners in average 32% of total radon exposure is due to domestic indoor exposure. The excess relative risk of lung cancer incidence in case of combine occupational and domestic radon exposure is 0.011 (– 0.003 ÷ 0.029) WLM^{-1}. After the excluding of uranium miners from analysis (46 men from case group and 9 from control group) the excess relative risk value was estimated as 0.0040 (– 0.003 ÷ 0.034) WLM^{-1}. The obtained results are in good agreement with BEIR VI data.

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Late effects of radioactive and chemical contamination

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Abstract

The influence of the polluted environment on health of the population has been studied for a decades in industrial countries. It is established that polluted environment can cause great variety of diseases in all categories of inhabitants. Most vulnerable groups are children and pregnant women because of possible genetic or teratogenic defects in the newborns. To prevent this situation, control of the environment is established. Besides the usual industrial pollutions, during the NATO air strikes on Serbia in 1999, many industrial facilities were destroyed and harmful, toxic substances were released in concentrations even ten thousand times higher than maximum permissible. Additionally, depleted uranium ammunitions were deployed in south of Serbia. In the years after, health surveillance of the soldiers who served in DU contaminated regions in Kosovo were organized and increased cancer incidence is noticed as well as non-specific diseases (cardiovascular, rheumatic) and psychological disturbances, including PTSP, alcohol abuse and social problems. According to EUROCAT protocol their children born after the strikes were followed-up too, and higher rate of congenital (skeletal, cardio-vascular) and chromosomal anomalies, immunological disorders and endocrine diseases were noticed. Similar results were found in the inhabitants of the most polluted regions.

Mortality due to gastrointestinal cancers in northern part of East-Ural Radioactive Trace

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Abstract
The objective of the study was to evaluate the register of causes of death, created using deaths certificates issued by rural municipalities at northern part of EURT, as a source of information for analysis of mortality caused by radiation induced gastrointestinal cancers. In the register the case (6158) and control (4844) groups were formed according to assessments of initial contamination of municipalities by Sr-90 – above and bellow 0.1 Ci per square kilometer respectively. Average colon dose 0.07 Gy and 440 gastrointestinal cancer deaths was observed in case group. The analysis included comparison of gastrointestinal cancers contribution to total number of deaths in the groups and estimation of excess number of deaths for sex, age and time since accident categories. To assess years of life lost (YLL) due to excess cancer cases the life durations were compared. By results of estimation total number of excess cancer deaths is 57±39 (with 90% CI) and total YLL is 893±279 for period from 1967 to 2000.

Introduction
Accidental explosion of waste storage tank at former Soviet Union plutonium production plant “Mayak” in 1957 had released considerable radioactivity to atmosphere and caused contamination of the environment (East-Ural Radioactive Trace, EURT). Available data on the radioactive contamination allowed the assessment of post-accident population radiation exposure. For analysis of radiation induced health effects the rural population of Kamensky rajon (Sverdlovsk oblast, Russia) was evaluated. That administrative district is situated in northern part of EURT, as far as 80-130 km from “Mayak” plant. The radioactive trace crossed territory of Kamensky rajon and caused contamination by ^{90}Sr up to 190 kBq/m^2.

Rural population of the district (more than 30 000 peoples in 1957) was chosen for analysis because its homogeneity in relation to social and environmental factors other than accidental radiation exposure. Radiation exposure of rural population directly associates with the levels of initial contamination of residential area and agricultural lands. Due to consumption of contaminated food and milk gastrointestinal tract received higher doses. This presentation describes the results of analysis of case of deaths related to cancers of gastrointestinal organs (esophagus, stomach, liver and colon) in rural population of northern part of EURT.

Material and methods

The primary source of information on causes of deaths of EURT population from Kamensky rajon was the archive of Sverdlovsk oblast civilian registry office. The death certificates issued by rural municipalities, which stored in the archive, contain the information as follow: unique identification of death certificate, sex, date of death, date of birth, place of birth, last place of residence, duration of life at last place of residence and the cause of death. Totally data on 15,685 cases of death for period 1954-2000 were obtained from the archuve. Obtained data were used to create Register of causes of deaths of rural population of Kamensky rajon (Register).

Cumulated doses absorbed due to radioactive contamination were estimated according to Guidelines "Reconstruction of doses accumulated by residents of the Techa river basin and the area affected by the accident at the "Mayak" plant in 1957" approved by Ministry of Health of Russia in 1995 (Guideline 2.6.1.024-95). The main factors contributed to radiation doses are consumption of contaminated crops, vegetables and milk, external exposure from contaminated soil and inhalation as well as external exposure during transfer of radioactive cloud. The Guideline suggests intake rates of radionuclides due to inhalation and ingestion and external exposure rates standardized per unit contamination by ^{90}Sr.

Case and control groups of the study were formed by dividing the Register due to level of initial contamination. Control group was formed from the residents of settlements with initial contamination level equal or bellow 0.1 Ci/km^2 and case group – from residents of settlements with initial contamination higher than 0.1 Ci/km^2. Thus control and case groups include residents of the same administrative territory.

Relative contribution of gastrointestinal cancers to the total mortality in the control group was estimated as ratio of number of deaths from gastrointestinal cancers (n) to total number of deaths (N). Estimated relative contribution was then applied to total number of deaths in case group (M) to assess expected number of deaths from gastrointestinal cancers in case group (E):

$$E = M\frac{n}{N}. \qquad (1)$$

Comparing the expected and observed numbers of deaths the excess number of deaths O-E and observed to expected ratio O/E were estimated. The variances of O and E were assessed accepting Poison distribution for n and O.

For each case of death at age A the years of expected life lost ($L(A)$), was accepted using data of demographic statistics for region. Then observed and expected total years of life lost (TYLL) for cases of gastrointestinal cancer deaths in case group were estimated using equations:

$$ObsTYLL = \sum_{\substack{Case\\Group}} L(A) \qquad (2)$$

$$ExpTYLL = \frac{M}{N}\sum_{\substack{Control\\Group}} L(A) \qquad (3)$$

Estimated expected and observed values were applied to assess excess TYLL of life lost $\boldsymbol{ObsTYLL - ExpTYLL}$ and relative value $\dfrac{\boldsymbol{ObsTYLL}}{\boldsymbol{ExpTYLL}}$.

To estimate the variance of TYLL the variance of expected life lost was accepted as follows:

$$\mathrm{Var}\left(\sum_{i=1}^{K} L_i\right) = \mathrm{Var}(L_0) + \ldots + \mathrm{Var}(L_i) + \ldots + \mathrm{Var}(L_K) \approx K \cdot \mathrm{Var}(L), \qquad (4)$$

To adjust assessments for sex and age at exposure factors the estimations were performed in subgroups formed according to categories of the factors (for age in 1957 five years categories were considered). Then the estimations were summarized to draw the final results.

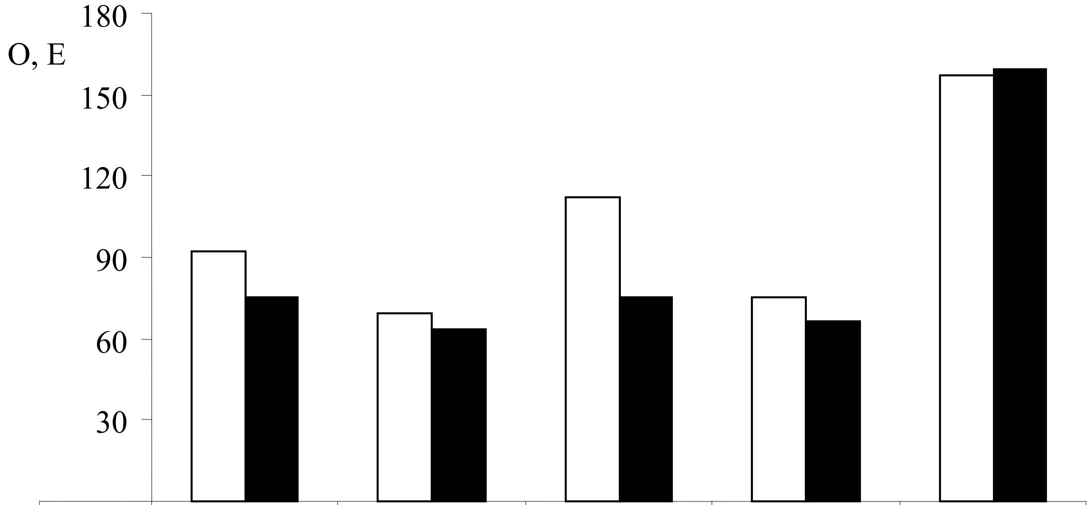
Results

General characteristics of case and control groups are presented in Table 1. Dependence of total observed and expected cases of gastrointestinal cancers in case group on calendar year is presented on Fig. 1. For periods before 1967 the difference between expected and observed values is insignificant. Significant difference ($p<0.05$) is obtained for period 1967-1977, for periods after 1977 the difference demonstrates a tendency of decreasing ($p>0.05$). Values of excess cancer deaths in case group are also presented on Fig. 1. After summarizing for period 1967-2000 estimated number of excess gastrointestinal cancer cases is equal to 57±39 (with 90% confidence interval). Estimated related excess TYLL in case group is 870±280 years, which is equivalent to about 15 years of life lost per excess case of cancer. For cancers other than gastrointestinal estimated number of excess cases (6±49) and excess TYLL (-222±410) in case group are insignificant.

Values of O-E and ObsTYLL-ExpTYLL in case group for period 1967-2000 in dependence on age in 1957 were tested (Table 2). For age at exposure interval 35-65 total value O-E = 54±39 (with 90% confidence interval). For age younger 35 in 1957 insignificant value of O-E is obtained. Significant values of excess TYLL in case group are observed for each ten years category in interval 35-65, where total value is 797±129 (with 90% confidence interval). For younger and elder ages at exposure the values of excess TYLL are insignificant.

Table 1. General characteristics of case and control groups

Characteristic	Case group	Control group
Number of entries	6 158	4 844
Male/female ratio	0.486/0.514	0.482/0.518
Total years of life after 1957	120 877	107 238
Mean colon dose, mGy	70	4.5
Number of gastrointestinal cancer deaths	440	278



Year	1954-1961	1962-1966	1967-1976	1977-1986	1987-2000
O-E	17±26	5.6±22	37±25	8.7±21	-2.3±32

Fig. 1. Observed (white boxes) and expected (black boxes) number and excess cases (O-E) of gastrointestinal cancer deaths in case group.

Table 2. Estimated number of excess of gastrointestinal cancer deaths and related excess TYLL in case group.

Age in 1957	O-E	ObsTYLL-ExpTYLL
<35	3.5±16	13±167
35-45	15±17	292±170
45-55	20±19	270±98
55-65	18±21	235±78
65-75	5.5±13	12±39
>70	-5.1±5	-

To compare risk of gastrointestinal cancers in case and control groups by age at exposure categories the relative risk parameters O/E and ObsTYLL/ExpTYLL were estimated. Two approaches yielded matching results though uncertainty of estimation by excess TYLL is in general lower than by excess cancer cases estimation. The parameter ObsTYLL/ExpTYLL demonstrates a tendency to decreasing with age at exposure from 1.5 (1.2-2.1 with 90% confidence interval) at age 35-45 to 1.0 (0.97-1.1 with 90% confidence interval) at age 65-75 in 1957.

Discussion

The crucial issue of the analysis is association of obtained excess cases of gastrointestinal cancer deaths and excess TYLL with accidental radiation exposure. In general the correlation of excess values with radiation exposure can be considered as reasonable evidence. However, the created Register is not sufficient to search such correlation. Nevertheless, we believe that the reasons such as follow support conclusion on relationship between estimated excess gastrointestinal cancer mortality in the case group of Register and the exposure. The case and control groups of studied rural population live within the same administrative and geographical territory and are homogeneous considering the social and demographic factors. It can be quite certainly supposed that the groups differ only by the factor of accidental radiation exposure. Considerable decreasing of gastrointestinal cancer mortality appeared within the period 10-20 years after accidental exposure, which can be related to the period of latency of the gastrointestinal cancers. There are no excess cases in the elderly population. Absence of excess cases deaths due to cancers other than gastrointestinal is in agreement with estimated nonuniform radiation exposure of organs and tissues.

Conclusions

Analysis of the register of causes of death of population live within northern part of EURT revealed increase of mortality due to gastrointestinal cancers, which can be associated with radiation exposure after accident at Mayak nuclear plant in 1957.

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Incidence of childhood leukaemia in the vicinity of Finnish nuclear power plants

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Abstract

Childhood leukaemia near nuclear installations has been investigated widely in the past decades, but the results have been inconclusive. Radiation exposure for the population appears negligible.

We investigated leukaemia incidence in children living near the two Finnish nuclear power plants (NPPs) using both cohort and case-control analysis.

The residential cohorts defined by censuses in 1980 and 1990 showed no increased risk of childhood leukaemia in the population residing within a 15-km zone from the NPPs compared to the 15–50-km zone.

In the case-control analysis with individual residential histories for 16 children with leukaemia and their 64 matched controls, residential distance from the NPP was not associated with leukaemia.

The results of the cohort and case-control analyses were consistent and neither of them indicated an increased risk of leukaemia. Even though the small sample size and lack of population residing within the 5-km radius limit the strength of the conclusions, the findings are reassuring from the public health perspective.

Introduction

Since 1983, childhood leukaemia near nuclear installations has been investigated widely but the results have been inconclusive so far (Laurier et al. 2008). The positive findings have been mainly limited to reprocessing sites while studies covering more than 100 nuclear power plants (NPPs) have shown little indication of increased risks. An exception was the recent German study reporting an increased risk of leukaemia among young children (aged <5 years) but not older ones living within a 5-km range around NPPs (Kaatsch et al. 2008). This finding raised public health concern and the need for information in Finland where a new NPP is being built but no published study has been available on childhood leukaemia and nuclear power.

We investigated leukaemia incidence near Finnish NPPs in children (aged <15 years) using cohort and case-control analyses which included information on residential history (Heinävaara et al. 2010).

Material and methods

Finland has four nuclear power reactors located in two nuclear sites, both having two reactors. Loviisa has been in commercial production since May 1977 and Olkiluoto since October 1979. Both sites are located at the seashore (Fig 1, large graph). The fifth nuclear power reactor is currently under construction in Olkiluoto.

Municipalities with any area within 15 km from the NPP were defined as being situated near to an NPP. According to this definition seven municipalities situate near to NPPs (Fig 1, small graphs).

In the cohort analysis, cohorts of people living near NPPs were formed based on census data at the end of 1980 and 1990. The coordinates of residence were obtained for these cohorts at the end of 1980 and 1990 from the Population Register Centre. Leukaemia cases diagnosed during the follow-up of the cohorts until the end of 2000 were obtained from the Finnish Cancer Registry. Leukaemia cases and data population counts were aggregated into 2 km * 2 km squares based on the coordinates of residence in the beginning of the follow-up. Leukaemia incidence in cohorts living within a 15-km zone around the NPPs was compared to that in reference cohorts living in the 15–50-km zone (Fig 1, small graphs). We calculated the indirectly standardized risk ratios (RRs) adjusting for age and socioeconomic status.

In the case-control analysis, the association between the residential distance from the nearest NPP and leukaemia was studied. Cases diagnosed with leukaemia between January 1, 1977 and December 31, 2004 and living in the municipalities near an NPP were identified from the Finnish Cancer Registry. Four controls were randomly chosen from the Population Registry Centre and matched to cases with respect to sex, age, and the municipality of residence at index date (i.e., date of diagnosis of the corresponding date). Residential histories of all cases and controls were obtained from the Population Register Centre. Case-control analysis included 16 children, 11 boys and 5 girls, with their 64 matched controls. All cases had acute lymphocytic leukaemia.

In the case-control analysis, residential history was taken into account from the date of production of the nearest NPP until the index date. The distances from the nearest NPP to each residence and the corresponding durations were calculated for each subject. Average distance was calculated as the sum of distances weighted by their relative durations. Categorized average distance was used as a primary measure for residential distance. Categorized distance at diagnosis and minimum distance were used as secondary measures for residential distance. Minimum distance was the shortest distance from any of the subjects' residence to the NPPs. The association between leukaemia and the residential distance measures was assessed with odds ratios (ORs).

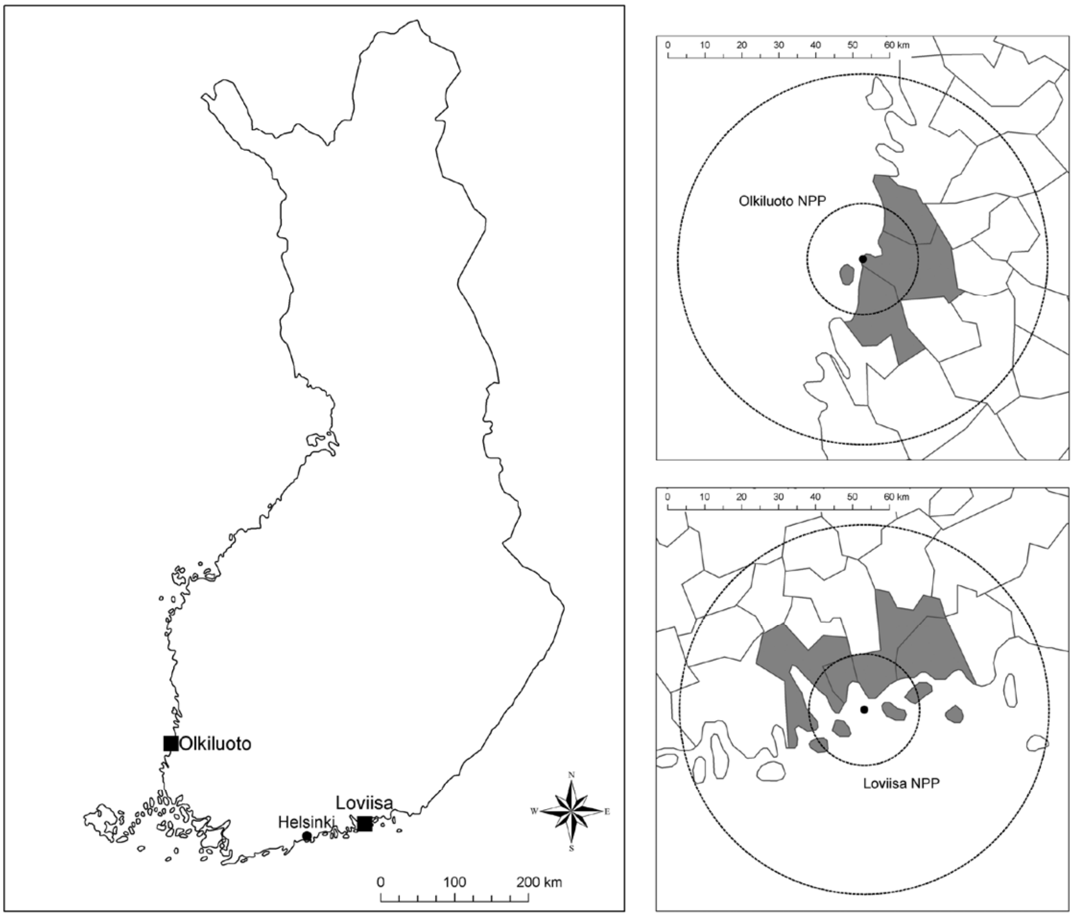

Fig. 1. Nuclear power sites Loviisa and Olkiluoto are located at seashore in the Southern and Western Finland, respectively (large graph). Municipalities near Olkiluoto site are Eurajoki, Luvia and Rauma whereas Loviisa, Ruotsinpyhtää, Pernaja and Pyhtää are near Loviisa site. Case-control analysis cover grey area. Small and large circles illustrate residential zones of 0-15 km and 15-50 km around the NPPs used in cohort analysis (small graphs). ***Source:*** **Municipal boundaries © National Land Survey of Finland, licence 53/MML/10.**

Results

In the cohort analysis, the rate ratios (RRs) of childhood leukaemia for the cohort living within the 15-km zone from an NPP were 1.0 (95% CI 0.3, 2.6) for the 1980 residential cohort and 0.9 (95% CI 0.2, 2.7) for the cohorts of 1990. For both cohorts, incidence of childhood leukaemia within the 15-km zone from the NPPs was comparable to that in the 15-50-km zone (Table 1).

Table 1. Observed leukaemia cases (Obs) within the 15-km zone from the NPPs and rate ratios (RRs) with 95% confidence intervals (95% CI). Expected incidence within the 15-50-km zone from the NPPs was used as a reference.

	Boys			Girls		
Childhood leukaemia	**Obs**	**RR**	**95% CI**	**Obs**	**RR**	**95% CI**
Residential cohort of 1980	3	1.1	0.2, 3.3	1	0.8	0.0, 4.5
Residential cohort of 1990	2	1.2	0.2, 4.5	1	0.6	0.0, 3.3

In the case-control analysis, the mean (maximum) of average distance was 18.4 km (59.7 km) km for leukaemia cases and 19.3 km (86.4 km) for controls. None of the children lived within the 5-km zone around the NPPs (Table 2). The odds ratio (OR) of leukaemia in the closest category (5-9 km) was 0.7 (95% CI 0.1, 10.4) compared to the reference, ≥30 km zone. None of the ORs related to categorized average distance differed significantly from the unity (Table 2). The result was the same for categorized distance at diagnosis and minimum distance.

Table 2. Odd ratios (ORs) with 95% confidence intervals (95% CI) of childhood leukaemia related to categorized average distance in the municipalities near the NPPs.

Average distance	Cases	Controls	OR	95% CI
0-4 km	0	0	-	
5-9 km	1	5	0.7	0.1, 10.4
10-19 km	11	41	0.9	0.2, 4.4
20-29 km	1	9	0.3	0.0, 3.6
≥30 km	3	9	1.0	

The results showed no heterogeneity between nuclear power sites, sexes, 5-year age groups and 10-year calendar period in cohort or case-control analysis.

Discussion

We assessed the risk of childhood leukaemia in populations living near the Finnish nuclear power plants. The results of the cohort and case-control analysis were consistent and did not indicate an excess of leukaemia.

The strengths of the study include comprehensive individual residential histories for all subjects in the case-control study. The nationwide, population-based cancer registry allowed identification of the complete roster of leukaemia cases in the study population. Combining two approaches in the study allowed a thorough evaluation of the leukaemia risk.

A key limitation of the analyses was the small number of cases. Therefore more detailed analyses, such as childhood leukaemia in young children aged less than 5 years living within a 5-km zone around the NPPs (corresponding to the findings of the German KiKK study), could not be assessed. As in other studies on the issue, exposure assessment was challenging as the monitoring of exposures indicates practically zero doses from the power plant exposures. If there is any excess radiation exposure to the population residing close to the power plants, it would have to be from internal exposure, which is more difficult to measure than external radiation. Yet, the concentrations of radionuclides are very low and calculated maximal dose contribution negligible.

Conclusions

This study showed no evidence of increased risk of leukaemia in children living close to the nuclear power plants in Finland.

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Radiation doses from global fallout and cancer incidence among reindeer herders and Sami in Northern Finland

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Abstract

People in the Arctic regions can be heavily exposed from the global radioactive fallout due to their diet rich in reindeer meat in which radionuclides accumulate. The primary aim of this study was to assess whether the estimated lifetime cumulative radiation doses for the Arctic population from atomic bomb testing have had detectable effects on the incidence of cancer. A cohort of the Arctic population in Finland (n = 34,653) was identified through the Population Register Center with grouping by reindeer herding status, ethnicity and radiation exposure. Annual average radiation doses, based on ^{137}Cs whole-body measurements, were assigned by birth-year, gender and reindeer herder status. Cancer cases of radiation-related cancer types (cancers of the bladder, female breast, colon, esophagus, liver, lung, ovary, stomach, brain and nervous system, bone and thyroid, non-melanoma of the skin, basal cell carcinoma of the skin, and leukemia) during 1971-2005 were identified from the Finnish Cancer Registry. A total of 1,580 cancer cases were observed versus 1,948 expected on the basis of incidence rates in Northern Finland [standardized incidence ratio (SIR) was 0.81 with 95% confidence interval (CI) of 0.77-0.85]. For the reindeer herders SIR was 0.72 (95% CI 0.60-0.85) and for the Sami people SIR was even lower, 0.47 (95% CI 0.37-0.60). No association between the lifetime cumulative radiation exposure from global radioactive fallout and cancer incidence in the Arctic population was found. Potential underestimation and misclassification of the radiation dose may affect the results and the findings should be interpreted with caution.

Introduction

Most of the atmospheric testing of nuclear weapons was performed during 1945-1963 and the highest global fallout from these tests occurred in the early 1960s (UNSCEAR 2000). ^{137}Cs is the most important component of human radiation exposure from the fallout. Its persistence in the environment is enhanced by a slow turnover in the northern ecosystems. After the A-bomb testing period further exposure resulted from the Chernobyl nuclear power plant accident in 1986. However, the ^{137}Cs deposition in Northern Finland was only a fraction of that received from atomic bomb testing. The route of exposure among Arctic population is mainly through the intake of food with elevated radioactivity, resulting in protracted internal exposure (Rahola and Suomela 1998). For human exposure, the lichen-reindeer-human pathway is important, because ^{137}Cs is enriched in the food chain (Golikov et al. 2004; Miettinen and Häsänen 1967). The diet of Arctic people in Lapland is rich in reindeer meat and they have several-fold higher ^{137}Cs body burdens from the global test fallout than other populations of the Nordic countries (Rahola and Muikku 2004; Rahola and Suomela 1998).

The Sami people are the indigenous population of the Northern Scandinavia and the Kola Peninsula, and are genetically distinct from other Northern European people (Lahermo et al. 1996). In all published epidemiological studies, the Sami populations have had a cancer incidence below the national and regional average (Haldorsen and Tynes 2005; Hassler et al. 2004; Hassler et al. 2005; Hassler et al. 2008; Soininen et al. 2002; Wiklund et al. 1990; Wiklund et al. 1991). The lower cancer risk of the Sami has been suggested to reflect lifestyle and/or genetic factors. However, the non-Sami population in the Arctic areas has not been investigated. Even though the radiation exposure has been the motivation for several studies on cancer risk among the Nordic indigenous populations, no information on radiation doses has been available in earlier studies.

Material and methods

The baseline cohort consisted of all 36,461 Finnish residents identifiable from the Population Register Center born in the northernmost municipalities of Finland (Figure 1). These areas were chosen because they have the highest numbers of reindeer herders and people of Sami ethnicity. Subjects in the baseline cohort were born between years 1860 and 2001.

People registered as reindeer herders in the censuses carried out in the end of 1970, 1975, 1980, 1985, 1990, or 1995 were identified from Statistics Finland. Persons resident in Finland on 1 January 1971 or later were included in the final study cohort (n = 34,653). Persons whose own mother tongue or that of both parents was any of the three Sami languages were defined as ethnic *Sami* (Table 1).

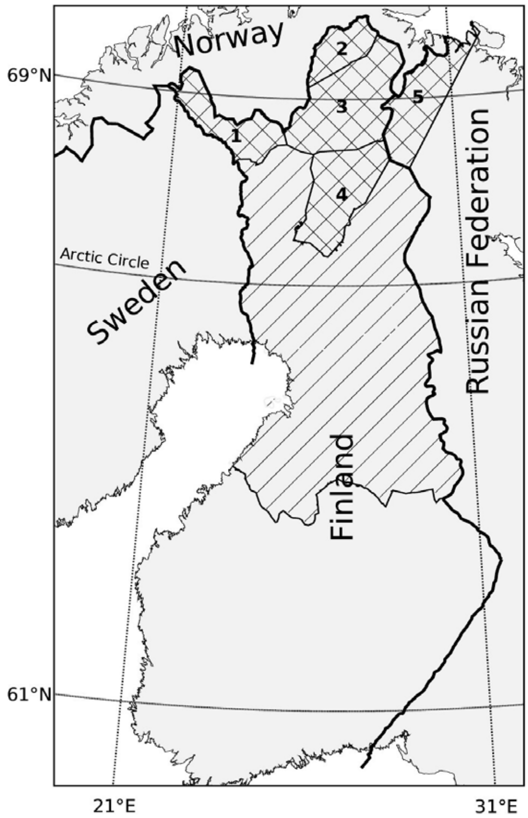

Figure 1. Birth municipalities of the study cohort members: 1 = Enontekiö; 2 = Utsjoki; 3 = Inari; 4 = Sodankylä; 5 =Petsamo (in Russia). The reference region for defining the expected numbers of cases (Cancer Care Responsibility Region of the Oulu University Hospital) is illustrated with a raster without a number.

Mean doses were assigned to groups defined by birth-year, gender, and reindeer herding status. Radiation dose assessment was primarily based on the ^{137}Cs body burden among the reindeer herders and other people in Lapland during 1961-2003. The ^{137}Cs body burden estimates from the whole-body measurements were available from surveys carried out by the Department of Radiochemistry, University of Helsinki in 1961-1977 and by STUK in 1986-2003 (Rahola and Muikku 2004). Annual doses were estimated for reindeer herders and non-reindeer herders according to gender (Figure 2). The cumulative whole-body dose was calculated as the sum of the assigned calendar-year-specific annual doses, taking into account the reindeer herding status. If a subject had a reindeer herding occupation in the earliest available census or if either parent of a child was a reindeer herder, the beginning of the reindeer-related exposure was assumed to start at three years of age. For spouses of reindeer herders, the beginning of the reindeer-related exposure was assumed to be the wedding year. Parents of a reindeer herder whose own occupation was unknown were assumed to have been reindeer herders, and calculation of reindeer-related exposure for them was assigned to start at three years of age. Subjects classified as reindeer herders remained in that exposure category regardless of subsequent changes in occupation or place of residence (assuming that their consumption of reindeer meat remained high).

Table 1. Description of the study population. Number of subjects (N) and person-years during 1971-2005.

		Total	
		N	Person-years
All		34,653	874,391
	Men	17,262	429,433
	Women	17,391	444,958
Reindeer herder		2,786	70,287
	Non-Sami	716,154	46,225
	Sami	917	24,063
Non reindeer herder		31,867	804,104
	Non-Sami	30,747	774,869
	Sami	1,120	29,236
Estimated cumulative radiation dose[a] (mGy)			
	0-0.9	13,523	172,293
	1-4.9	9,916	192,209
	5-9.9	19,665	376,65
	>10	6,612	133,24

[a] Based on ^{137}Cs whole body measurements. A person may change exposure category during the follow-up period and therefore contribute to numbers of more than one category

The cumulative dose (D) was assigned as a function of follow-up

$$D = \sum_{t_o}^{t_1} d_{ijk}$$

where t_0 is the start of exposure and t_1 is the time of observation. The annual dose, d, depends on: gender (i), calendar year (j), and reindeer herding status (k).

Incident cancer cases occurring in 1971–2005 were identified from the Finnish Cancer Registry. Results are given for *a priori* selected radiation-related cancer sites. The category of "*radiation-related cancer sites*" is defined to consist of cancers consistently associated with radiation in previous studies, i.e., cancers of the urinary bladder, female breast, colon, esophagus, liver, lung, ovary, stomach, bone, basal cell carcinoma of the skin, brain and nervous system and thyroid, as well as leukemia.

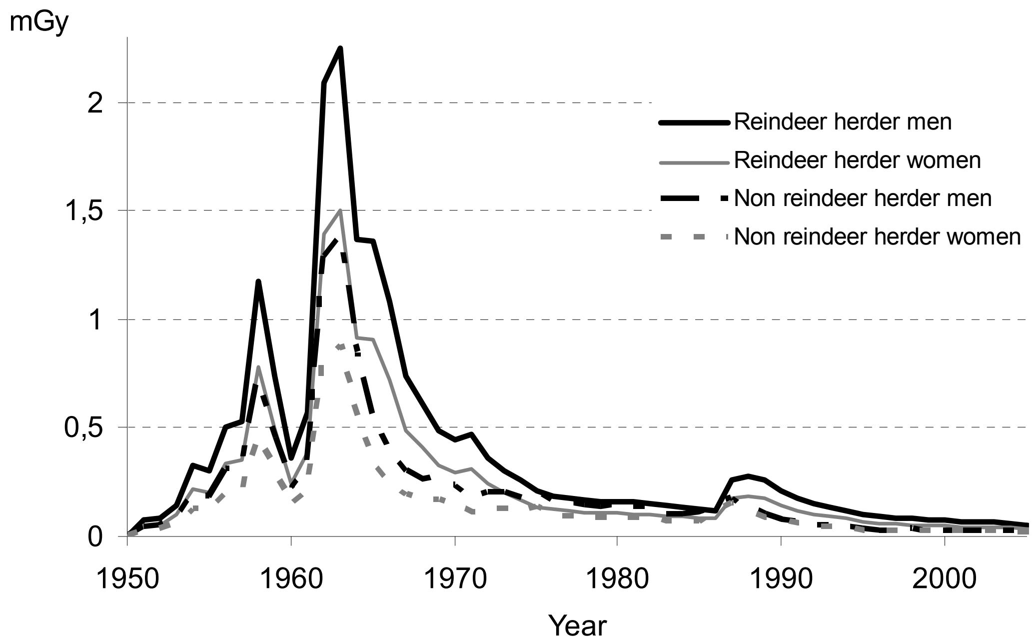

Figure 2. Average annual internal doses (mGy) from the ^{137}Cs for reindeer herders and other persons in Lapland.

The observed numbers of cancer cases and person-years at risk in the study cohort were counted according to sex and five-year age groups, separately for five calendar periods. The person-years were calculated until date of cancer diagnosis, death, emigration or the end of 2005, whichever came first. Follow-up in a given dose category was started when the cumulative dose reached the threshold level of the category with lagging by the induction period (latency). A latency of two years was allowed for leukemia, five years for thyroid cancer and ten years for other cancers. The expected numbers of cancer cases were calculated by multiplying the number of person-years in each stratum by the corresponding average cancer incidence in Northern Finland (Figure 1). The standardized incidence ratio (SIR) was defined as the ratio of observed to expected number of cases. The exact 95% confidence interval (95% CI) for each SIR was estimated on the presumption that the number of observed cases followed the Poisson distribution.

An ethical review of the study protocol was conducted by the Radiation Protection Advisory Board. The Population Register Centre gave the permission to use the population database for identifying the study population. The permission to use the census data on occupation was obtained from Statistics Finland. The National Research and Development Centre for Welfare and Health gave permission to use Finnish Cancer Registry data.

Results

A total of 1,580 cases of radiation-related cancer were observed versus 1,948 expected in the study population, indicating lower cancer risk in the study area than in the entire Northern Finland. The cancer incidence among the Sami was lower than in the rest of the study cohort, independent of the reindeer herding status. Reindeer herders had a slightly lower cancer incidence than non-reindeer herders (Table 2). The SIR for radiation-related cancers was not associated with the estimated cumulative radiation dose (Table 3).

Table 2. Observed (Obs) and expected (Exp) number of cancer cases and standardized incidence ratios (SIR) and 95% confidence intervals (95% CI) of radiation-related cancer sites by the reindeer herding status and ethnicity in the Arctic population in Finland.

	Sami		Non-Sami		All	
	Obs/Exp	SIR (95% CI)	Obs/Exp	SIR (95% CI)	Obs/Exp	SIR (95% CI)
Reindeer herder	35/67	0.52 (0.36-0.73)	102/124	0.82 (0.67-1.00)	137/191	0.72 (0.60-0.85)
Non reindeer herder	33/77	0.43 (0.29-0.60)	1,410/1,680	0.84 (0.80-0.88)	1,443/1,757	0.82 (0.78-0.86)
All	68/144	0.47 (0.37-0.60)	1,512/1,804	0.84 (0.80-0.88)	1,580/1,948	0.81 (0.77-0.85)

Table 3. Observed (Obs) and expected (Exp) number of radiation-related cancer cases and standardized incidence ratios (SIR) and 95% confidence intervals (95% CI) of the Arctic population in Finland by estimated radiation dose based on ^{137}Cs whole body measurements during 1971-2005.

Dose	Obs/Exp	SIR (95% CI)
0-0.9 mGy	137/191	0.72 (0.60-0.85)
1-4.9 mGy	33/44	0.76 (0.52-1.06)
5-9.9 mGy	811/966	0.84 (0.85-0.97)
>10mGy	307/378	0.81 (0.72-0.91)

Discussion

We evaluated cancer risks associated with estimated cumulative radiation doses, based on ^{137}Cs whole-body measurements, among the population in northernmost Finland. The cancer incidence for several primary sites was lower than among the other inhabitants of Northern Finland. No association was found between the cumulative radiation dose and radiation-related cancer sites.

The dose estimation of the current study is prone to underestimation and misclassification, mainly because reindeer meat is also rich in natural polonium, while only dose from ^{137}Cs was considered. The estimated group-level radiation doses have significant individual uncertainty. Due to uncertainties in the quantitative dose estimation quantitative risk estimation was not performed in the current paper. Northern people have a relatively high intake of smoked and salted food and a low intake of fresh vegetables and fruit, but on the other hand, a high consumption of berries. These etiologic factors constitute potential confounders that we were unable to control. Interaction between radiation and dietary risk factors is also a possibility.

Conclusions

The overall cancer incidence in Arctic population cohort was lower than in the general population in Northern Finland. Ethnic Sami people had a lower cancer incidence than the rest of the cohort. No association was observed between the lifetime cumulative radiation exposure from global radioactive fallout and radiation-related cancer risk.

Acknowledgements

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Health impairments in occupational ionizing gamma exposure

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Abstract

Work involving the preparation and assay of radiopharmaceuticals is associated with the highest occupational medical staff exposure from gamma emitters (^{99m}Tc, ^{131}I).

Our health status surveillance, a 5 years follow-up, of 16 subjects (18.11±13.76 years exposure length, 31,3 % easy-smokers) included mandatory national recommendations effect indices: clinical, haematological, cytogenetic micronuclei-MN in peripheral blood. Moreover we performed MN in sputum, oral exfoliated cells, oxidative stress indices (whole blood superoxide dismutase activity – SOD, serum lipoperoxides - Lpox).

During follow-up, 18.7% subjects over 12 years ^{131}I exposure were monitored for occupational thyroidal pathology, other 12.5% for dermic papillose, 56.3% for cardiovascular disorders or allergic rhinitis. Lymphocytes decreased level correlated inversely with exposure length in 31.2% cases. HGB, HCT modifications were associated with reticulocytes response in 12.5% subjects, but inverse correlated with years of exposure (r= -0.42). In 56.3% cases were neutrophil modifications with a 3 years tendency of normal values return at 28.1%. For 6.3% were found numerical and structural blood MN disorders due to workload or no radioprotection rules used. In 36.4% cases MN high level in oral mucous epithelial cells were indirect correlated with exposure length but direct correlated with smoking habit (r=0.58). Sputum type II cytology showed 16.7% ferruginous bodies over 20 years exposure, no correlation with smoking habit. Despite of allowable external exposure limits recorded, chronic exposure increased the SOD activity (148.8±106.6 U/ml) in 12.5% cases and Lpox level (2.4±1.9 µmol/l) at 18.7%, having no correlation with tobacco use.

We emphasized that continuously clinical and bioassay monitoring in nuclear medicine practices can reveal early changes of radiation induced effects. New proposals on national legislation for improving health surveillance of medical staff will be made.

Introduction

Nuclear medicine practices have a greater contribution to the exposure of medical staff involved as compared to other medical uses of radiation. While effective doses for radiologists in diagnostic radiology are at average of 1 mSv/year with increased values for interventional radiology, for medical personnel from diagnostic or therapeutic nuclear medicine is associated with high exposure average doses of 1-2 mSv/year from gamma emitters (^{99m}Tc, ^{131}I), low-linear-energy-transfer (LET) radiation (UNSCEAR 2000, 2006).

Occupational exposure control is usually made for both workplaces– external exposure (dose rate and radioactive contamination) and individual monitoring– individual dose estimation (dosimetry).

Internal dose assessment can be quantified by radioactive measurements (whole body counting, activity excreted in body fluids or in the breathing zone).

Health status assessment of medical staff with ionizing radiation exposure is annually performed by occupational medicine physicians with radio-pathology empowerment and based on national recommendations for effect indices in nuclear medicine exposure (clinical, haematological, cytogenetic micronuclei-MN-in peripheral blood, psychological test, thyroidal hormone determinations) (MPH 2007).

Material and methods

In a 5 years surveillance study of 16 subjects (81.2% females, 31.3% easy smokers, 18.11±13.76 years mean exposure length) we investigated effects indices, mandatory in Romanian national recommendations (MPH 2007): complete physical examination, haematological tests (blood count and blood cells` morphology: HGB, HCT, RBCs, RBC indices, WBC, platelets, reticulocytes, lymphocytes), cytogenetic test – unstable aberrations - number of cells with micronuclei (MN)/ 1000 counted cells in peripheral blood lymphocytes cultures (Popescu 2006) and specialty exams (endocrinology - T_3 triiodothyronine and T_4 thyroxine blood level, ophthalmology, psychology examinations).

Moreover we performed blood oxidative markers as whole blood superoxide dismutase activity (SOD), serum lipoperoxides (Lpox) and MN in sputum and oral exfoliated cells- by mucous cell brushing - a non invasive, not expensive method useful as screening technique.

In the final interpretation, clinical and biodosimetry results were correlated with individual exposure recorded by individual film and electronic dosimeters. Analytical methods used in this study were: t-Student, correlation and descriptive statistics by EXCEL functions.

Results

During follow-up, 18.7% subjects over 12 years ^{131}I exposure were monitored for occupational thyroidal pathology (polinodular goitre) with T_4 levels inversely correlated with exposure length.

Other clinical findings were pigmentation and skin papillose (12.5%), cardiovascular disorders (18.7%) as hypertension and respiratory arithmia, allergic rhinitis (18.7%) or musculoskeletal diseases (12.5%) and minor epileptic strokes (6.2%).

Severe anaemia ($2.98*10^6$ RBC, HGB 8.8 g/dl, HCT 26.8%) were found in one case with 40 years of ionizing radiation exposure.

As haematological marrow cells are the most affected even after short time of exposure, we found the lymphocytes decreased level correlated inversely with exposure length in 31.2% cases.

Trombocytes and neutrophil granulocytes modifications found in 56.3% cases. HGB, HCT and erythrocytes modifications were associated with reticulocytes response in 12.5% subjects, but indirect correlated with years of exposure (r= -0.42).

Genomic instability such as micronucleation were found for 6.3% of subjects, both numerical and structural blood MN disorders related to workload or no radioprotection rules used. In 36.4% cases MN high levels in oral cells were indirect correlated with exposure length but strongly direct correlated with smoking habit (r=0.58). The modifications trend had maintained during 5 years follow up.

Sputum type II cytology in 36.4% of subjects (Babes-Papanicolau classification) showed 16.7% ferruginous bodies over 20 years exposure, but with no correlation with smoking habit.

Despite of allowable external exposure limits recorded, chronic exposure increased the stress oxidative markers such as SOD activity (148.8±106.6) in 18.8% cases and Lpox level (2.4±1.9) at 31.3%, also in our study, having no correlation with smoking.

Discussion

Doses control in nuclear medicine practices implies radioprotection regarding inhalation or ingestion during preparation or injection the radiopharmaceuticals.

In ^{99m}Tc handling the external exposure is important, with high annual dose rates up to 500 mSv especially at hands and fingers, while the average annual effective dose to monitored medical workers averaged over five - years period can vary between 1.04 mSv and2 mSv. (UNSCEAR 2000, 2006).

In our follow-up study for effective dose estimation due to external exposure were monthly recorded the individual dose equivalent Hp (10) which did not exceeded 170 µSv, admitted value for occupational exposure for occupational ionizing radiation exposed personnel. (NCNAC 1996, BSS 1996, ICRP 2007).

There were rare cases of slight exceeding of Hp (10) limit (180µSv/month) in 5.8% of subjects. The cumulative film badge, dose for medical staff were under 24 mSv/year.

Gamma ionizing radiation-induced health effects are not specific and the targeted damages are observed as deterministic effects in hematopoetic system, lungs, thyroid, skin, eye, gonads, embrio/faetus and as stochastic effects by direct DNA damage or delayed effects – carcinogenesis (Little 2000), potential for biological effects expressed within one or two cell generation radiation risk (Morgan et al.1996, Little 2000, Ward 1999, 2002).

We focused to monitor in hematopoetic system the number of lymphocytes, with knowing tendency of decreasing immediately after exposure. In 32.1% cases the decrease of lymphocytes number (95% CI: 29.56-37.68) % were inverse correlated with exposure length (r=-0.35) (fig.1).

Platelets number had no significant changes in all years of monitoring period (fig. 2).

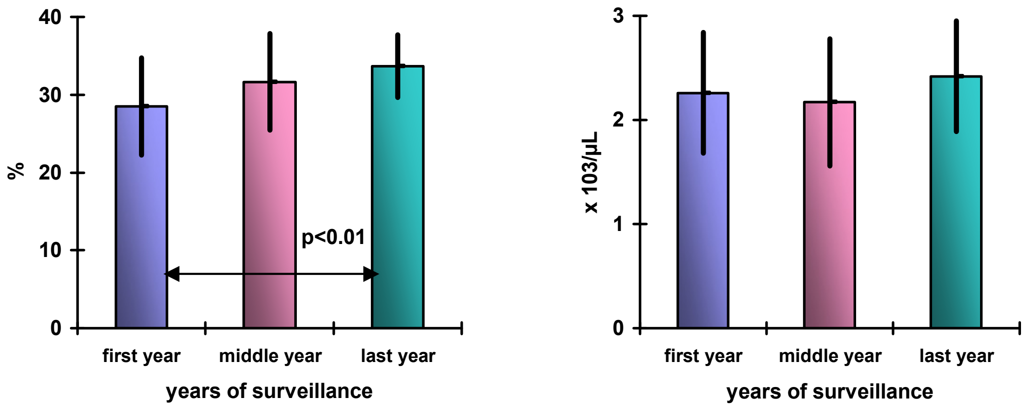

Fig. 1. Lymphocytes mean values in surveillance period.

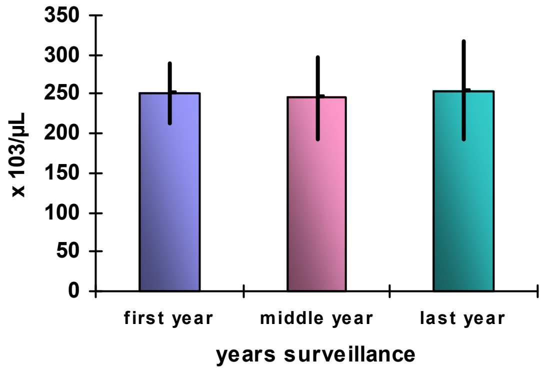

Fig. 2. Platelets mean values in surveillance period.

Numerical modifications (95%CI: 49.60-61.22) % - neutrophil granulocytes immunostimulation / immunosuppressed were found in 56.3% cases correlated inversely with occupational length (r=-0.31) and smoking habit for easy smokers: Brinkmann index under 200 (r=-0.28), statistically non-significant (p>0.05). We noticed at 28.1% subjects the trend to normal values recovery after 3 years of ionizing radiation exposure (fig. 3).

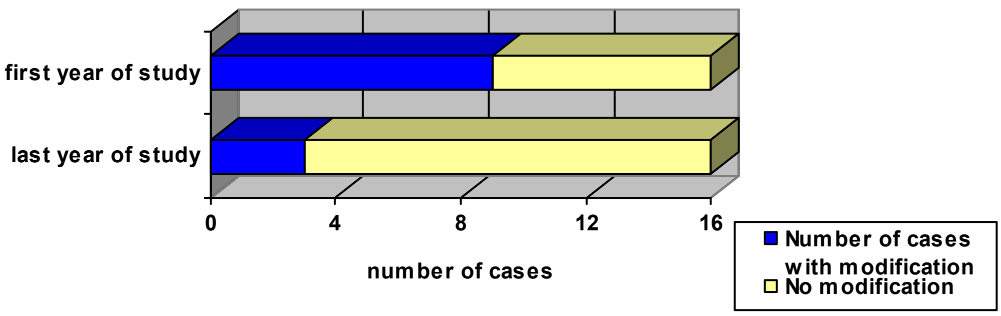

Fig. 3. Number of neutrophil granulocytes distribution in surveillance period.

Erythrocytes, which smoothly decrease after weeks of exposure (UNSCEAR 2000, 2006), in our study this trend was maintained in same values with no significant variations (fig. 4).

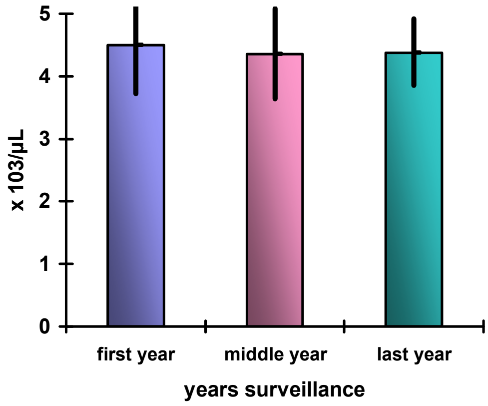

Fig. 4. Erythrocytes mean values in surveillance period.

Also HCT (95% CI: 34.42, 42.74) % correlated inversely with exposure length (r=-0.42), anaemic syndrome being observed in 12.5% of cases, one severe type. HGB (95% CI: 11.43, 14.59) g%, HCT and erythrocytes modifications were associated with reticulocytes response (95% CI: 6.06, 22.96) ‰ in 12.5% subjects, but significantly inverse correlated with years of exposure (r= -0.42).

Eosinophils granulocytes (95% CI: 0.62, 5.56) % in 12.5% of cases had a constantly numerical increasing trend over a 5 years-surveillance but had significantly positive correlation only with cigarette smoking for moderate smokers: Brinkmann index over 220 (r=0.79, p<0.01).

At subjects over 12 years ^{131}I exposure were monitored, clinical and therapeutic, an occupational multinodular goitre with hyperthyroidism, T_3 and T_4 levels trend were inverse correlated with exposure length. (r=-0.47).

Non targeted effects, like bystander effects (Yier 2000, Belyakov 2001), genomic instability (Emerit 1994, Morgan 1996, Wright 1998, 2000), adaptive response (Wolf 1998) are specific significant at low doses and this evidence is a new model for classical theory of radiation biology (Baverstock 2005).

We performed cytogenetic tests based on the assay of unstable aberrations such as MN in peripheral lymphocytes, sputum and oral exfoliated cells, visible in cellular cytoplasm at the first post-irradiation mitosis (Muller 1996).

These methods available at present for our study seemed useful to estimate the individual effects in moderate – low LET radiation doses. For 6.3% subjects were found numerical and structural blood MN disorders due to workload, meaning an acute recent exposure. Blood MN changes had no correlation with smoking habit but had an inverse correlation with occupational exposure length (fig. 5).

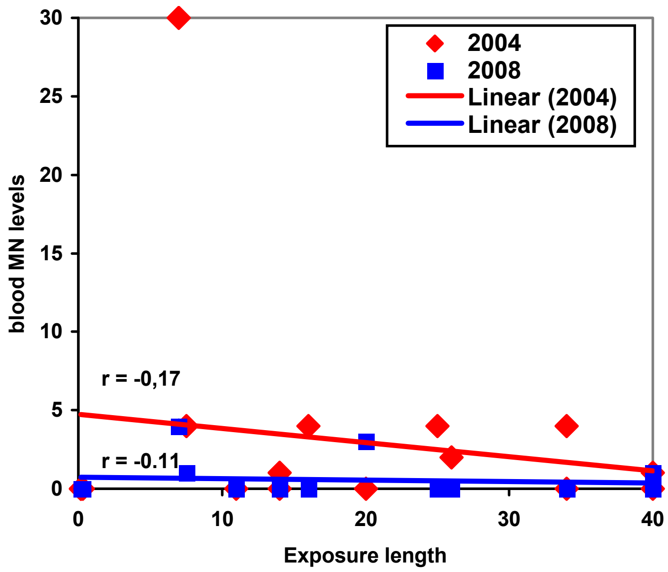

Fig. 5. Correlation between blood MN and exposure length during study surveillance.

Also, we observed a direct correlation of cases with blood/oral MN level modifications during 5 years surveillance (r=0.44). In 36.4% cases MN high numerical level in oral cells were inverse correlated with exposure length (r=-0.44) but significantly positive with smoking habit (r=0.58) (fig. 6).

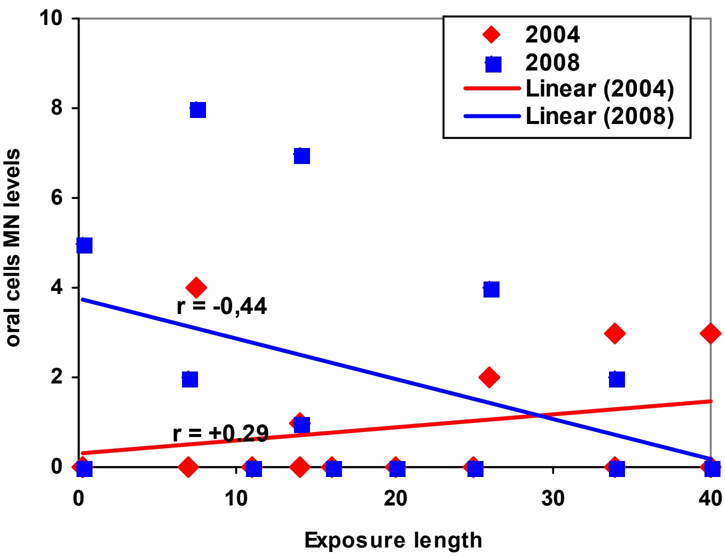

Fig. 6. Correlation between oral cells MN and exposure length during study surveillance.

Sputum type II cytology in 36.4% of subjects showed also 16.7% ferruginous bodies over 20 years exposure, but with no correlation with smoking habit. It seems that non-uniform/not constantly medical staff monthly exposure (for gamma generators) and adaptive mechanisms are responsible for the indirect correlation between length of exposure and MN alterations either in sputum or oral cells. Recently it was demonstrated that chromosome instability in cells can develop specific cell variants more resistant to radiation exposure (Limoli 2001).

The bystander effect is cell type dependent and energy metabolism or reactive oxygen species (ROS) (Mothersill 2000) are involved in mediation of this kind of response. (Emerit 1994, Pollycove 1998, 1999, Lyng 2001). Antioxidant enzymes SOD/Lpox annihilate ROS, stress oxidative markers disturbances indicating a continuously oxidative processes and cellular stress aggression of health status caused by ionizing radiation exposure (Kłucinski et al. 2008).

For medical personnel involved in our study SOD activity (95% CI: 42.25-255.51) U/ml had increased levels in 12.5% of cases with no correlation with exposure length or tobacco use (Kłucinski et al. 2008). Lpox varied (95% CI: 0.4-4.38) µmol/l from 18.7% high levels directly with occupational exposure, to 6.2% decreased level after 7 years of occupational exposure at smoker personnel (fig. 7).

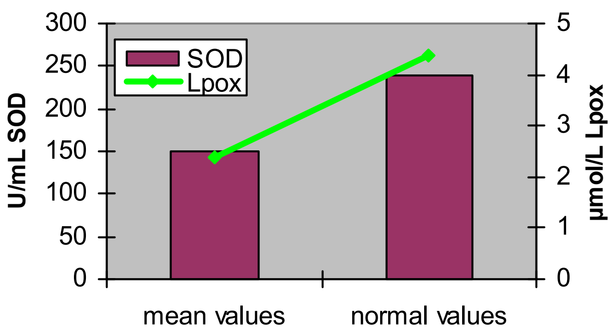

Fig. 7. Comparison trend of SOD and Lpox mean values in study surveillance.

An accurate assessment of oxidative stress in biological system is hard to perform, because a comprehensive surveillance must include measurements of all antioxidants, but that would be expensive, long time processing and technically difficult (Olisekodiaka MJ et al. 2009).

Conclusions

We emphasized that continuously clinical and bioassay monitoring in nuclear medicine practices – haematological, cytogenetic and oxidative stress status - can reveal early changes of radiation-induced effects.

Long term exposure to gamma-rays changes the antioxidant response of medical staff, having no correlation with cigarette smoking.

New proposals (such as MN in oral mucous epithelial cells) on national legislation for improving health surveillance of health care staff will be made.

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Spatial correlations of microdosimetric parameters and biological endpoints associated with radon inhalation

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Abstract

Radon and its progenies can be inhaled and deposited in the airways. Many of the decay products of the radioactive radon isotopes are short lived alpha, beta and gamma active atoms interacting with the epithelial tissue after their deposition and decay in the lung. Some of the radon and thoron decay products are short lived alpha-emitters and can transmit large amount of localized energies to the surrounding cells causing cellular and tissue damages. Since these interactions take place at a microscopic scale, the averaging of radiation dose to the whole body or the whole lung seems not to be fully appropriate for the quantification of the health effects. The aim of this study was to reveal the exact spatial correlations of different microdosimetric quantities and biological endpoints in the epithelium following the deposition and decay of short lived radon progenies. For this purpose complex numerical methods were developed. Air and particle transports in reconstructed three-dimensional airways were simulated. Particle deposition coordinates were recorded and alpha tracks generated. The cell nucleus architecture of the epithelium was reconstructed and interactions of alpha tracks with the nuclei were modelled. Our results demonstrate that in some clusters of nuclei the probabilities of single and even multiple hits are high even at low levels of macroscopic exposure. For the cells located in the deposition hot spots the probability of cell killing and transformation is more than one order of magnitude higher than the corresponding values assuming uniform surface activities.

Introduction

Radiation protection directives and regulations are based on our knowledge about health effects of radiations. Although by the accumulating experience, development of experimental and computational techniques and the emergence of new, multidisciplinary approaches a series of questions have been answered, some others are still to be clarified. One of these challenging tasks regards the biological response triggered by low dose densely ionising radiations. Densely ionizing radiation produces a unique type of damage in which multiple lesions are encountered within close spatial proximity. The degree of damage clusterization can be further accentuated by the nonuniform radiation source distributions. Such a nonuniform surface activity distribution is produced by the inhaled radionuclides deposited within the airways. The site specificity of radioaerosol deposition in the lungs has been demonstrated both experimentally and by computational methods. However, the consequences of the inhomogeneous radionuclide deposition at cellular and tissue level are not fully explored to date. Although a comprehensive experimental study on the effect of spatial distribution of hit cells on different biological endpoints is still missing, it is clear from the existing experimental results that the spatial distribution of activity and related cellular alpha hit distributions are essential for the induced biological damage.

This study proposes to quantify the spatial distribution of the cell nucleus hits and the related biological endpoints, such as cell killing and cell transformation. Knowledge of distances between hit, killed and transformed nuclei instead of providing only their number distribution or simply assuming that they are uniformly distributed may be an important step towards a qualitatively better risk assessment. Elucidation of spatial correlations is essential especially for low doses, where neither epidemiology nor the experiments could establish a statistically convincing plausible dose-effect relationship.

Methods

Based on detailed histopathological studies (e.g. Saccomanno et al. 1996) the radon induced carcinomas arise preferentially in the upper lobes, especially in the right upper lobe. Therefore, in the present study a representative central airway segment consisting of a single bronchial bifurcation was considered (Figure 1). The 3D geometry model was constructed based on the numerical technique described by Hegedűs et al. (2004). To get the expected locations of the depositing radioactive particles it is necessary to track them within the airways. Since particle trajectories are strongly influenced by the airstreams, only a coupled (particle-air) approach can yield appropriate results. Air- and particle transport within the model airway bifurcation was simulated by the FLUENT CFD (computational fluid dynamics) code (FLUENT User's Guide 2001). Breathing parameters characteristic of light physical activity were selected. Radiation exposure conditions were characteristic of the former Czech uranium mines derived from Tomasek et al. (2008) publication. Inhaled particles were tracked until they deposited in the model airway or left the geometry. Deposition coordinates were recorded for further processing. Inhomogeneity of the deposition pattern was quantified by the help of activity maps. Activity maps were constructed by scanning the surface of the airway bifurcation by a regular triangle shaped surface element (surface area = 0.43 mm^2) and assigning to each patch a relative activity density value. Activity density is given by the sum of the activities provided by the individual radionuclides deposited on a patch divided by the area of the patch (computational cell). The ratio of the local activity density to the average activity density yields the relative activity density.

In order to model the interaction of the radiation with the radiosensitive epithelial cell nuclei, alpha tracks were generated and a three-dimensional bronchial epithelium was constructed (see also Szőke et al. 2008). Alpha-tracks were simulated as straight lines with randomly selected directions. Their lengths (ranges) were derived based on the initial kinetic energies of alpha particles and published stopping power functions for air and tissue (ICRU 1993). Near wall and far wall alpha tracks were distinguished based on their direction to the nearest surface element. By definition, tracks entering directly into the tissue are near wall alpha tracks, whereas tracks first entering the airway lumen then penetrating into the epithelium are far wall alpha tracks. The reconstruction of the three-dimensional bronchial cell nucleus structures is based on the histological data of Mercer et al. 1991. In this work, basal, indeterminate, ciliated, preciliated, goblet and other secretory cell nuclei types were distinguished. Hit probabilities and cell killing probabilities were computed for each type of cells, including both proliferative and terminally differentiated cells. However, transformation probabilities were computed only for the radiosensitive basal and goblet cells (see Figure 1). For the simulation of cell killing and cell transformation, results of the *in vitro* cell irradiation experiments from the open literature were used.

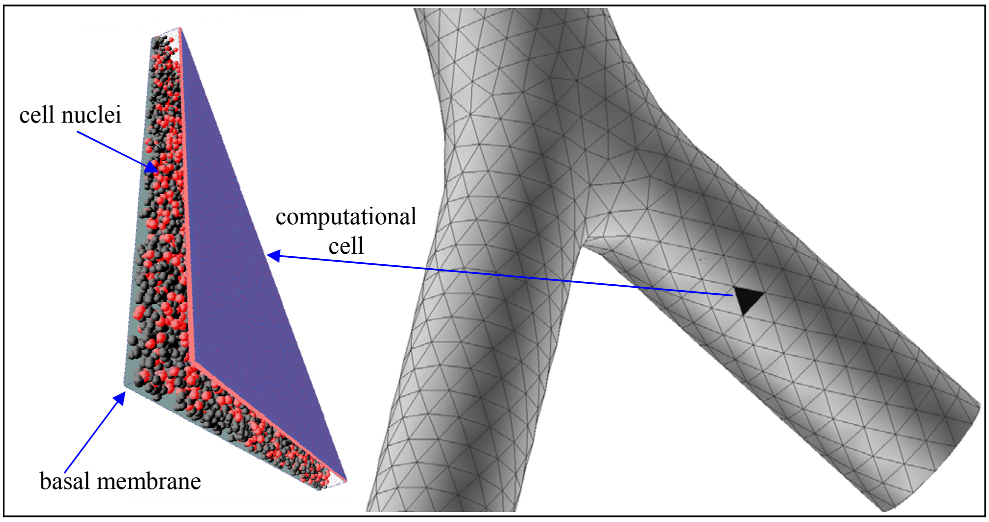

Fig. 1. Discretized (meshed) model airway bifurcation geometry and reconstructed epithelial cell nuclei.

Results and discussion

Figure 2 depicts the deposition patterns of attached (red dots) and unattached (blue dots) radon decay products corresponding to 0.01 WLM. As expected, particles deposited nonuniformly along the airway bifurcation, with the dividing spur as a preferential deposition site. The deposition efficiency, defined as the ratio of the number of deposited to the number of entering particles, was 0.46 % and 16.2% for the attached and unattached radon daughters, respectively (tracheal airflow rate was 50 l/min). Furthermore, the left panel of Fgure 2 demonstrates that the deposition of the molecular (unattached) radon progenies is less inhomogeneous than that of the attached ones. Quantification of the inhomogeneity of deposition distribution is demonstrated in the right panel. In the most exposed patch, the activity is roughly 40 times higher than the activity averaged over the whole bifurcation geometry. This suggests that cells located within this patch can be at considerable risk even if the average activity is not so high. To verify this, hit probability distributions were computed. The scanning technique presented at the description of activity maps (Methods section) was used to count the number of hits for each computational cell, then the distribution of hit numbers per patch was plotted for the 0.25 WLM exposure level (Figure 3, left panel). To highlight the importance of considering realistic deposition distributions, the hit number per patch distribution in case of uniform activity distribution is also presented for the same exposure level (Figure 3, right panel).

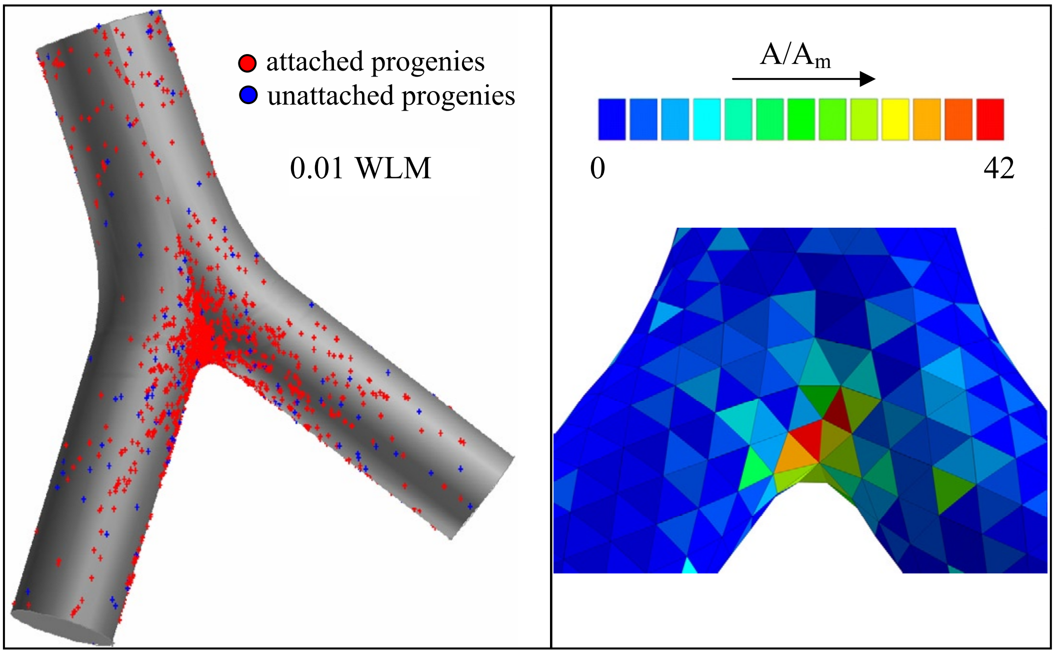

Fig. 2. Deposition pattern of attached and unattached radon progenies assuming an exposure level of 0.01 WLM (left panel) and activity density map in the vicinity of the carinal ridge normalized to the mean activity density (right panel).

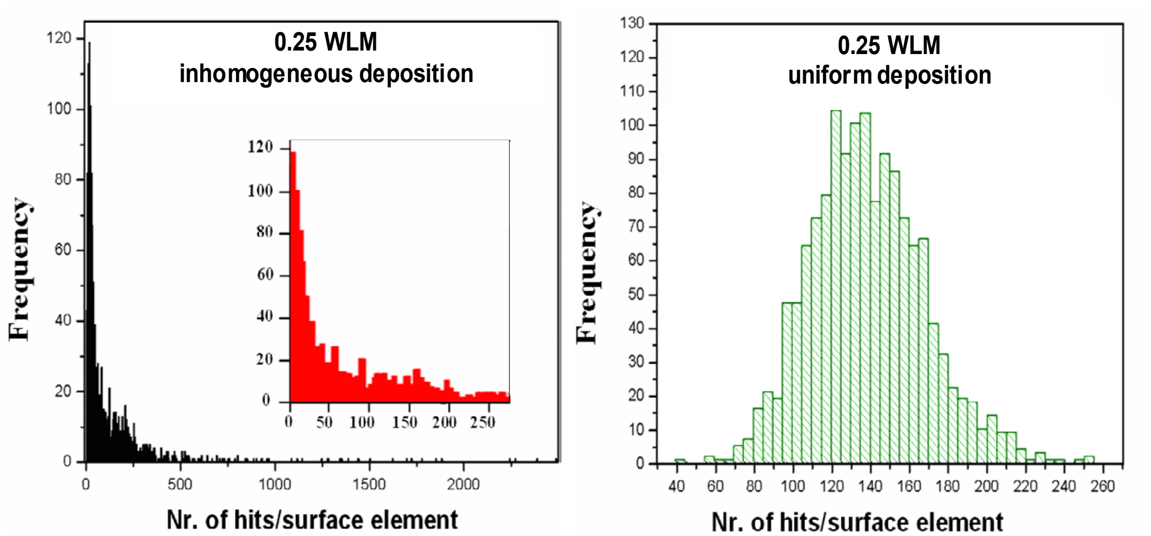

Fig. 3. Distribution of the number of hit cell nuclei located in a pre-specified surface element assuming inhomogeneous (left panel) and uniform (right panel) radioaerosol deposition distributions corresponding to 0.25 WLM radiation exposure.

Figure 3 reveals that the two hit distributions are quite different. Considering realistic deposition distribution while most of the patches receive no hit or only a small number of hits, a few patches receive hundreds and even thousands of hits. In contrast assuming uniform deposition, like most of the current models, the number of hits per patch is closely distributed around a mean value of about 130. This means that for the same exposure the degree of clusterization of the hit cells is much higher in case of inhomogeneous deposition compared to the case of uniform deposition. To check whether this observation is valid for other exposure levels, as well, the mean and maximum number of hits per patch were plotted as a function of WLM. Figure 4 demonstrates that the inhomogeneity in the distribution of hit cell nuclei per patch is present at any exposure level in the low dose range. This result suggests that although the proportion of cell nuclei receiving hits is quite low in the low dose range, there are areas, where the number of hit nuclei is very high and thus the most exposed cells are close to each other. The above fact is usually neglected by current risk models, which operate with average hit numbers.

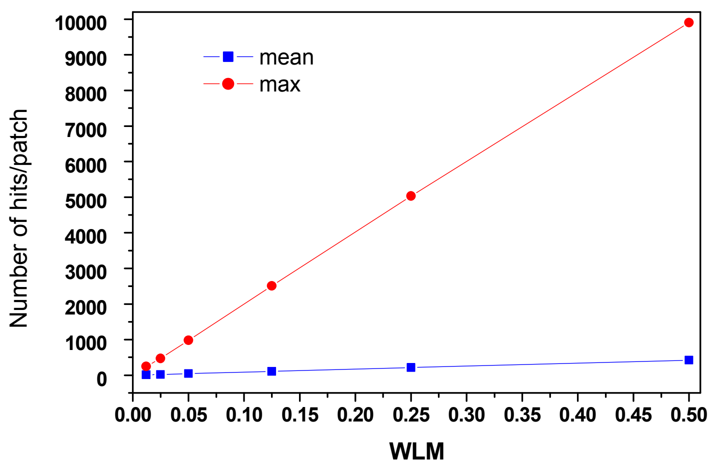

Fig. 4. Average and maximum number of cell nucleus alpha hits per patch (computational cell) as a function of working level.

It is worth mentioning that computations related to Figures 3 and 4 do not account for multiple hits. Taking into account the multiple hits the inhomogeneity becomes even more accentuated. Location of the cell nuclei receiving multiple hits are illustrated in Figure 5. The figure demonstrates that assuming realistic particle deposition patterns, the probability that a cell nucleus located in the carinal ridge of the airway bifurcation receives multiple hits is much higher than it is for a nucleus from the cylindrical parts of the bifurcation. In addition, the mean value of the distances of cell nuclei receiving multiple hits to their multiple hit neighbours (d_m) is also significantly lower than the same parameter characteristic of uniform deposition, indicating that cells at direct risk are closer to each other when assuming realistic deposition scenarios. To verify whether the finding holds for other exposure levels, as well, the probability that a cell nucleus receives at least one and at least two hits was calculated for a range of WLM values (see Figure 6). Although assuming uniform radionuclide deposition, in the considered working level interval, the probability of at least one and at least two hits increases linearly with the increase of the exposure level, there is little chance for a nucleus to receive at least one hit below 6 WLM and the probability of more than two hits is almost zero. In contrast, in the hot spot of the inhomogeneous deposition, the curves are nonlinear and the probability of at least one and at least two hits can be high even at low exposure levels, suggesting that for these regions of the tissue the “low dose problem” is actually a” high dose problem”.

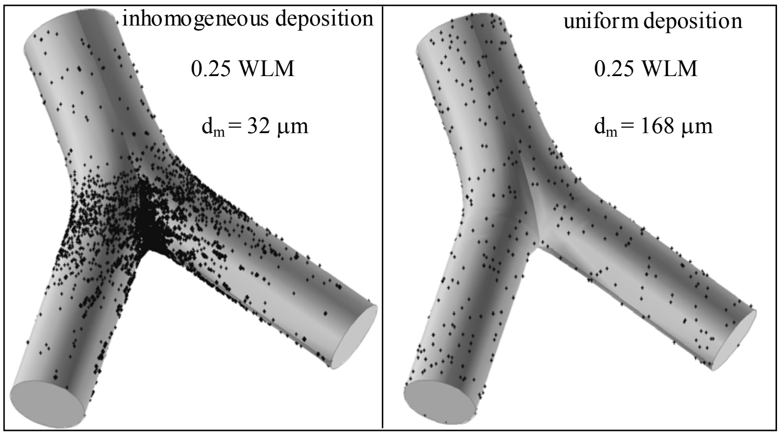

Fig. 5 Locations of cell nuclei receiving multiple hits in case of realistic nonuniform (left panel) and uniform (right panel) radionuclide surface distributions. d_m denotes the mean value of the distances between the nuclei receiving multiple hits and their nearest neighbours receiving multiple hits.

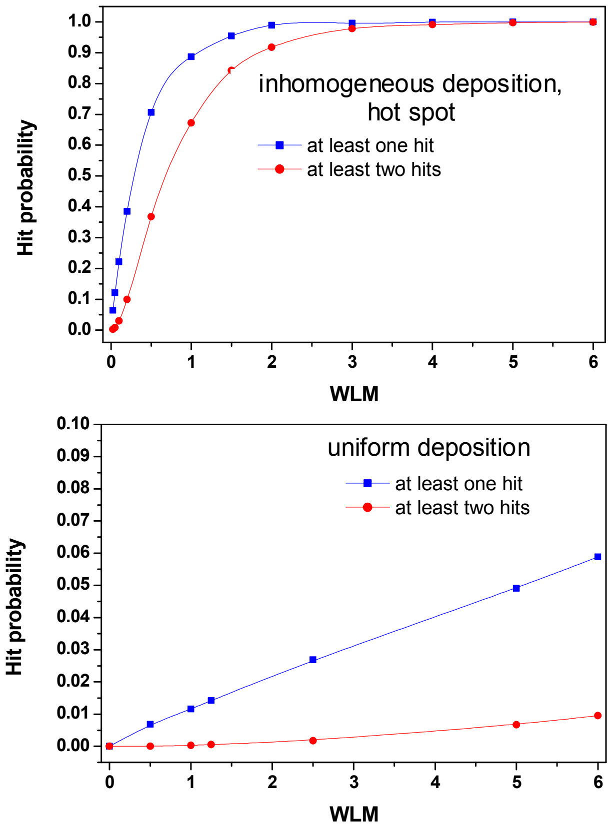

Fig. 6. Probability that a cell nucleus is hit at least once or twice as a function of WLM. The upper panel refers to nuclei in the deposition hot spot of the inhomogeneous deposition pattern, whereas the bottom panel presents the case of uniform deposition.

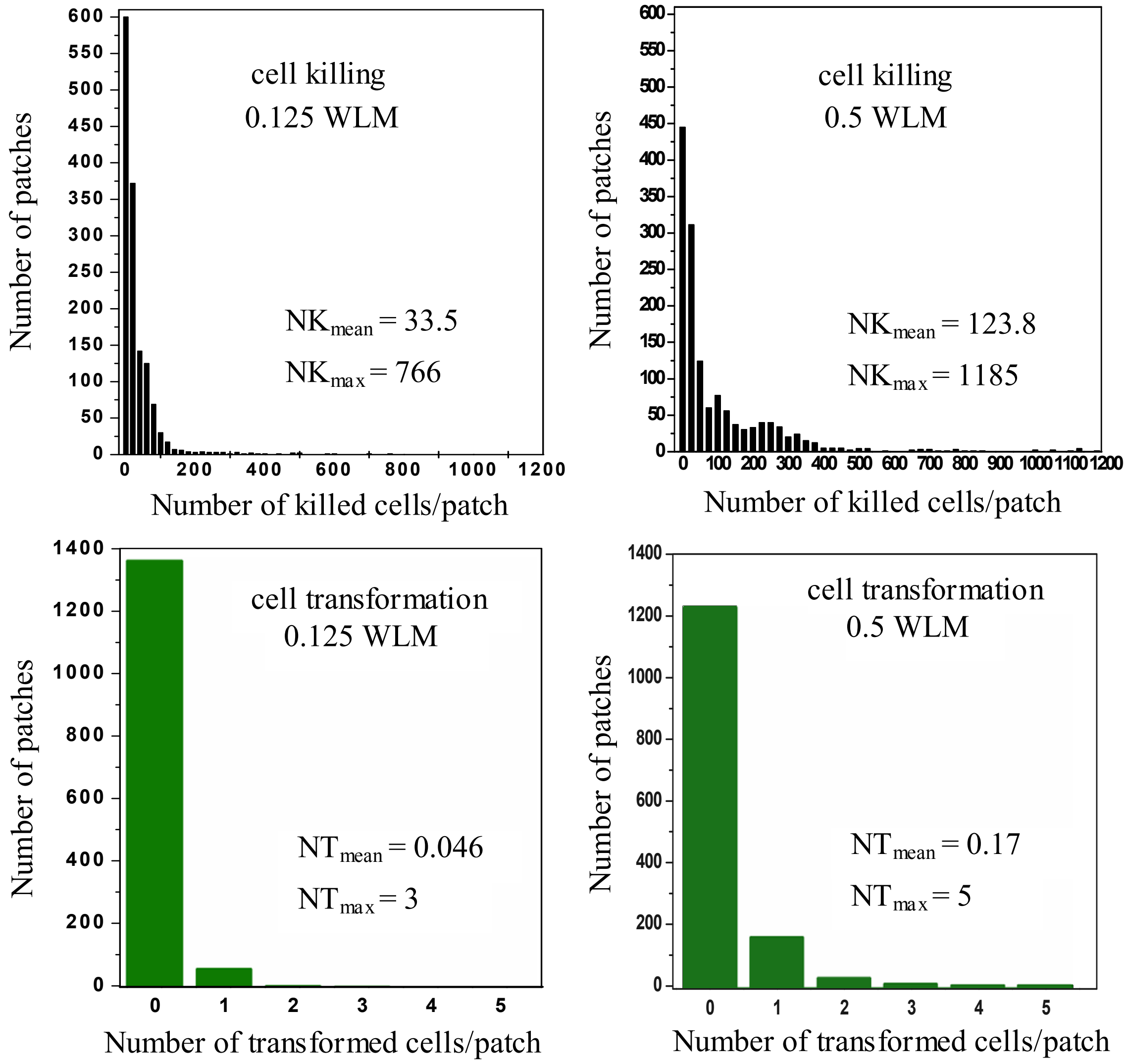

Fig. 7. Distribution of the number of killed (upper panels) and transformed (bottom panels) cells per patch at 0.125 WLM (left panels) and 0.5 WLM (right panels) exposures, assuming inhomogeneous deposition patterns. NK_{mean} – mean number of dead cells per patch; NK_{max} – maximum number of dead cells per patch; NT_{mean} – mean number of transformed cells per patch; NT_{max} – maximum number of transformed cells per patch.

To reveal the cell biological consequences of the nonuniform deposition, cell death and cell transformation distributions were computed. The results of these simulations are presented in Figure 7. The figure reveals that following the tendencies concerning the microdosimetric parameters illustrated in Figs. 3-6, spatial distribution of the biological endpoints is also heterogeneous. While the average number of inactivated cells in a patch is 33.5 at 0.125 WLM, it reaches 766 in the hot spot. The corresponding values for 0.5 WLM are 123.8 and 1185, respectively. It has been demonstrated by different experiments that the transformation frequency due to ionising radiation is extremely low in the low dose range (e.g. Sawant et al. 2001). For instance, according to Miller's microbeam irradiation experiments performed on CH310T1/2 cells (Miller et al. 1999) only one cell out of ten thousand surviving cells receiving exactly one hit will be transformed. Applying the computational schemes described in the methods section, the average number of transformed cells on a patch is 0.046 at 0.125 WLM and 0.17 at 0.5 WLM. However, our computations demonstrated that even at low exposures and hence low average transformation probabilities, there were three transformed cells at 0.125 WLM and 5 at 0.5 WLM in the hot spots. It is worth mentioning that one patch corresponds to about 10^4 cells (or nuclei). The existence of significant number of cell death and cell transformation events in restricted areas suggests that, extended detrimental effects might occur even in the low or intermediate dose range.

Conclusions

In this study, spatio-temporal distributions of microdosimetric parameters and related biological endpoints were determined in the central bronchial airways by complex computational methods. As expected, the deposition patterns of the inhaled attached and unattached radon progenies are nonuniform within an airway bifurcation with primary activity hot spots located at the dividing spur (carinal ridge). Cell nucleus hits of radon progeny alpha particles were also heterogeneous in space, although to a slightly lower extent than the distribution of the deposited radionuclides. The probabilities of single and multiple hits were quite high in the deposition hot spot even at low doses and increased in a nonlinear manner with increasing exposure level. The mean distances between hit nuclei were significantly reduced for hot spots compared to the corresponding mean distances in case of uniform deposition. Our computations revealed high cell killing rates in the deposition hot spots at low macroscopic doses. Oncogenic transformation probabilities are also significantly higher at the carinal ridges where transformed cells are closer to each other. Present results may help in the elucidation of different aspects concerning the biological consequences of radio-aerosol inhalation and may serve as inputs for future risk assessment models.

Acknowledgements

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Cardiovascular risk after low-dose radiation exposure

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Abstract

Introduction: There are considerable uncertainties concerning health effects of low doses of ionising radiation on heart. The Life Span Study of the Japanese atomic bomb survivors shows excess radiation-associated risk for cardiac disease even at doses of 0.5 Gy.

Aim: We have used both a human endothelial cell line and a mouse model to study proteomic alterations after low-dose irradiation.

Methods: The human endothelial cell line Ea.hy926 was irradiated with 0.2 Gy gamma rays with two different dose rates and the cells were harvested 4h and 24h after the irradiation. The proteome changes were analysed using 2 DE-DIGE techniques. Using a C57/Bl6 mouse model functional and proteomic alterations in mitochondria isolated from irradiated and sham-irradiated murine hearts 4 weeks after heart-focussed irradiation (0.2 Gy, 2 Gy X ray) were analysed.

Findings: In the endothelial cell line, out of more than fifty protein spots that showed significant alterations in their expression 22 proteins were identified. Among the pathways affected by the low-dose ionising radiation are Ran and Rho/Rock pathways, stress response and glycolysis. In the mouse model, no differences between sham- and irradiated cardiac mitochondria were found in swelling, respiratory coupling and production of ATP. However, we could identify significantly increased ROS formation in cardiac mitochondria 4 weeks after the exposure to 2 Gy ionising radiation.

Conclusions: The immediate response to low-dose ionising radiation in the human endothelial cell line includes alterations in the expression of small GTPases (or their regulators) such as Ran and RhoA, the expression of which is known to be regulated by the production of reactive oxygen species (ROS). Similarly, we find in the long-term response of irradiated murine heart mitochondria an increased production of endogenous ROS. We conclude that changes in the oxidative stress may be an important factor in the development of radiation-induced cardiac disease.

Modelling and experimental studies on adaptive response

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Abstract

The ability of a low "priming" radiation dose to decrease the cell response to a subsequent higher "challenging" dose is termed adaptive response (AR). The main proposed mechanisms to explain AR are: increased efficiency of DNA repair, induction of anti-oxidant enzymes, alteration of cell cycle progression, changes in chromatin conformation. The model proposed by Curtis in 1986 that considers a modulation of the efficiency of DNA repair activity and of the level of anti-oxidant enzymes, starting from the framework of lethal-potentially lethal (LPL) model, was extended with the inclusion of the dynamical variables representing the efficiency of repair, the levels of radiation induced radicals and of anti-oxidant enzymes. In the simulation code particular attention was devoted to the induction of anti-oxidant enzymes as adaptive response mechanism, even if the more relevant mechanism remains the modulation of the repair efficiency, by which the cells processes the initial radiation damage. Furthermore, in order to describe different radiation qualities, the weight of the two factors (i) production of direct damage and (ii) production of free radicals have been made variable in the model. Our model is able to describe the protective effect of a priming dose. Moreover, in agreement with the literature data, the simulations show that the AR happens in a given priming dose and priming dose rate ranges only, and requires at least 4 hours to develop. In order to get more insights on the role of cell-cell communication as factors affecting the AR, experimental studies have been performed using sparse or confluent AG1522 cell monolayer. The results obtained after gamma-irradiation suggest that cell density is a crucial factor for observing an AR. The possibility of an AR induced as bystander effects and as a function of radiation quality is under study.

The α-particle irradiator at the ISS: a useful tool for low dose/dose rate studies

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Abstract

An α-particle irradiator has been developed for radiobiological studies at the Istituto Superiore di Sanità (ISS), Rome, Italy. It consists in a 200 mm diameter stainless steel chamber that can be equipped with alpha sources of different activities, allowing modulation of dose and dose-rate.

In the present configuration the irradiator, equipped with ^{244}Cm or ^{241}Am sources, provides a useful facility for irradiation of cultured mammalian cells with α-particles at dose rate in the range of 1-100 mGy/min, with spatial variations of less then ± 7%. Moreover, for both sources the photon doses calculated at the cell entrance are negligible compared to the α–particle dose. A further important feature of the irradiator is that the seal necessary to keep the helium gas inside the chamber is provided by the Mylar® base of the sample holder. This solution decreases energy degradation and eliminates the problems related to the estimation of the air layer between the exit window and the sample holder. Mylar®-based Petri dishes with different irradiation area were designed that can house permeable membrane inserts, mimicking the same geometry of commercial cell culture insert companion plates. Due to the limited residual range of α-particles, this feature is particularly valuable for co-culture experiments aimed at investigating bystander effects.

The small size allows the irradiator to be easily positioned into a standard CO2 incubator, avoiding use of "*ad hoc*" and separate devices for temperature control and for keeping the cells in the proper air/CO_2 mixture. This important feature makes it possible to carry out long term irradiation and/or post-irradiation incubation of cells in physiological conditions.

The irradiator has been already successfully used to continuously irradiate confluent primary human fibroblasts for up to 14 days as well as to investigate cellular and molecular end points in directly hit and in bystander primary human fibroblasts.

Using of radioprotectors and antidotes during the external radiation exposure and radionuclide incorporation

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Abstract

We have studied experimentally on mice the compatibility of radioportectors (indralin and riboxin) and antidote drugs (potassium iodide and ferrocin) during the external gamma ray exposure and incorporation of ^{131}I and ^{137}Cs. We have shown that during combined action of both external and internal radiation factor radioprotectors and antidotes do not affect the efficiency of each other. In the absence of antidotes the radioprotectors alternates the exchange kinetics of the radionucildes. Application of the indralin alone increases the concentration of ^{131}I in the thyroid and kidneys and ^{137}Cs in the liver. Application of riboxin alone decreases the concentration of ^{137}Cs in organs and tissues at early times. The accumulation rate and concentration values for radioniclides in different organs depends on the time of radioprotector application.

Low-dose irradiation delays neuronal differentiation during early embryonic brain development

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Abstract
Recent interest into the effects of low-dose ionizing irradiation on the human brain has been rather limited partly due to the fact that the adult brain (postmitotic neurons) is very insensitive to radiation damage. However, epidemiological studies on individuals who were exposed to ionizing radiation *in utero* after the atomic bombings in Hiroshima and Nagasaki, showed an increased incidence of mental retardation and behavioural defects. These defects were most outspoken when the irradiation occurred between weeks 8 and 15 of gestation, which is a critical time for human embryonic neuronal development. This suggests a higher sensitivity of neurons during their early differentiation.

Initial experiments in our laboratory showed that young adult mice that had been irradiated with doses (1 Gy) *in utero* at day 12 of gestation (E12) and at lesser extent with lower doses (0.2 Gy) at day 11 of gestation (E11) suffered from memory and learning defects as assessed by Morris Water Maze testing. In order to identify possible pathways involved in neurogenesis that could be affected by low-dose irradiation during early neuronal development, we irradiated pregnant mice at day 10 (E10) and 11 of gestation at 0, 0.1 or 0.2 Gy. Three hours after irradiation, embryonic brains were isolated for total RNA extraction and subsequent microarray analysis using the MoGene 1.0 ST Arrays (Affymetrix, USA).

In line with earlier observations, our analysis revealed upregulation of several genes involved in p53-responsive pathways in a dose-dependent way in both E10 and E11 irradiated embryos, indicating that these pathways are important for the response of the embryonic brain to low doses of irradiation. Furthermore, we observed that low doses of irradiation attenuate many functions of the normal physiological changes in gene expression, and in particular functions related to neurogenesis and synaptogenesis. This suggests that normal neuronal development may be delayed by low-dose irradiation. A significant amount of the affected genes involved in neurogenesis were predicted targets of the neuronal gene silencer RE-1 silencing transcription factor (REST) of which the expression in the embryonic brain decreases during differentiation. Interestingly, expression of REST was increased when E11 brains were

irradiated both at 0.1 and 0.2 Gy, while the expression of the REST target genes dose-dependently declined.

Together, our data indicate that the cognitive defects observed in young adult mice that had been irradiated *in utero* may be related to delayed neuronal differentiation due to a repression of neuronal genes following increased expression of the neuronal silencer REST. The exact mechanism behind the radiation-induced stimulation of REST remains to be elucidated.

This work is financially supported by the NOTE project EC contract: FI6R-036465.

EU GENRISK-T project: Thyroid cancer risk in the Balb/c mouse strain after exposure to low doses of X-rays

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Abstract

Thyroid cancer is one of the malignancies that are induced by radiation exposure as supported by epidemiological studies of different radiation-exposed groups such as: the survivors of atomic bombing in Japan, Marshall islanders exposed to nuclear test fall out and children undergoing head or neck radiotherapy or accidentally exposed to radiation like in Chernobyl accident. In these two last groups, the age was an important determinant of the radiation-exposure outcome.

Besides the age at the time of exposure, genetic susceptibility is one of the factors that can interfere in the development of thyroid cancer. The aim of the FP6 EU GENRISK-T project is to define the genetic component influencing the risk of developing thyroid cancer after exposures to low dose radiation as there is currently no accurate model of the dose response curve for thyroid cancer at low doses. To shed light on the cause-effect relationship between low radiation exposure, thyroid cancer and genetic background we used a mouse model with low sporadic rate of developing thyroid cancers. Such studies of genetic susceptibility would be impossible to conduct in humans due to the significant incidence of sporadic thyroid cancers and to the presence of "confounding" non-radiation induced tumours.

Balb/c mice were irradiated at 12 weeks of age with either low (25cGy, 50cGy, 100cGy) or high (4Gy) X-ray doses and sacrificed 4 hours, 9 or 18 months post-irradiation. Thyroid cancer was assessed via histopathology examinations to detect any morphological alteration in the thyroid tissue structure. In addition, we used immunohistochemistry to detect the level of some proliferation markers (PCNA and Cyclin-D1) that would be indicative of the proliferation status of the thyroid tissue.

Based on histopathology examinations of thyroid tissue sections from the 9 month mouse group, we found that no mice had developed thyroid tumors even at the highest doses. However, at a dose of 1 Gy, we found two specific cases, one of microfollicular hyperplasia and the other one of acute infiltrate with granulocyte. Regarding the immunolabeling with Cyclin-D1 and PCNA, we found a dose-dependent increase of these molecules significant from 25cGy onwards compared to the controls. Examination of thyroid tissue sections from the remaining mouse groups (4 hours and 18 months) is currently under progress. In parallel, high-throughput microarray technology is currently performed.

This work is financially supported by the EU Euratom programme (GENRISK-T : contract FP6-36495).

Identification of health risks in workers staying and working on the terrains contaminated with depleted uranium

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Abbreviations:

L-ALP	Leukocyte alkaline phosphatise
MCV	Mean corpuscular volume
IU	international units
DU	depleted uranium
NU	natural uranium
TLD	thermo luminescent dosimeters
DOE	duration of occupational exposure
NATO	North Atlantic Treaty Organization
FRY	Federal Republic of Yugoslavia

Abstract

Objectives: This study investigated health risks in workers permanently or occasionally residing in the contaminated terrains by low ionizing radiation doses originated from ammunition containing depleted uranium (DU).

Methods: The studied population had been composed of two test groups (T-I, T-II) occasionally .exposed to DU, and two referent (R-I, R-II) groups not exposed at any time to DU. All of them had been evaluated for the following: complete clinical examination and blood count, presence of immature forms and blasts, leukocyte alkaline phosphatase activity and cytogenetic tests (lymphocyte karyotype and chromosomal aberrations). The probability of onset of the characteristic complete biomarkers – chromosomal aberrations, had been specially analyzed using logarithmic function of the Poisson regression.

Results: The difference between the groups was found in potential exposure to DU, which was not found in the reference group R-I and R-II being probable in the test group T-I and T-II. Estimated function of density of probabilities of Poisson distribution of the chromosomal aberrations in the test group T-II was drastically different from the corresponding distribution of the referent group R-I and to the somewhat lesser extent from the group R-II; Wilcoxon test exactly confirms presence of significant difference between the reference group R-II and test group T-II, $p<0.05$.

Conclusion: Damages of chromosomes and cells, had been used for the relative radiation risk assessment, were highest in the test group T-II of workers additionally occupatioanlly exposed to DU. Group of workers T-I, had been exposed to DU working on contaminated terrain, have had certain risks of cell and chromosome damages, and that risk was not greater than the risk to the referent group R-II of workers occupationally exposed to ionizing radiation.

Introduction

The research studies directly related to depleted uranium (DU) and its utilisation for military purposes are relatively rare in comparison to the studies related to the natural uranium.[1] Natural uranium is normal component of the lithosphere (in the region of Serbia, averagely ranging between 0.5–5g/1 ton of the soil) and it is composed of 3 mutually balanced isotopes: 234, 235 and 238 (depleted).[2 3]

Utilization of ammunition containing depleted uranium (DU) has resulted in contamination of the terrains after the bombing and continuous exposure of the living world to small ionizing radiation doses,[4 - 6] in addition to the already existing natural (e.g., uranium) and artificial (e.g., radio cesium) radionuclides. Local human population living and working in the contaminated environment has been also exposed to continuously increased radioactivity to DU.[7 8]

Two ways of DU transfer from the contaminated environment to humans are possible. Ingestion is the predominant form of DU contamination transfer from the environment to the human bodies in the post-conflict period. DU containing in the soil finds its different ways to be included in the food chain. Through contamination of underground and ground water, radionuclide comes into the plants and animals, to be finally consumed by human population.

The mechanism of internal contamination through inhalation is also possible. The aerosol forms are deposited in the soil, to be thereafter returned to the aerosol forms under the influence of wind or human activities, thus coming into the human body through inhalation.[3]

The study is aimed at identification of health risks in workers permanently or occasionally residing in the contaminated territory associated with contamination of the terrain by low ionizing radiation doses resulting from utilization of ammunition containing DU.

Methods

According to the health consequences monitoring program on the contaminated terrain proposed by the World Health Organization in 1996,[9] the studied population was evaluated for the following:

- Complete clinical examination;
- Complete blood count;
- Leukocyte formula and leukocyte/lymphocyte ratio;
- Microscopically observed morphological changes; presence of immature forms and blasts;
- Leukocyte alkaline phosphatase (L-ALP) activity;
- Cytogenetic tests (lymphocyte karyotype and chromosomal aberrations).

The method had been limited to analysis of the peripheral blood cells and cytogenetic analysis of lymphocyte karyotype as the most sensitive parameters of low dose influence, due to short time period (5 years after initial contamination) for clinical presentation of the possible diseases. Nevertheless this is sufficiently long time of period to expect positive findings of DU blood and urine values, regardless of the previous presence of absence of contamination.

Identification of environmental radionuclide presence effects on the body was performed indirectly, using analysis of blood count parameters: total count of red blood cells, haemoglobin concentration, mean corpuscular volume (MCV), platelets, white blood cells, leukocyte formula. Particular attention had been given to lymphocyte/leukocyte ratio, leukocyte alkaline phosphatase (L-ALP) activity, chromosomal aberration and lesion frequency, probability of onset of chromosomal aberrations.[8 10 - 12]

The venous blood was used for the blood cells count. An automatic counter counted erythrocytes, reticulocytes, platelets and leukocytes.

Blood smears stained with May-Grunewald-Gyms were studied through the optical microscope with immersion for the differences the white cell components, presence of morphologically altered blood cells, young precursors, immature forms and blasts.

Percentages of the white cell components of the blood (specific leukocyte shapes) were used to determine the absolute number of lymphocytes.

The capillary blood smears were stained for alkaline phosphatase using a modified Kaplow's method.[8] The colourless cytoplasm was marked 0, the mildly stained 1, the clearly stained 2, that with numerous granules 3, the intensely stained 4. Every hundred of counted cells was multiplied by the stain intensity index. The sum of these products makes the L-ALP score, i.e. the value of enzyme activity. Score of enzyme activity was presented as the international units (IU).

Chromosomes were observed in peripheral blood lymphocytes. Moorhead's method and conventional cytogenetic techniques were used for preparation of lymphocytes.[10 12] The cells in metaphase were microscopically examined in stained (2% M.G. Gyms) smears under immersion (magnification 100x16). The karyotypes of 200 prepared metaphase lymphocytes were analyzed, when the chromosomes were arranged in equatorial plane. The most characteristic aberrations were dicentric chromosomes. Ring chromosomes and acentric fragments were considered the equivalent to dicentric (chromosome aberrations – ca). Chromatid and chromosomal breaks and chromatid exchanges were designated as chromosomal lesions – cl. Lymphocytes having karyotype damages were marked as damaged cells – dc.[12]

The final studies were performed during 2004, five years after the initial contamination caused by DU radioactive ammunition used in 1999, and the whole five-year period had been taken into account.

Environmental follow-up measurements were performed on several occasions (2001, 2002 and 2004) applying the well known methods, according to which, the initial contamination of the terrain was determined.[5 6] The above evidenced chronically increased population exposure due to DU transfer into the biosphere.[8]

Measurement of the total drinking water alpha-activity resulting from both and DU (DU) from different sources (the total of 18 points) in the contaminated region revealed the values below 1 Bq/l, i.e., averagely 10mBq/l.

The specific DU activity in water samples ranges between 0.03 and 0.21Bq/l.

The total alpha-activity in the samples that are indicators of biosphere contamination (animals – rabbit; and plants – vegetation moss and lichen) ranges between 1340± 200 and 1740 ±260 (rabbit); moss 1370±210 and lichen 1860±280 Becquerel at the average.

The specific activity of the DU in meat (rabbit) was 2.3Bq/kg., while in plants it was 16-48 Bq/kg, averagely: 23± 5Bq/kg.

Cesium 137gamma activity as the indicator of radioactive contamination (even before the conflict) was lower, 0.16 ±0.05Bq/kg (rabbit) and 2.0± 0.1(vegetation).

Statistical analysis included two groups of subjects (table 1): reference group (without potential risk of exposure to DU) and test group (with potential risk of exposure to DU).

Reference group was designed to include two characteristic samples. The first reference sample was obtained from the population considered to be exposed one to natural phone without occupational exposure to ionizing radiation, physical or chemical mutagens. The group comprised young workers employed at the Federal Customs Administration, averagely aged 36.6 years, exclusively males with average 7.6 years of service and zero duration of exposure (group R-I).

As opposed of the above group, another reference group (R-II) was introduced composed of workers employed at the Institute of Oncology in Belgrade, who had been occupationally exposed to ionizing radiation effects. They were constantly subjected to dosimetric and medical control. Dosimetric control included monitoring doses by personnel dosimeters. The absorbed external doses of ionizing radiation to the bodies were measured by personnel thermo luminescent dosimeters (TLD) for the duration of occupational exposure (DOE). The TLD measurements were expressed in mSv, as equivalent doses. Average annual doses for the observed five-year period (1999-2004) have been presented (table 1).

Medical control included periodic check-ups based on the program proposed by the Ionizing radiation protection law,[11] incorporating ICRP recommendations, in order to get the insight into general health status, symptoms and clinical signs (complete clinical evaluation).

The group was averagely aged 42.7 years, with 28.6% of males, 15.6 years of service at the average and 23.8 years of occupational exposure at the average.

Test group comprised two specific samples. The first sample (T-I) included the workers employed at the Radio Television of Serbia from Belgrade, who had stayed, during the North Atlantic Treaty Organization (NATO) bombing of Federal Republic of Yugoslavia (FRY), on several locations including Pljačkovica hill near Vranje, the town in south Serbia ,in order to repair malfunctions on the TV antenna hit by DU ammunition. The group was not occupationally exposed to ionizing radiation before that. However, due to the nature of their occupation, the test group was exposed to increased electromagnetic radiation present on TV antenna. Their average age was 53.7 years, 95% of them were males with 28.7 years of service and 0 years of occupational exposure.

The second test group (T-II) included health care workers from Vranje, who were occupationally exposed to ionizing radiation since they have lived and worked in vicinity of the terrain contaminated with DU.

The workers have been constantly subjected to dosimetric and medical check-ups, similarly to R-II group.

The average age of the second test group (T-II) was 43.3 years, 59.6 % of them were males with 14.8 years of service and 11.1 years of occupational exposure.

Statistical methods

The samples were compared based on the observed parameters using the estimated Poisson distribution and the significance of the difference was tested using Wilcoxon rank sum test with 95% confidence interval and significance threshold at the level of 0.05. The probability of onset of the characteristic complete biomarkers was specially analyzed using logarithmic function of the Poisson regression and it was quantified by lambda parameter (λ).

Linear regression correlation analysis and Student's t-test comparison at the probability level of 0.05 were used.

Results

Measurements performed at the working sites confirmed that the exposure doses absorbed by the occupationally exposed medical professionals in Belgrade (group R-II) and Vranje (group T-II) were admissible and low. The readings were performed in the reference group II using thermo luminescent dosimeters (TLD) and the equivalent doses ranged between 1.00 mSv and 2.04 mSv (1.34mSv/year, at the average). As for the test group II, the mean annual dose was 2.08 mSv, with minimal and maximal values being 0.9 mSv and 4.98 mSv, respectively (Table 1).

The obtained results were tabularly presented in the context of the comparative pair analysis (table 1), both with negative clinical finding of the diseases associated with low ionizing dose radiation effects:

1. Referent group I (R-I) and test group I (T-I);
2. Referent group II (R-II) and test group II (T-II).

Table 1. Statistical parameters of reference and test groups with average annual dose.

Statistical parameters		Groups			
		Reference (R)		Test (T)	
		R-I	**R-II**	**T-I**	**T-II**
N° (persons)		25	42	20	52
Equivalent dose (TLD)		*< 1 mSv*	*1.34 mSv*	*< 1 mSv*	*2.08 mSv*
Age	μ	36.6	42.7	53.7	43.3
(year)	σ	5.8	9.5	12.1	9.7
α_ci = 0.05	ci_lower	34.2	39.8	48	40.6
	ci_upper	39	45.7	59.3	46
Sex	M	25	12	19	31
masculinum –M, versus femininum-F	%	100	28.6	95	59.6
	F	0	30	1	21
	%	0	71.4	5	40.4
Years of services (year)	μ	7.6	15.6	28.7	14.8
α_ci = 0.05	σ	5.9	9.3	12	9.6
	ci_lower	5.2	12.7	23	12.1
	ci_upper	10.1	18.5	34.3	17.5
DOE	μ	0	13.8	0	11.1
(year)	σ	0	9.3	0	9.2
α_ci = 0.05	ci_lower	0	10.9	0	8.6
	ci_upper	0	16.7	0	13.7

TLD - term luminescence dosimeter
DOE - duration of occupationally exposure
μ – mean value – central tendency measure
σ – standard deviation
α_ci = 0.05; ci_ lower, and ci_ upper = confidence interval and deviation below and above it

The R-I/T-I pair differs with respect to average age and average years of service, in absence of any differences related to sex and years of exposure which has had zero value in both groups. The difference in sample size may be considered negligible. The difference was found in potential exposure to ionizing radiation – it was not found in the group R-I, being probable in group T-I because of exposure to DU.

As for the pair R-II/T-II, i.e., health professionals employed in the ionizing radiation zones in two cities (Belgrade and Vranje) in Serbia, no differences were found related to age, years of service, years of exposure, occupation (approximately the same level of qualification). However, the difference was found with respect to their age, mean value (1.34mSv/2.08mSv) and range of annual equivalent ionizing radiation dose over the observed five-year period (1.0-2.04 mSv/0.9-4.98mSv). The equivalent dose of ionizing radiation in T-II was by 55% higher from the dose in R-II as a consequence of the working site conditions. Workers from the test group II have head higher work load (greater number of radiological procedures per single shift). The difference between the groups was also found in potential exposure to DU, which was not found in R-II being probable in the T-II.

The analysis included blood count elements (Tables 2 and 3).

Table 2. Statistical parameters of reference and test groups of blood count.

Blood count		Groups			
		Reference (R)		Test (T)	
		R-I	R-II	T-I	T-II
	N° (persons)	25	42	20	47
Hemoglobin (Hb)	Standardized interval	120 – 160g/l			
	Mean value of standardized interval	140			
α_ci = 0.05	μ	**139.1**	**133.1**	**132**	**133.2**
	σ	±3.8	±12.9	±7.1	±17.4
	ci_lower	137.5	129.1	128.6	128.1
	ci_upper	140.6	137.1	135.3	138.3
	Below limit	0	4 (9.5%)	0	10 (18.9%)
	Above limit	0	2 (4.8%)	0	2 (3.8%)
	Without limit	**0**	**6 (14.3%)**	**0**	**12 (22.7%)**
Eerythrocytes (Er)	Standardized interval	4.0 – 5.5x10^{12}/l			
	Mean value of standardized interval	4.75			
α_ci = 0.05	μ	**4.71**	**4.418**	**4.61**	**4.64**
	σ	±0.350	±0.475	±0.363	±0.445
	ci_lower	4.56	4.27	4.44	4.51
	ci_upper	4.85	4.56	4.78	4.77
	Below limit	0	3 (7.1%)	0	2 (3.8%)
	Above limit	0	2 (4.8%)	0	3 (5.7%)
	Without limit	**0**	**5 (11.9%)**	**0**	**5 (9.5%)**
Mean corpuscular volume	Standardized interval	80.0 – 94.0			
	Mean value of standardized interval	87.0			
α_ci = 0.05	μ	**91.9**	**82.0**	**92.9**	**87.9**
	σ	±4.0	±23.9	±6.3	±7.5
	ci_lower	90.2	74.5	90.0	85.6
	ci_upper	93.5	89.4	95.8	90.3
	Below limit	0 (0%)	9 (21.4%)	0 (0%)	4 (7.6%)
	Above limit	7 (28%)	6 (14.3%)	10 (50%)	5 (9.4%)
	Without limit	**7 (28%)**	**15 (35.7%)**	**10 (50%)**	**9 (17%)**
Retikulocytes (Ret)	Standardized interval	0.5 – 1.5 %			
	Mean value of standardized interval	1.0			
α_ci = 0.05	μ	**0.80**	**1.11**	**1.11**	**1.06**
	σ	±0.316	± 0.803	±0.174	±0.440
	ci_lower	0.674	0.864	1.028	0.817
	ci_upper	0.934	1.365	1.192	1.305
	Below limit	0 (0%)	5 (11.9%)	0	1 (1.9%)
	Above limit	2 (8 %)	11 (26.2%)	0	1 (1.9%)
	Without limit	**2 (8 %)**	**16 (38.1%)**	**0**	**2 (3.8%)**
Platelets (Plt)	Standardized interval	150 – 350x10^9/l			
	Mean value of standardized interval	250			
α_ci = 0.05	μ	**286**	**250**	**303**	**270**
	σ	±22.0	±69.1	±50.8	±65.3
	ci_lower	277.3	228.5	278.7	249.3
	ci_upper	295.5	271.5	326.3	291.0
	Below limit	0	2 (4.8%)	0	2 (3.8%)
	Above limit	0	2 (4.8%)	0	2 (3.8%)
	Without limit	**0**	**4 (9.6%)**	**0**	**4 (7.6%)**

α_ci = 0.05 Confidence 95%; probability p at 0.05
MCV – mean corpuscular volumeof erythrocytes
Standardized interval – adopted from the Ionizing radiation protection law and based on ICRP 2005 and WHO 1996 recommendations (7, 15)
μ– mean value – central tendency measure
σ – standard deviation
α_ci = 0.05; ci_ lower; ci_ upper = confidence interval and deviation below and above it
Below limit – below standardized interval borderline value
Above limit – above standardized interval borderline value
Without limit – below or/and above (without) standardized interval borderline value

Table 3. Statistical parameters of reference and test samples for total leukocytes, leukocyte formula and alkaline phosphatase.

Leukocytes, leukocyte formula, and alkaline phosphatase		Groups			
		Reference (R)		Test (T)	
		R-I	R-II	T-I	T-II
	N° (persons)	25	42	20	47
Leukocyte	Standardized interval	$4.0 – 9.0 \times 10^9/l$			
Le	Mean value of standardized interval	$6.5 \times 10^9/l$			
α_ci = 0.05	μ	**6.44**	**6.54**	**6.35**	**6.95**
	σ	1.41	2.18	1.69	2.27
	ci_lower	5.86	5.87	5.57	6.28
	ci_upper	7.03	7.22	7.14	7.62
	Below limit	0	2(4.8%)	0	1(1.9%)
	Above limit	2 (8%)	5(11.9%)	2 (10%)	8(15.1%)
	Without limit	**2 (8%)**	**7(16.7%)**	**2 (10%)**	**9(17%)**
Granulocyte	Standardized interval	0.51– 0.61			
G	Mean value of standardized interval	0.56			
α_ci = 0.05	μ	0.598	0.577	0.630	0.597
	σ	0.060	0.113	0.152	0.087
	ci_lower	0.573	0.542	0.558	0.571
	ci_upper	0.622	0.612	0.701	0.622
	Below limit	1(4%)	10(23.8%)	1(5%)	3(5.7%)
	Above limit	0	1(2.4%)	0%	2(3.8%)
	Without limit	4%	11(26.2%)	5%	5(9.5%)
Lymphocyte	Standardized interval	0.21– 0.35			
Ly	Mean value of standardized interval	0.280			
α_ci = 0.05	μ	0.319	0.326	0.288	0.357
	σ	0.0805	0.0815	0.0363	0.0829
	ci_lower	0.2860	0.3006	0.2705	0.3322
	ci_upper	0.3524	0.3513	0.3045	0.3809
	Below limit	2 (8%)	6(14.3%)	8 40%)	4(7.6%)
	Above limit	1 (4%)	5(11.9%)	1(5%)	11(20.8%)
	Without limit	3 (12%)	11(26.2%)	9(45%)	15(28.4%)
Monocyte	Standardized interval	0.04 – 0.08			
Mo	Mean value of standardized interval	0.060			
α_ci = 0.05	μ	0.0500	0.0657	0.0425	0.0430
	σ	0.0144	0.1008	0.0234	0.0203
	ci_lower	0.0440	0.0343	0.0316	0.0370
	ci_upper	0.0560	0.0971	0.0534	0.0489
	Below limit	3(12%)	12(28.6%)	6(30%)	12(22.6%)
	Above limit	1(4%)	3(7.1%)	0	1(1.9%)
	Without limit	4(16%)	15(35.7%)	6(30%)	13(24.5%)
Leukocyte alkaline phosphatase L-ALP	Standardized interval	20.0 – 80.0 IU			
α_ci = 0.05	Mean value of standardized interval	50.0			
	μ	**64.9**	**57.1**	**74.9**	**67.1**
	σ	8.8	15.3	9.8	16.3
	ci_lower	60.3	51.8	70.3	61.8
	ci_upper	69.5	62.3	79.5	72.3
	Below limit	0	0	0	0
	Above limit	0	0	7 (35%)	4(7.6%)
	Without limit	0	0	**7(35%)**	**4 (7.6%)**

α_ci = 0.05 – Confidence 95%; probability (p) at 0.05
Standardized interval – adopted from the Ionizing radiation protection law and based on ICRP 2005 and WHO 1996 recommendations (7, 15)
μ – mean value – central tendency measure
σ – standard deviation
α_ci = 0.05; ci_ lower, and ci_ upper; = confidence interval and deviation below and above it
Below limit – below standardized interval borderline value
Above limit - above standardized interval borderline value
Without limit – below or/and above (without) standardized interval borderline value

Mean erythrocyte values were not statistically significant for probability threshold condition α=0.05 for the subject pair R-I/T-I (table 2). The subjects from the both groups were within the standard range for the erythrocytes. Mean haemoglobin values were also within standard range in the both groups. The difference was not found with respect to the criterion of falling of the measured haemoglobin values out of the standard interval. However, mean haemoglobin values were statistically different with respect to the probability threshold condition α=0.05.

The parameter of the erythrocyte volume (mean corpuscular volume -MCV) indicating haemoglobin content in the erythrocytes and cellular size and shape relevant for its function, was not statistically significant, likewise reticulocyte number. No significant difference was found between the subject pair R-I/T-I with respect to red blood cell line parameters (table 2).

Mean platelet values were not statistically different. All the subjects from both groups were within standard range for platelets (table 2).

As for the subject pair R-II/T-II (table 2) mean red blood cell values ranged within standard limits and the difference found between them cannot be considered statistically significant with respect to the probability threshold condition α=0.05. The difference was found with respect to the criterion of falling of the measured values out of the standard interval for erythrocytes. However, it cannot be considered statistically significant.

As for the test group (T-II) the total number of subjects found to be out of the standard range was 5 (11,9%) being in the reference group (R-II) 5 or 9,5%, with their deviation ratio of 1,2. Mean haemoglobin values were within standard limits and the difference between them cannot be considered to be statistically significant for the condition of probability threshold α=0.05. The difference was found with respect to the criterion of falling out of the measured values out of the standard interval for haemoglobin, MCV and reticulocytes and it must be considered statistically significant.

As for the test group T-II, there was significantly higher number of subjects who were out of the standard interval for haemoglobin when compared to the reference group, while the number of subjects out of the standard interval for MCV was significantly lower when compared to the reference group R-II (table 2). Therefore, red blood cell line generally cannot be considered significant in this pair as well with respect to the ionizing radiation influence. As it may be seen in table 2, no differences in platelet line were found for the test pair R-II/T-II. The mean values do not differ significantly while deviations from the standard interval are approximately the same in the both groups.

As for the pair comprising reference and test group R-I/T-I (table 3) mean values of white blood cells were within standard limits and they were not statistically different for α=0.05 (confidence 95%). Additionally, no difference was found with respect to falling out of the borderline values of the standard while blood cell interval, while deviation above the upper limits was almost the same in the both groups. No significant difference was found in number of lymphocytes. The average distribution in the leukocyte formula was 0.32% vs. 0.29% (table 3), while mean values of the absolute number of lymphocytes were 2.1± 0.5 vs.1.8±0.5. The difference was found with respect to the criterion of deviation of the measures values from the standard interval for lymphocytes. As for the reference group I, the number of subject falling out of the standard interval was significantly lower (12%) in comparison to the test group I (45%).

As for the R-II/T-II pair, no difference was found in white blood cell count, being 16.7% for the reference group and 17% for the test group, meaning that this difference in deviation was not significant. Significant difference was not found with respect to other elements of the leukocyte formula. Basophiles were not evidenced. Moreover, no immature cells or blasts were observed.

Although the comparison between the groups revealed no statistical difference, deviation from the standard intervals for white blood cells and lymphocytes was still present, however it was approximately the same in the both groups (reference II and test II), which included health care professionals.

In comparison to the reference group I (free of ionizing radiation risk) the deviations are prominent (tables 2 and 3).

It has been known that, in addition to alteration of white blood count, ionizing radiation **also** influences leukocyte alkaline phosphatase activity changes as well as the changes of the natural proportions of the leukocytes formula (table 3). As for the L-ALP activity, the changes were characterized by increased activity and they cannot be considered statistically significant, particularly owing to the fact that they follow the increase in white blood count, which is normal phenomenon.

Due to specific lymphocyte sensitivity, lymphocyte-related changes are also particularly characteristic with respect to their number and correlation with white blood cells as well as with respect to chromosomal aberrations in their nuclei.

The correlation analysis of leukocyte and lymphocyte relation was performed for all subjects from the reference and test groups. The analysis of linear regression of the relation between leukocytes and lymphocytes gave regression coefficient for the reference and test groups R-I, R-II, T-I and T-II: 0.76, 0.83, 0.90 and 0.78, respectively. The results of comparison of the linear regression parameters using parametric Student's t-test with significance threshold of 0.05 did not show any difference. In none of the cases statistical difference of white blood cells and lymphocyte proportions was identified for the probability level of 0.05.

Presence of chromosomal aberrations was analyzed in the reference and test groups (Tables 4 and 5).

Table 4. Poison distribution of chromosomal aberrations, chromosomal lesions and damaged cells in the reference and test groups.

Chromosomal aberration, lesion, and damaged cells			Groups			
			Reference (R)		Test (T)	
			R-I	R-II	T-I	T-II
N° (persons)			25	42	19	41
n° (cells)			5000	8050	3800	7920
Chromosomal aberrations –ca	N° persons with ca		4	7	6	20
	%		16.0	16.7	31.6	48.8
	Total ca		4	14	8	32
	% - frequenci ca		**0.08**	**0.17**	**0.21**	**0.48**
	Poison	λ	**0.1600**	**0.3333**	**0.4211**	**0.9268**
	distribution	ci_ lower	0.0269	0.1483	0.1353	0.5853
	α_ci = 0.05	ci_ upper	0.5038	0.6390	0.9778	1.3892
Chromatid lesions - cl	N°persons with cl		1	16	0	17
	%		4.0	38.1	0.0	41.5
	Total cl		1	29	0	24
	%		0.02	0.36	0.00	0.30
	Poison	λ	0.0400	0.6905	0	0.5854
	distribution	ci_ lower	0.0002	0.4049	NaN	0.3233
	α_ci = 0.05	ci_ upper	0.2972	1.0947	0.2789	0.9694
Damaged cells - dc	N°persons with dc		5	18	6	26
	%		20.0	42.9	31.6	63.4
	Total dc		5	27	8	47
	%		0.10	0.34	0.21	0.60
	Poisson	λ	0.2000	0.6429	0.4211	1.1463
	distribution	ci_lower	0.0431	0.3688	0.1353	0.7614
	α_ci = 0.05	ci_upper	0.5660	1.0356	0.9778	1.6516

λ - lambda parameter of density of probabilities of Poisson distribution
α_ci = 0.05 Confidence 95%; probability (p) at 0.05
α_ci = 0.05; ci_ lower, and ci_ upper; = confidence interval and deviation below and above it

Table 5. Significance of difference in chromosomal aberrations and lesions between the reference and test groups.

Reference groups	Chromosomal aberration, lesion, and damaged cells	Test group 1		Test group 2	
		Confidence 95% alfa=0.05	p	Confidence 95% alfa=0.05	P
Referent group 1	Chromosomal aberration	No	0.2103	**Significance *<0.01***	**p=0.0041**
	Chromatid lesion	No	0.4089	*Significance*	9.86e-004
	Damaged cells	No	0.3531	*Significance*	1.29e-004
Referent group 2	Chromosomal aberration	No	0.3449	**Significance *<0.01***	**p=0.0045**
	Chromatid lesion	*Significance*	0.0022	No	0.7582
	Damaged cells	No	0.3513	*Significance <0.05*	p=0.0203

alfa=0.05 - Confidence 95%; probability (p) at 0.05

Observation of the first pair from the groups R-I and T-I evidenced presence of chromosomal aberrations (table 4.) in the both group. Poisson distribution parameter λ differs, however the corresponding confidence intervals are crossed over for the significance threshold α=0.05, which indicates that the difference is not statistically significant. The former is confirmed by Wilcoxon test of matching of the estimated parameters of the Poisson distribution. The former indicates the presence of tendency toward chromosomal aberrations (higher risk of their onset) in the group T-I in comparison to the comparative reference group R-I. However, the difference in incidence of chromosomal aberrations is not significant, at the average (tables 4 and 5), although the frequency of chromosomal aberrations (table 4) is higher in the group T-I (0.21% vs. 0.08%) in comparison to the reference one.

The equivalent analysis for the test pair R-II and T-II shows similar tendencies, however they are more prominent and exceed the limits of statistic tolerance in testing of differences (do not belong to the same set - tables 4 and 5). Estimated function of density of probabilities of Poisson distribution of the chromosomal aberrations in the test group T-II is drastically different from the corresponding distribution of the reference group R-I and to the somewhat lesser extent from the group R-II; Wilcoxon test exactly confirms presence of significant difference between the reference group R-II and test group T-II, $p<0.05$ (tables 4 and 5).

The difference in chromosomal aberrations is observed between the reference groups R-I and R-II which is expected since the groups are mutually different with respect to the occupational exposure to low ionizing radiation doses (table 4). Reference group II has significantly higher incidence of the unspecific chromosomal lesions ($p=0.0012$) in comparison to R-I group, as well as significantly higher number of the damaged lymphocytes for that reason (damaged cells – dc, p=0,0054 Frequency of chromosomal aberrations characteristic for radiation is increased (0.17% vs. 0.08%) however the increase is non-significant ($p=0.13$, Wilcoxon test). Poisson distribution parameter λ is higher (0.33 vs. 0.16) in the group R-II. This indicates that the probability for onset of chromosomal aberrations is higher in R-II in comparison to R-I group (tables 4).

The results of the chromosomal aberration analysis indicate that the test group I, which has been temporarily occupationally exposed to the effects of low environmental ionizing radiation doses as well as to non-ionizing radiation, had been different from the groups R-I and T- II being at the same time the most similar with the results of the health-related risk analyzes to the referent group R-II, which was chronically occupationally exposed to ionizing radiation.

Group T-I has been significance low chromatid lesions than R-II ($p=0.0022$, table 5). Frequency of chromosomal aberrations characteristic for radiation is increased (0.21%) and Poisson distribution parameter λ is higher than in referent groups (λ=0,42), but they are no significant.

Test group II, despite test group I, had been significantly different (confidence 95%; table 5) from the both of referent groups (R-I and R-II) taking into consideration chromosomal aberrations in lymphocytes ($p<0.01$) and damaged cells ($p<0.05$).

Accordance to logarithmic function Poisson regression, complete biomarkers (chromosomal aberration – ca) showed that relative risk (RR) in group T-II was 4; in group T-I is 2 and in group R-II it was 3, while in group R-I exposed only nature radiation, relative risk had been 1.

Discussion

Identification of field contamination resulting from application of radioactive ammunition risks on the health on individuals living and working on the contaminated territory is significant for studying of the admissible levels of absorbed radiation of different population groups, particularly individuals occupationally exposed to the ionizing radiation as well as their patients and general population.

The obtained results indicate the presence of the objective risk if exposure of workers living and working on the known contaminated terrain on the south of Serbia as a result of DU originating from the military conflicts that took place on the terrain in 1999.

DU ammunition has become a source of environmental contamination.[4 5 6] On one hand, it is superimposed to the natural phone, while on the other, it penetrates to the soil and ground water to come thereby into the biosphere. Thus, DU becomes a residual radiological risk for human life in the course of their life on the affected terrain.[13] As for the non-occupationally exposed population, each dose above 1mSv/1year is conspired to be a significant dose. Low environmental doses are cumulated during the stay in such regions and within five-year period, if they are above 1 mSv per year, they may reach 6mSv, which represents the upper limit for the occupationally exposed individuals in the category B zone.[11] For this reason, the individuals working and living on the contaminated territories are subjected to health monitoring.[3 6 9]

The analysis included blood count elements, as the most sensitive parameters associated with the reaction of the organism to ionizing radiation effects, which may be monitored and point out to the early signs of the diseases using simple techniques. Blood element differences were observed between the studies groups, however they were insufficiently specific or statistically significant, and thus the analyses were not sufficient for risk assessment.

Lack of specificity of the blood count elements as health indicators for assessment of ionizing radiation risk points out to the need of analysis of chromosomal aberration presence.[12 - 14]

As for the workers living in a wider region with verified contaminated zones who are occupationally exposed to low doses of ionizing radiation, statistical differences were identified upon their comparison with the referent group of the occupationally equivalent subjects, particularly with respect to onset of the chromosomal aberrations. For that reason, the number of damages lymphocytes is higher, which is essential for their organism defence role.

No statistically significant difference was found with respect to unspecific, single-stranded chromatid changes (lesions), which do not form chromosomal figures (dicentrics). Most probably, they are also the result of other causes, in addition to radiation (chemical metal toxicity, habits, smoking etc.) and they may influence karyotype instability.[12]

Additionally, it has also been evidenced that onset of these lesions on DNA is accompanied by the same number of the stable aberrations (deletion, inversion and translocation), which indirectly suggests the probability of onset of mutations (due to stable aberrations fixed in the cell division).

Test group II (workers in the radiation zone in the contaminated region in the southern Serbia) had been composed of two groups of subjects. The first one was composed of individuals at risk of uniform low-dose irradiation on their working sites, while the second one comprised the individuals at risk of contamination with DU. For this reason, the radiation risk was the highest in this group and it has been quantified by the highest probability of onset of chromosomal aberrations and the highest frequency of chromosomal aberrations and damaged lymphocytes. Damaged lymphocytes lost function because since 50% of them tend to disappear after the initial divisions, and before division ten, all the remaining damages ones also disappear.[8 10 12 14]

Only referent group I was free of any of the 2 above mentioned risks – radiation at the working site or radiation at the contaminated terrain. Therefore, it is clear why the referent groups are also mutually different, since the referent group II had been composed with the individuals chronically exposed to low ionizing radiation doses at their work sites. Nevertheless, the difference was found between the reference groups I and II with respect to chromosomal lesions, i.e., higher risk expressed as probability of onset of lesions was found in the referent group II composed of the occupationally exposed health care professionals from Belgrade. The group was also proved to have higher frequency of unspecific lesions in comparison to the test group I, in which occupational exposure was not continuous but only occasional during repair works on the TV antenna on the contaminated terrain. This group was not exposed to increased radiation and did not have relative risk due to exposure to DU. It was not significantly different from other referent groups with respect to other parameters, however if was different from the test group II composed of subjects continuously staying in the vicinity of the contaminated region.

Accordingly, statistically significant differences related in comparison with the referent group had not been observed, although changes values of certain parameters were evidenced in workers who had lived out of the contaminated regions but who were occupationally exposed for some time due to their stay in the contaminated zone performing their professional tasks.

Therefore, life in the contaminated region is associated with higher radiological risk, particularly for those who work in the ionizing radiation zone and probability of potential occupational diseases may be higher and thus workers rights related to occupational diseases, treatment and assessment of their working abilities should be considered. Additionally, the criteria for recognition of radiation-induced occupational diseases must be also reconsidered in the new circumstances associated with increased risk of environmental contamination as well.

The above fact points out need for further investigations of the association between occupational exposure to the ionizing radiation effects and effects of the underlying environmental contaminants, particularly within the context of the increased relative risk that way quantitatively verified in this study. Accordingly, the aspect of legal regulations in the fields of occupational protection should be also reconsidered, since the situation is new and uncovered by the current regulations. This is particularly important for the health risks of medical professionals in the ionizing radiation zone in southern Serbia, having in mind the fact that their work load is higher due to higher morbidity rate among the population in the underdeveloped region.[9 15]

Conclusions

For all the above, the latent period before onset of the disease is prolonged, and thus the consequences of the DU effects are yet to be expected. The former points out to the need of reassessment of the admissible level of radiation in case of occupational exposure. Decrease of occupational exposure (admissible dose level) may suppress the influence of the environmental doses, since natural phon is increase by superimposed contamination with DU.

Increased health risk of the workers exposed to ionizing radiation caused by profession and additionally by contamination from the contaminated environment by DU, has become because of cumulative radiobiology effects of small doses over the continual exposition and it depends to the time of period of exposure duration.

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Current challenges for radiation protection in medical X-ray and molecular imaging

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Abstract
The increasing radiation exposure from medical diagnostic procedures is becoming a subject of concern and social interest and is given increased attention by health professionals, authorities and manufacturers. Also patients are now more and more concerned about their exposure situation. New imaging technologies that use x-rays or radioactive materials are continuously developed. These include faster CT-scanners, new interventional techniques, increased use of positron emission tomography (PET), and digital imaging techniques. The techniques are often combined into a single investigation, such as PET/CT and SPECT/CT. As the availability and quality of radiological imaging methods has continued to grow, increases in image quality rather than a reduction in patient dose continues to characterise modern practice. The benefits of the new techniques are so rewarding that a tendency to overuse the technology has occurred. The aim of the presentation is to inform about developing technologies and clinical techniques, and their associated potential radiation risks to patients and staff. The presentation also includes a discussion on tools for optimising investigations and improved dose management, and most importantly, ways to create greater awareness of the physicians prescribing the examinations. It is important that the advances in imaging technology also provide increased medical benefits to large numbers of patients, while maintaining the efforts to keep unnecessary exposure to a minimum. Therefore, persons educated and trained in radiation protection have to be fully involved in planning, dose measurement and optimisation and have to initiate necessary radiation safety training for the medical staff, including those who refer the patients for an investigation. Strict guidelines need to be developed so that referring physicians carefully weigh the benefits against the potential risks and base their decisions on medically relevant data.

Justification of medical exposure in diagnostic imaging – a way forward

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IAEA

Collective doses from medical exposures: an inter-comparison of the "TOP 20" radiological examinations based on the EC guidelines RP 154

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Abstract

The network of Heads of European Radiation Control Authorities (HERCA) has appointed a working group (WG6) to test the feasibility of the European Commission guidance on estimating population doses from medical x-ray procedures, Radiation Protection Report No 154. Fourteen European countries have collected frequency and/or dose information for at least the "TOP 20" examinations that in RP154 were identified as most contributing to the total collective effective dose. The national surveys were aimed to establish updated national information on doses representative for current apparatus and scan techniques. Around 2002, the collective effective dose for the "TOP 20" examinations ranged from 303 to 1421 mSv per 1000 population in Europe. In 2008 the figures still range from 331 to 1521 mSv per 1000 population. However, the contribution to the "TOP20" from conventional radiography examinations has decreased, while the contribution from CT now is in the range 46 – 81%. The challenges and uncertainties in estimating frequency and dose recognised in RP154 were still present. A need to update the "TOP 20" list was recognised.

Introduction

The use of ionizing radiation in diagnostic radiology makes a major contribution to population dose. The introduction of new modalities such as multi-detector computed tomography has tended to increase the doses; it is therefore of concern for the radiation protection authorities to follow trends in the medical use of radiation. An EU-funded project called DOSE DATAMED (2004 - 2007) was set up to develop harmonized methods for future surveys of population exposure from medical x-rays. The project group was recruited from radiation control authorities or expertise institutes in ten European countries. The guidance was published in 2008 by the European Commission as report No 154 in the Radiation Protection Series [1]. Since then, the network of Heads of European Radiation Control Authorities (HERCA) has appointed a working group (WG6) to test the feasibility of RP154 for ongoing European surveys. Some of the countries are performing full national surveys based on the identification and frequency counting of 225 specific x-ray examinations or 70 broader categories of examinations. Others are testing the methodology based on the set of 20 examinations identified as the ones contributing most significantly to collective effective dose. Pitfalls and challenges associated with the different approaches will be identified. All countries are collecting dose values for at least the “TOP 20” examination types, to establish updated national information on doses representative for current apparatus and scan techniques. Around the turn of the century, the collective effective dose for the “TOP 20” examinations ranged from 303 to 1421 mSv per 1000 population in Europe. Comparison of the collective effective dose averaged over the population based on the frequency in 2006-2008 and updated dose information from the “TOP 20” examinations will be presented, looking for trends and information from more European countries.

Material and methods

How to count the “TOP20” examination types” in 2008

Detailed descriptions of the 20 types of examination that were consistently found to be amongst the highest contributors to the collective effective dose in the ten DOSE DATAMED countries, the ‘Top 20 Exams’, are provided in RP154 Appendix 1 Table A1 [1]. They are sorted according to imaging modality:

- PLAIN RADIOGRAPHY (without contrast media): 1. Chest/lung, 2. Cervical spine, 3. Thoracic spine, 4. Lumbar spine, 5. Mammography, 6. Abdomen and 7. Pelvis & hip
- RADIOGRAPHY/FLUOROSCOPY (mostly involving use of contrast media): 8. Barium meal, 9. Barium enema, 10. Barium follow, 11. IVU (Intravenous Urography) and 12. Cardiac angiography
- COMPUTED TOMOGRAPHY (CT): 13. CT head, 14. CT neck, 15. CT chest, 16. CT spine, 17. CT abdomen, 18. CT pelvis and 19. CT trunk
- INTERVENTIONAL PROCEDURES: 20. Coronary angioplasty (PTCA)

The recommended definition of an x-ray examination is [1]: ‘*An x-ray examination or interventional procedure is defined as one or a series of x-ray exposures of one anatomical region/organ/organ system, using a single imaging modality (i.e. radiography/fluoroscopy or CT), needed to answer a specific diagnostic problem or clinical question, during one visit to the radiology department, hospital or clinic*’.

Table 1. Overview of the national frequency survey methods.

Country	National source of information	No of hospitals and entities in the survey (%)
Belgium	National Health Insurance Institution code system (RIZIV / INAMI): about 130 (double - quadruple coded) types of diagnostic and/or interventional X-ray procedures (dental X-ray excluded)	All (7 university hospitals, 61 public hospitals & 135 private hospitals), medical institutes, and practices (100%)
Denmark	Part of "Sundhedsvæsenets Klassifikationssystem" (SKS) www.medinfo.dk/sks. 206 x-ray codes	All public hospitals report examinations to SKS (100%)
Estonia	Estonian Health Insurance Fund code system: 73 types of diagnostic and interventional X-ray procedures, including CT and dental X-ray	All
Finland	The national code system; About 800 codes	Covered about 97%
France	CCAM, 224 codes (including radiography, CT and diagnostic IR)	
Germany	Health insurance companies (codes designed to meet the German system of reimbursement)	10% of hospitals, 100% of practices (tot.:70% of *exams*)
Iceland	The codes are based on old Swedish codes which have been modified over the years.	All
Lithuania	The Lithuanian Health Information Centre provides only total numbers. For radiography/fluoroscopy the distribution in UNSCEAR 2000 (Table 30) was used to establish the TOP 20 numbers, while the CT numbers were gathered from hospitals.	
Netherland	Radiological code system (CTG)	All (university hospitals, regular hospitals, private clinics, screening)
Norway	Radiological code system (NORAKO), The Norwegian association of radiologists, version 2008 178 codes for X-ray based exams (302 in total)	ALL (100%): 50 hospitals, 25 private X-ray institutes, 10 private hospitals & screening
Sweden	Classification of radiological procedures from the National board of Health and Welfare (SoS 1991) No of codes: 299	20 entities (hospitals, health care centres) corresponding to 20 % of the whole country
Switzerland		Small pilot, survey runs now
UK	National Standard Representation of Clinical Imaging Procedures. No of procedures: 3220	

Most countries used a kind of radiological code system to collect frequency data (Table 1). These coding systems are in many countries also used for reimbursement purposes, and are not directly reflecting the number of examinations in the way they are defined in RP154. Each country had to develop a method for the translation of code frequencies into actual numbers of examinations. Some countries collected information from only a part of the activities and scaled up to reflect the whole country, while other countries collected frequency data from all hospitals and x-ray institutes. Lithuania had only the total number of radiography/fluoroscopy available but used the distribution between various types of examinations provided by UNSCEAR [2]. Switzerland is running a national survey now; the data are preliminary scaled up from a small sample, also considering the distribution in the Swiss 2002 data. Estonia and Luxembourg are also currently running their surveys and may provide more data in near future.

The dose survey methods

The majority of the dose figures were collected from 2006 – 2008, while some countries have used dose data from 2000 – 2006. The dose data were in most countries collected from calibrated x-ray equipment by appointed professionals in the hospitals and sent in to the authority. In most countries the dose data were collected in terms of the “practical dose quantities” as identified in RP154; the dose-area product (DAP) for radiography and fluoroscopy, mean glandular dose (MGD) for mammography, and dose-length product (DLP) for CT examinations, i.e. in these countries the national dose figures were based on the same data material as used for the establishment of national diagnostic reference levels. Here, the conversion coefficients from the practical dose quantities to effective dose, based on the ICRP 60 definition and tissue weighting factors [3] were used. On the other hand, some countries (Finland, France, Switzerland) based their national dose figures on the use of various kinds of software that calculate the effective dose based on exposure and scan parameters. In Finland for example, they used PCXMC (STUK, Finland) to calculate the effective dose for plain radiography, both according to ICRP 60 [3] and ICRP 103 [4].

Results

Trends in frequency in X-ray based radiological examinations during the last six years

The frequency for the sum of the “TOP20” examinations in 2008 is presented in Table 2 with comparison to 2002 for the countries that contributed to RP154 [1].

Table 2. No exams/y/1000 population in 2008 compared to 2002 for the “TOP 20” [1].

Participating country	2002	2008	Change (%)
Belgium	790	690	– 13 %
Denmark	279	327	+ 17 %
Estonia	n.a.	354	
Finland	n.a.	432	
France	503	572	+ 14 %
Germany	723	612	– 15 %
Iceland	n.a.	481	
Lithuania	n.a.	761	
Luxembourg	572	n.a.	
Netherland	335	341	+ 2 %
Norway	507	464	– 9 %
Sweden	355	382	+ 8 %
Switzerland	444	570	+ 28 %
UK	305	335	+ 10 %
Average (min – max)	481 (279 – 790)	487 (327 – 761)	

On average, the “TOP20” examination frequency has been quite stable during the last six years, and it can be noticed that Belgium and Germany that counted for the highest frequencies in 2002 have lower frequencies now. The frequency for each of the “TOP 20” examination types for the countries in the survey is presented in Table 3. Typically, there is only a factor 2 – 9 between the highest and lowest frequency among the countries in the use of plain radiography, while there are huge differences in the use of fluoroscopic examinations; some countries have almost stopped using those procedures. There are also huge differences in the use of CT.

Table 3. The range in examination frequency normalised to per 1000 population for the “TOP 20” examination types as defined in RP154 [1] in fourteen European countries anno 2008.

TOP 20 Exam type	Specific exams included in ‘Exam type’	Average frequency	lowest frequency	highest frequency
1. Chest/lung	Lungs & ribs, Thoracic inlet	182.7	104.7	428.1
2. Cervical spine	Cervical spine	16.3	6.5	45.3
3. Thoracic spine	Thoracic spine	10.5	4.4	20.6
4. Lumbar spine	Lumbar spine Lumbo-sacral joint Sacro-iliac joints Sacrum & coccyx	33.8	14.9	59.5
5. Mammography	Symptomatic & Screening	61.3	21.7	85.0
6. Abdomen	Abdomen (plain film)	22.3	0.7	56.0
7. Pelvis & hip	Pelvis (one or both hips)	53.7	38.2	89.9
8. Ba meal	Ba meal (stomach & duodenum)	2.0	0.04	7.1
9. Ba enema	Ba enema (colon)	2.4	0.2	12.5
10. Ba follow	Ba follow (small intestine) Small bowel enema	0.8	0.1	1.8
11. IVU (Intravenous Urography)	IVU (kidneys, ureter and bladder)	2.4	0.1	11.4
12. Cardiac angiography	Coronary angiography Left or right ventriculography	5.2	1.7	15.0
13. CT head	Head, brain, facial bones	32.6	18.7	58.0
14. CT neck	Soft tissue in neck, cervical spine	6.9	0.8	34.4
15. CT chest	Chest/thorax	18.8	4.8	32.6
16. CT spine	CT of lumbosacral spine	9.0	0.9	33.7
17. CT abdomen	Abdominal organs	23.9	3.6	44.8
18. CT pelvis	Pelvic bone &/or organs	4.8	0.6	24.5
19. CT trunk	CT of chest, abdomen & pelvis. CT of thoracic/ abdominal aorta	19.1	0.5	106.0
20. Coronary angioplasty (PTCA)	PTCA	2.3	1.3	5.8

Trends in radiation doses per patient examination during the last decade

Effective dose values for each of the “TOP 20” examination types is presented in Table 4, all referring to ICRP 60 tissue weighting factors [3]. There are huge variations in average dose figures between reporting countries also in 2008. The factor between the highest and lowest country average dose figures are in the range 2 – 68. The variations in CT doses seem to be somewhat lower, a factor 2 - 5 between the countries which have submitted data recently. There are also considerable variations in doses regarding angio- and interventional procedures, cardiac angiography and PTCA.

The effective dose for all plain radiography examinations seems to have decreased during the last six years. This tendency also applies to the fluoroscopy examinations (except barium enema). On the other hand, the effective doses in CT seem to have been quite stable, even though a slight decrease in dose is seen for CT of chest and abdomen.

Table 4. The range in national average effective dose values for the “TOP 20” examination types in 2008 compared to similar values reported in 2002 [1].

TOP 20 Exam type	Effective dose 2002 (mSv)	Range (min – max)	Effective dose 2008 (mSv)	Range (min – max)
1. Chest/lung	0.12	(0.02 – 0.30)	0.09	(0.01 – 0.29)
2. Cervical spine	0.27	(0.02 – 1.10)	0.23	(0.02 – 1.10)
3. Thoracic spine	1.01	(0.30 – 3.50)	0.81	(0.30 – 3.50)
4. Lumbar spine	1.86	(0.40 – 4.10)	1.32	(0.40 – 4.10)
5. Mammography	0.33	(0.10 – 0.60)	0.20	(0.03 – 0.42)
6. Abdomen	1.53	(0.40 – 3.60)	1.19	(0.40 – 2.93)
7. Pelvis & hip	0.89	(0.40 – 2.00)	0.70	(0.25 – 2.00)
8. Ba meal	7.71	(2.30 – 18.50)	6.38	(2.00 – 18.50)
9. Ba enema	8.60	(5.40 – 15.90)	9.83	(2.60 – 25.73)
10. Ba follow	10.48	(2.20 – 42.30)	9.08	(0.63 – 42.30)
11. IVU (Intravenous Urography)	4.03	(2.10 – 7.90)	2.91	(2.10 – 4.25)
12. Cardiac angiography	9.12	(4.30 – 12.00)	7.15	(1.20 – 14.40)
13. CT head	2.07	(1.20 – 2.60)	1.95	(1.20 – 3.05)
14. CT neck	2.53	(1.30 – 3.40)	2.70	(1.10 – 5.00)
15. CT chest	8.29	(4.10 – 11.50)	5.35	(3.50 – 7.37)
16. CT spine	5.47	(2.90 – 9.10)	7.04	(3.10 – 11.81)
17. CT abdomen	12.54	(8.40 – 18.60)	10.28	(5.30 – 17.90)
18. CT pelvis	8.73	(7.00 – 10.60)	7.65	(0.80 [i] – 14.48)
19. CT trunk	13.54	(7.90 – 24.40)	15.80	(8.40 – 33.40)
20. Coronary angioplasty (PTCA)	14.05	(9.00 – 23.0)	15.04	(2.84 – 23.00)

[i] The lowest dose value is due to that CT pelvis was interpreted as CT pelvimetry

Trends in collective effective doses per 1000 population (limited to the TOP20 exams)

The collective effective dose per 1000 population from the list of "TOP20" examinations has on average been increasing from 744 to 864 mSv for the group of countries participating in the two surveys around 2002 and 2008. Furthermore, there are still major differences in 2008 in the total collective effective dose per 1000 population between the countries, the variations being about the same as in the previous survey; 331 – 1521 mSv per 1000 population. France, Netherland and Switzerland are countries where the collective effective dose seems to have increased significantly, more than expected from the increase in frequency (Table 2).

Table 5. Trends in the collective effective dose per 1000 population in 12 European countries in the last decade, based on twenty identified examination types most contributing to S around the year 2000 [1].

Example of table	**DOSE DATAMED [1]**		**New surveys based on the year 2006/08**	
Participating country	**S/y/1000 pop [mSv] for the "TOP 20" i)**	**Year of survey data ii)**	**S/y/1000 pop [mSv] for the "TOP 20" iii)**	**Trend in S from previous survey to current situation (%)**
Belgium			1387.7	
Denmark	351	1995	454.9	+ 30%
Finland	500 iv)	1995	346.1	
France	597	2002	1110.1	+ 86%
Germany	1421	2000	1521.5	+ 7%
Iceland			1276.1	
Lithuania			466.6	
Luxembourg	1388.5	2002	n.a.	
Netherland	303.3	2002	477.0	+ 57%
Norway	942.2	2002	942.9	0
Sweden	550	1995	591.1	+ 7%
Switzerland	831	1998	1366	+ 64%
UK	312.8	2001	331.2	+ 6 %
In average	**744**		**856**	**+ 12 %**
Range in TOP20	**[303 – 1421]**		**[331 – 1521]**	

i) From APPENDIX 3 to ANNEX 1 – DD Report 1 in Radiation protection N° 154 [1], In the table, data in square "TOTAL 1-20 (excl. 'All' groups)" was selected

ii) The year when the frequency of the examination was counted in last survey. The dose data could be older in some countries.

iii) Current surveys done according to European guidance RP No 154 [1] based on frequency from 2008 (except German data from 2006 and French from 2007) and updated national dose figures.

iv) The data includes all examinations and is therefore not directly comparable to the "TOP 20"

Contribution of examination types to the collective effective dose in 2008

The contribution to the collective effective dose from the “TOP 20” examination types sorted according to the imaging modality is presented in Figure 1. We do not yet have the values of total collective effective dose from all countries in the survey, and cannot assume that the distributions of the modalities within “TOP20” are representative for all examinations. It is still interesting to notice that CT examinations now account for 46 – 81% of the “TOP20” collective effective dose, while the corresponding numbers in 2002 were 33 – 72%, i.e. the contributions from CT are higher now. On the other hand, the collective effective dose from plain radiography and radiography/fluoroscopy contributed 17 – 51% to the “TOP20” total in 2008, while the corresponding numbers from 2002 were 16 – 47% for the different countries (same range even though average has changed).

The only interventional procedure included in “TOP20” is PTCA. Obviously many other interventional procedures contribute to the total collective effective dose. The collective effective dose from PTCA ranged from 6.8 to 55.5 mSv per 1000 population between countries in 2008. The reported figures also represent a major increase from 2002 to 2008 in some countries. This might be due to some kind of underreporting of frequency (Table 3) or misinterpretation of the dose (Table 4).

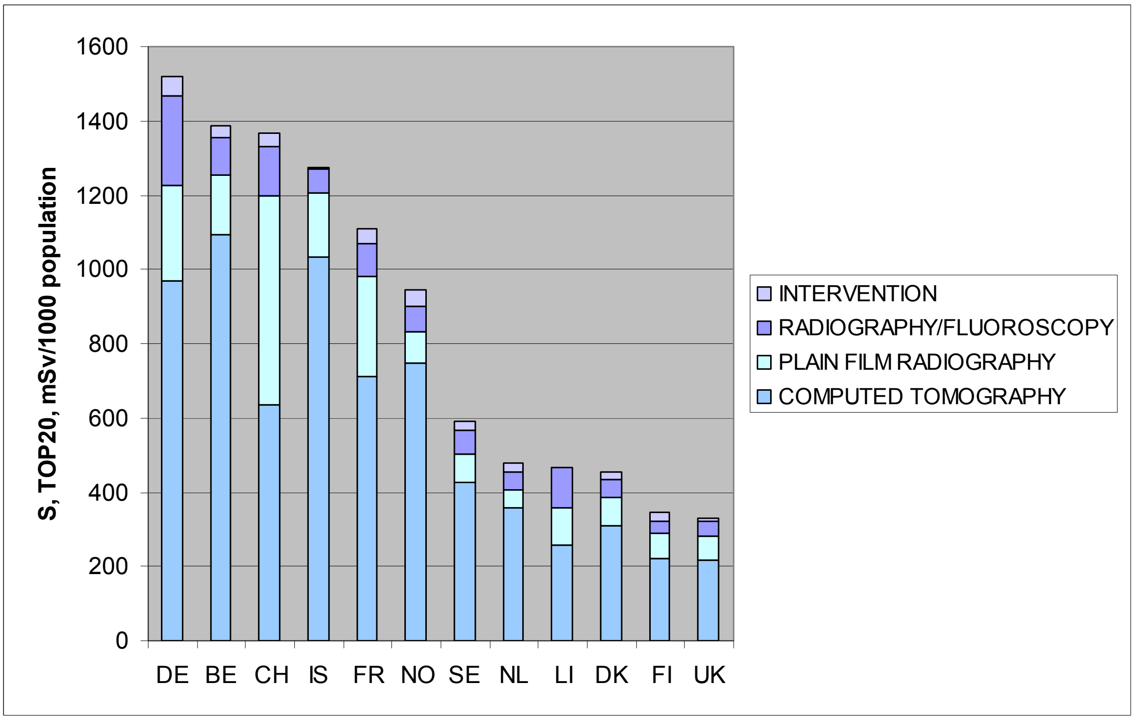

Fig. 1. Contribution of collective effective doses from different imaging modalities to the total from all “TOP20” examinations for twelve European countries anno 2008.

Discussion

All the participating countries tried to collect data as reliably as possible paying attention to the recommendations given in RP154 [1]. However, the conditions for collecting data differ between countries, and this was also foreseen in ref [1] where alternative procedures with different levels of ambition are presented. Of course that means that the uncertainties will vary with the methodology applied. There are a number of problems and pitfalls in the process of assessing the data required which are not always recognised or could not be avoided.

In the estimation of frequencies e.g. the following difficulties have to be coped with: The identification of all examinations belonging to the TOP20 categories is not straight forward. The coding system sometimes comprises the evaluation of an already performed examination, there can be local variations, there is not specified how examinations of pairs of organs are counted (one or two examinations). These problems can be overcome by checking all the details in the files with examinations and e.g. eliminate the merely administrative codes, but this can be very time-consuming.

The source from which data are retrieved has also impact on the quality of data. The majority of the participating radiation safety authorities are entitled to request data on frequency and dose from the health care sector. With the RIS/PACS rather complete data can be retrieved, but there is the risk for missing those examinations not connected to the normal RIS. This could be e.g. mammography or PTCA performed outside the x-ray department. Others have received data from the insurance companies. There the coding system usually is not well adapted to the needs for this collection of data and the registers will normally cover only a special selection of all patients.

The dose assessment could be even more cumbersome. Mostly the average dose assessed in a country for a particular examination is based on just a few measurements in some x-ray departments. Surveys have shown that the average patient dose for a specific examination can vary with a factor of 10 and more between different x-ray rooms, and taking the figures for just a few rooms will not necessarily reflect the average for the whole country.

Each of the TOP 20 categories may comprise several types of examinations, with different radiological techniques and different typical doses. However, mostly one type will constitute the vast majority of examinations within one category and using the same dose in the estimates for the remaining examinations as for the dominant one will only introduce a small error.

Despite all these possible pitfalls the data are reliable, certainly more than the data for 2002 with which comparison is done. It takes time and experience to learn how to deal with the problems of e.g. a coding system with ambiguities or assessing examinations to the TOP 20 categories. This can be achieved (and was in this study) by checking the retrieved data in detail and i.e. correct for wrong affiliation or searching alternative pathways to get the missing data.

To the results, both the examination frequency and the collective effective dose for the “TOP 20” examinations have slightly increased during the last six years. We can however not conclude with respect to the total for all x-ray based examinations. Some of the survey countries have collected information from the whole range of examinations; it has to be analyzed further how new procedures like CT colonoscopy, CT urography and CT angiography affect the total frequency and collective dose in a country. The material should be analysed in more detail, also with respect to single examinations and groups of examinations. It is for example recognised that the frequency of CT examinations has increased significantly during the last six years, but this trend has been offset by the decrease in plain film and fluoroscopy examinations. The trends in CT doses will also be analysed further, for a better understanding of how the technological development influence the doses.

Conclusions

The HERCA WG6 network consists of countries with certain traditions of population dose surveys. The radiological systems used to categorize imaging procedures in the fourteen countries have many similarities, but are very different in detailing level. The problems and uncertainties in using them for the estimates of examination frequency are recognised. Still, all participating countries managed to apply the RP154 methodology for estimating the frequency for the twenty examinations identified as the "TOP 20". Most countries also have running programs to collect doses from the radiological departments in terms of RP154 "practical dose quantities", i.e. it was possible to establish country mean values for the effective dose for the "TOP 20" examinations by means of the provided conversion coefficients and assess collective effective doses from those "TOP 20" examinations, or alternatively pick values from RP154. The list works well for comparison purposes, huge differences both in frequency and doses are identified between European countries, which should be analyzed further. However, new imaging procedures have been introduced since 2002, thus it is not possible to estimate the total collective dose based on the old "TOP 20" list. The significant efforts needed to perform a high quality population dose survey should not be underestimated.

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Medical exposure of the French population in 2007

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Introduction

The Institute for Radiation protection and Nuclear Safety (IRSN) and the National Institute for Public Health Surveillance (InVS) have been collaborating since 2003 to provide updated data on medical exposure of the French population, as requested in the European Directive 97/43/Euratom. The first report published in 2006 was based on 2002 dataset.

This study is related to 2007 dataset. It only includes ***diagnostic*** procedures from the following imaging modalities: conventional radiology including dental radiology, computed tomography (CT), nuclear medicine and interventional radiology restricted to diagnostic procedures. The objective of the study is to characterise the exposure of the French population in 2007 according to these imaging modalities, to anatomical areas and to age and sex of the patient. The part of the population really exposed for diagnostic purposes has also been estimated.

Material and methods

To characterise the diagnostic medical exposure of the population, it is necessary to estimate:

- the number of each type of examination,
- the distribution of these examinations by age and sex,
- the mean effective dose associated to each type of examination.

According to the EC 154 report, the dosimetric quantities used in this study are the “collective effective dose” and the “annual average effective dose per inhabitant”.

Each type of examination is defined by a unique code in the French Common Classification for Medical Care (CCAM). For the four imaging modalities listed above 376 codes have been studied.

For private practice, a continuous and representative sample of about 1% of the population (about 500 000 persons) has been followed by the National Health Insurance since 2006. All cares performed to people of this sample by private practitioners are registered. The 2007 data related to the 376 codes of examination studied have been analysed according to the type of examination, age and sex of the patient.

For public practice, as no global registration was available, two national surveys have been specifically conducted:

- one survey within 50 representative radiological departments in public hospitals: the total activity of each department in 2007 and dosimetric information for each examination (DAP or DLP) have been collected and analysed according to the type of examination, the sex and the age of the patient,
- one questionnaire sent to all the 127 nuclear medicine departments in public hospitals: the total 2007 activity and dosimetric information for each examination (radiopharmaceutical and administered activity) have been collected and analysed according to the type of examination.

A mean effective dose has been associated to each type of examination, using the national guidelines in radiology or in nuclear medicine, data collected to update the French Diagnostic Reference Levels, recent French and European studies and dosimetric information collected through the two national surveys.

Results

Medical exposure

74.6 million examinations have been performed in 2007. The conventional radiology represents 63% of them, dental radiology 24.7%, computed tomography 10.1% and nuclear medicine 1.6% (fig. 1).

The annual average effective dose per inhabitant has been estimated to 1.3 mSv/year/inhabitant in 2007. Computed tomography contributes to 58%, conventional radiology to 26% and nuclear medicine to about 10% of the annual dose (fig.1).

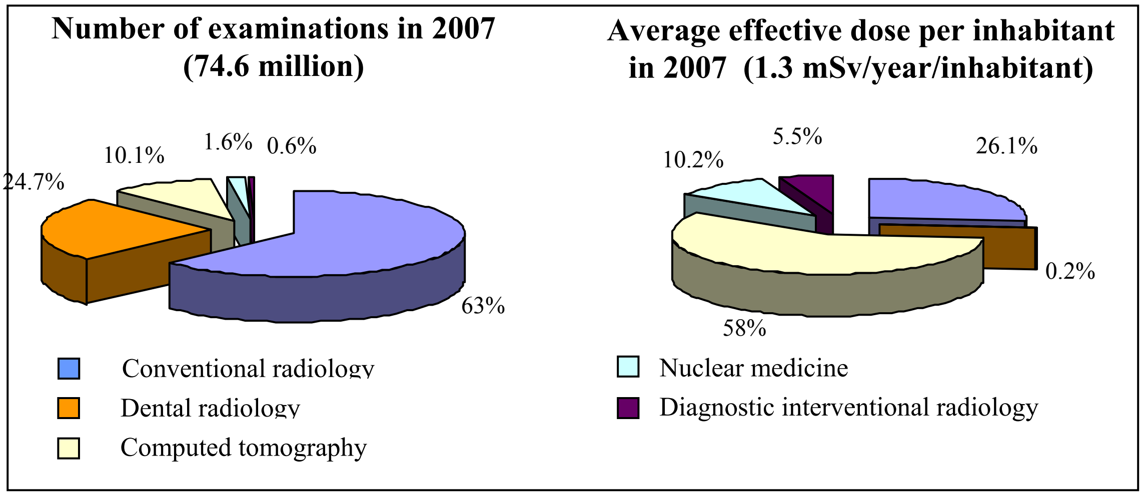

Fig. 1. Distribution of examinations and associated effective dose according to imaging modalities in 2007.

Following the recommendations of the EC Radiation Protection report n°154, the contribution of the “TOP 20 exams” has been estimated. These “TOP 20 exams” have been defined as the 20 examination types that contribute most to the collective effective dose (excluding nuclear medicine). Figure 2 shows the annual number of the “TOP 20 exams” and the associated contributions to the effective dose, per inhabitant, in 2007.

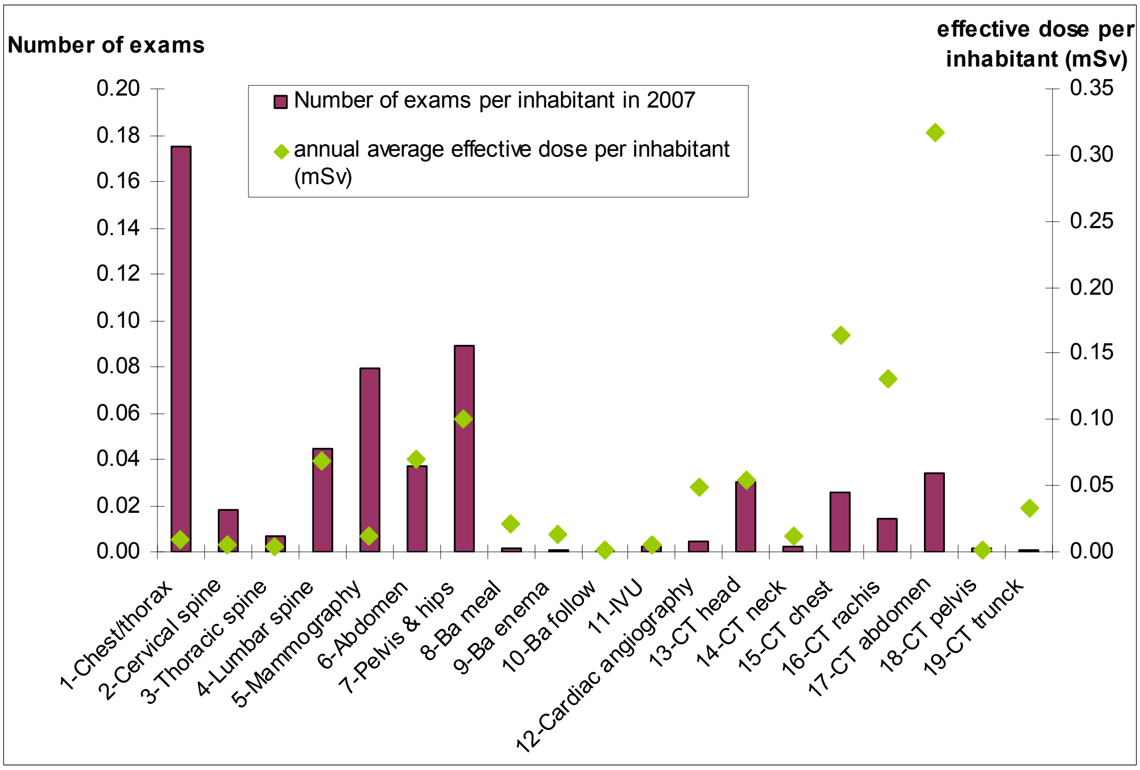

Fig. 2. Number of examinations per inhabitant in 2007 and associated contributions to the effective dose per inhabitant, for « TOP 20 exams » defined in the EC Radiation Protection report n°154.

Analysis to age and sex of the patient

For conventional radiology and computed tomography the information on sex and age of the patient was available. It was possible to study the age and sex distributions of x-ray procedures and associated effective dose (fig. 3 and 4).

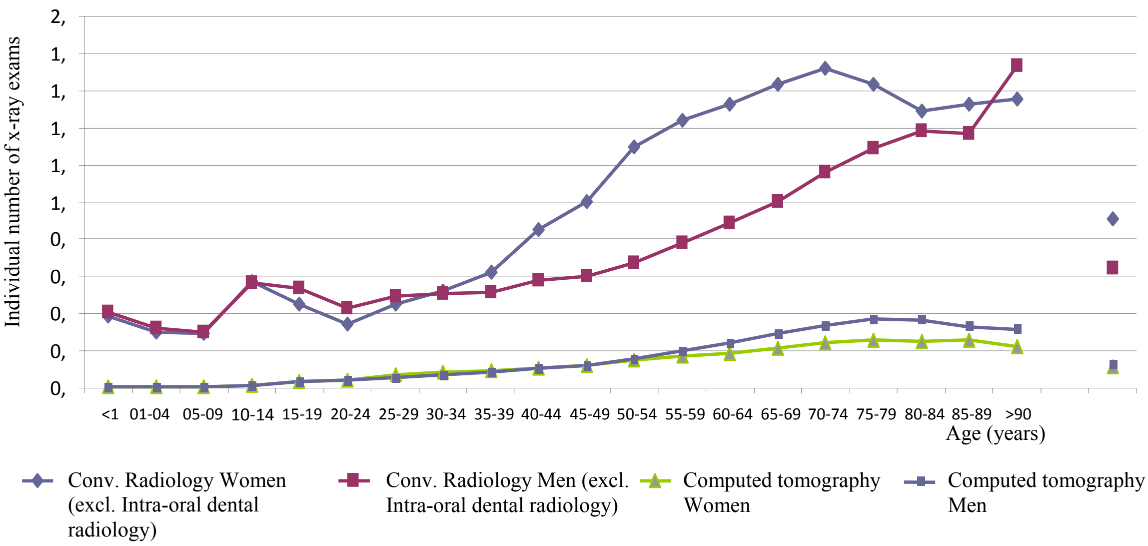

Fig. 3. Age and sex distribution of the annual number of x-ray examinations per inhabitant in 2007.

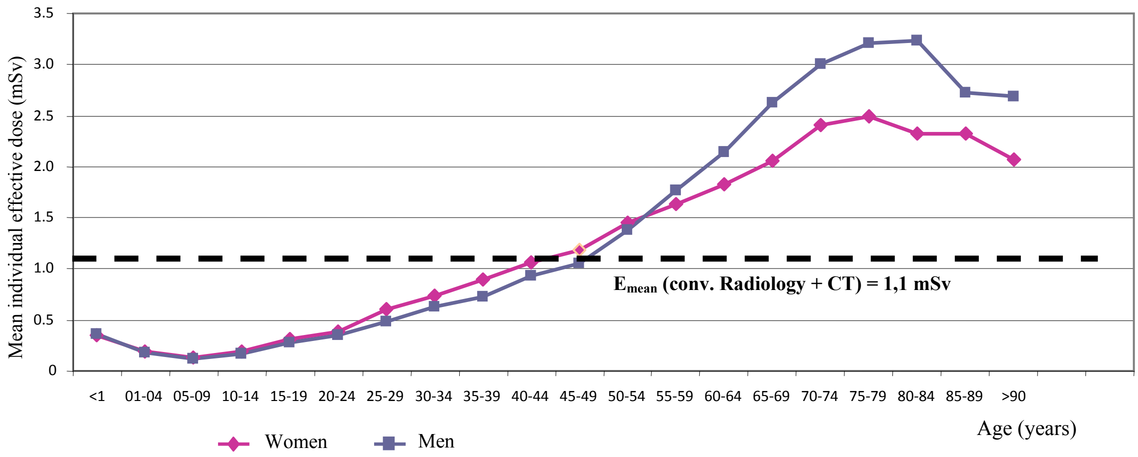

Fig. 4. Age and sex distribution of the annual average effective dose per inhabitant in 2007.

Part of the population really exposed to ionizing radiation for medical purpose

The average effective dose per inhabitant has been estimated to 1.3 mSv/year/inhabitant in 2007. Nevertheless it is obvious that this mean value does not reflect the real situation: people who did not perform any examination received no dose in 2007; other people received much more than 1.3 mSv per year because of numerous examinations. It was therefore interesting to estimate and to characterise the population really exposed.

The data to achieve this objective was only available for private practice. In 2007, the part of the French population exposed to at least one examination in the private sector has been estimated to 27.7% (radiology, CT or nuclear medicine). In terms of effective dose:

- 72.3% received no dose,
- 17.9% received less than 1 mSv,
- 6% received between 1 and 5 mSv,
- 3.8% received more than 5 mSv.

The average individual effective dose received by the 27.7% of the population exposed for medical purpose has been estimated to 2.5 mSv per year.

Discussion

The annual average effective dose per inhabitant due to medical exposure increased by a factor of 1.6 between 2002 and 2007 (0.83 vs 1.3 mSv/year/inhabitant). The main reasons to explain this large increase in dose in that period are:

- a better knowledge of the examination frequency due to the CCAM classification,
- a large increase of the number of CT and nuclear medicine examinations (+ 26% and +38 % respectively),
- a more important part in 2007 of CT exams that exposed chest, abdomen and/or pelvis, organs that highly contribute to the effective dose,
- a large increase of PETSCAN examinations.

Conclusions

The annual average effective dose per inhabitant has been estimated to 1.3 mSv/year/inhabitant in 2007 in France. This value is lower than the one of the Unites States (3 mSv per inhabitant in 2006) and in the range of the European values published: from 0.4 mSv in the UK to 2 mSv in Belgium (EC radiation protection report n°154). The method developed in this study ensures reproducibility and accuracy of assessment of medical exposure. Finally, it could contribute to assess patients' protection policy.

All these results have been published in March 2010 and the report can be downloaded: www.irsn.org or www.invs.sante.fr.

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Feasibility study of a simplified method to determine population doses from medical x-ray imaging

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Abstract

Medical X-ray imaging has been the largest man made source of population doses for many years, and is rapidly increasing due to an increased use of computed tomography. It is important to obtain information about the contribution from different examinations to these doses in order to select appropriate radiation protection actions. Guidance for assessing population doses is provided in a publication of the European commission with amongst others a simplified method where the collection of data is restricted to a sample of representative hospitals for those 20 x-ray examination categories contributing mostly to the population dose. The purpose of this study was to check the feasibility of this approach.

The national coding system was used to identify the 20 categories. From a sample of hospitals (around 20 % of the country) frequency data was collected from the radiological information system (RIS). Problems envisaged were differences in the code system between hospitals and that one category could include several codes. This was solved by editing the data manually. The frequencies were scaled up to represent the whole country using information from a complete assessment from 2005. Dose values were collected from national dose surveys and when not available from tabulated values in the guidance.

The population dose was calculated to 5700 manSv. The contribution to the population dose from examinations not included in this figure was estimated to be less than 10 %. The total uncertainty of the population dose was expected to be relatively low. This could be achieved with a modest workload, using a rather small sample. However, due attention has to be paid to the ambiguities in the coding systems, to the selection of the sample of hospitals and to the methods of extrapolation to a national value.

Estimation of exposure from medical exposure

The importance of estimating exposure to the populations from different sources has been emphasized in the radiation protection community, among others by UNSCEAR [1]. UNSCEAR has evaluated the exposure of the population on a global level. There it is shown that a large part of the radiation dose from artificial radiation sources comes from medical exposures. This figure has increased significantly in some countries [2]. Evaluating level and trends of these doses from medical exposures is thus an important issue that can help to set priorities for radiation protection activities.

The EU directive MED [3] requires that the member states must assess the distribution of radiation doses from medical exposure for the population and for relevant reference groups, without specifying in more details how this should be carried out, e.g. what quantity to be used and which relevant reference groups were envisaged. To use collective effective dose [4] is one option. This quantity has been introduced in order to describe the radiation to a group of persons. Collective effective dose has been used to compare the radiation dose incurred by different sources, often on a national level. However, the quantity has to be used with care, the group of persons included could be exposed very unevenly when most of the radiation dose is received by only a few people. The collective effective dose alone does not say anything about the dose distribution within the group. Therefore this figure has to be complemented with other information in order to be useful. The results of an assessment of collective doses may be dependent on the methodology.

In order to calculate the collective effective dose from medical exposures the numbers of examinations have to be known together with the effective doses. The ideal method would be to assess the effective dose for every patient for all single examinations and to sum up these figures. However, this is not feasible; the amount of data to be handled must be reduced. One method that has been proposed is to restrict the dose estimate to those examinations that contribute most to the collective effective dose. This method has been developed in a European project, called DOSE DATAMED, aiming at proposing unified methods that can be used in Europe [5]. One of the methods proposed there is called TOP 20. In this method, 20 examination categories contributing most to the collective effective dose are identified and the assessment of the total collective effective dose is restricted to these 20 categories. The frequency of these categories of examinations performed in the country has to be assessed and the average effective dose for each of them be estimated. More details about this method are found in [5].

The TOP 20 method was applied in Sweden. This presentation addresses some key issues: the importance of specifying and categorizing the examinations, difficulties and pitfalls encountered in data collection and uncertainties concerning effective dose and frequency.

Material and methods

Description of categorization

In 1991 the National Board of Health and Welfare in Sweden issued a coding system (KRÅ) for all radiological examinations that were performed at that time [6]. The classification was intended to be used for registration of radiological examinations e.g. in the radiological information system (RIS). The first task was to identify which codes in the Swedish KRÅ are corresponding to each of the TOP 20 examination categories in [5], listed in table 1. This task was performed by an experienced radiologist.

Table 1. The TOP 20 list as defined in the [5].

TOP 20 categories		
Radiography 1 chest/Thorax 2 Cervical spine 3 Thoracic spine 4 Lumbar spine 5 Mammography 6 Abdomen 7 Pelvis & hip	**Flouroscopy** 8 Barium meal 9 Barium enema 10 Barium Follow 11 Intravenous Urography (IVU) 12 Cardiac angiography	**Computer tomography (CT)** 13 CT head 14 CT neck 15 CT chest 16 CT spine 17 CT abdomen 18 CT pelvis 19 CT trunk **Interventional Radiology** 20 Cardiac dilatation/stenting (PTCA)

Frequency collection

Frequency data from all radiological examinations during 2008 were collected from a sample of hospitals in Sweden. From each hospital, data was retrieved from the local RIS comprising local RIS codes, examination description and number of examinations. Most, but not all local RIS codes corresponded to the KRÅ codes. Non-matching codes could be identified by manually checking the description of examination given together with the code. With this time consuming procedure a rather complete estimate of the frequencies for the TOP 20 examinations could be performed. However, some data are missing in the reporting, for those examinations that are not always registered in the normal RIS, e.g. screening with mammography and PTCA outside the radiological department. Frequency data from the sample was scaled up to represent the whole country by using information from a national survey from 2005 [7], which included all examinations and hospitals in Sweden. The scaling factor was determined by taken the data from 2005 and calculate the ratio between the number of examinations from the sample of hospitals and the total number of examinations from the whole country.

Unlike in the data collection from the sample in 2008 the national survey in 2005 did provide reliable figures for the frequency of mammography examination. Therefore the frequency data from the 2005 survey was taken for the year 2008. For comparison frequency data from the 2005 survey was used to create a corresponding TOP20 list for 2005. Also for PTCA and cardiac angiography a national reporting system for these procedures exists [8] including frequencies and doses, expected to be more reliable than the data from the sample.

Doses

The radiation safety authority has established diagnostic reference levels for 12 examinations [9]. Both in 2005 [10] and in 2008 [11] all Swedish hospitals reported the patient doses for these 12 radiological examinations, as the average for approximately 20 normal sized patients for every examination room. These examinations correspond to 11 of the TOP 20 categories and hence the average of the radiation doses reported was taken as the national dose value for the respective examination. For the remaining 9 categories, tabulated dose values from the guidelines [5] were used. The effective dose was derived by multiplying each dose figure with the corresponding conversion factor in table 14 in the report. [5].

Results

Categorization

In total 84 KRÅ codes were found to correspond to the TOP 20 categories. The number of codes in one category varied between 1 and 16. The KRÅ system did not contain any code for CT-trunk, the category number 19 of the TOP 20. However, some hospitals reported local RIS codes for complex examinations, i.e. multitrauma which were added to the CT – trunk category.

Frequency

Data from 20 hospitals were collected. These hospitals represent 4 out of 5 hospital categories available in Sweden, table 2. In the 2005 survey, these hospitals performed 18 % of the radiography/radioscopy, and 21 % of the CT examinations in Sweden.

Table 2. Total numbers of different categories of health care entities in Sweden and the numbers used for the sample.

Hospital category	Number collected	Total number in Sweden
University Hospitals	1	8
Central (county) Hospitals	4	18
Small Hospitals	4	55
Health Centers (small radiological units)	0	28
Private radiological units	11	26

The data for all the TOP 20 categories from 2005 and 2008 showed that the total number of examinations increased with 3% during this time period. However, this figure varies considerably between the different radiological techniques. Plain radiography examinations and fluoroscopic decreased with 1% and 32%, respectively. On the other hand CT and PTCA increased with 28% and 8% respectively. In total, approximately 3.5 million TOP 20 examinations have been performed in Sweden during 2008. Approximately 74% of these were plain radiography examinations, 3% were fluoroscopy, 22% were computed tomography (CT) and 1% coronary angioplasty. In the 2005 survey, the TOP20 examinations represent 71% of the total number of examinations. Table 3 shows the frequencies for the different modalities for 2005 and 2008.

Table 3. Number of TOP20 examinations performed with different modalities in 2005 and 2008.

Frequencies			
TOP 20	**2005**	**2008**	**Difference (%)**
Plain radiography	2627120	2611702	-1
Fluoroscopy	173238	117749	-32
CT	600650	765998	28
PTCA	17671	19120	8
Total	3418679	3514569	3

Collective effective dose

For approximately 90% of the total collective dose national data was used and for the remaining 10% tabulated data was used from the guidelines [5]. A comparison between the reported data from 2005 and 2008, show that the collective doses for plain radiography and fluoroscopy have decreased with 19% and 36%, respectively. This is due to a decrease of the number of examinations and a decrease of the mean dose values (see table 4). The collective dose for CT and PTCA has increased during these three years with 30% and 8%, respectively, mainly because of the increased number of examinations. The mean dose values for CT have decreased slightly (see table 4). The dose data for PTCA was available for 2008 in [8], the same value was taken for 2005.

For all examinations the dose per examination has decreased or remained unchanged, but the total collective dose increased with 8% from 2005 to 2008. This can mainly be attributed to the increased number of CT examinations.

The total collective dose for the TOP 20 examinations in 2008 is approximately 5700 manSv, and 11% of the total collective dose is caused by plain radiographic examinations, 12% by fluoroscopy, 72 % by CT examinations and 5% by PTCA. CT of the abdomen is the single examination contributing most to the total dose, to 40% of the total collective dose. The collective dose from the TOP 20 examinations is estimated to represent 90% of the total collective dose from all examinations in Sweden.

The details of the distribution of the collective doses and how they have changed since 2005 are shown in table 4. The changes in dose are presented as those resulting from change in frequencies (Δfreq) alone (column 3 & 4), from change of dose (Δdose) per examination alone (column 5 & 6) and from both of them (column 7 & 8).

Table 4. Collective doses from 2005 and 2008 including separate contribution from changes in frequency and DSD.

Collective dose (manSv)							
Exams	**2005**	**2008 due to ∆freq**	**Diff (%)**	**2008 due to ∆dose**	**Diff (%)**	**2008 due to ∆freq and ∆dose**	**Diff (%)**
Plain radiography	863	835	-3	730	-15	699	-19
Fluoroscopy	1023	833	-19	812	-21	658	-36
CT	3142	4261	36	3020	-4	4093	30
PTCA	247	268	8	247	0	268	8
Total	5274	6197	17	4809	-9	5718	8

Uncertainties

It was easy to get reliable data for those examinations that were stored in the local RIS. It took some effort to stratify the data, i.e. to decide which examinations belong to which TOP 20 category and to eliminate codes for administrative measures. For those examinations the uncertainty is expected to be low. It was obvious after the collection of data that figures for mammography were missing. For mammography the values from the 2005 survey were used also for 2008, because there are no indications that the frequency of examinations should have changed considerably in this time period, no screening activities have been stopped and no have been added. For coronary angiography and PTCA the collected frequency data from a national quality system were used. This system comprises all examinations/procedures performed in the country resulting in low uncertainties. The scaling of the sample to the whole country is inevitably introducing uncertainties. Collected values for CT trunk are unreliably since there are no national codes for complex examinations. Some hospitals have introduced local codes.

Some of the uncertainties in dose estimation are due to the fact that 10% of the collective dose is assessed using the dose figures listed in [5], which might or might not be representative for the Swedish practices. In addition, the national doses do not cover all KRÅ codes for the various TOP 20 categories.

Feasibility of the TOP 20 approach

This study has been performed with confined resources, where data from a limited number of hospitals was analyzed and an extrapolation to the national level was performed. The uncertainties from the assessment of the frequencies originate from: The uncertainty of the reported frequencies in the sample and the uncertainty arising from the scaling to the entire country. For the majority of the TOP 20 categories the reported values are accurate. However, as pointed out above, the data had to be checked in detail in order to be able to remove e.g. administrative codes not representing the conduct of an examination. There are also large uncertainties for the category “CT trunk” for which no national code exist. The problem with underreporting of data for mammography, coronary angiography and PTCA has been overcome by using other sources. For example the PTCA frequencies would otherwise have been underestimated by 28%.

We believe that this study was performed with relatively low uncertainties. Due to the earlier performed nation wide survey a small sample could be used for the study without introducing considerable uncertainties. A nation wide survey including all examinations and all hospitals should be performed in regular intervals, 5 to 7 years. That not only enables facilitates study as the present one but can also be used for a revision of the TOP 20 list which is, as shown in this study, not static at all

Although a national code is used specifying different examinations it is not unambiguous what to include to the TOP 20 list of examinations, this interpretation will contribute to the overall uncertainties in the suggested method. Even though the national coding system is rather logical, different hospital within Sweden used the codes differently. When standardized evaluation method, e.g. a computer reading of the data is used, systematic errors in the frequency data can be introduced. Thus it is important to both manually check the data and actually enquire how the hospital has used the national coding system. The example of coronary angiography and PTCA showed that there could be other sources for data on frequencies and doses than those commonly used. It could be worth while to look for similar national data bases that might exist for special procedures.

The confidence in and uncertainties of the result heavily depend on available dose data. In this study, the dose data from national dose surveys was used for examinations contributing to 90% of the total collective effective dose from the TOP 20. These dose surveys reveal a large range of average doses between hospitals. That implies that there is a large uncertainty when the doses from only a few hospitals are used as the dose for calculating the collective effective dose. The same argument holds for the use of tabulated dose values that will also introduce a large uncertainty.

This study shows that the simplified “TOP 20” method can give reliable results. Of great help for this achievement was the availability of an earlier nation wide survey and that national dose data exist for the most important examination.

Conclusions and future work

In this study we show that the total examination frequency remained almost constant and the collective dose increased with about 8 %. This is due to that the decrease of both frequency and dose for conventional x-ray examinations is counterbalanced by the increase of the frequency of CT examinations. The TOP 20 method proved to be adequate for providing the data needed with sufficient accuracy,

In the future such surveys could be made easier by using the data at the hospitals more efficiently. The data system in the hospitals already contains all information needed, but in slightly different technical platforms. Some hospitals make use of this information internally already for there own planning and optimization work. It could be useful for the estimation of collective effective dose to have a common system with validated radiation dose data and frequency of well defines examinations.

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Paediatric computed tomography (CT) exposure and radiation induced cancer risk: setting up of a French cohort study

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Abstract

Introduction: The increasing use of computed tomography (CT) scans for paediatric population raises the question of a possible impact of such ionising radiation (IR) exposure on the occurrence of secondary leukaemia and other cancers. We describe the preliminary results of an ongoing large-scale cohort study of cancer risks among children exposed to CT scan in France, performed in collaboration with the "Société Francophone d'Imagerie Pédiatrique et Prénatale" (SFIPP). This study will be included in a trans-national collaborative cohort study for long term follow-up.

Material and methods: In 18 paediatric radiology departments, demographic and IR exposure information concerning children less than 5 years, who underwent at least one CT scan between 2000 and 2006, are under collection. Absorbed organ doses are calculated with the "CT-expo" software.

Results: Until now, 28,190 children free of cancer or leukaemia at the first CT scan examination have been included. Age at first CT scan exposure was less than 1 year for 42%, 1 to 2 years for 19%, and 13% per year of age up to 4 years old. The mean number of CT examination per child was 1.5 (min 1, max 30) and concerned head in 66% of the cases, thorax in 22%, abdomen and pelvis in 10% and other localisations in 2%. Highest cumulated organ doses were observed for brain and lens during head exposure and mean absorbed doses were 27 mGy (range: 2.8 – 478 mGy) and 30 mGy (0.5–620 mGy), respectively.

Conclusion: This cohort allows to better characterizing organ doses associated with CT scan exposure in childhood. Relatively high doses to radiosensitive organs (lenses, ovaries, breast, etc...) have been observed, as well as quite large dose ranges according to the protocols used. This underlies the need for optimization of paediatric CT protocols, or even standardization. The follow-up of the cohort will be first based on cancer incidence up to age 15, and on mortality afterward to quantify a possible excess cancer risk.

Variing CT settings to decrease patient dose in PET/CT studies without degradation in attenuation correction

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Abstract
PET/CT hybrid devices are using CT for attenuation correction as well as for anatomic or/and diagnostic purposes. To gain optimal images, the companies install fixed protocols for PET/CT which is not optimized in every case.

This pose the question, how CT parameters can be varied, particularly voltage and current, with the aim to lower patient dose but preserving useful attenuation correction.

To measure the effects of different CT parameter settings on the PET image we used a self constructed "Attenuation Phantom". This phantom consists on eight hollow acrylic cylinders with different diameters and several bones (two femoral, a spine with rips and a pelvic bone). The Scanner we used was a Siemens Biograph PET/CT.

First, the phantom was filled with 18F, whereas the cylinders, due to NEMA conditions - received eight times more 18F than the remaining phantom volume. A PET-scan was performed with and without attenuation correction based on a CT-scan. Following CT parameters were used, whereas the first four settings were carried out with care dose and the last four settings with minimum current-time-product (mAs):

140 kV/138 mAs; 120 kV/138 mAs, 100 kV/138 mAs, 80 kV/138 mAs, 140 kV/30 mAs; 120 kV/27 mAs, 100 kV/13 mAs, 80 kV/8 mAs.

The results demonstrate that the setting parameters have little to no effect on the qualitative and quantitative output of the attenuation correction, hence the calculated activity concentration. Therefore a reduction of both current-time-product and voltage are justified for the sake of lowering the effective dose to the patient for medical examinations where a high-quality diagnostic CT is not necessary.

Introduction
The simultaneous imaging of anatomic features and metabolic activities with Positron tomography (PET) and Computer tomography (CT) in one device is state-of-the-art for numerous medical examinations. One of the properties of the CT is the attenuation correction, a quantitative technique to correct images for the effects of attenuation. Older PET-scanner used line sources (e.g. 68Ge), which rotated around the analyzed volume. Today this expensive and time-consuming method is replaced by the CT.

PET/CT devices mostly rely on the setting parameters of the CT for high-quality resolution which on the other side is not justified by using CT just for attenuation correction.

This work aims to establish guidelines relating to parameters at PET/CT-scanners for attenuation correction with the objective to limit unnecessary exposure of the patient. This is done by applying different setting parameters to the CT and their subsequent use for the attenuation correction.

Material and methods

Attenuation coefficients of analyzed areas of a human body have to be simulated with a special phantom. In practice, the CT measures attenuation coefficients which are used for the correction of the activity concentration at the PET. Therefore a phantom for body simulations at PET/CT-devices should have both scatter characteristics like the human body as well as contrast elements with strong different attenuation coefficients, such as lung equivalent and compact bone.

Existing phantoms in nuclear medicine were designed for quality assurance and are just partial suitable for the purpose of this work. The NEMA PET-Body – phantom [1], [2] comes close to these requirements, but has to be modified. It consists of six hollow spheres of acrylic glass and a cylinder filled with polystyrene-balls which simulate the lung. For testing attenuation correction over the complete phantom volume, the spheres are replaced by hollow acrylic cylinders in different diameters (10 mm; 15 mm; 25 mm and 38 mm), so that the respective volumes span the entire phantom's height . For the measurements these cylinders and the remaining phantom volume are filled with 18F, the concentration in the cylinders being eight times as high as for the remaining phantom volume due to NEMA conditions. To simulate bone density three backbone segments from a pig are placed in the centre of the phantom with a 10 mm acrylic hollow cylinder in its midst. At the edge of the phantom two pig's thigh bones with pelvic bones are situated (Fig.1).

The cylinders were filled with an activity concentration of 51 kBq/cm^3 whereas the remaining phantom volume contains 5.9 kBq/cm^3 at the time of measurement (concentration ratio: 8.6:1). The measurements included three series where each run started with a Topogram (overview screen) followed with four CT-expositions with respectively fixed voltage and four different currents and ended with a PET-scan. Afterwards the evaluation of the PET-scan was carried out with and without attenuation correction.

A Siemens Biograph 64 PET/CT Hybrid device was used for the measurements. 16 different setting parameters of the CT were chosen, including the standard setting for attenuation correction (140 kV; 30 mAs) which is referred to as the "Care dose" (Table 1).

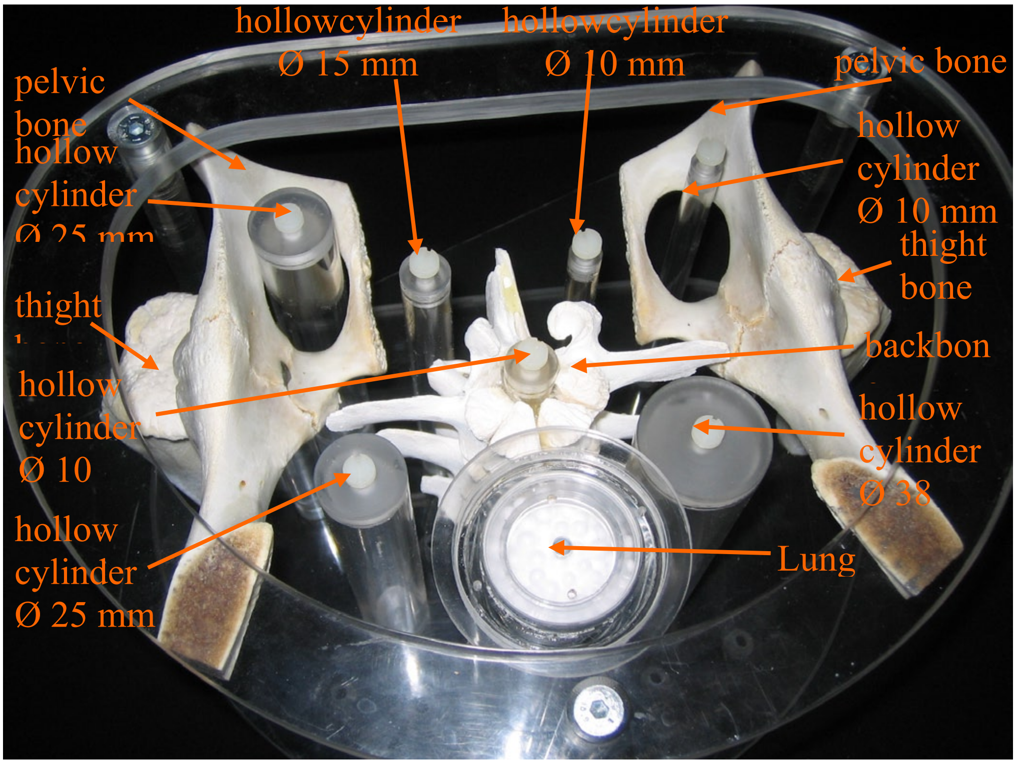

Fig. 1. Phantom with hollow acrylic cylinders, the lunge-equivalent, the three backbone segments in the middle and the two thighbones with pelvic bones at the edge.

Table 1. Setting parameters of the different CT-serials.

	Voltage	Current	Pitch
1.	140 kV	138 mAs	1,5
2.	140 kV	100 mAs	1,5
3.	140 kV	65 mAs	1,5
4.	140 kV	30 mAs	1,5
5.	120 kV	138 mAs	1,5
6.	120 kV	95 mAs	1,5
7.	120 kV	51 mAs	1,5
8.	120 kV	27 mAs	1,5
5.	100 kV	138 mAs	1,5
6.	100 kV	95 mAs	1,5
7.	100 kV	51 mAs	1,5
8.	100 kV	13 mAs	1,5
9.	80 kV	138 mAs	1,5
10.	80 kV	95 mAs	1,5
11.	80 kV	51 mAs	1,5
12.	80 kV	8 mAs	1,5

The analyses of the PET measurements were carried out with Imagej (Vers. 1.41c). In every slice inside the cylinders as well as in the remaining phantom volume a ROI (Region Of Interest) was constructed, so that fringe effects can be avoided. Inside the constructed areas the means and standard deviations of the activity concentration were calculated.

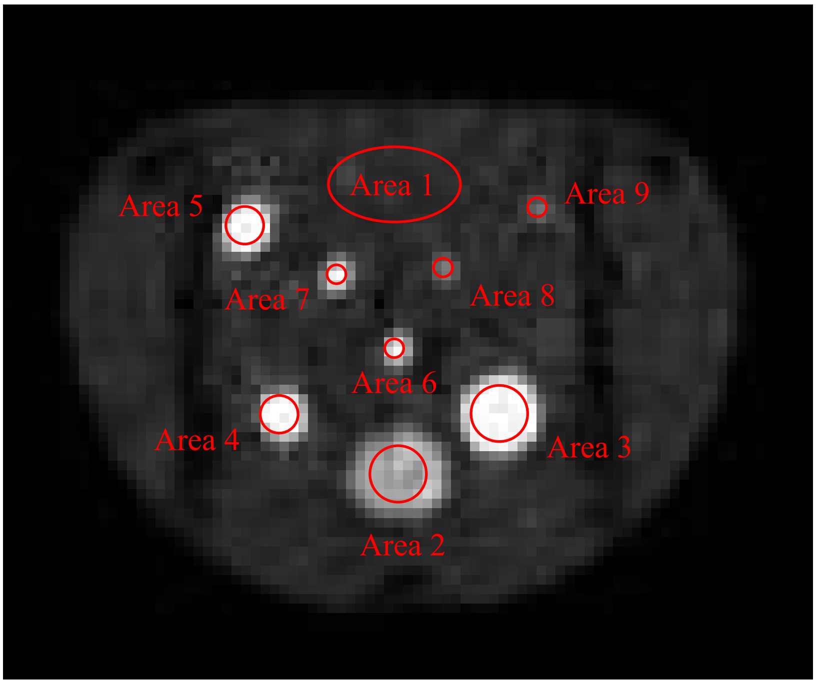

Fig. 2. Attenuation corrected PET-pictures (140 kV; 30 mAs, Slice 48) with ROI`s for calculating the respective means and standard deviations of the activity concentration. Area 2-9 are the selected ROI`s of the cylinders and Area 1 represents the remaining phantom volume. The two dark lines origin from the pelvic bones due to the zero 18F accumulation of the bones.

Results

Surrounding tissue of a radioactive source, especial high density tissue, can falsifies the true activity concentration of an area and even the form of the area them self. The optimum attenuation correction is given by exact reconstruction of the form of the area and the activity concentration in it. A first analysis of the results shows that a smaller cylinder, filled with ^{18}F, correlates with a worsening of the attenuation correction. This effect gets even worse with bone structured surrounded areas. Fig. 3 displays this effect, where area 8 and area 9 with Ø 10 mm are reconstructed with very little activity concentration according to area 3 with Ø 38 mm. Area 1,3,4 and 5 are also relatively homogenous over the entire scan length in comparison to Area 6,7,8 and 9. This is due to the insufficient attenuation correction of the cylinder areas with small diameters (< 15 mm) and the partial-volume-effect, wherein different tissue types are mixed within a pixel (resolution of the PET-scanner: 4 mm per pixel). However the setting parameters of the CT cannot be accounted for this effect as seen in Fig. 4 where the attenuation corrected images are shown with the highest and the lowest setting parameters. There the maximum deviation of displayed activity concentration in the same ROI and the same slice is less then 6 %!

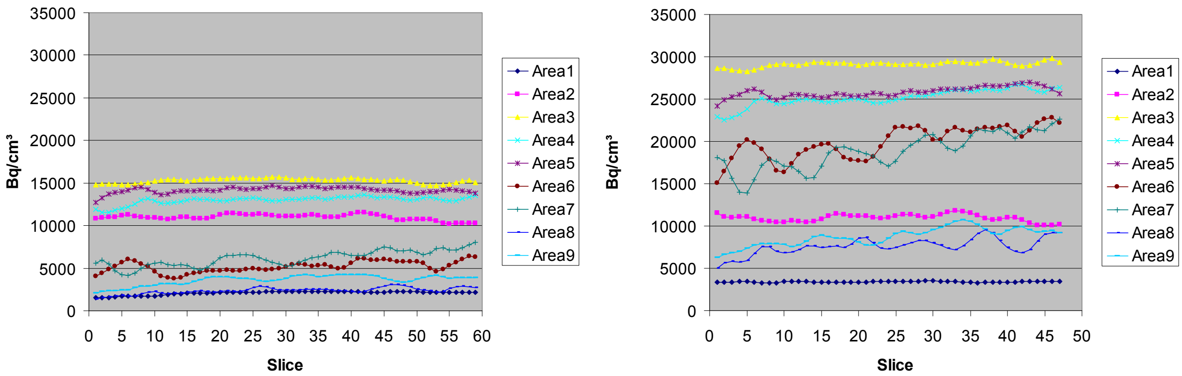

Fig. 3. Left: uncorrected PET-image. Right: PET-Image left attenuation corrected (140 kV; 30 mAs). This Image consists of fewer Slices due to the interpolation algorithm.

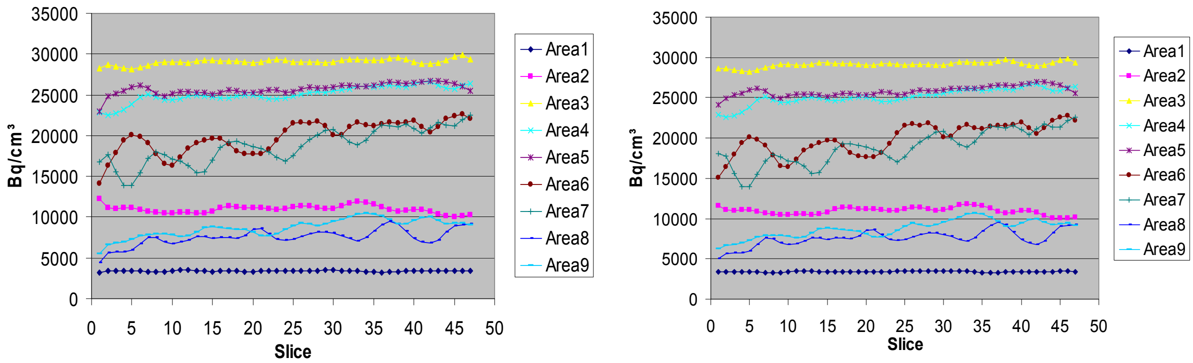

Fig. 4. Left: attenuation corrected PET-image (140 kV; 138 mAs). Right: attenuation corrected PET-Image (80 kV; 8 mAs).

On the basis of the computed tomography dose index (CTDI) which is given by the respective CT software for every setting parameter it is possible to estimate the effective dose of a patient 0,0. 0 illustrates the relation between voltage, current-time-product and the resulting effective dose for pelvic CT images of a male patient. One can clearly see the linear ascent of the effective dose as a function of the current-time-product with voltage determining the steepness.

So in order to reduce patient dose i.e. lower the effective dose to a value less than the one which corresponds to setting parameter of the "Care Dose" (140 kV and 30 mAs) one has to stick to the setting parameters displayed in Tabble 2. For example, at a voltage of 140 kV the current-time-product has to be adjusted to ≤ 30mAs.

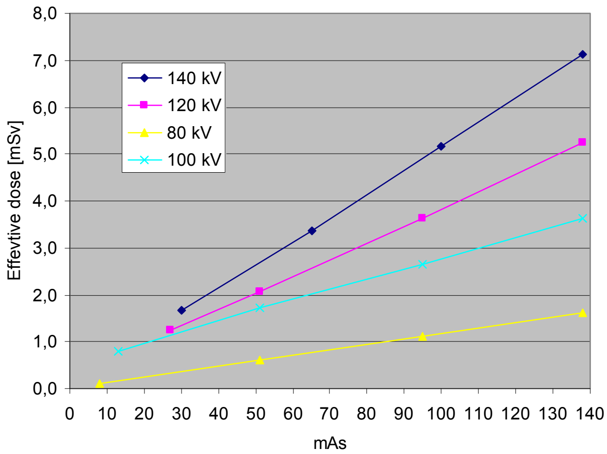

Fig. 5. Estimated effective dose of a female patient for pelvic CT images with different setting parameters. Each line represents a different voltage.

Table 2. Setting parameters to reduce patient dose at Siemens Biograph 64 PET/CT hybrid device.

Voltage	Current-time-product
140 kV	≤ 30 mAs
120 kV	≤ 39 mAs
100 kV	≤ 48 mAs
80 kV	≤ 139 mAs

When selecting the lowest possible voltage and current-time-product a decrease of the effective dose of 92 % can be achieved. In the case of a pelvic exposure of a male patient at a Siemens Biograph 64 PET/CT Hybrid device this corresponds to a reduction of the effective dose from 0,7 mSv to 0,1 mSv. However it should be noted that the effective dose strongly depends on the exposed body region, sex, age, weight and other factors., Nonetheless its reduction as suggested above is independent of these parameters. Finally it also must be mentioned that a PET/CT hybrid device also can be used for anatomical imaging and therefore changing the setting parameters can yield unworkable images.

Discussion

This project demonstrated that the reduction of patient dose is possible by decreasing setting parameters of the CT for attenuation correction purposes using a Siemens Biograph 64 PET/CT hybrid device. The variation of the setting parameters has proofed to have very little effect on the quantitative outcome of the attenuation correction.

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Trigger Levels to prevent tissue reaction in interventional radiology procedures

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Abstract
It is well known that the large use of fluoroscopy in interventional radiology procedures may induce unintended patients' skin injuries. For that reason assessment of skin dose for these procedures is getting more and more important.

Aim of the study is to investigate the role of cumulative air kerma (CK) as on-line dose indicator and to evaluate the pos-sibility to define a local CK trigger level which can help opera-tors to identify situations with high probability to exceed a peak skin dose of 2 Gy, the threshold dose for transient skin erythema. Cerebral angiography, aneurysm embolisation and chemoembolisation of liver cancer have been identified as the interventional procedures where high skin doses could be delivered.

Dosimetric data (CK, air kerma area product (KAP) and fluoroscopy time) have been collected in a sample of procedures and peak skin dose (PSD) have been measured from dose distributions measured with large area radiochromic films (Gafchromic, IPS, USA) located between table top and patient.

PSD varied in very wide range and in a few cases were close to the threshold for main erythema and epilation (6 Gy). The correlations between PSD and CK have been successfully assessed for each procedure type and a trigger level for CK has been derived to alert the interventionalist on the probability to have reached a PSD of 2 Gy.

In our center local trigger levels of 5200 mGy and 2500 mGy has been established for brain aneurysm embolisation and chemoembolisation procedures respectively.

As suggested by ICRP in the publication No. 85, a follow-up for patients whose estimated peak skin dose was 3 Gy or greater has been implemented as a routine practice.

Introduction
The extensive use of fluoroscopy in interventional radiology procedures may induce unintended patients' skin injuries varying from erythema to necrosis. Some cases have been described in a review paper by Koenig et al.

As stated by the European regulation 97/43/Euratom patient dose should be periodically evaluated to guarantee optimisation and justification of the practice.

Moreover the ICRP (International Commission on Radiological Protection) recommends to provide an adequate follow-up and eventual treatment of these injuries, for patients whose skin dose has been 3 Gy or greater.

Maximum skin dose (MSD) could be directly measured using film or thermoluminescent dosimeters (TLDs). In this way the operator doesn't have an immediate knowledge on the amount of dose received by patient. It is important to perform an "on-line" evaluation (e.g.: during the procedure) of patient's skin dose to individuate those procedure in which it could be greater than threshold for deterministic effects. Therefore dosimetric indicators for estimating and monitoring patient skin dose in routinely practice should be individuated.

For interventional radiology procedures kerma-air product (KAP) and cumulative air kerma (CK) at the interventional reference point (IRP) could be used as dosimetric indicators for skin dose.

Aim of this work is investigating the correlation between these parameters and the skin dose directly measured in a sample of procedure.

The dosimetric indicator that better correlate with the MSD could then be used to define trigger levels that indicate the overcoming of threshold for deterministic effects and necessity of medical follow-up for possible radiation injuries, respectively.

Material and methods

In the period October 2007 – October 2008 data from procedures, both diagnostic and interventional, performed in two interventional rooms in Udine University Hospital (Udine, Italy) have been collected.

The procedures were performed with two angiographic system equipped with a digital flat panel detector by four experienced radiologists.

The peak skin dose (PSD) has been measured with radiochromic films (Gafchromic XR-typeR, IPS, USA), placed between patient and couch in a sample of 61 procedures. Radiochromic films have been calibrated under an angiographic beam for comparison with an ionisation chamber (Radcal, Model 2026C 6cc ion chamber).

Films were read with a flatbed scanner (Epson 1680pro) in reflective mode. Images obtained were evaluated in terms of dose with a Matlab home-made routine.
The correlations between PSD and KAP and PSD and CK have been investigated for each procedure type.

Results

Data collected are summarized in Table 1, in which are reported: type of procedure, number of patients, mean values of fluoroscopy time (FT), kerma-area product (KAP) and cumulative air kerma (CK) at IRP.

Table 1. Mean and SD fluoroscopy time (FT), air kerma-area product (KAP) and cumulative air kerma (CK) for a sample of interventional procedure performed at Udine University Hospital in the period October 2007- October 2008.

Procedure	No	FT (min)	KAP ($Gycm^2$)	CK (mGy)
Cerebral Angiography	197	6.1 ± 8.2	71.1 ± 49.2	770.9 ± 887.4
Aneurysm Embolization	76	26.6 ± 13.5	135.4 ± 60.8	2153.7 ± 1345.3
Chemo-embolization	144	14.1 ± 7.7	210.5 ± 138.6	1136.3 ± 767.9
Embolizations	57	26.2 ± 41.6	269.7 ± 320.8	1384.7 ± 1472.0
Peripheral Angiography	145	1.4 ± 1.9	43.4 ± 29.3	154.5 ± 106.1
Lower limb angioplasty	44	15.6 ± 9.9	24.7 ± 37.6	149.0 ± 237.6
Carotid angioplasty	73	9.4 ± 5.5	53.7 ± 26.0	247.3 ± 135.7
Iliac angioplasty	45	11.4 ± 9.8	80.5 ± 89.5	401.9 ± 293.8
Below-knee angioplasty	27	17.9 ± 10.4	8.9 ± 14.3	101.8 ± 326.0
Renal angioplasty	12	7.7 ± 3.5	48.8 ± 54.8	308.6 ± 270.3
AAA/AAT	13	11.6 ± 5.1	87.6 ± 50.3	495.7 ± 248.6
Brachyteraphy	9	22.6 ± 19.8	16.4 ± 14.6	104.1 ± 94.5
Cavography	7	7.5 ± 6.5	66.1 ± 53.8	273.3 ± 216.4
Fibrinolysis	10	19.9 ± 11.0	28.1 ± 29.6	113.2 ± 102.9
Caval Filter	10	7.0 ± 7.3	64.1 ± 98.3	236.7 ± 282.7
Fistolography	10	4.6 ± 4.0	4.5 ± 14.6	28.4 ± 109.2
Flebography	26	6.5 ± 16.1	28.4 ± 58.9	300.2 ± 803.9
HVPG measurement	10	9.1 ± 7.1	29.2 ± 18.1	167.0 ± 100.1
TIPS	13	20.5 ± 13.3	117.3 ± 74.1	827.5 ± 609.5
Epiaortic trunk angiography	13	3.6 ± 3.3	40.9 ± 30.5	221.3 ± 150.4
Vertebroplasty	13	13.0 ± 16.1	51.4 ± 26.0	392.7 ± 157.3

Mean values of FT, KAP and CK indicate that cerebral angiography, aneurysm embolisation and chemoembolisation of liver cancer are procedure where high skin doses could be delivered. PSD has been measured in a sample of these procedures: results are summarised in Table 2.

Table 2. Peak skin dose (PSD) (mean values, SD and range) for a sample of selected interventional procedures.

Procedure	No	PSD (mGy)	Range (mGy)
Cerebral Angiography	25	352.4 ± 145.4	98.8 ¸ 561.9
Aneurysm Embolization	18	1072.5 ± 1085.2	332.2 ± 4941.9
Chemo-embolization	38	1343.8 ± 915.7	343.4 ± 4135.5

The correlation between PSD and KAP and CK was investigated. Results for cerebral angiography are represented in figures 1 and 2 for KAP and CK respectively.

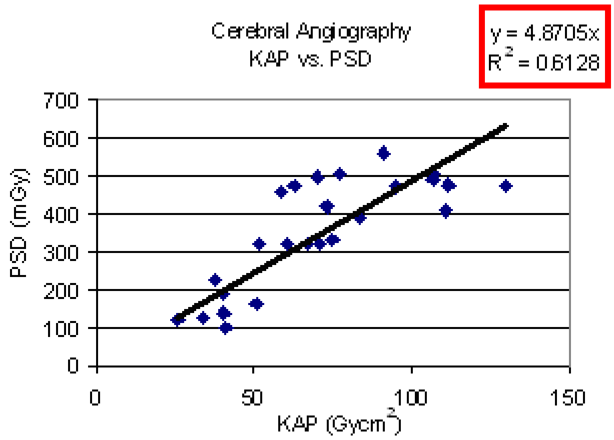

Fig. 1. Correlation between KAP and PSD for cerebral angiography.

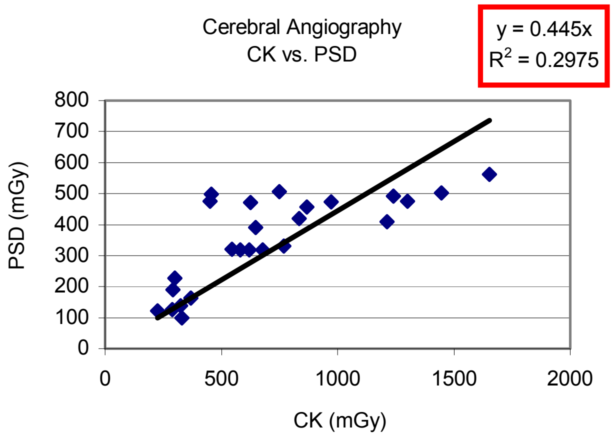

Fig. 2. Correlation between CK and PSD for cerebral angiography.

Results for aneurysm embolisation are reported in figures 3 and 4 for KAP and CK respectively.

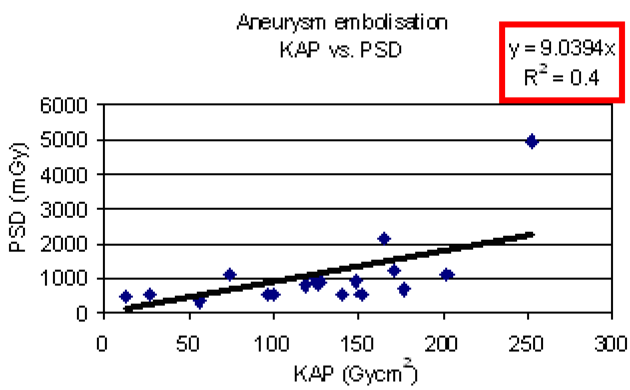

Fig. 3. Correlation between KAP and PSD for aneurysm embolisation.

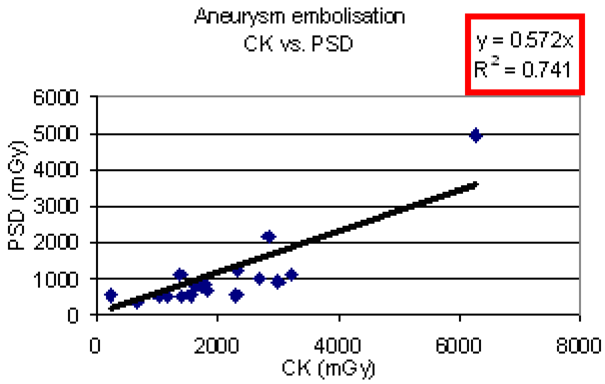

Fig. 4. Correlation between KAP and PSD for aneurysm embolisation.

Results for chemoembolisation are reported in figures 5 and 6 for KAP and CK respectively.

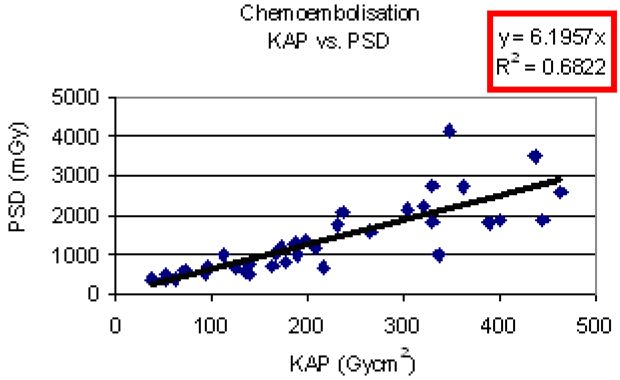

Fig. 6. Correlation between KAP and PSD for chemoembolisation.

Discussion

Registered values of KAP, CD and FT are high and suggest high doses to patient' skin, in particular for cerebral angiography, aneurysm embolisation and chemoembolisation for liver cancer.

Therefore it's important to understand the meaning of these values in terms of skin dose. Moreover it's important to allow the operator to have an immediate knowledge of the amount of dose received by patient and to recognize those procedures in which the dose could be greater than the threshold for deterministic effects.

For that reasons the correlation between dosimetric indicators as KAP and CK given by the equipment and PSD directly measured has been assessed and the possibility to establish trigger levels to be used in routinely practice has been investigated for cerebral angiography, aneurysm embolisation and chemoembolisation.

A. *Cerebral angiography*

For cerebral angiography procedures, correlation between PSD and both dosimetric indicators is weak (R2=0.61 and R2=0.3 respectively for KAP and CK). This is probably due to the high number the projections at different angles used in these procedures.

As a consequence the definition of a trigger level wasn't possible.

However doses measured in the sample are quite low (maximum value of 561.9 mGy).

A comparison between FT, KAP and CK of the sample in which PSD has been measured and the values of the entire database October 2007-October 2008 was performed: mean values were not statistically different.

This allow to assert that doses delivered in this type of procedure are generally not high and the definition of trigger levels in this case is not relevant.

B. ***Aneurysm** embolisation*

For aneurysm embolisation procedures, correlation be-tween PSD and KAP is weak (R2 = 0.4), instead correlation between PSD and CK is quite strong (R2 = 0.74), as reported in figures 3 and 4.

A retrospective analysis on the entire database was done using the linear coefficient to estimate PSD from CK. Estimated PSD exceeded the threshold for transient erythema (2 Gy) 10 times (13% of procedures) and 2 times the threshold for temporary epilation (3 Gy).

Therefore it's necessary in this case define a trigger level that indicate the possible overcoming of 3 Gy for PSD.

The value established in terms of CK for that type of procedure in our centre is 5200 mGy.

C. *Chemoembolisation **for** liver cancer*

For chemoembolisation procedures, correlation of PSD resulted good (R2 = 0.68) with KAP and very strong with CK (R2 = 0.93).

For these procedures, a retrospective analysis on the entire database, using the linear coefficient between PSD and CK found, was done as well. PSD exceeded the threshold for transient erythema (2 Gy) in 25 procedures (17% of the total number) and in one procedure PSD was estimated to be higher than 5 Gy.

Therefore, also in his case, it's necessary to define a trigger level that indicate the possible overcoming of 3 Gy for PSD.

Conclusions

Doses delivered to patient' skin during some interventional procedures could be very high and overcome the threshold for tissue reactions.

For procedures at highest doses it's opportune to define trigger levels based on cumulative air kerma values (CK) measured by the equipment to inform the operators that the dose delivered in a procedure could be higher than the threshold for skin damage.

Skin doses measured for cerebral angioplasty were well below the threshold for deterministic effects and no trigger level has been established.

For aneurysm embolisation and chemoembolisation PSD measured were quite high. For that reason trigger levels have been established in terms of CK. In our centre these levels are 5200 mGy and 2500 mGy, respectively for aneurysm embolisation and chemoebolsation.

As suggested by ICRP in the publication No. 85, a follow-up for patients whose estimated peak skin dose was 3 Gy or greater has been implemented as a routine practice.

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International project on individual monitoring and radiation exposure levels in interventional cardiology

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Abstract
Within the Information System on Occupational Exposure in Medicine, Industry and Research (ISEMIR), a new IAEA initiative, a working group on interventional cardiology (WGIC) aims to assess staff radiation protection levels and to propose an international database of occupational exposures.

A survey of regulatory bodies (RB) has provided information at the country level on radiation protection practice on IC. Concerning requirements for wearing personal dosimeters, only 57% of RB specify the number and position of dosimeters for staff monitoring.

Less than 40% of RB could provide occupational doses. Reported annual median effective dose values (often <0.5 mSv) were lower than expected considering validated data from facility-specific studies, indicating that compliance with continuous individual monitoring is often not achieved in IC. A true assessment of annual personnel doses in IC will never be realized unless knowledge of monitoring compliance is incorporated into the analysis.

Introduction
The United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR) has the mandate within the United Nations system to assess and report the levels, effects and risks of exposure to ionizing radiation from all sources, including the collection of dose data on occupational, public and patient radiation exposure. Complementary to the work of UNSCEAR, in 2009 the International Atomic Energy Agency (IAEA) initiated the Information System on Occupational Exposure in Medicine, Industry and Research, referred to as the ISEMIR project. ISEMIR is now undergoing a three-year test period.

The ISEMIR project was established to help to improve occupational radiation protection practice in targeted areas of medicine, research and industry, where non-trivial occupational exposures occur. Within each area, its main objectives are to identify good practices and deficiencies, define actions and key issues that need to be addressed, and disseminate information and knowledge so as to contribute to the reduction of risk of radiation exposure and facilitate the implementation of internationally agreed upon radiation protection practices. A final goal is to make proposals for establishing an international database for the regular collection of occupational dose data in targeted areas of radiation use in medicine, industry and research.

Within ISEMIR, the first Working Group on Interventional Cardiology (WGIC) was set up to assess radiation protection practice and occupational exposure to workers (cardiologists, electrophysiologists and other staff members) involved in IC. Its main objectives are to gain a worldwide overview of the current situation in this field, identify good practices and deficiencies, and define actions to be implemented to improve occupational radiation protection practice in IC. The working group intends to come up with recommendations for harmonizing worldwide monitoring procedures and set up a system for regular collection of occupational doses in IC.

This paper reports the results of an international survey of national regulatory bodies on requirements for staff dosimetry and on dose data availability at the country level.

Material and methods

In its first year (2009) the WGIC performed an international survey on requirements for staff dosimetry and on dose data availability for IC at the country level, using the following questions addressed to the radiation protection (RP) regulatory body:

1. Number of workers (total number, physicians, other professionals) with personal dosimetry involved in IC procedures in 2008.
2. Values of occupational doses (effective dose) in the national authority's database (or database accessible by the national authority) - median and 3rd quartile for each year (2006, 2007, 2008) and each occupational grouping (physicians, other professionals, and all).
3. Does the RP Regulatory Body define the number and position of dosimeters for staff monitoring in IC?
4. Does the RP Regulatory Body require a person to have specific RP training to perform fluoroscopy in IC?
5. Does the RP Regulatory Body require a person to have a specific license or certification in RP to perform fluoroscopy in IC?

Questionnaires were distributed by the IAEA to RP regulatory bodies. The questionnaire was emailed to 191 RP regulatory bodies in 136 countries, reaching, finally, about 180 regulatory bodies. Some Member States have a federal system of government, with the radiation protection regulations covering the use of X-rays being administered at the state level. State regulatory bodies were contacted for such countries where contact information was available.

Results

Responses were received from 81 regulatory bodies (56 national regulatory bodies and 25 state regulatory bodies), giving an overall country response rate of 40% (Table 1). The responding regulatory bodies have jurisdiction over countries and states whose summed population is about one-quarter of the world's total population. Few responses were received from the regulatory bodies of countries with large populations.

Table 1. Numbers of regulatory bodies (RBs) contacted, numbers and percentages (in parentheses) of responses received and portion of the world population represented.

Region	Countries contacted	Countries responded	RBs contacted	RB responses	Total regional population, 10^6	Total population of responding countries, 10^6
Africa	35	10	35	10 (29)*	1000	212 (21)
Asia Pacific	29	14	37	17 (46)	3879	815 (21)
Europe	49	26	49	26 (53)	731	222 (30)
Latin America	21	5	21	5 (24)	679	151 (22)
North America	2	2	49	23 (47)	341	212 (62)
Global	136	57	191	81 (42)	6630	1612 (24)

* Values in parentheses are percentages of the corresponding total.

Questions 1 and 2 asked about the number of monitored workers involved in IC in 2008 and the values of occupational doses registered for physicians, other workers and all workers. Averaged results suggest that monitored physicians in IC represent about 40% of all persons in IC being individually monitored.

About 50% of regulatory bodies provided some occupational dose data (question 2). Of these there was a wide variety of responses, ranging from detailed, accurate dose values to data that were inconsistent and/or ambiguous. It was often necessary to write back to regulatory bodies to obtain more detailed information about their responses.

About half the regulatory bodies stated that they were not able to provide occupational dose data for IC, citing the following reasons:

- no central dose register in the country;
- no easy access to the central dose register by the regulatory body;
- regulatory body only has records of doses if they exceed some particular threshold (e.g. investigation or action level);
- no specific classification for interventional cardiologists, or other persons in IC in the database.

Some of the dose data supplied were not suitable for further analysis for the following reasons:

- reported dose data were "contaminated" with doses from other occupational classes and functions, such as interventional radiology;
- corrected and uncorrected doses were mixed – e.g. doses were corrected for wearing position only if they exceeded some threshold, and these corrected values were then entered back into the original database of raw doses;

- reported doses were from a database that contained only doses above some action level.

After excluding dose data for the above reasons, occupational dose data from 29 countries were analyzed giving the following results:

- For those regulatory bodies reporting data for IC physicians as a group, in 2008 the average median effective dose was 0.73±0.62 mSv per year, and the average 3rd quartile effective dose was 1.09±0.69 mSv per year. The 2008 results are based on reported monitoring results from 23 countries, for a total of 1432 interventional cardiology physicians. Data were analysed on a per country basis.
- For 2006 and 2007 the average median effective doses for IC physicians were 0.67±0.64 and 0.78±0.60 mSv/year respectively, and the average 3rd quartile effective doses were 1.80±2.54 and 1.35±1.25 mSv/year respectively.
- For those regulatory bodies reporting data for other professionals in IC as a group, in 2008 the average median effective dose was 0.76±0.68 mSv per year, and the average 3rd quartile effective dose was 1.10±1.09 mSv per year. The 2008 results are based on reported monitoring results from 17 countries, for a total of 825 other professionals working in IC. Data were analyzed on a per country basis.
- For 2006 and 2007 the average median effective doses for other professionals were 0.42±0.38 and 1.07±1.17 mSv per year respectively, and the average 3rd quartile effective doses were 1.28±1.06 and 1.46±1.12 mSv per year respectively.
- For those regulatory bodies reporting data only for all persons in IC combined, in 2008 the average median effective dose was 0.56±0.47 mSv per year, and the average 3rd quartile effective dose was 1.68±0.21 mSv per year. The 2008 results are based on reported monitoring results from only 4 countries, for a total of 391 persons working in IC. Data were analysed on a per country basis.
- For 2006 and 2007 the average median effective doses for all persons in IC combined were 0.59±0.34 and 0.76±0.39 mSv per year respectively.

Distributions of median and third quartile annual effective doses from the regulatory bodies' reported data for interventional cardiologists, for the years 2006 to 2008, are given in Figure 1.

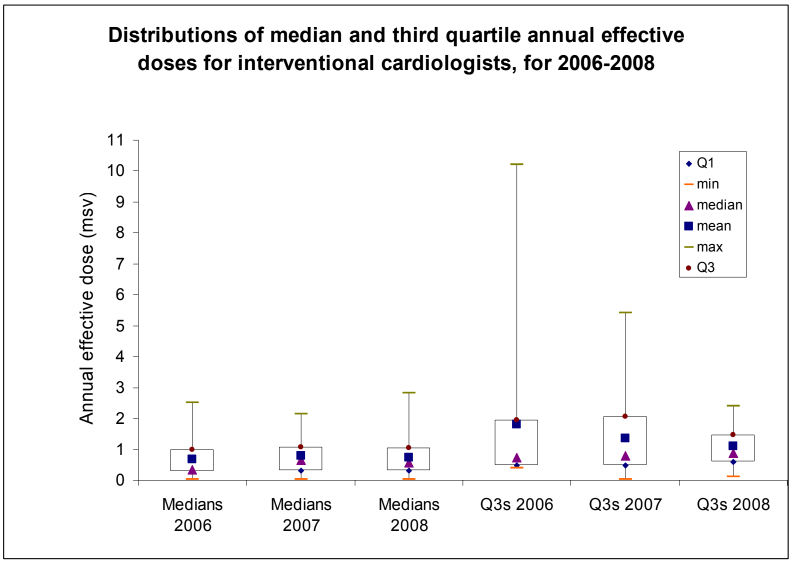

Figure 1. Distributions of median and third quartile annual effective doses from the regulatory bodies' reported data for interventional cardiologists, for the years 2006 to 2008.

The similarity in the values of doses reported for the IC physicians as a group and for the other professionals as a group deserves comment. Emphasis has traditionally been placed on the IC physician as being the person with the most potential for being occupationally exposed. Radiation protection training promotes the use of additional radiation protection tools, such as the ceiling suspended screen, to bring about a lower level of occupational exposure for the physician. The other professionals, such as the theatre nurse may not be afforded the same access to these additional radiation protection tools, having to then rely on the protective apron and distance as the main means of protection. Attention may need to be given to providing additional protective tools for these other professionals if occupational radiation protection in IC is to be truly optimized.

In all cases, annual effective doses remain well below the 20 mSv occupational dose limit, recommended by the ICRP. Specific dose values for extremities and eyes were not requested. It may be useful to obtain these in the future.

Despite some vetting of the dose data provided, other issues remain. Often personnel who have moved into more administrative duties remain on the monitored list, thus lowering average doses for that occupational group in the facility. It is very difficult to keep track of the doses for interventional cardiologists who may work in more than one facility, and reported doses may not be total doses across all workplaces. The treatment of doses at the limit of detection may differ from one regulatory body to another – a zero dose may be assigned, or a nominal minimum reporting dose or even some other nominal value. This may affect the statistical analysis, especially the mean.

But the largest potential shortcoming of the reported results is whether the dosimeters were actually being worn by the interventional cardiologists whenever they were performing IC procedures. The reported annual median effective dose values were lower than expected considering validated data from facility-specific studies, indicating that compliance with continuous individual monitoring is often not achieved in IC. Reasons for non-compliance with monitoring range from simple negligence to deliberate avoidance because of the fear of exceeding some dose threshold that leads to regulatory or administrative investigation (often as a result of an above-the-apron dose value being used as a surrogate for effective dose with no correction). All of these reasons would indicate that the results reported above are likely to be an under-estimate of the real situation.

Table 2. Number of regulatory bodies that mandate the number and position of personal dosimeters for monitoring in IC.

Region	Numberof responding RBs	Number and position mandated by the RB?		
		Yes	No	Not answered
Africa	10	3 (30)*	4 (40)	3 (30)
Asia Pacific	16	8 (50)	8 (50)	0 (0)
Europe	26	19 (73)	7 (27)	0 (0)
Latin America	5	2 (40)	2 (40)	1 (20)
North America	22	13 (59)	8 (36)	1 (5)
Global	79	45 (57)	29 (37)	5 (6)

* Values in parentheses are percentages of the corresponding total.

Concerning requirements for wearing dosimeters, 57% of regulatory bodies define the number and position of dosimeters for staff monitoring in IC (Tables 2 - 4).

Among those regulatory bodies defining the number and position of the dosimeter(s):

- 40% require the use of one dosimeter, worn above the apron (80%), and at the collar level above the apron (50%);
- 20% of regulatory bodies require the use of two dosimeters, worn beneath and above the apron in almost all cases,, except where the second dosimeter was to be used for extremity monitoring;
- 2% of regulatory bodies require the use of three dosimeters, one above and one beneath the apron, and a third finger type dosimeter;
- 38% did not specify their defined wearing position.

Table 3. Number of personal dosimeters mandated by regulatory bodies for IC.

Region	No of RBs mandating the number of dosimeters	Number of dosimeters required:			
		1	2	3	Not specified
Africa	3	1	1	0	1
Asia Pacific	8	0	2	0	6
Europe	19	10	2	1	6
Latin America	2	0	1	0	1
North America	13	7	3	0	3
Global	45	18 (40*)	9 (20)	1 (2)	17 (38)

* Values in parentheses are percentages of the corresponding total.

Table 4. Mandated location of personal dosimeters in IC when one dosimeter was mandated.

Region	No of RBs mandating only 1 dosimeter	Mandated wearing position:					
		Worn above the apron:			Worn below the apron:		
		Chest or trunk	Collar or shoulder	Unspecified	Chest or trunk	Collar or shoulder	Unspecified
Africa	1				1		
Asia Pacific	0						
Europe	10	4	2	2	2		
Latin America	0						
North America	7		7				
Global	18	4	9	2	3	0	0

Analysis of the responses on radiation protection training requirements showed that just over one half (55%) of regulatory bodies mandated RP training for personnel in order to be able to perform IC procedures. Clearly there is a need for many countries to strengthen the regulatory framework for occupational RP in IC.

The question on licensing or certification requirements for persons to be able to perform fluoroscopy in interventional cardiology unfortunately yielded ambiguous results. Analysis of responses and accompanying comments indicated that there was confusion about who needed to be licensed (e.g. the physician or the radiographer), what the licence was for (e.g. use of radiation or practice of medicine), and who issued the licence (e.g. radiation protection regulatory body or medical registration body or similar). No meaning results could be concluded, except that there was a spectrum of (radiation protection) licensing systems in use throughout the world, ranging from the physician not needing to have a licence to use radiation in interventional cardiology through to the physician needing such a licence.

Conclusions

This study presents the results of about eighty responses from regulatory bodies on staff exposure in IC practice. The results are helping the WGIC of ISEMIR formulate recommendations to improve monitoring methods and facilitate more adequate collection of data at the country level.

Obtaining reliable data on occupational exposures in IC from RP regulatory bodies proved to be difficult. Many regulatory bodies have limited access to these data, and, even if they do have access, the data are often not detailed enough to provide the required information. A further complicating factor is that recorded doses may underestimate true occupational exposure because compliance of IC personnel can be poor, and because an individual's exposures from different IC facilities may not be summed. Alternative strategies for the collection of IC occupational dose data will need to be utilized if a worldwide database of such information is to be established under the ISEMIR project.

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Radiation-induced cancer risk from recurrent CT scanning

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Abstract

Radiation induced carcinogenesis is a well known potential risk of diagnostic radiology. Especially the use of relatively high dose CT and exposure at a young age may lead to increased risk. Therefore the purpose of our study was to estimate the radiation-induced cancer risk from recurrent computed tomography scanning and associated lifetime attributable risk.

The cohort comprised 11,968 patients who underwent two or more diagnostic CT's from January 2005 to January 2009. In total 44,449 CT examinations were performed. Associated risk was estimated using the BEIRVII methodology on the basis of sex, age at each exposure and estimated mean dose. Each patient's cumulative risk was estimated by combining the LAR for each individual CT.

In the cohort, the youngest patient was 2 days old, the oldest 108 years old. The average age was 55.0±17.5 years with a median of 58 years. On patients of ten years or younger 593 (1.3%) examinations were performed. Ten or more CT examinations were performed on 551 (4.6%) patients, and between 15 and a maximum of 44 CT examinations were performed on 155 (1.3%) patients. An estimated cumulative effective dose of more than 100 mSv was received by 312 (2.6%) patients, and an estimated cumulative effective dose between 150 and a maximum of 314 mSv was received by 107 (0.90%) patients. Of the patients receiving more than 100 mSv, associated LAR had mean and maximum values of 0.87% and 5.6% for cancer incidence and 0.12% and 1.2% for cancer mortality, respectively. 120 (1.0%) patients of the cohort had an estimated LAR for cancer incidence greater than 1%, of which 29 (24%) patients had no cancer history. Recurrent CT scanning in this cohort may be responsible for 16 radiation-induced cancers and 10 cancer deaths in further life.

Recurrent CT scanning added to a significant increase in cancer risk in our cohort which may be a serious problem in long term follow-up of non-cancerous patients or patients who survived their (previous) cancer.

Introduction

The use of diagnostic radiology has shown a rapid growth in recent years [1]. Especially the use of computed tomography as a 3D technique contributed to this growth [1,2]. CT usage over the past quarter of a century has risen approximately 12-fold in the UK and 20-fold in the US [2]. It has been estimated that more than 62 million CT scans per year are obtained in the US, including at least 4 million children [1]. However, radiation induced carcinogenesis is a well known potential risk of diagnostic radiology. Moreover, especially the use of relatively high dose computed tomography (CT) and recurrent exposure at young age may lead to increased tumor risk and radiation-induced mortality [3,4,5]. It has been estimated that 29,000 future cancers could be related to CT scans performed in the US in 2007 [5]. Because some patient groups with chronic diseases or for the long term follow-up of cancer are repeatedly exposed to CT and sometimes a relatively young ages, it is very important to estimate the associated lifetime attributable risk (LAR) of these exposures. Therefore, the purpose of our study was to estimate the radiation-induced cancer risk from recurrent computed tomography scanning and associated lifetime attributable risk.

Material and methods

This retrospective study was performed at a main university medical center in The Netherlands with more than 1,300 beds and 190,000 radiologic examinations in 2009. From the database of the Radiology Information System (Radi, Hiscom, Leiden, The Netherlands) all patients were extracted that underwent two or more diagnostic CT scans from January 2005 to January 2009. Re-assessment of CT scans performed in other hospitals was excluded. In addition, sex, date of birth, examination type and examination date were extracted. From the examination date and date of birth, the age of the patients at the time of exposure was calculated. From the Hospital Information System (ZIS, Hiscom, Leiden, The Netherlands) the patient history and pathology was extracted.

From the Biological Effects of Ionizing Radiation (BEIR) VII report (fig 1), the associated LAR was estimated on base of sex and age at the time of exposure in terms of the probability of tumor induction and tumor death [6]. The total radiation induced tumor probability LAR_{tot} for each patient was calculated by using the LAR at each exposure:

$$LAR_{tot} = 1 - \prod_{i=1}^{N} (1 - LAR_i) \qquad (1)$$

where LAR_i is the lifetime attributable risk at exposure i and i runs over all N exposures. The probability of radiation induced tumor death was calculated in an analogue way.

Each examination was assigned an effective dose. In table 1 a summary of the effective doses is given for the anatomical regions. The effective doses were assigned on base of typical adult doses and are within range of common published estimates. The effective dose of the individual patient, as well as the scanner type and date of examination were not taken into account. Blank and contrast scans were assigned the same effective dose and two phase contrast scans were assigned twice the effective dose.

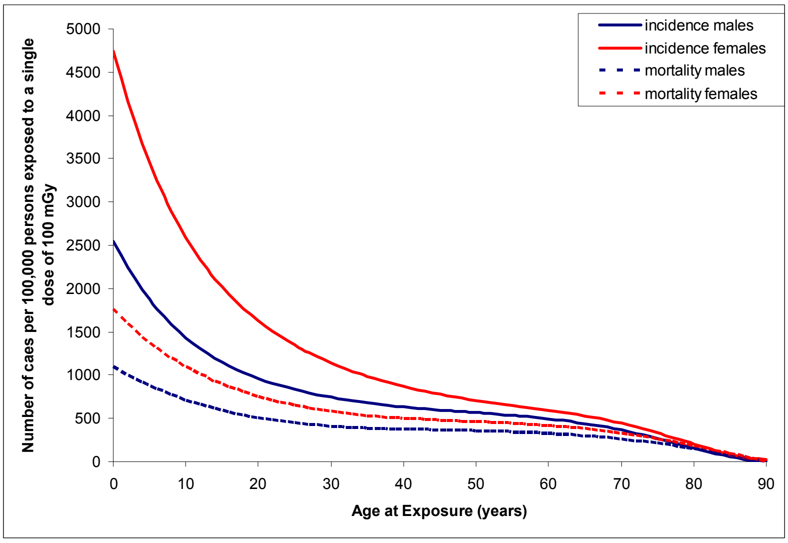


Fig. 1. Estimation of the number of radiation induced tumors (solid lines) and radiation induced mortality (dashed lines) as a function of age at exposure for males (blue) and females (red) according to BEIR VII.

Table 1. Estimated Effective dose per anatomical region in mSv.

Anatomical region	Scan	Blank [mSv]	Contrast [mSv]	Two phases [mSv]
Head	Skull	1.1	1.1	2.2
	Sinus	0.2	0.2	0.4
	Mastoid	0.3	0.3	0.6
	Mandibula	0.8	0.8	1.6
	Face	0.9	0.9	1.8
Neck	Neck, CWK	1.5	1.5	3.0
Thorax	Thorax	4.0	4.0	8.0
	TWK	5.0	5.0	10.0
	Shoulder	1.0	1.0	2.0
Abdomen	Abdomen, kidneys	10.0	10.0	20.0
	LWK, sacrum	8.0	8.0	16.0
Extremities	Upper leg	1.0	1.0	2.0
	Other	0.3	0.3	0.6

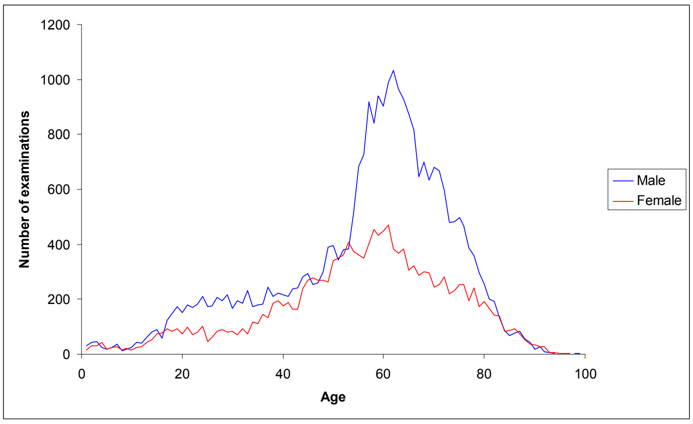
Results

The cohort comprised 11,884 patients (63.5% male). In the cohort, the youngest patient was 2 days old, the oldest patient was 108 years old. The average age was 55.0±17.5 years with a median of 58 years (table 2).

Table 2. Patient demographics.

sex	No of patients	Minimum age [y]	Maximum age [y]	Mean age [y]	Standard deviation [y]	Median age [y]
Male	7,595 (63.5%)	0	108	55.4	17.2	59
Female	4,373 (36.5%)	0	97	54.3	18.0	56
All	11,968 (100%)	0	108	55.0	17.5	58

In total 44,449 CT examinations were performed. 2,426 (5.5%) examinations were performed on patients of 20 years or younger of which 593 patients (1.3%) were 10 years or younger (fig 2). More than 70% of the examinations where performed in the thorax and abdomen region (fig 3).



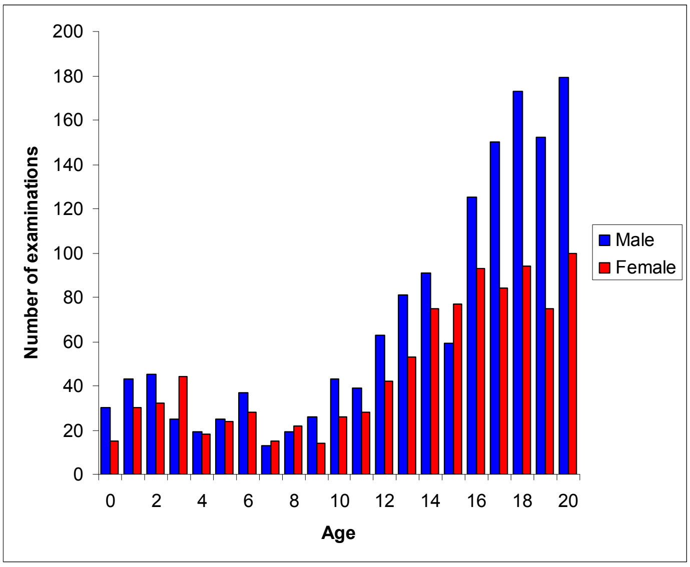


Fig. 2. Distribution of age at examination for males and females.

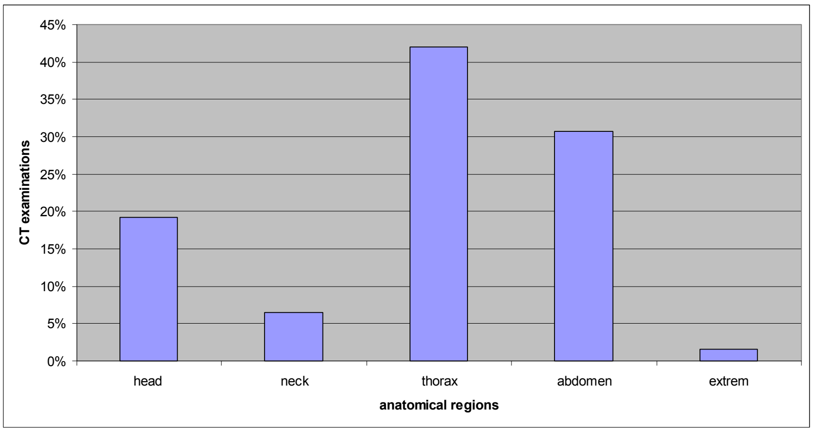


Fig. 3. Distribution over the anatomical regions for in total 44,449 CT examinations.

2,575 (22%) patients underwent 5 or more CT examinations, 551 (4.6%) patients underwent 10 or more CT examinations, and 163 (1.4%) patients underwent between 15 and a maximum of 44 CT examinations (fig 4).

312 (2.6%) patients received an estimated cumulative effective dose of more than 100 mSv, and 107 (0.90%) patients received between 150 and 314 mSv (fig 5).

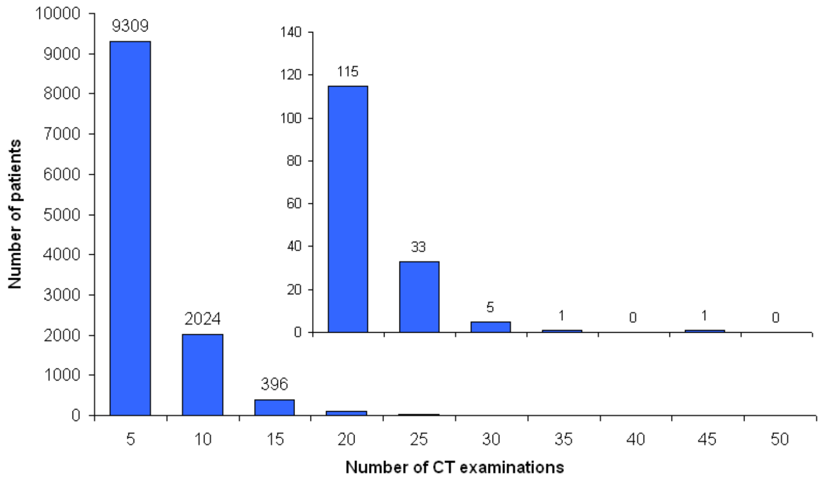


Fig. 4. Histogram of the number of CT examinations per patient.

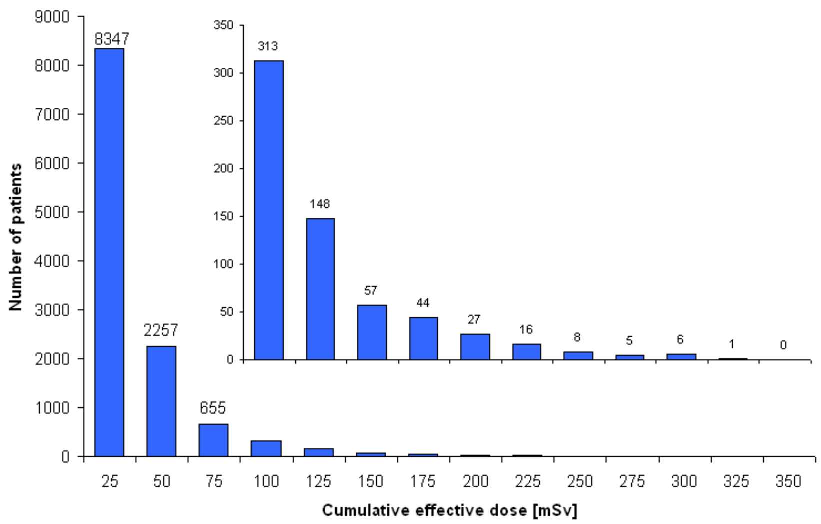


Fig. 5. Cumulative effective dose per patient.

Of the patients receiving more than 100 mSv, associated LAR had mean and maximum values of 0.87% and 5.6% for cancer incidence and 0.12% and 1.2% for cancer mortality respectively (fig 6, 7). 120 (1.0%) patients of the cohort had an estimated LAR for cancer incidence greater than 1%, of which 29 (24%) patients had no cancer history (table 3). These values imply that recurrent CT scanning in this cohort may be responsible for 16 radiation-induced cancers and 10 cancer deaths in further life.

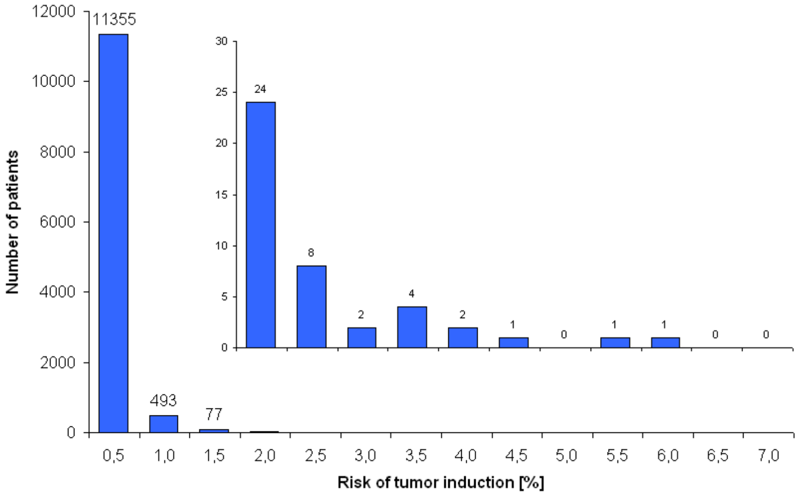


Fig. 6. Histogram of radiation induced cancer incidence.

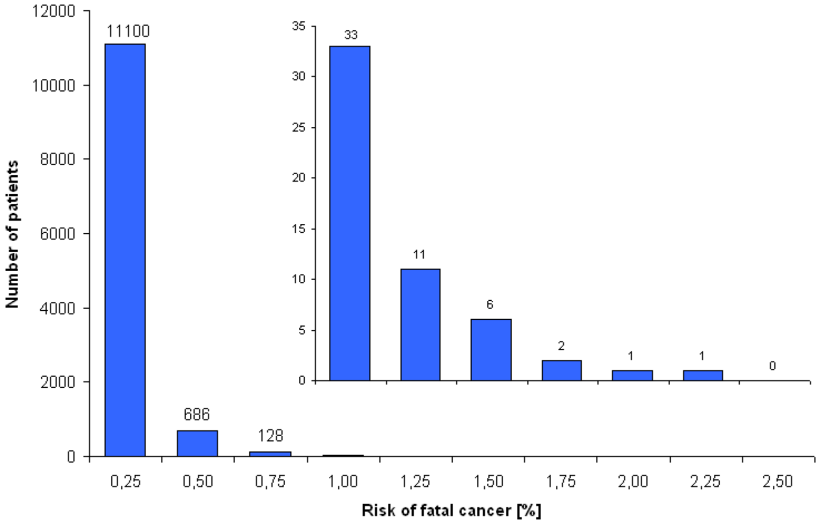


Fig. 7. Histogram of radiation induced fatal cancer.

Table 3. Non-cancer patients with a tumor-induction probability lager than or equal to 1.0%.

patient	sex	Age [y]	Cumulative Dose [mSv]	LAR incidence [%]	Number of examinations head	thorax	abdomen	total	patient's pathology
1	F	14-17	186	3.7		9	11	20	Liver transplantation
2	F	1	70	3.1			4	4	Liver transplantation
3	F	48-52	276	1.9		14	13	27	Liver transplantation
4	F	0	40	1.9			2	2	Liver failure
5	F	0	40	1.9			2	2	Liver failure
6	F	30-31	160	1.8			12	12	Rectal prolapse
7	F	10	60	1.5			3	3	Liver transplantation
8	M	2	68	1.5		2	4	6	Short bowel syndrome
9	M	15	130	1.5			8	8	Necrotizing pancreatitis
10	F	4-5	40	1.4			3	3	Complex appendicitis
11	F	19-21	84	1.4		1	4	5	Ulcerative pancolitis
12	F	44-45	224	1.3		1	13	14	Necrotizing pancreatitis
13	F	19-20	81	1.3	1		5	6	Liver transplantation
14	F	5-6	40	1.3			2	2	Oesophageal varices
15	F	19-20	80	1.3			4	4	Liver failure
16	M	59-61	263	1.3	1	3	14	18	Liver transplantation
17	F	53	194	1.3		4	10	14	Necrotizing pancreatitis
18	F	15-16	64	1.3		1	5	5	Appendicitis
19	F	54-55	190	1.2			11	11	Kidney transplantation
20	F	2	30	1.2			2	2	Morbus Hirschsprung
21	F	2	30	1.2			2	2	Chronic inflammation
22	F	21-23	80	1.2			4	4	Hepatic adenoma
23	F	31-32	110	1.2			6	6	Necrotizing pancreatitis
24	F	43-44	146	1.2		4	9	13	Liver transplantation
25	F	4-6	34	1.2		1	2	3	Liver transplantation
26	F	2	40	1.2			2	2	Pancreatitis
27	F	23	80	1.2			4	4	Hepatic adenoma
28	F	56	170	1.1	1	2	12	15	Pancreatitis
29	M	0	40	1.0			2	2	Biliary atresia

Discussion

We have shown that recurrent CT scanning added to a significant increase in radiation induced cancer risk in our cohort. 1.0% of our patient had an estimated LAR for radiation induced cancer incidence larger than 1%. However, of these patients as much as 24% had no cancer history. These figures might cause a serious problem in the long term follow-up of non-cancerous patients or patients who survived their (previous) cancer.

Recently Sodickson et al. concluded also that recurrent CT and associated cumulative CT radiation exposure added incrementally to baseline cancer risk [7]. The cohort they studied was taken from 22 years of CT examinations from a tertiary academic medical center. Therefore, the number of patients in their cohort was much larger and the number of examinations was much larger also. In addition, they observed that 15% of their patients had a cumulative effective dose of more than 100 mSv, whereas we observed that only 2.6% of our patients had a cumulative effective dose of more than 100 mSv. As a consequence, the maximum LAR they estimated was 12% and 6.8% for tumor induction and tumor death respectively, whereas we estimated a maximum LAR which was much smaller, i.e. 5.6% and 1.2% respectively. Also the fraction of their cohort with a LAR larger than 1% was much larger than in our study, 7% versus 1%. Finally they observed that 40% of this fraction had no history of cancer, whereas we observed that only 24% of this fraction has no history of cancer. The relatively larger LAR observed by Sodickson et al. can be partly explained by the larger fraction of women in their cohort and the relatively larger fraction of thorax examinations. In addition the maximum cumulative effective dose in their cohort was much larger, 1375 mSv versus 314 mSv and also the maximum number of recurrent CT scans in their cohort was much larger, 132 versus 44. This might also partly explain the relatively larger LAR observed in their study. However, because we used larger dose values for two phase examinations and we also included patients under the age of 11 in our cohort, this would increase the LAR in our study. Recently Griffey et al. reported a maximum estimated LAR for cancer incidence of 5.6% in emergency department patients undergoing multiple CT which corresponds very well with the maximum estimated LAR for cancer incidence of 5.9% in our study [8].

In our cohort, 5.5% of the CT examinations were conducted on patients of 20 years and younger and 1.3% on patients of 10 years and younger. A study from the US showed much more patients in the same age groups, with 18% of the CT scans conducted on patients up to 20 years and 6% of the CT scans conducted on patients up to 10 years [9]. Furthermore, their dose was slightly higher for CT-head (1.5 mSv) and CT-chest (5.4 mSv) compared to the maximum effective doses applied in our study (1.1 mSv and 5.0 mSv respectively). These figures indicate that for young patients in the US the LAR from recurrent CT scanning might be even higher than the estimated LAR values for young patients in our study. In addition, it was recently shown that as much as 30 percent of all CT examinations conducted on patients younger than 35 years were unjustified and could have been replaced by magnetic resonance imaging or would not have been needed [10]. When we apply these numbers to our population then about 2041 CT examinations conducted on our patients younger than 35 years may have led to an unjustified risk of cancer incidence.

It is well known that the individual effective dose of a patient depends on the scanner type, the specific protocol used and varies with specific patient parameters [11]. We did not adjust the effective dose estimates for these effects but used average effective dose estimates per anatomical region. This approach has several limitations, and we plan to extend our study for individual patient effective dose estimates. In addition, patients with a single CT scan were not included in our study. However, because of the relative low effective dose of one single CT scan with respect to the cumulative effective dose of multiple recurrent CT scans, we estimate that the influence of including single CT scans in our cohort is relatively small. Finally, also interventional CT procedures were included in our cohort. Because the effective dose in an interventional CT procedure depends strongly on the total scan time, an average effective dose estimate would only provide a crude estimate of the patient’s individual dose. However, because the number of interventional CT procedures in our database is relatively small and we estimate that this would not change our results. Finally, the errors in the cancer risk model as presented in the BEIR VII report are estimated at a factor of 2 [6]. Therefore, the error in the cancer risk model is thought to be dominant over the other limitations listed above.

Conclusions

Recurrent CT scanning added to a significant increase in cancer risk in our cohort which may be a serious problem in long term follow-up of non-cancerous patients or patients who survived their (previous) cancer.

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National co-ordination of Clinical Audits for medical radiological procedures

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Abstract

According to the EC directive 97/43/EURATOM the Member States are required to implement clinical audits for medical radiological practices in accordance with national procedures. Correspondingly, the Finnish legislation requires that the radiological units have to implement self-assessments and external clinical audits of their practices. To ensure the quality and consistency of clinical auditing, an Advisory Committee for the coordination, development and follow-up of the clinical audits was established in 2004 by the Ministry of Social Affairs and Health. The committee is a multi-disciplinary group of clinical experts, independent of any auditing organizations. Its tasks include, among other things, evaluating the suitability and coverage of the criteria used in clinical audits and collecting summaries and reviews of the results.

The Advisory Committee has issued four recommendations dealing with the competence, experience and independence of the auditors, developments for the second audit round, how to take into consideration the accreditations of nuclear medicine units and the ten points of interest given for clinical audits in the legislation. Two more recommendations are being prepared concerning audit reports and how to take into consideration the new European guideline on clinical audit. Current other activities include efforts to encourage the national scientific and/or professional societies to contribute to the development of the criteria of good practice, and a review of clinical audits in some other countries in order to propose improvements of the Finnish implementation. The Committee has also conducted a survey of the results of the complete first round of audits in Finland and organized a particular seminar on clinical audits. Further information (in Finnish) is available from “www.clinicalaudit.net”. In this paper, the achievements and experiences of the national coordination efforts are discussed in detail.

Introduction

The EC directive 97/43/EURATOM (MED-directive) introduced the concept of Clinical Audit for the assessment of medical radiological practices and the Member States are required to implement clinical audits in accordance with national procedures. The directive has been transposed to the Finnish legislation with a requirement that the radiological units have to implement self-assessments and external clinical audits for their radiological practices. The practical methods of the external clinical audits are not specified in detail in the legislation, and in particular, no particular organization is designed to carry out the audits. Therefore, more than one approach for the practical implementation had been expected and a national coordination of efforts for the quality and consistency of auditing has been considered necessary.

In 2004, a national steering group, or an Advisory Committee for Clinical Audit, for the development and follow-up of the clinical audits in Finland was established by the Ministry of Social Affairs and Health. This Committee is a multi-disciplinary group of clinical experts, independent of any auditing organizations. It has representatives of all three disciplines (diagnostic radiology, nuclear medicine and radiotherapy) and of all major professional groups (physicians, medical physicists and radiographers), and in addition, a representative of Finnish Accreditation Service (FINAS) and the Radiation and Nuclear Safety Authority (STUK). The Committee has been established in three years' terms.

Objectives

The objectives of the Advisory Committee are:

- To co-ordinate and develope clinical audits for the medical use of radiation
- To promote good quality and the consistency of criteria for good practices in clinical audits

The detailed tasks of the Advisory Committee have been specified as follows:

- To follow-up the implementation of clinical audits in Finland and to prepare national summaries
- To advice on the competence and training requirements for auditing organizations and auditors
- To assess the relevance and adequacy of audit criteria in different sectors (diagnostic radiology, nuclear medicine and radiotherapy)
- To advice on examination and treatment specific audit criteria and objectives and to promote their implementation
- To evaluate the importance of other quality audits in medical practice (such as audits for accreditation or for the certification of quality systems) for clinical audits, and give recommendations on their exploitation in clinical audits
- To follow-up international developments in clinical audits and in order to develop the Finnish practice of clinical audits

Actions

One of the main actions of the Advisory Committee in order to achieve its objectives has been to prepare and issue recommendations. By now, the Advisory Committee has issued six recommendations:

- Requirements of competence, experience and independence of the auditors (Recommendation 1, 2005).
- Development of clinical audits: Recommendations by the Advisory Committee for the second audit round (Recommendation 2, 2006)
- Taking into consideration of accreditations in the clinical audits of nuclear medicine (NM) units (Recommendation 3, 2006)
- Taking into consideration of the ten points of interest given in the Degree (423/2000) in clinical audits (Recommendation 4, 2008).
- Recommendation on the audit reports (Recommendation 5, 2009)
- Recommendation on how to take into consideration the new European guideline on clinical audit (Recommendation 6, 2009).

The first recommendation specifies for example that the auditors are required to have practical clinical experience in the field to be audited, the lead auditors must have at least one week specific training on the audit techniques, and the composition of the audit team must generally include a physician (e.g., radiologist, oncologist, or nuclear medicine expert and a radiation technologist, and in certain cases (e.g. for radiotherapy), also a physicist.

The second recommendation set out the priorities for the next audit round, recommending specific topics of interest in the three fields (diagnostic radiology, nuclear medicine and radiotherapy) including more detailed “in depth” assessments of selected examinations and treatments. It also gave references to some recommended criteria of good practices. The third recommendation describes specific considerations to be taken into account in clinical audits of NM units which have been accredited, in order to avoid unnecessary overlap with the audits carried out for the accreditation. The fourth recommendation considers in detail the contents of the ten points of interest set out in the Degree (423/2000) and specifies the sub-items of each point where clinical audit should have the highest weight. The purpose of this recommendation is to unify auditor’s interpretations of these ten points of interest and to avoid unnecessary overlap of clinical audits with regulatory inspections.

The fifth recommendations gives some particular guidance on clinical audit reports, e.g. that the reports must be stored for at least 12 years. The sixth report deals with the EC Guidelines of Clinical audit [1] raising some key points of the Guidance to ensure its implementation in the Finnish practice of clinical audits.

Besides giving recommendations, the Advisory Committee has conducted a survey of the results of the complete first round of audits in Finland (2000-2006). The summary of this survey has been published (in Finnish) in the Committee web-site and is briefly reviewed below. In 2006, at the end of its first term of operation, the Advisory Committee organized a half-day seminar on clinical audits, where the recommendations of the Committee and the summaries and future development of the clinical audits, on point of view of auditing organizations and authorities, were presented and discussed. In the beginning of 2010, after the end of its second term of operation, the Advisory

Committee organized another half-day seminar on clinical audits, where the new recommendations of the Committee and the EC Guidance on clinical audits were introduced and the future development of the clinical audits (criteria of good practices, audit methods) were discussed.

Besides the published recommendations and presentations in the above two seminars, the Committee has given several lectures on clinical audit at national meetings and training events, and replied and given comment or statements on several questions by the auditing companies, auditors or individual persons from audited organizations. The Committee has also participated in two meetings between the auditing organizations and the authority (STUK), where the possible overlap of clinical audits and regulatory inspections has been discussed.

The Committee has also actively followed and participated in the international developments of clinical audits. The Committee has participated and provided scientific support on organizing of an international Symposium (2003) and an international Workshop on clinical audits (2008) in Tampere, Finland. An EC project leading to the publication of the EC Guidance on clinical audits was lead by the Secretary of the Committee, and also Committee's chairman and one other member participated in this project. Committee's members have given several presentations and published several papers on the topic of clinical audits (see the list of publications and presentations below).

Currently, the Committee is addressing the national scientific and/or professional societies in order to encourage the societies to contribute to the development of the criteria of good practice. Further, the Committee has reviewed the development of clinical audits in some other countries in order to propose improvements of the national implementation of clinical audits. The future working plan includes also the development of guidance for self-assessment of practices.

From the beginning of its establishment the Advisory Committee has maintained a dedicated web-site "www.clinicalaudit.net", where all recommendations, proceedings of the seminars and all other documents and information on the Committee have been published. The wed-site also publishes information and maintains links to the international activities where the Advisory Committee has been a contributor (e.g. the EC Guidance on Clinical Audits).

Summary of clinical audits

After the first five years period of auditing (2000-2006), the National Advisory Committee conducted a survey of the results of the audits by reviewing the audit reports, altogether 346 reports, with the permission of the audited health care units. The following gives a brief summary if this survey.

All of the first clinical audits have been carried out by the same organization, employing a total of 38 auditors from volunteered health care professionals. The audits have typically been carried out by a team consisting of a medical expert (physician) and a radiographer, in several cases (e.g. for radio-therapy, nuclear medicine and large units of diagnostic radiology) also a physicist. The duration of the audit has varied between 0,5 and 4 days depending on the size of the unit, being most typically 1-2 days. The audits have been based on the guidance and check lists developed by the auditing organization. The criteria of good practices have been derived partly from the legislative requirements, partly from existing recommendations for good clinical practices (e.g. referral guidelines, image quality criteria), while the audits to a great extent have also relied on the professional judgement of the auditors. Besides interviews and reviews of documents and data, the audits have included assessment of the image quality for a sample of patient images.

A significant number of recommendations to improve the practices have been issued in the audits, on the average 4-7 recommendations per health care unit. An example of the main topics of recommendations for diagnostic radiology units is shown in Fig. 1. The results have also suggested that the criteria of good practices, now mainly based on legislative requirements, should be supplemented by more clinical criteria in order to avoid unnecessary overlap with regulatory inspections.

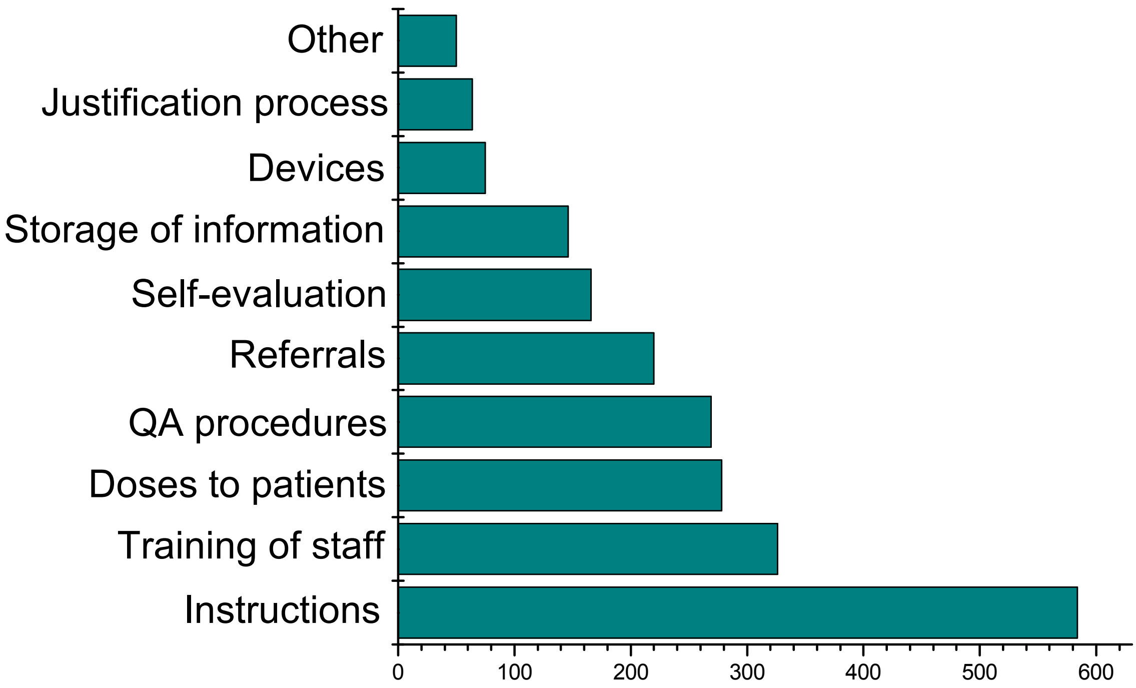

Fig.1. The total number of recommendations given at the first external clinical audits of diagnostic radiology units, classified according to the points of interest from Decree 423/2000.

Conclusions

The organization and criteria for external clinical audits in Finland have provided the radiological units with easy access to such audits. The role of the National Advisory Committee, in collaboration with the Radiation and Nuclear Safety Authority (STUK), has proved to be important for overall co-ordination, review and development of the national efforts as well as maintaining good quality and consistent methods of the audits.

Publications and presentations

The following is a list of publications and presentations by the Committee or Committee's members (besides the Recommendations presented above).

1. Implementation of clinical audits in Finland (2000-2006) and the recommendations given in the audits. The final report by the Advisory Committee for Clinical audits, Helsinki 1.12.2008 (in Finnish).
2. Outcome of the International Symposium on Clinical audit in Tampere 2003, Seppo Soimakallio, a pp-presentation. International Workshop on Practical Implementation of Clinical Audit for Medical Exposure to Ionizing Radiation, Tampere, Finland, 7-10 September 2008. www.clinicalaudit.net.
3. Objectives and structure of the EC Clinical audit project, Hannu Järvinen, a pp-presentation, International Workshop see reference 2.
4. Summary of the results of the European questionnaire with conclusions, Hannu Järvinen, a pp-presentation, International Workshop see reference 2.
5. Critical review of the Guidelines - Quality/competence assessment organizations, Tuija Sinervo, a pp-presentation. International Workshop see reference 2.
6. Major feedback on the draft EC Guidelines, Hannu Järvinen, a pp-presentation, International Workshop see reference 2.
6. Summary of clinical audits in Finland after the first complete audit round, S. Soimakallio, H. Järvinen, A. Ahonen, K. Ceder, M. Hirvonen-Kari, T. Lyyra-Laitinen, T. Sinervo, T. Sipilä and T. Wigren, a pp-presentation. International Workshop, see reference 2.
7. Steering actions by a National Advisory Committee for Clinical Audits in Finland, S. Soimakallio, H. Järvinen, A. Ahonen, K. Ceder, T. Lyyra-Laitinen, M. Paunio, T. Sinervo and T. Wigren, a Poster, International Workshop see reference 2.
8. Clinical Audit programme for diagnostic radiology: Intent, design and early experiences, B. Abdullah, P. Butler, K. Faulkner, H. Järvinen, D. McLean, M. Pentecost, M. Richard, a Poster, International Workshop see reference 2.
9. Is Clinical Audit Useful in Continuous Improvement of Quality in Radiological Department? S. Soimakallio, P. Dastidar, R. Järvenpää, L. Mäkelä and T. Paakkala, a Poster, International Workshop see reference 2.
10. Experiences Of A Clinical Audit In A Finnish Radiotherapy Clinic, S. Hyödynmaa and T. Wigren, a Poster, International Workshop see reference 2.
11. Implementation of Clinical Audit in the context of the EC directive 97/43/EURATOM: Finnish experiences, Hannu Järvinen, a pp Presentation, Dublin 2007 and IAEA 2007.
12. How to organize clinical audit at national level, Hannu Järvinen, a pp Presentation, MIR-Management in Radiology - Scientific Congress, Ateena 2008.
13. How to organize clinical audit in an Imaging Department, Seppo Soimakallio, a pp Presentation, MIR-Management in Radiology - Scientific Congress, Ateena 2008.
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15. Cover Story: Clinical audit in the European Union, Bringing the Gap Between Guidelines and Implementation, Hannu Järvinen, Imaging Management Vol 9, Issue 4, 2009, p.14-16.
21. Implementing External Clinical Audits in radiological Practices, The Experience in Finland, Seppo Soimakallio, Imaging Management Vol 9, Issue 4, 2009, p. 19-20.
22. European wide perspective on clinical audit, Hannu Järvinen, a pp Presentation, European Radiology Congress, Wien, 2009.
23. EC Clinical Audit Guideline: Justification is among the Priorities of an Audit Program, Hannu Järvinen, a pp Presentation, International Workshop on Justification of Medical Exposure in Diagnostic Imaging, Brussels, September 2009.
24. Clinical Audits; who should control what? - European guidelines, Hannu Järvinen, a pp presentation, International Conference on Modern Radiotherapy: Advances and Challenges in Radiation Protection of Patients, Versailles, 2009.
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Radiation protection of patients in medical radiology: proposal for a worldwide action plan

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The highest doses of ionizing radiations of artificial origin delivered to the population result from medical exposures of patients (UNSCEAR data). In developed countries such as USA and Japan, they are the first source of exposure to radiations of the population, before natural exposure. Medical exposures represent an average of 1 mSv per capita per year worldwide. By averaging the data over the whole population when only about 10% of it is exposed annually, the average value of 1 mSv underestimates the real average dose effectively received by the persons medically exposed.

Exposures of the populations to ionizing radiations from medical origin are constantly increasing worldwide. The trend in dose increase in diagnostic radiology and nuclear medicine seems unavoidable and is due to a combination of factors :

- There is a clear increase worldwide in the number of medical equipments which deliver the highest doses, i.e., CT and PET, because they are the most useful;
- There is an increase in the number of examinations which deliver the highest doses : whole body scanning, virtual colonoscopy, CT coroscanning … Whole body CT may deliver in one single examination almost the maximum effective dose that a worker can receive in one year, i.e., 20 mSv;
- Radiologic imaging results in a net health benefit for the patients :
    - Medical imaging is a decisive tool in combination with laboratory dosages for making the diagnosis of diseases ;
    - Medical imaging is also an efficient non effractive tool to treat patients. Interventional radiology as a subspecialty of radiology is also a growing field because it provides minimally invasive ways of diagnosis and therapy under image guidance. It is widely used to replace surgery whenever possible for treating heart, brain, kidney … vascular diseases or tumours ;
    - Finally, medical imaging is decisive to drive the therapeutic strategy and to follow up the efficacy of patients' treatments.

Recurrent publications advocate an excess of cancers due to medical ionizing radiations. Hypothetical numbers of cancer in excess are calculated on the basis of the ICRP risk estimates for low doses of ionizing radiations.

However, attention needs to be paid to limit the medical doses delivered since they may reach in some patients in whom examinations are repeated the level of the upper limit of the so called "low doses", i.e., about 50 mSv in children and 100 mSv in adults, above which statistically significant increase of cancers has been established. Thus if medical exposure of patients continue to grow, time will come in few decades where an excess of cancers will be demonstrated by epidemiology studies.

Thus time has come for a worldwide action plan to limit the growth of medical exposure to ionizing radiations.

Action plan

Since medical exposures to ionizing radiations are the highest of all exposures, and since the trend in the increase of dose due to medical imaging seems unavoidable, the radiation protection authorities, the international organizations in radiation protection and the medical professionals could slow the rate of increase through the following actions :

1. Radiation protection principles to be revisited and enforced :
   - Justification of new radiology procedures, e.g., whole body CT, CT coroscanning, CT virtual colonoscopy,… Request to the medical professionals to identify the real indications/non indications of these new techniques and to complete the referral criteria for imaging guides. Support of the recent WHO initiative for the development of an international referral criteria for imaging guide;
   - Optimization of CT and interventional radiology. Request to the medical professionals to revisit the guides of procedures in the sense of their optimisation;
   - Support of the IAEA initiatives in radiation protection of patients;
   - Dedicated attention to children and young adults.

2. Substitution by other medical imaging modalities (MRI and ultrasound) to be encouraged :
   - Request to the professionals to revisit the referral criteria for imaging guide by highlighting when MRI and ultrasound must be preferred to radiology and nuclear medicine procedures when the results are equivalent ;
   - Request to medical policy makers to favour the installations of MRI equipments at an appropriate level to be determined on the basis of the delay of appointment for an examination.

3. Optimisation of CT and digital imaging equipments :
   - Request to industry to optimise, lower the doses delivered by the equipments, e.g., pulsed radiology for interventional radiology, and develop the corresponding standards in relation with standardisation organisations;
   - Request to industry to provide the radiologists with in line dosimetric parameters for optimisation, to include these data in the headers (DICOM) and to provide software for the statistical use of these data as an experience feedback;
   - Continuous development of quality assurance, especially regarding the new digital equipments;
   - Support of the recent HERCA initiative.

4. Specific training of all medical radiology professionals
   - Enforcement of a continuous training in radiation protection. Modules of training to be developed and distributed;
   - Development of peer training regarding new technologies, especially in interventional radiology.

5. Development of concerted actions for the audit of radiology practices, especially for the procedures delivering the highest doses, i.e., CT, interventional radiology and PET;
   - Elaboration of guidelines for peer review of medical practice. Support of the European Commission initiative (RP n°159);
   - coordination with the authorities, if any, responsible for the control of medical device safety as a whole (e.g., AFSSAPS in France).

6. Organization of periodic scientific meetings to make the point on all radiation protection issues in medicine

7. Information and education of patients regarding the risks of ionizing radiations

TLD postal dose audit of radiotherapy beams calibration: creation of national quality assurance system in Ukraine

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Abstract

According to IAEA/ICRU requirements for effective radiotherapy and prevention of radiation effects the discrepancy between stated and delivered absorbed dose in tumor should not exceed ±5 %. In Ukraine the metrological control of radiation parameters of radiotherapy units in hospitals are carried out once per 2 years. In hospitals a control of radiation beam calibration are carried out by medical physics using own clinical dosimeters once per quarter or year. But it is necessary to organize the independent control of calibration quality. In 1969 IAEA with WHO established the TLD postal programme to verify the calibration of radiotherapy beams in developing countries. The main purpose of the programme is to provide an independent quality audit of dose delivered by radiotherapy units using TL-dosimetry. Ukraine was included to IAEA programme since 1998. During 1998 – 2008 there were 152 radiation beam checks of 76 radiotherapy units. Although the participation in IAEA/WHO TLD-audit was voluntary and confidential some hospitals (about 25%) never took part in this programme. IAEA/WHO TLD dose audit shows that only 47 – 75% of radiation beams in different years (average value – 64,5%) were within the acceptance limits of ±5 % at first stage of TLD-audit, moreover about 10 – 15.5% of radiotherapy beams had errors of calibration outside ± 10%. After two stages only 69.6 – 89.5% (average value – 77,6 %) of radiotherapy beams were in the acceptance limits of ± 5%. The main reasons of such results were outdated state of radiotherapy units and clinical dosimeters in hospitals, absence of qualified medical physics and using old methodology of absorbed dose estimation in tissues. Thanks to technical and methodological support of IAEA, Ukraine has a possibility to create national system of TLD postal dose audit and provide the control of quality calibration of radiotherapy beams for all hospitals on regular and obligatory basis that will assist in improving quality of radiotherapy.

Making the best use of the radiation in diagnostic radiology – improvement is needed

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Abstract
In the radiological protection community there is an increasing concern about justification. This is partly, due to the remarkable increase in the number of CT examinations in the last 10 years in Europe and in the US, and partly because it is recognized that there is a large potential for averted dose if a substantial part of the examinations is unnecessary. Because of this concern a study was conducted in Sweden in order to investigate the level of unjustified examinations. The study included all patients, in all departments, from large university hospitals to small health care units, examined with a CT scanner during one specific day. There were patients without known disease, patients before, during and after treatment of disease. The evaluation was performed by a group of radiologists and physicians. Some findings; a substantial part of the examinations were unjustified, about 20 %. Certain types of examinations had a higher and some a lower level of justification compared to the average number. Examinations of younger patients had a slightly lower level of justification. But as anticipated the observers were not always unanimously in the grading. The first conclusion is obviously, that justification is important and there is a need for the radiation protection community to take active part in the work. Furthermore, it is not fruitful to compare the data in this study with others because the result depends on the methodology and the examinations included. The study also indicates that it is necessary to improve the situation for specific types of examinations. In order to improve the situation much work is needed. One key issue is to develop a system for assessing the cost and the benefit of the procedures. It should be emphasized that risk comprises not only radiation risk. In order to succeed a joint effort involving different professionals is required. This is a challenging task but is essential when striving for making the best us of the radiation.

Concern has been raised
Diagnostic radiology is used in different situations and for different purposes, e.g. to verify disease or in planning treatment. Thus, there are different reasons for performing examinations. Which type of equipment is available differs, conventional radiology, computer tomography, magnetic resonance imaging or ultra sound examinations can be chosen, sometimes for the same purpose. Different methods might be applied with

specific equipment and new equipment, techniques and applications is introduced. One important question is how these complex situations can be managed in the health care sector, so that the diagnostic procedures have positive impact on the health of the individual patient.

The consumption of radiology differs between countries, e.g. the number of examinations per capita differs between the European countries [1]. One example, the total number of x-ray examinations per 1000 inhabitants and year differ with a factor of 4 between the UK and Germany. The reasons for such differences are not obvious, it might be due to differences in the availability of health care, diagnostic equipment, medical doctors, radiology specialists, or to factors such as reimbursement systems. No clear answers have been given, and most remarkable a medicine based reason has not been presented yet.

Common to all countries the rapidly increasing number of computer tomography (CT) examinations [2]. Serious concerns for this development have been expressed. Some are concerned about the health care costs and some about insufficient competence for performing complex examinations, and others are concerned about the increased radiation dose. The increased number of examinations might, at least partly, be explained by an increasing number of unjustified examinations. The present study was designed to cover this aspect.

Finding a method to assessed justification

The assessment of justification is not an easy task. On a more generalised level the outcome of different diagnostic procedures has to be evaluated systematically, which gives evidence on what impact the diagnostic procedure has on the handling of specific group of patients. Unfortunately, for many of the procedures evidence on this level is not available [3]. Therefore evidence on a lower level has to be accepted in the guidelines for the use of diagnostic procedures that have been worked out [4]. There are also hazards, including radiation, that have to be addressed in order to truly evaluate a diagnostic procedure. Thus this is a complex situation.

For the individual patient it has to be investigated whether a certain diagnostic procedure will give a positive impact on that particular person's health. A complete thorough analysis has to be performed for every patient, including the outcome of the investigation and the impact on health. This requires large resources and the involvement of different professionals. The studies available use simpler methods of estimating the justification level [5, 6, 7], as did the Swedish study which was based on the evaluation of referrals [8].

The Swedish study design and results

The referrals for all CT-examinations that were conducted in Sweden during one day were requested, which resulted in a total of 2435 CT-examinations. Eighteen physicians were engaged as evaluators of the justification of the examinations. Predominantly each examination was evaluated by two physicians, one clinician and one radiologist. A CT-examination was regarded as justified when the examination was judged as accurate for this patient.

In addition to patient data, information was available on the organisation as well as health care region the prescriber belonged to and on the anatomical region examined. About 20 % of the examinations were found to be not justified. Further analysis is required.

Making use of the results from the assessments

Three aspects will be discussed here; differences found for different types of examination, examinations performed in different health care regions and for different prescribers.

Different health care regions

Due to the fact that the study involved the whole country results for different health care regions can be derived, figure 1.

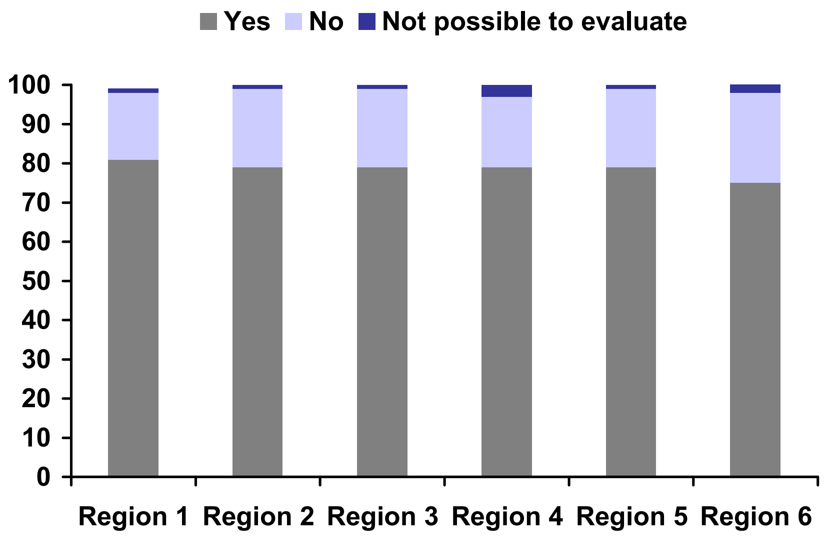

Fig 1. Level of justification (%) in different regions, all examinations.

No difference can be found between the health care regions. However, the data comprise all types of examinations, referred to for various reasons. These highly aggregated data could very well hide important findings.

Different types of examinations

In figure 2 the level of justification for different types of examinations is shown.

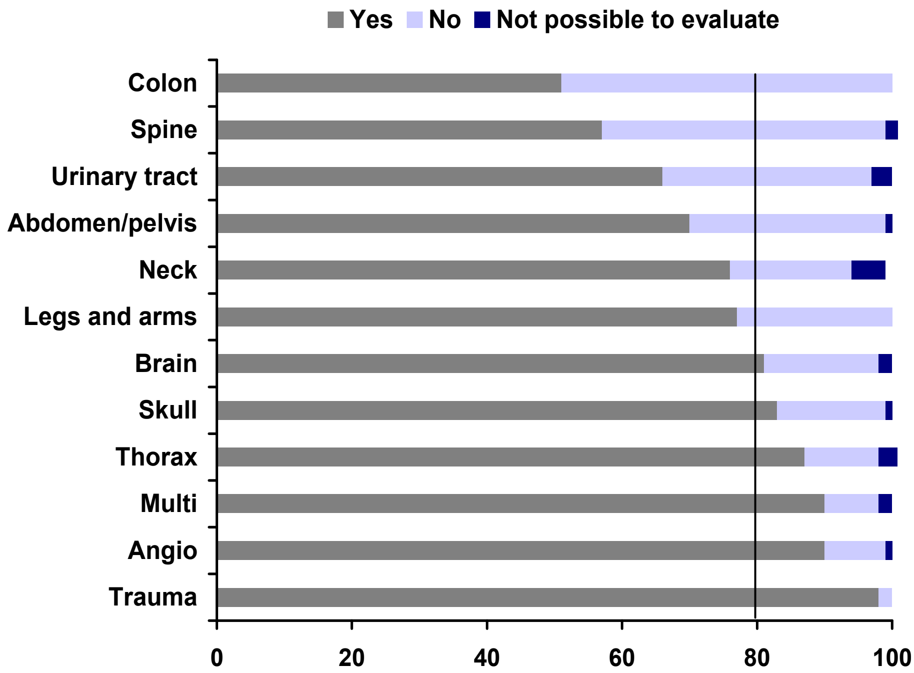

Fig 2. Level of justification (%) for different types of examinations.

The level of justification is different for different types of examinations. For example, more than 45 % of the colon and spine examinations were unjustified. On the other hand, for the trauma and angiographic examinations the figure was less than 10%.

Different health care regions and different types of examinations

Large differences between health care regions were found when going into more detail in the analysis. Figure 3 gives fraction of those spine examinations (in %) in the various regions which that should have been replaced by a MR or ultrasound examination according to the opinion of the evaluators.

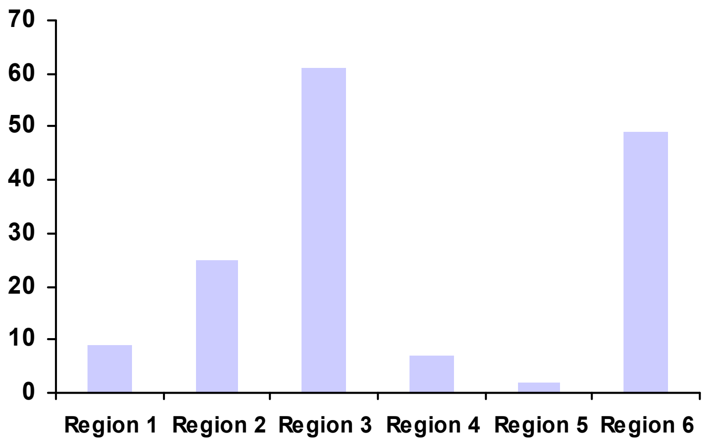

Fig 3. The fraction of spine examination that should have been replaced by a MR or ultrasound examination.

For region 3 and 6 more than 50 % of the CT examinations should have been replaced by MR or ultrasound examinations according to the opinion of the evaluators. For the regions 1, 4 and 5 this figure was less than 10 %. No obvious reason for these differences was found.

Prescribers from hospitals and health care centres

It is also interesting to study the significance of the prescriber's place of work. The results for examinations referred from hospitals and from health care centres respectively, are shown in figure 4.

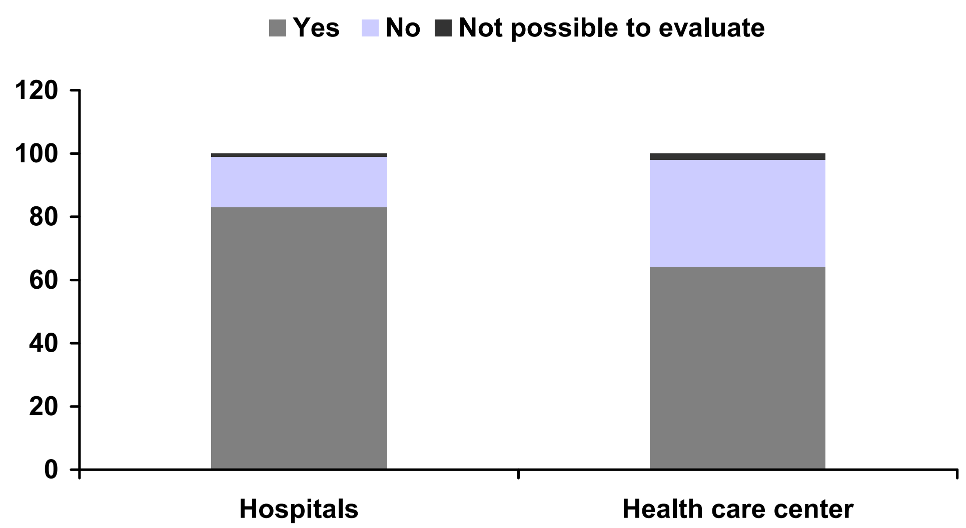

Fig 4. The level of justification (%) for examinations from hospitals and health care centres.

Clearly a difference exists. A larger fraction of examinations referred from health care centres were unjustified. In this context, it should not be forgotten that the radiology departments did agree to perform the examinations.

Experiences from the study

The study showed that assessment of justification is not an easy task, and unambiguous results cannot be expected. As could be predicted the evaluators did not always agree on what was justified since consensus not exists when to perform what type of examination. The need for more elaborated guidelines and discussions about these guidelines is evident.

The methodology differs between different studies; therefore when comparing the results it is important to consider the design of the studies. Our study gave an overall figure of 20 % unjustified CT examinations, however, when looking at specific examinations this figure varies between 2 and 50 %. Other studies have shown figures for unjustified examinations varying between 4 to 40 %. This is an indication that unjustified examinations are a considerable problem that must be dealt with [5, 6, 7].

Addressing radiation risk

In the Swedish study we have not specifically addressed radiation risk, apart from the fact that CT is a relatively high radiation dose examination. The risk, any risk, is difficult to assess for one individual in advance. The concept of effective dose and risk calculations is not directly applicable for an individual [9]. Thus, the risk assessment, any risk assessment, has to be performed for a group of individuals. Therefore the radiation risk has to be accounted for in the evaluation of different types of examinations, on the more generalised level.

Conclusion - more multidisciplinary work is needed

This study is an assessment of justification, but it is not a full evaluation of diagnostic procedures. That will include evaluating diagnostic performance, diagnostic impact, hazards (including radiation), costs and impact on health taken into account. It is also important not to forget that a prerequisite for a justified examination is a high quality (optimised) examination and a qualified evaluation of the examination. In order to improve the situation a multidisciplinary approach is needed.

This study shows that there are disagreements between different professionals. Much work has to be done on a national and sometimes international level in order to minimise these differences. The subjects that could be addressed are; what to expect (and not to expect) from the diagnostic procedure, when to perform (and not to perform) a certain diagnostic procedure and how (and how not) to perform the diagnostic procedure. This should include a true evaluation to what extent the diagnostic procedure is indicated and contributes to good health. For the radiation protection community a well-known example of assessing justification is mammography screening for early detection of breast cancer. Here the evaluation included the cost, the radiation risk, the diagnostic performance, the ability to treat disease, preparedness for handling false positive findings and the acceptance of false negative results [10, 11]. This example shows that there are interdisciplinary issues so complex that collaboration on a European level would be recommended.

For the individual patient a robust system including guidelines, sufficient clinical indicators, communication with the patient and between professional and competent experienced judgments remains a prerequisite making the best use of diagnostic radiology. In the development of guidelines the radiation risk and optimisation of examinations are important issues to deal with and this is not possible without involving the radiation protection community.

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On the patient dose recording and reporting

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Abstract
Romanian authorities built a regulatory framework for implementing the radiation protection requirements in the field of medical exposures. In this regard, among other actions, it was issued a legal binding document that requires a system for recording and reporting, at national level, the patient doses for each X-ray generator or irradiation source used in diagnostic or treatment. The responsibility for recording the data belongs to the practitioner, and the correctness check is radiological safety officer's duty. Moreover, in case of diagnostic radiology, the patient shall be informed in a written form about the dose he received during the procedure. The aim of this paper is to make an analyse on the appropriateness of parameters required by regulation and on the patient radiation protection benefits and costs in the context of a recognised need for harmonisation of quantities, terminology and procedures for dosimetry in diagnostic.

Introduction
The Article 12 of Council Directive 97/43/EURATOM requires Member States to ensure that the distribution of individual dose estimates from medical exposure is determined for the population and for relevant reference groups of the population in Member States. Following the request regarding the implementation of Directive's requirements the necessary regulations and administrative provisions brought into force by the Romanian health authorities by issuing the Order no.1542/2006 of the Ministry of Health. The provisions of this regulation establish a mechanism to collect and to report the patient's doses at the national level, involving in this process all the diagnostic radiology, nuclear medicine and radiotherapy departments within the country. In order to harmonize at the national level the reporting system, the Order no. 1003/2008 of the Ministry of Health has been published. Among the other information, the effective dose of all patients shall be reported. The regulations seem to be burdensome and difficult to be respected because the provisions have no regard to ICRP recommendations and there are no adequate procedures and available resources.

Regulation on recording and reporting the dose of the patients
The objective of the current regulation is to build a system for recording and reporting of the data regarding the medical exposure to ionizing radiation of the population. In this regard, for each X-ray generator or irradiation source used for medical purposes, it

is required to be ensured an individual recording system either electronically or on the paper for monitoring the data results from medical exposures to ionizing radiation of the population.

According to the legally binding provisions, the recording system shall include at least the following information:

1. Patient data:
   a) name and surname;
   b) date of birth;
   c) personal identification number;
   d) gender;
   e) height;
   f) weight;
   g) in cases of females of childbearing potential it shall be mentioned the possibility of pregnancy and during pregnancy, if necessary;

2. data on individual exposure parameters used, as appropriate:

Diagnostic radiology	Nuclear Medicine	Radiotherapy
a) Projection (examination)	a) Examination or treatment type	a) Procedure type
b) Source-patient distance (cm)	b) Radiopharmaceutic/radionuclide	b) Source-patient distance (cm)
c) Field (cm x cm)	c) Administrated activity (Bq)	c) Field (cm^2)
d) kV		d) Target volume (cm^3)
e) mA		e) The dose/session (Gy)
f) mAs		
g) timp (s)		
h) DAP(mGy x cm^2)		

3. data for patient dose assessment, as appropriate:

Diagnostic radiology	Nuclear Medicine	Radiotherapy
a) DAP indication (mGy x cm2)	The effective dose estimated	a) Number of sesions
b)Skin absorbed dose (mGy)		b)The total dose in target volume
c) Absorbed dose rate at the intensifier screen surface (mGy)		
d) CTDI (mGy)		

The responsibility for recording the above data is assigned by the regulation to the practitioner and the responsibility for checking the accuracy belongs to the radiation protection officer. The license holder shall centralize the data at the institution level according to the information system of the Ministry of Public Health. Moreover, the patient shall be notified in writing regarding the result of the examination and the dose received due to the medical exposure to ionizing radiation. The data registered could be controlled by the competent authorities and shall be archived for a minimum 5 years period. The centralized data shall be reported quarterly by the license holder to the ionizing radiation hygiene laboratory based on territorial assignation. The Ionizing Radiation Hygiene Laboratory forwards the centralized data to the county public health authorities and Bucharest, up to the end of the next month of the reporting month. Counties and Bucharest public health authorities shall annually report to the Ministry of Public Health the centralized data on the medical exposure of the population in the counties and in Bucharest. The Institute of Public Health centralizes the collected data reported by counties and Bucharest, issues an annually report and informs the Commission of Radiology, Medical Imaging and Nuclear Medicine which belongs to the Ministry of Health. The Commission of Radiology, Medical Imaging and Nuclear Medicine review the report provided by the health authorities and proposes an action plan for optimizing the radiological procedures. The annual report data on medical exposure of the population and the plan for optimizing radiological procedures are subject to approval by Minister of Public Health and shall be forwarded to the National Commission Nuclear Activities Control, and international institutions concerned, as appropriate. Violating the provisions of this Order shall be punished administrative, disciplinary, or criminal contravention, as appropriate, under the law. In order to detail the requirements of the Order 1542/2006 on recording and reporting the patients' dose and to establish a unitary system for recording and reporting, Ministry of Public Health issued the Order no. 1003 of May 19, 2008 for approving the use of forms for registration and reporting data on medical exposures to ionizing radiation. The provisions of this regulation establish the forms for registration of the individual parameters of exposure in diagnostic radiology, in nuclear medicine and in radiotherapy, for recording data on individual medical exposure to ionizing radiation and for the centralized reporting data on medical exposure to ionizing radiation by the hospitals that provide radiology, nuclear medicine and radiotherapy. In order to be in accordance with these requirements the hospitals should report, among other information, the number of examination/type of procedure for different age/sex distribution and the corresponding dose, mean dose, and effective dose, including for the X-ray dental examinations and for nuclear medicine.

Discussion

Actually, the issued regulations require a national study without having the necessary resources to perform it properly because guidance on the effective dose assessment methodologies and medical physicists with particular expertise in diagnostic radiology dosimetry are available neither in hospitals nor in the Ionizing Radiation Hygiene Laboratories. Moreover, the mandatory requirements focus only on the dose patient and do not take into account the image quality indicator and the fact that there are large uncertainties in the coefficients used to convert dose measurements to effective dose. According to the ICRP recommendations, the main uses of effective dose are the prospective dose assessment for planning and optimization in radiological protection, and demonstration of compliance with dose limits for a regulatory purpose which is not the case of medical exposures. Nevertheless, the effective dose is useful as a comparator for relative magnitudes of doses for different types of examination, but it must not attribute to this more certainty than is justified. The use of effective dose for assessing the exposure of patients has severe limitations that must be considered when quantifying medical exposure. Effective dose can be of value for comparing doses from different diagnostic procedures and for comparing the use of similar technologies and procedures in different hospitals and countries as well as the use of different technologies for the same medical examination. However, for planning the exposure of patients and risk-benefit assessments, the equivalent dose or the absorbed dose to irradiated tissues is the relevant quantity. Moreover, the regulations do not take into account that for the examinations where particular radiosensitive organs receive significant doses, it is recommended to give estimates of doses to these organs in the assessment, rather than effective dose. Also, it is recognized that for complex examinations, detailed analyses of dose for each projection should be undertaken. This could allow an indication of effective dose to be obtained for particular examinations. However, this level of detail requires a significant effort to provide information that is of limited value and involve an unjustified cost and very limited benefit for the patient. Therefore, the absence of a certain level of strictness will determine a high level of inaccuracy that may negatively influence the results and conclusions despite of workload spent and obtaining no significant benefits for the patients.

Conclusions

A national study performed according to Radiation Protection no.154 - European Guidance on Estimating Population Doses from Medical X-Ray should replace the regulations for the estimation of population doses as required by the Article 12 of the EC Medical Exposure Directive. It is recognized that the collection of data on medical exposures is important to provide a large national data bank of doses for the specific range of hospitals, but this goal could be achieved without burning regulations and unacceptable costs. It should be considered the suitability of the involvement of the medical radiology professional societies, in order to establish the priorities, methods, quantities, cost effectiveness and for the assessment of the results. Moreover, in order to decrease the workload and to increase the accuracy of the collected data for future population dose estimates it should be established electronic information systems in hospitals as input to the national database. In addition, the approach that includes the dental X-ray examinations in the scope of the medical exposure data collection of the population should be reconsidered.

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The progress in Minimizing Activity and Dose with Enhanced Image Quality by Radiopharmaceutical Administration project

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Abstract

The introduction of new and more beneficial diagnostic procedures combining CT and PET or SPECT is related to a rapid growth of medical exposures. The European collaborative project MADEIRA (Minimizing Activity and Dose with Enhanced Image Quality by Radiopharmaceutical Administration) within the 7th EURATOM FP aims to improve 3D nuclear medical imaging in terms of increase of spatial and temporal resolution and also of reduction of the radiation exposure. The project is organized in 5 work packages, which are identified as:WP1 assessment of clinical data, WP2 PET magnifier probe development, WP3 physics-based image processing; WP4 biokinetic and dosimetric modelling; WP5 training and dissemination activities. In particular in WP1 quantitative biokinetic data were collected in patients after administration of 18F-choline and those data were used to develop detailed compartmental models and to evaluate the internal dose to the patient. Moreover, the biokinetic models were used as a basis to define more efficient protocols for data collection (WP4). The collected tomographic images were used to test different reconstruction algorithms (WP3) including some newly developed ones for which a patent is pending. In addition a special phantom for checking image quality in nuclear medicine imaging was designed and also for this a patent is pending. WP2 deals with the construction of a PET insert probe. The different modules of the probe have been developed and their performances monitored, and the probe prototype is going to be tested in clinical conditions with the use of the developed reconstruction methods. Finally, training and dissemination activities include the organization of two training courses (Milan, November 2008; Malmö November 2009), and the Malmö Conf. on Medical Imaging (June 2009) as well as the tutoring of graduate and doctoral students and young post-doctoral researchers.

A study for the application of prospective approaches for safety assessment in new radiotherapy techniques

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Abstract

Modern and advanced planning and delivery technologies and techniques, recently introduced in radiation therapy for improving the treatment outcome, call for new challenges in terms of radiological protection of the patient and risk management of potential accidental exposure. The increased complexity related to these technological and process changes, places indeed new demands on quality assurance programmes and new attitudes and approaches of safety culture. In particular, prospective techniques for risk assessment in modern radiotherapy practices are recommended. A research project aimed to study these aspects and to sensitize the Italian scientific community in the use of prospective approaches has been recently launched by a working group of the Italian Association for Medical Physics (AIFM). In the first phase of the project intraoperative radiation therapy (IORT) practices carried out with mobile electron linear accelerators are considered. Process flow diagrams describing the activities, the instruments and the adopted procedures are properly developed in view of the identification of possible failure modes and the assessment and management of the associated risk. The methods and the analyses performed during this first phase are presented in this work. In a second phase, similar approaches will be considered and applyed for the risk assessment in other modern radiation therapy techniques, including hadrontherapy.

Introduction

Radiotherapy technologies and methodologies to plan and accurately deliver the dose to the target volume significantly improved in the recent years. Intensity modulated radiation therapy (IMRT) is becoming the standard technique in most of the developed countries, and, since the start-up marketing, more than 200 tomotherapy machines are operating throughout the world. Similarly, other modern radiation therapy techniques like intensity modulated arc therapy (IMAT), image guided radiotherapy (IGRT), stereotactic body radiotherapy (SBRT) are gradually gaining ground in many radiation therapy departments. Also the number of patients treated with intraoperative radiation therapy (IORT) is increased as effect of the recent development and rapid growth of dedicated treatment units. Finally, centres and facilities devoted to hadrontherapy are spreading in various countries.

The increased complexity related with the technological and process changes, places new demands on quality assurance (QA) programmes and new attitudes and approaches of safety culture. In fact, traditional instruments and procedures for QA, for the characterisation of radiation beams, and for the verification of the delivered dose may be no more adequate and effective for the modern radiotherapy techniques.

A number of accidents in conventional external radiotherapy have been extensively investigated and the lessons learned from events with major consequences have been reported and disseminated with the aim to prevent reoccurrence, following the so called “retrospective approaches” (IAEA 2000; ICRP 2000). Some general lessons learned from accidents occurred with conventional radiotherapy can be applicable also to new technologies. In particular, the presence of well-trained staff with necessary education background and specialised training; the draw up of written and comprehensive procedures and the identification of responsibilities are key-elements for a successful treatments, independently of the techniques used (Ortiz Lopez 2009).

However, in order to assess and manage the new risks deriving from the use of innovative radiotherapy methodologies, retrospective approches are not enought adequate since they have the intrinsic limitation of being confined to reported experience, thus unreported events or other latent risks remains unaddressed. Therefore, prospective approaches have to be implemented with the aim to find out all the elements that could go wrong and to identify, *a-priori*, all the potential hazards that might occur during a radiotherapy treatment. Prospective approaches are widely applied in high-risk industries, such as the nuclear, aviation, and chemical industries to assess and manage risk. Recently, the interest in using these methodologies for safety assessment in complex medical practices, like radiotherapy is gaining importance, and literature on this topic is rapidly increasing (Ekaette et al. 2006; Ekaette et al. 2007; Ford et al. 2009; Huq et al. 2008; Ramirez 2009; Thomadsen et al. 2003; Vilarguit et al. 2008).

On the basis of this new needs, at the beginning of 2010, a multidisciplinary working group (WG) involving experts from the European Institute of Oncology (IEO), from the National Centre for Oncological Hadrontherapy (CNAO) and from the Physics Department of the State University of Milan has been established in the frame of the Italian Association for Medical Physics (AIFM) with the aim to sensitize the Italian scientific community in the use of prospective approaches in radiotherapy practices. The components of the working group have skills and experience in various areas including radiation therapy, oncology, health physics, dosimetry, radiation protection, statistics, computer science and modelling complex systems. Considering the experience on IORT gained by the components of the WG, it was decided to start to become familiar with prospective techniques using this irradiation methodology.

Intraoperative radiation therapy (IORT) using electron beams is a special treatment modality involving the delivery of a single-fraction large radiation dose to the exposed tumour or tumour bed at the time of surgery. Due to its multi-disciplinary nature, this procedure needs a well-coordinate team approach by anaesthesia, surgery, radiation oncology and medical physics staffs. For many years, IORT has been commonly performed using a conventional linear accelerator located in a radiotherapy treatment vault or, only in a few cases, in a dedicated and shielded operating room (OR) (Palta et al. 1995; Beddar et al. 2006, ISS 2003). The recent development and rapid growth of dedicated treatment units, namely mobile linear accelerators, have changed the scenario in a relevant way, allowing a larger number of patients to undergo normal surgical procedure and immediate irradiation without the need of transportation outside the operating theatre.

Mobile linear accelerators are quite compact and only operate in electron mode up to 10-12 MeV for radiation protection restrictions, thus offering the great advantage that they can work in almost any conventional OR.

At the European Institute of Oncology (IEO) in Milano (Italy), since 1999 more than five thousands patients have been treated with IORT, using two dedicated mobile linear accelerators installed in different ORs: a Novac7 (NRT, Italy) and a LIAC (Sordina, Italy). At IEO, IORT is mainly used in the conservative treatment of early-stage breast cancer (figure 1), either exclusively for partial breast irradiation at the dose of 21 Gy or as an anticipated tumour-bed boost at 12 Gy (Orecchia et al. 2003; Veronesi et al. 2005; Ciocca et al. 2003, 2006, 2009; Ivaldi et al. 2008).

In terms of the risk of accidental radiation exposures to the patient, the main critical aspect in IORT seems to consist in its single-shot nature itself, unlike conventional radiation treatment modalities, where daily dose fractionation can somehow permit dose compensation if a systematic error in the whole treatment process is detected. Furthermore, in IORT up to now no image-based treatment planning is normally performed prior to the dose delivery, treatment parameters (like field size, beam energy, blocks or surface bolus) being defined in the minutes immediately before the irradiation and Monitor Units consequently calculated by the medical physicist.

The aim of this work was to present the preliminary results of the application of the failure mode and effects analysis (FMEA) prospective approach to IORT, starting from the definition of both process and fault trees, then through the assignment of a score for each potential failure mode using the risk probability number (RPN) and finally suggesting additional safety measures for process improvement.

Material and methods

IORT process tree

Following the guidelines recently proposed by the WHO (Who 2008), the whole radiotherapy treatment process can be divided into ten stages: 1) assessment of patient, 2) decision to treat, 3) treatment protocol prescription, 4) positioning and immobilization, 5) simulation, imaging and volume determination, 6) planning, 7) treatment information transfer, 8) patient set-up, 9) treatment delivery, 10) treatment verification and monitoring. Nonetheless, the strong specificity of IORT using electron beams, implying single-fraction dose delivery, makes steps from n. 4 to 8 not applicable. The state-of-art of IORT assumes the lack of a pre- or peri-operative simulation and CT-based treatment planning, while future improvements will necessarily require the availability of dedicated imaging devices with volumetric capability (CT or cone-beam CT modality) located directly inside the OR. The first three steps were not included in this study, since they are part of the medical decision making process and they will need to be analysed at a different level of the work. The successful performing of the tasks of commissioning, calibrations and maintenance of the equipment was taken as a base for the scope of this analysis. Thus, the IORT process tree was restricted to one phase, including treatment delivery and verification, and sub-processes were identified.

Failure mode and effects analysis (FMEA) for IORT

Failure mode and effect analysis is an established pro-active method, widely used in industry, that helps to identify and counter weak points occurring in a complex process. This technique has been also suggested by the Task group 100 of the American Association of Physicists in Medicine (AAPM) as a powerful tool for the implementation of suitable QA programs in modern radiotherapy. Consequently, it has been recently applied for safety evaluation in radiation oncology (Ford et al 2009), including IMRT (Huq et al. 2008).

FMEA focuses on the sub-processes, trying to identify what could go wrong (the failure mode), the potential causes of failure and the potential related effects. Since the goal of FMEA is to rank the failure modes in order of importance, for each failure three attributes are assigned: the occurrence rating (O); the severity rating (S), and the detection rating (D). Conventionally, a 10-point scale is used for scoring each of these categories, with 10 being the most severe, most frequent, or least detectable. The product of these three indices gives the risk probability number (RPN=S x O x D). The failure modes that have the highest RPN indicate the areas of greatest concern where corrective actions should be implemented in order to reduce the overall risk of accident.

Results and discussion

Within the stage of treatment delivery and verification, the identified sub-processes in IORT using a mobile electron linear accelerator are reported in table 1.

Nine potential failure modes were identified and for each of them potential causes and effects were investigated (table 2). Following the FMEA approach and on the basis of the authors' experience, a risk probability number will be assigned to each failure mode and presented. Among the nine identified failure modes, the undetected linac malfunctioning, wrong MU calculation and misalignment of internal shield appeared as the most critical. The former two failures modes are strictly connected to the specificity of IORT technique, where the treatment machine in the OR is usually switched on few minutes before each patient irradiation, then switched off, unlike conventional stationary linacs in the Radiotherapy Department, which are always kept in the operational mode during all the working day; moreover, in IORT the treatment parameters cannot be chosen in large advance and consequently MU calculation has to be rapidly performed. Concerning the internal shield, it is commonly used for breast irradiation to protect underlying normal tissues, such as the lung, heart, ribs, pectoral muscles, so potential misalignment, combined with the use of high beam energy, could result in undesired acute and late toxicity.

The following additional safety measures to be implemented aiming at the IORT process improvement in terms of patient safety were identified: a) implementation and extensive use of in-vivo dosimetry procedures, using real-time suitable detectors, like MOSFET sensors, allowing immediate action levels to be applied in case of detected dose discrepancies; b) independent, double-checking of MU calculation and data entry to be stated as mandatory, together with the implementation of an automatic system for MU calculation; c) identification of a dedicated radiotherapy staff, composed by the radiation oncologist, therapist (also in view of safe manoeuvring of the mobile linac, especially critical when close to the patient) and medical physicist, adequately trained to IORT procedures.

Conclusion

The FMEA method as a prospective approach to avoid accidental exposures has been applied to IORT using mobile and dedicated electron linear accelerators directly operating in the operating room. Potential failure modes, causes and effects of failures have been investigated, as well as three additional safety measures for process quality improvement proposed. Prospectively, further investigation will be oriented to the application of more complex prospective methods, such as probabilistic safety assessment (PSA) to IORT and other newer radiation therapy technologies, like hadrontherapy.

Figure 1. Example of an IORT procedure in the operating room at IEO, Milan, for a patient affected by early-stage breast cancer, using a mobile electron linear accelerator.

Table 1. Sub-processes of the stage "Treatment delivery and verification" in IORT. Legend of abbreviations: CTV=clinical target volume; RO= radiation oncologist; RT=radiation therapist; MP=medical physicist; OR=operating room; MU=monitor units.

Sub-processes of the stage "Treatment delivery and verification" in IORT.
• Machine start-up
• Preparation of surgical bed (for optimal exposure of CTV to radiation beam)
• Normal tissue protection (mechanical displacement, internal shielding)
• Selection of applicator (size and bevel angle)
• Selection of beam modifiers (bolus, blocks)
• Evaluation of target thickness
• Confirmation of dose prescription
• Beam modifier placement
• In-vivo dosimeter placement
• Applicator placement
• Treatment volume documentation (photographs, clips, x-rays)
• Beam stopper placement (below surgical table)
• Applicator docking to the linac
• Temporary vertical shielding placement (around surgical table)
• OR evacuation (all personnel)
• Selection of beam energy
• MU calculation
• Dosimetry sheet compilation and signing (MP)
• Data entry (treatment parameters) on treatment console
• Physical delivery of radiation dose (start button pressed)
• Confirmation of dose delivery using dosimeters
• Applicator-to-linac undocking
• Removal of all IORT devices and dosimeters
• Linac switch-off
• Treatment recording, reporting (in pt chart or medical record) and signing (RO/RT/MP)

Table 2. Application of failure mode and effects analysis (FMEA) in IORT, for the stage "Treatment delivery and verification".

Step	Potential failure mode	Potential causes of failure	Potential effects of failure
Normal tissue protection; applicator placement	Misalignment of internal shielding	Shielding displacement, wrong applicator placement	Unintended normal tissue irradiation
Selection of the applicator	Inadequate safety margins	Underestimation of CTV extension, inadequate selection of the applicator	CTV underdose
Evaluation of target thickness; selection of beam energy	Inadequate energy selection	Human error in the measurement of target thickness or consultation of dosimetry atlas, failure in the communication between operators	CTV underdose or viceversa unintended normal tissue irradiation
Applicator placement	Inadequate preparation of the area to be treated or inaccurate placement of the applicator base	Biologic fluid accumulation, tissue protrusion inside the applicator, air gap presence	Wrong dose distribution
Applicator placement	Geographic miss of the CTV	Inadequate localization of the CTV; applicator displacement due to physiological movements	CTV underdose; unintended normal tissue irradiation
Applicator docking to the linac	Inaccurate docking (for soft-docking systems)	Malfunctioning and tolerances of the alignment optical system	Wrong dose distribution
MU calculation	Wrong MU calculation	Human error in the calculation, failure in the communication between operators	Wrong dose delivery
Data entry at the treatment console	Incorrect data entry at the treatment console (beam energy, MU, field size)	Human error in manual data entry, failure in the communication between operators	Wrong dose distribution
Physical delivery of radiation dose	Undetected failure of the linac	Linac malfunctioning, linac operated in physics rather than clinical mode, linac start-up procedures not correctly followed	Wrong dose delivery and/or distribution

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Radiation doses received by children during CT examinations

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Abstract

Introduction: Computer Tomography revolutionized radiological diagnosis, the number of CT examinations growing quickly. Due to its nature, computer tomography implies higher radiation doses than those in classical radiology. An increased number of examinations, if justified, must be regarded as a benefit for the patient. Monitoring the patients' trend in CT is of particular importance as technology evolves rapidly towards extremely efficient equipments capable of acquiring 4, 8, 16, and 64 slices per rotation. At the same time, reducing examination time led to increasing CT usage as a diagnosis means.

Material and method: The present study refers to four departments of radiology, three of them being private. The data regarding the number and type of performed examinations as well as the examination protocols come from the registers of these departments. One of the equipments is single-slice (SSCT) and the others are multi-slice (MSCT), all of them having instruments to compute and display the dose length product (DLP) and the dose index (CTDI), according to the protocol in use. We estimated the effective dose and compared them with the reference values in other studies.

Results: The skull and abdomen examinations represented a large majority. The effective dose average varied from one department to another. In comparison to other studies, the average abdominal effective doses were below the values in other studies for SSCT (5.90 mSv compared to 12 mSv), while for MSCT were above them (7.29 mSv compared to 6.59 mSv).

Conclusions: The variation of the patients' dose for a particular type of procedure represents a premise for improving the optimisation process of protecting the CT examined patients. In order to diminish doses, it is very important to adjust some parameters such as the multi-amperage or the number of examined sections depending on the clinical requirements, without compromising the image quality.

Presentation withdrawn

The risk of medical exposure to ionizing radiations in pediatrics

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Abstract
Radiological examination of children is extremely significant since it stands for the safe and most accurate investigation method. However, one shouldn't neglect the risk of cancer induction that is up to three times higher for children than for adults. This risk is given by the children's anatomic features and their longer life expectation. A special attention is paid to premature children and to those showing clinical complications leading to longer hospitalization periods, thus increasing the number of radiological examinations.

The study has been conducted in the New Born and Pediatrics Clinics of Dolj County Hospital, Romania, between 2007 and 2008. The focus was on the frequency of radiological examinations and the entrance surface doses within the most relevant radiological procedures. The measurements have been carried out using the multi-functional device for testing the quality of radiological systems RMI-242.

The lung X– ray is the most frequently used type of examination, the measured doses being up to four times higher than the reference levels in the literature.

The utilized radiological equipment is not especially designed for the purpose of pediatric radiology, making thus impossible to choose the specific parameters which lead to smaller doses without altering the quality of the radiological image.

What's worrying is not the dose per type of procedure, but the large number of examinations performed per one hospitalization period. All these lead to cumulative doses, to which it is likely that several others be added during childhood.

Introduction
Medical diagnosis using ionizing radiations to all groups of children requires a special attention because of the children stand for that category of population with the highest radio – sensitivity and for whom the risk of cancer induction is 2 up to 3 times higher than that for adults. Such a risk is determined by both a longer life expectancy in children, this meaning a higher potential of expression of any possible effects of ionizing radiations, as well as by the anatomic, biochemical and physiological characteristics, different from one age group to another and which are also very distinct from an adult. The big number of exposures, the incorrect use of the physical

parameters of radiological equipments, the poor performances, these may all eventually lead to higher entrance surface doses to patient (Iacob et al. 2007).

Pediatric radiologists should very well know the relation between the benefit of radiological examinations and the risk of cancer induction. The ALARA principle is the best guidebook radiologists can use in order to establish the risk benefit - balance when talking about radiological examinations (Cook et al. 1998).

Knowing the current trends of the doses received by patients during radiological examinations in each and every section, may represent a guidebook when choosing the measures required to be taken for the dose reduction in view of an efficient optimization of the patients' radioprotection and for the purpose of minimizing the risk that such an exposure to ionizing radiations involves (Cook et al. 2001).

Material and methods

The study has been carried out in the anesthesia and intensive care unit (ATI) and in the pediatric department, in the greatest hospital in Dolj County, between 2007 and 2008. The study describes on the one hand the frequency and distribution of the radiological examinations, and on the other hand it estimates the entrance surface doses for the patients during the most significant radiological procedures. Radiological examinations are carried out with two types of installations:

- a mobile installation with X rays - the Polymobil 10 type -, with a total filtration of 3.4 mmAl
- a fix installation of the ELTEX 400 type, with a total filtration of 2.5 mmAl.

The measurement of physical parameters for the two installations and of the entrance surface doses have been carried out with a multi – functional instrument for testing quality of radiological systems of the RMI – 242 type. A statistical processing has been conducted, by comparing the average values, obtained for the entrance surface doses with the reference values by means of the Student test (Guide 2002).

Results

Frequency of radiological examinations within the ATI department:

Table 1. Radiological examinations frequency per years and types of procedures.

Examination type	2007	2008
Pulmonary - abdomen	270	249
Pulmonary	11	12
New born babies	2974	3221

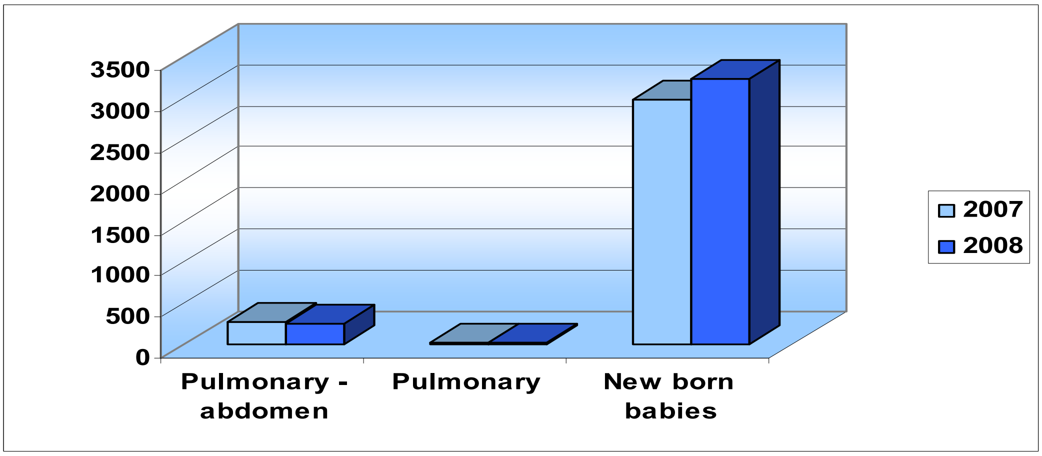

Fig. 1. Radiological examinations frequency per years and types of procedures.

One may find that the most frequently performed radiological procedures have been the pulmonary – abdominal ones, as a consequence of breathing and digestive problems of new born babies and of the premature babies. In 2008 one may find a slight decrease in the number of radiological investigations, although the number of new born babies has increased. Such decrease might be explained by the fact that both the neonatology physician and the radiologist have tried to get, wherever possible a number of radiological information or medical records which are relevant so as to avoid any useless medical irradiations.

The distribution per child of the investigations performed was:

Table 2. The distribution per child of investigations performed.

No. of x – rays	***1***	***2***	***3***	***4***	***5***	***6***	***7***	***8***	***9***
No. of children	132	234	53	27	25	41	15	10	5

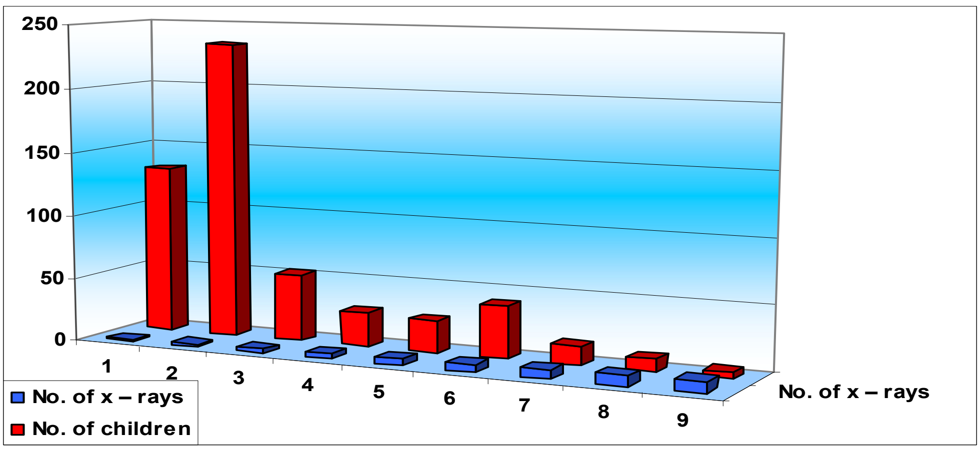

Fig. 2. The distribution per child of investigations performed.

The longest hospitalization periods within the Intensive Care Unit (maximum 70 days), have been for prematurely new born babies, 2 being the high number of radiological examinations conducted per child. The maximum number of x – rays examinations carried out per child is significantly low, namely nine, and they have been performed to premature babies having the smallest weight at birth. (1.5 kg).

Table 3. Frequency of investigations within the pediatric radiology department.

Year	Pulmonary	Skull	Spine	Extremities	Pelvis
2007	2550	1066	424	1331	114
2008	2607	785	478	2805	165
Total	4157	1851	902	4136	279

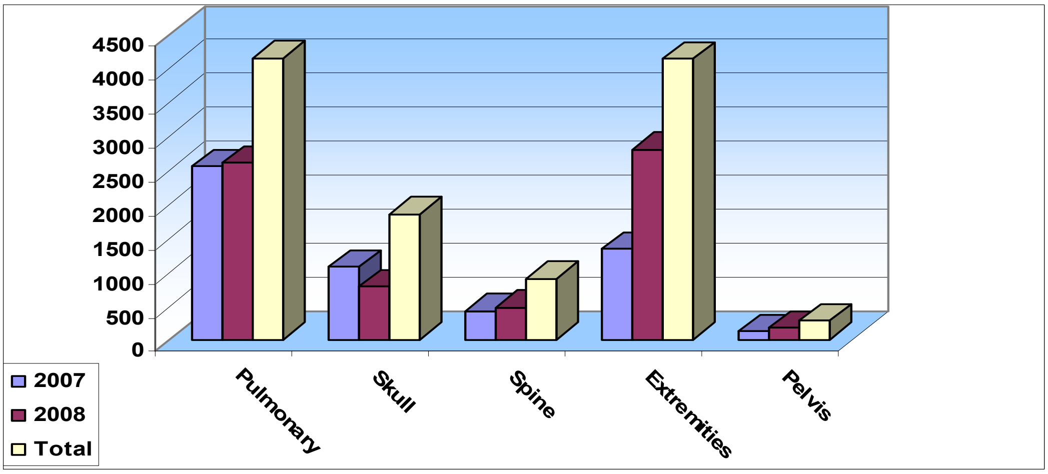

Fig. 3. Frequency of investigations within the pediatric radiology department.

In terms of the distribution of radiological examinations per anatomic areas, one finds that pulmonary and extremities radiological examinations are the most frequent ones, being the most characteristic for this category of population.

The average values for the entrance surface doses per type of examination for the children within the Anesthesia and Intensive Care Unit are:

Table 4. Entrance surface doses.

Examination Type	Entrance surface doses µGy	kV/mAs	Ord. 285* µGy
Pulmonary -abdomen	94	52-58/2.5	-
Pulmonary	90	52-57/2.5	80

* Order no. 285/79/2002 of the Ministry of Health and Family and of the National Commission for Nuclear Activity Control regarding people's exposure to ionizing radiations in case of medical examinations.

The highest value of the entrance surface doses is in the case of the pulmonary – abdominal examinations, the difference from the pulmonary examination being relatively small. Even if the doses for the first type of an investigation are higher, it is preferable that when the physician requires information for the two areas, one shall perform only one single exposure, thus avoiding the superposing. The increased values of the doses might be explained by the fixed value of filtration (3.4 mmAl), without any possibility of an additional filtration (Wraith et al. 1995).

Here are the average values for the entrance surface doses per type of examination and per age groups, for the children within the pediatric department and the comparison with the reference levels are presented in table 5.

Table 5. Average values for the entrance surface doses per type of examination and per age groups in mGy.

Examination Type	0 – 1 years	1 – 3 years	4 – 7 years	5 years	8 – 11 years	12 – 15 years
	Craiova	Craiova	Craiova	Ord. 285	Craiova	Craiova
Pulmonary AP, PA	0.28±0.01	0.36±0.02	0.40±0.02	0.1	2.88±0.10	3.01±0.12
Pulmonary LAT	0.39±0.01	0.40±0.02	0.61±0.04	0.2	3.12±0.14	3.25±0.15
Skull AP	0.88±0.03	1.01±0.07	1.52±0.06	1.5	3.39±0.14	3.45±0.16
Thoracic spine AP	0.83±0.02	1.18±0.09	2.81±0.11	-	3.14±0.12	3.75±0.17
Pelvis AP	0.50±0.01	1.01±0.08	1.89±0.08	0.9	3.32±0.11	3.44±0.12
Extremities	0.60±0.02	1.29±0.09	1.70±0.08	-	2.91±0.09	3.55±0.13

When comparing with the reference dose levels, for one single exposure and for the 5 year old child, one finds that in the case of the lung, the doses that we have measured are 4 and respectively 3 times higher for the two projections and two times higher in case of the pelvis. We have found comparable values with the existing reference levels, in terms of the skull x – ray examination (Sorop et al. 2008).

The values of the Student test show statistically significant differences in terms of the pulmonary x - rays (AP and LAT) and in the case of the pelvis AP ($p<0.001$) as well as a series of statistically insignificant differences in the case of the skull AP ($p>0.05$). For all the other types of procedures there are no reference levels.

Discussion

The pulmonary x – ray examinations and the extremities examinations hold the first position, being given by the age specific pathology.

The radiological equipments used both in the ATI department for babies and in the pediatric departments are not especially designed for pediatric radiology and they are not adequate from the point of view of the requirements of the standards regarding the technical parameters playing a very significant part in terms of the doses received by each and every patient (Lindskoug 1992).

For the purpose of getting accurate diagnosis–related information, a very important part is played by the quality of the radiological image, which is conditioned by the technical parameters (Mooney, Thomas 1998).

The doses received by children (which are three and four times higher than the reference levels) may also be explained by the use of the low kilo - voltage technique (which involves soft X radiations beams, associated with high doses given to the patient).

In order of getting a child's diagnosis using radiations, with a minimum risk, it is important for the medical staff to have a special training, knowing the child's anatomy and physiology, and for the environment where this medical act is carried out, to be warm and friendly.

Every clinician shall have to understand his/her responsibility, as well as the collective responsibility, for providing the quality of the medical act. All these might lead to the decrease of the number of unjustified investigations and also to the decreasing of the doses per patients.

The cooperation deficiencies and any functional particularities (faster heart beats, fast breathing and incapacity of holding breath upon command) are also other reasons for a deterioration of the quality of the image, leading to the repetition of the x – ray examinations.

The development of certain working protocols, at hospital level, would be an opportunity of improving the clinical system and the working procedures, in such a way as to increase the medical care offered to the patient (Bahnarel et al. 2007).

Conclusions

In the case of new born babies, what we should worry about is not the doses resulted per type of procedure but the fact that such examinations also have the tendency of being repeated during the child's hospitalization period, and such repeated procedures lead to cumulative doses, with several others likely to be added during childhood.

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The role of radiology technologists in radiation protection of children

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Abstract

Introduction: Radiation protection depends on many factors. This study deals with the human factor, the radiology technicians. If all technical malfunctions are excluded they are responsible for the patient dose. Depending on their education and experience technicians perform X ray examinations with various end results: image quality, entrance surface dose, patient interaction etc.

Patients and methods: The study group of 60 children (for each of three technicians 20 children) was chosen that had a clinical indication for a chest X ray examination (standard PA radiogram). The age of the patients varied between 6–12 years, the weight between 20–50 kg and the height 110–150 cm. All parents were informed about the aim and the experimental details of the study. A Shimatzu X ray machine was used in all cases. Radiophotoluminescent (RPL) and thermoluminescent dosimeters (TLD) were applied at the entrance of the beam in center of the X ray field to measure the entrance surface dose (ESD). The three differently experienced technicians were unaware of the point of the study. Parameters that were noted were the kV, mAs and the size of the field. The radiologists interpreting the radiogram, as well as the scientists interpreting the dose were blinded.

Results: The dosimetry results with both dosimetry systems show a good correlation of ESD as a function of mass body index (MBI) between two technicians. Doses were significantly higher for the third one. The ESD values varied between 0.05–0.5 mSv.

Discussion: This work shows the importance of continues education and good teamwork for dose reduction. In a sequel study after additional education interference with the same three technicians we hope to have results that would show a better dose reduction.

Issues and practices concerning pregnancy and ionising radiation

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Abstract
A literature review in relation to the irradiation of patients and staff during pregnancy was carried out. A number of publications describing procedures to be followed in the case of a pregnant patient who needs to have a radiological examination and in the case of pregnant staff who work in a department where they could be exposed to radiation were reviewed.

A review of existing practices in 13 European countries was carried out by sending a questionnaire to representatives in each country. The questionnaire addressed the following issues : the application of the 10 day rule, patients who are pregnant and need a radiological examination or a nuclear medicine examination , staff and carers who are or may be pregnant and who are working in a radiology department or in a Nuclear medicine department.

From the review, it was found that the existing practices with respect to irradiation of patients and staff during pregnancy vary enormously from country to country. There is no harmonisation on this issue at the European level. From the literature review and the review of practices, a number of ethical issues were identified and exposed and a number of conclusions were drawn.

Introduction
The Medical Exposure Directive 97/43 (C.E.U. 1997) requires special attention for the protection of the unborn and breast fed child who are exposed to ionising radiation for medical purposes. This is because the unborn child, the breast fed child as well as young children are particularly vulnerable to ionising radiation. Along with other risks, there is the risk of malformation and mental retardation for the unborn child and there is a risk of radiation induced cancer which may be three times as high as for the average population (ICRP 2007) . However for diagnostic exposures the risk of deterministic effects appearing although not impossible is extremely unlikely. The potential benefit of the examination or treatment involving ionising radiation will in most cases be for the mother and only indirectly for the child, whereas it will incur a risk. This is different to the normal situation where one person namely the patient, incurs the risk but also derives the benefit of the examination or treatment (E.C. 100 1998). Pregnancy is also an important issue for staff working with ionising radiation. The moment the pregnancy is declared the employer is obliged to ensure the protection of the fœtus. This means

that the working conditions of the pregnant worker, after the declaration of pregnancy, should be such as to make it unlikely that the additional dose to the conceptus will exceed 1 mSv during the remainder of pregnancy (C.E.U. 1997). A number of ethical issues concerning irradiation of patients and staff during pregnancy were identified and these are presented in this paper. A review of practices concerning pregnancy and ionising radiation in various European countries was also carried out, the results of which are also presented.

Literature review in relation to irradiation of patients and staff during pregnancy

A literature review in relation to irradiation of patients and staff during pregnancy was carried out.

Council Directive 97/43 Euratom of 30 June 1997 on health protection of Individuals against the dangers of ionising radiation in relation to medical exposures. (C.E.U. 1997). The Medical Exposure Directive requires special attention for the protection of offspring of pregnant and breastfeeding patients exposed to ionising radiation for medical purposes. The prescriber of the exposure and the practitioner are obliged to ask the woman of childbearing age if she might be pregnant or missed a period. In the case where pregnancy cannot be excluded the practitioner and the radiographer shall treat the patient as if she were pregnant.

Radiation protection 100. Guidance for protection of unborn children and infants irradiated due to parental medical exposures (E.C. 100 1998). This guidance is addressed to prescribers, practitioners responsible for diagnosis or treatment, to nurses, to medical physicists and other professional staff who are in contact with the patient, e.g. such as midwives and gynaecologists. Furthermore the report is of interest to regulatory authorities.

Advice on Exposure to ionising radiation during pregnancy. Diagnostic Medical Exposures. Joint guidance from National Radiological Protection Board (NRPB), College of Radiographers, Royal College of Radiologists (NRPB 1998). The main objective of the NRPB advice concerning in utero exposures to ionising radiation is "to prevent unnecessary exposure of the foetus when medical diagnostic procedures involving ionising radiation are indicated during pregnancy". This is achieved by the introduction of the 10 day rule.

Medical and Dental Guidance Notes. Institute of Physics and Engineering in Medicine (IPEM 2002). A lot of information in these guidance notes has been published in various other references but it is helpful to have it brought together in one authoritative source covering the healthcare sector. The Guidance Notes is an excellent and thorough publication, which combines useful advice and a comprehensive list of relevant references.

Working safely with ionising radiation: Guidelines for expectant or breastfeeding mothers (HSE 1999). This leaflet provides advice for female employees who may be exposed to ionising radiation during their work. It is specifically aimed at those women who are thinking of having a baby or are already pregnant or breastfeeding.

ICRP publication 84. Pregnancy and medical radiation (ICRP 2000). This publication is concerned with the diagnosis of pregnancy, informed consent, and radiation effects on the embryo/fœtus, and the management of pregnant patients undergoing diagnostic radiology, nuclear medicine or radiotherapy. Occupational exposure of pregnant workers and research involving pregnant females is also covered.

Radiological protection of the unborn child. Recommendations of the commission on radiological protection and scientific grounds (Berichte 2006). This publication concerns the radiological protection of the unborn child in the case of occupationally exposed women who work with radionuclides.

The 10-day rule and the 28-day rule

The National Radiological Protection Board issued ASR8 (Exposure to ionising radiation of pregnant women: advice on the diagnostic exposure of women who are or may be, pregnant) in 1985. This advice suggested that there would be no risks to the conceptus following irradiation during the first 10 days of the menstrual cycle and that subsequent risks in the remainder of the first 4-week period would be likely to be so small that no special limitation on exposure was required – sometimes known as the "28-day rule" (NRPB 1998). Diagnostic exposure of women only during the first ten days after the beginning of the last period is what is called the ten day rule. All requests for radiological examinations of female patients, which place the uterus in or near the primary beam or nuclear medicine examinations which are likely to result in a dose to the unborn child up to 10 mGy, should include the date of the last menstrual period. The prescriber and practitioner or radiographer should ask a patient beyond day 10 of the menstrual cycle whether she might be pregnant. This enquiry and the patients answer should be recorded in writing. If the answer is no, the examination may proceed. If the answer is yes or uncertain, the examination should not proceed. In cases of medical emergency, the practitioner or the prescriber, if necessary following discussion with the practitioner or radiographer and taking justification into account may decide to proceed with the examination (R.S. 2000).

Review of existing practices in various European countries concerning the 10-day rule

Thirteen countries were reviewed by sending them a questionnaire. The U.K., Ireland, Cyprus, Bulgaria and Hungary apply the 10day rule for high dose procedures. In Italy and Austria it is the prescribing physician and the radiologist who ask whether the patient is or might be pregnant. In Greece and in Luxembourg it is the radiographers who ask whether the patient is or might be pregnant and in Spain the 10day rule is not applied. In Poland and Luxembourg a urine pregnancy test is carried out on women of childbearing age for interventional radiology (IR) procedures. In Poland the 10day rule is applied only for (IR) procedures. In Holland and in Luxembourg a sign is usually present in the radiology departments (in the changing cubicles) on the risk of pregnancy and radiation. In Slovenia the 10day rule used to exist in the previous radiation protection legislation but it wasn't applied. In the new legislation there is no such rule.

Review of existing practices in various European countries concerning patients who are pregnant and need a radiological examination

Thirteen countries were reviewed. In all the countries reviewed special attention to justification and optimisation issues is given when a pregnant patient needs a radiological examination. In Bulgaria, Italy, Austria and Poland X-ray examinations on pregnant women are not allowed unless it is an emergency and the life of the patient is in danger. In Hungary it is the prescriber who decides on whether the examination needs to be performed or not. In Italy, Greece and Cyprus a medical physicist is involved in order to calculate the dose to the foetus. Italy seems to be the only country where this matter is regulated by law

Review of existing practices in various European countries concerning staff and carers who are or may be pregnant and who are working in a radiology department

In Bulgaria, Slovenia, Hungary and Italy the worker is obliged to declare her pregnancy. In Luxembourg the worker is automatically placed in another department. In Hungary the worker must announce her pregnancy and after this she isn't allowed to work with ionising radiation anymore. She can continue working in the X-ray department but only to carry out administrative work. In Bulgaria, Slovenia, Italy, Poland, Greece, Spain, Cyprus, Holland and Ireland the worker can continue to work as long as the dose to the child to be born doesn't exceed 1mSv (C.E.U. 1996). In Cyprus, Spain, Italy and Slovenia the worker can choose not to work with radiation. In most countries pregnant carers are not allowed to work in X-ray Dpts.

Review of existing practices in various European countries in nuclear medicine concerning staff, patients and carers who are or may be pregnant

In all countries a nuclear medicine examination is carried out on a pregnant patient only if it is absolutely necessary and then it is optimised to give a dose as low as achievable. Breastfeeding patients are informed of the need to either interrupt or cease breastfeeding depending on the diagnostic or therapeutic procedure involved. In most countries the pregnant staff can continue working as long as there is no risk of contamination. In Luxembourg, Greece, Cyprus and Slovenia the pregnant staff is removed and placed in another department.

Identification of existing issues concerning patients

The International Commission on Radiological Protection in its publication 84 says the following: "Thousands of pregnant patients and radiation workers are exposed to ionising radiation each year. Lack of knowledge is responsible for great anxiety and probably unnecessary termination of many pregnancies. For many patients, the exposure is appropriate, while for others the exposure may be inappropriate, placing the unborn child at an unjustified increased risk." (ICRP 2000). One may well ask why this is the case. From this statement a number of issues arise. Notably the issue of education and information. It is important that the public be informed of the risks of ionising radiation during pregnancy. This can be achieved with simple information leaflets, articles in journals or through television programs. There seems to be a lack in general of information on the risks of pregnancy and ionising radiation for the public as well as the healthcare personnel.

In countries where large foreign communities exist it is important that the risks of pregnancy and ionising radiation be explained in the appropriate language. There is a risk that the female patient may not understand the enquiry on whether she is pregnant, or breastfeeding.

The 10 day rule should be applied if a diagnostic examination or treatment is planned which involves a high dose to the uterus or else a pregnancy test should be performed (E.C. 100 1998).Although the efficacy of a pregnancy test is another issue that needs to be looked at. Most pregnancy tests look for the presence of human chorionic gonadotropin (hCG) in the blood or urine. hCG can be detected in urine or blood after implantation, which occurs six to twelve days after fertilization. This results in false negatives if the test is performed during the very early stages of pregnancy (Wilcox et al. 1999). If a pregnant patient needs a radiological procedure in which the X-ray beam irradiates the foetus directly, and this cannot be delayed until after the pregnancy, it should be adapted so as to give as low a dose as possible to the foetus (ICRP 2000) . The principles of justification and optimisation are in this case of extreme importance. The procedure should be carried out by an experienced radiologist, cardiologist in case of a cardiac interventional procedure, and a medical physicist should be involved.

There is limited information on research and pregnancy in the council directive 97/43 (C.E.U. 1997) however the involvement of pregnant females in research should be discouraged. Pregnant women should not be asked to take part in any research projects involving irradiation to their unborn child, unless the pregnancy itself is essential to the research or therapeutic research which may be life saving for the mother (E.C. 99 1998). Another important ethical issue is that the pregnant patient has the right to know the magnitude and type of potential radiation effects that might result from in-utero exposure (ICRP 2000) .

Identification of existing issues concerning staff

Training of the personnel on the subject of pregnancy and radiation protection is very important. Lack of training leads to the problem of radiophobia. It is important that radiation protection issues be properly explained to pregnant staff so that she can make an informed choice about what she should undertake. Female personnel should have the right to decide if they want to continue to work in a radiology department and not be automatically placed in another department. Another ethical issue that could be looked at is whether it is appropriate to apply the same dose limit to the fœtus as a member of the public, given that the fœtus is generally considered to have a higher radiosensitivity (Faulkner 2005).

Discussion

From the literature it can be seen that most diagnostic procedures which are performed correctly present no measurable increased risk of prenatal or postnatal death, developmental damage including malformation, or impairment of mental development over the backround incidence of these entities. High dose procedures such as abdominal and pelvic computed tomography that deliver doses to the foetus of some tens of mGy may carry significant risks with a doubling of the natural cancer risk in the unborn child. However this is unlikely to be a reason for the termination of the pregnancy. Absorbed doses below 100 mGy should not be considered a reason for terminating a pregnancy. Most nuclear medicine procedures do not result in high doses to the embryo and foetus. However, some radiopharmaceuticals that are used in nuclear medicine can pose significant risks to the embryo and foetus (NRPB 1998, ICRP 2000). In reference to the above mentioned introduction of the ICRP publication 84 it would seem that education, training and information for the healthcare personnel and the public is of great importance and that a number of the above mentioned ethical issues could be solved this way.

From the review of practices it has been seen that there no harmonisation at a European level and this is something that still needs to be looked at.

Conclusions

The 10day rule is not applied in all the countries reviewed. The manner in which this issue is treated differs from country to country. The person who determines whether a woman of childbearing age is or might be pregnant differs from country to country. In only two countries a urine pregnancy test is carried out regularly on women of childbearing age who have to undergo IR procedures.

Special attention is given to optimisation and justification of the examination in all countries. Who decides whether the examination should be done i.e. the radiologist or the prescriber differs from country to country. A medical physicist is involved in this issue in only a small number of countries.

The existing practices in various European countries concerning staff and carers who are or may be pregnant and who are working in a radiology department vary enormously. There is no harmonisation on this issue at a European level

Special attention is given to optimisation and justification of the examination of a pregnant patient in Nuclear Medicine. A medical physicist is involved in this issue in only a small number of countries. The existing practices in various European countries concerning staff who are or may be pregnant and who are working in a Nuclear Medicine department vary enormously. There is no harmonisation on this issue at a European level.

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Quality development and dose optimisation in native pediatric radiography in a maternity hospital

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Abstract

Kätilöopisto is the biggest maternity hospital in Finland. In addition to radiography of newborns and mothers, also the native X-ray examinations of children under 16 years in Helsinki area were earlier centrally taken care by Kätilöopisto hospital. When X-ray imaging was thus increasing, the need for systematic estimating and developing the technology used in the examinations, measuring radiation doses, assessing the clinical image quality and own working methods became current. The hospital started a quality improvement project in 1999 in cooperation with Radiation and Nuclear Safety Authority (STUK). The radiologists and radiographers of the Department of Radiology have participated in the developing work. The aim is to produce an optimal radiodiagnosis with as low X-ray dose as possible following the common good practice.

The methods and results of the development project, changes to radiology practices and factors affecting these changes from the past 10 years are presented. The project has also involved monitoring radiation doses and assessing the clinical image quality in children's X-ray examinations and pelvimetric examinations. Special attention has been paid to radiation doses of newborns and the quality of examinations. Among others, in certain examinations, the radiation dose has been reduced to one third of the earlier dose. The project has proved an excellent starting point for developing good practices. On the basis of the results of the study, objective reference levels of the examinations of the department have been set on picture quality and on radiation dose. The quality development project is still continuing.

Comparison of results of quality control tests of the mammography screening in Mazovia province in 2007 and 2008

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Abstract

In 2007 and in 2008, quality control tests were performed respectively in 48 and 46 facilities performing mammography screening. For this purpose a comprehensive system and methodology of testing was established.

On the basis of existing legal regulations in Poland, it was decided that as a part of quality control testing the following items should be checked at a facility: equipment, organization of the examinations, correctness of the basic and specialist tests (only in 2007). Additionally a group of tests, so called supervision tests, was listed. One part of the test dealt with physical and technical parameters essential for the quality of mammography images, the other part was used to evaluate the doses absorbed by the examined women. The tests were carried out by the physicists from Mazovia Regional Coordinating Centre. These physicists had to prepare a report according to standard protocol, providing the evaluation of particular tests.

A controlled facility could get maximum score of 84 points in 2007 and 365 points in 2008. An evaluation of the examinations in a facility was rated according to a percentage of scored points: at least 80% – high level, from 79% to 60% – medium acceptable level, below 60% – low non-acceptable level.

In 2007, according to these criteria only 8% of facilities were scored at high level quality of screening examinations, 35% were scored at medium acceptable level, and 56% at non-acceptable level. In 2008, only 7% of facilities were scored at high level quality of screening examinations, 59% were scored at medium acceptable level, and 35% at non-acceptable level.

Obtained results indicate a slight improvement of the quality of mammography screening examinations. However, the quality is still not acceptable and sholud be further improved in order to be able to contribute to the decrease of breast cancer mortality which is the prime aim of the screening.

Evaluation of dose for routine exposure in mammography screening in Mazovia province in 2007 and 2008

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Abstract

There is a significant risk of radiation-induced carcinogenesis associated with x-ray mammography. Therefore determinaton of average glandular dose is an important part of quality control of mammographic imaging systems. This study presents estimated doses for routine exposure using 45 mm thick PMMA blocks at mammography facilities in Mazovia Province in 2007 and in 2008.

Material for this study constituted data from most of mammography screening facilities in Mazovia Province in 2007 and 2008. The method consists of two parts, i.e, the determination of tube voltage (kV), focal spot charge (mAs), type of anode and additional filter for routine exposure of standard breast. The second part was measurement of entrance surface air kerma (wihout backscatter) at these parameters. Afterwards, the entrance surface air kerma was calculated at the upper surface of the PMMA. Furthermore the half value layer (HVL) was calculated for tube voltage used during routine examinations. The average glandular dose was calculated according to Dance. Estimated average glandular dose was compared with achievable value for 45 mm thick phantom (i.e 2 mGy) from "European Guidelines for Quality Assurance in Breast Cancer Screening and Diagnosis".

In 2007, established average glandular doses at all mammography facilities ranged from 1,19 mGy to 4,14 mGy. In 2008, established average glandular doses in all mammography facilities ranged from 1,07 mGy to 3,95 mGy. The average glandular dose exceeded limiting value at 52% mammography facilities in 2007 and at 28% mammography facilities in 2008.

Obtained results of average glandular doses for 45 mm thick phantom reached alarming high values and they depended on wrong setting of technical parameters of mammography equipment.

Doses received by women in screening mammography examinations in Poland in 2007

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Abstract

Purpose: The aim of this study was the estimation of the radiation risk for women undergoing screening mammography examinations. This is the first study of this kind in Poland concerning radiation protection of patients at the national level.

Methods and material: Materials for this study are the data from 250 screening mammography facilities in Poland. The following parameters were collected: the breast thickness after compression, the high voltage, the mAs values, material of the anode, additional filters used during the exposition. Data from 44992 expositions were collected. For every mammography facility standard average glandular doses for routine exposures were calculated. Furthermore, average glandular doses for individual mammography examinations, according to the methods proposed by Dance, were calculated. Tolerances were taken from the "European guidelines for quality assurance in breast cancer screening and diagnosis".

Results: Established average glandular doses in 250 mammography facilities in Poland in 2007 range from 0.12 mGy to 14.56 mGy with an average of 1.99 mGy. At only 32 mammography facilities all expositions did not exceed acceptable levels of the average glandular dose. At only 18 mammography facilities all expositions did not exceed achievable levels of the average glandular dose.

Conclusions: Average glandular doses obtained for women undergoing mammography screening examinations reach high values and they depend on technical parameters of mammography equipment. In order to reduce the radiation risk for women undergoing screening mammography examinations, the facilities which do not conform to the standards recommended by the European Commission should improve the quality of their examinations or be closed down.

Implementation of quality assurance/quality control programme in mammography practice at the University Hospital of Osijek

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Abstract
The radiologist's objective is to obtain accurate diagnostic information from mammography while keeping radiation dose to the breast acceptable. Image quality and patient doses in mammography depend on large number of factors such as film speed and contrast, film processing, tube voltage, breast size and composition, and automatic exposure control. Also, image viewing conditions can considerably influence overall diagnostic accuracy of mammography, especially its sensitivity. Due to number of steps involved in this process, the potential for mistakes increases. Reproducible high-quality images with reasonably low radiation doses are, in general, assured by comprehensive quality assurance / quality control (QA/QC) program and this is demanded by the Croatian radiation protection law.

The national audit of mammography equipment performance, image quality and dose in Croatia was done recently. Results showed large variations in image quality and patient doses and revealed the need to continuously monitor and optimize mammography techniques. Survey pointed out that main problem in the Croatia is the lack of written QA/QC procedures. Consequently, equipment performance, image quality and dose are unstable and activities to improve image quality or to reduce the dose are not evidence based.

At the University hospital of Osijek mammography QA/QC programme, based on EC (European Commission) guidelines, was implemented during the year of 2007. After we did the corrective actions based on measured data and image quality assessment, technical parameters are now within acceptable levels given by EC guidelines. Image quality is improved significantly and rate of rejected images is much less than before implementation of programme.

Introduction

The radiologist's objective is to obtain accurate diagnostic information from mammography while keeping radiation dose to the breast acceptable (1, 2). Reproducible high-quality images with reasonably low radiation doses are, in general, assured by comprehensive quality assurance / quality control (QA/QC) program (2, 3) and this is demanded by the Croatian radiation protection law.

The national audit of mammography equipment performance, image quality and dose in Croatia was done recently (4). Results showed large variations in image quality and patient doses and revealed the need to continuously monitor and optimize mammography techniques.

At the University hospital of Osijek (UHO) mammography QA/QC programme, based on European Commission (EC) guidelines (2), was implemented during the year of 2007. In this work the impact of its implementation will be presented as comparison of the rejection rate and the image quality before and after QC implementation.

Material and methods

As a first step, the film reject rate and the image quality assessment within two weeks were done at 4 mammography facilities (A, B, C and D) in eastern part of Croatia in order to compare the obtained data with the data obtained after implementing QA/QC programme based on EC guidelines (2).

The image quality assessment was done according EC image quality criteria (5). The patients whose images were assessed were randomly picked at facility A, 20 of them before QC and 22 after QC. In facilities B, C and D the number of randomly picked images was 20, 20 and 19 respectively. The quality of images was assessed and scored by on site radiologists. At all facilities same radiologist performed the image quality assessment before and after implementation of QC. Four images (cranio-caudal (CC) right breast, CC left breast, medio-lateral-oblique (MLO) right breast, MLO left breast) were evaluated individually for each patient using the questionnaire according to EC quality criteria (5).

Causes of the film rejections fit into seven different categories: too light, too dark, mechanical problem, artefact, positioning problem, patient motion and administrative mistakes. Afterwards, the plan was to make necessary corrections and repeat the survey.

Only minor corrections were made in three departments and image quality was slightly improved. The most common corrective actions were education of technologists, cleaning screens and developing unit maintenance. More complicated and/or more expensive corrections couldn't be done. The main reasons were the lack of financial resources, inadequate staff training and the lack of willingness to cooperate. For this reason, QA/QC programme based on European guidelines [2] was installed only at the University hospital of Osijek (UHO) where these resources already existed.

Siemens Mammomat 3000 Nova, dedicated developing unit (Curix 60, AGFA) and old non-dedicated image viewing box has been used for mammography in UHO at the time of installation. To measure the performance level of mammography practice in UHO before implementing QA/QC programme, following parameters were tested: X-ray tube voltage accuracy and reproducibility, X-ray output, dose to standard breast, half-value layer (HVL), viewing conditions, dark chamber, automatic exposure control

(AEC) performance and film-processing data. Some parameters are changed to comply with typical values of EC guidelines (2).

Afterwards, QC with frequencies of the tests, typical values and tolerances are implemented according to EC guidelines (2). Constant analyses of processing unit as well as constant checks of optical density at the centre of phantom are done on daily basis and other tests are done weekly or annually. Image quality assessment and rejected image analysis is performed once a year, again within two weeks. The intention of quantitative analysis of film rejection causes is to detect the most common problems. The images rejection rates obtained at the UHO as well as the rate of causes of rejection before and after QC implementation were compared and analysed. The rate of image rejection and image quality assessment for other facilities was analysed and presents the situation encountered before and after corrective actions.

Results

Table 1 shows the characteristic parameters of mammography before and after implementing the QA/ QC programme and comparison with tolerances and typical values (2).

Table 1. Characteristic parameters of mammography before and after implementing the QA/ QC programme and comparison with tolerances and typical values.

	Before QC	After QC	Tolerance	Tipical values
D_{min}	0.22	0.23	0.15 - 0.25	
D_{max}	3.74	3.90	-	~ 4.00
M_{Grad}	3.60	3.40	-	3.0 - 4.0
$Grad_{1,2}$	3.36	4.40	-	3.5 - 5.0
Speed	1.83	1.70	-	-
OD	1.16	1.65	1.5 – 1.9	
AGD (mGy)	0.8	1.1	< 2	
Resolution (lp/mm)	14.5	14.5	>12	
Large detail contrast (%)	2	1	<1.5	

Table 1 show that the ODs of the images were lower than recommended (2). Lighter images were used because of luminance of viewing box was 1200 cd/m^2. Mammography dedicated viewing box was acquired and OD increased.

Image developing parameters before and after implementing QA/QC programme are shown in Figures 1, 2, 3, 4.

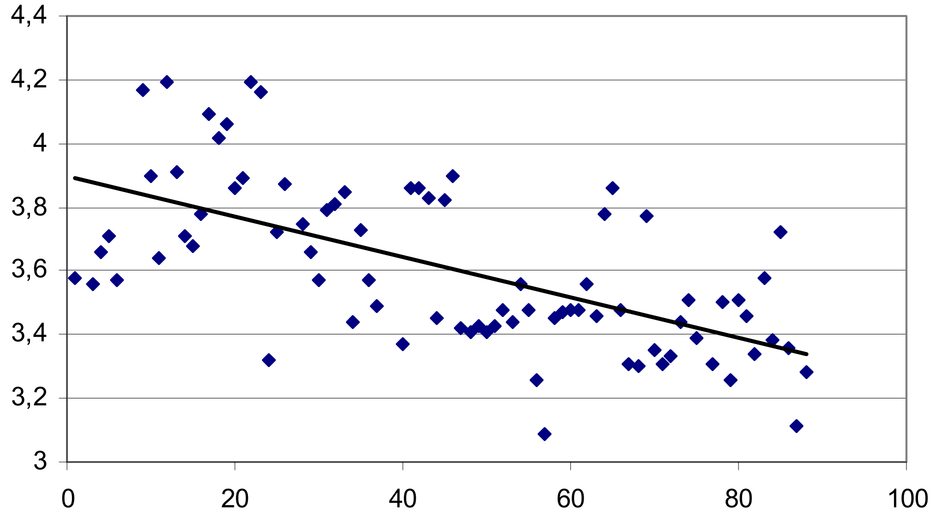

Fig. 1. M_{Grad} values before replacing developing unit. On x axis are days.

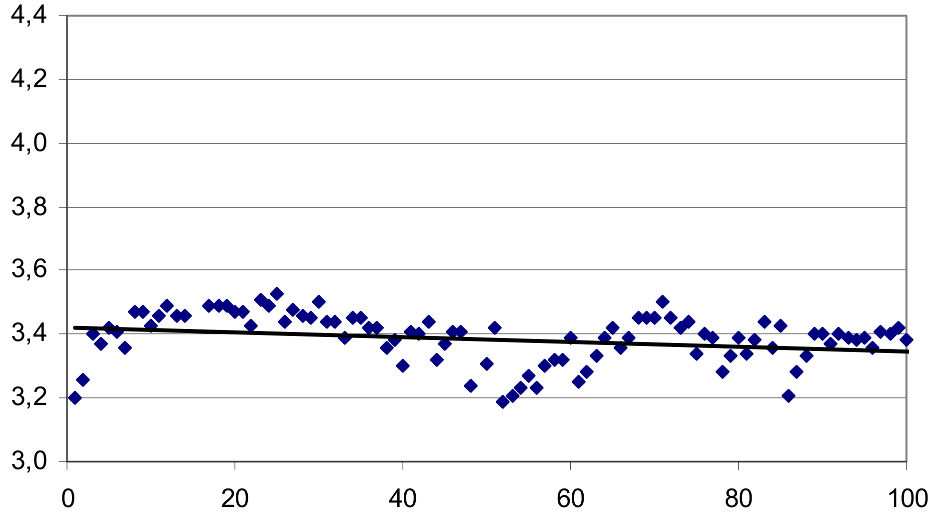

Fig. 2. M_{Grad} values after replacing developing unit. On x axis are days.

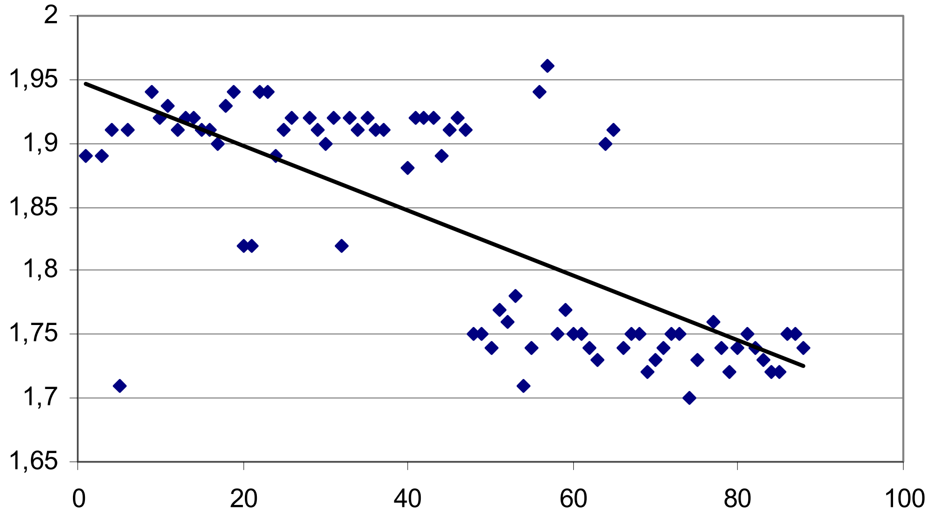

Fig. 3. Speed values before replacing developing unit. On x axis are days. On day 50 the ventilation is installed.

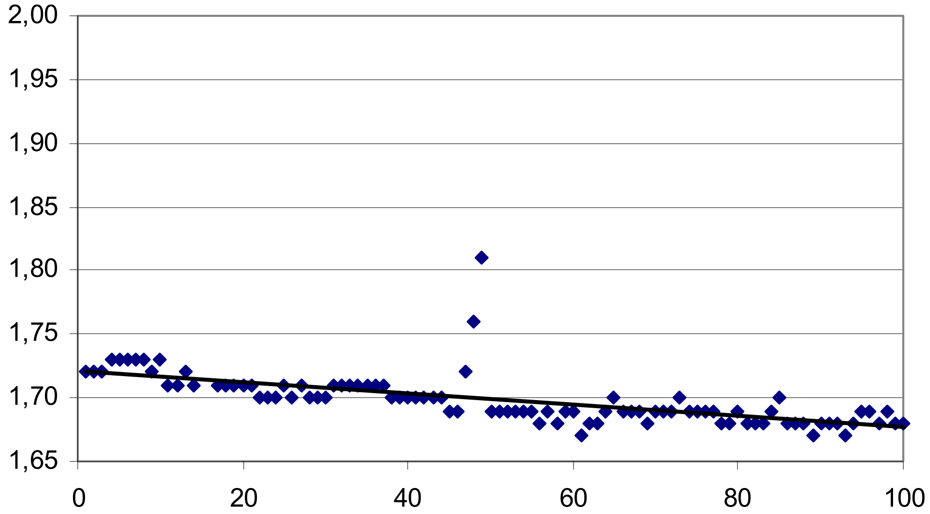

Fig. 4. Speed values after replacing developing unit. On x axis are days.

Image developing parameters at the beginning varied in time because of non-adequate developing unit (the unit is for low-volume processing or backup).

Figure 5 shows image rate rejection before and after corrective actions (in UHO after QA/QC programme implemented) and Figure 6 shows causes of image repeating.

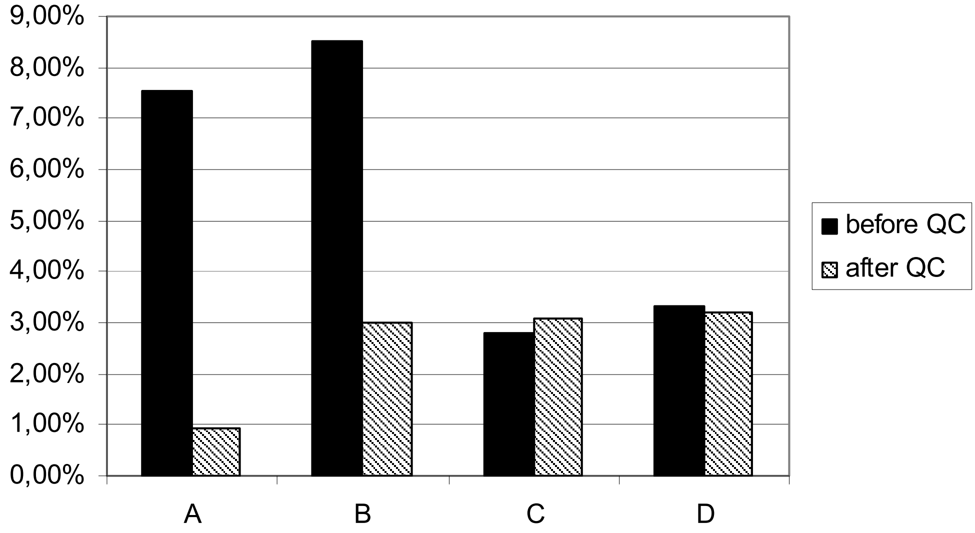

Fig. 5. Image rate rejection before and after corrective actions (in UHO after QA/QC programme implemented).

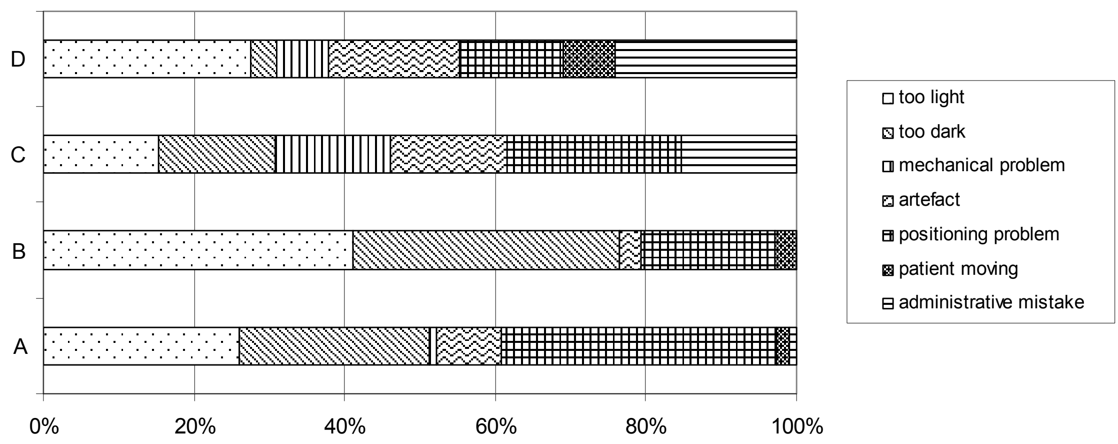

Fig. 6. Causes of image repeating in all four facilities.

The main reason of image repeat at dpt B was too dark or too light image. The reason was non functional AEC. Maintenance was called and AEC enabled.

Figures 7 and 8 show the EC criterion fulfilment, before and after implementation of QA/QC programme or corrective actions, in CC and MLO projections, respectively for all dpts.

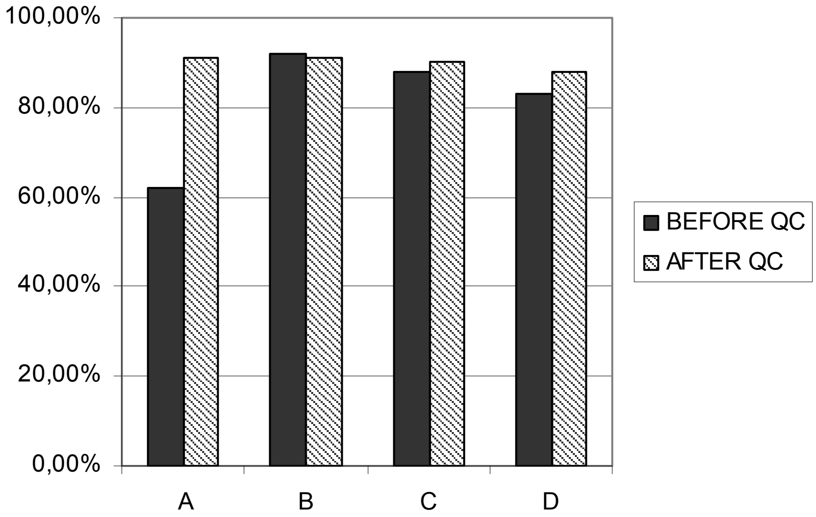

Fig. 7. EC criterion fulfilment, before and after implementation of QA/QC programme or corrective actions, in CC projection for all surveyed facilities.

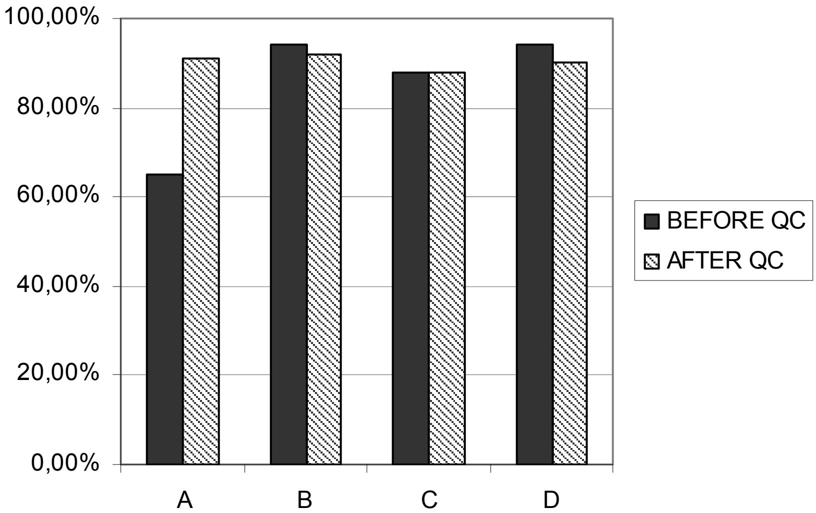

Fig. 8. EC criterion fulfilment, before and after implementation of QA/QC programme or corrective actions, in MLO projection for all surveyed facilities.

Discussion

The mammography is, in a technical sense, one of the most demanding radiographic procedures, image quality and the patient dose critically depend on equipment and its performance, as well as on radiographers' skill. We showed that even simple corrective actions as cleaning of screens, education or somewhat complicated as AEC enabling can considerably improve mammography image quality (Figures 6, 7, 8). It was surprising that rate rejection was low and fulfilment of EC criterions high in C and D dpts. As discussed before (6), the reason can be in high tolerances of radiologists and not high image quality.

In UHO comprehensive QA/QC programme was implemented and the data analyzed. According to these analyses, new equipment has been acquired (processing unit and viewing box) and working modality (higher OD) has changed. Though higher OD means higher patient dose (Table 1) it improves diagnostic information (3, 7).

Conclusions

There are many proofs of improvements in image quality and patient dose reduction in mammography by implementation of simple and low-cost measures (6, 8). IAEA highlighted the need to evaluate the situation in different countries, to identify the points where corrective actions are needed and to document the improvements in order to achieve an effective optimization of mammography practice (8).

In Croatia, QA in diagnostic radiology is a regulatory requirement. Furthermore, national committee for QA in mammography has been established by the National Radiology Association, but there is still no detailed framework for QA implementation, only pilot projects in a few University hospitals where physicists are available.

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Cooperation of the Nordic radiation protection authorities in the field of X-ray diagnostics

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Abstract

The Nordic countries (Denmark, Finland, Iceland, Norway and Sweden) have many similarities as for the status of using radiation in health care and in the regulatory control of radiation practices. Therefore, the importance of co-operation between the Nordic radiation protection authorities has long been recognized and manifests through Working Groups (WG) established for several fields of interest. The Nordic WG for X-ray diagnostics has been operational for over 30 years. The main tasks are to exchange information on the national activities in the field of X-ray diagnostics, to discuss problems for regulatory issues and to undertake joint efforts of research. The results are published as Nordic reports or scientific papers in international journals. In a recent joint project, patient doses in paediatric CT examinations were studied in order to pay attention to the increased need for careful optimization. Comparison of the characteristics of X-ray diagnostics in the Nordic countries indicate that the level of professional resources is somewhat varying. For example, the number of radiologists per number of X-ray examinations is rather constant (about 0,15 radiologists/1000 examination) but the number of physicists in diagnostic radiology varies from about 0 to 13 physicists/million examinations. The annual joint meetings, joint research and other joint activities have ensured effective exchange of information for developing the national regulatory activities and for promoting consistent methods of quality assurance and dosimetry. This paper compares some characteristic features of the use of radiation for X-ray diagnostics in the Nordic countries and the activities of the radiation protection authorities in this field, in particular for the modern X-ray diagnostic practices (multi-slice CT etc) highlighting the importance and achievements of effective co-operation between the authorities.

Introduction

The Nordic countries (Denmark, Finland, Iceland, Norway and Sweden) have many similarities as for the status of using radiation in health care and in the regulatory control of radiation practices. Therefore, the importance of co-operation between the Nordic radiation protection authorities has long been recognized and very practical co-operation has existed for decades. The practical co-operation manifests through Working Groups (WG) established for several fields of interest. The WGs have sometimes been established for specific tasks, but usually they have mandates for working more permanently on a specific topic area.

One of the permanent Nordic WGs deals with X-ray diagnostics and has been operational for over 30 years. The main tasks are to exchange information on the national activities and achievements in the field of x-ray diagnostics, to discuss problems and development for regulatory issues and to co-ordinate or undertake joint efforts of research. The results of the work are published in the series of the Report on Nordic radiation protection co-operation or as scientific papers in international journals. In a recent joint project [1], patient doses in paediatric CT examinations were studied in order to pay attention to the increased need for careful implementation of the optimization principle.

This paper compares some characteristic features of the use of radiation for X-ray diagnostics in the Nordic countries and the activities of the radiation protection authorities in this field, highlighting the importance and achievements of effective co-operation between the authorities.

Health care structure and characteristics of radiological practices

All Nordic countries have a good public health care system while there are also private hospitals and practitioners. In Sweden and Norway, the health care system is organized by county or Health Trust, which is a legal entity and has the licence for radiological procedures. The radiation protection legislations are rather up-to-date while the need of some improvements has been identified in some cases, e.g. modification in the supervision of screening programs (Finland), non-ionizing radiation (Sweden; covered only partly by the present regulations; Denmark; no legislation exist covering the medical field), Diagnostic Reference Levels not established (Iceland), training and education in radiation protection (Norway; define adequate level of competence and skills), medical physicists in hospitals (Norway; quantification) and changing the legislation to more general terms (Denmark; to better match modern equipment).

Some characteristics of the medical use of radiation in the Nordic countries are compared in Table 1. While there are no significant differences between the countries, the level of professional resources is somewhat varying. The number of radiologists per 1000 examinations is almost the same, 0,13-0,15, while there are higher variation in the number of physicists and radiographers. This may partly be associated with different responsibilities of these professionals, but generally the number of physicists seems very low.

Activities of the authorities

All authorities take care of the licensing procedure for radiological practices and perform regulatory inspections to ensure the compliance of practices with the legal requirements. A challenge within licensing issues is the rapid development in modern X-ray diagnostic practices used outside radiological departments, like 3D mobile C-arms for use in surgery and Cone Beam CT for use in dentistry. Further, the development in MSCT raises the question about justification of screening of asymptomatic patients outside national screening programs. The inspection policy and methods vary somewhat between the Nordic countries and is described in more detail below (see section Inspection Workshop).

Besides the actual regulatory activities, all Nordic authorities undertake training efforts, prepare and issue guidance on radiation protection and radiation measurements, maintain the national measurement standards for radiation metrology and, to varying extent, undertake research on radiation metrology, radiation protection and quality assurance techniques.

Training of the users of radiation has been considered as one of the key activities of radiation protection authorities, in order to promote the safe use of radiation. Therefore, the actions for training seem very intensive as can be seen from the examples in Table 2.

All authorities also undertake research activities, some of them being joint Nordic efforts, resulting in a number of scientific publications annually. Some recent research topics have been devoted to:

- Determination of patient doses in paediatric examinations (Finland)
- Patient doses in diagnostic radiology (Iceland, Sweden)
- Pilot study: Occupational personal doses in interventional radiology (Denmark)
- Testing of the IAEA Code of practice for x-ray diagnostic dosimetry (Finland)
- Dentists and veterinarian surveys (Norway)

All authorities also have relatively extensive participation in national and international cooperation. Most of the authorities participate in working groups or other activities of, e.g., HERCA (Cooperation between European radiation protection authorities), IEC (International Electrotechnical Commission, standardization activities), EMAN (European Medical Alara Network) and EURAMET (European Metrology cooperation).

Table 1. Comparison of some statistical information between the Nordic countries, on the use of radiation in health care.

	Denmark	Finland	Iceland	Norway	Sweden	Average
Number of inhabitants, million	5,52	5,26	0,32	4,81	9,26	
Number of x-ray examinations/year, million	3,00	4,14	0,26	3,90	5,20	
Number of exams/population	0,54	0,79	0,81	0,81	0,56	0,70
Number of CT units	110	70	10	130	170	
CT units/million population	19,9	13,3	31,6	27,0	18,4	22,0
Number of radiologists	430	550	35	600	1000	
Radiologists/1000 examinations	0,14	0,13	0,14	0,15	0,19	0,15
Number of medical physicist for X-ray diagnostics	25	15	0	30	65	
Physicists/million examinations	8,3	3,6	0,0	7,7	12,5	6,4
Number of radiographers for x-ray diagnostis	1400	2200	200	2700	3000	
Radiographers/1000 examinations	0,47	0,53	0,79	0,69	0,58	0,61

Table 2. Training activities of the Nordic radiation protection authorities in the field of diagnostic radiology.

Training events for...	Training in 2008				
	Denmark	Finland	Iceland	Norway	Sweden
Medical physicists		2 days		1 day	
Radiologists					4 days
Radiographers	7 days	2 days	1 day		
Radiation protection officers or equivalent	2 days	1 day	1 day	1 day	
Other	12 days	1 day			4 days
Lecturing in university courses	no	no	yes	yes	no
Lectures in different national meetings etc	yes	yes	yes	yes	yes

Inspection Workshops

A part of the co-operation between the Nordic radiation protection authorities has been a regular "Inspection Workshop", which has now been organized for three times (every second year). The last one was organized in Norway in 2009 with 20 participants. The Workshops has not only dealt with X-ray diagnostics but covered all medical disciplines, i.e. X-ray diagnostics, nuclear medicine and radiotherapy. In the Workshops, the status of inspections in the medical field in the various countries has been reviewed, followed by working group discussions, comparisons and analysis of the inspection methodology and results. Common findings from the inspections have been recorded. Some specific topics have also been addressed, such as implementation of the justification principle or level of skills for radiation protection in cardiology.

The inspection frequency varies between the countries, but the type of inspection and the inspection procedure are fairly similar, although Denmark and Finland seem to have more technical/measurement elements in the inspection. Denmark is currently revising the inspection procedures towards fewer measurement elements. Comparison of the number of inspections is not straightforward because some countries count inspections for each piece of equipment and others for each county or health trust. A typical finding from inspections, common for all countries, is the lack of competence among the radiation users, especially outside Radiological departments.

Summaries of the observations from the latest Inspection Workshop are shown in Tables 3 and 4.

Table 3. Characteristics of regulatory inspections for X-ray diagnostics in the Nordic countries.

Country	Frequency, y	Fee	No of inspectors/ inspection	Inspector training
Finland	3-5, mammo-screening 2	Inspection fee per equipment	1-2	On-job training. Hospital background
Sweden	5-10; in future 5	Annual fee	2	Special course. Hospital background
Norway	5-10	No fee	2	On-job training. Special course. Supervision by seniors
Iceland	2-4	Inspection fee per equipment + lisencing fee	1-2	On-job training. Internal training program with certification
Denmark	5	Fee per equipment		On-job training. Internal courses

Table 4. Organization of regulatory inspections for X-ray diagnostics in the Nordic countries.

Country	Choice of object	Procedure	Focusing	Follow-up
Finland	By risk based pre-defined periodical system	Specialized inspectors in diffrent areas	No. Focused topics in Clinical audits.	Recorded in a register with a deadline.
Sweden	By order and risk analysis	Opening meeting. Inspection. Interviews. Closing meeting	Both general and focused inspections.	Report on corrective actions required. Special re-inspection.
Norway	By knowledge and prioritized areas	Request for documentation. Opening meeting. Inspection. Interviews. Closing meeting.	Competence outside radiology department	Report after inspection with schedule for corrective actions required.
Iceland	By activity and risk analysis, also by findings in provious inspections.	Both technical and QA reviews. Closing meeting.	Patient doses and QA procedures	Report on corrective actions required in 1-6 months, less important on next inspection. Followed up.
Denmark	Reported problems, high doses, new types of equipment No fixed intervals	QA systemm revies for hspitals and larger clinics	QC and shielding	Discussion at the time of inspection; report on corrective actions required

Conclusions

The annual joint meetings, joint research projects and other joint activities of the Nordic WG on X-ray diagnostics have ensured effective exchange of information in the field of diagnostic radiology. This has been of high importance for developing the national regulatory activities and for promoting consistent methods of quality assurance and dosimetry.

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Diagnostic reference levels for diagnostic x-ray examinations in the Baltic and Nordic countries

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Abstract

Diagnostic reference levels (DRLs) are important tools for optimisation in diagnostic radiology. The European directive on medical exposures [1] requires member states to promote the establishment and use of DRLs, and thus Denmark, Finland, Sweden and all of the Baltic countries are required to establish DRLs. Individual patient doses should not be compared with the DRL, but average patient doses for a given examination in a hospital are to be compared with the DRL.

Common Nordic DRLs were recommended in 1996 for 6 conventional examinations [8]. These values are still in use in Iceland, while – based on national surveys etc. – Denmark, Finland and Sweden have set and revised national DRLs, and Norway have just set national DRLs. Additional DRLs have gradually also been set for CT and mammography as well as for cardiologic and paediatric examinations. DRLs were first set in Lithuania 5 years ago, and have been modified once, while DRLs are currently being set in Estonia.

The Nordic Working Group on X-ray diagnostics has in cooperation with representatives from Estonia and Lithuania conducted a study of DRLs. The choice of dose quantities measured as well as current and previous values of DRLs are compared among the Nordic and Baltic countries, and the process of revision in the different countries is described.

The trend for radiographic and fluoroscopic examinations is that the dose levels and thus the DRL values have decreased over the years. For CT, the dose levels are more constant.

Introduction

Diagnostic reference levels (DRLs) are important tools for optimisation in diagnostic radiology. The concept of DRLs was introduced by the 1990 recommendations of ICRP [4]. The European directive on medical exposures [3] requires member states to promote the establishment and use of DRLs, and thus Denmark, Finland, Sweden and all of the Baltic countries are required to establish DRLs. Individual patient doses should not be compared with the DRL, but average patient doses for a given examination in a hospital are to be compared with the DRL.

DRLs were introduced in the Nordic countries through a common set of recommended values for three conventional and three fluoroscopic examinations, published by the Nordic Working Group on X-ray diagnostics (collaboration between the radiation protection authorities in the five Nordic countries) [8].

DRLs are set based on surveys of the patient doses used for a given type of examinations of a group of patients of standard size. For setting of national DRLs, a national survey of patient doses is thus necessary. In most cases, the DRL is set as the third quartile of the patient doses found in the dose survey. DRLs are conventionally set in dose quantities that are relatively readily available to the user of the x-ray equipment. For conventional examinations, the main quantity is the total DAP for the examination while in some cases the entrance surface dose (ESD) for different projections is used. For fluoroscopy, the main quantities are the total DAP for the examination and the fluoroscopy time. For CT, DRLs can be set for the CTDI as well as for the total DLP for the examination.

Material and methods

Based on collaboration between the Nordic Working Group on X-ray diagnostics and representative from the radiation protection authorities in Estonia and Lithuania, previous and current national DRLs from the 7 countries have been collected. The DRLs have been taken from the national guidelines and from results of dose surveys carried out by the authorities. Estonia does not have national DRL values yet, but recommended DRL values have been published [5, 7] based on limited surveys.

DRLs can be set both nationally and locally, in this document only national DRLs will be considered.

Results

The number and type of examinations for which diagnostic reference levels have been set and the values of the DRLs have been compared between the seven countries in the study.

Number of diagnostic reference levels and types of examinations

Table 1 shows the number of currently valid national DRLs in the different countries. The main number shown is the number of different examinations for which DRLs have been set. In some countries, DRLs for several quantities (e.g. total DAP and fluoroscopy time) have been set for the same type of examination, and the number in parentheses is the total number of DRLs set. Estonia has no national DRLs yet, and has not been included in the table.

Table 1. Number of examinations for which DRLs have been set. In cases where DRLs have been set for more than one quantity for each type of examination, the number in parentheses shows the total number of DRLs. The last row shows the average number of examinations of adults for which DRLs have been set and the average number of DRL values.

Country	Adult patients					Paediatric patient
	Conv.	Fluorosc.	Cardiac	CT	Mammogr.	
Denmark (DK)	4	1		9	1	4 (6)
Finland (FI)	5§ (12)	1	2 (4)	4 (9)	1	3 (8)
Iceland (IS)*	4 (8)	2				
Lithuania (LT)	7 (22)	3 (6)		5	1	
Norway (NO)	8	1	1	10 (20)	2	
Sweden (SE)	4	1	1	4 (8)	2 (4)	
Average	14 (22)					

§ Additionally, Finland has DRLs for two types of dental examinations
* Identical to the common Nordic DRLs from 1996

All countries have DRLs for conventional examinations of thorax, pelvis and lumbar spine, while all but Lithuania have DRLs for IV urography. Finland, Lithuania and Norway also have DRLs for various other examinations, e.g. abdomen, skull, thoracic spine and hip joint. Only Finland and Lithuania have DRLs for several quantities for the same examination, in both cases for entrance surface dose (ESD) for different projections as well as for the total DAP for the examination, while the other countries have DRLs only in terms of DAP.

The only fluoroscopic examinations, for which all countries have set DRLs is barium enema. Lithuania in turn has DRLs for chest and for barium swallow and Iceland for barium meal. All countries have DRL values for the total DAP for the examination, while Finland and Lithuania also have values for the fluoroscopy time.

Only Finland, Norway and Sweden have DRLs for cardiac examinations. All three have values for coronary angiography, and Finland has a value for PTCA also. All three countries have DRLs for total DAP, while Finland in addition has DRLs for fluoroscopy time.

All countries except Iceland have DRLs for CT examinations of head, thorax and abdomen and for different parts of the spine. In addition, Lithuania and especially Denmark and Norway have DRLs for a range of specialised CT examinations in the pelvic and abdominal region. Finland, Norway and Sweden have DRLs for the total DLP of the examination as well as for the CTDI value, while Denmark and Lithuania uses only the total DLP.

Denmark, Finland and Lithuania have DRLs for clinical mammography, and Norway and Sweden have DRLs for both clinical and screening mammography. Sweden has DRLs for each exposure as well for the full examination, while the other countries have only one value (either for one exposure or for the full examination).

Most DRLs are set for adult patients of standard size, as is also the case for all of the values mentioned above. Due to the large variations in size, it is more difficult to set DRLs for paediatric patients. Some examinations are carried out primarily on special age groups (or an age group can be selected as reference), and DRLs can be set for these special age groups. Denmark and Finland have set DRLs for a few paediatric examinations of special age groups: abdomen for infants up to 12 month (DK), sinuses for children 7 – 15 years (FI) and Mictiocystography (MCU) for infants up to 12 month and for children 1 – 5 years (DK and FI). For other examinations patients of all ages are included, and one approach here is to set the DRL as a function of the size of the patient. Denmark and Finland have used this approach to set DRLs for thorax examinations (DK and FI) and for pelvis examinations (DK). In Denmark, the equivalent diameter (based on the height and weight of the child) [2] has been used to describe the size of the patient, while in Finland the actual thickness of the patient has been used. For MCU, both countries have set the DRL in terms of DAP. For the remaining examinations, Denmark has used ESD, while Finland has used both ESD and DAP.

In a previous project in the Nordic Working Group on X-ray diagnostics [1], the aim was to set DRLs for paediatric CT examinations also, but it turned out that the number of examinations carried out was too small to give the required statistical basis.

Table 2. Overview of the development of diagnostic reference levels (DRLs) in the Nordic countries, Estonia and Lithuania.

Country	
Denmark (DK)	Nordic recommendations: 1996 National DRLs: May 2001 Revised DRLs: January 2007
Estonia (EE)	Limited dose surveys: 2002-2006 (recommended DRLs published in scientific papers) National DRLs: Currently being set
Finland (FI)	Nordic recommendations: 1996 National DRLs: December 2000 Revised DRLs: January 2006 Revised DRLs: April 2007 Revised DRLs: January 2009
Iceland (IS)	Nordic recommendations: 1996 (still used as national DRLs)
Lithuania (LT)	National DRLs: December 2004 Revised DRLs: January 2009
Norway (NO)	Nordic recommendations: 1996 National DRLs: Values are set, publication planned around May 2010
Sweden (SE)	Nordic recommendations: 1996 National DRLs: October 2002 Dose survey: 2005-2006 Dose survey: 2008

Values of diagnostic reference levels

Table 2 shows an overview of how DRLs have been/are being set and revised. The first values were the common set of recommended DRL values established in 1996 by the Nordic countries. When the EC Medical directive was implemented in the beginning of the 2000s, Denmark, Finland and Sweden set new national DRLs, partly based on the common Nordic values but supplemented by values for CT and mammography, and in Sweden also for cardiac angiography. Estonia and Lithuania joined the EU in 2004, and Lithuania set national DRLs at the end of 2004. In Estonia limited dose surveys were conducted about 5 years ago and recommended DRLs were published in scientific papers [5, 7], but national DRLs are only now under way. Iceland and Norway are not members of the European Union, and are thus not required to set national DRLs. Iceland has continued to use the DRL values from the Nordic recommendation. Norway has recently set new national DRL values, which are included here; publication is planned for May 2010.

Denmark, Finland and Lithuania have revised their national DRLs based on national dose surveys within the last few years. In Denmark, only the values for the conventional examinations were revised, a revision of the remaining values is still pending. In Finland and Lithuania, the values for all types of examinations were revised, and also certain types of examinations were added to or removed from the list of DRLs. In Sweden, the DRLs have not been revised, but nationwide dose surveys have been carried out twice, and the results (third quartile) from these surveys have been included here. These results from the dose surveys will also be referred to as DRLs and be included in the comparison below. Also, even though the values from Estonia are not official national DRLs, they have also been included in the comparison.

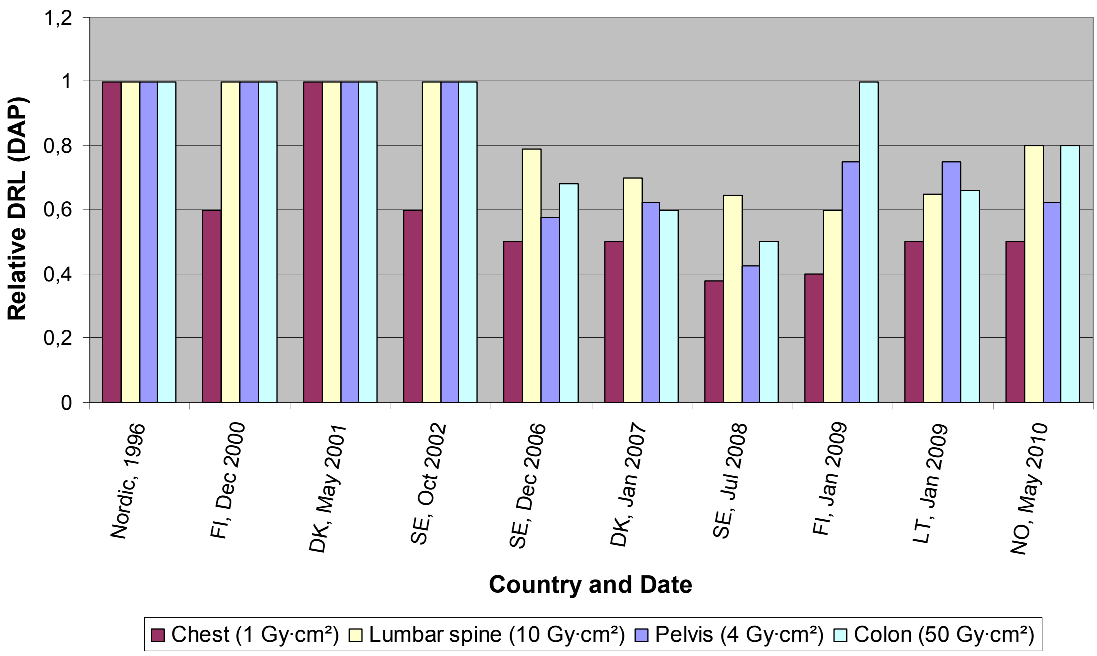

Figure 1. Development of the value of DRL (total DAP for the examination) for three conventional and one fluoroscopic examinations. The value of each DRL is shown relative to the value of the original common Nordic DRL, the value of which is shown in the legend.

Figure 1 shows the development in the DRL values for total DAP for conventional and fluoroscopic examinations, as shown by the ratio of the relevant DRL to the common Nordic DRL from 1996. The examinations, three conventional and one fluoroscopic, for which DRLs have been set by nearly all countries (in Estonia, no value for colon has been proposed), have been included in the figure. Since the first set of DRLs from Lithuania and the recommended values from Estonia were given only in terms of ESD they could not be included in figure 1. Instead, the development in the DRL values for ESD for selected individual projections of the three conventional examinations is shown in figure 2. The figures show that in general, the DRLs have been decreasing over the last decade. Looking at any single country, the DRLs have been constant or decreasing, but there is some variation in the values between the different countries. The DRL values from Lithuania and Estonia for ESD are generally higher than the Nordic values, but the 2009 values from Lithuania are considerably lower than the 2004 values. The 2009 DRL values for DAP from Lithuania are also in the same range as the newer values from Denmark, Finland, Norway and Sweden.

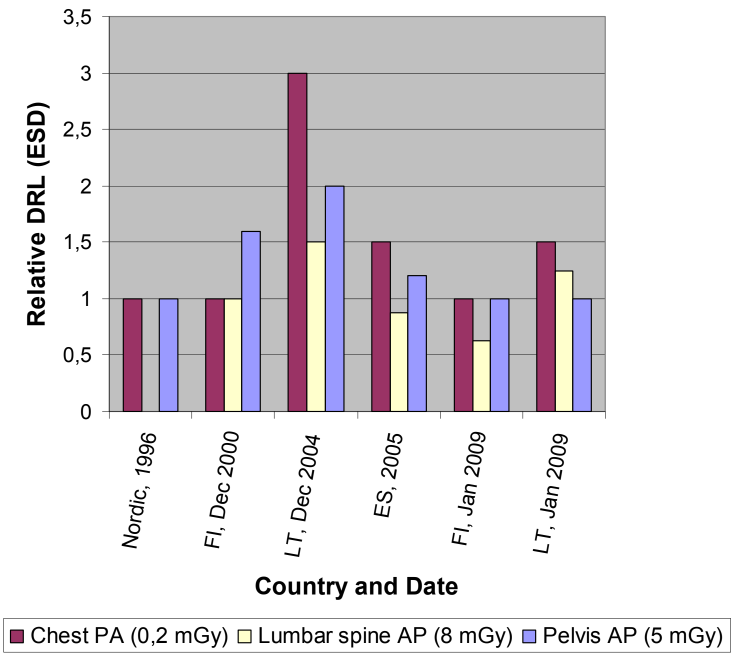

Figure 2. Development of the value of DRL (ESD for one projectin) for three conventional examinations. The value of each DRL is shown relative to the value of the original common Nordic DRL/DRL from Finland 2000, the value of which is shown in the legend.

Figure 3 shows the development in the DRL values for DLP for CT examinations, as shown by the ratio of the relevant DRL to the first DRLs values set, the DRL values of Finland from 2000. Included in the figure are the three CT examinations for which DRLs have been set by the largest number of countries – all except Estonia and Iceland. The first set of DRLs for Lithuania had to be excluded, since the DRLs were not given as DLP values. Overall, figure 3 shows that the DRLs for CT examinations have been relatively constant.

The DRLs for mammography and for cardiac examinations have not been compared between the countries here. For mammography, some countries have DRLs for the entire examination, while others have DRLs for single exposures. This makes a direct comparison very difficult. For cardiac examinations, DRLs have only been set by three countries.

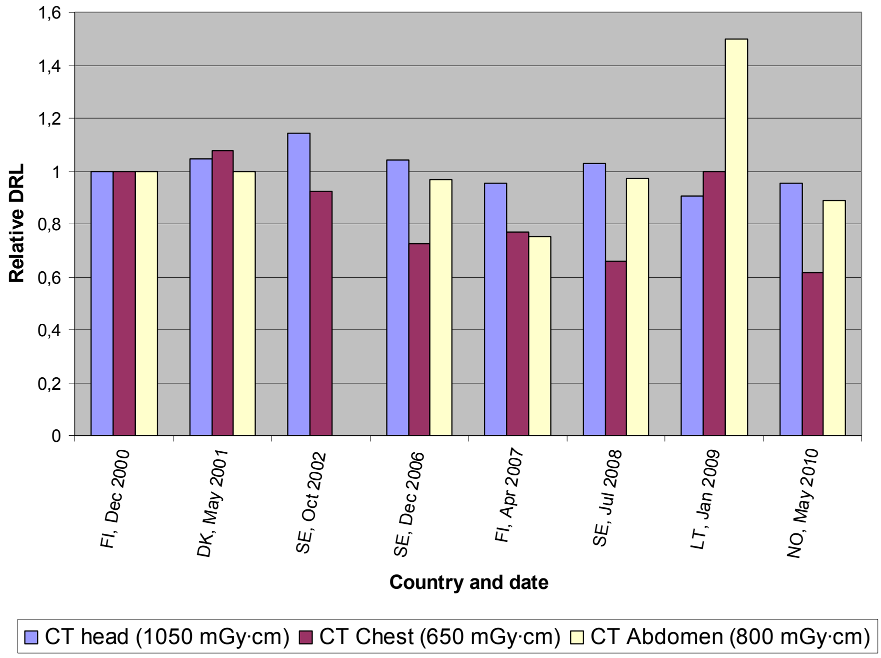

Figure 3. Development of the value of DRL (DLP for the examination) for three CT examinations. The value of each DRL is shown relative to the value of the original common Nordic DRL, the value of which is shown in the legend.

For paediatric examinations, only Denmark and Finland have set DRLs. For MCU, the Danish DRLs were based on the Finnish values, consolidated by results from a Danish survey, so the values are identical. The only other paediatric examination where both countries have DRLs is chest. Figure 4 shows a comparison of the DRL values for ESD as a function of the size of the child. The Danish DRLs are given in terms of the equivalent diameter, but have been converted approximately to actual thickness by using data for growth curves for Danish children [2].

Discussion

As can be seen from Table 1, the number of examinations for which DRLs have been set as well as the total number of DRL values vary considerably between the different countries. The number of DRLs for examinations of adult patients varies between 6 and 22 with an average of 14. The number of examinations for which DRLs have been set is quite similar in Denmark, Finland, Lithuania and Sweden, while Iceland and Norway have set DRLs for considerably fewer and more examinations, respectively.

Finland and Lithuania normally have DRLs for several quantities for the same type of examination, while Denmark and Iceland in general only have one DRL per examination. Norway and Sweden are in between, having set DRLs for several quantities for some examinations and only one DRL for some.

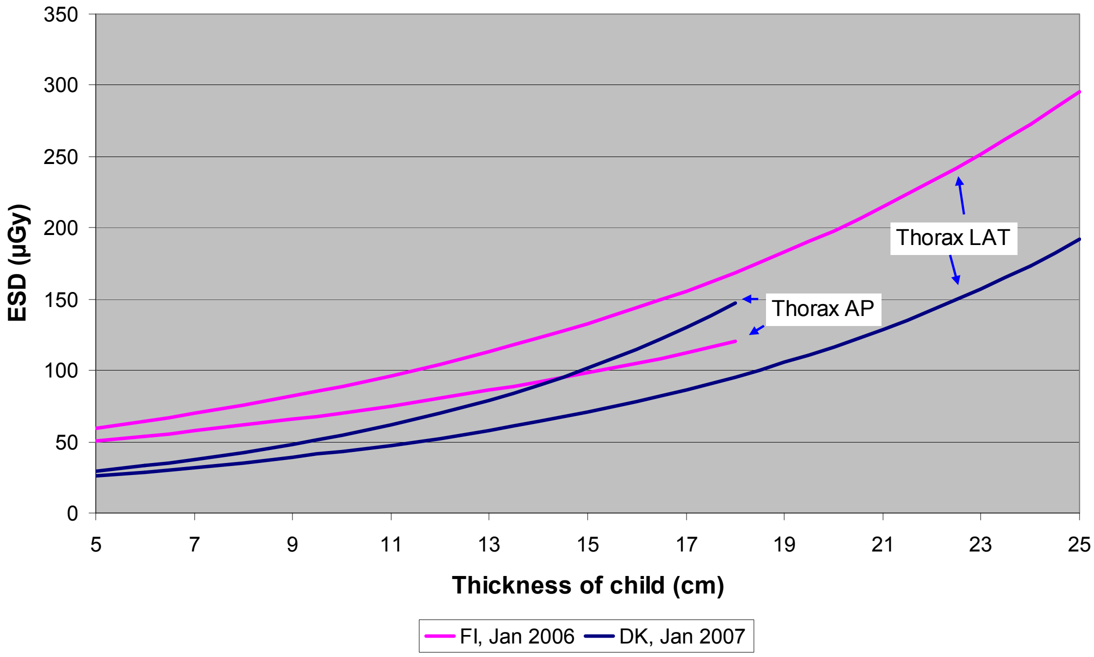

Figure 4. Comparison of the DRLs for paediatric chest examinations.

The development in the DRLs values in the Nordic countries has been quite similar over time, although there are also some differences between the countries. For conventional examinations, the values have decreased between 25 % and 60 %, reflecting the development in equipment and probably also the focus on optimization during this period. For the fluoroscopic examinations, represented by colon (barium enema), the decrease is smaller, and in Finland the value is unchanged. For CT, the DRL values have been quite constant, except for CT chest were the recent values from Sweden and Norway are 40 % lower than earlier values.

The DRL values from the two Baltic countries, Lithuania and Estonia from around 2005 were generally higher than the values from the Nordic countries, especially the Lithuanian DRL for the ESD for Chest PA, which was 200 % higher. However, the recent DRLs from Lithuania are much lower and for conventional and fluoroscopic examinations quite similar to the recently set Norwegian values. For CT, the recent DRLs from Lithuania are still somewhat high, especially the DRL for CT abdomen, which is 50 % higher than the Nordic values.

For the paediatric DRLs for Chest AP and LAT, the dependence on patient size is quite similar for the Danish and Finnish values. The DRL values for Chest AP are quite similar, although the Danish values increase slightly more with increasing patient thickness than the Finnish. Thus, the Danish values are lowest for small patients, but higher for larger patients. The Danish values for Chest LAT are consistently lower than the Finnish values, from 30 % lower for the larger patients and up to nearly 60 % lower for the smaller patients. Comparing the values for the two projections, there is also an interesting difference. The Danish values for LAT are lower than the values for AP, while the Finnish values for LAT are higher.

Conclusions

The DRLs of the Nordic countries and two of the Baltic countries, Estonia and Lithuania, have been compared. For conventional and fluoroscopic examinations, the general trend is that the DRLs have decreased over the last 10 years, indicating that the patient doses for these examinations have decreased. For CT, the DRLs are more constant, reflecting that the patient doses have not decreased.

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Modification of nuclear medicine diagnostic reference levels in Austria

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Abstract

Aim: Diagnostic reference levels (DRLs) are implemented to optimize diagnostic medical examinations using ionizing radiation. In Austria DRLs were established for the first time 2005 by the Austrian medical radiation protection regulation. In a nation wide survey we asked for daily routine patient details. The aim was to adjust the DRLs in Austria, considering new device developments as well as modified acquisition protocols.

Materials and methods: Using pooled data of 5320 patients from the survey, descriptive statistics was done. Frequency, age distribution and mean applied activities were determined, whereby we estimated also an individual effective dose for particular examinations. An extrapolation concerning the cumulative exposure of the population were also taken into consideration.

Results: In Austria the most frequent examination was thyroid scintigraphy using 99mTc-pertechnetate (23.2%), followed by heart (19.7%), bone (18.7%), renal (8.6%), PET-FDG (7.7%) and lung (7.5%) scintigraphy. About 150000 nuclear medicine procedures in 71 nuclear medicine facilities were performed 2008. An extrapolation revealed 35000 thyroid, 28000 bone, 29500 myocardial 12900 renal, 11500 PET-FDG, 11200 lung and 21900 miscellaneous scintigraphies. Regarding the number of nuclear medicine examinations per 1000 Austrian residents, Austria is located on the fourth place in comparison with 8 other European countries. The average effective dose for a nuclear medicine procedure was 3.9 mSv (range: 0.2–22.2 mSv). A mean effective dose of 0.07 mSv per year and resident was estimated. The additive part of the nuclear examinations compared to the total natural effective dose of 3.8 mSv per year could be stated as marginal.

Conclusion: Due to our survey results we suggested a modification of the DRLs in 73%. These alterations will be presented and implemented in a revised version of the forthcoming amendment of the Austrian medical radiation protection regulation.

Medical exposure to ionizing radiation in Switzerland – implementation of national diagnostic reference levels in radiology and nuclear medicine

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Abstract

The frequency and complexity of medical radiological procedures using ionizing radiation has been increasing continuously over the past years. As a direct consequence also patient doses have been increasing significantly. Since there are no dose limits for patients when applying ionizing radiation in medicine radiation protection is based on two basic principles – justification and optimization.

The concept of diagnostic reference levels (DRLs) first introduced in 1996 by the International Commission on Radiological Protection represents a powerful tool for dose optimization of radiological procedures. In Switzerland, DRLs are implemented in the national legislation. Radiology departments are obliged to compare periodically patient doses to DRLs and to optimize their procedures if necessary.

The Federal Office of Public Health (FOPH) provides national DRLs in the field of nuclear medicine (based on a national survey in 2004), interventional radiology (IR) and cardiology (based on a national survey in 2002) and computed tomography (CT) (partly based on a national survey in 1998 and on literature). However, effective optimization of radiological procedures is only feasible if DRLs are periodically adapted to the local practice and the actual state of technology. Therefore, national surveys have been repeated in IR and cardiology in 2007 and in CT in 2007-09 to collect actual dose data. These data are statistically analyzed and DRLs will be adapted. In a further project, examination parameters in radiography were collected to derive DRLs for the entrance-surface-dose. National DRLs are published in technical bulletins and are available for free on the FOPH webpage.

Periodical audits of the radiology departments and the on-site comparison of local patient doses to national DRLs are most effective for optimizing radiological procedures. To assess the change of patient doses over time for each radiology department the FOPH is establishing a national data base of medical radiological examinations.

Introduction

Over the past years, a continuous increase in the frequency and complexity of radiological procedures applying ionizing radiation has been observed. Today, annually worldwide, more than 3,600 million X-ray examinations are performed in radiography and around 37 million nuclear medicine and 7.5 million radiotherapy procedures are carried out (WHO 2008). As a direct consequence, the contribution of the medical application of ionizing radiation to the collective dose has also been increasing. Although the diagnostic and therapeutic benefits to the patients are enormous the widespread use of radiological procedures becomes a major concern in health care.

Since in medicine there are no dose limits for patients radiation protection is mainly based on two pillars – justification (i.e. provide more good than harm to the patient) and optimization (following the ALARA principle "as low as reasonably achievable") of radiological procedures, as first introduced by the International Commission on Radiological Protection (ICRP 1996). One powerful tool for optimization of radiological procedures represents the concept of diagnostic reference levels (DRLs). In diagnostic radiology, DRLs are derived empirically by determining the 75th percentile of the distribution of a defined dosimetric quantity for routine conditions and for standard patients. In nuclear medicine, DRLs are defined by the median of the applied activities. DRLs are specific to a country or region assessing the local radiological practice and should be periodically updated. The application of the concept of DRLs helps the radiologists, nuclear medicine professionals and radiographers to identify and optimize those radiological procedures which provide unusually high patient doses. In Switzerland, the concept of DRLs has been implemented in the national legislation (Swiss Federal Council 1994). Article 37a of the Radiological Protection Ordinance enables the Federal Office of Public Health (FOPH) to issue recommendations on radiation doses for diagnostic procedures in the form of DRLs taking into account data from national surveys and international recommendations. Furthermore, radiology departments are obliged to compare periodically patient doses to national DRLs and to optimize their procedures if doses exceed systematically corresponding DRLs.

The FOPH provides national DRLs in the field of nuclear medicine (based on a national survey in 2004), interventional radiology (IR) and cardiology (based on a national survey in 2002) and computed tomography (CT) (partly based on a national survey in 1998 and on data from literature). These radiological procedures contribute most to the collective dose of the Swiss population (Aroua et al. 2003) and thus DRLs are considered to be an essential tool for dose optimization in these fields. However, effective optimization is only feasible and reliable if DRLs are periodically adapted to the local practice and the actual state of technology. Therefore, national surveys have been repeated in IR and cardiology and in CT to collect actual dose data. Taking into account that by far most of the examinations are performed in radiography a further project of the FOPH was launched in 2009 to collect examination parameters of common radiographic examinations and to derive DRLs for the entrance-surface-dose. Simultaneously to data collection, a data base has been developed allowing further statistical analysis of dose data and the derivation of actual DRLs.

This paper provides a summary of actual national DRLs in nuclear medicine and diagnostic radiology in Switzerland.

Material and methods

Nuclear medicine

In 2004, a nationwide survey was performed and following data was collected: age and gender of the patients; type of examination; applied radionuclide and pharmaceutical; type of application; and applied activity (Roser 2006). Median values of the distribution of the applied activities were calculated. Based on these median values, practical experiences of nuclear medicine professionals, as well as DRLs of the neighboring countries Germany and Austria national DRLs were defined.

IR and cardiology

The first survey in the field of IR and cardiology in Switzerland was conducted in the university hospitals in 2002 to study the clinical practice of 8 fluoroscopic examinations. From 2006-2007, a second survey was performed including 32 small and medium sized hospitals (Trueb et al. 2009). For each center, dose data of 20 patients for the most frequent examinations were collected. These two surveys provide the basis for the derivation of the DRLs for the dose-area-product (DAP), the fluoroscopic time (T) and the number of images (N).

CT

DRLs in CT were introduced for 21 indication-based CT protocols which based partly on a national survey performed in 2004 (Aroua et al. 2004) and on recommendations of the European Commission (European Commission 1999). In 2007, a new project was launched by the FOPH with the aim to collect dose data (volume computed tomography dose index ($CTDI_{vol}$) and dose-length-product (DLP)) for every Swiss radiological institute operating a CT scanner and to adapt the previously defined DRLs to the actual practice (Treier et al. 2009).

Radiography

38 private radiology institutes and hospital radiology departments operating a digital X-ray system (either computed radiography (CR) or digital radiography (DR)) participated in a survey that was performed last year (Theiler 2009). Each radiology site provided exposition parameters (tube voltage, tube current, and focus-skin-distance) of 20 patients for three common types of radiographs (chest, pelvis, and lumbar spine, all in ap or pa projection). This allowed the entrance-surface-dose (ESD) to be calculated. The resulting DRLs were compared to the DRLs based on the European directive (European Commission 1999).

Data base

A national data base (DAMEX: data base of medical radiological examinations) has been developed based on Microsoft Access 2003 to collect and store patient doses and examination parameters of the national surveys. So far, DAMEX contains dose data of the surveys in CT and radiography as well as data specifying the X-ray units and radiology departments.

Results

National DRLs for nuclear medicine, IR and cardiology, CT, and radiography are published in technical bulletins of the FOPH and are available for free at

http://www.bag.admin.ch/themen/strahlung/00044/00071/index.html?lang=de
for nuclear medicine (L-08.01.pdf) and at

http://www.bag.admin.ch/themen/strahlung/02883/02885/02889/index.html?lang=de
for IR and cardiology (R-06-05.pdf), CT (R-06-06.pdf), radiography (R-06-04.pdf).

In the following paragraphs selected results of these technical bulletins are presented.

Nuclear medicine

The most frequently used radionuclide in nuclear medicine is technetium (Tc-99m) in combination with the pharmaceutical MIBI (cardiolite). In case of positron-emission-tomography (PET) examinations fluorine (F-18) together with the pharmaceutical FDG is most widely applied. In table 1 DRLs for common examinations in nuclear medicine using Tc-99m and F-18 are presented.

Table 1. DRLs (median) for common examinations in nuclear medicine using Tc-99m MIBI (cardiolite) and F-18 FDG.

Examination	Radionuclide	Pharmaceutical	Application	DRL [MBq]
Thyroid	Tc-99m	MIBI (cardiolite)	intravenous	170
Parathyroid	Tc-99m	MIBI (cardiolite)	intravenous	550
Myocardium	Tc-99m	MIBI (cardiolite)	intravenous	300 + 900 * 2 x 600 **
Tumors	Tc-99m	MIBI (cardiolite)	intravenous	740
Tumors (PET)	F-18	FDG	intravenous	350
Neurology (PET)	F-18	FDG	intravenous	200
Heart (PET)	F-18	FDG	intravenous	250

* one-day protocol: 300 MBq and 900 MBq

** two-days protocol: two times 600 MBq

IR and cardiology

DRLs for five types of interventional examinations (coronary angiography, percutaneous transluminal coronary angiography (PTCA), cerebral angiography, angiography of the lower limbs, and PTA of lower limbs) are shown in table 2. Values for small and medium sized hospitals are compared to those of university hospitals.

Table 2. DRLs (75th percentile) for DAP [Gy·cm²], fluoroscopic time T [min], number of images N.

Examination	Measure	Small and medium sized hospitals	University hospitals	Ratio
Coronary angiography	DAP T N	110 10 1500	80 7 1400	1.38 1.43 1.07
PTCA	DAP T N	150 20 1800	110 20 1500	1.36 1.00 1.20
Cerebral angiography	DAP T N	160 8 240	125 15 480	1.28 0.53 0.50
Angiography of lower limbs	DAP T N	70 8 150	210 8 150	0.33 1.00 1.00
PTA of lower limbs	DAP T N	70 22 180	460 25 200	0.15 0.88 0.90

For coronary angiography, PTCA, and cerebral angiography the DRLs of the DAP in the small and medium sized hospitals are 30-40% higher than the DRLs in the university hospitals. In contrast, DRLs for angiography and PTA of lower limbs are smaller by a factor 3-7.

CT

Standard CT protocols of the skull/brain, thorax, and abdomen/pelvis are most frequently applied in Swiss radiology departments. Results of the surveys in 2004 and 2007-2009 for these 3 examination types are summarized in table 3. Interestingly, changes in DRLs between the two surveys are very small.

Table 3. DRLs (75th percentile) for $CTDI_{vol}$ and DLP of standard examinations of the skull/brain, thorax, and abdomen/pelvis.

Examination	Indication	Survey 2004 $CTDI_{vol}$ [mGy]	Survey 2007 $CTDI_{vol}$ [mGy]	Survey 2004 DLP [mGy·cm]	Survey 2007 DLP [mGy·cm]
Skull/brain	Metastases, abscess	60	65	1000	1000
Thorax	Angiography, pulmonary embolism	15	15	450	450
Abdomen/pelvis	Emergency, abscess	15	15	700	650

Radiography

Results demonstrated that the calculated DRLs based on the survey were significantly lower than the existing DRLs as proposed by the European Commission (table 4). Further analysis has shown slightly lower ESD values for DR-systems compared to CR-systems (data not presented) mainly caused by the improved detector quantum efficiency of solid state detectors compared to phosphorous imaging plates.

Table 4. DRLs for ESD [mGy] of radiographs of the chest, pelvis, and lumbar spine.

Examination	European Commission	Survey 2009	Ratio
Chest ap/pa	0.3	0.16	0.53
Pelvis ap	10	3.19	0.32
Lumbar spine ap/pa	10	6.58	0.66

DAMEX

In total, data of 187 hospitals and private radiology institutes participating in the surveys in CT and radiography were recorded. 214 CT scanners and 43 radiography units provided 1973 and 2320 patient dose data sets, respectively. The structure and complexity of DAMEX is depicted in figure 1.

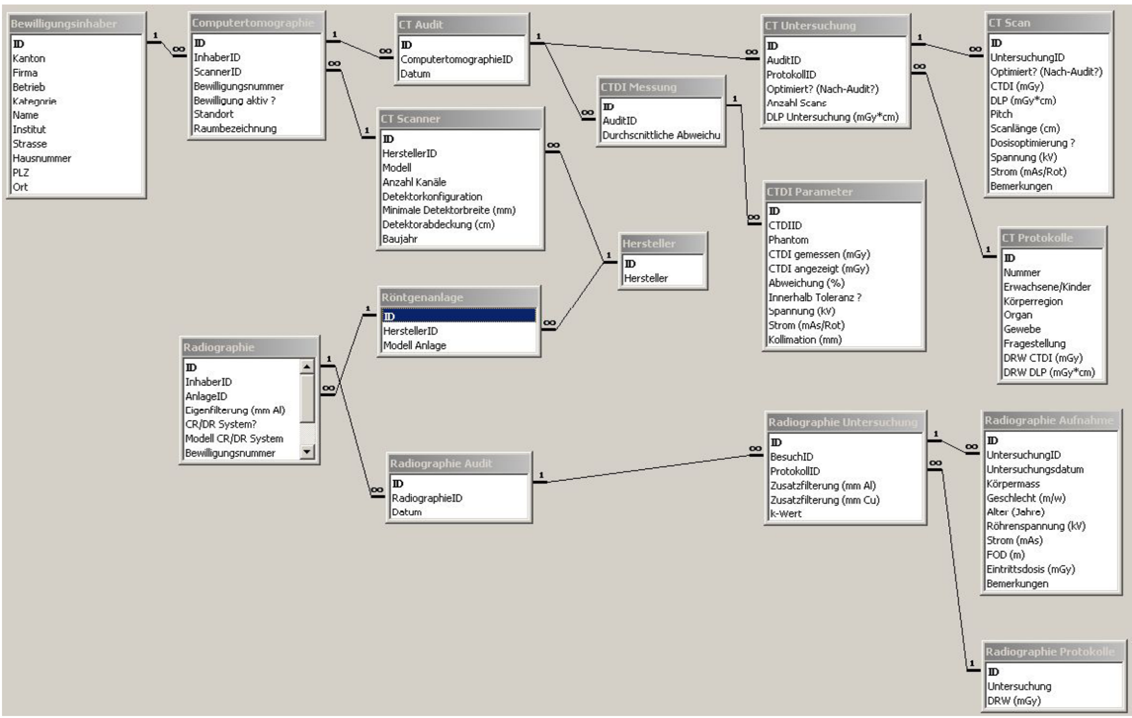

Fig. 1. Screenshot illustrating the tree structure of DAMEX and the relationships between the different tables.

Discussion

DRLs for nuclear medicine, IR and cardiology, CT, and radiography were derived based on nationwide surveys performed in the last few years in Switzerland. In IR and cardiology and CT surveys have been repeated to adapt DRLs to the actual practice. DRLs are published in technical bulletins and uploaded on the FOPH website.

Results for IR and cardiology have shown significant differences in the DRLs between small and medium sized hospitals and university hospitals. The reasons are manifold. University hospitals often treat complex cases where an increased dose is necessary. Also, the experience of the operator plays an important role. Junior doctors more often work in university hospitals and need more time to perform certain examinations. To avoid these problems DRLs could be defined separately depending on the size of the hospital. For CT, the change in DRLs from the first to the second survey was only minor although technical advances of CT scanners have significantly improved the radiation efficiency. This indicates that the concept of DRLs has not yet been correctly applied for CT examinations in clinical routine. In radiography, national DRLs were much lower compared to those defined by the European Commission. One reason for this remarkable result represents the high optimization potential of digital imaging systems (particularly DR-systems) when compared to screen-film combinations.

For data collection and statistical analysis of patient doses a data base using the software package of Microsoft Access has been developed. Data for IR and cardiology and CT have already been entered. An extension of the data base is scheduled since more surveys will follow.

Conclusions

In Switzerland, much effort has been made to optimize radiological procedures aiming at the reduction of patient doses. National DRLs for various fields in radiology have been defined and published in technical bulletins. Periodical audits of the radiology departments are performed and local patient doses are compared on-site to national DRLs. By establishing a national data base and continuously collecting dose data the development of patient doses over time can be assessed. However, the other basic principle in radiation protection, i.e. justification of radiological procedures, has been almost completely neglected so far. Thus, the FOPH is planning to launch a project to introduce clinical audits in Switzerland. Clinical audits are systematic and continuous assessments of all radiological processes performed during the clinical pathway of the patient. The aim is to identify and eliminate those radiological procedures that are not justified and to optimize the justified procedures. For a successful implementation of clinical audits a close collaboration of all the involved stakeholders (nuclear medicine professionals, radio-oncologists, radiologists, radiographers, medical physicist, and authorities) is mandatory.

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The study of patient doses for establishing diagnostic reference levels of exposure in Ukraine

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Abstract

Medical exposure of patients in diagnostic radiology is the highest source of population artificial irradiation. Optimization of radiation protection in diagnostic radiology should be focused on reducing patients doses obtained from most wide-spread medical examinations while keeping high quality of diagnostic information. International Basic Safety Standards (BSS-115) established guidance levels of patient doses for diagnostic radiography. Different countries can use these values until national diagnostic reference levels are defined. Since 2009 the scientific research of patient doses in diagnostic radiology has started in Ukraine. The main purpose of this work is to evaluate the structure and parameters of X-ray diagnostic examinations, the current state of X-Ray diagnostic units, radiation output of X-ray units in different hospitals, and to measure patient doses for 5 main radiography examinations: chest; cervical spine, thoracic spine, lumber spine, pelvis (for main projections). The measurements of radiation parameters were carried out for 35 X-ray diagnostic units from 12 large hospitals of 5 Ukrainian regions. The entrance surface doses were measured for more than 600 patients with TL-detectors LiF type MTS-N (Poland). For different types of X-ray units the values of entrance surface doses differ in 20–40 times. For chest X-Ray the average dose was about 0.83 mGy. The maximum average patient doses were observed for thoracic spine diagnostic examinations – 19.7 mGy (AP), 35 mGy (LAT) and for lumber spine: 18.5 mGy (AP) and 46 mGy (LAT). At present the measurements of patient doses in Ukrainian hospitals are continued. To establish diagnostic reference levels for main radiography examinations the study of patient doses distribution will be done.

Real-time measurement of ovaries dose burden for I-131 treatment patient

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Abstract

The ovaries dose burden for post-operation thyroid cancer patient undergoing I-131 treatment is estimated by directly measuring ovaries skin entrance dose. Iodine-131 environmental dose rates are determined in real-time basis at 1 meter distance from the patient body to check for the regulatory compliance.

Dual digital dosimeters are deployed at the patient's ovaries proximity skin location (OS-1 meter) and at 1 meter distance away from the body surface (EV-1 meter). The OS-1 meter is operated in accumulated mode to totalize the overall dose burden during the I-131 treatment and the EV-1 meter is operated in dose-rate mode to monitor instant environment dose rate read-outs, respectively. Both of the measurements are recorded in hourly basis for up to 4 consecutive days. The times and multiple trips to the bathroom during 4-day period are also been tracked by patient herself. The ovaries dose burden determined by the ovaries skin entrance dose is at a 0.075 rad/mCi (mGy/3.7MBq) value for 4 days accumulated data in this study which is about half of the value from the nuclear medicine reference manual. Iodine effective half-life by using the imaging ROI data is at a 15.4 hours in our case of study, but the whole body iodine clearance half-lives can vary between 6 to 16 hours depending on foods and water taken and how often bathroom trips conducted. The correlation ratios between OS-1 and EV-1 meters read-outs can vary between 22 to 38 from time to time due to I-131 bio-distribution differences during our 4 days of observations. The ovaries dose burden can be measured directly by using real-time digital dosimeter. Real-time environmental dose rate verification, below than 50 micro-Sv/h required, can be used to meet the safety compliance when we are allowed to discharge an I-131 patient.

Introduction

When post-operation thyroid cancer patients were treated with large doses of Iodine-131 for targeted therapy, he or she would have significant radiation doses delivered to extrathyroidal tissues that might induce some side effects. Recently, these side effects, including bone marrow insufficiency and late occurrence of extrathyroidal cancer and leukemia, have been published extensively elsewhere. (Sawka 2009; Brown 2008;

Remy 2008) There was also a concern about young females with I-131 ablation treatment after Chernobyl accident. (Travis 2006) With the completion of our new I-131 therapy ward, we have been receiving more thyroid cancer patients who are still in reproductive age. We are more interested in ovaries dose-related infertility among young females who need larger doses of I-131 therapy and would like to monitor environment dose rate for radiation safety compliance for the general public.

In our present of study, the ovaries dose burden for post-operation thyroid cancer patient undergoing I-131 treatment is estimated by directly measuring ovaries skin entrance dose. Iodine-131 environmental dose rates are determined in real-time basis at 1 meter distance from the patient body to check for the regulatory compliance before releasing a patient. Iodine effective half-life is also determined by using time-profile of gamma scintigraphic images at the remnant of thyroid tissues for bio-effective or bio-clearance evaluations.

Material and methods

Real-time, dual digital G-M dosimeters are deployed at the patient's ovaries proximity skin location (OS-1 meter) and at 1 meter distance away from the body surface (EV-1 meter). These miniature dosimeters were selected and evaluated among others earlier for the dose-response accuracy, reproducibility, linearity check that was verified for the suitable use in our dose assessment applications. The OS-1 meter is operated in accumulated mode to totalize the overall dose burden during the I-131 treatment and the EV-1 meter is operated in dose-rate mode to monitor instant environment dose rate read-outs, respectively. Both of the measurements are recorded in real-time basis during awakening hours continuously for up to 4 consecutive days. The frequency of multiple trips to the bathroom during 4-day period is also been tracked and recorded by patient herself.

The effective half-live of iodine uptake by the thyroid remnant is determined by using selected region of interest (ROI) total counts of gamma scintigraphic images at the neck area at 6 hours and 24 hours after I-131 drug being administered. Other whole body I-131 clearance rate was estimated for various time-frames, which was strongly affected by the patient foods consumed and frequency of bath-room trips. The average whole body clearance rates were calculated by the EV-1 meter read-outs located at one meter distance away from the patient.

Results

There are 40 consequent data taken during about 88 hours time span after I-131, with total activity of 75 mCi (2.78 GBq), is orally administered. Some real-time raw data are listed in Table 1 for demonstration purpose, including the initial and final readouts on the EV-1 and OS-1 dosimeters, respectively. Based on the last integral dose reading on the OS-1 meter, the ovaries dose burden determined at the ovaries skin entrance location gives a value of 56.3 mSv (5.63 rad). This is equivalent to a value of about 0.075 rad/mCi (mGy/3.7MBq) that is about half of the value from a nuclear medicine reference manual. (Kuni 1997) Iodine-131 effective residence half-life of the thyroid remnant by using the imaging ROI data is about 15.4 hours in our case of study. The whole body iodine clearance half-lives can vary between 6 to 16 hours depending on foods and water taken and how often bathroom trips conducted. The whole body I-131 clearance half-lives are calculated from Table 1 EV-1 dose rate meter at one meter distance. The correlation ratios between OS-1 and EV-1 meters dose rate read-outs can vary from 22 to 38 at different time based on our 4 days of observations. (Table 2)

Table 1. Data recorded on the dual dosimeters based on 4 days observation.

Approximate Time Elapsed	OS-1 Dose Integral Meter (milli-sievert)	EV-1 Dose Rate Meter (micro-sievert/h)
0 h (started)	0	149
6 h	14.4	77.3
12 h	24.0	58.9
24 h	39.4	39.2
2 d	50.8	7.80
3 d	54.5	6.39
88 h (ending)	56.3	4.16

Table 2. Correlation ratios between OS-1 and EV-1 dose rates at various time frames.

Approx. Time Elapsed	Derived OS-1 Skin Entrance Dose Rate (microsievert/h)	EV-1 Dose Rate (microsievert/h)	Correlation Ratio (OS-1/EV-1)
1 h	3140	82	38.2
2 h	3340	94	35.5
6 h	1910	77.3	24.7
12 h	1510	58.9	25.6
24 h	867	39.2	22.1
48 h	300	7.80	38.5
3 d	144	6.39	22.5

Conclusions

In this case study, we have demonstrated that the ovaries gamma dose burden can be measured directly by using real-time digital dosimeter at ovary skin entrance location.

This dose burden has assumed a minimum beta dose contribution within ovary tissues. Effective residence half life of I-131 at thyroid remnant can be determined by the gamma scintigraphic imaging at the neck location. Real-time environmental dose rate verification can be conducted by the EV-1 meter, that is required to be less than 50 micro-Sv/h, to meet the safety compliance when we are allowed to discharge an I-131 treatment patient.

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Evaluation of CT dose and image quality in three states of Brazil

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Abstract

The objective of this work was to perform a dosimetry study of 11 CT scanners located in Brazil and to evaluate the image quality of these equipments. The volume CT air kerma index (C_{VOL}) and the air kerma length product (P_{KL}) were estimated for each procedure on the basis of the normalized weighted air kerma indices in CT standard dosimetry phantoms ($_nC_W$), supplied by the ImPACT group. The scanning parameters for adults were collected for head, chest, high-resolution chest, abdomen and pelvis examinations. For each procedure and each institution 15 examinations, randomly selected, were evaluated and the following parameters were registered: tube voltage and current, gantry rotation time, collimation and slice thickness, pitch or increment, scan length and the scanner dose index indication. The image quality was evaluated using the ACR CT accreditation phantom, manufactured by Gammex. The results indicate that efforts should be concentrated in the maintenance of the equipments and specific training of the technicians. Most of the scanners have showed some non-conformity. For dose evaluation, the results showed a wide variation of air kerma values for the evaluated institutions. The range of C_{VOL} values for head scans varied between 18 and 105 mGy and the P_{KL}, from 256 to 1632 mGy.cm. For chest scans, the C_{VOL} values varied from 8 to 30 mGy, and P_{KL} values from 153 to 1120 mGy.cm For abdominal scans the CVOL values varied from 8 to 19 mGy and PKL values from 240 to 805 mGy.cm. Conclusions: Although the C_{VOL} and PKL values for most procedures performed in the institutions are compatible to the European reference levels, the image quality did not attend all ACR CT accreditation requirements.

Introduction

In Brazil, in the last years, there was an exponential increase in the number of CT scanners installed, especially multi-slice CT scanners (MDCT). This new technology created important advances in imaging modalities. New applications such as CT angiography and virtual endoscopy are a reality in many centers. However, as a result of the rapid development of these technologies, its use has become more widespread and the patient dose should be a health concern. According to UNSCEAR 2000, CT scans represent 34% of the annual collective dose of all X-ray diagnostic procedures. Recent studies indicate that the tendency of the dose is to increase, due to the possibility to carry one procedure in fraction of seconds with high special and temporal resolution. The tendency is also to increase the irradiated volume (thorax-abdomen-pelvis).

However, there is a lack of preparation of the professional involved in this practices. There is no quality assurance programs implemented in the services and the evaluation of patient dose is not a practice in the services. Consequently, the protocols are used without one plan for optimization specially, for paediatric patients.

The amount of radiation dose received from a CT scan depends on two main factors, the design of the scanner and the way it is used. Especially the MDCT can potentially result in higher radiation risk to the patient due to its increased capabilities such as long scan lengths at high tube currents with faster acquisition times and multi-phase contrast studies. The selected scanning parameters, such as kV, mAs, table increment/pitch and scan length, will mainly affect the radiation dose to the patient.

These parameters should be evaluated for dose reduction including those that can be modified based on patient size, study indication such as tube current, gantry rotation time, pitch, tube potential, scan covered, radiographic shields, automatic exposure control techniques, and noise reduction filters (Bongartz, 1999;Tsapaki, 2006).

A national survey never has been done in Brazil so far. However, in 2007, some Brazilian institutions participated in a survey carried out in Latin America under the IAEA Technical Cooperation Regional Project RLA/9/057: Radiological protection of patient and in medical exposure (Dias et al, 2007). The objective of this work was to evaluate the patient dose in different states of the country in order to be able to conduct a national survey in a near future.

Material and methods

In this study, institutions located in Rio de Janeiro, Pernambuco and Paraná have participated. The Table I show the distribution of the equipments in the states, the main characteristics of the equipments.

Methods for Estimate the CT dose indices

Values of Cvol and PKL were calculated for each axial or helical sequence on the basis of the representative n*C*W coefficients published by IMPACT [4] or when available the n*C*W value obtained in the dosimetry. For scanners operated in auto dose reduction mode with automatic tube current modulation, doses were calculated using reported values of (average) tube current or mAs that included the effects of modulation [5].

The volume CT air kerma index (C_{VOL}) and the air kerma length product (P_{KL}) were estimated for each procedure on the basis of the normalized weighted air kerma indices in CT standard dosimetry phantoms ($_nC_W$), supplied by the ImPACT group .The scanning parameters for adults were collected for: head, chest, high-resolution chest, abdomen and pelvis examinations. For each procedure and each institution, in Table 1, 15 examinations, randomly selected, were evaluated and the following parameters were registered: tube voltage and current, gantry rotation time, collimation and slice thickness, pitch or increment, scan length and the scanner dose index indication.

Table 1. Main characteristics of the evaluated CT Scanners.

State	Institution	Manufacturer/ Model	Scanner/Type	Number of Chanels	Year of Installation
Rio de Janeiro	1	Philips, Brilliance 64	MSCT	64	2006
	2	Philips, Brilliance 4	MSCT	4	2009
	3	Philips, Brilliance 40	MSCT	40	2008
Curitiba	4	GE, Hi Speed	SSCT	1	2008
	5	Toshiba Aquilion 64	MSCT	64	2007
Pernambuco	6	Philips Ct Aura	SSCT	1	2001
	7	Toshiba auklet	SSCT	1	2001
	8	GE HiSpeed FX/i	SSCT	1	2003
	9	GE HiSpeed FX/i	SSCT	1	2003
	10	Toshiba Asteion	SSCT	1	2003
	11	Siemens Somatom Sensation 64	MSCT	64	2003

Obs: SSCT: Single-Slice CT Scanner; MSCT: Multislice CT Scanner

Methods for Estimate the CT dose indices

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The volume CT air kerma index (C_{VOL}) and the air kerma length product (P_{KL}) were estimated for each procedure on the basis of the normalized weighted air kerma indices in CT standard dosimetry phantoms (${}_{n}C_{W}$), supplied by the ImPACT group The scanning parameters for adults were collected for: head, chest, high-resolution chest, abdomen and pelvis examinations. For each procedure and each institution 15 examinations, randomly selected, were evaluated and the following parameters were registered: tube voltage and current, gantry rotation time, collimation and slice thickness, pitch or increment, scan length and the scanner dose index indication.

Guidelines to optimize the protection during CT procedure include diagnostic reference levels (DRL) (Tsapaki,2006). The dose parameters suggested are the weighted CT kerma index (CW) and the kerma-length product (PKL).

Image Quality Evaluation

In this study, the ACR accreditation phantom, from Gammex, model 464, was used. The phantom allows evaluating the accuracy of positioning and alignment, accuracy of CT number, slice thickness, low and high resolution and uniformity of the CT number.

According with the instructions of the phantom manufacturer, the phantom images were acquired using the facility's typical head, high-resolution chest and abdomen exam protocols depending on the parameter evaluated.

Results

For each state, the number of participating hospitals varied. The scanners mode were approximately 46% multislices and 54% single slices.

Evaluation of Cvol and PKL mean values

The technical factors used by the participating hospitals revealed that in all procedures the tube voltage was 120 kVp but a very large variation of mAs was verified. The maximum values were higher than the values reported in the literature. Consequently, the inadequate choice of the exposure parameters resulted in unnecessary doses.

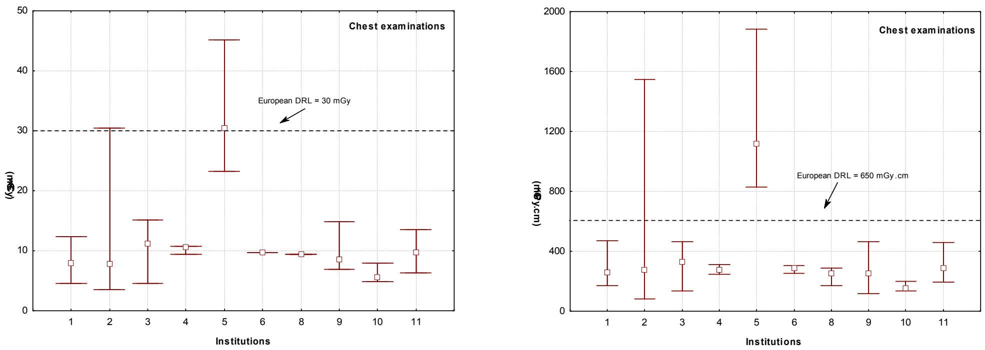

Fig. 1. Variation of the values of Cvol (a) and PKL (b) of the Chest examination in each institution.

The ranges of C_{vol} and P_{KL} are observed in the Figure 1 (a) and (b) respectively. Only the Institution 5 presented values higher than the DRL for chest. In the same institution is observed a large variation in the Cvol and DLP values, especially in the institution 2 and 5. In the sample, the average values were very similar, except in the institution 5.

The results of the values of Cvol and PKl for hi-hesolution can be shown in Figure 2.

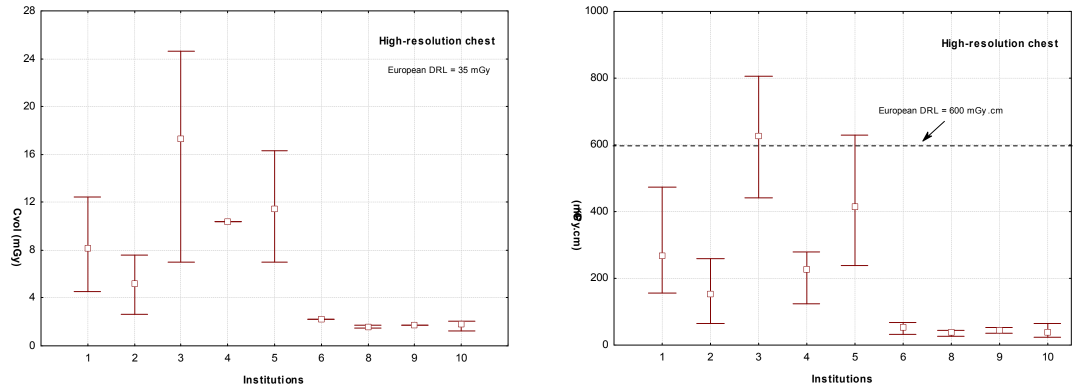

Fig. 2. Variation of the values of Cvol (a) and PKL (b) of the Hi-Resolution chest examination in each institution.

Figure 3 and 4 presents the results for abdomen and pelvis respectively. One institution from Curitiba doesn't carry out the pelvis procedure. All institutions from Recife don’t carry out both exams in their routine .

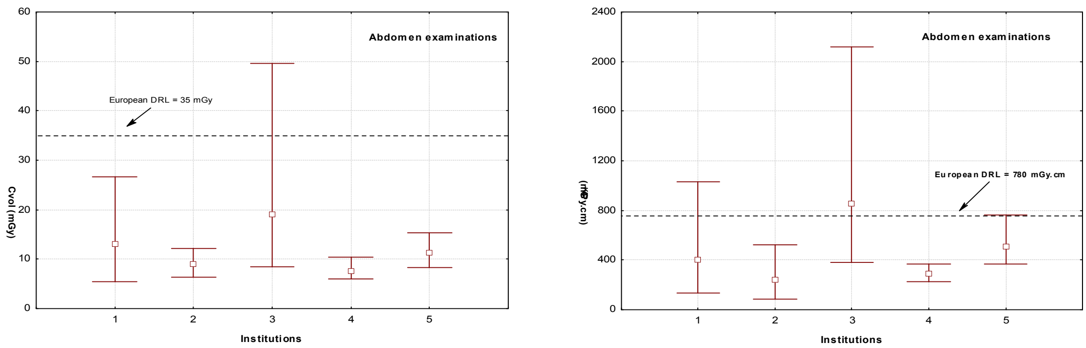

Fig. 3. Variation of the values of Cvol (a) and PKL (b) of the Abdomen examination in each institution.

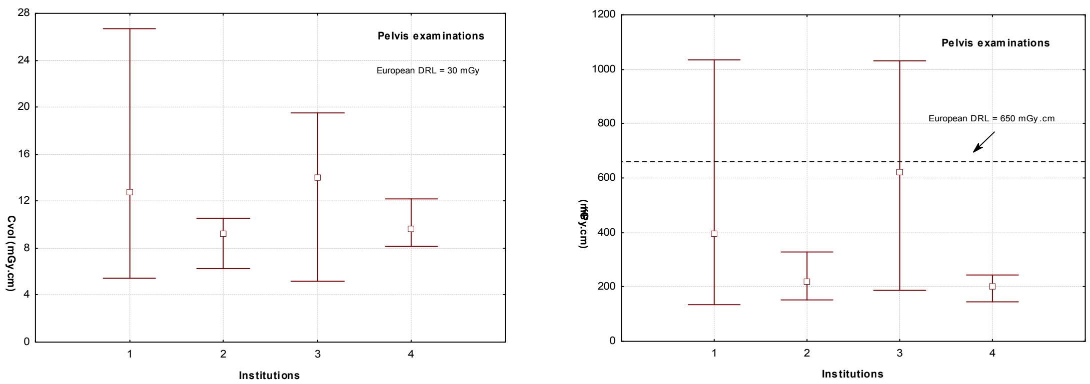

Fig. 4. Variation of the values of Cvol (a) and PKL (b) of the pelvis examination in each institution.

Evaluating the cvol and PKL values in abdomen and pelvis procedures we can observe that the scan lengths between these patients for each procedure presented consistency. The Institution 3 showed the biggest difference range in Cvol and PKL in the abdomen procedure. In the case of pelvis procedure, the biggest difference was in the Institution 1. It was also observed that most of the institutions presented values lower than the DRL for both procedures.

In head procedures (Figure 5), in four institutions the averages Cvol were higher than the DRL. The institution 2 showed a substantially wide range of doses in their centers.

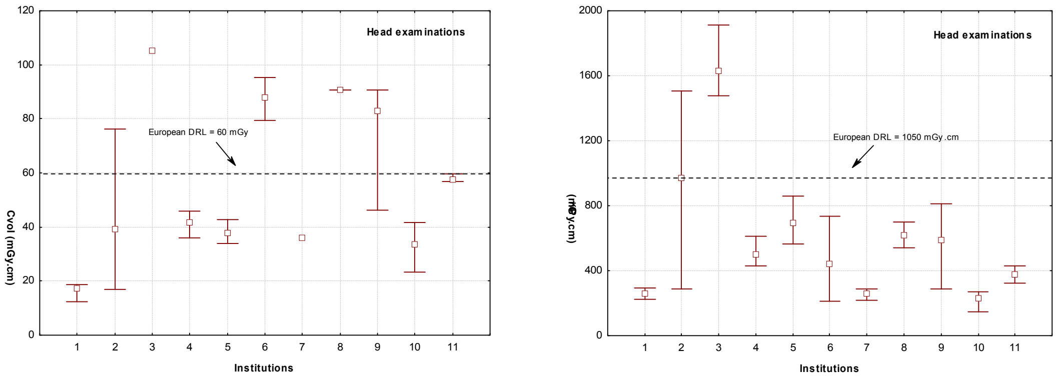

Fig. 5. Variation of the values of Cvol (a) and PKL (b) of head examination in each institution.

In the Figure 6 are presented the comparison of the dose between the different states for the five different procedures.

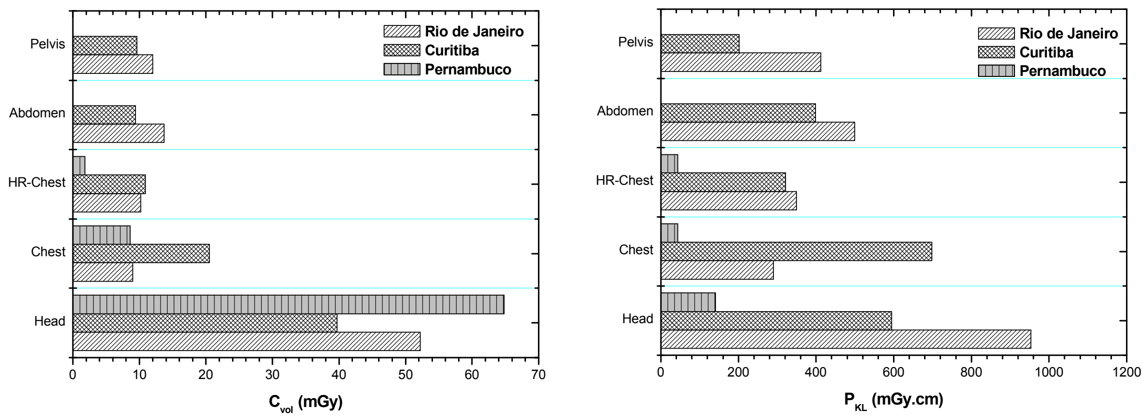

Fig 6. Variation of the values of Cvol (a) and PKL (b) for the five procedures in each state of Brazil.

Evaluation of Image quality

The Figure 7 indicates the percentage of conformity of the parameters with the international requirements. It is possible to observe that only the uniformity and slice tickness was completely adequate in all equipments and only on scanner comply with the requirements of CT number.

All institutions presented at least one no conformity. Many scanners failed in more than 50% of the parameters evaluated and in some cases did not comply up than 80% of the ACR requirements. The conformity of low contrast resolution was observed only in 67% of the institutions.

Only slice thickness accuracy and uniformity was adequate in 100% of the scanners but only one scanner comply with the requirement of accuracy of CT number. Also image noise was verified in 50% of the scanners and all of them were artifact like a ring.

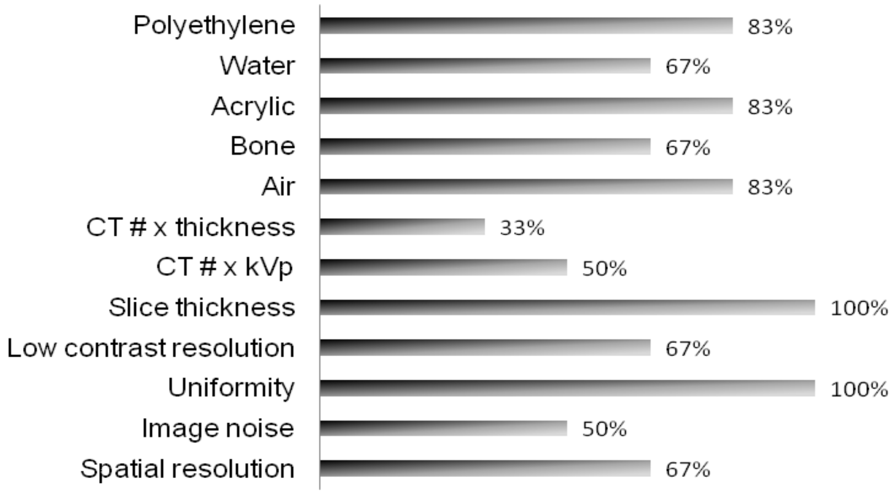

Fig 7. Frequency of acceptability of each criteria of image quality.

Discussion

The first information we have had during this work was the lack of appropriate training and especially of one program of continuum education. The professionals did not know all the equipments resources and the possibility to get better results in the procedures.The recommendation is to carry on the exams according exactly to the protocol, independently of the age, weight and size of the patient.

For the conduction of the tests many times was necessary to chance some parameters. This was a special problem for them because they didn't know how to conduct some procedure that was out of their routine practice.

The results of the Chest for Institution 5 of C_{vol} and PKL values were 1,5% and 72 % respectively higher than the DRL for chest. Considering the maximum C_{vol} and PKL, the values were respectively 45% and 189% higher than the DRL.

The wide range of conformity in low contrast observed in most of 60% of the all CT evaluated increase the risk of poor diagnostic, due the poor resolution doesn't allow visualization an indication of disease.

The spacial resolution test verified the ability of the scanner to detect small lesions that has attenuation values which differs greatly from those of surrounding tissue. The results showed that only 67% of the institutions are within the limits minimum acceptable. The physical size is also thought of in terms of detail to be discerned. It is expecting to resolve at least 5lp/cm for adult abdomen and 6 lp/cm for high-resolution chest.

In addition, the staff should know the equipment and the system with which they work to know how to identify the error messages and possible causes of failures. To ensure the production of images with a diagnostic quality, the equipment must be properly calibrated and subjected to a rigorous routine of quality control, including regular tests and verification of the professionals routine.

The conformity only in 50% of the CT's in noise level implies that the accuracy of detectability of small or low contrast lesions could be seriously affected .

Conclusions

In all Brazilian institutions the C_{vol} and P_{KL} values obtained in this survey was lower that the DRL European guidelines. However in all procedures, large ranges were observed in patient radiation doses in the different centers.

For these routine examinations the European Guidelines (Bongartz, 1999) recommends a pitch equal a 1. However in this study, the pitch ranged from 0,2 to 1,35 in all procedures.

It's necessary to inform Institution 5 about the dose values for chest procedures and maybe orientate for protocols review.

In Brazil, only few institutions carry out image quality evaluation. In general, there isn't a quality assurance implemented. Also, we don't have a CT accreditation program. institutions perform the exams without an adequate control of the performance of the scanners.

Also, we verified the lack of training of the professionals involved in the exam, especially in the centres where the new technologies were implemented.

Ackowledgments

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Is the radiation dose received by patients undergoing multiple CT radiofrequency ablations too high?

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Abstract

Objective: Radiofrequency ablation (RFA) of tumors has gained significant interest and acceptance in the last decade due to its potential to produce a large volume of cancer cell death in a controlled fashion. It is successfully used under CT guidance to treat certain tumors of lung, liver, kidney, and bone. The objective of the study was to calculate the effective and skin doses in patients that underwent repeated CT guided radiofrequency ablations (RFA).

Materials and methods: The retrospective review included patients having more than 3 RFA ablations during the study period (June 2003-June 2009). The dose length product (DLP), effective dose (E), skin dose profiles, as well as the peak skin dose (PSD) for both the bottom and upper patient skin surface were calculated for each RFA using appropriate methods and software developed during the study. For each patient, the cumulative DLPc and Ec were also calculated from the sum of the respective figures of each individual procedure. To calculate the PSD, the skin dose profiles of each procedure were projected on the same scale using anatomical points for reference and then they were summed up.

Results: Five patients were analyzed. Four of these patients had totally 3 RFA and one had 11 RFA during the study period. DLPc, Ec and PSDc range values were 5569-9642 mGycm, 83-173 mSv and 774-3364 mGy, respectively. Median E and PSD for each RFA were 35 mSv and 400 mGy. For comparison purposes, a routine abdominal CT imparts approximately 8-10 mSv.

Conclusions: DLPc, Ec and PSDc vary greatly even for the same patient. However, the high radiation dose associated with repeated RFA shows the need to monitor in more detail these patients.

Introduction

Computed tomography (CT) is one of the imaging techniques currently used to perform Radiofrequency ablation (RFA). This interventional technique has gained significant interest and acceptance in the last decade due to its potential to produce a large volume of cancer cell death in a controlled fashion. It is successfully used under CT guidance to treat certain tumors of lung, liver, kidney, and bone. The main advantages of CT guided RFA compared to ultrasound or magnetic resonance are that it can be repeated many times if required, with much lower cost compared to MRI and that it provides a more detailed imaging than ultrasound. [Laspas F]. Patients can be treated more than once with CT guided RFA. It should be noted however, that the issue of radiation dose in CT has received a lot of attention during the last years. There is evidence that 1.5-2% of cancers may eventually be caused by the radiation dose used in CT [Brenner DJ]. Furthermore, the media occasionally contains exaggerated articles regarding CT radiation doses. These articles can be very easily misinterpreted by the general population who is not sufficiently knowledgeable to accurately assess such information, thus leading to unnecessary stress to the patients [McCollough CH]. Taking all these factors into consideration, we evaluated patient doses in patients who undergone more than 3 CT guided RFA procedures.

Material and methods

A wide range of CT guided interventions are performed in the CT department of Konstantopoulio general Hospital such as biopsies, RFA's, abscess drainages and nephrostomies. Patients who underwent more than 3 RFA procedures were included in the study. During the study period (June 2003-June 2009), 15 patients underwent more than 3 RFAs. Seven of the patients were treated 7 times, 6 patients were treated 4 times, 1 patient was treated 6 times and finally 1 patient was treated 11 times. The X-ray machine used was a single-slice helical CT scanner without CT fluoroscopy capability (Prospeed SX Power, General Electric Medical Systems, Milwaukee; WI, USA). All procedures were performed by an operator with more than 25 years of experience in various CT guided interventional techniques. To evaluate the lesion's immediate response to ablation and to identify any potential complications, dual-phase dynamic contrast-enhanced CT was performed. Follow-up was performed including dual-phase dynamic contrast-enhanced CT at 1, 3 and 6 months after RFA.

Using the DICOM data, the Dose Length Product (DLP), the effective dose (E), the patient skin dose profiles, as well as the peak skin dose (PSD) for both the bottom and upper patient skin surface were calculated for each RFA using appropriate software developed for this purpose. For each patient, cumulative DLPc, effective dose (Ec) and PSDc were also calculated from the sum of the respective figures of each individual procedure. More specifically, to calculate the cumulative PSDc, the skin dose profiles of each procedure were projected on the same scale using anatomical points for reference and then they were summed up.

Results

Mean age of patient sample was 71 ± 9 years. DLPc, Ec and PSDc range values were 5569-9642 mGycm, 83-173 mSv and 774-3364 mGy, respectively. Median E and PSD for each RFA were 35 mSv and 400 mGy. More analytically, Table 1 shows the radiation doses in terms of E and PSD for 4 patients having 3 RFAs. Table 2 shows the radiation doses in terms of E and PSD for a patient that had 11 RFAs. The age of each patient is also presented. The youngest patient (53 years) underwent the biggest number of RFA techniques (11 in total) having 2 lesions treated. The results varied greatly even for the same patient depending on the treatment site, the number of lesions, or lesion size. Total Ec and PSDc also varied immensely, depending not as much on the number of RFAs performed but mainly on the difficulty of each one. Indeed in our study, the larger dose values were observed for patient 1 who had 3 RFAs and not for the patient 5 who had 11 RFAs. Patient 1 had one anatomical location treated (PSDc reached approximately 3.4 Gy), whereas patient 5 had 2 anatomical locations treated (PSDc reached 0.95 Gy in the one anatomical site treated and 1.6 Gy in the other).

Table 1. Radiation doses in terms of effective dose (E) (in mSv) and Peak Skin Dose (in mGy) are shown for 4 patients having 3 radiofrequency ablations (RFA). The last row presents total E and PSD values.

	Patient 1 (75y)		**Patient 2 (77y)**		**Patient 3 (76y)**		***Patient 4 (59y)***	
	E	**PSD**	**E**	**PSD**	**E**	**PSD**	**E**	***PSD***
1	39	849	50	701	31	402	34	*988*
2	64	1404	36	449	27	290	15	*415*
3	70	1548	28	584	35	397	33	*522*
Total	***173***	***3364***	***114***	***1273***	***92***	***1071***	***83***	***1505***

Table 2. Radiation doses in terms of effective dose (E) (in mSv) and Peak Skin Dose (in mGy) are shown for the patient that had 11 radiofrequency ablations (RFA). The last row presents total E and PSD values.

Patient 5 (53y)		
N	**E**	***PSD***
1	26	*280*
2	44	*422*
3	42	*392*
4	33	*287*
5	40	*250*
6	33	*236*
7	44	*93*
8	35	*216*
9	36	*215*
10	29	*163*
Total	***112***	***1666***

Discussion

A major drawback of the study at this stage is that the estimation of radiation dose (PSD and E values of each RFA) requires the manual extraction and inputting of all DICOM data in the software developed for this purpose in our department. This results in a very time consuming estimation of PSD and E, making the process extremely difficult to do in routine. Initial results, however, indicate that radiation doses can be rather high. It was not possible to compare with other studies because no similar data existed until the completion of this paper. Despite the high E, it should be noted that these repeated procedures are usually carried out in patients where malignant tumours are already present and their medical condition is life threatening. On the other hand, it is not uncommon to have young patients undergoing multiple RFA procedures and one must take this into account when attempting to treat them. Just for comparison purposes, a routine abdominal CT imparts approximately 8-10 mSv to the patient, which is four times less than a typical CT guided RFA.

Conclusions

Initial results show that high radiation doses are associated with repeated RFA and there is definitely need to monitor in more detail these procedures. It is important to note that the particular techniques could not be prohibited on the grounds of radiation risk. The reasons are both prolongation of life expectancy and improvement of the quality of life. These factors certainly justify small radiation risks. However, the fact that high amount of radiation dose can be accumulated must be taken into account when attempting to treat young patients.

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Dose-area-product to effective dose in interventional cardiology and radiology

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Abstract

The implementation of the European Directives introduced a number of new tasks to the radiology departments. It was stated that the determination of radiation doses are an important issue in the framework of radiation protection of the patient. And special attention is given to high-dose procedures, like interventional cardiology (IC) and radiology (IR).

Radiological departments are legally obliged to register dose-area-product (DAP) values for every patient undergoing IR and IC procedures. A large national multi-centre project on dose evaluations for IC and IR performed a few years ago, showed that centers with similar average DAP-values, could still result in significant different average effective dose values. Simply register DAP-values is therefore not always the only and most adequate tool for optimization at IC and IR. The additional calculation of effective dose could enable medical physicists to determine and evaluate dose values which will more connect to radiation risk evaluation, if necessary.

In the past conversion coefficients (CCs) from DAP to effective dose have been calculated systematically for different anatomical regions and projections for conventional radiological procedures. The use of these published CCs, however are not appropriate for the calculation of effective dose for IC and IR. The irradiated field sizes deviate from those in conventional radiology. Moreover, the requested CCs according to the beam qualities used for these complex procedures are not included in the published conversion tables. In the framework of patient dose optimization, however, there is a need to the availability of systematic tables with CCs who will allow the calculation of the effective dose for the complete offer of IC and IR procedures.

In this paper is described how the CCs are calculated for the most common and used radiation fields in IC and IR. The systematic tables with CCs can be obtained at the Federal Agency of Nuclear Control (www.fanc.fgov.be).

Introduction

The implementation of the European Directives (97/43/Euratom) into the Belgian legislation introduced a number of new tasks to the radiology departments. It was stated that the determination of radiation doses are an important issue in the framework of radiation protection of the patient. And special attention is given to high-dose procedures, like interventional cardiology (IC) and interventional radiology (IR).

The need of dose auditing and patient dosimetry is emphasized in relation to optimization of radiological procedures. Moreover, radiological departments are legally obliged to register dose-area-product (DAP) values for every patient undergoing IR and IC procedures. As these procedures were considered as a priority, the Belgian Federal Agency of Nuclear Control financed a very large national multi-centre project on dose evaluations for IC and IR procedures a few years ago (Bosmans 2006). From this project, however, we learned that only DAP registration for optimization purposes is not always adequate. Measurements in almost 20 hospitals showed that centers with similar average DAP-values, could still result in significant different average effective dose values. This was caused by a different use in copper filtration during the procedures. With complex interventional procedures the risk of deterministic skin damage to the patient exists. To prevent this, the use of additional filtration is recommended. At new modern equipment, the amount of filtration is automatically introduced by the system, depending on the procedure and the patient. More and more also the use of other types of filtration is investigated for these kinds of procedures.

Simply register DAP-values is therefore not always the only and most adequate tool for optimization at interventional procedures. The additional calculation of effective dose could enable medical physicists to determine and evaluate dose values which will more connect to radiation risk evaluation, if necessary.

The effective dose can be calculated by multiplication of the registered DAP-values and appropriate conversion coefficients (CCs). In the past such coefficients have been calculated systematically for different anatomical regions and radiation projections for conventional radiological procedures. The use of these published CCs, however are not appropriate for the calculation of effective dose for IC and IR procedures. The irradiated field sizes and regions deviate from those in conventional radiology. Moreover, the requested CCs according to the beam qualities used for these complex procedures are not included in the published conversion tables. In literature (Kemerinck et al. 1999), (Schultz et al. 2003), (Kemerinck et al. 2003), (Kicken et al. 1999) some CCs can be found for specific interventional procedures, calculated according to the need of the specific study. In the framework of patient dose optimization, however, there is a need to the availability of systematic tables with CCs who will allow the calculation of the effective dose for the complete offer of IC and IR procedures, if needed.

In this paper is described how the CCs are calculated for the most common and used radiation fields in interventional cardiology and interventional radiology. The systematic tables with conversion coefficients can be obtained from the authors or at the Federal Agency of Nuclear Control (www.fanc.fgov.be).

Material and methods

Choice of Monte Carlo code

The Monte Carlo code that is used in this project is *MCNP-X (v 2.5.0)* (Pelowitz et al. 2005). This code has been used frequently in medical physics and allows reliable dose calculations for photon radiation sources. The calculations are performed by 2 institutes: the department of medical Physics of the University of Ghent (UGent) performed the necessary calculations for the IC procedures and the Belgian Nuclear Research Centre (SCK•CEN) performed all calculations for the IR procedures. The same and most recent libraries, containing all cross sections and material data, are used by both institutes.

X-ray source definition

An important input parameter for the Monte-Carlo simulations is the X-ray source design. Different possibilities exist to model an X-ray source in the *MCNP-X* environment. A small intercomparison with a simplified geometry showed that the approach of both institutes resulted in the same values.

With respect to the definition of the X-ray spectrum, the "IPEM-78 – Catalogue of Diagnostic X-ray Spectra" (Cranley 1997) was used. This publication provides a valuable software tool for generating X-ray spectra based on parameters kVp, filtration, anode angle and kV signal ripple. In clinical practice a large variety of X-ray spectra are being used. For the interventional applications in this project, kVp-values typically range from 60 to 130 kVp. Filtrations can be based on a single aluminium filtration (2.5 – 6 mm Al) or on a combination with copper filtration (0.1 – 0.9 mm Cu). As it is practically impossible to simulate all possible kV/filtration combinations for all clinical projections, an alternative approach is used within the project.

We have decided to use X-ray spectra definitions based on half-value layers (HVL). In fact, different kVp/filtration combinations may result in the same HVL-values as illustrated in figure 1.

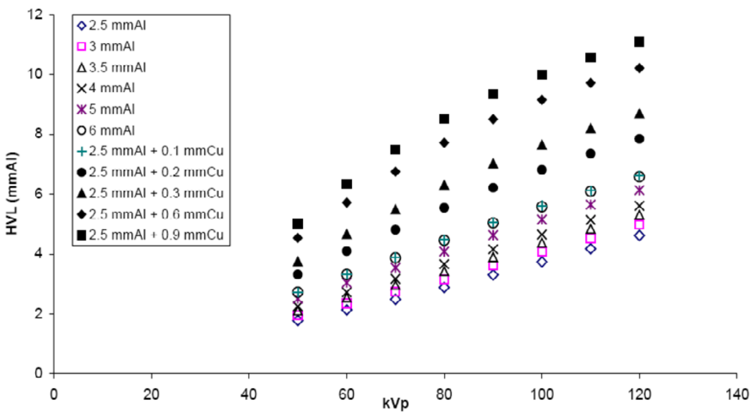

Fig. 1. **Half-value layer values for different kVp and filter combinations.**

Therefore, simulations based on a HVL-range that is clinically relevant would be interesting. In order to test the feasibility of this approach, Monte Carlo calculations were performed on a mathematical anthropomorphic phantom using different kVp/filtration settings – all resulting in the same HVL. The latter simulations showed that DAP-to-effective dose conversion coefficients simulated with the "HVL method" deviated maximum 5% of the values simulated with the exact spectrum.

For interventional radiology procedures, an anode angle of 14° was used, whereas for interventional cardiology procedures the spectra are generated based on an anode angle of 9°.

X-ray fields and projections

The X-ray field sizes and projections are other important factors to be taken into account for the simulations. In table 1 and 2, an overview of typical clinical settings is given for IR and IC, respectively.

Table 1. X-ray fields and projections considered for interventional radiology procedures.

	Application	Projection	Field size at image intensifier (cm)
1	Head	LAO[1] 45°	28
2	Head	RAO[2] 45°	28
3	Head	PA[3]	28
4	Head	LLAT[4]	28
5	Head	RLAT[5]	28
6	Neck	LAO 45°	28
7	Neck	RAO 45°	28
8	Neck	PA	28
9	Thorax	LAO 45°	28
10	Thorax	RAO 45°	28
11	Abdomen	LAO 45°	40
12	Abdomen	RAO 45°	40
13	Abdomen	PA	40
14	Abdomen	LLAT	40
15	Abdomen	RLAT	40
16	Pelvis	LAO 45°	40
17	Pelvis	RAO 45°	40
18	Pelvis	PA	40
19	Upper legs	PA	40
20	Lower legs	PA	40

[1]LAO: Left Anterior Oblique / [2]RAO: Right Anterior Oblique
[3]PA: Posterior-Anterior / [4]LLAT: Left Lateral
[5]RLAT: Right Lateral

Table 2. X-ray fields and projections considered for interventional cardiology procedures.

	Projection RAO/LAO	Projection CRAN[1]/CAUD[2]	Field size at image intensifier (cm)
1	RAO 30°	CAUD 25°	17
2	RAO 30°	CAUD 0°	17
3	RAO 30°	CRAN 25°	17
4	LAO 45°	CRAN 25°	17
5	LAO 45°	CAUD 0°	20
6	LAO 45°	CAUD 25°	17
7	LAO 90°	CAUD 0°	17
8	LAO 0°	CAUD 25°	17
9	LAO 0°	CAUD 0°	20
10	LAO 15°	CAUD 0°	17
11	LAO 30°	CAUD 0°	17
12	RAO 30°	CAUD 0°	20

[1]CRAN: cranial / [2]CAUD: Caudal

Choice of anthropomorphic phantom

In view of the new recommendations of ICRP (ICRP 2008) and the definition of additional radiation sensitive organs (salivary glands, adipose tissue, connective tissue, extra thoracic airways, heart wall and lymphatic nodes) for the calculations of the effective dose, the choice of an anthropomorphic phantom was not straightforward. Current mathematical phantoms are not appropriate for calculating the revised definition of the effective dose as they do not contain these 'new' organs. The choice of a voxel-phantom, for which a larger amount of organs are segmented, seemed more appropriate. At the start of the project, the standard ICRP voxel phantoms were not available. Other appropriate and available phantoms with standard dimensions are the MAX06 and FAX06 phantoms (Kramer et al 2006). These are very detailed phantoms constructed from voxels of 1.2 x 1.2 x 1.2 mm³. The total MAX06 phantom consists of 474 x 222 x 1359 = 143.004.582 voxel elements. All necessary organs are present and realistically segmented. However, when this phantom is converted into a format suitable for *MCNP-X*, input files are created from 20 to 30 MB. Such large input files require a lot of computer memory and some test calculations demonstrated that this memory capacity is not available in both UGent and SCK•CEN. Hence, the calculations could not be performed with these phantoms.

A large family of voxel phantoms is available at the Helmholtz Zentrum München – German Research Center for Environmental Health and we decided to test the Golem (Zankl, Wittman 2001) and Laura phantoms. These phantoms have body characteristics similar to the reference persons. Golem is constructed from voxels of 2.08 x 2.08 x 8.0 mm³. His height is 176 cm and weight 68.9 kg. Laura is constructed from 1.875 x 1.875 x 5.0 mm³ voxels. Her height is 167 cm and weight 59 kg. A cross section of both phantoms is given in figure 2. Both phantoms have a realistic number of voxels that can be handled by *MCNP-X*. Test runs showed that CPU time of maximum 130 minutes are needed for the transport of 10E06 particles (for F6 and *F8 tallies) on the computer clusters of UGent and SCK•CEN, resulting in relative errors for the organs in the radiation field lower than 1%.

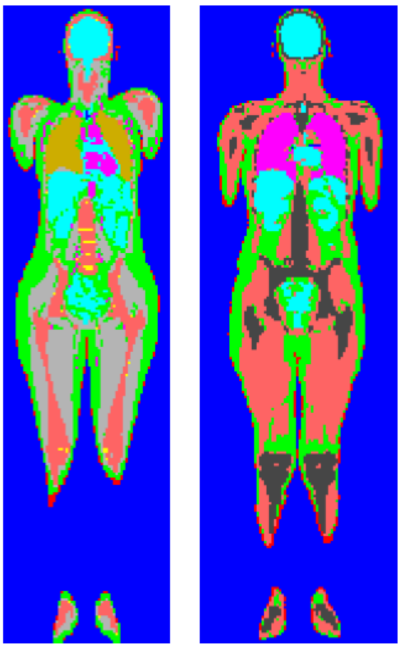

Fig. 2. Cross sections of the LAURA (left) and GOLEM (right) voxel phantoms.

As red bone marrow and bone surface are not segmented within the Golem/Laura phantoms, correction factors to the mean skeleton dose were calculated based on the material composition and density of red bone marrow, yellow bone marrow and cortical bone structures throughout the human body (Zankl et al. 2007). For the gall bladder and small intestine no distinction is made between wall and contents in the Golem phantom. Golem does not have breast (glandular tissue) and salivary glands and both phantoms do not have oral mucosa nor lymphatic nodes. The dose to the oral mucosa was approximated by the dose to the tongue and the dose to the lymphatic nodes was approximated by that to other distributed tissue, like muscle or adipose tissue.

Results

For interventional radiology, the following spectra are considered:

- Head: from 80 to 100 kVp in steps of 10 kVp
  from 3 mm Al to 6 mm Al in steps of 1 mm
  from 0 to 0.3 mm Cu in steps of 0.1 mm

This resulted in a HVL range from 3.5 – 8.5 mm Al

- Neck: from 60 to 100 kVp in steps of 10 kVp
  from 3 mm Al to 6 mm Al in steps of 1 mm
  from 0 to 0.3 mm Cu in steps of 0.1 mm

This resulted in a HVL range from 2.5 – 8.5 mm Al

- Thorax, abdomen and pelvis: Idem as Neck
- Legs: from 60 to 80 kVp in steps of 10 kVp
  from 3 mm Al to 6 mm Al in steps of 1 mm
  from 0 to 0.3 mm Cu in steps of 0.1 mm

This resulted in a HVL range from 2.5 – 6.5 mm Al

Spectra are considered in steps of 1 mm Al HVL for both the Golem and the Laura phantoms and each projection as given in table 1. This resulted in a total number of 262 calculations.

For interventional cardiology, the following spectra are considered:

- Thorax: from 60 to 130 kVp in steps of 10 kVp
  from 0 to 0.9 mm Cu for 2.5 mmAl
  from 0 to 0.3 mm Cu for 3 and 4 mm Al
  in steps of 0.1 mm Cu en 1 mm Al

This resulted in a HVL range from 2.5 to 11.5 mm Al.

For both phantoms and all projections considered in table 2, this resulted in a total number of 240 calculations.

Before the systematical calculation campaign started, a final intercomparison was performed between UGent and SCK•CEN. This should reveal possible differences in processing the voxel-based data. The simulations for this intercomparison was based on the same input file (PA thorax irradiation of the Golem phantom, field size 520 cm^3, 70 kVp, 4 mm Al and 10E06 particles). The results are presented in table 3.

The results show an excellent agreement between the SCK•CEN and UGent simulations with respect to the effective dose/DAP. Individual organ doses show small deviations smaller than 1%. The latter simulation run had a CPU time of 55 min for the SCK•CEN simulations and of 79 min for the UGent simulations.

The results of the systematic calculation campaign are given in organ dose/DAP for both phantoms separately and effective dose/DAP for both phantoms together using the new ICRP103 organ weighting factors. The conversion coefficients between DAP and effective dose are given in table 4a and b for interventional radiology and interventional cardiology, respectively. All tables, including the DAP-organ dose conversion coefficients can be obtained from the authors and from the Federal Agency of Nuclear Control (www.fanc.fgov.be).

Table 3. Results of an intercomparison for one typical input file.

	organ dose D/DAP SCK/CEN [Gy/Gy cm²]	organ dose D/DAP UGent [Sv/Gy cm²]
RBM	1,96E-05	1,98E-05
colon	7,56E-06	7,55E-06
lung	1,30E-04	1,31E-04
stomach	9,93E-05	9,92E-05
bladder	1,23E-07	1,22E-07
oesophagus	2,39E-04	2,39E-04
gonads	1,47E-08	1,45E-08
liver	1,04E-04	1,03E-04
thyroid	1,03E-04	1,04E-04
bone surface	4,46E-04	4,50E-04
brain	1,53E-06	1,54E-06
kidneys	1,87E-05	1,86E-05
salivary glands	4,41E-08	4,39E-08
skin	4,19E-05	4,20E-05
remainder:	8,99E-05	8,95E-05
effective dose/DAP	6.72 Sv/Gycm²	6.72 Sv/Gycm²

Table 4a. DAP to effective dose conversion coefficients for interventional radiology procedures.

	HVL (mm Al)						
Effective dose (mSv/Gycm²) (*)	**2,5**	**3,5**	**4,5**	**5,5**	**6,5**	**7,5**	**8,5**
Abdomen PA	0,089	0,136	0,169	0,200	0,224	0,259	0,283
Abdomen LAO 45°	0,074	0,102	0,135	0,155	0,179	0,202	0,219
Abdomen RAO 45°	0,068	0,100	0,117	0,140	0,161	0,187	0,204
Abdomen LLAT	0,205	0,259	0,320	0,360	0,380	0,428	0,456
Abdomen RLAT	0,118	0,158	0,195	0,217	0,244	0,266	0,286
Head PA	-	0,036	0,043	0,050	0,055	0,063	0,068
Head LAO 45°	-	0,036	0,045	0,052	0,059	0,065	0,070
Head RAO 45°	-	0,039	0,046	0,053	0,060	0,067	0,073
Head LLAT	-	0,045	0,056	0,063	0,066	0,075	0,081
Head RLAT	-	0,049	0,060	0,066	0,074	0,081	0,086
Upper legs PA	0,022	0,028	0,039	0,048	0,055	-	-
Upperlegs-knees PA	0,004	0,005	0,006	0,007	0,008	-	-
Knees-lower legs PA	0,004	0,004	0,005	0,006	0,006	-	-
Neck PA	0,044	0,065	0,079	0,093	0,103	0,118	0,129
Neck LAO 45°	0,056	0,071	0,088	0,099	0,111	0,123	0,131
Neck RAO 45°	0,061	0,079	0,092	0,105	0,118	0,130	0,139
Pelvis PA	0,095	0,134	0,164	0,190	0,210	0,236	0,254
Pelvis LAO 45°	0,057	0,077	0,100	0,116	0,133	0,149	0,161
Pelvis RAO 45°	0,059	0,084	0,099	0,118	0,134	0,154	0,167
Thorax LAO 45°	0,065	0,091	0,122	0,141	0,163	0,186	0,202
Thorax RAO 45°	0,052	0,079	0,093	0,113	0,131	0,156	0,171

Table 4a. DAP to effective dose conversion coefficients for interventional cardiology procedures.

Effective dose (mSv/Gycm²) (*)	HVL (mm Al)									
	2,5	3,5	4,5	5,5	6,5	7,5	8,5	9,5	10,5	11,5
LAO 0° CAUD 0°	0,117	0,185	0,245	0,285	0,313	0,360	0,394	0,425	0,453	0,466
LAO 0° CAUD 25°	0,102	0,168	0,228	0,269	0,293	0,343	0,379	0,411	0,441	0,457
LAO 15° CAUD 0°	0,123	0,196	0,261	0,305	0,334	0,386	0,424	0,458	0,490	0,505
LAO 30° CAUD 0°	0,135	0,199	0,255	0,294	0,321	0,364	0,396	0,399	0,451	0,464
LAO 45° CAUD 0°	0,172	0,245	0,307	0,350	0,387	0,429	0,462	0,493	0,519	0,531
LAO 45° CAUD 25°	0,105	0,169	0,218	0,252	0,278	0,314	0,342	0,367	0,389	0,400
LAO 45° CRAN 25°	0,134	0,206	0,262	0,301	0,331	0,372	0,403	0,431	0,455	0,468
LAO 90° CAUD 0°	0,118	0,183	0,241	0,280	0,309	0,352	0,385	0,414	0,440	0,453
RAO 30° CAUD 0°	0,176	0,254	0,320	0,365	0,404	0,448	0,484	0,517	0,545	0,555
RAO 30° CAUD 0°	0,173	0,250	0,316	0,361	0,399	0,444	0,480	0,513	0,540	0,551
RAO 30° CAUD 25°	0,110	0,167	0,216	0,250	0,276	0,312	0,341	0,366	0,388	0,399
RAO 30° CRAN 25°	0,140	0,210	0,271	0,312	0,345	0,389	0,423	0,454	0,481	0,492

Discussion

The ICRP standard voxel phantoms REX and REGINA were not available at the time of the study, so instead the phantoms GOLEM and LAURA are used. The dimensions of these phantoms are really close to the standard phantoms, but still some radiosensitive organs were missing, to be able to calculate effective dose according to the new ICRP103 organ weighting factors. One of the organs missing in the GOLEM phantom is the breast. In this study, the 'average' breast dosis for both phantoms for the effective dose calculation only comes from the breast of LAURA. We could consider a breast dosis of zero for GOLEM to calculate the average breast dose, but the authors believe that a larger error is made compared to only consider the breast dose from LAURA.

In general, we do not expect drastic changes for the conversion coefficients when these calculations would be repeated with the standard REX and REGINA phantoms.

Conclusions

The possibility to calculate organ doses and effective dose from online DAP measurements for interventional radiology and cardiology procedures can be a useful tool and completion for dose optimization purposes. In this project complete and systematic tables with organ dose and effective dose conversion coefficients are calculated from DAP values for these kinds of procedures.

Acknowledgement

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Artifacts found in computed radiographic system

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Abstract

Computed Radiography Systems (CR) are replacing the standard analog systems. In CR, However, as in other diagnostic imaging, artifacts can lower the image quality, simulate pathological lesions and mask the correct diagnosis. Artifacts similar to those found in analog systems, such as artifacts generated by the patient or operator´s mistakes, are very common. Further, there are artifacts related with the system itself: Image Plate (IP), the reader, to soft copy display or the printers. Some CR artifacts have still unknown reasons. *I*n this paper we present some artifacts found in CR systems and solutions to remove them are proposed.

Introduction

Quality Control Programs (QCP) are required for Computed Radiography systems (CR), which are replacing the conventional analog systems. CR systems are based on a new technology for storing the image in phosphorus photostimulable (Solomon et al 1991).The QCP has to evaluate the consistency of the equipment, and ensure the production of diagnostic images with excellent quality.

In this sense, the elimination of possible artifacts in the image is a task that requires the identification of all processes involved in producing the image, since the acquisition until the displaying on a monitor or on a medical radiographic film (Cesar, 2001, Willis et al 2004). The artifacts in the CR images can have their sources in the machine hardware (e.g., the X-ray equipment, grid, image plate, reader, detector), in the software (e.g., algorithms), or even coming from the patient (e.g., positioning, motion) (Seibert et al 2006). If there is a failure at any step of the image production, the final image will be compromised, requiring exam repetition, increasing the radiation dose in the patient (Lth, 2000). Hence, it is necessary to minimize the artifacts.

The main objective of this study is to assess the possible causes of some artifacts, and, when it is possible, propose solutions to remove them. It is important to mention that not all artifacts detected in the CR system have any known cause

Material and methods

Weekly the cassettes integrity are checked and their surface are cleaned with clorhexidine 0.2%. Monthly external cleaning is carried out using ethyl alcohol 99%.

The internal cleaning is only performed when the image plate (IP) is dirt, and this cleaning is done carefully according to manufacturer recommendation. The artifacts caused by dirty points in the IP are similar to those found in analog systems. The cassette identification number in the hard-copy makes the IP localization easier, and thus the cleaning task is quickly performed.

Artifacts related to X-ray examination start to appear soon after the new CR system started to be used. Thus, the physicians were instructed to save the images for future evaluation by the quality control team. Weekly, the physicians report the situation in which the artifacts appeared (patient standard, cassette used, the error message of equipment, etc.) and identify the possible causes of them, seeking for a solution

Results

Since the installation of the computed radiography system in our institution, September of 2006 medical images artifacts were found. The artifacts sources are extensive and may be related to the image plate, the reader, the display, the printer or the operator of the system (Willis et al 2004)..

Figure 1 is an example of a ghost image. Ghost images are the registered image that remains in the IP, even after the image is read and the IP is erased by the reader system. It is related to the erasure cycle efficiency of the reader (Seibert et al 2006). This type of artifact can occur when an improper processing menu is selected, resulting in incorrect histogram normalization and dynamic range scaling. Other causes of ghost images are miss-positioning the object, collimation errors that can occur in high scatter situations, and unusual anatomic variations that confuses the algorithms that identify the useful image information on the detector (Cesar, 2001). The solution could be the use of a smaller IP and to collimate the beam to the specific region of interest of the body.

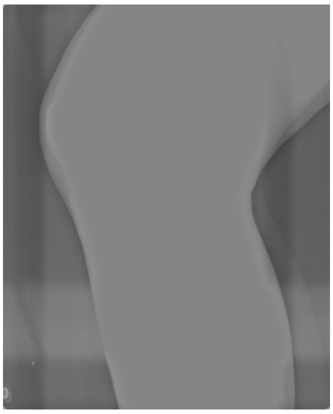

Fig. 1. Ghost image.

However, after working about 4 years with CR systems from the same manufacturer, in two different institutions, we have seen that there is more to learn about this problem. For example, in institution A we detected problems with a worker that did not correctly collimated the beam, and we have never had this kind of artifact.

Nevertheless, in institution B, we had this artifact everyday, with a worker that used the smallest IP practicable and collimates the beam, with adequate dynamic scale (without overexposure).

It is important to mention to the manufacturer that they need to analyze their equipment, not only trying to prove that the failure is in the workers that make the examinations. Of course, the employees should always be directed to comply with all requirements, minimizing the chances of artifacts.

In Figures 2 and 3 shown the appearance of two overlapping exams. This can occur if the latent image on the IP was not complete erased, either by a failure of the reader, or by the incorrect selection of the erasure mode by the technologist. This situation is also called double exposure, when two exposures occur without reading and erasing the first one (Willis et al). Other cause for this artifact is the selection of the incorrect algorithm to read the IP.

Fig. 2. Overlapping examination of foot and shoulder.

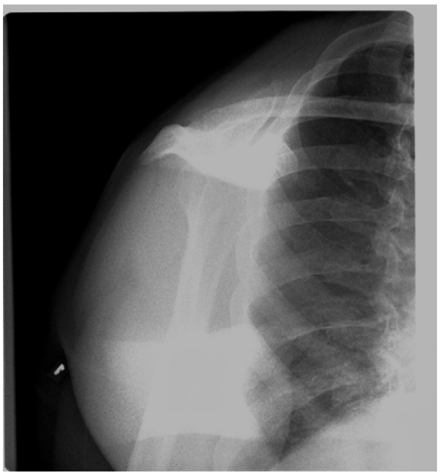

Fig. 3. Overlapping examination.

The system has the algorithms pre-determinate to read the IP, which means that the digital image for CR system is pre-processed. For each anatomical examination there is a specific algorithm which is prepared to read the signal from that anatomical region. If the physician that made a chest exam, but selected the algorithm for pelvis, for example, the images could be like those in figures 2 and 3. Sometimes this kind of problems is correlated with Figure 1.

Other artifact are small black dots, Figure 4. Image points are formed in IP, which causes dark spots on the image acquired later. Images were sent to the system's manufacturer and they informed that these artifacts occurs in tropical countries, where the air has a much higher relative humidity.

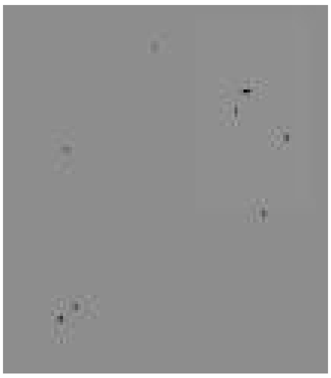

Fig. 4. Artifact due to high air humidity.

It also occured the appearance of objects from previous examination, as in Figures 5 and 6. In Figure 5, can be seen the lead marker (red arrow) used to identify the right side of the patient on an exam of the knee. On an exam of the chest, performed after the examination of the knee, the marker appears on the portion of trachea, and it's also possible to observe subtly the anatomical borders of the first examination (blue arrow). Figure 6 shows a lateral chest examination, where can be visualized the needle used in the urography examination done before.

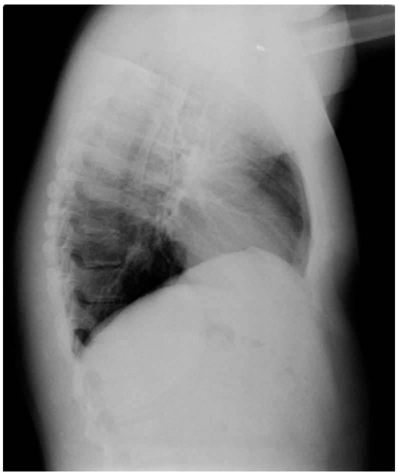

Fig. 5. Residual signal on the image from the lead marker.

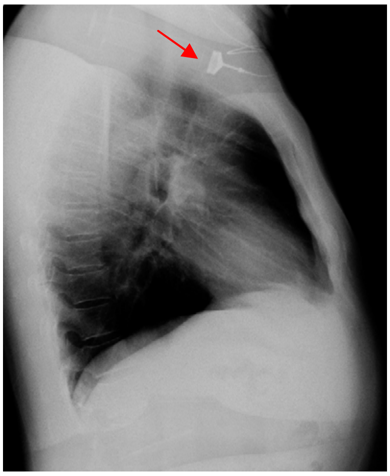

Fig. 6. Residual signal on the image from the urography examination.

It was also detected the presence of white lines in some images of X-ray examinations in both the horizontal direction, as shown by the red arrow in Figure 7, as in the vertical direction, evidenced by the red arrow in Figure 8. Artifacts in this pattern may be caused by dirt on the deflectors mirrors of the reader and should therefore be required the cleaning of the reader.

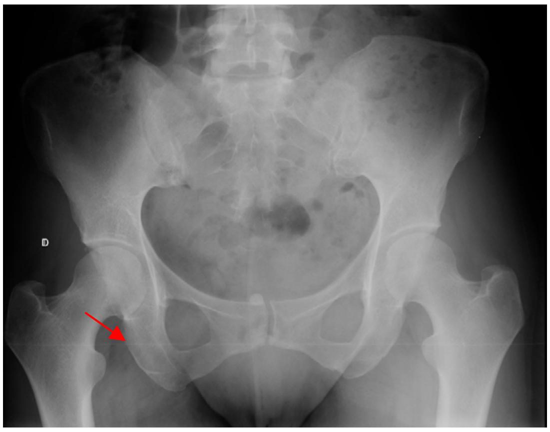

Fig. 7. Image with horizontal pattern of line.

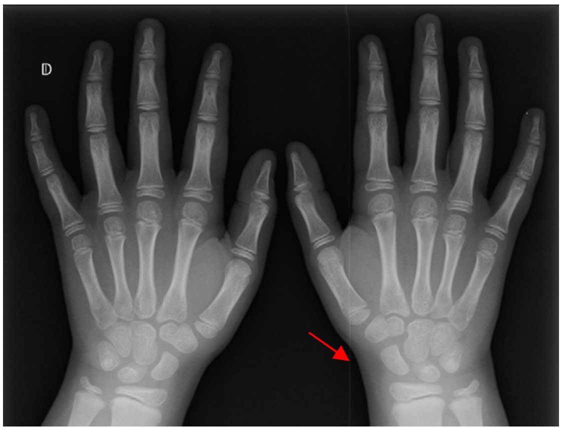

Fig. 8. Image with vertical pattern of line.

Another situation observed during the clinical routine was the appearance of artifacts due to the presence of backscatter radiation. The non-exposed cassette was stored in one side of the cabinet which is located next to the bucky of the X-ray equipment. We verified that when examinations were performed in the bucky and when the stored cassettes in the cabinet are exposed, the examination appear with artifacts, as shown in Figures 9 and 10. The corrective measures in this case was the exchange of the local storage of cassettes and perform the test of influence of other sources of radiation, noting that the local was suitable for the storage, not having been occur this artifact later.

Fig. 9. Image with artifact due the backscatter radiation.

Fig. 10. Image with artifact due the backscatter radiation.

Discussion

Images with artifacts that occur due to operators errors of the CR system, such as inadequate positioning, improper collimation, under or overexposure were shown. To avoid these artifacts an appropriate training period to the worker helps to avoid faults that are not committed during the examinations.

In addition, the staff should know the equipment and the system with which they work to know how to identify the error messages and possible causes of failures. To ensure the production of images with a diagnostic quality, the equipment must be properly calibrated and subjected to a rigorous routine of quality control, including regular tests and verification of the professionals routine.

Conclusions

As the production process of the image in computed radiography system differs from the analog system, this new technology also creates the appearance of new artifacts, without eliminating, however, the appearance of artifacts already known.

In addition, a number of factors affect the image quality in computed radiography, because this modality has distinct processes for the production of the image: data acquisition, processing, IP erasing, viewing and printing and/or storage the image. No device or professional is immune to produce unacceptable images. Therefore, the technologists need appropriate training to use effectively the new technologies and produce images with diagnostic quality. It's the responsibility of the quality control team reporting and taking the corrective measure when artifacts appeared in medical images in order to try minimizing or eliminating the causes of them.

And the manufacturer needs to analyse their equipments failures more seriously, not only trying to prove that the failure is the professional who makes the examinations. To realize the routine test of reports is very important, meanwhile, a bad operation of the reader may interfere in the total process.

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Patient dosimetry investigations of a dental panoramic unit and a digital volume tomograph

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Abstract

Patient dosimetry measurement methods were investigated in dental radiology comparing irradiations in a panoramic x-ray unit and a new digital volume tomograph DVT. Results of thermoluminescence dosimetry methods were validated by ionisation chamber dosimetry. Free-air and various in-phantom measurements were carried out in the rotating x-ray fields using maximum adult patient examination settings. Determination of absorbed dose was made in a cylindrical PMMA phantom and in an anthropomorphic Alderson-Rando head phantom. Reference radiation fields at the Dosimetry Laboratory Seibersdorf were used for calibration and determination of x-ray energy correction factors for both detector types. Resulting free-air and in-phantom dose values were up to several mGy whereas effective dose estimates were about 8 µSv and 81 µSv for panoramic radiography and DVT imaging, respectively.

Introduction

Radiation protection of patients during medical exposure is regulated throughout Europe (Council Directive 97/43/Euratom, 1997). The corresponding Austrian regulation necessitates the determination of individual patient dose values or of relevant parameter settings by the holder of authorisation. National diagnostic reference levels were established for a number of clinical x-ray imaging techniques except for dental radiographic applications. Panoramic machines are routinely used in dental radiology applications using a thin vertical beam and synchronously rotating radiographic source and film cassette. New DVT (digital volume tomography) scanners are based on CBCT (cone beam computed tomography) technology for 3D x-ray imaging. The number of DVT systems dedicated to maxillofacial imaging has increased for several years providing different source and detection technologies (Ludlow, 2008). Conventional CT (computed tomography) imaging is also used for various examination modalities in dental practice but was not considered in the current patient dosimetry investigations.

In this study free in air and various in-phantom dose comparisons were carried out in the rotating x-ray fields of a panoramic unit and a DVT using adult patient examination settings, see Figures 1 to 3. Thermoluminescence detectors (TLDs) distributed inside an anthropomorphic phantom was used for determining equivalent dose to organs and effective dose. The best way of calibrating TLDs is within the x-ray radiation field which is to be investigated using a reference detector such as an ionisation chamber. The drawback of this method is the time and effort and the problem of inhomogeneous spatial field distributions as would be expected for rotating x-ray scanners. One of the goals of this study was the introduction of appropriate energy correction factors for calibrating TLDs in reference ^{137}Cs gamma radiation fields.

Fig. 1. Alderson-Rando phantom positioned inside the panoramic unit (left) and slice 7 with four TLD positions 29 to 32 (right).

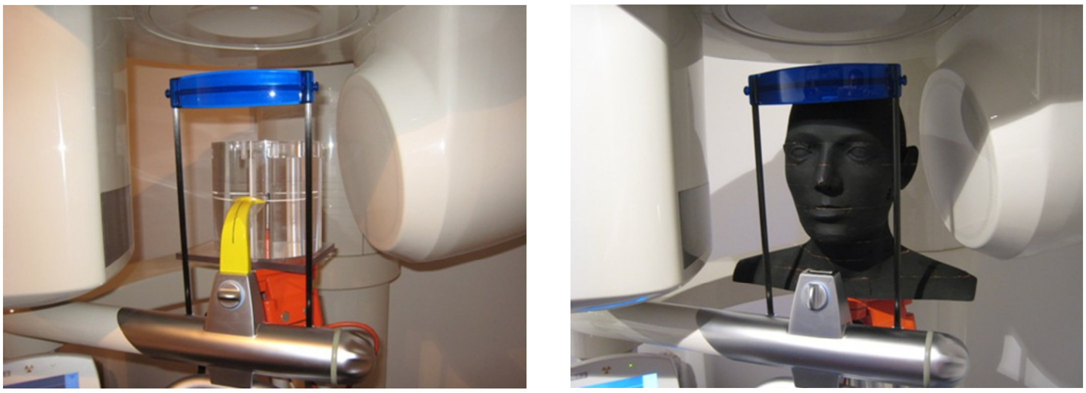

Fig. 2. Phantoms positioned inside the DVT, cylindrical PMMA phantom (left) and Alderson-Rando head phantom (right).

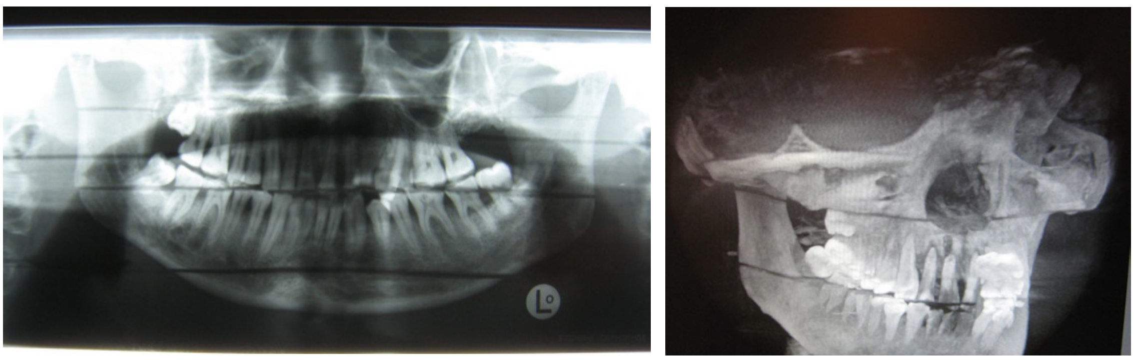

Fig. 3. X-ray images of the Alderson-Rando phantom by the panoramic (left) and DVT unit (right). The air gap between the slices of the phantom can be clearly seen in both images.

Materials and methods

Two x-ray facilities with rotating tube and detector geometries were investigated: a conventional panoramic x-ray unit (Sirona ORTHOPHOS Plus DS) and a new digital volume tomograph (Sirona GALILEOS). The DVT provides a 204° rotating pulsed beam with 200 x-ray images detected by an image intensifier of 21.5 cm diameter within about 14 seconds. A tungsten anode and a total equivalent filtration of at least 2.5 mm Al equivalent were stated by the manufacturer for both units. The "maximum adult" patient settings were used at the DVT, i.e. a tube voltage of 85 kV and a current-time product of 35 mAs. For panoramic radiographs also known as orthopantomograms the maximum tube voltage of 90 kV was chosen with a resulting 130 mAs current-time product.

Two different phantoms were used: a cylindrical PMMA phantom and an anthropomorphic tissue-equivalent Alderson-Rando phantom. The PMMA phantom is a standard CTDI head phantom with a diameter of 16 cm, with a central hole and with four periphery holes 1 cm below the surface. The Alderson-Rando phantom consists of 2.5 cm thick slices but only the first eleven slices of the head and neck region were used. A total of 55 selected locations for TLDs were available at slice 2 to 11. The phantoms were mounted on a stand inside the scanners using bite blocks to achieve preferably reproducible positioning similar to a real patient, see Figures 1 and 2.

Two different detector types were used: a 0.125 cm^3 thimble ionisation chamber (PTW 31010 with UNIDOS electrometer) and passive 3.2 x 3.2 x 0.89 mm LiF:Mg,Cu,P (Thermo Fisher Scientific TLD-700H) thermoluminescence detector chips. Dose results are based on calibration factors and correction factors determined at the Dosimetry Laboratory Seibersdorf traceable to the national standards of the BEV (Austrian Federal Office of Metrology and Surveying). Ionisation chamber readings were corrected for influences of the air density using a mobile temperature and pressure detection device (Thommen HM-30). TLD glow-curves were read out on a 3500 Harshaw manual reader with planchet heating system and WinREMS readout software using an optimized time-temperature-profile (Hranitzky et al. 2008), see Figure 4.

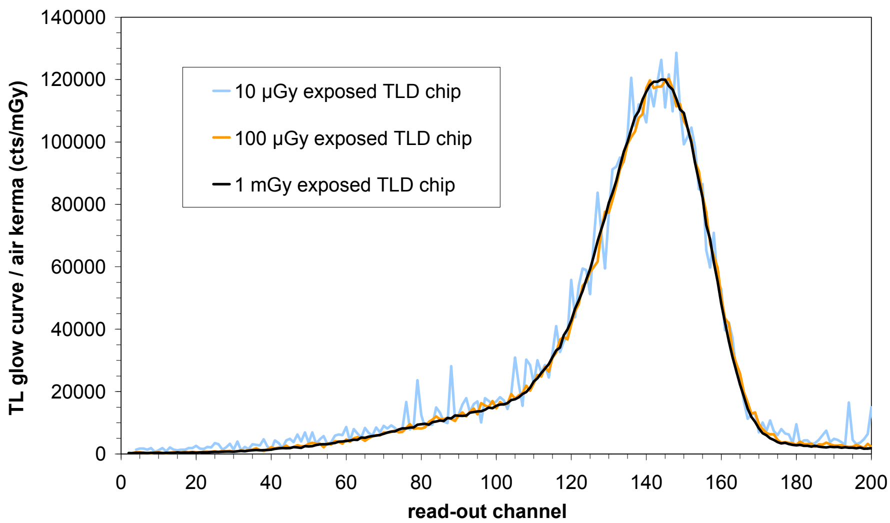

Fig. 4. Examples of TL glow curves at air kerma levels between 10 µGy and 1 mGy normalized to the corresponding air kerma. Read-out channels are distributed within a read-out time of 26.7 s.

TLD dose calculation, air kerma free in air and absorbed dose to water, is based on summing up the whole glow curve signal resulting in a single *TL* value. The dosimetry model for determining dose D_i of a single chip number "i" is based on a simple linear equation:

$$D_i = N \cdot k_Q \cdot (k_i \cdot TL_i - TL_{bg}) \qquad (1)$$

Individual sensitivity correction factors k_i and an average background signal TL_{bg} were considered. The group of calibration TLDs were irradiated at reference conditions in a ^{137}Cs calibration facility at 1 mGy air kerma free in air. X-ray energy correction factors also called radiation quality correction factors k_Q were determined by a diagnostic reference x-ray radiation quality RQR-7 (IEC 61267 Ed2, 2005) with a tube voltage of 90 kV and a first half-value layer of 3.25 mm Al.

Results

TL dosimetry

The relative standard deviation of the mean of TLD dose values were between 1% and 2% using at least 5 TLD chips closely put together in the free-air holder or in the PMMA phantom. TLD calibration was carried out for each measurement using a group of about 10 TLD chips resulting in calibration factors *N* of about 2.1 x 10^{-7} mGy/cts (±3%). The uncertainty estimate is dominated by the influence of the reader stability over several hours of readout time. The radiation quality correction factors k_Q were determined by the response values at 90 kV reference x-ray radiation. Resulting TLD correction factors averaged over five measurements are 1.11 ± 3% and 1.22 ± 3% for air kerma and absorbed dose determination, respectively. The "high" values of the correction factors are due to the energy dependence of the TLDs using a calibration at ^{137}Cs photon energy of about 662 keV. Details on k_Q determination can be found elsewhere (Halda, 2008). For each measurement a group of at least 10 unirradiated TLDs was read-out resulting in an average background TL signal TL_{bg} of about 2 µGy. All measurements starting from TLD annealing to TLD read-out were carried out within a single day. Generally the influence of background subtraction on calculated dose values was negligible. An uncertainty contribution due to linearity and coefficient of variation for low dose values could be neglected in this study.

Ionisation chamber dosimetry

The ionisation chamber calibration was performed in a certified ^{60}Co gamma radiation beam of therapeutic dose rate level at 5 cm depth within a water phantom. Therefore the indication of ionisation chamber is in terms of absorbed dose to water taking into account an air density correction with respect to reference conditions. For measurements free in air a conversion coefficient of 0.92 ± 1% form absorbed dose to air kerma was determined for the ionisation chamber in the ^{60}Co calibration field. A radiation quality correction factor of 0.96 ± 1.5% was determined for 90 kV reference x-ray radiation free in air averaged over seven measurements. The value is consistent with the energy dependence of the ionisation chamber published by the manufacturer for both quantities, air kerma and absorbed dose. Therefore the value of 0.96 was used as radiation quality correction factor for all measurements of the panoramic unit and the DVT. For absorbed dose measurements in the PMMA phantom an additional correction factor for the displacement of the effective point of measurement of the air filled ionisation chamber was taken into account. Based on the dose gradient of the 90 kV x-ray radiation field at the measurement position inside the phantom an effective point of measurement correction factor of 0,98 ± 2% was estimated. The maximum value of the air density correction factor for environmental influences during ionisation chamber measurements was 1,06 ± 0.5% due to a temperature of 24°C and an air pressure of 968 hPa. The uncertainty estimate was based on repeated comparisons of the indicated temperature and pressure to the stationary calibrated reference instruments of the laboratory over a period of one month.

Dosimetry results free in air and in PMMA phantom

The total measurement uncertainty was about 3% and 4% for ionisation chamber and TLD dose results, respectively. 5 detectors chips were closely put together at the same measurement position. All measurements were performed twice within two months for both radiographic units, see table 1. Only one air kerma free in air value is given for the panoramic unit because the first of two measurements was carried out for testing purposes.

Table 1. Comparison of air kerma free in air and absorbed dose in the centre of the cylindrical PMMA phantom, for the panoramic unit (left) and the DVT (right).

PANORAMIC	TLD	ionisation chamber	(TLD-IC)/IC
K_a (mGy) free in air	3.94 (±4%)	3.21 (±3%)	23%
D_w (mGy) in PMMA phantom centre position	1.06 (±4%)	0.98 (±3%)	9%
	0.92 (±4%)	0.94 (±3%)	-2%

DVT	TLD	ionisation chamber	(TLD-IC)/IC
K_a (mGy) free in air	3.73 (±4%)	4.01 (±3%)	-7%
	3.68 (±4%)	3.97 (±3%)	-7%
D_w (mGy) in PMMA phantom centre position	3.18 (±4%)	3.02 (±3%)	6%
	3.12 (±4%)	3.09 (±3%)	1%

Differences between the TLD results and the ionisation chamber results are within several percent. Especially for the air kerma results in the panoramic unit bigger deviations can be seen mainly due to the difficulties in positioning detectors consecutively at the same measurement point. Due to the broad beam geometry in DVT resulting differences were smaller as compared to the thin beam x-ray field of the panoramic unit. As can be seen, air kerma values were quite similar in both radiographic units. For the panoramic unit, the air kerma value depends strongly on the actual measurement position. There is no static rotation axis of the panoramic x-ray unit but its path is curved to follow the shape of the jaw. The central area is exposed up to three times by the fan-shaped radiation beam (Kaeppler, 2007). Using a stack of TLDs a full width at half maximum FWHM of 3.5 mm of the panoramic x-ray beam was determined free in air, which is smaller than the diameter of the ionisation chamber detection volume. Within the phantom a broader beam can be expected due to scattering effects.

Table 2. Comparison of the absorbed dose in the cylindrical PMMA phantom at three different positions: at the centre, at the rear (neck), and at a side position.

DVT	TLD	ionisation chamber	(TLD-IC)/IC
***D* w (mGy)** in PMMA phantom centre position	3.18 (±4%)	3.02 (±3%)	6%
***D* w (mGy)** in PMMA phantom rear position	5.49 (±4%)	5.58 (±3%)	-2%
***D* w (mGy)** in PMMA phantom side position	3.63 (±4%)	3.84 (±3%)	-5%

Only for the DVT x-ray radiation field, measurements at the different positions within the cylindrical PMMA phantoms were carried out. The maximum dose of about 5.5 mGy was found at the neck position. Differences between the available positions can be explained by the path of the x-ray tube which rotates about from one side to the other at the rear of the head with the detector at the opposite side close to the teeth and jaws of the patient. Differences between TLD and ionisation chamber results are again within several percent and therefore consistent within measurement uncertainties.

Alderson-Rando dosimetry results

Table 3. Comparison of the absorbed dose values at TLD positions distributed in slice 2 to 11 of the Alderson-Rando phantom for the panoramic unit and the DVT, respectively.

	***D* w (mGy)**	
	PANORAMIC	**DVT**
slice 2		
pos 1	0.02 (±3%)	0.20 (±3%)
pos 2	0.02 (±1%)	0.24 (±2%)
pos 3	0.02 (±3%)	0.21 (±2%)
pos 4	0.02 (±3%)	0.25 (±2%)
pos 5	0.03 (±10%)	0.29 (±1%)
pos 6	0.02 (±1%)	0.25 (±4%)
pos 7	0.02 (±1%)	0.22 (±8%)
pos 8	0.02 (±3%)	0.28 (±2%)
pos 9	0.02 (±5%)	0.24 (±2%)
pos 10	0.02 (±2%)	0.19 (±7%)
slice 3		
pos 11	0.02 (±9%)	0.30 (±1%)
pos 12	0.04 (±1%)	0.44 (±6%)
pos 13	0.04 (±1%)	0.51 (±2%)
pos 14	0.04 (±3%)	0.46 (±2%)
pos 15	0.06 (±16%)	0.63 (±4%)
pos 16	0.05 (±2%)	0.55 (±5%)
pos 17	0.06 (±6%)	0.78 (±5%)
pos 18	0.05 (±1%)	0.60 (±6%)
pos 19	0.03 (±1%)	0.45 (±4%)
slice 4		
pos 20	0.05 (±2%)	0.71 (±4%)
pos 21	0.07 (±2%)	1.18 (±7%)
pos 22	0.09 (±2%)	1.74 (±4%)
pos 23	0.07 (±5%)	1.26 (±4%)
pos 24	0.13 (±4%)	1.62 (±1%)

	***D* w (mGy)**	
	PANORAMIC	**DVT**
slice 5		
pos 25	0.27 (±4%)	2.58 (±1%)
pos 26	0.25 (±3%)	2.63 (±2%)
pos 27	0.27 (-)	2.57 (±3%)
slice 6		
pos 28	0.35 (±1%)	3.15 (±1%)
slice 7		
pos 29	0.73 (±1%)	5.10 (±1%)
pos 30	0.59 (±1%)	4.69 (±1%)
pos 31	0.69 (±1%)	5.15 (±1%)
pos 32	0.27 (±1%)	1.49 (±1%)
slice 8		
pos 33	0.17 (±4%)	4.03 (±2%)
pos 34	0.15 (±4%)	4.10 (±3%)
pos 35	0.14 (±6%)	3.64 (±2%)
slice 9		
pos 36	0.04 (±9%)	0.69 (±1%)
pos 38	0.06 (±1%)	0.91 (±10%)
pos 39	0.06 (±3%)	0.88 (±6%)
slice 10		
pos 41	0.02 (±3%)	0.27 (±1%)
pos 43	0.03 (±3%)	0.30 (±3%)
pos 45	0.02 (±1%)	0.35 (±6%)
pos 47	0.03 (±4%)	0.37 (±3%)
slice 11		
pos 49	0.01 (±3%)	0.12 (±1%)
pos 51	0.01 (±1%)	0.18 (±5%)
pos 54	0.02 (±5%)	0.18 (±6%)
pos 55	0.02 (±1%)	0.19 (±6%)

Two TLDs were positioned inside a hole at selected sites inside the Alderson-Rando head phantom but not all of the 55 positions were loaded with TLDs. In table 3, average absorbed dose values of two TLDs are presented. For both x-ray fields, panoramic and DVT, the maximum dose is located within slice 7, which includes the teeth (see Figure 1). The maximum absorbed dose values are 0.73 mGy and 5.2 mGy for the panoramic and the DVT x-ray field, respectively. It has to be noted that a conversion from absorbed dose to water to absorbed dose to tissue was omitted in this study. Values in brackets of table 3 are relative standard deviations of the two TLD dose values representing a measure of repeatability. The average relative standard deviation of two TLDs is about 3.5% for both units and for all dose levels. Generally absorbed dose values of the DVT irradiation is a factor of about 10 higher compared to the panoramic irradiation at corresponding locations within the Alderson-Rando head phantom.

Equivalent dose to organs and effective dose

The quantity effective dose can be used for estimating nominal risk of stochastic effects associated with radiological procedures. Effective dose is reported to be of value for comparing doses from different diagnostic procedures and for comparing the use of similar technologies (ICRP, 2007). Effective dose E can not be measured directly but is calculated as the sum of equivalent dose values of a number of organs and tissues H_T multiplied by the corresponding tissue weighting factors w_T (ICRP, 2007). Due to a radiation weighting factor of 1 for photon radiation fields, values of equivalent dose in tissue are equal to the average absorbed dose to tissue values. Absorbed dose to tissue D_T was calculated by the mean value of TLD measured absorbed dose D and weighted according to the ratio of the organ mass in the head to the total organ mass in the body, see table 4. A discussion of the mass weighting factors for large organs, especially skin, muscle, and red bone marrow, can be found elsewhere (Looe et al., 2008). Average absorbed dose values were calculated for four radio-sensitive organs, i.e. brain, salivary glands, red bone marrow, and thyroid. Other radio-sensitive organs and tissues in the head and neck region according to (ICRP, 2007) were neglected because of missing TLD positions.

Table 4. Comparison of measured average absorbed dose values D, organ and tissue dose weighted by the organ mass ratio H_T, and effective dose results $E = \Sigma H_T w_T$ for the panoramic (left) and DVT (right) exposure.

PANORAMIC	D (mSv)	H_T (mSv)	$H_T w_T$ (µSv)
brain	0.07	0.07	0.7
salivary glands	0.39	0.39	3.9
red bone marrow	0.08	0.01	1.7
tyhroid	0.03	0.03	1.4
		effective dose (µSv)	**8**

DVT	D (mSv)	H_T (mSv)	$H_T w_T$ (µSv)
brain	0.79	0.79	8
salivary glands	4.03	4.03	40
red bone marrow	0.90	0.15	18
tyhroid	0.37	0.37	15
		effective dose (µSv)	**81**

Conclusions

Resulting free in air and in-phantom dose values were up to several mGy whereas effective dose estimates were about 8 µSv and 81 µSv for panoramic and DVT imaging, respectively. Although dose values of both x-ray units were directly faced in this study it has to be stressed that the results of the two units and parameter settings are not comparable. Maximum adjustable x-ray tube voltage and tube current-exposure time product values were chosen for both units. The higher absorbed dose and effective dose values, in average about a factor of 10, of DVT irradiations as compared to panoramic x-ray irradiations can not be generally applied to other units or parameter settings. Especially image quality and clinical indications were not considered in this study.

A maximum dose value of 0.65 mGy inside an Alderson-Rando phantom was published for panoramic radiography (Cohan, 2002) for an Orthophos C scanner model. This value is close to the measured value of 0.73 mGy in this study. An average effective dose of 10 µSv for panoramic radiography in the UK was reported by the Health Protection Agency (NRPB, 2002). (Looe et al., 2008) reported effective dose values based on (ICRP, 2007) tissue weighting factors of 6 and 16 µSv at about 80 kV for a Philips and a Gendex panoramic unit, respectively. The effective dose value of 8 µSv in this study is in agreement to the published literature.

A maximum dose value of 4.1 mGy inside an Alderson-Rando phantom was published for a DVT 9000 scanner (Cohan, 2002). The maximum absorbed dose to water value of 5.2 mGy in this study is comparable to the reported dose value. The manufacturer of the GALILEOS DVT system stated an effective dose of 90 µSv in the manual based on published data (Ludlow, 2008). The calculated effective dose of 81 µSv agrees with the published results. It has to be noted that the locations of the TLD sites within the available phantom were not perfectly distributed according to the radio-sensitive organs of (ICRP-103, 2007). Additionally, skin, bone surface, oesophagus and remainder tissues and organs in the head and neck region, i.e. muscle, oral mucosa, and lymphatic nodes were not included in the current evaluations for effective dose. Nevertheless their contributions are expected to be rather low to the effective dose results. MCNP Monte Carlo simulations were carried out to validate the measured dose distributions and resulting effective dose values. Currently, simplified simulation models were investigated only for planar x-ray exposure (Halda, 2008). Future investigations are concentrated on simulations of ICRP reference computational voxel phantoms for radiation protection applications.

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Radiation dose in selective nerve root blocks

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Abstract

Objective: Selective nerve root blocks (SNRB) are useful in the treatment of radicular symptoms, due to the infiltration of large amounts of steroids around a particular nerve root resulting in reduction of pain. Fluoroscopic guidance is usually used to perform these procedures. The objective of this study was to measure the radiation dose received by these patients.

Materials and methods: The study included 24 patients undergoing SNRB using a fully digital fluoroscopy X-ray machine Philips Allura FD 20 installed in 2008 in Konstantopoulio Hospital. The following technical parameters were recorded: Kerma Area Product (KAP), cumulative dose (CD), fluoroscopy time (T) and total number of images (F). Patient data recorded were: Age (A), Weight (W) and Height (H).

Results: Median values of KAP, CD, T and F were 26.7 $Gycm^2$, 360 mGy, 3.4 min and 123 images, respectively. Ranges of KAP and CD were 1.8-218.4 $Gycm^2$ and 24-856 mGy, respectively. No correlation was found between KAP and F (0.12), as well as KAP and T (0.04). Furthermore, no correlation was found between CD and F (0.09) and moderate correlation between CD and T (0.56).

Conclusions: Radiation doses vary substantially in SNRB procedures. The fact that they can be repeated to the same patient in certain time intervals calls for more detailed investigation and close monitoring of patients.

Introduction

Selective nerve root blocks (SNRB) are useful in the treatment of radicular symptoms, due to the infiltration of large amounts of steroids and anesthetic drugs around a particular nerve root resulting in reduction of pain [Wagner, 2005]. They have long been a staple of anesthesiologists and orthopedic surgeons, but have been increasingly performed by interventional radiologists [Wagner, 2004]. It is a minimally invasive outpatient technique that can be performed with minimal sedation.

Fluoroscopic guidance facilitates greatly in the accuracy of needle placement, and thus the specificity of the examination. With injection of contrast media, the correct position of the needle in the nerve root sleeve and not in the nerve root itself is confirmed. Furthermore, it helps to confirm that the needle is not positioned in a vascular structure, a situation that could remove the injectate from the injection site and cancel any success of the SNRB [Fenton 2003]. The use of fluoroscopy, however, imparts a certain radiological burden to the patient. It should be noted that there are no data in the recent literature on the level of radiation dose received by the patient during SNRB using fluoroscopic guidance. Taking all these into account, the objective of this study was to measure and evaluate the radiation dose received by patients undergoing SNRB procedures.

Material and methods

SNRBs were performed by a single operator on a Philips Allura FD 20 large area flat panel angiography system installed in 2008 in Konstantopoulio Hospital. All procedures were executed by a single operator with more than 15 years in interventional techniques. SNRBs were performed with the patients in the prone position on the fluoroscopy table. The interventional radiologist is standing next to the patient so as to position the needle and the technologist is seated next to the radiologist so as to move the C-arm in the appropriate position and to facilitate the radiologist during the SNRB. The difference between this technique and other interventional techniques is that the radiologist must stand close to the back of the patient and therefore be very close to the X-ray machine and the scattering medium (patient body). More importantly, the hand of the radiologist is seen in the X-ray image during his attempt to accurately position the needle using the “barrel view” technique.

The following technical parameters were recorded: Kerma Area Product (KAP) and cumulative dose (CD) both of which are provided by the X-ray system, fluoroscopy time (T) and total number of images (F). As patient data are concerned, the following parameters were also recorded: Age (A), Weight (W) and Height (H) of the patient.

Results

The study included 24 patients undergoing SNRB of the lumbosacral anatomical region. The distribution of KAP and CD did not exhibit a normal distribution so the median, minimum and maximum values are estimated. These values are shown in Table 1 together with corresponding values of T, F, A, W and H. Large variation in results was found especially in KAP with max/min ratio of 121, whereas corresponding value for CD was 36. The correlation of KAP and CD was investigated with technical parameters (T, F) and patient data (W, A). The only correlation found (moderate) was between KAP and W (0.4). Furthermore, the only correlations of CD with collected data were for T (0.6) and W (0.4).

Table 1. The Selective Nerve Root Block (SNRB) technical and patient data are presented.

	W (kgr)	H (cm)	A (years)	F	T (min)	DAP (Gycm²)	CD (mGy)
median	78	168	72	124	3.4	26.7	360
min	55	155	30	6	1.0	1.8	24
max	117	195	88	254	10.3	218.4	856

It was not possible to compare with the literature since so similar study existed. There are a number of studies presenting data on SNRB and CT or CT fluoroscopy (Wagner 2004, Carlson et al 2001, Nawfel et al 2000)

Nawfel et al estimated a patient skin dose of 830 mGy when applying 80 seconds of CT fluoroscopy, by using a 90-mA and 120-kVp exposure. This value, however, so a theoretical calculation using the technical parameters, therefore being not specific for the SNRB. It must also be noted that the SNRB can be repeated to the same patient in certain time intervals.

Conclusions

The literature has practically no information on radiation dose levels regarding SNRB. Our study showed that radiation doses can vary substantially. Taking into consideration that: 1) they can be repeated to the same patient in certain time intervals and 2) the limitation of information on radiation dose, shows that more detailed investigation and close monitoring of patients should be done.

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Managing radiation dose in interventional radiology

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Abstract
The number of interventional radiology procedures has been shown to increase over time due to their clinical safety and efficacy. Nevertheless, these procedures may be related to elevated patient radiation dose. Stochastic risk management can be improved by monitoring the dose-area product and comparing it to available dose reference levels (DRL). On the other hand, deterministic effects can be managed by monitoring cumulative dose at the intervention reference point (D_{IRP}). The objective of this study was to provide interventionists a simple and reliable method to control risks associated with the use of ionizing radiation. Patient data for a seven-month period were collected and analyzed for hepatic embolization and cerebral angiography. Data included dose-area product values, fluoroscopy time, number of images obtained during the procedure and D_{IPR}. Patient dosimetry was performed by using high-sensitivity MOSFET dosimeters. The SPSS statistical program was employed for data analysis. Maximum skin dose reached 8.9 and 5.5 Gy during hepatic embolization and cerebral angiography, respectively. Good correlation was found between the MOSFET dosimeters and D_{IRP} during patient study. The 75% percentile of patient dose was calculated and compared with the national DRLs. D_{IRP} should be recorded during the procedure and included in patient's report file for post-procedure reference. Actions have to be taken towards dose optimization. Patient follow-up has to include check for recognized or suspected radiation injury.

Patient dosimetry during biventricular I.C.D. insertion

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Abstract

Recent years have seen a great increase in electrophysiological cardiac procedures number and radiological systems are constantly improved to obtain good image quality by upgrading technological performance. Even if patient benefits are clear, these procedures expose patients to X-rays for a long fluoroscopic time; the result is progressively higher patient radiation exposure, both in local skin areas (for example, the skin of the back) and in terms of effective dose and related risks.

In order to evaluate the mean patient dosimetric parameters, we have analyzed 30 biventricular implantable cardioverter defibrillators (ICD) procedures, measuring DAP (dose area product) during the entire procedure and registering all geometrical and physical data that are necessary to reconstruct, via a Monte Carlo simulation, the dosimetric quantities for ICD therapy.

For this reason, for each patient we have collected all X-ray spectrum data (such as kV, mA and time during fluoroscopy and fluorography) related to irradiation geometry (such as focus-to-patient distance, X-ray tube angulations, dimension of X-ray field of view) to obtain a "cumulative dataset" that permit to simulate the intervention.

Using a Monte Carlo code, we have evaluated the dose for each patient (both organ dose as effective dose, to obtain "dosimetrical indexes" for the procedure) to produce a statistical analysis that can be compared with other interventional therapies (for examples, hemodynamic procedures as stent implantation, angiography, percutaneous transluminal coronary angioplasty, etc.).

Even if the dose variability between patient data is great, primary due to the complexity of the ICD insertion, our results demonstrate that patient dose indexes for this particular electrophysiology procedures are not significantly different from the X-ray guided hemodynamic therapies.

Introduction

While several research studies focus on the analysis of the patient dose under hemodynamics procedures, fewer deal with electrophysiology RX guided procedures *[Cusma et al. 1999, Huyskens et al. 1995, Likckfett et al. 2004].*

For this reason, we have decided to investigate this issue and to proceed with the analysis of real patient cases at the Cardiovascolary Department of the University of Bologna.

Since electrophysiology procedures are very diversified and numerous, we chose the procedure that probably involves the highest patient dose values: the biventricular ICD (implantable cardiostimulator defibrillator) insertion for the therapy of heart failure.

In fact, this intervention needs longer fluoroscopy time than the implantation of a single chamber pacemaker for the insertion of multiple guides and three electrodes (the first one in the right atrium, the second one in the right ventriculum and the third in coronary sinus) to resinchronize the cardiac heartbeat.

As one of the scopes of our work is to present a dataset of EPs dose values directly comparable with other interventional procedures, the choice of dose indexes becomes fundamental.

Many authors have decided to focus their works on possible risks for specific areas of the patient body (for example, the skin), analyzing the so-called Maximum Skin Dose (MSD) as dose index for hemodynamic procedures *[Balter 2006, Van de Putte et al. 2000, Wittkampf et al. 2000].* This choice, related with the usage of detectors located on defined positions of the patient skin (such as thermoluminescent detectors or gafchromic films [*Chu et al. 2006*]), permits to estimate the X-ray energy on patient with good precision, but has a spatial limitation due to the detector dimensions (a possible solution is the use of large films located on the operating table, with an underestimation of dose in lateral projections).

In order to obtain not only a single organ data, we have decided to collect the DAP (Dose Area Product) values for every shot (i.e., projection) and reconstruct the irradiation geometry and patient anatomy via Monte Carlo simulation.

Material and methods

The X-ray system

The X-ray machine used in the operative room dedicated to EP procedures at Cardiovascular Department of the University of Bologna is a Sias Synthesis (*www.sias-spa.com/synthesis.htm*; nominal high voltage of 120 kV; half value layer of 3.8 ± 0.1 mmAl equivalent at 100 kV; image intensifier of 9 inches input, with different electronic zoom usable by cardiologists).

In this system, the X-ray tube and the image intensifier (I.I.) are mounted on a U-shaped arch that can rotate around the longitudinal and transverse axis of the patient.

Patient investigations are principally characterized by fluoroscopy, in order to define the guidance of catheters and their localization inside patient thorax, and sometimes by venography for the analysis of positions with high resolution and/or contrast media. During fluoroscopy, the X-ray tube works in continuous mode and uses the automatic exposure control (AEC) to set high voltage (i.e. kV_p) and current (i.e., mA).

The venography instead works in pulsed mode with current fixed at 120 mA and regulates with the AEC only the voltage in order to obtain a defined signal level on the image intensifier.

No changes are introduced during this study, but all the radiological parameters and procedures are executed by cardiologists as usal, in order to obtain significant and realistic data.

Patient group

The choice of patients suitable to represent the procedure had required the selection of candidates who had anatomical and pathological features comparable. In practice, starting from the list of all patients, our cardiologists selected a group of patients to obtain a sufficiently uniform distribution of physical parameters around values described by the so-called ”standard man” *[ICRP 103, 2007]*. This process led us to obtain a group of 30 patients whose characteristics are summarized in Table 1.

Table 1. Population overview.

	Mean	SD
Age (years)	61	12
Height (m)	1,75	0,09
Weight (Kg)	77	15
Avarage BMI (kg/m²):	26,4	-
Number of males	23	-
Number of females	7	-

Procedures

The biventricular implantation (BIV) procedure involves the insertion of catheters guided by fluoroscopy - normally using the so-called brachial access - and positioned by means of venography.

Due to the variability associated with the patient specificity, each operation was divided into radiological projections that compose it. A physicist has followed every procedure in the operating room recording all the information required for dosimetric evaluation of the patient. This allows to know the contributions of each shot to the total dose, and to compare the procedures between them. Data collected for each projection are divided into: X-ray tube parameters and geometric parameters.

X-Ray Parameters

The values of high voltage, current, time, frame numbers and collimation are collected from the console of the system. Since the AEC changes the parameters during the same projection depending on the workload, the values we save for each projection are the mean value of the data parameter presented in the console.

Geometric parameters

In order to know the radiation field investing the patient not only in terms of hardness (i.e., the capability to penetrate inside the body, described by X-ray spectrum) but also in terms of geometrical distribution on the patient body, the relative positions between the X-Ray tube and patient have been studied.

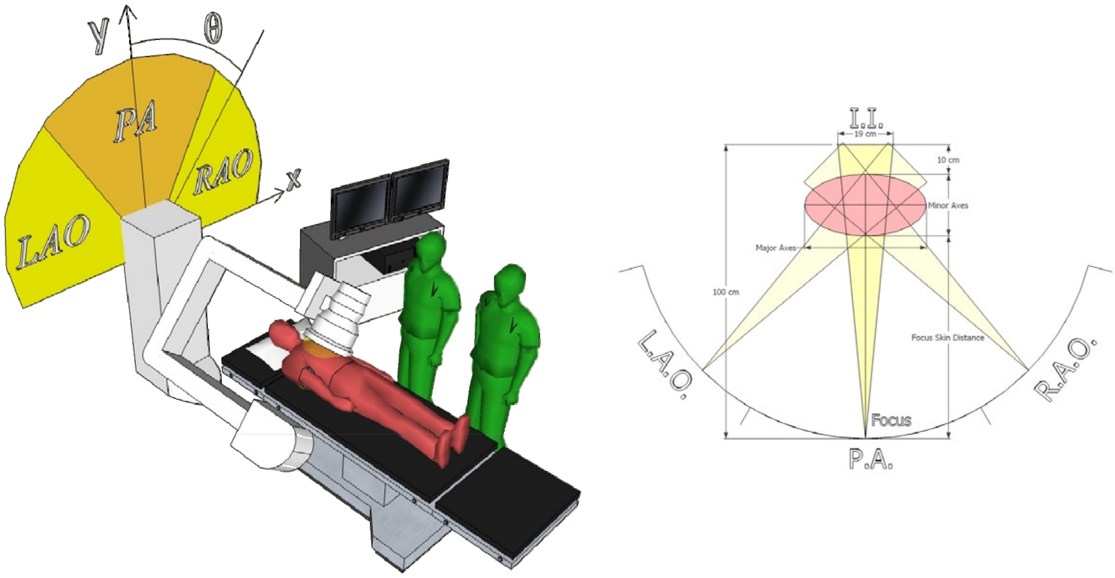

Fig.1-left: Schematic representation of the relative position of the subjects involved during the procedure.Fig 1-right: Representation of irradiation geometry and ICRU model for torax dimension.

The primary beam (which can be considered conical with the vertex on the X-ray focus) rotates around the patient's heart in the XY plane perpendicular to the axis of the patient Z. In angular coordinates, the focus position can be described by θ (theta angle, see Fig. 1), that is the rotation angle respect to the Y axis, and with the distance r. We decided to group the rotation angles based on the position of the tube with respect to the axis Y as in figure 1($\theta = 0 \pm 30°$ is collected as Postero - Anterior (PA) projections, while $-90° < \theta < -30°$ there are Right Anterior Oblique (RAO) projections and $30° < \theta < 90°$ are Left Anterior Oblique (LAO) projections).

Obviously, during the Monte Carlo simulations, this choice introduces an error in the determination of the organ volumes invested by primary X-ray beam as in the evaluation of the relative doses: we have estimated this error and decided to group the angles in order to not excede an uncertainty of 10%, with a computing time reduction of about 30%.

To evaluate the distance between the patient and the X-ray tube we used the model of the elliptical cross section described in ICRU 2005 (Fig.1 right). In that model the average value for our set of patients are min axis 21cm and max axis 42 cm. Each simulation was done with the patient's own geometrical parameters. An evaluation of the relative positions between patient and I.I., as between X-ray tube and I.I., permits to estimate for our sample of patients the FSD (focus skin distance) for every projection. On average, in our sample of patients, we estimate for the PA projection a FSD of 69,1 cm (SD=1,8 cm) while in RAO is 59.6 cm (SD=2.6 cm) and LAO is 67.1 cm (SD=2.8 cm)

Evaluation of dose indexes: organ and effective dose calculations

Starting from the geometrical and the radiological data collected, a commercial Monte Carlo software (PCXMC) was used: the simulation for every patient was divided in all the projection angles described (LAO, RAO, PA), using the tools that allow not only the definition of the X-ray beam shape (i.e., the radiological parameters as kV, mA, etc.) for every shot, but also a free adjustment of the X-ray beam dimension on phantom body.

Obviously, the Cristy and Eckerman phantom implemented in PCXMC was adjusted in height and weight in order to simulate the anatomy of every patients. For everyone, the dose contributions are being estimated both in terms of organ dose (as described by ICRP 103), both in terms of effective dose in order to compare these procedures with other interventional ones.

The Monte Carlo code simulate a number of stories of photons chosen to have a good ratio between statistics and calculation time: our goal is to have a simulation error for the primary organs around 0.2 %. This allow us to choose a number of stories equal to $N_s = 5 \cdot 10^5$ for each projection.

Results

Results can be divided in two parts: the first part concerns the procedure itself with the data collected in the operating room and the second part concerns the organs and the effective dose evaluated by Monte Carlo simulations.

Radiological data

The radiological data are summerized in Table 2. The work parameters of the two diagnostic phases (fluoroscopy and fluorography) are divided in geometric components as described above (RAO, LAO, PA).

Table 2. Median system setup parameters for each projection and measured DAP values.

		PA			RAO			LAO		
		Median	Min	Max	Median	Min	Max	Median	Min	Max
Fluoroscopy	Tube voltage (kV)	93	82	116	104	86	120	101	86	120
	Tube current (mA)	3,0	2,0	5,0	3,1	2,0	5,2	3,1	2,2	4,0
	Time (s)	630	144	2468	40	3	867	360	9	1231
	DAP (mGy×cm²)	23'378	2'937	115'763	1'415	86	28'271	15'891	200	64'366
	DAP ptc	57,5%	-	-	3,5%	-	-	39,1%	-	-
Fluorography	Tube voltage (kV)	81	63	108	91	73	110	88	72	120
	Numer of frame	295	27	876	119	26	968	282	20	1347
	DAP (mGy×cm²)	4'660	844	39'053	2'045	774	21'063	5'207	651	34'039
	DAP ptc	39,1%	-	-	17,2%	-	-	43,7%	-	-
Projection	DAP (mGy×cm²)	26'629	2'937	132'315	2'824	546	49'334	17'804	651	71'993
	DAP ptc	56,3%	-	-	6,0%	-	-	37,7%	-	-
Total	DAP (mGy×cm²)	54'426 (21'693 - 140'069)								

Organ and effective dose calculation

In this study we have focused our attention on the value of organ dose rather than the value of effective dose because our goal is not only to compare ICD implantation with other cardiac procedures, but to have dose indexes directly related to the energy absorption inside patient.

The simulation of each projection gives us the absorbed dose for each organ: the total organ dose is the sum for each patient of the dose absorbed in a specific organ. The distribution of dose values in our population does not follow a gaussian distribution (for example, see Fig. 2 that plots the lung dose of the population). This concerns that the statistical variation of the values are not centred on an expected value and so we treat the representative parameters under the term of median (and range) and not under term of average (and standard deviation).

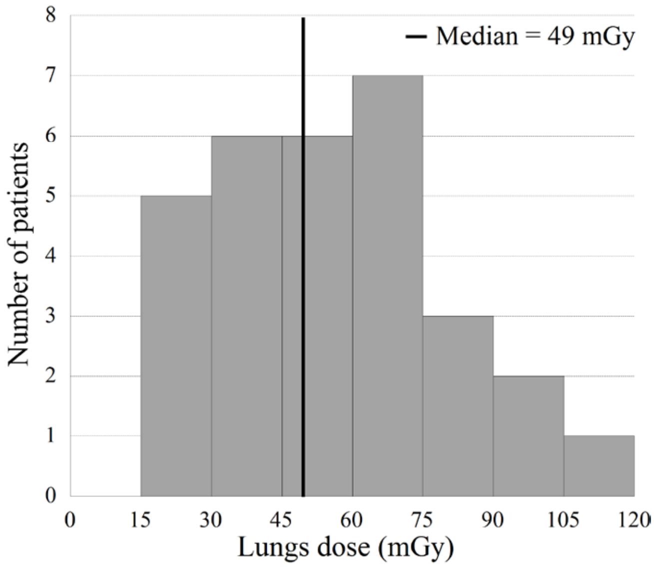

Fig 2. Histogram of the lung dose for our population.

We decided to divide the study of the organ dose in two parts: the skin dose and the internal organs dose. This choice is due to the observation that the region of the skin invested by the X-ray beam certainly receives the greater part of the energy, if compared with all the other organs.

Skin Dose

We associate all the skin doses calculated by PCXMC to the skin mass irradiated by primary X-ray beam. With this operation, we consider as skin organ not all the skin of the body but only the two regions (rear and front) irradiated by the primary beam. The proportion between the total skin area and the irradiated skin area gives us the dose absorbed by the skin. The total skin area is given by ICRP 89:

$$Area_{Skin}(m^2) = 0.0235 \cdot \left(Height(cm)^{0.42246}\right) \cdot Mass(kg)^{0.51456}$$

In this way we can evaluate the skin dose as it is summarized in table 3.

Table 3. Skin dose values evaluated for the different projections.

	Median (mGy)	Min (mGy)	Max (mGy)
PA	115,4	12,7	587,0
RAO	13,4	5,4	260,1
LAO	62,9	3,0	367,4
TOTAL	276,1	98,0	658,5

The skin dose values estimated by this method are comparable with other studies related to cardiac procedures *[Geise et al. 1999, Bor et al. 2009]*.

Dose for internal organ

The Monte Carlo software gives us the dose for each organ defined in ICRP 103. Because of the little region irradiated, there are only some organs in the primary beam or with a dose value reasonably different from zero. In table 4 we present the median values of the internal organ dose for our population while in table 5 we present the organ dose percentages divided in the two diagnostic phases (fluoroscopy and fluorography) and in the three geometrical projections (PA, LAO, RAO).

Table 4. Median organ dose for the BIV insertion procedure.

Organ dose (mGy)				
Organ	**Median**		**Min**	**Max**
Active bone marrow	16,7	(6%)	6,3	43,4
Adrenals	22,7	(8%)	9,3	78,2
Breasts	6,9	(2%)	1,8	14,9
Heart	46,2	(17%)	16,6	95,8
Liver	7,7	(3%)	2,6	22,0
Lungs	49,4	(18%)	16,4	108,2
Oesophagus	35,5	(13%)	13,8	84,4
Pancreas	10,6	(4%)	3,9	31,1
Skeleton	26,7	(10%)	9,7	64,3
Spleen	9,3	(3%)	2,5	32,2
Stomach	6,3	(2%)	2,0	16,2
Thymus	16,6	(6%)	0,0	0,0
Remainder organs*	25,4	(9%)	-	-
Median total dose	304,4	-	101,3	1281,4

Table 5. Median partial contribution of each projection to the organ dose. First column describes the percentage inside the projection, while second column the percentage related to the total organ dose.

Organ	PA		RAO		LAO		Fluoroscopy		Fluorography	
Active bone marrow	7%	(3,6%)	5%	(0,4%)	5%	(2,1%)	6%	(4,8%)	7%	(1,3%)
Adrenals	10%	(4,9%)	6%	(0,4%)	7%	(3,0%)	9%	(6,9%)	7%	(1,5%)
Breasts	3%	(1,3%)	2%	(0,1%)	2%	(0,8%)	2%	(1,9%)	2%	(0,4%)
Heart	15%	(7,2%)	15%	(1,1%)	19%	(8,5%)	17%	(13,5%)	16%	(3,2%)
Liver	3%	(1,4%)	5%	(0,4%)	1%	(0,6%)	2%	(1,8%)	2%	(0,5%)
Lungs	17%	(8,3%)	24%	(1,6%)	20%	(9,0%)	19%	(15,0%)	19%	(3,9%)
Oesophagus	13%	(6,5%)	12%	(0,8%)	12%	(5,4%)	13%	(10,3%)	12%	(2,4%)
Pancreas	3%	(1,7%)	2%	(0,2%)	4%	(2,0%)	4%	(3,1%)	4%	(0,7%)
Skeleton	10%	(4,9%)	9%	(0,6%)	7%	(3,3%)	9%	(6,9%)	10%	(2,0%)
Spleen	3%	(1,3%)	1%	(0,0%)	5%	(2,4%)	4%	(3,0%)	4%	(0,7%)
Stomach	2%	(1,0%)	1%	(0,1%)	3%	(1,3%)	2%	(1,9%)	2%	(0,4%)
Thymus	4%	(2,0%)	8%	(0,5%)	6%	(2,6%)	5%	(4,0%)	6%	(1,1%)
Remainder	9%	(4,2%)	9%	(0,6%)	9%	(3,8%)	9%	(7,0%)	9%	(1,8%)
Total	100%	(48%)	100%	(7%)	100%	(45%)	100%	(80%)	100%	(20%)

Effective dose

The estimation of the effective dose (as defined in ICRP 103) may be useful not only for estimating absolute risk associated with BIV implantation but rather permit a direct comparison with other cardiac interventions *[Wielandts et al. 2010, Martin 2007, Drexler et al. 1993]*.

The median value of calculated effective dose is 13.6 mSv in a range of (4.9÷35.1) mSv.

Figure 4 presents a comparison between the effective dose estimated in this study and E values for other common cardiac interventions executed inside Sant'Orsola University Hospital of Bologna *[Compagnone et al. 2005]*.

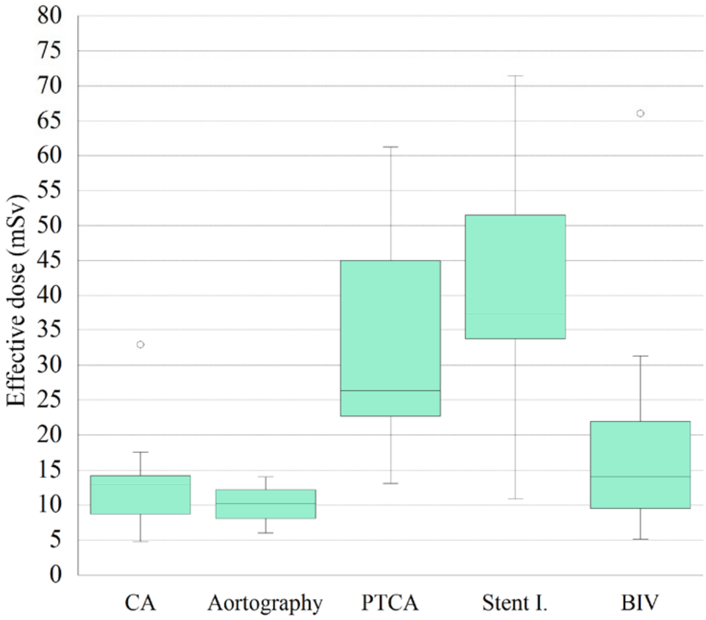

Fig 3. Comparison between effective doses evaluated for different cardiac RX-guided procedures: CA (coronarography); PTCA (Percutaneous Transluminal Coronary Angioplasty); STENT insertion and BIV

Conclusions

This paper has attempted to provide an indication of the patient dose levels associated to a specific electrophysiology procedure (biventricular ICD insertion) in order to have data that can aswer at the patient dose questions presented by many cardiologists *[Smith et al. 2008]*.

In fact, even if the EP procedures are not clearly defined as high dose procedures, organ doses and effective dose values seems to be comparable with – for example - angiographic ones.

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Cancer risk from medical radiation exposure: a *pro*gnosis-based *l*ifetime *a*ttributable *r*isk *a*pproximation (*PROLARA*)

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Abstract

Purpose: To evaluate the impact of the reduced life expectancy of patients (compared to a non-patient group with same age distribution) on their nominal risk of developing solid cancer from the diagnostic use of radiation.

Method: We define a "*pro*gnosis-based *l*ifetime *a*ttributable *r*isk *m*odifier" (*PROLARM*) as the ratio of risks for non-patient and patient, a dimensionless quantity which indicates how strongly the lifetime attributable risk (*LAR*) is reduced due to the patient's prognosis. We show that an approximation to this ratio can be calculated (named *PROLARA*) which depends only on patient's age at exposure and his/her life expectancy, but is independent of the exact choice of values for the baseline risk of cancer incidence (which varies across populations) and the excess relative risk (*ERR*) from radiation exposure (which remains controversial in the scientific community). *PROLARM* and *PROLARA* were computed for a cohort of n=4285 female patients with metastatic breast cancer, for which all necessary input data where available.

Results: *LAR* of solid cancer is significantly decreased for these patients: *PROLARM* > 20 for all ages at exposure ≤ 65 years. For any reasonable choice of function for *ERR*, the approximation *PROLARA* gives a lower estimate of the reduction in risk. The risk for a patient from the above cohort, exposed at age 50, is decreased by a factor of 29 (*PROLARM*) rsp. 27 (*PROLARA*). In other words, 20 mSv for a patient with metastatic breast cancer correspond risk-wise to less than 1 mSv for a healthy person of the same age.

Conclusion: A major portion of the total dose from diagnostic medical exposures does not constitute an additional cancer risk due to the poorer prognosis of patients compared to non-patients of same gender and age. Our new *PROLARA* concept allows an estimation of the reduction in risk for any pathology when the associated survival is known.

Introduction

Calculations of cancer risk from diagnostic medical use of ionizing radiation have resulted in predicted numbers of cancer cases ranging into the thousands for the most developed countries (Berrington de Gonzalez and Darby 2004, Brenner and Hall 2007).

Those predictions are based on applying the Linear-No-Threshold (LNT) model which may not hold against new radiation biological data, but nevertheless remains the foundation of current practice in radiation protection.

Even with the LNT model, however, the risk from diagnostic radiation exposure is significantly smaller than conventionally projected because those calculations generally neglect the fact that the exposure is not evenly spread among the population, not even among subgroups of a certain age range. Rather, the healthy part of the population will hardly be exposed at all, while most of the procedures, especially the ones with high effective doses, will be applied to patients who are seriously ill.

We have therefore investigated quantitatively the influence of the illness of one particular patient cohort, i.e. their prognosis or life-expectancy, on their nominal risk of developing solid cancer from any diagnostic radiation exposure, compared to the risk for healthy persons of the same age. The method is transferable to any other patient group.

Such calculations involve using values for the excess relative risk (*ERR*) of cancer incidence per unit dose for the exposed population and transporting them into lifetime attributable risk (*LAR*). As there is no generally accepted "best set" of such *ERR* values, we present a novel approach that avoids having to choose a particular *ERR* function.

Material and methods

Using the nomenclature of (Vaeth and Pierce 1990) and (Kellerer et al. 2001), the lifetime attributable risk *LAR* (of a radiation-induced malignancy to become clinically manifest) is a function of age at exposure, *e*, and given by

$$LAR^i(e) = \int_{e+L}^{a_{\max}} ERR^i(a)m(a)S(a)/S(e)\,\mathrm{d}a = \int_{e+L}^{a_{\max}} ERR^i(a)m(a)S'(a,e)\,\mathrm{d}a \qquad (1)$$

where *e* and *a* are age at exposure and attained age, respectively, *m*(*a*) is the baseline cancer incidence rate and *L* is the latent period. The *survival function*, i.e. the probability at birth to reach at least age *a*, is denoted by *S*(*a*). The ratio *S*(*a*)/*S*(*e*) is the conditional probability of a person alive at age *e* to reach at least age *a*. The superscript ^{i} denotes incidence (which is the quantity we are interested in, rather than mortality).
For patients (subscript $_P$), we have

$$LAR_P^i(e) = \int_{e+L}^{a_{\max}} ERR^i(a)m(a)S'_P(a,e)\,\mathrm{d}a = \int_{e+L}^{a_{\max}} ERR^i(a)m(a)P(a,e)S'(a,e)\,\mathrm{d}a \qquad (2)$$

with $P(a,e)$ relative survival of the patient population compared to a standard population of same gender and age.

We define the "prognosis-based lifetime attributable risk modifier" (*PROLARM*) as the ratio of risks for non-patient and patient, a dimensionless quantity which indicates how strongly the *LAR* is reduced due to the patient's lower life expectancy:

$$PROLARM(e) = LAR^i(e)\big/LAR_P^i(e) \qquad (3)$$

It would be desirable to have a quantity which is independent of the exact values of *ERR(a)*, because these remain controversial in the scientific community, and also independent of the baseline risk of cancer incidence, *m(a)*, which varies across populations. This can be achieved by defining the "prognosis-based lifetime attributable risk approximation" (*PROLARA*) which depends only on patient's age at exposure and his conditional life expectancy. With any of the generally accepted choices for *ERR(a)* such as the preferred functions from (NRC 2006) or (ICRP 2007), the following relation holds:

$$PROLARA(e) = \int_{e+L}^{a_{\max}} S'(a,e)\,\mathrm{d}a \Bigg/ \int_{e+L}^{a_{\max}} S'_P(a,e)\,\mathrm{d}a \le PROLARM(e) \qquad (4)$$

In words: the ratio of the integrals over conditional survival for normal and patient is always (slightly) lower than the ratio of the *LAR* integrals in eqn. (3), i.e. *PROLARA* as the ratio of the survival integrals gives a lower limit to the *PROLARM*.

Application to patient cohort

We computed *PROLARM* and *PROLARA* according to eqns. (3) and (4) for a patient cohort where all necessary input data where available. The group consists of n=4285 female patients with metastatic breast cancer. The following input data were used:

Baseline survival (life expectancy): The sex-specific survival function was taken from the latest available statistical data for Germany (www.destatis.de 2009). The data can be modeled by a Gompertz expression of type $S(a) = \exp(-c_1 \cdot \exp(c_2 \cdot a))$ with fitted parameters c_1=5.5E-05, c_2=1.13E-01. (Fig. 1; values differ from those in (ICRP 1991) due to today's longer life expectancy).

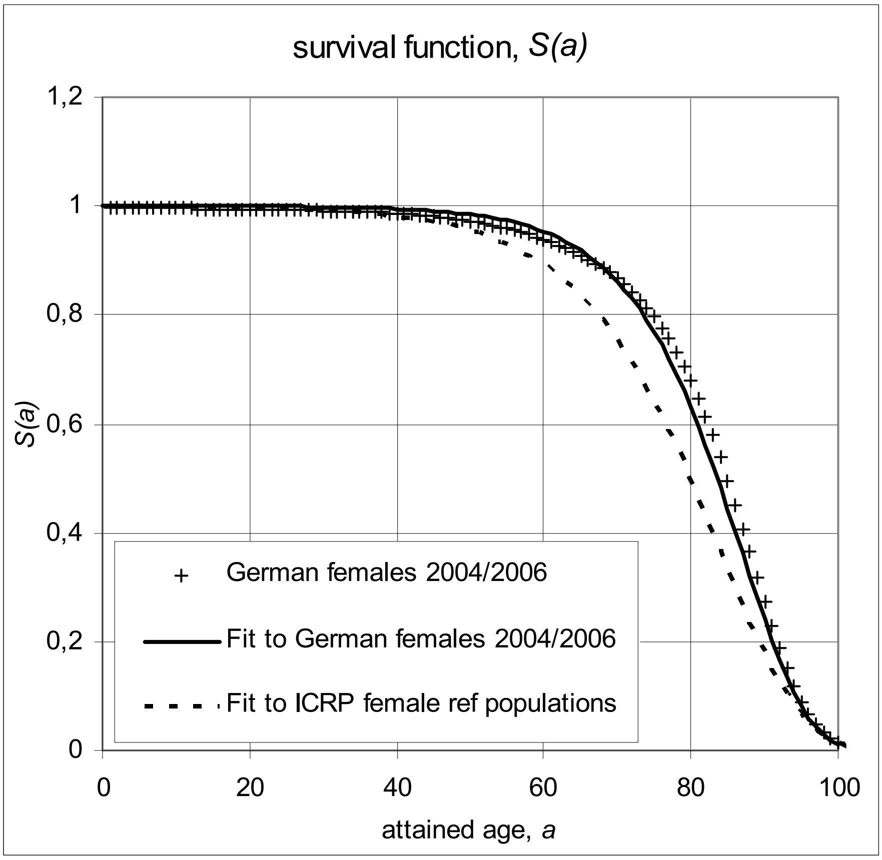

Fig. 1. Baseline survival function for women in Germany and fit to these data as well as fit to the 5 reference populations used in ICRP 60.

Baseline cancer incidence: Although there is no nationwide cancer registry in Germany, extrapolations of the available incidence data to the whole German population have been published (RKI 2008). They are listed for most of the ICD-10 cancer types and various age intervals. We used the data for all solid cancers except non-melanoma skin neoplasms (ICD-10 C00-97 except C44). Fitting a power function $m(a) = k \cdot (\, a / 60\,)^r \cdot \exp(\, -c \cdot (\, a / 60\,)^r\,) + b$ to those data yields parameters k=0.008, r=4, c=0.14 for females, while b, which corresponds to the rate at very young ages, was conservatively set fixed to b=0.0002. Fig. 2 shows the published data and the corresponding fit.

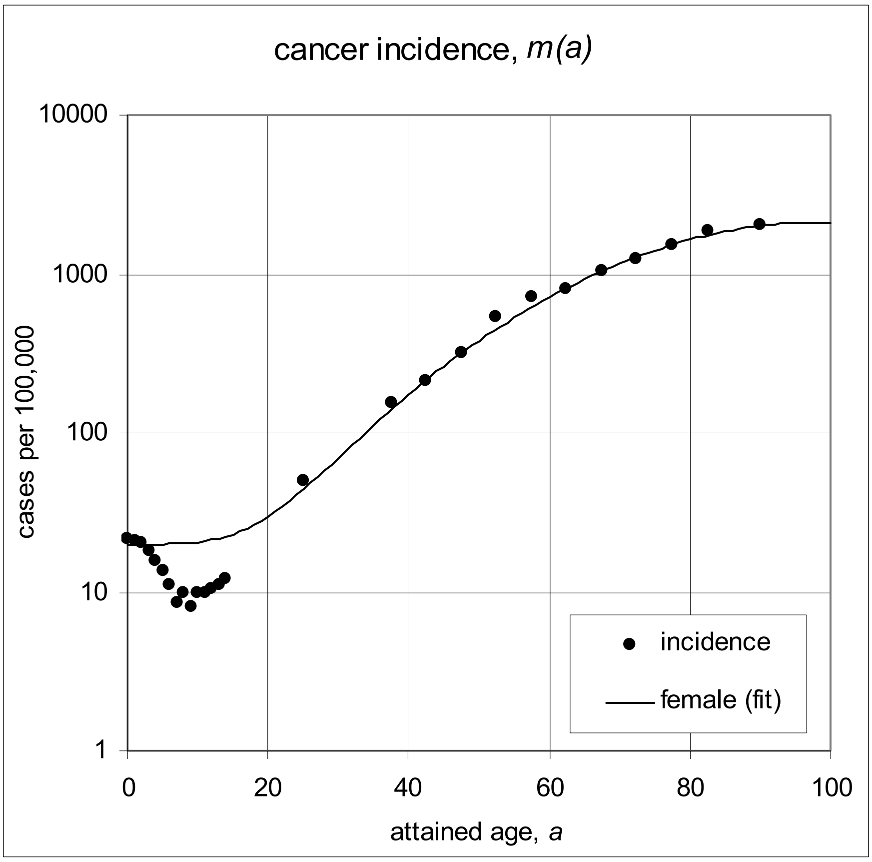

Fig. 2. Age specific cancer incidence for all solid malignancies except non-melanoma skin cancer in women in Germany, 2003. Cases per 100,000 in each age group.

Excess relative risk, ERR: We use the values published in BEIR VII Phase 2 report. The preferred function there for ages at exposure, e, greater than 30 years is of type $ERR(D,S,a) = \beta_S \cdot D \cdot (\eta \log(a / 60))$ where D is dose (Sv), β_M and β_F are the ERR/D for males and females exposed at age 30 at attained age 60, and η is the exponent of attained age. Subscript $_S$ denotes sex. For all solid cancer incidence (again excluding non-melanoma skin cancers), the fitted parameters are β_M=0.33, β_F=0.57, and η=-1.4. The result is displayed in Fig. 3.

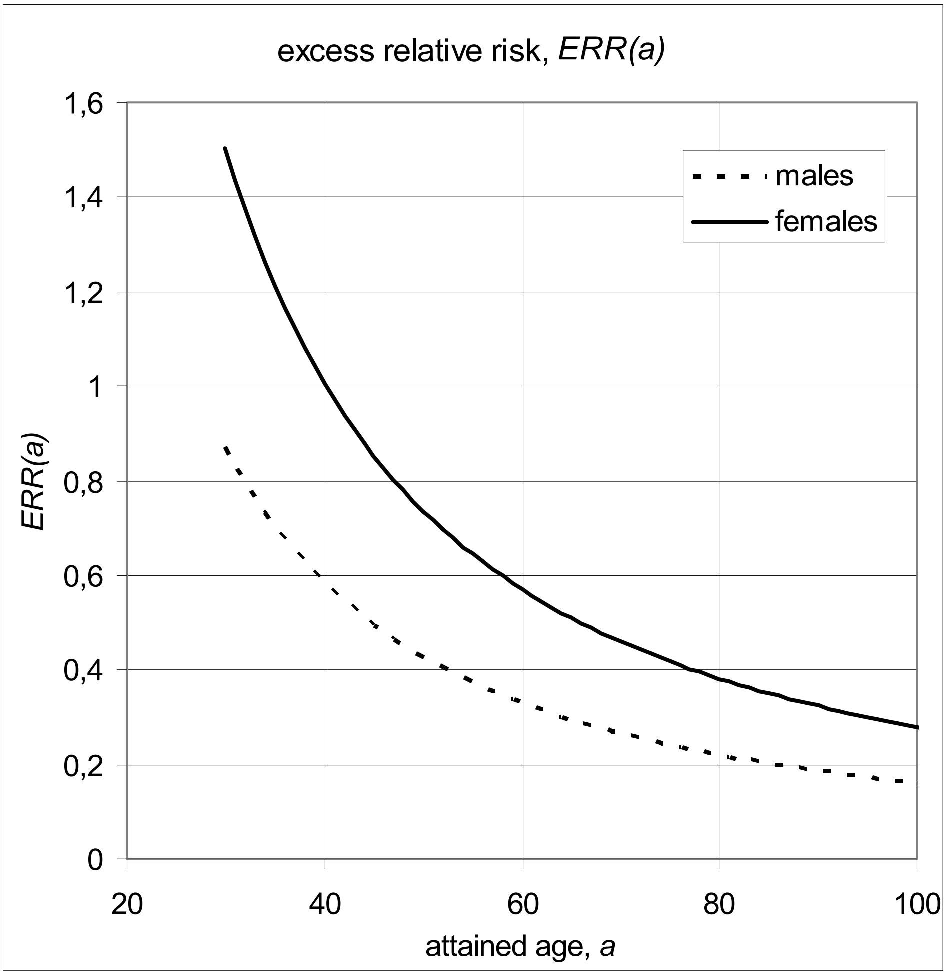

Fig. 3. ***ERR*** **per unit dose for incidence of all solid cancers as function of attained age** ***a*****, for age at exposure e ≥ 30.**

Survival / life expectancy of patients: Survival data (Kaplan-Meier curves) for this particular patient cohort were supplied by the Munich Cancer Registry. Overall survival is available for up to 20 years after diagnosis of metastasis, and survival for sub-cohorts by age group for between 3 and 15 years (Fig. 4). To perform the integration according to eqns. (2) to (4), a fit to these data is needed. In order not to underestimate survival in later years, a logistic function of type $S(a-e)=\left(1+\left(\frac{a-e}{c_1}\right)^{c_2}\right)^{-1}$ is used, with c_1=2.246, c_2=1.27 for the full patient cohort. It is assumed that the radiation exposure under investigation occurs when metastasis is detected, such that Fig. 4 depicts survival after exposure, i.e. the abscissa (years after metastasis) equals attained age minus age at exposure, $a-e$.

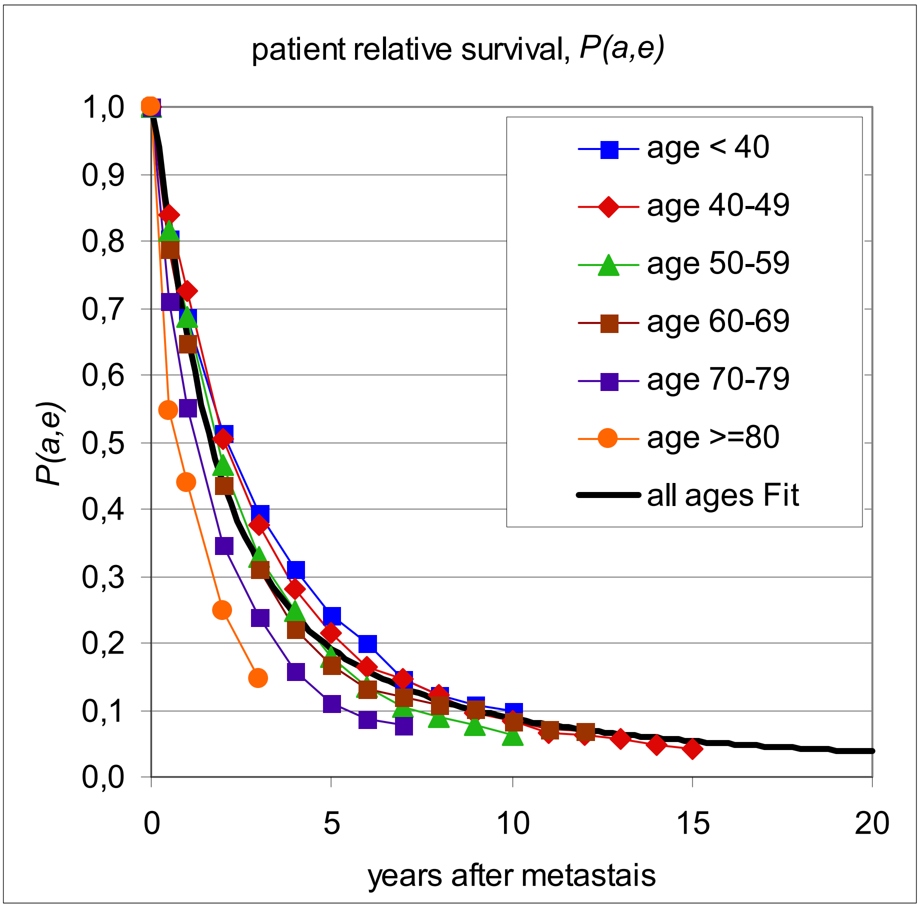

Fig. 4. Relative survival for patients from sub-cohorts by age (data from Munich Cancer Registry).

Results

The above input data were used to calculate *PROLARM* and *PROLARA* for this patient cohort. The effect of patient survival (prognosis) on *LAR* is depicted in Fig. 5a, which plots the *differential LAR* (*dLAR*), i.e. the integrand in eqns. (1), (2), vs. attained age, *a*, in this example for age at exposure *e* = 50 years. *PROLARM* is simply the ratio of the areas under the respective curves.

A very similar diagram can be generated by plotting the conditional survival *S'(a,e)* for patient and non-patient (Fig. 5b). The ratio of the areas under the curves from Fig. 5b yields the *PROLARA* for the given age at exposure, according to eqn. (4).

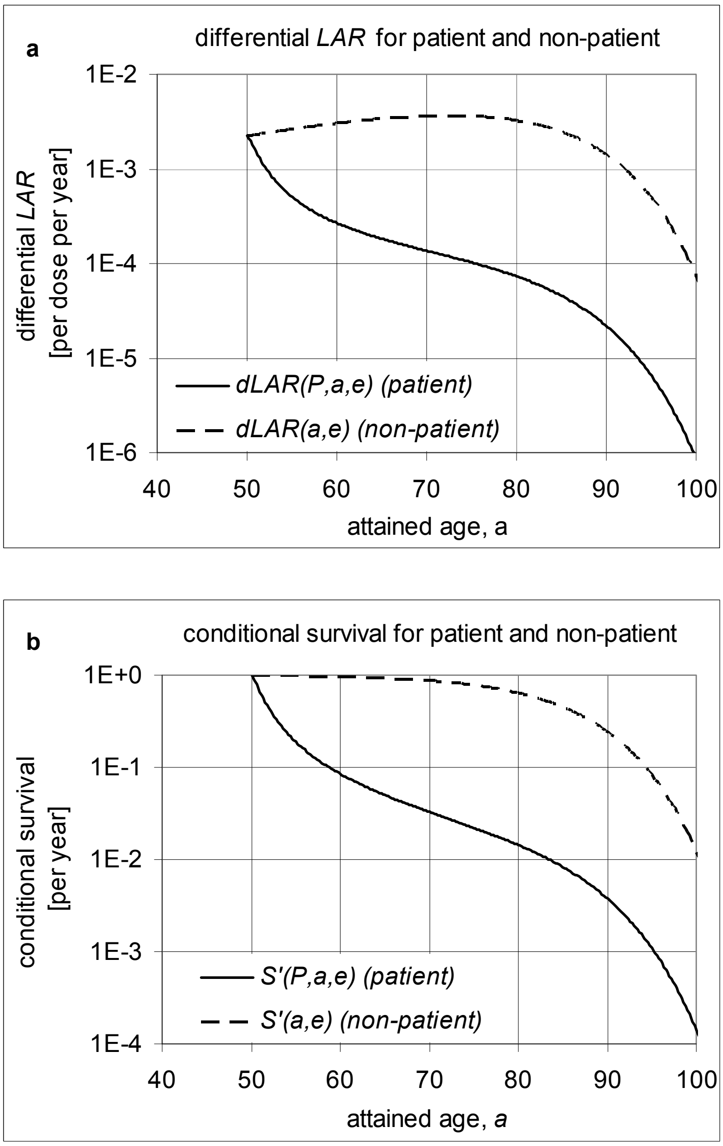

Fig. 5. Comparison of risk (a) and survival (b) for patient and non-patient
a Differential *LAR* (*dLAR*) for patient, *dLAR(P,a,e)*, and non-patient, *dLAR(a,e)*, for *e* = 50 a.
b Conditional survival for patient, *S'(P,a,e)*, and non-patient, *S'(a,e)*, for *e* = 50 a.

Performing the integration for all ages at exposure, *e*, and plotting the quotients of the integrals for non-patient and patient yields the desired diagram of *PROLARM* and *PROLARA* values (Fig. 6; parameters used in integration: latency L=10 years, maximum age a_{max}=100 years.)

From Fig. 6, the following can be seen:

- *LAR* of solid cancer is significantly decreased for patients of any age at exposure compared to non-patients. At younger ages, this decrease is more pronounced (*PROLARM* as well as *PROLARA* > 20 for $e \leq 65$). Using fitted functions for *ERR* as proposed in BEIR VII Phase 2 report, *PROLARM* is always greater than the ratio of the survival integrals. The approximation thus gives a lower estimate of the reduction in risk.
- Integrating over the curves in Fig. 5a for an effective dose of 10 mSv at age $e = 50$ yields *LAR* values of 1.2E-3 for non-patient and 4.3E-5 for a patient from the above cohort, respectively. That is a reduction in risk by a factor (*PROLARM*) of 29. From the approximation using only survival data, i.e. integrating over the curves in Fig. 5b, that ratio (*PROLARA*) is 27.

Thus, a diagnostic procedure with an effective dose of 50 mSv for a patient with metastatic breast cancer corresponds risk-wise to ca. 2 mSv for a healthy person of the same age.

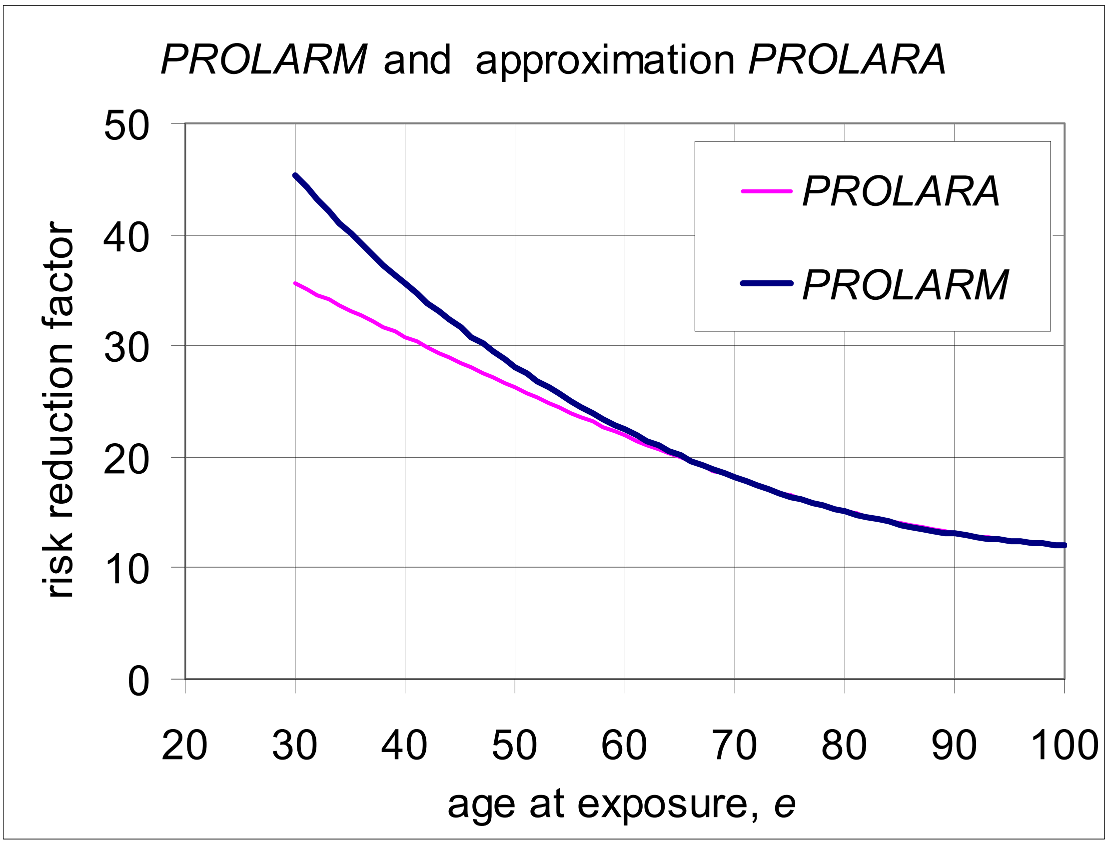

Fig. 6. *PROLARM* and *PROLARA* for the patient cohort under investigation as function of age at exposure, *e*.

Discussion

We introduce a novel method to approximate the reduction in cancer risk from diagnostic radiation exposure for patients based on their difference in life expectancy as compared to non-patients. We had investigated the topic before on an ad hoc basis (Smolarz et al. 1998), but this formal approach allows a general treatment of the topic. Performing the exact computation to yield the *PROLARM* value requires knowledge of the baseline cancer risk for the population under investigation as well as a choice for the excess relative risk (*ERR*). Calculation of the approximation *PROLARA* does not need those but only requires knowledge of the baseline and patient survival curves. Although the *PROLARA* concept was introduced here for only one particular patient cohort, it is transferable to any other.

In order not to rely solely on data from BEIR VII report, we tested *ERR* values from other sources like the (ICRP 2007) recommendations and could verify that the choice of values has only a minor influence on the quantitative results.

In terms of the number of radiation-induced cancers, our results indicate that even within a large cohort of patients exposed to any diagnostic procedure, cancer induction will be undetectable. A major portion of the total dose from diagnostic radiation exposures does not result in any increase in cancer risk at all. To quantitate that amount for other patient groups and diseases shall be the topic of future investigations.

Conclusions

Calculations of cancer risk from diagnostic radiation exposure must take into account the patient's prognosis, i.e. his or her reduced life expectancy compared to an otherwise comparable healthy individual. Our *PROLARA* concept allows for an approximation of the reduction in such risk resulting from any pathology. The approach is both practical and solid in that it only needs data on the patients' survival but is independent of values for the excess relative risk.

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Extracranial radiation doses in children undergoing Gamma Knife radiosurgery

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Abstract

It has been long known that patients treated with ionizing radiation carry a risk of developing radiation induced cancer in their lifetimes. Factors contributing to the recently renewed concern about the radiation induced secondary cancer include improved cancer survival rate, younger patients population as well as emerging treatment modalities that can potentially elevate secondary exposures to healthy tissues distant from the target volume. Gamma Knife radiosurgery stereotactically delivers a high single dose of external radiation to a small well-defined intracranial lesions. Due to a large amount of dose delivered in a single fraction (10–30 Gy) in stereotactic radiosurgery, dose outside the treatment volume is an important issue. The aim of study was to measure the out-of-field doses during the Leksell Gamma Knife Model C radiosurgery for children. The children population was chosen due to their higher susceptibility to radiation. Also, due to smaller size of their bodies, the larger doses are expected to all out-of-field tissues and organs than for adults for the same irradiation conditions. The purpose was to identify doses delivered to the eye lens, thyroid glands, breasts, sternum, upper abdomen, gonads and knees. According to the phantom measurements of Hasanzadeh et al., Phys. Med, Biol. 51, 4375–4383 (2006), the surface dose is comparable to depth dose and we placed dosimeters at the surface of the skin to measure depth (organ) dose. At the every point of the measurement two types of thermoluminescent dosimeters were placed, LiF:Mg,Ti and LiF:Mg,Cu,P and two radiophotoluminescent glass dosimeters (GD-352M). Preliminary results obtained for two children show very good agreement between different dosimetry methods but very different doses at the sites of the particularly organs, for instance for eyes 1.74% (patient A) and 2.02% (patient B) and for thyroid 0.53% (patient B) and 1.2% (patient A). Doses were expressed as percentage of the mean target dose.

A cohort study of childhood cancer following diagnostic X-rays

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Abstract

Introduction: Ionising radiation is an established cause of cancer, yet little is known about the health effects of doses from diagnostic examinations in children. The risk of childhood cancer was studied in a cohort of 92.957 children who had been examined using diagnostic X-rays in a large German hospital during 1976–2003.

Material and methods: Radiation doses were reconstructed using the individual dose area product and other exposure parameters, together with conversion coefficients developed specifically for the medical devices and standards used in the radiology department. Newly diagnosed cancers occurring between 1980 and 2006 were determined through record-linkage to the German Childhood Cancer Registry.

Results: The median cumulative effective radiation dose was 7 μSv. 87 incident cases were found in the cohort: 33 leukaemias, 13 lymphoma, 10 central nervous system tumours and 31 other tumours. The standardized incidence ratio (SIR) for all cancers was 0.99 (95% CI: 0.79 1.22). No trend in the incidence of total cancer, leukaemia or solid tumours with increasing radiation dose was observed in the SIR analysis or in the multivariate Poisson regression. Risk did not significantly differ in girls and boys.

Discussion: Overall, while no increase in cancer risk with diagnostic radiation was observed, the results are compatible with a broad range of risk estimates.

Modelling electron and photon interactions for applications in brachytherapy

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Abstract
One of the main purposes of specific brachytherapy techniques is positioning radiation sources inside or next to the tumour in order to minimize irradiation of surrounding healthy tissues while assuring delivery of high dose levels to the target volume. For example, Ru-106 plaques (Georgopoulos et al. 2003) and I-125 seeds (Thomas et al. 2008, Fuller et al. 2004) are commonly used for eye melanoma and prostate cancer treatments, respectively. Both applications require a precise dose distribution map defining the appropriate radioactive source implantation by minimizing damage to sensitive areas and concentrating dose inside the tumour volume. Following the method we recently published (Muñoz et al. 2008), here we apply energy deposition models at the molecular level for high energy electrons (0-3.5 MeV) and photons (50-100 keV) as a complementary tool for clinical treatment planifications. Briefly, this model consists of a new Monte Carlo simulation programme which as its input parameters uses the interaction probabilities (cross section) and energy loss distribution function we have previously measured or calculated. Special attention is paid to the effect of low energy secondary electrons which are the main responsible for induced damage via molecular dissociations. Detailed energy deposition images and radiation damage evaluation can be derived from the single particle track structures (photon and electrons) predicted by the present model.

Introduction
It is well known that during the irradiation of biological tissues by different kinds of incident radiation, a large portion of energy dose is eventually deposited in the target material by secondary electrons through multiple collisions. However, only recent discoveries have shown that molecular damage (e.g., molecular dissociations or strand breaks in DNA) can be induced very efficiently even by sub-ionising electrons through

molecular resonances (Boudaïffa et al. 2000, Huels et al. 2003) and dissociative electron attachment (Hanel et al. 2003, Abdoul-Carime et al. 2004). This new understanding of the importance of slow electrons for radiation interactions in biomaterials has still not found its way into diagnostic or therapeutic medical practice, where Monte Carlo calculations become faster by using sophisticated condensed-history algorithms while employing rather simple physical models. For example, most relevant interaction models disregard low-energy electrons by forcing them to instantly deposit all of their remaining energy below a certain cut-off value. This neglects the fact that collectively, low-energy secondary electrons can carry away a considerable amount of energy from the primary particle's path and produce molecular damages there.

The full Monte Carlo code Low-Energy Particle Track Simulation (LEPTS) has been specifically designed by us with the intention to translate the information available through recent investigations into the simulation approach. This is achieved by introducing substantially more details in the electron interaction model and giving a molecular-level description of the processes involved in energy degradation (ionization, neutral dissociation, electronic, vibrational and rotational excitation, and dissociative electron attachment). After having gathered the necessary input data for applications in a water environment (Muñoz et al. 2007b, 2008a), we here show two applications to brachytherapy treatments of our electron transport simulation.

First, LEPTS is used to simulate brachytherapy of the eye with the beta-emitter ^{106}Ru. Uveal melanoma and other malignancies of the eye can be effectively treated by surgically implanting a concave ruthenium applicator tightly around the eyeball (see fig. 1). Second, LEPTS is combined with PENELOPE (Baró et al. 1995) in order to simulate the interaction of photon radiation with water. In particular, we investigate photon radiation with an initial energy distribution as measured for ^{125}I seeds that are used for radiotherapy of prostate cancers but can also be employed for treating lesions affecting the eye. In both cases, the localized dose deposition by the radionuclide benefits treatment outcome by sparing healthy patient tissues while delivering high doses to the clinical target volume. At the same time, these isotopes are suitable for longer-term or permanent implants, assuring the radioprotection of medical staff and third persons in close contact with patients.

Computational methods

Monte Carlo code

Our simulation model LEPTS is implemented in a C++ programme and has been described in detail elsewhere (Muñoz et al. 2005, 2007a). Comparing to other existing Monte Carlo codes like EGSnrc (Kawrakow 2000), Geant-4 (Agostinelli et al. 2003), MCNPX (Hendrick et al. 2007), PARTRAC (Friedland et al. 1998, 2003) or PENELOPE (Baró et al. 1995), the main differences achieved with LEPTS in electron tracking are summarized as follows. First, the present code distinguishes different types of inelastic interactions (ionization, neutral dissociation, electronic, vibrational and rotational excitations where applicable) and treats them with specific routines, thereby permitting that absorbed energy dose be not strictly proportional to the total number of charge pairs generated. Secondly, input parameters (integral and angular-differential interaction cross sections and electron energy loss distributions) are based on experimental data whenever available, which are deemed to generally match the simulated conditions more closely than theoretical predictions and which do not depend on the approximations or simplifications needed for calculating data. Thirdly, all electrons are followed until their complete thermalization (meV range) in the irradiated material without any cut-off limits and without applying any range or energy limits for secondary electron generation. Additionally, all electron tracks are fully simulated and no kind of condensed-history algorithm is applied, implying that the exact locations of all energy losses (and their magnitude) or elastic collision events are revealed through the simulation. The present interaction model thus offers a description of radiation-matter interaction at the molecular level, providing details about interaction types and locations that can be used to identify the spatial distributions of certain events of interest, e.g. molecular dissociations or large-angle scattering events.

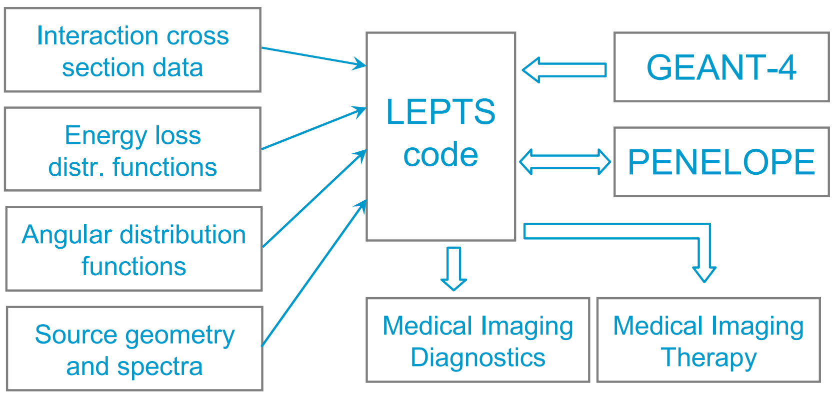

Fig. 1. Scheme of the Monte Carlo code LEPTS. The programme uses geometrical/material definition facilities, sampling mechanisms and graphical output generation from the GEANT4 toolkit but substitutes the electron interactions classes used there by two new routines corresponding to elastic and different kinds of inelastic scattering. Photon physics from either GEANT4 or PENELOPE can be used if necessary.

Incident radiation spectra

106Ru is a β emitter (endpoint energy 39.4 keV) that slowly (T1/2 = 373.59 d) decays to 106Rh. This has a half-life period of 29.8 s before decaying to 106Pd (stable) by different beta decays with a maximum energy of 3.541 MeV. Subsequent γ emissions from palladium have a maximum energy of 1.5623 MeV. The effective electron emission spectrum of the applicator for use in the simulation was determined experimentally by means of three silicon (SiPAD) detectors measuring in coincidence (Muñoz et al. 2008b). The lowest-energy electrons emitted directly from ruthenium are absorbed within the applicator material (see fig. 2b), reducing the applicator's emission to the keV-MeV spectrum of its daughter nucleus rhodium.

Photon emission spectra that were measured with standard solid state spectrometers (Muñoz et al. 2008b) in order to check for possible contamination of the source revealed gamma energies in excellent agreement with the disintegration scheme. However, in order to quantitatively relate electron and photon radiation, relative intensities of the gamma emissions were taken from the Lund Nuclear Data Service (Chu et al. 1999).

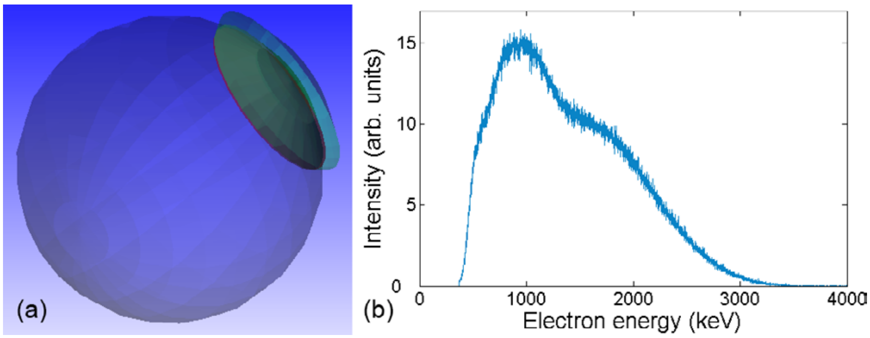

Fig. 2. Ru/Rh applicator used in the simulations: (a) Geometric scheme of the eyeball with the applicator (diameter: 12mm). (b) Experimental β emission spectrum of the applicator.

^{125}I decays with 100% probability to the 35.49 keV state of ^{125}Te by electron capture with a 59.408 day half-life period. There are no transitions to the fundamental level of Te-125. Due to the subsequent relaxation of nucleus and shell, a gamma and X-ray photon emission takes place which is considered as the primary radiation in this model. A standard calibrated solid state Si(Li) spectrometer has been used to determine the energy and intensity of the photons emitted by a ^{125}I brachytherapy seed (see Fig. 3). A representative spectrum shows the γ peak and various X-ray lines in the range 27-32 keV of tellurium. Additionally, we observed X-ray lines between 22-26 keV with an intensity comparable to some of the photons emitted by Te which are attributed to silver (present as the core of the seed onto which the radioactive iodine is adsorbed). These "contaminations" of the spectrum need to be taken into account for realistically modelling applications in brachytherapy (Rivard et al. 2004), and are thus included.

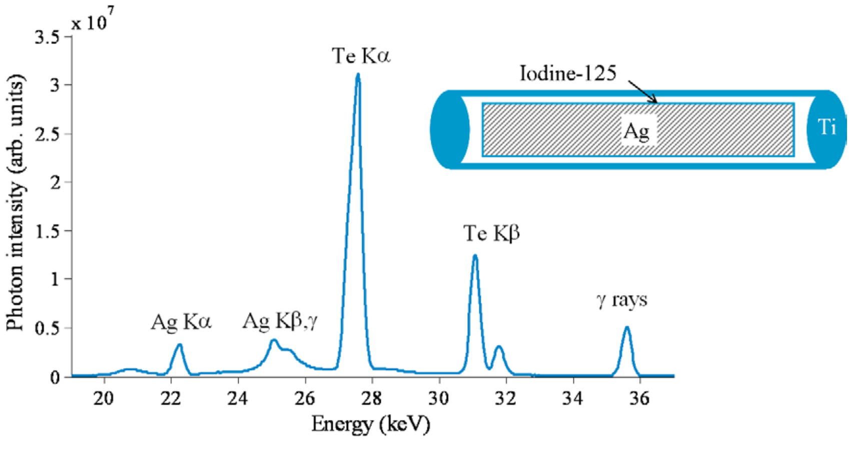

Fig. 3. Photon emission spectrum of an I-125 seed (Amersham Health model 6711, see scheme) measured in perpendicular geometry. Gamma photons (35.49 keV) produced during ^{125}I/^{125}Te-decay as well as the most intense Kα and Kβ X-rays of Te are observed. Additional X-rays at about 22keV and around 25-25.5 keV can be attributed to silver.

Interaction data for electrons in water

The total and partial cross sections needed in order to follow different electron scattering processes in water were obtained from experimental results whenever possible.

Total electron scattering and integral ionization cross sections in water vapour were previously measured between 50 eV and 5 keV (Muñoz et al. 2007b) with a transmission beam technique and using synchronized electron and ion extraction pulses, respectively. Below 50 eV, data from Čurík et al. (2006) and Szmytkowski (1987) (total CS) and Straub et al. (1996, ionization) were used. For elastic collisions and the remaining inelastic channels (rotational excitation for applications in the gas phase, vibrational excitation, electronic excitation, dissociative electron attachment, neutral dissociation), integral CS were derived by combining experimental data (Brunger et al. 2008, Thorn et al. 2007a,b, Itikawa and Mason 2005, Kedzierski et al. 1998, Harb et al. 2001, Cho et al. 2004) with theoretical calculations that were carried out with an optical potential method based on an independent atom approximation including screening corrections (details are described in Blanco and García 2003a,b and 2007). The theoretical total cross sections show very good agreement with experimental ones in the overlapping energy range > 10 eV, hence confirming the validity of the approach and indicating calculated elastic and inelastic CS to be trustworthy as well. For high energies, the electron-molecule collision can be treated as a plane wave interaction with a sum of atoms in the framework of the first Born approximation. Integral elastic and inelastic interaction CS can then be represented by simple energy-dependent formulae (Inokuti 1971, García and Blanco 2000). For want of other data, this method was used at energies greater than 10 keV. Angular distribution functions for scattered electrons were taken from our calculations.

Results and Discussion

Ru-106 plaque therapy

Using a generic setup frequently found in brachytherapy of the eye with concave Ru/Rh plaques (sketched in fig. 2), radiation-induced processes in the volume of interest – the eyeball approximated by liquid water – have been simulated. In this case, the incident photons where simulated using the photon interaction processes integrated in GEANT4. Once a secondary electron is generated, it is further tracked using LEPTS, identically to primary electrons. This permits obtaining the exact location of each interaction event as well as the type of collision produced, the energy deposited, the change of momentum suffered by the photon or electron, and the energy and direction of the secondary electron produced in case of ionizations as the programme output. Figure 4 shows a 2D projection of a small example data set obtained. Each coloured point indicates an inelastic interaction event. Figure 5 shows lateral and transversal sections through the energy deposition map corresponding to the same situation, calculated after simulating approximately 2×106 primary particle histories. The relative depth dose curve along the central axis obtained after normalization (100% at 2 mm depth) is shown in fig. 6. It can be seen that even though the dose gradient is very steep within the first few mm inside the eye, there is still an appreciable amount of energy deposited in greater depths (e.g., ~ 16% in 1 cm depth and ~ 4% in 2 cm depth) due to the applicator geometry (curved surface), the greater penetration of incident photons (compared to electrons which account for the main dose in the entrance region), and secondary electrons depositing small amounts of energy in multiple collisions while slowing down continuously.

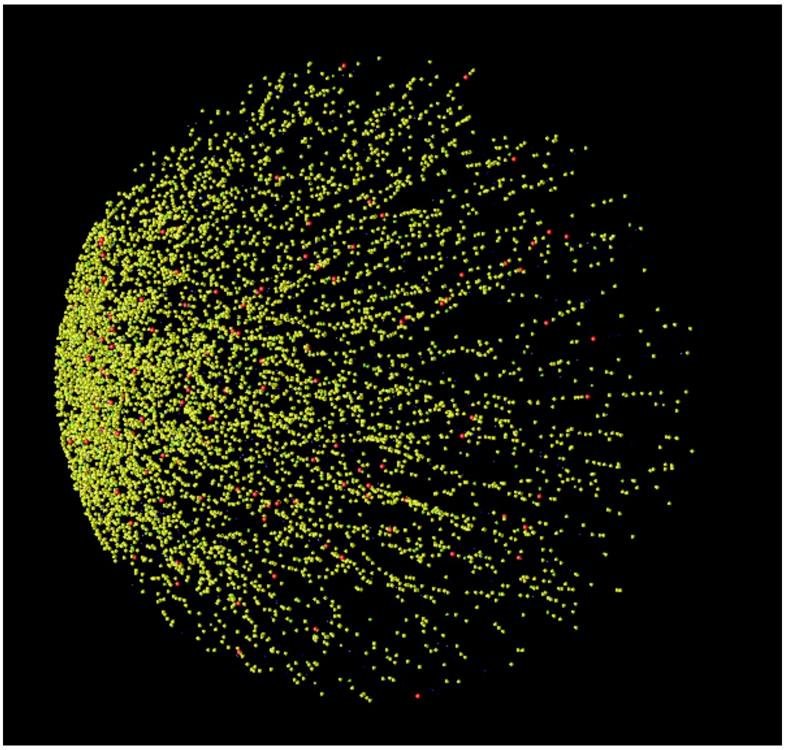

Fig. 4. Map of interactions produced inside the eyeball when irradiated with a Ru/Rh plaque from the left. Inelastic scattering events are colour-coded as follows: yellow, ionization; chartreuse, neutral dissociation; green, dissociative electron attachment; blue-green, electronic excitation; light blue, vibrational excitation; red, Auger electron emission. Photon-induced events are not explicitly shown but their main effect is to generate high-energy secondary electrons which are then followed identically to primary electrons.

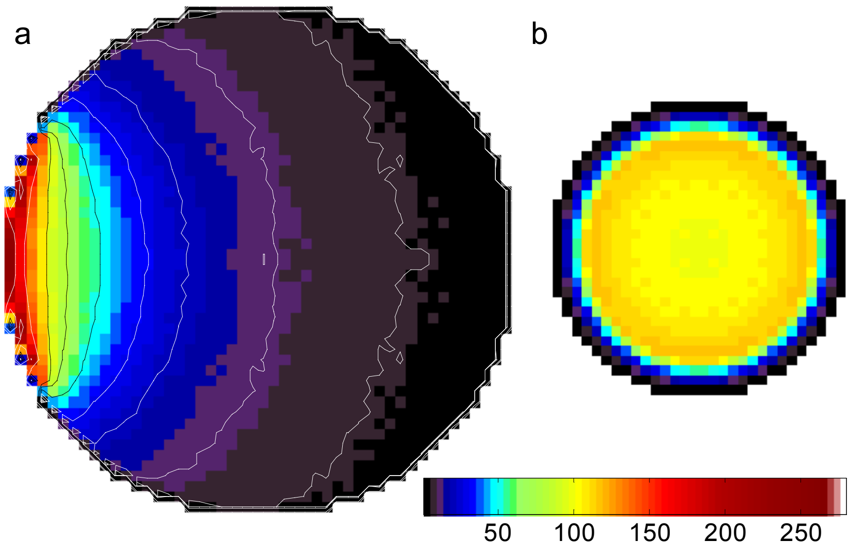

Fig. 5. Relative dose distributions (normalized to 100% in 2mm depth at the central axis) produced by the ^{106}Ru plaque in the eyeball (voxel size: 0.5 mm). (a) Lateral section through the central axis with 200%, 150%, 100%, 75%, 50%, 30%, 20%, 10% and 5% isodose lines shown (left to right). (b) Transversal section in 2mm depth. Both distributions use the same colourmap.

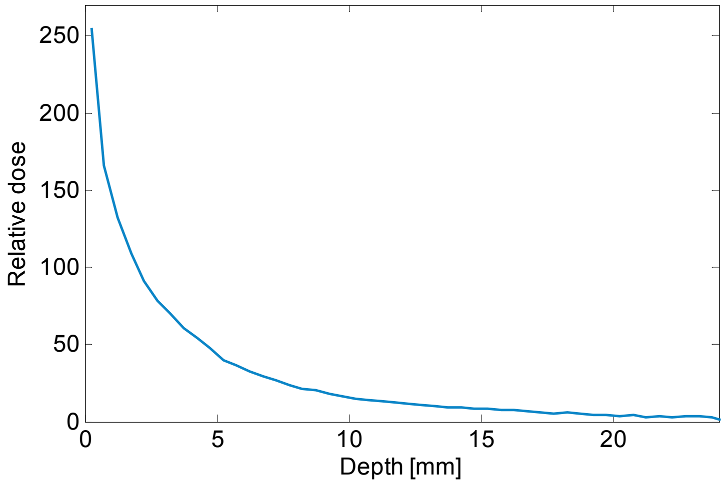

Fig. 6. Relative depth dose curve obtained in the water sphere along the central axis of the applicator. Normalization as in fig. 5.

I-125 seeds

Fig. 7. Simulation of 100 000 photons emitted from a 4 mm (diameter) disk according to the energy distribution measured for I-125 brachytherapy seeds. The simulated volume consists of 10x10x10cm^3 of water vapour at a density of 0.7320 g/cm^3. Different types of interactions are colour-coded as follows: red, photoelectric effect; green, Compton scattering; blue, Rayleigh scattering.

Using LEPTS in combination with PENELOPE (for handling all photon interactions), we simulate the interaction processes induced in water vapour when exposed to photon radiation according to the emission spectrum measured for 125I seeds. The resulting interaction map with H2O molecules, at a density similar to that of liquid water, is shown in Fig. 7. It can be observed that in the geometry used, the photon beam remains laterally well defined, however small contributions to energy deposition can still be found near the boundary of the simulated volume. Photon interactions are coloured according to the type of event produced. It can be observed that photoeffect (red dots) prevails for 125I in water (which, for many purposes, is an acceptable approximation for human tissue), indicating that the main effect of the incident photons is to generate high-energy secondary electrons. This highlights the necessity to include accurate electron interaction data and models into simulation approaches for medical applications, even if the primary radiation consists of photons as with I-125.

In figure 8, one photoelectron track, shown from its generation until complete thermalization, illustrates details of the energy degradation mechanisms. As can be seen, the secondary particle undergoes successive different inelastic interactions with target molecules and thus deposits energy at many points along its track. Furthermore, still more electrons are generated by ionization events.

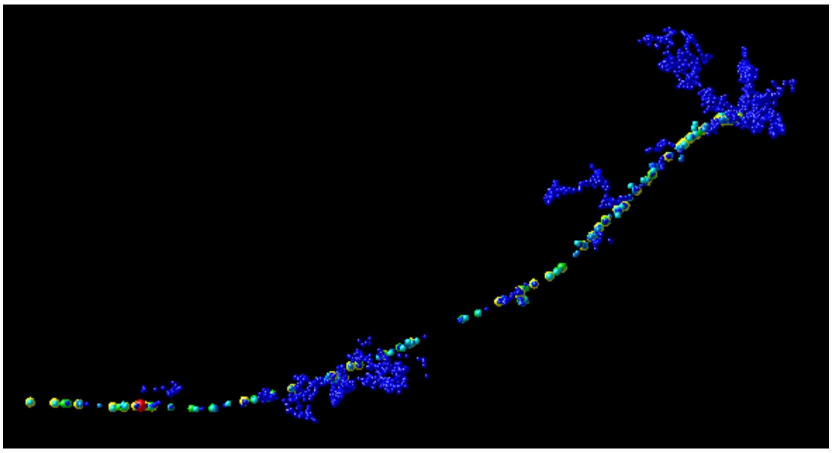

Fig. 8. Example of a photoelectron trajectory (from left to right) showing details about the electron-molecule interactions produced. Collisions are depicted using orange (ionization), yellow (neutral dissociation), green (electronic excitation), cyan (vibrational excitation), light blue (rotational excitation), and dark blue (elastic collision). For clarity, only some of the numerous elastic collisions are shown.

Fig. 9 depicts the energy deposition corresponding to a parallel photon beam of 4 mm diameter penetrating water at a density of 0.7320 g/cm^3. Most of the incident energy is lost immediately after entrance into the medium. 58.4% of the total energy is deposited within the first 2 cm, and 90% of the energy is deposited within 5.1 cm.

Conclusions

Brachytherapy of many tumours requires increased spatial precision due to the sometimes small dimensions of the treatment volume and the close proximity of organs at risk. A high accuracy energy deposition model may thus improve dose calculations and, consequently, treatment outcome. Motivated by this, the recently developed model LEPTS, based on Monte Carlo simulation using critically selected input data, has been applied to determine the energy deposition in water of Ru-106 and I-125, two radionuclides commonly used in brachytherapy. The electron transport model provides detailed information about secondary electron tracks, energy deposition and interaction processes at the molecular level and may thus yield an improved picture of radiation damage in a biomedical context. In order to validate the present Monte Carlo approach, further studies will be aimed at a more realistic simulation of clinical treatments (prostate / eye cancers), including exact geometries of the relevant patient tissues.

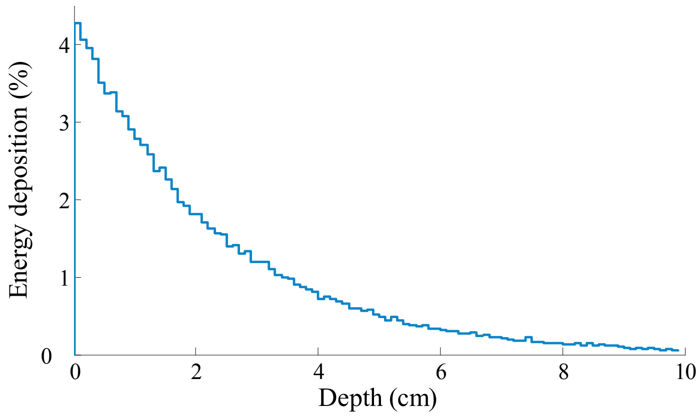

Fig. 9. Histogram depicting the energy deposition by an I-125 source in water vapour supposing parallel photon emission. All contributions for a given depth are summed. The total energy deposited in the simulated volume (same as fig. 7) is normalized to 100%.

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Usefulness of specific sensitivity factors of radionuclide calibrators used in nuclear medicine

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Abstract

In nuclear medicine, the activity of a radionuclide is measured with a radionuclide calibrator that has often an efficiency factor independent of the container type and filling.

To determine the effect of the container on the accuracy of the activity injected to the patient, we simulated a commercial radionuclide calibrator and 18 container types representative of the clinical practice. The instrument sensitivity was computed for various container thicknesses and filling levels. Mono-energetic photons and electrons as well as five common radionuclides were considered. The quality of the simulation with gamma-emitting sources was validated by an agreement with measurements better than 4% in four critical situations.

Our results show that the measured activity can vary up to a factor 2 depending on the type of container. The filling and wall thicknesses only have a marginal effect for radionuclides of high energy but could induce differences up to 5%.

We conclude that dose calibrators used for clinical applications should have efficiency factors specific to a given container type.

Introduction

The dose received by the patient in nuclear medicine depends on the administrated activity. The optimization principle of radiation protection states that this dose should be kept as low as reasonably achievable . For diagnostic exams, "reasonably" means that the dose should not be so low as to prevent the searched pathology to be detected. In radiation therapy, this implies that the dose should be high enough to cure the disease but as low as possible for the healthy tissue. Whatever the context, the first step is to measure the activity with a sufficiently low uncertainty.

What is commonly understood by "sufficiently low uncertainty" depends on the context. Measurement uncertainty (defined here and in the rest of the text as one standard deviation or k=1 [1]) is typically below 1% for primary laboratories operating with liquid sources of well-defined geometries. At the clinical level, activities used for diagnostic examinations are generally administrated with uncertainties smaller than

10%. For radiotherapy treatments, many would like to reach values similar to the common practice of external radiotherapy, around 3% [2].

Prior to injecting patient the activity of radiopharmaceuticals is verified most of the time by re-entrant ionization chambers called radionuclide calibrators or activimeters [3]. It is not clear if activity measurements performed in nuclear medicine departments should take specific account of sensitivity factors specific to the radionuclide, container type (syringe or vial) and measurement geometry. Several radionuclide calibrators found on the market are used with sensitivity factors only dependent on the radionuclide whatever the container type or filling level. Knowing that some manufacturers do provide their instruments with more specific geometric factors, it is of interest to estimate the relevance of this approach for a wide range of conditions encountered in practice.

The goal of this study was to estimate the effect of specific sensitivity factors on the measurement uncertainty for a variety of measurement geometries. We used a Monte Carlo (MC) simulation method that was first validated by comparison with a series of measurements. MC simulations were then performed for different specific radionuclides used in clinic in order to estimate the variation of the sensitivity factor (1) for a large range of container types, (2) for different filling levels and (3) for different container thicknesses. Finally, we performed the same MC simulations for pure photon- and electron-emitters at fixed energies in order to explain the observed results with the specific radionuclides.

Material & method

Dose calibrator

We considered the dose calibrator Veenstra model VDC-405 [5]. The sensitive volume is filled with gaseous argon at a pressure of 12 bars. The chamber walls are 2.5 mm aluminum. The chamber is protected against contamination by a 3.3 mm Plexiglas sleeve, and from external radiation by 3.0 mm of lead.

The details of the measurements of Y-90, Sr/Y-90 and F-18 can be found in [6]. The measurements of Co-57 and Co-60 were performed in the frame of verification procedures at the Lausanne University Hospital. The measurements of I-131 and Tc-99m were collected from various follow up measurements of the dose calibrators at the Lausanne University Hospital.

Monte Carlo code

In this work, we used the version 4.8 of the Monte Carlo code Geant. In our previous publication, we used the version 3 [4]. The main improvements of the present version are the lowering of the cut thresholds, which affects particularly the low density materials such the interior of the chamber, and a better treatment of the ionization process, with the LowEm package. The new version is written in C++, and allows an easier implementation of the geometry. The version 4 gives actually a relative discrepancy between simulation and measurement about two times lower than the version 3 for gamma emitters.

Simulated sources

We simulated mono-energetic sources of pure photon emitters from 20 keV to 4 MeV and of pure electron emitters from 50 keV to 4 MeV. We also simulated five gamma-emitting radionuclides (Tc-99m, In-111, I-123, I-131, Cs-137), one beta plus emitter (F-18) and one pure beta minus emitters (Y-90).

Containers geometry

A representative sample of each container was physically measured in order to estimate the variations of the geometries. The containers were first cut open to allow the measurements of inner diameters and thicknesses. Syringes are manufactured by plastic injection and do not suffer from large geometry variations. The glass vials on the other hand are more prone to variations. However, the measurement in the dose calibrator has a cylindrical symmetry. The variation of the thickness of a vial along its circumference is typically ±0.2 mm. The main geometrical characteristics of the containers used in this study are presented in Table 1.

Table 1: Characteristics of the containers investigated in this study. ID letter "V" is for vial and letter "S" is for syringe.

ID	Long name	Material	Internal diameter [mm]	Height [mm]	Volume [ml]	Standard filling [ml]
V01	Penicillin vial	glass	25.3	53.2	10.0	5.0
V02	Elution vial	glass	22.0	47.0	11.0	5.0
V03	Cis-Bio glass vial	glass	14.3	32.0	2.0	1.0
V04	Bayer glass vial	glass	25.0	44.6	13.0	10.0
V05	Cis-Metas glass vial	glass	25.5	53.5	15.0	3.0
V06	Metas glass vial	glass	27.0	57.0	20.0	20.0
V07	Zinsser scintillation vial	plastic	26.5	59.0	20.0	20.0
S01	Omnifix - F 1ml	plastic	6.6	88.0	1.0	1.0
S02	Omnifix 2ml	plastic	10.8	64.7	3.0	2.0
S03	Omnifix 5ml	plastic	13.7	75.7	5.0	5.0
S04	Omnifix 10 ml	plastic	17.3	99.0	10.0	10.0
S05	BD Plastipak 1ml	plastic	6.5	88.0	1.0	1.0
S06	BD Plastipak 10ml	plastic	16.2	97.0	12.0	10.0
S07	Terumo 2ml	plastic	10.2	63.2	2.5	2.0
S08	Terumo 5ml	plastic	14.6	70.5	5.0	5.0
S09	Terumo 10ml	plastic	17.4	91.0	10.0	10.0

Parameters tested

For each container, we started by simulating the standard filling condition used in clinic. We then simulated the situation corresponding to the full filling condition and the half-standard condition. Finally, in order to estimate the response of the instrument after injection to the patient, we simulated the situation in which the container only contains a small fraction of the source at its bottom.

Because our physical measurements of the containers showed that a variation of the wall thickness of typically 0.1 mm was encountered in the practice, we also simulated the dose calibrator response to a variation of this amplitude.

Validation of the simulations

Only considering the results of Monte Carlo simulations without comparison with actual measurements is meaningless. We therefore compared our simulations with real measurement in the standard filling conditions. We used container V05 for the Sr-90/Y-90 and the Y-90 sources and container V01 for the other nuclides (Tc-99m, In-111, I-123, I-131, Cs-137, F-18).

Before doing this, we have to notice that a radioactive source induces a current in the ionization chamber that is proportional to the amount of energy deposited in the sensitive volume. During a measurement, the actual current is usually not directly available: The dose calibrator transforms the current by applying a specific calibration factor implemented by the manufacturer in order to display the estimated activity. For the Monte Carlo simulation we have directly access to the energy deposited in the sensitive volume per unit of activity in the source.

The comparison between experimental and simulated results was based on responses normalized to a reference Cs-137 solution in the V01 geometry. All experimental measurements were performed by forcing the dose calibrator calibration factor on the "Cs-137 position".

The metrological traceability of the measured source was guaranteed by a measurement of each source by a secondary standard instrument of our laboratory directly linked to the international reference system (SIR) of the Bureau International des Poids et Mesures (BIPM).

Results & discussion

Validation of the simulations

Experimental and simulated responses normalized to Cs-137 are presented in Table 2. It can be seen that the agreement is better than 4% for the investigated gamma emitters and virtually the same for the beta plus emitter (F-18). The result is however not as good for the two beta minus sources (Y-90 and Sr-90/Y-90) with a statistical difference of 18%.

We can therefore conclude that our simulation model is adequate for gamma-emitting sources while probably in need of improvement for the beta plus emitters.

Table 2: Comparison between measurements and simulations normalized to the Cs-137 condition. An absolute value of z-score below 1.96 indicates no statistical difference at the 95% confidence level.

Condition	Cs-137	Co-57	Co-60	Tc-99m	I-131	F-18	Y-90	Sr-90/Y-90
measurement	1.00	0.669	3.65	0.586	0.822	1.79	0.0394	0.0443
std-uncertainty [%]	0.30	0.45	0.35	3.2	3.5	2.1	2.0	0.60
simulation	1.00	0.687	3.55	0.592	0.789	1.79	0.0322	0.0362
std-uncertainty [%]	0.03	0.29	0.19	0.58	0.34	0.47	1.4	1.7
sim/measurement	1.00	1.03	0.97	1.01	0.96	1.00	0.82	0.82
std-uncertainty [%]	1.7	1.8	1.8	3.7	3.9	2.7	3.0	2.8
z-score	-	1.5	-1.6	0.30	-1.0	0.14	-6.1	-7.5

Sensitivity factors in standard conditions

As an example, Figure 1 presents the dose calibrator sensitivity for monoenergetic photons and electrons. The sensitivity is defined as the energy deposited in the sensitive volume divided by the kinetic energy of the emitted radiation. For photons, it can be seen that the sensitivity peaks above 1% at about 40 keV and plateaus at around 0.1% for higher energies. The sensitivity for electrons is 2 to 3 orders of magnitude below the photons and monotonically increases with the energy. Below around 3 MeV, the electrons only deposit their energy through the bremsstrahlung produced in the source or the instrument surrounding. At 4 MeV, the sensitivity suddenly increases because the electrons start to have energies sufficiently high to directly interact with the sensitive volume.

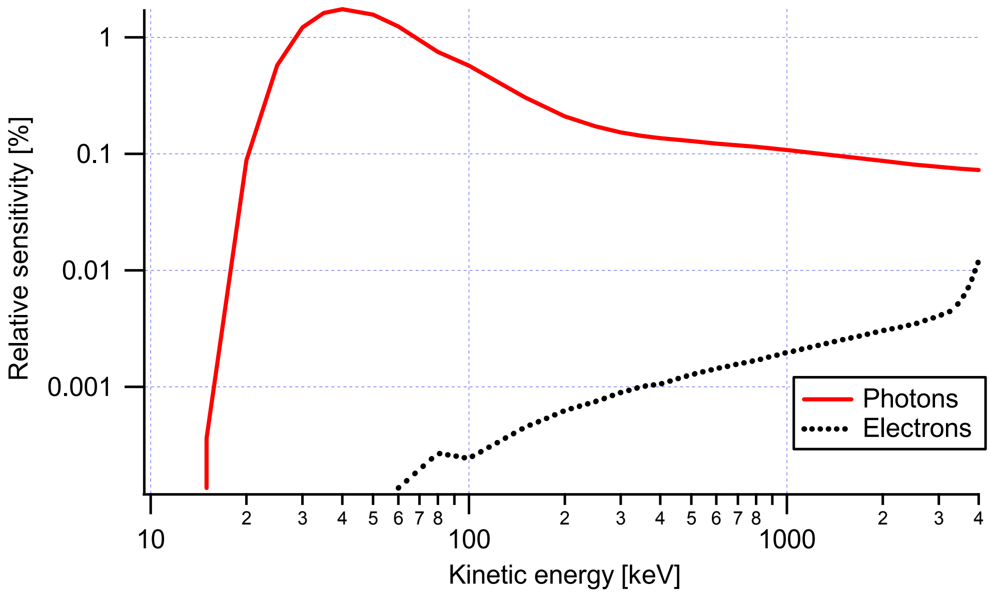

Figure 1. Example of the dose calibrator sensitivity for monoenergetic radiations (container V01 in standard filling conditions; glass density 2.33 g/cm^3).

Figure 2 and Figure 3 show the sensitivity factors of the instrument for each container of interest in standard filling conditions. The values have been normalized to the one of vial V01 with standard filling.

One can see that the sensitivity for photon radiation is smaller than 1% for glass containers. For plastic syringes it is slightly higher with values between -3% to +4%. For electrons, the change of container is relatively modest for the glass vials (V01 to V06) with a variation that can rise up to 16%. For plastic syringes, a change of container for energies smaller than 1 MeV could induce a variation of sensitivity up to 50%. The most dramatic effect arises for plastic syringes containing high-energy electrons for which the sensitivity can increase up to a factor 12.

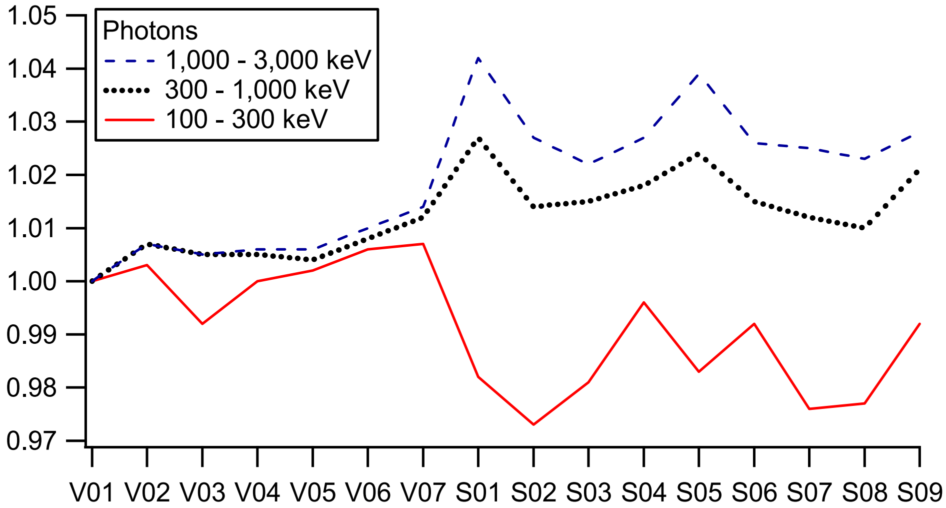

Figure 2. Relative sensitivity factors to photons normalized to container V01 averaged over three ranges of photon energies. Standard filling conditions.

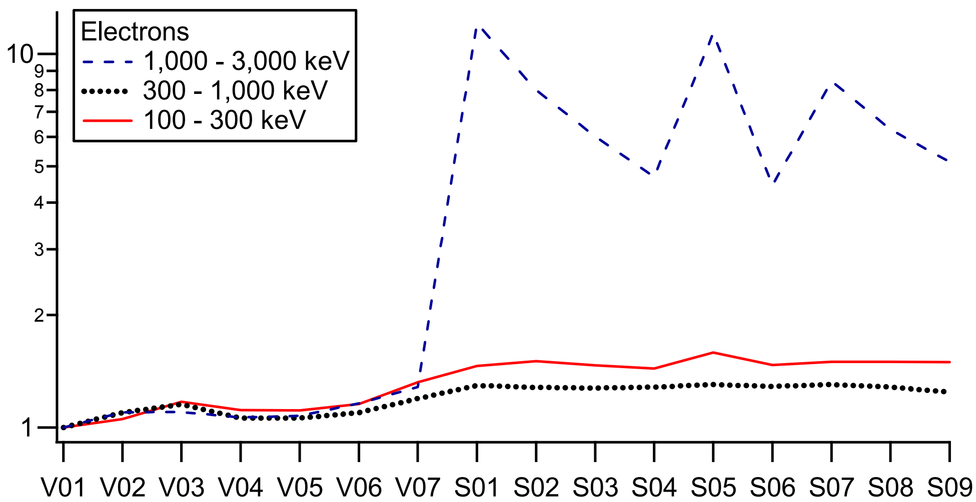

Figure 3. Relative sensitivity factors to electrons normalized to container V01 averaged over three ranges of photon energies. Standard filling conditions.

Figure 4 shows the relative sensitivity factors for a selection of radionuclides at standard filling level. It can be seen that the type of container has the least effect for Tc-99m, F-18 and Cs-137 with variations lower than 4% for glass vials and up to 8% for plastic syringes. For the other nuclides, the variations are also smaller for glass vials than for plastic syringes. However they can go up to a factor of 2.4 for I-125.

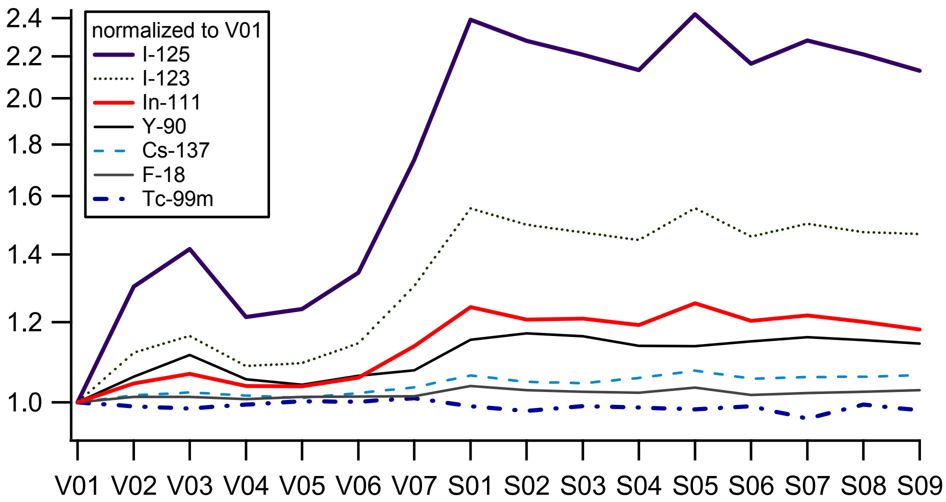

Figure 4. Relative sensitivity factors to different nuclides normalized to container V01. Standard filling conditions

Effect of filling level

The sensitivity factors were computed for different filling conditions as an average of the corresponding energy range for the mono-energetic photons and electrons sources and for the individual nuclides. Each sensitivity factor was normalized to the sensitivity factor in standard filling-condition with the same container (obtained in the previous section) in order to only show the effect of the filling level.

Many containers have standard-filling conditions that are close to their full capacity. This is especially the case for syringes for which a full filling can only be up to 50% above the standard level. In this case, totally filling the container changes the sensitivity factor less than 1% for monoenergetic photons and less than 5% for monoenergetic electrons. The glass vials are usually not filled at their maximum capacity and some of them have a maximum level that can be up to 5 times the standard level. However, the effect on the sensitivity is well below 1% for photons and below 7% for electrons.

The same simulations computed for our set of nuclides show that a full filling of the containers induces a variation on the sensitivity factor less than 3% in most cases. The only exception is for the glass vial V01 totally filled with F-18 (decrease of the sensitivity factor by 9%) and glass vial V04 filled with In-111 (increase of the sensitivity factor by 19%).

Filling only half the containers has a negligible effect on the sensitivity factor for monoenergetic photons: the variation is within Monte Carlo uncertainties (below 1%). For monoenergetic electrons, the effect is significantly higher with a variation up to 10%.

For our selection of nuclides, filling half the containers typically changes the sensitivity factor by less than 4%. The only exception is for glass vial V05 containing Y-90. In this case, the sensitivity decreases by 12%.

We finally computed the sensitivity factors for the container containing only a small volume of radioactive liquid. The effect is small for monoenergetic photons (smaller than 3%). However, for monoenergetic electrons, it can be up to 40%. For our set of nuclides, the effect really depends on the container and the nuclide. The effect is less than 5% for all nuclides except for Y-90 in all glass vials where the effect typically 25%.

Effect of wall thickness

The sensitivity factors were calculated for a variation of wall thicknesses of 0.1 mm. The effect on the sensitivity factor is negligible for all monoenergetic photons and low-energy monoenergetic electrons (below 1 MeV). For electrons in the 1 MeV-3 MeV energy range, the induced effect on the sensitivity factor can go up to 4%.

A change of 0.1 mm of the container wall thickness has a negligible effect for each nuclide of our list. The only exception is I-125 in glass vials for which a difference up to 4% is observed.

Conclusion

Our Monte Carlo model has been validated by measurements for gamma-emitting sources: the observed difference is below 4%. For pure beta emitters, the simulation is not as good and the observed difference is almost 20% for Y-90.

For pure gamma-emitting sources of energies higher than 100 keV, using the calibration factor of one container and measuring with another has a negligible effect with glass vials and an effect up to 4% with plastic syringes. Doing the same with pure beta emitters has a much stringent effect: for glass vials this can induce an error of about 15%, whereas for plastic syringes, the error can be up to 50%. Simulating the same operation for actual sources can produce very small effects for relatively high-energy gamma emitters whereas the error can become higher than a factor 2 for a nuclide such as I-125.

The amount of radioactive liquid actually contained in the container has typically an effect of about 5% for gamma-emitting sources. For electron emitters, the effect is typically twice higher.

Using the dose calibrator for measuring the fraction remaining in a container after injection is not disastrous for gamma emitters: the error is typically within 5%. However, the effect can be much higher for pure beta emitter, with value going up to 25%.

The imperfections of the container wall thicknesses only have negligible effect on gamma emitters of energies above 100 keV. A variation of 0.1 mm in the wall thickness can however induce an error of the order of 5% for high-energy beta emitters or for low-energy gamma emitters such as I-125.

These results therefore suggest using a specific calibration factor for each container and filling level. This is especially the case beta-emitting sources and plastic containers.

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Radiation detection in hemodialysis room for a uremia patient with Y90-microsphere SIRT – Initial eExperience in KMUH, Taiwan

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Abstract

Y90-microsphere selectively intravascular radiation therapy (SIRT), which is radiation embolism, becomes an alternative method for advanced hepatic malignant control in hepatic metastasis and primary hepatoma. Y90 with pure beta ionizing radiation is an available therapy radionuclide in clinic. In southern Taiwan, Y90-microspheres SIRT in hepatic malignant control is a beginning technique. During past year, we have total six-case procedures. Among them, a uremia patient has colon cancer and massive hepatic metastasis. Due to failure of treatment response to conventional chemotherapy and target therapy, she is advised to receive Y90 microspheres SIRT. Under inform consensus and document approval, the survey procedures including hepatic angiography, abdominal CT and Tc99m MAA shunt study are performed before treatment. Base on metastatic tumor burden and calculated dosimetry, 2.5 GBq of Y90 microspheres is arranged to deliver to hepatic lobe. After successful technique performance, the patient received regular dialysis for medical care in the next two day after radiation embolism treatment. First, we arrange independent dialysis space for this special medical condition in our hospital. For environmental exposure dose evaluation, we directly measure X ray from patient by GM counter. There is inhomogeneous radioactivity distribution in patient's body for example: higher in liver, heart and brain. The exposure dose does not exceed 0.15 mR/ h in one meter distance from body surface. Initially, contamination of radionuclide Y90 in dialysis equipments is a potential issue. However, there is no significant radioactive residual in dialysis tube as waste storage in nuclear medicine department. In conclusion, dialysis is not a contraindication post Y90 microspheres radiation embolism treatment. In this special medical condition, we suggest arrangement of relatively independent dialysis space and awareness of nursing care as patient with conventional nuclear medicine procedure.

Introduction

New radionuclide treatment becomes growing in recent year, while radioiodine has been familiar with ablation and treatment in differentiated thyroid cancer over fifty years. Yttrium90 (Y90) is a pure β-emitter with a physical half-life of 64 h and decays to stable zirconium-90 (Zr90). The average energy of the β particles emitted from 90Y transformation is 0.94 MeV with a mean tissue penetration of 2.5 mm. Y90 microspheres selective internal radiation therapy (SIRT) is a novel approach approved in 2002 by the U.S. FDA for the regional treatment of colonic hepatic metastasis [1, 2]. Recently, unresectable hepatocellular carcinoma is also one of indications [3-5].

As we know, elimination of radionuclide from body is dependent on function of kidney and liver. Therefore, patients with end stage renal disease, uremia become special issue of radiation protection in clinic. Almost non- occupation dose related medical staff concerns radiation exposure or contamination when they care patients with radionuclide treatment. Browsing previous literatures in radioiodine treatment patients with dialysis, there are not few of publications [6-12]. However, it seems few of documents to consider the issue of radiation protection associated with Y90 microspheres SIRT and dialysis. In southern Taiwan, Y90-microsphere SIRT in hepatic malignant control is a new development technique. During past year, we have total six-case procedures. Among them, a uremia patient has colonic massive hepatic metastasis. Herein, we share our primitive experience of clinic hemodialysis scheduling and radiation protection for non-occupation dose related medical staff during hemodialysis after Y90-microsphere SIRT.

Material and methods

This is a 68-year-old female patient with history of uremia under regular hemodialysis and diagnosis as advanced descending colonic cancer recent year. Initially, she received standard surgery and chemotherapy treatments. Unfortunately, progressive hepatic metastasis and minimal ascites were diagnosed by CT imaging. Due to unresectable metastatic massive mass and failure respond to chemotherapy, Y90-microsphere hepatic radioembolization treatment was suggested. Under inform consensus and official approval of DOH in Taiwan, therapeutic procedure of Y90-microsphere SIRT was allowed to execute.

Before treatment, several evaluations as cancer condition by ECOG, the hepatic function and hepatic vascular anatomic survey by angiography were performed to exclude improper patient. During first time angiography, prevent coil was set in pancreatico-duodenal arteries to avoid reflux. In the meanwhile, 99mTc-MAA was injected into hepatic vessels to simulate distribution of microspheres to evaluate exact tumor burden and detect hidden vascular shunt to avoid radiation complication. In our case, 99mTc-MAA scintigraphy showed an acceptable lung/liver shunt fraction (8%), less probability of radiation pneumonitis. For evaluation of treatment response, modern FDG PET/CT scan was arranged before 90Y-microsphere treatment.

Considering issues of efficacy and radiation protection, duration between secondary angiographic Y90-microsphere radioembolization and schedule of regular hemodialysis should be arranged carefully. After consultation with an Australian specialist, hemodialysis was performed before radioembolization procedure. And then the next following hemodialysis was arranged after 72 h later. Based on tumor burden and vascular shunting, therapeutic hepatic dose was delivery into left hepatic lobe with 2.5 GBq of Y90-microspheres.

The radio-pharmaceutics of Y90-microspheres was prepared in department of nuclear medicine, first. Plastic bottle of sterilized Y90-microspheres was transported to angiographic room carefully. In secondary therapeutic angiographic procedure, all of equipment, floor and bed should be covered by water-proof paper to avoid contamination of Y90 with β particle emitter. All of operative staff should wears prevent clothes, either. Radiation safety officers in hospital will check contamination during procedure. After procedure, patient was acquired bremsstrahlung imaging to identify location of Y90-microspheres in nuclear medicine.

After successful technique, the patient received regular hemodialysis in the next two days. Because of primitive experience, independent dialysis space was arranged carefully. Let other patients and nursing staff as far away as possible. Radiation safety officers provide environmental exposure dose evaluation. We measure one-meter distance exposure dose (X ray) and contact skin surface dose of patient by counter. Potential contamination in dialysis machine was survey as detail as possible.

About 4 weeks after treatment, follow-up FDG PET/CT scan demonstrated a dominant reduction of FDG avid metastatic mass in the left hepatic lobe. Unfortunately, the patient was expired due to infection at the 6th week after treatment.

Results

Treatment Response on FDG PET/CT

In figure 1, there is significant reduced FDG avidity in left hepatic lobe with previous Y90-microspheres delivery. Treatment response is demonstrated between pre and post treatment imaging of FDG PET/ CT.

In Angiographic Room

Base on exact preparation of radiation protection as standard protocol and careful catheter procedure by experienced radiologist, there is background environmental activity during counter measurement. In figure 2, angiographic therapeutic procedure and radiation protection equipments were shown.

Bremsstrahlung Imaging

In figure 3, Bremsstrahlung imaging (energy setting around 100 kev and acquisition by Angle gamma camera) demonstrated braking radiation of spectrum X ray from β particle emitter of Y90 via human body (water). Successful localization of microspheres in liver was identified on imaging.

In Hemodialysis Room

Space

The patient was arranged to stay in a corner of dialysis room and away from other patients and nursing staff as possible in our primitive experience. Figure 4 shows the environmental radiation detection.

Measurement

Dose distribution of one-meter distance exposure and skin surface was shown as figure. The maximum dose rate was found to be 0.15 mR/h at one meter distance from body surface and 5 mR/h at patient's chest surface respectively.

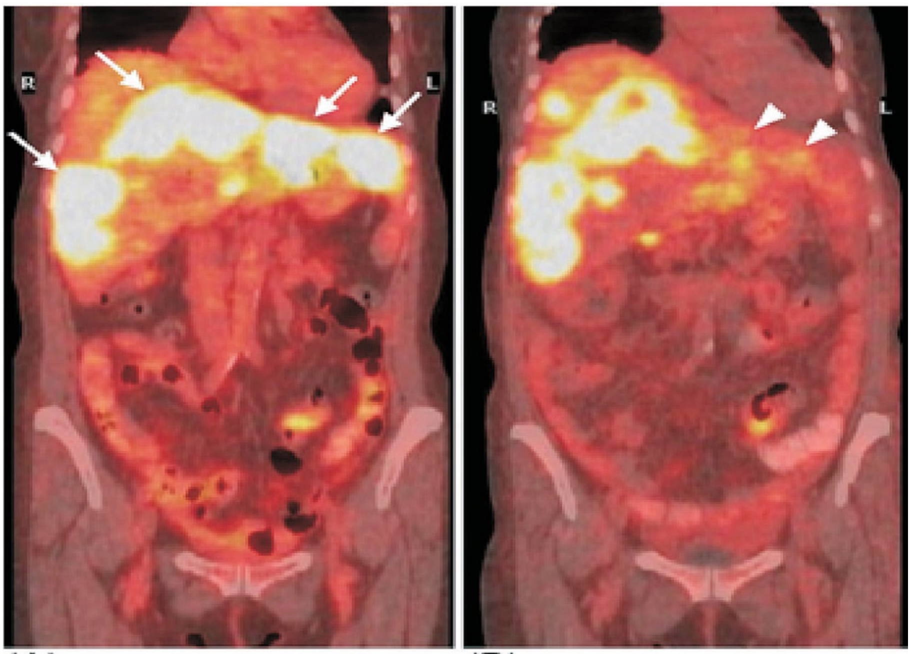
Base on bremsstrahlung imaging and direct counter measurement during hemodialysis, low energy X ray is still a source of external exposure for non-occupation dose related medical staff. Therefore, they should be aware of principle of radiation protection when they care patient closely. We consider lead-cloth wearing is not necessary.

Base on sterilization, hemodialysis equipment is closed system. Therefore, partial Y-90 microspheres via blood stay in exchanged tube and can't permeate through membrane. There is no issue of Y90-microsphere contamination in massive diluted dialysis solution. In fact, very low activity is measured in exchanged tube after dialysis.

However, any contamination of body fluid, as blood or urine from patient with Y90- microspheres treatment should be aware of prevention.

Storage of Y90 waste in nuclear medicine

All of residual waste were double packing with plastic bags and put into cages for a period of time for decay, then discharged as Taiwan's regulation.

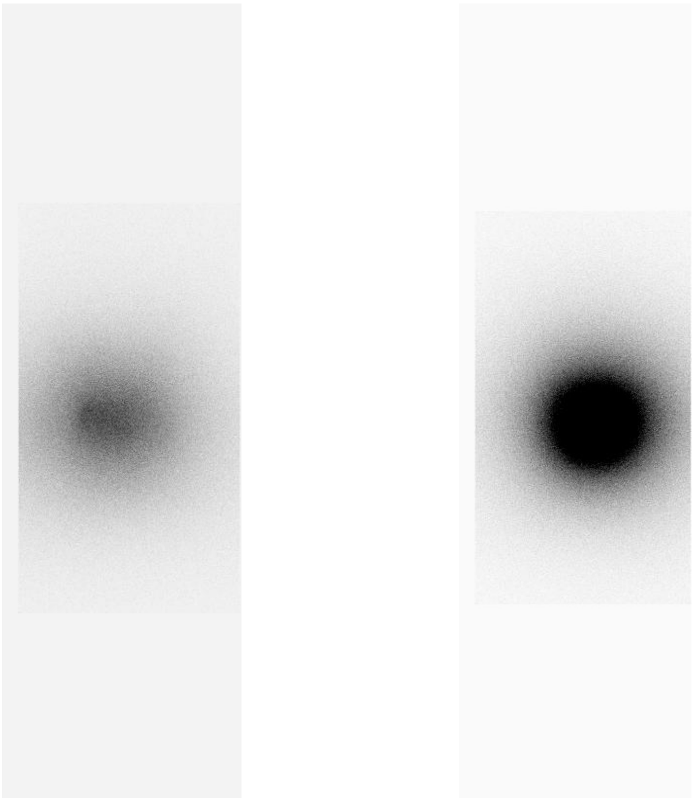


(A) (B)

Fig. 1. In comparison with pre and post treatment of FDG PET/CT imaging, significant reduced glucose metabolic activity in left hepatic lobe is seen. The imaging demonstrates treatment response of Y90-microspheres in left hepatic lobe. (A: FDG PET/CT in pretreatment; B: FDG PET/CT in post treatment).

(A) **(B)**

Figure 2. In angiographic room, Y90-microspheres are set in a plastic vial and covered by outer plastic cage for radiation protection (A). After procedure, survey of contamination is performed by radiation safety officer in KMU hospital.



(A) **(B)**

Figure 3. Bremsstrahlung imaging shows intense Y90 avidity in abdomen (liver) and demonstrates exact localization of microspheres post catheterization. (A: anterior view; B: right lateral view).

A

B

Figure 4. In hemodialysis room, measurable body surface dose around patient's head is shown (A). Surface dose is detectable around in hemodialysis exchanged tube (B).

Discussion

Y90-microsphere radioembolization treatment becomes a choice for hepatic metastasis or primary hepatocellular carcinoma in unresectable condition. More and more clinic application will happen in future day. Consideration of dose exposure of non-occupation dose related medical staff should be aware in the meanwhile. Exact radiation education and assessment of good-practice guidance of radiation protection will be helpful for new medical radionuclide treatment technologic development in clinic.

Although Y90 is a pure β emitter, low energy of X ray is still detectable in distance. There is a physic phenomenon, called "bremsstrahlung". Bremsstrahlung is also called as "braking radiation" or "deceleration radiation". It is electromagnetic radiation produced by the acceleration of a charged particle (β particle) when deflected by another charged particle, such as an atomic nucleus. Bremsstrahlung is a type of "secondary radiation" and has a continuous spectrum. The phenomenon was discovered by_Nikola Tesla during high frequency research he conducted between 1888 and 1897. In our cases, the β particles of Y90 through water of human's body will produce a spectrum of X ray. This phenomenon is demonstrated by gamma camera imaging. No matter why, there is still dose exposure measurement in body surface and one-meter distance away from surface. In comparison with radioiodine treatment, the issue of environmental dose exposure is less considerable. There is no suggestion of lead protection. In general, patient with Y90-microspheres radioembolization allow to stay a single-room ward, not radiation isolated ward. However, non-occupation dose related medical staff should be aware of radiation protection during close care of patient. There is similar protection principle when uremia patient with hemodialysis, as distance, time and shielding.

Contamination of body fluid from patient with radionuclide treatment will become an important issue in clinic care. However, we don't find significant contamination in hemodialysis device due to close system design. But there is still dose exposure in exchanged tube transiently. As our primitive experience, we suggest it seems not necessary changing dialysis schedule base on radiation protection.

Allowable doses to members of public are generally established at 2 mSv per year. Procedure of Y90-microspheres radioembolization potentially provides radiation exposure in non-occupation dose related medical staff. Education program of radiation protection is necessary for modern hospital workers to elucidate the spirit of ALARA.

Conclusions

hemodialysis is not a contraindication post Y90 microspheres radioembolization treatment. In this special medical condition, we suggest arrangement of relatively independent dialysis space and awareness of nursing care as patient with conventional nuclear medicine procedure.

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Radiographer students learning dose management of the patients

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Abstract
Radiographers have their Bachelor degree in Universities of Applied Sciences in Finland. It takes 3,5 years and is 210 ECTS. The students have 75 ECTS (2000 hours) in clinical practice in radiological departments. The students (N=21) collected patient data during the clinical practice (16 chest, 1 sinus and 4 lumber spine). The students collected data of ten patients: height, weight, sex, kV, mAs, focus to skin distance, added filtration and equipment. Each student counted for the patients ESD (Entrance surface dose) and compared them to the reference levels given by Radiation and Nuclear Safety Authority in Finland (STUK). Students managed the ESD dose calculation from air kerma (Kinetic Energy Released in Matter) very well. All have sent their signature reports with results and remarks to the department where the data was collected. Radiographers in clinical practice use these reports as a part of QA-system and clinical audit's self-assessment material.

Introduction
In Finland the Radiographers have their Bachelor degree in Universities of Applied Sciences. The Finnish credit system has been updated to comply with the European Credit Transfer and Accumulation System (ECTS) so that the Finnish credits are equivalent to the ECTS credits. The reform came into force on the 1st of January, 2005. Higher education studies are measured in credits (cr). Courses and study modules are credited according to the amount of work they require to attain the required objectives. A student's average study effort of 1,600 hours required for the completion of studies during one academic year corresponds to 60 credits.

It takes 3,5 years (210 ECTS) to get the Bachelor in degree programme of Radiography and Radiation Therapy. The students have 75 ECTS (2000 hours) in clinical practice in radiological departments having all modalities which are used in radiology including also nuclear medicine and radiotherapy. The students start their clinical practice in the end of first year in Health Centres where often only one or two radiographers are working. This offers very individual guidance to the student and possibility to plan the clinical practice together with the staff, teacher and student. The students write a reflective diary in every clinical practice period. They have also different tasks in order to deepen their skills and theoretical knowledge in some area during the practice. The tasks are more meaningful to the student if the results of it can be used in order to develop the practise, not only to make the tasks because they have to be done because of the demands of the curricula.

The principles of justification, optimisation and dose limits for the staff and general public, and the establishment and use of diagnostic reference levels (DRL) are important part of the European Medical Exposure Directive Council Directive 97/43. It has been implemented to Finnish legislation by the statement of the Ministry of Social and Health 423/2000. Dose limits do not apply to patients undergoing medical exposure but the exposures must be justified by a net benefit to the patient before making the decision about exposure. When performing x-ray examinations the exposures must be optimised by maximising radiation protection. The balance between dose and image quality taking in to account the indication for the examination must be found with every patient and every time when having an exposure to a patient. The image quality needed may be high, medium or low. (Bush 2004, International Commission on Radiological Protection 93 2004.)

It is considered that patient protection can be improved by continuously comparing local practice with the dose reference levels of doses to patients (International Commission on Radiological Protection 73 1996). Leitz (2003) believes in a five percent dose reduction, which means 800 manSv/year in Scandinavia. All measured doses should be saved together with the individual exposure data (kV, mAs, grid use, image receptor system and the generator). This is important because the dose received by a patient should be estimated afterwards (Sosiaali- ja terveysministeriö 2000) and according to many studies, the effective dose as well as ESD varies widely - even in the same examination and with similar imaging receptors (e.g. Egbe et al 2009, Willis 2009, Shaefer-Prokov et al 2008).

According to Finnish Radiation and Nuclear safety Authority in Finland (STUK) the doses of the patients must be monitored either as ESD (Entrance Surface Dose) or by DAP (Dose Area Product). The Entrance Surface Dose (ESD) can be defined as the absorbed dose to air at the point where the x-ray beam enters the surface of the patient stressed by the backscatter factor (from 1.29 to 1.46). The backscatter factor depends on the quality of the radiation beam (energy filtration of the x-ray beam and field size) (Harrison 1983, Tapiovaara 1985, Grosswendt 1990, Cranley et al. 1991, European Commission 1996, Harju et al. 2003). The unit for Entrance Surface Dose is mGy. The Dose-Area Product (DAP) is the absorbed dose in the air averaged over the area of the radiation beam multiplied by the area of the beam in the plane where dose is measured. It is normally given in $Gycm^2$ or $mGycm^2$. The dose-area is the same wherever in the beam it is measured as long as the chamber receives no scatter from the patient. Before use, the DAP meter must be calibrated. (Le Heron 1992, Toivonen et al. 2001)

The doses are monitored as an average of ten standard size patients at least every third year for typical examinations. Every year must be done some estimation. The normal size of adult patient is said to be 70 kg +/-15 kg. (STUK 2004.) These levels dose reference levels are not expected to be exceeded for standard procedures when good and normal practice is applied (Council Directive 97/43 Euratom). The dose reference level corresponds to the 3^{rd} quartile, 75% of individuals received a dose less than this value, i.e. these levels are expected not to be exceeded for standard procedures when good and normal practice regarding diagnostic and technical performance is applied (Office for Official Publications of the European Communities 1996, Council of the European Commission 1997). If the doses are too high, in the upper part of dose range, patients may get unnecessarily high doses and if doses are in the lower part, the image quality may be poor (Papadimitriou et al. 2001). It is therefore important to assess image quality in relation to patient dose and be able to get a striking and significant dose reduction.

Flat panel detectors (FPD, direct radiography, DR) offer possibilities for dose reduction but still having at least as good image quality as with computed radiography CR (e.g. Willis, Slovis 2004; McEntee et al 2006; ICRP 93 2004, Bush 2004). According to studies (e.g. Al Khalifah, Brindhaban 2004, Lanca, Silva 2008), lower levels of radiation dose settings can be established also for CR systems than the standard values used for film/screen systems to produce diagnostic images of equal quality.

By monitoring the doses regularly, the creping of doses higher (to reach the better image quality) can be found. The dose reference levels can also be reset on a lower level when having new, more effective detectors (Uffman et al 2008). This means higher DQE (Detective Quantum Efficiency) with better MTF (Modulation Transfer Function).

Material and methods

The first year radiographer students (N=21) collected patient data during the first six weeks or during second three weeks clinical practice in 21 health centres' radiological departments. The focus in these clinical practice periods are on skeleton and chest project (plain) examinations. Mostly the departments used were small, only one or two radiographers on each. In these departments the radiologist visited once a week or rarely. The teleradiology connections were available in all places in order to get consultation when needed. In two departments were ten radiographers and one or two radiologist working all the time. In these places also ultrasound examinations and mammography examinations were made. A physicist was only consulted when needed, usually once a year or less.

The dose monitoring as been regularly done in these departments at least for ten years and the clinical audit has been done once or twice in each place. Each student planned together with the tutoring radiographer and the mentoring teacher, about which plain (projection) examination the material would be most useful to collect. First, because the student needs at least ten patients coming to certain examination, and secondly because the goal is that staff can use the material as an official part of self-assessment for clinical audits. It is useful to have dose monitoring from quite common examination and not from same one every year. The dose reference levels are set to the most common examination for adults (Table 1) and some for children and also for computed tomography (CT) and cardiac interventions (Säteilyturvakeskus 2008).

Table 1. Dose reference levels for common project examinations for adult patients given by STUK 2009.

Projection	ESD / projection mGy	Projection	DAp of the whole examinationGyCm2
Chest PA	0,2	Chest pa+lat	0,4
Chets lat	0,8		
Lumbar spine ap	5	Lumbar spine ap+lat	6
lumbar spine lat	15	Abdomen ap or pa	3
Pelvis	5		
Urography / exposure	5	Urography	20
Abdomen ap or pa	5	Colongrafia	50
Mammogram CC, MLO LAT	10	OPTG	0,12
one tooth	5		

The students collected data of ten patients with a data collection sheet. The information about patient’s height, weight and sex was recorded for later use in calculation of effective dose to the same patients. Imaging technique including tube voltage (kV), tube current (mAs), size of exposure area, focus to skin distance, amount and material of inherent and added filtration and was recorded. The air kerma with different kVs were found from the measurements made yearly by the technical staff. Also the use of added filtration was taken account in these measurements (Figure 1).

Säteilyn tuotto eri suodatuksilla (arvot uGy,) isofokus
10 mAs, 100 cm 20x20cm

kv	OmmAl	2mmAl	0,1Cu+1mmAl	0,2Cu+1mmAl
40	111,8	49,7		
50	224,9	117,4		
60	353,7	201,0		
70	497,0	297,0		
80		407,5		
90		536,3		
100		676,5	516,6	373,0
110		829,2	650,4	487,5
120		993,3	796,3	612,4
130		1163	950,4	744,6

Mitattu
7.4.2010

Figure 2. Measured Air kerma with different kVs and added filtration .

Backscatter factors used in calculations were found from STUK’s publication säteilyaltistuksen määrittäminen stuk tiedottaa. If DAP meter was in use, they recorded also the dose given by DAP. The exposure indexes were recorded in order to analyse later the connection between exposure index and patient dose.

Table 2. BSF in different projection (Säteilyturvakeskus 2004).

Examination, projection	BSF (Back Scatter Factor)
Abdomen AP	1.40
Lumbal spine AP	1,35
Lumbal spine Lat	1,34
Lumbal spine L5-SI lat	1,28
Chest lat	1,40
Chest PA	1,41
Pelvis ap	1,40
Skull	1,27

Each student calculated ESD for ten patients on the bases of collected information of the patient and technical values. The doses were compared to the dose reference levels given by STUK. The whole data consist of 16 chest, one sinus and four lumbar spine examinations produced to adult patients in 21 health centres.

Results

On the point of view of the calculated doses all patients were not 55-85 kg in weight, as recommended. Still the average weight was in the recommend range. The chest x-rays were taken with high voltage technique 125-133 kV. Inherent filtration of the x-ray tubes varied from 2,8 mmAl to 3,2 mmAl. Added filtration between 0.1 -0.3 mmCu+1 mmAl was used in eleven (11) places. In five (5) wall stand equipment the added filtration was not available. The doses varied in chest pa (posterior-anterior) from 0,06 mGy to 0,23 mGy and in lateral view 0,08 mGy to 1,3 mGy. The dose reference levels are 0,2 mGy to chest pa and 0,8 mGy to chest lateral view. LÄHDE

In sinuses the lowest dose was 2,28 mGy and highest 2,69 mGy. There was no possibility to use added filtration (inherent filtration was 3 mmAl) and 85 kV was used to all patients. The dose reference level to sinus pa is 5 mGy for adult patient.LÄHDE

In lumbar spine ap doses varied from 1,1 mGy to 6,5 mGy and in lateral from 2,0 to 8,6 mGy, which are under the dose reference levels 6 mGy and 15 mGy. In lumbar spine ap the kV varied between 75 to 83 kV, and 2 mmAl added filtration was used. In lumbar spine lateral kV varied from 85 kV to 98 kV. In one place they used 0,1mmCu+1mmAl as added filtration, in other places only 2 mmAl.

On the point of view of learning dose calculation, all students were able to calculate doses. The formula which was used is

$$ESD = BSF*(h_c/h_p)^2*Q*Y$$

where Q is the output of x-ray beam at distance h_c, BSF is backscatter factor, h_p is the distance from the focus to patient skin and the Y is the mAs used in examination (Parry et al. 1999, Toivonen et al. 2001). The formula gives a reliable result if data collection is made carefully (Toivonen et al. 2001, Kepler et al. 2003).

The most asked questions concerned about the technical measurement sheet (figure 2 and Backscatter factors.

If the air kerma was not measured exactly for the used kV, the student had to draw a graph and take the air kerma /mAs from that. The most difficult information to find out from the technical measurement sheet was the distance used in measurements and what mAs was used in measurements.

The tables with the information of patients and doses were mostly done correctly. Doses were also compared to dose reference levels given by STUK 2009. If the patients were out of range of 55kg - 85 kg the student made a comment of that and discussed the effect to doses based on a larger patient group

Table 3. Example of one students data collected in one health care centre.

	ESD-pa/mGy	ESD-lat/mGy	ESD-yht/mGy	M/N	Height/cm	Weight/kg
Patient 1.	0,071	0,199	0,27	M	187	69
Patient 2.	0,12	0,447	0,567	M	158	57
Patient 3.	0,13	0,389	0,519	M	180	80
Patient 4.	0,206	0,559	0,765	M	176,5	94
Patient 5.	0,093	0,33	0,423	N	166	66
Patient 6.	0,124	0,573	0,697	M	171	78
Patient 7.	0,13	0,282	0,412	M	184	81
Patient 8.	0,132	0,556	0,688	N	158	72
Patient 9.	0,099	0,247	0,346	M	184	65
Patient 10.	0,133	0,309	0,442	M	172,5	74
Average	**0,1238**	**0,3891**	**0,5129**		**173,7**	**73,6**

ESD-PA –min: 0,071 mGy ESD-lat –min: 0,199 mGy
ESD-PA –max: 0,356 mGy ESD-lat –max: 0,573 mGy

Each report had the information of the X-ray department, equipment used with technical information and time schedule of data collection and student's own personal information. Before this official paper was sent back, teacher checked the calculations. Also the technical measurement data sheet was attached for the teacher who checked the calculations, in order to notice that the students had found all information for the calculations correctly. The reports were returned after the period of clinical practice.

Discussion

The education of radiographers is in universities of applied sciences in Finland. The Bachelor degree of radiography and radiation therapy takes 210 ECTS, 3,5 years. The studies include both theoretical studies and clinical practice, which actualises in x-ray departments all over Finland in health centres and hospitals.

According to constructive learning it is more meaningful if the learning tasks are based on real need. The students have from the first clinical practice tasks aimed to deepen some areas in the clinical practice.

Patient's dose follow up is very important part of radiographer's work and offers possibility to deepen and expand the daily work. For this task the students collect data and calculate doses for ten patients. The results are also used in the x-ray department for yearly dose follow up and as a tool of self assessment. The student has to be aware of factors affecting to ESD when collecting the data. Also the technical part, such as air kerma measurements, become familiar. he technical measurement sheet has many parameters and is not so simple to understand. The students have to understand the meaning and effects of added filtration.

The students learned very well the dose calculation and found also the reference levels and there meaning. Also the dependency between patient's size and amount of radiation is easy to point out. To find out the reasons why the doses varies in different places, the students have a discussion in small groups later. Reasons which they have found are e.g. level of used kV, speed class of the detector, use of grid, total filtration, SID (source to image distance), patient's size and size of exposure field.

The students can also compare the Finnish dose levels and technique to those in other countries. It is interesting to find differences for instance in the kV used. In some countries in chest x-rays it is high, 125-135 kV and in some others lower. ESD is reliable, easy and inexpensive way for dose follow-up (Edge 2009, Matthwes, Brennan 2009).

The latest results of research in radiography have shown that there are differences in the focus of education; some programs are more technical oriented, some more caring oriented (Akimoto et al 2009) but in any case more research and discussion of the results is needed (Malamanetiou 2009).

Conclusions

Students managed the ESD dose calculation from air kerma very well. Each student made a table including the information of the department, date of data collection, equipment used in plain examination and minimum, maximum and average doses. The doses were compared to reference levels. After teacher's comments, everyone has sent the signature report with results and remarks to the department where the data was collected. Radiographers in clinical practice use these reports as a part of QA-system and clinical audit's self-assessment material. Also the authority and clinical auditors have noticed the reports and recommend this way of learning and developing the good practices together in co-operation with the clinical staff, radiographer students and universities of applied sciences.

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Neutron field analysis for a proton therapy installation

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Abstract
A proton therapy centre is planned to be sited in Rome, Italy. It will be based on a medium energy proton accelerator and should be associated to a National Health Institute. At least two treatment station should be realized: a 140 MeV area for shallow tumors therapy and the 200-250 MeV full energy station for deep tumors treatment.

Additional experimental areas are foresee for studying the interactions of low energy, high LET, protons with tissues and the effectiveness of proton therapy on specific pathologies. The project is now in a preliminary design phase. The accelerator is under study, the building layout has to be defined and the preliminary safety solutions are considered as well in this phase. The radiation protection approach requires that all the radiation fields are well known, even those that are not useful for the treatment itself. In this frame the neutron field due to proton interactions in solid, liquid and gaseous materials has to be analyzed during the design phase in order to reduce this component to its minimum extent and to address the radiation protection program. The injector of the proton accelerator has been installed at the ENEA Research Center in Frascati, Rome and will be used to perform the preliminary low proton energy testing and the benchmarks for the simulations needed to assess the neutron field. In the paper the simulation models and the calculation performed with Monte Carlo codes are described. The related results are presented together with the comparison with the low energy benchmarks and the data found in the literature for similar projects. Considerations about workers and patients protection are issued taking into account different technical options and related advantages and disadvantages.

Introduction
In Italy a new project is close to be launched with the aim of realizing an innovative proton therapy facility. The TOP-IMPLART project is based on the use of a linear accelerator for producing the beam. TOP-IMPLART are the acronym of Terapia Oncologica con Protoni (Oncological Therapy with Protons) and Intensity Modulated Proton Linear Accelerator for Therapy. The project benefits of previous studies, designs

and tests done under the funding of the TOP project carried on in the years 1998 – 2005 by ENEA and ISS (The National Institute of Health). The TOP-IMPLART project has recently been considered suitable for a substantial funding by the Department of Innovation of Regione Lazio. A final decision about this is expected to be taken in the very next months. The work will be carried on in collaboration between ENEA, ISS and IFO (Istituti Fisioterapici Ospedalieri, the most important oncological hospital in Rome). The base of the TOP-IMPLART project is substantially the same as the TOP Project. The accelerator, that constitutes the main peculiar characteristic of this design is a linear accelerator, or , better to say, a sequence of linear accelerators. The low energy part is a commercial 7 MeV proton linac produced from AccSys-Hitachi, that itself is a sequence of a source, a RFQ and a DTL operating at the frequency of 425 MHz.

The project is aimed to develop a proton irradiation facility that could be devoted to different applications taking advantage of the modular nature of the linear accelerators. Using a linear machine instead of a compact circular accelerator (syncrotrons and cyclotrons) permits the possibility to proceed by steps in the construction and operation process and makes it possible the combined use of different irradiation stations at various energies between the minimum (about 7 MeV) and the maximum (about 250 MeV). The sequential setup of each partial irradiation module and its clinical application also before the whole facility has been completed will match the financial support flux that will be discontinue and spread over many years. This process will provide clinical and social advantages in a shorter time than the one that should be required for the construction and operation of the whole facility. The first 7 MeV module of the accelerator, is already installed and has been tested at the ENEA Research Center in Frascati, Rome. Additional modules will be added to the injector leading proton energy to 30, 70 and 150 MeV in a step by step project. The present study is finalized to the neutron field analysis for the radiation protection of the workers involved in the testing activities of the first two modules of the linear accelerator, with a final proton energy of 30 MeV. The main irradiation model considers the proton beam hitting a cell-culture-in-water target. The study has been performed using FLUKA code [1], a powerful computer program based on the Monte Carlo method, implemented to simulate the experimental setup in order to evaluate the neutron field parameters.

Material and methods

The TOP LINAC concept

Usually proton linear accelerators used for research purpose, operate with intense beams, run at low frequencies and must therefore have large apertures. This setup is unnecessary for tumor treatments, where very weak intensities are required (the average beam currents are only about 10 nA), and therefore an high frequency technology is applicable. The idea of using 3 GHz structures with gradient of the order of 15 MeV/m is at the basis of the studies initiated in 1993 by the TERA Foundation. This machine was the first 3 GHz linear accelerator dedicated to proton therapy [2, 3]. It has been designed to produce a continuously variable energy beam, promoting a xyz scanning. This requires a pulse repetition rate as high as possible (400 Hz) to irradiate a single pixel at least twice during a 1 min treatment, and a precise control (2-3 % in a relative intensity range of 50-100) of the dose rate of the single pulse. The fine energy variation is achieved by amplitude variation in the last module still under power, giving a pretty smooth energy variation with a small energy spread (about 0.3%) in the range above 130 MeV. Xyz scanning totally avoids to use passive systems such as absorbers, scattering foils and collimators, whose thickness and shape need to be designed and milled for each tumor shape and volume to provide the desired angular and energy distribution of the beam.

The cost-effectiveness of this facility is a particularly relevant issue. In fact, many cost-effectiveness reviews suggest that at the moment no clear evidence can be drawn either in favor or against proton radiotherapy as opposed to conventional photon radiotherapy. Some authors suggest a net advantage of proton therapy for a limited number of tumor sites, such as melanomas and others ocular tumors, skull base chordomas and chondrosarcomas, medulloblastoma in pediatric patients [4, 5]. For other pathologies such as breast, prostate, head-and neck tumors, similar evidence has been reported for selected patient subgroups [6, 7, 8]. A cost reduction in building proton therapy facilities equipped with robotic systems for patient positioning instead of rotating gantries, is expected to reveal more clearly the clinical advantage of proton versus photon therapy demonstrating improved dose distribution.

IMPLART Frascati test facility and related models

The test facility that will be sited in Frascati will be located inside a narrow, 13 m long shielded room, also referred as IMPLART bunker. The bunker is sized to accommodate the first two accelerating sections up to 30 MeV, with an average beam current of about 10 nA. Geometrical dimensions and shielding material compositions, as derived from the FLUKA database, are shown in the tables from 1 to 4.

Table 1. Shielding materials.

Shielding	Material	Thickness [cm]
Ceiling	Concrete	50
Wall on each side of the beam	Concrete	100
Additional shielding	Lead	10
Wall in front of the beam	Concrete	150
Wall opposite to the beam	Concrete	100
Floor	Steel	2+2

Table 2. Composition of concrete, ρ = 2,34 g cm^{-3}.

Element	Atom fraction
C-Carbon (Z=6)	23,0
O-Oxygen (Z=8)	40,0
Si-Silicon (Z=14)	12,0
Ca-Calcium (Z=20)	12,0
H-Hydrogen (Z=1)	10,0
Mg-Magnesium (Z=12)	2,0

Table 3. Composition of typical stainless-steel, ρ = 8,0 g cm^{-3}.

Element	Atomic fraction
Cr-Chromium (Z=24)	18,0
Fe-Iron (Z=26)	74,0
Ni-Nickel (Z=28)	8,0

Table 4. Composition of kapton membrane, ρ = 1,42 g cm^{-3}, thickness 80 μm.

Element	Mass fraction
H-Hydrogen (Z=1)	0,026362
C-Carbon (Z=6)	0,691133
N-Nitrogen (Z=7)	0,07327
O-Oxygen (Z=8)	0,209235

FLUKA Monte Carlo code has been implemented to simulate a 30 MeV proton beam hitting a cell culture in water enclosed in a Petri's capsule and positioned on the beam axis at about 2 cm from the kapton membrane located at the end of the vacuum accelerating channel. 10^7 particle stories were used in the Monte Carlo execution. The calculation model includes the following main sections:

- Final segment of the accelerating section;
- Petri's-capsule target;
- Shielding walls.

The following kind of results were obtained:

- Neutron fluences inside the bunker, inside the shielding and outside the bunker;
- Neutron dose equivalent inside the bunker, inside the shielding and outside the bunker;
- Double differential neutron fluence spectrum inside the bunker;

These results allow a preliminary assessment of the effectiveness of the existing shielding in relation to the neutron fields. The final segment of the accelerating section has been shaped like a stainless-steel hollow cylinder, with vacuum inside, end-closed by a kapton membrane. The target is a Petri's capsule: a glass or plastic cylindrical box that is the most common container for cell cultures. The simplified model consists of an hollow cylinder containing three layers of material. The hollow cylinder simulate the Petri's capsule with an air-equivalent-plastic (ICRU C-552); the two outer layers simulate the water while the inner layer simulate the cell culture with a tissue-equivalent material, as shown in figure 1.

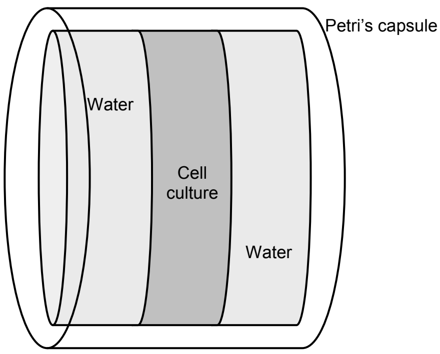

Fig. 1. Schematic view of the target (not in scale).

The tables from 5 to 7 show the FLUKA material composition of the target.

Table 5. Composition of the air-equivalent-plastic (ICRU C-552), ρ = 1,76 g cm^{-3}.

Element	Mass fraction
H-Hydrogen (Z=1)	0,02468
C-Carbon (Z=6)	0,50161
O-Oxygen (Z=8)	0,004527
F-Fluorine (Z=9)	0,465209
Si-Silicon (Z=14)	0,003973

Table 6. Composition of water, ρ = 1,0 g cm^{-3}.

Element	Atom fraction
H-Hydrogen (Z=1)	2,0
Oxygen (Z=8)	1,0

Table 7. Composition of tissue-equivalent material, ρ = 1,0 g cm^{-3}.

Element	Mass fraction
H-Hydrogen (Z=1)	0,101866
C-Carbon (Z=6)	0,10002
N-Nitrogen (Z=7)	0,02964
O-Oxygen (Z=8)	0,759414

Results

FLUKA calculations provide results in term of quantity per single source proton that have to be scaled accounting for the actual particles flux in the beam. The data obtained in the following will be than multiplied by the proton-per-hour yield of the Implart modules, that is about 2,25 10^{14} h^{-1}, giving the dose rate in the same time unit.

In the figures from 2 to 5 the final segment of the accelerating section (thin strip) and the target are located at the right, inside the bunker. In the figures 2 and 4, the thin strip below outside the bunker represents the second steel layer. In the figures 3 and 5, the thin sector at the top, inside the bunker, represents the additional lead shielding.

Figures 2 and 3 show neutron fluence due to the proton interaction on the system target as a result of the FLUKA run.

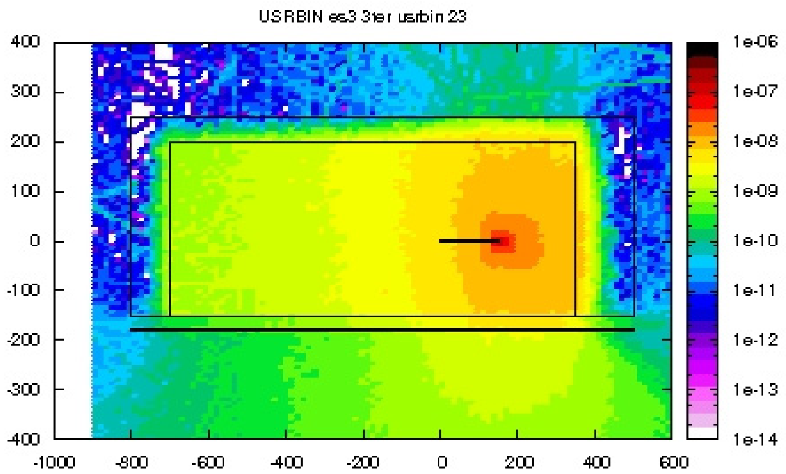

Fig. 2. Spatial distribution of the neutron fluence. YZ plane (vertical section). The abscissas and the ordinates show the linear dimensions in cm, while the spatial distribution is represented by color scale in unit of part/cm^2. The result is normalized per source-particle or primary.

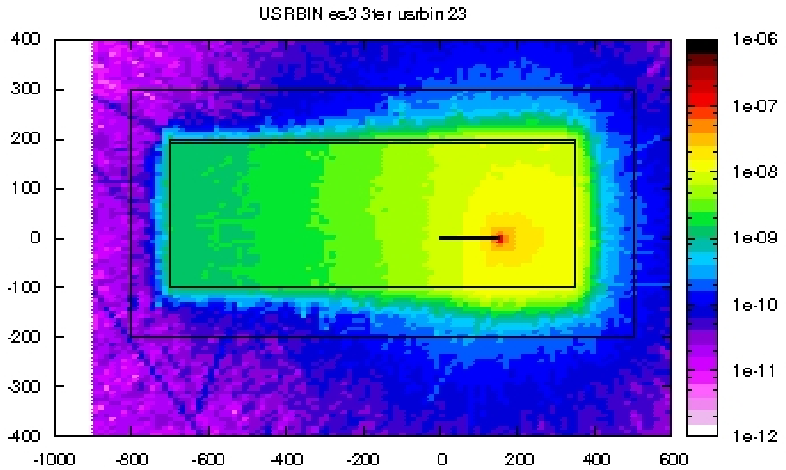

Fig. 3. Spatial distribution of the neutron fluence. XZ plane (orizontal section). The abscissas and the ordinates show the linear dimensions in cm, while the spatial distribution is represented by color scale in unit of part/cm^2. The result is normalized per source-particle or primary.

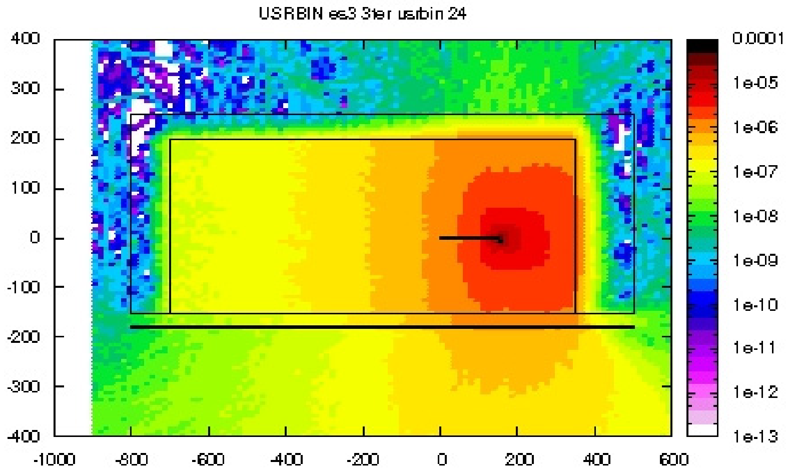

Fig. 4. Spatial distribution of the neutron ambient dose-equivalent. YZ plane (vertical section). The abscissas and the ordinates show the linear dimensions in cm, while the spatial distribution is represented by color scale in unit of pSv. The result is normalized per source-particle or primary.

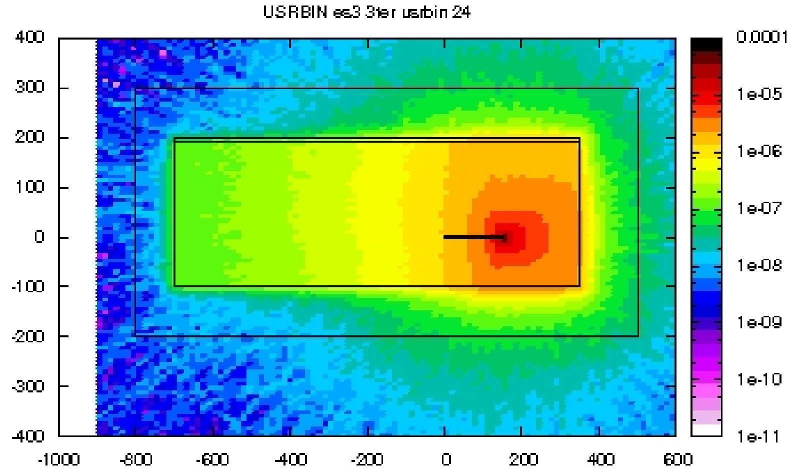

Fig. 5. Spatial distribution of the neutron ambient dose-equivalent. XZ plane (orizontal section). The abscissas and the ordinates show the linear dimensions in cm, while the spatial distribution is represented by color scale in unit of pSv. The result is normalized per source-particle or primary.

In the Figures 4 and 5 the main result of the simulation is shown. The neutron ambient dose-equivalent is normalized to the single source proton. In the chromatic graphs the walls of the bunker are reported in order to assess the related shielding effect.

A general result is represented in Figure 6 showing the neutron fluence spectrum in air, at 10 cm from the target.

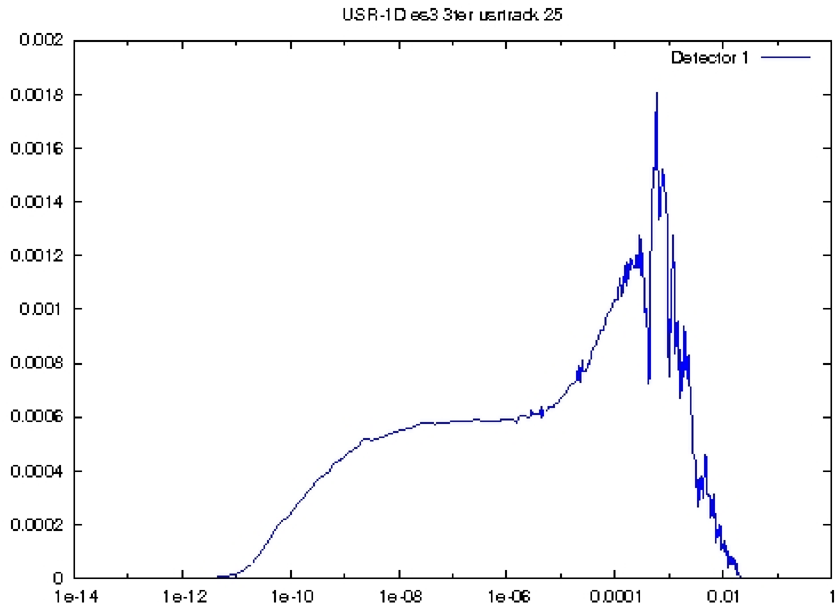

Fig. 6. Differential distribution of the neutron fluence. The abscissas show the energy in unit of [GeV], while the ordinates show the differential distribution of the fluence dΦ/dE in unit of [cm^{-2} GeV^{-1}]. The result is normalized per source-particle or primary.

Discussion

The neutron field is generated by interaction of the proton beam with the nuclei of the target, the air and the shielding. It represents an inevitable and undesirable radiation field which could cause undue dose to the operators.

Figure 2 shows the neutron fluence in vertical projection (YZ plane):

- around the target we have about 1e-07 part cm^{-2};
- inside the bunker it ranges from 1e-08 to 1e-09 part cm^{-2}, depending on the distance from the target;
- outside the bunker it varies from 1e-11 part/[cm^{-2}] beyond the concrete, to 1e-09 part/[cm^{-2}] in the room below the steel layers;

Figure 3 shows the situation in a horizontal projection (XZ plane). The fluence shows a general behavior similar to that shown in the vertical plane, with a slightly lower intensity.

The neutron ambient dose equivalent shown in Figure 4 gives the trend of the field in vertical projection (YZ plane):

- around the target, is about 5e-05 pSv (i.e. 11,2 mSv/h, for a 10 nA current);
- inside the bunker it varies from about 5e-06 to 1e-07 pSv, depending on the distance from the target;
- outside the bunker it can be seen as it varies, in the points of greatest field strength, from about 5e-08 pSv (i.e. 11,2 μSv/h, for a 10 nA current) beyond the concrete, to 1e-06 pSv (224 μSv/h, for a 10 nA current) in the room below the steel layers;

Figure 5 shows the situation in the horizontal projection (XZ plane).

The difference from fluence to dose-equivalent is due to the application of the conversion coefficients that for neutrons are strongly dependent from their energy.

In the case study the neutron energy has the spectral trend shown in Figure 6. The curve shows that the neutron fluence spectrum has the main peak at about 0,598 MeV, then decreases rapidly dying at about 21 MeV and also shows a tail starting from the thermal region at about 4,361 meV with a plateau from about 1e-06 MeV up to 1e-03 MeV. The thermal component could be caused by slowing of fast neutrons in concrete and subsequent backscatter inside the bunker.

Conclusions

The TOP-Implart project could provide important medical applications suggesting the needing of testing the first sections of the accelerator in an existing facility. The current work describes the initial analysis of the neutron field produced by the 30 MeV proton beam provided by the first two accelerating sections.

The preliminary result obtained with the FLUKA simulation shows that the existing shielding walls and structures, effective in reducing the dose rate at an acceptable level for the initial module of 7 MeV protons, have to be improved to keep the dose rate at a similar low level for the upgrade to 30 MeV.

In the simulation wall shielding appears to be able to slow down the dose-equivalent by a factor of about 10 only. A better attenuation seems to occur for the neutron fluence due to the neutron spectrum and the conversion coefficients distribution vs. neutron energy. Anyway the dose rate assessed in the simulation is greater than 10 µSv/h in many areas beyond the shielding walls and this would limit the system workload at less than 1000 hour per year.

In conclusion the simulation model has been setup but further analysis is needed to correctly size the shielding walls or to better define distance and materials. Possible local shielding or beam dumps have to be considered too. If needed, also the accelerator workload and average beam current could be reduced during the testing phase.

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Where is Bulgaria in radiation protection in medicine?

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Abstract

Medical uses of ionizing radiation are the major man-made source of exposure of the Bulgarian population. The requirements of the International Basic Safety Standards for protection against ionizing radiation and the EURATOM Directive 97/43 were fully harmonized in the Bulgarian legislation and many actions have been undertaken for their practical implementations. In October 2005 new regulation has been enforced for radiation protection at medical exposure. A department for Radiation Protection at Medical Exposure was established at the National Centre of Radiobiology and Radiation Protection in order to lead the practical implementation of the Ordinance. New objective requirements to radiological equipment were set and licensees were obliged to put into operation Quality control program. First Diagnostic Reference doses were given in 2005 based on the limited own experience from the First national survey in conventional performed in 2002–2003 within the Bulgarian-German twinning project. Further enlargement of dose surveys and improvement of RP practice at medical use of radiation were realized within the Bulgarian-Finnish Twining project “Strengthening of administrative structures for radiation protection and safety use of ionizing radiation in diagnostics and therapy”. New radiation protection training programs for medical specialist were elaborated including recent knowledge on patient protection. Several national training courses with international lecturers were performed.

A national system on registration and reporting of medical exposure in Romania

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Abstract

The 97/43/EURATOM Directive was fully transposed into Romanian legislation, starting with the "Norms on radioprotection of the public from medical exposure", which were joitly approved in 2002 by the minister of health and the president of the nuclear regulatory authority (C.N.C.A.N.) and also by other specific regulations regarding radiation protection of the patient. As in most European countries, the medical exposure is the main man-made radiation exposure of the population in Romania. It is now an alarming increase in use of high radiation dose examinations , such as computed tomography (CT) and of X-rays guided interventions (like coronary angioplasty), with similar doses and sometimes even higher (particularly to the skin). As part of the implementation plan of the 97/43/EURATOM Directive, a national system on registration and reporting of medical exposure was recently developed in Romania. The system is establishedby two specific regulations of the Ministry of Health, approved by the ministerial orders No.1542/2006 and No.1.003/2008. It refers to all medical radiological procedures: diagnostic radiology, interventional radiology, nuclear medicine and radiotherapy. The system includes three main parts :

– registration (electronic or on files) of basic data (patient data , exposure parameters and individual patient doses, if available), at the level of each user (X-ray machine, nuclear medicine department and/or radiotherapy facility);
– on-line data reporting to the Radiation Hygiene Laboratory (RHL) , belonging to the local Direction of Public Health and to the specialized network of the Ministry of Health (quarterly);
– on-line reporting of data , by each RHL, to the Institute of Public Health-Bucharest (annually). The Institute of Public Health – Bucharest has to evaluate all data and to issue in June 2010 a first report on data from 2009.

Analysis of doses in x-ray computed tomography

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Abstract

Computed tomography is known to be relatively high dose diagnostic procedure. The need to justify, to optimize, and to evaluate the exposure in tomography examinations is therefore widely accepted. This is especially true for children, who are more sensitive to radiation, and for whom unnecessary overexposure is possible in case of non-optimized scanning protocols. The ALARA principle should, however, be obeyed for all patients.

In this study dose related data were collected for patients examined on several different CT units. The patients included adults and children of various age. The examinations were done either on purpose of diagnosis or for planning of radiation therapy.

Collected data were analyzed in order to establish, whether the parameters of examinations were routinely adjusted accordingly to patient size, especially in case of children. Doses, expressed in terms of $CTDI_{vol}$ and DLP, were compared with reference levels published by the European Commission and by the NRPB. Effective doses were estimated using αconversion factors published by NRPB, and then used for comparison between CT units and groups of patients.

The analysis helped to reveal situations, in which scanning protocols were clearly non-optimized as far as radiological protection of patients is concerned.

Radiographers' safety culture in medical use of radiation

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Abstract

Safety culture is one of the stone basis in medical use of radiation. Safety culture means that organisation and its personnel ways of action and attitudes are so that maintaining and continuous developing of safety is possible. Safety culture in medical use of radiation has been refered mostly theoretically but not empirically. The purpose of this study is to describe and interpret radiographers' safety culture in the medical use of radiation. Another objective is to yield new information and highlight inductively characteristics of safety culture. The data were gathered from two university hospitals and one central hospital. The data consisted of interviews (n = 20), fieldwork, articles in a professional journal (n = 457) and documents used by a radiographer in his work. Focused ethnographic studies and discourse analysis were drawn on to interpret radiographers' shared meanings of safety culture.

Radiographers' safety culture consisted of four shared meanings: challenges of knowledge and skills structuring safety culture, dimensions of cooperation enabling safety culture, disorientation conditioning safety culture and multidimensional professionalism as the foundation of safety culture. The use of technology was considered an essential part of a radiographer's work, contributing to patient care. The radiographers' varied assignments, their conventional opinions, the work and roles of both head radiographers` and physicians, and their responses to current social challenges in modern working life, all were seen as contributing to cooperation.

Protection against radiation was understood to be an important part of the radiographer's standard activities. Radiographers were confused by the different directions and practices in implementing radiation protection. The results of the study can be used in developing and organizing radiographers' work and education. They may also clarify safety culture as a phenomenon and a concept.

Ethical dilemmas in radiographer's work in diagnostic radiology

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Abstract

Purpose: The purpose is to describe ethical dilemmas in diagnostic radiographer's work. The aim of the study is to create new information since there is comparatively little research focusing on this subject.

Methods and materials: Data was collected during spring and summer 2008 and consisted of thematic interviews of diagnostic radiographers (N=8), whose working experience varied from 4 to 31 years. Data was analysed with qualitative inductive content analysis.

Results: Ethical dilemmas were found to consider the use of radiation and radiographer's work community. In the use of radiation, implementation of justification and optimisation principles were found to be lacking. There were problems in the safe use of equipments and especially lacks in referrals. There was neither found common safety culture, because the projections, number of them and dose optimisation was insufficient. Also the use of lead shields varied individually and although it was noticed that some colleges didn't use them, there was no discussion about the common practice. Dilemmas in work community consisted of problems among employees and insufficient practice. Background factors of these dilemmas were found to be both dependent on and independent of the employee, resulting in worsened well-being at work and seek for change. Current processing methods of dilemmas were found to be insufficient, and suggestions for better processing methods were made.

Conclusions: Daily ethical considerations were described by respondents, but not recognised as ethical dilemmas. Ethics in work as a concept may seem separate and distant, despite the fact that in most cases, ethical dilemmas occur in everyday situations. Recognition of dilemmas is important in order to be able to intervene in them. Respondents repeatedly described powerlessness and inability to intervene in ethical problems. Quite many of those problems can be seen as problems or even lack of safety culture.

Effects of Computed Tomography on paediatric patients: towards an understanding of gene expression profiles and the potential role of radiation induced cytokines

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Abstract

Computed Tomography (CT) is a diagnostic imaging technique that uses X-rays for assessing a variety of disorders in both adults and children. Within the patient population undergoing CT examination, paediatric patients, following interventional cardiac catheterization CT scans, showed low dose hypersensitivity. A recent study showed an *in vivo* induction of γ-H2AX foci, caused by interventional X-rays (Beels et al., Circulation, 2009). It is suspected that the mechanism behind this hypersensitivity is related to non-targeted or bystander effects.

In collaboration with multiple centres, blood samples will be collected from paediatric patients' pre-CT and post-CT; as well as blood samples from volunteers will be irradiated *in vitro*, helping us to understand the kinetics of gene expression and track possible differences between *in vivo* and in *vitro* radiation effects. Using microarrays, we will be able to investigate global gene expression, thus pointing out differential gene profiles before and after exposure to CT scans. This could lead to an understanding of the biological mechanisms behind the low dose hypersensitivity; on the other hand, we might be able to specify certain early radiation induced biomarkers, thus using it to assess the risk associated with CT exposures.

Bystander effect is the induction of DNA damage response in non-irradiated cells that are neighbouring irradiated cells, and is considered to amplify the effects of radiation by increasing the number of affected cells. Potential players that can promote bystander effect are cytokines; these are signalling molecules that are involved in cellular communication during an immune response. We will use multiplex array assay to determine levels of different cytokines following the radiation induced stress.

Part of this project is financially supported by the Federal Agency of Nuclear Control (FANC-AFCN), Belgium.

Radon associated lifetime risk

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Abstract

Quantification of the risk associated to radon exposure is a major public health issue. Since the 1970s, 13 cohort studies provided results on the exposure-risk relationship in miner populations. Since the 1990s, more than 20 case-control studies evaluated the risk of lung cancer associated to indoor radon exposure in the general population. Nevertheless, due to variations in the study designs, in the characteristics of the studied populations and in factors that modify the exposure-risk relationship (age at exposure, attained age, time since exposure or exposure rate), it is difficult to verify the coherence of published results. Calculation of lifetime excess absolute risks (LEAR) is a way to compare the results of risk coefficients or models derived from different populations when applied to the same scenario of exposure. We present here some LEAR estimates based on the most recent epidemiological results. Different models are applied, derived from both miner studies (BEIR VI, Eldorado, French-Czech models) and indoor studies (European pooling project). The background reference rates are those proposed by the ICRP. Different scenarios of chronic low rate exposure are considered, including those used by the ICRP. Results illustrate the impact of age and time modifiers on the estimated LEAR. Results obtained using different models derived from miner studies are highly coherent. A good agreement is also observed between LEAR estimates obtained from miners and from indoor studies when applied to adapted scenarios. To conclude, calculation of LEAR allows demonstrating a very good coherence in currently available radon associated risk estimates. These results provide support to the elaboration of radiation protection measures regarding radon exposure.

European atlas of natural radiations: status of the indoor and geogenic radon maps

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Abstract

Based on its political and legal mission, in 2006 the JRC started to design the project of a European atlas of natural radiations. As first step a map of indoor radon was envisaged, given the radiological importance of the related exposure pathway. Since a seminal meeting in Prague, autumn 2006, the indoor Rn map is under way, with contributions from so far (August 2009) 15 countries. Individual measurement data are aggregated into a common grid by the participants. Grid data, which consist of grid cells filled with statistics on individual data, are then further processed by the JRC. In the aggregated dataset, no information on individual houses is available. We present the preliminary results in terms of descriptive statistics and maps, and some further results based on modelling steps, like spatial risk estimate. Indoor radon concentration is controlled by geogenic, climatic and anthropogenic factors. Among the latter are house and room types and living habits. In order to assess the hazard, or potential risk of indoor Rn exposure at a location, independent of anthropogenic factors, one needs a standardized quantity which is only controlled by the geogenic (and possibly average climatic) factors. Since a start-off meeting in Oslo, summer 2008, we are developing a harmonized European Rn index which can be estimated from different input data, as available in participating countries. Among these are indoor Rn concentrations, data on Rn in soil gas, geological classes, geochemical data and external dose rate. Currently algorithms are under discussion to convert such multivariate data, different country by country, into one radon index variable. Future steps of the European atlas of natural radiations may be maps of cosmogenic exposure and geochemical maps of natural radionuclides.

The Austrian Radon Programme – Past and future

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Abstract

As early as at the beginning of the 20th century radon measurements in water were conducted in Austria whereas radon measurements in air started in 1949. Several thousand measurements of radon in air in workplaces (caves, mines) and homes were conducted until 1972 by Pohl and Pohl-Rüling. Later measurement campaigns took place in Innsbruck, Salzburg, and Vienna but it was not until 1991 that a systematic and coordinated investigation of the radon situation in Austria began.

The main effort was to establish the Austrian radon map in the framework of a project called The Austrian Radon Project (ÖNRAP) with Harry Friedmann from the University of Vienna as project leader (1992 – 2004). Other main projects were to test various mitigation techniques (SARAH, Maringer, 1996 – 1998) and preventive measures. One project dealt with the influence of building characteristics on the radon concentration and the use of a Blower Door to determine the mean radon concentration of a building (RACODE, Ringer, 1997 – 2001).

Substantial efforts were undertaken to determine the radon exposure in kindergartens, schools, and town halls including the mitigation of the buildings with elevated radon levels.

The knowledge and experience gained by the projects led to the issue of three Austrian standards for radon measurement, mitigation and prevention, and to the issue of a radon information CD and a radon brochure.

In 2006 the Austrian Centre for Radon was established with the aim to better coordinate the radon efforts. Main tasks at present are the setup and maintenance of the official radon website and the design and implementation of the Austrian radon database. Other projects deal with the radon exposure in show caves and tourist mines, the investigation of the house to house variation of radon in a community and the investigation of the causes of that variability, and the determination of radon levels outdoors.

This paper will present and discuss past and future efforts in Austria to determine the radon exposures in homes and workplaces and to reduce the radon risk of the population.

Introduction

"Bergkrankheit" (or mountain sickness), among sixteenth century pitchblende (radium) miners in the Erz Mountains of Germany, was the earliest evidence of the health hazard of inhaled radon and its progeny. However, it was not until the 1920's that the lung disease was attributed to radiation, as opposed to inhaled arsenic (Miller 1990). And it was not until the 1970's that residential exposure to indoor radon began to be considered as a potential public health risk (Nero 1988).

In 1988 Radon was classified as a human carcinogen by the International Agency for Research on Cancer (IARC 1988).

Since the 1980's substantial efforts had been undertaken in many countries to evaluate the radon problem. The annual dose from inhalation of radon gas and its decay products represents typically about one-half of the effective dose received by members of the public from all natural sources of ionizing radiation (UNSCEAR 2008).

This paper describes the studies and projects with respect to radon in Austria in the past, discusses the lessons to learn, and gives a foresight to the future radon strategy.

Radon in Austria until 1991

As early as at the beginning of the 20th century radon measurements in water were conducted in Austria (balneology) whereas radon measurements in air started in 1949. Several thousand measurements of radon in air in workplaces (caves, mines) and homes were conducted until 1972 by Pohl and Pohl-Rüling. Later measurement campaigns took place in Innsbruck, Salzburg, and Vienna but it was not until 1991 that a systematic and coordinated investigation of the radon situation in Austria began (BMGSK 1992).

Radon in Austria after 1991

At the beginning of the 1990's publications from EU, ICRP, WHO, and IAEA gave an international framework to the radon issue (EU 1990, ICRP 1990, WHO 1993, ICRP 1993, IAEA 1996).

In 1991 a project with the aim to evaluate the radon situation in Austria and to provide a concept with respect to a national radon programme was conducted (BMGSK 1992).

Based on EU 1990 and ICRP 1990 the Austrian Radioprotection Commission passed a recommendation concerning 'action levels for indoor radon' in 1992 (BMGSK 1994). Key elements of that recommendation are

- action levels for existing and new buildings (400 and 200 Bq/m^3, respectively)
- the recommendation to set up a radon potential map of Austria
- the recommendation to set up guidelines for mitigation and prevention
- the recommendation to set up a folder to inform the population

In the following legislative matters and the major projects and studies are outlined.

Radon in Austrian legislation and standards

Indoor radon is dealt with in the recommendation of the Austrian radiation protection commission as mentioned above and is not subject to specific legislation.

On the other hand, radon in work places is regimented by the Radiation Protection Law 2006 and the Natural Radiation Sources Ordinance 2008. The Ordinance applies to the following four working areas: water supplies, underground work places in mines, tunnels, etc., show caves and tourist mines, and radon spas. So far, the response of the concerned companies to the legal requirement to induce an evaluation of the radon exposure to the staff was very weak.

A series of Austrian Standards (ÖNORM S 5280 Radon) has been issued dealing with Part 1: Measuring methods and their range of applications, Part 2: Technical precautionary measures in the case of buildings and Part 3: Remedial measures on buildings.

The Austrian Radon Project (ÖNRAP, 1992-2004, Friedmann)

The main project was the establishment of the Austrian radon map (Friedmann 2005). Radon measurements were conducted in about 16,000 living and sleeping rooms in Austrian dwellings. The results were normalised to a standard situation (no basement, ground floor level, …) and therefore the normalised data are an indicator for the radon potential, or in other words, the probability to measure a certain radon level in a specific dwelling in a municipality. Fig. 1 shows the Austrian radon potential map.

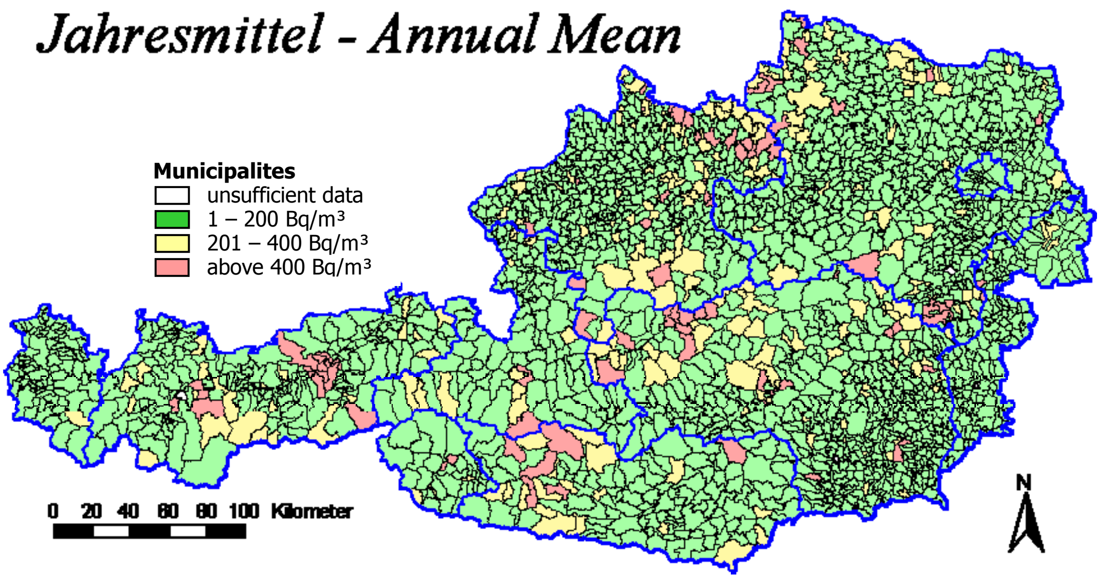

Fig. 1. The Austrian radon potential map.

Radon surveys and mitigations in kindergartens, schools, and town halls

The radon levels in about 800 kindergartens, 350 schools, and 450 town halls have been determined in Austria (mainly in the province of Upper Austria) in the last 10 years. 2,6% of all kindergartens and schools and 1,6% of all town halls showed mean radon levels above 1000 Bq/m³ and 5,9% (kindergartens, schools) and 12% (town halls) above 400 Bq/m³. In total about 130 buildings needed to be mitigated (or are in the process of mitigation since the priority was linked to the radon level) (Ringer 2002, Maringer 2004, Ringer 2004, AOOELR 2009).

Radon in low energy and passive houses (2009-2011, Ringer)

Airtight building shells, low air exchange rates and specific construction and heating technologies like air-soil heat exchanger gave rise to the concern that radon levels may be elevated in low energy and passive houses. A project is currently investigating the effect of those new building types on indoor radon. This project is a subtask of the EU-project "Radon Prevention and Remediation (RADPAR)". (http://web.jrc.ec.europa.eu/radpar/index.cfm)

Complete radon survey in three communities (2009-2011, Ringer)

The aims of the project are to study the dwelling to dwelling variation within a municipality, to evaluate the influence of building characteristics and geology, to verify the results of the Austrian Radon Project, to train local building professionals to mitigate dwellings, and to raise awareness with respect to radon within the municipalities and beyond. For that purpose the radon and thoron concentration of every single dwelling in three municipalities with elevated radon potential is to be measured. Furthermore, the soil gas concentrations in the main geological formations in this area will be determined.

After a public information event and after the distribution of the track etch detectors, presently, the long-term radon and thoron measurements are conducted (January – July 2010).

Mitigation of dwellings with elevated radon levels (SARAH, 1996–1998, Maringer)

This study served to test mitigation methods adapted to typical Austrian houses, to gain experience, and to evaluate typical mitigation costs.

Three dwellings with elevated radon levels were mitigated by applying sub-slab depressurisation, sub-house depressurisation and sub-slab ventilation, respectively, as mitigation method.

The main results are given in Table 1.

Table 1. Main results of the mitigation project.

	Gutau	Königswiesen	Traun
method	sub-slab depressurisation	sub-house depressurisation	sub-slab ventilation
mean radon concentration before mitigation (Bq/m³)	500	900	600
mean radon concentration after mitigation (Bq/m³)	250 (passive) 50 (active)	180	360
cost (EURO)	14000	7000	400
specific radon reduction costs (€/m^{2}Bqm^{-3})	0.23	0.11	0.015

Radon Potential Determination by Controlled Building Depressurisation (RACODE, 1997–2001, Ringer)

In this study a Blower Door (BD) was used to mechanically depressurise houses in a controlled and reproducible manner to determine the radon entry rate. Furthermore, diagnostic tests like "Opening A Door" or "Guard Zone" were carried out to characterise the leakage pattern (i.e. Effective Leakage Area ELA and flow exponent n) of the house. With these data and (simplified) modelling mean indoor radon concentrations are deduced and compared with the results of long term passive radon measurements.

Besides the determination of the mean radon concentration this new method should be useful for the rapid assessment of the effectiveness of mitigation measures if the same kind of measurements at defined pressure conditions are performed before and after mitigation (Ringer 2005).

Radon in work places

With respect to radon exposure in work places major studies – apart from those studies in kindergartens, schools, and town halls as described above – concerned water supplies, tourist mines and show caves.

The results show that the radon concentrations in the water supplies were lower as expected, being in 55% of all measurement sites below 1000 Bq/m³, in 91% below 5000 Bq/m³, and with a maximum value of 38700 Bq/m³. This leads to exposures that are below 2 MBqh/m³ (corresponding to approx. 6 mSv/a) in 42 water supplies. However, for the remaining three water supplies maximal occupational exposures due to radon of 2.8 MBqh/m³ (~ 10 mSv/a), 15 MBqh/m³ (~ 50 mSv/a), and 17 MBqh/m³ (~ 56 mSv/a), respectively, were determined. These water supplies were mitigated. Mitigation measures were: evacuation of the outlet air of the vaporizer by means of a fan, installation of a fan in the exhaust air duct of the compensating reservoir, sealing of drain shafts, mechanical ventilation of the office. In all water supplies the radon exposure was reduced to below 0.8 MBqh/m³ at a cost of approx. € 750,- to € 1.000,- (Ringer 2006, Ringer 2008).

For the purpose of determining the radon exposure of guides and of evaluating the determining factors on the radon concentration, such as climate, structure and geology, long-term time-resolved measurements over 12 months have been carried out in 6 tourist mines and 2 show caves - with 5 to 9 measuring points each - to obtain the course of radon concentration throughout the year. First results are presented in Ringer 2009. The final report is expected by the end of 2010.

Information of the public

A number of folders and brochures has been published to inform the public on the radon issue. One folder deals with radon in water supplies.

Furthermore, a Radon-CD was produced which presents basic information on radon in a multimedia way and a specific radon website was set up by the Ministry for Environment (www.strahlenschutz.gv.at) in addition to the existing Austrian Radon Project website (http://homepage.univie.ac.at/harry.friedmann/Radon/welcome.htm).

Presently, a centralised, web-based radon database is being developed. The purpose is to gather all indoor radon related data like measurements, mitigations, preventive measures, building characteristics etc. in one single database. These data are periodically evaluated to monitor the radon situation in Austria and to provide up-to-date information to the public.

The Austrian Centre for Radon

Based on the Radiation Protection Law the Austrian Centre for Radon was established in 2006 within the Competence Centre of Radioecology and Radon of the Austrian Agency for Health and Food Safety. The principal aim is to coordinate the radon efforts in Austria.

Present tasks are the design and implementation of the above mentioned Austrian radon database and the setup and maintenance of the official radon website. Other main projects are mentioned in this paper (radon in show caves and tourist mines, radon in low energy and passive houses, complete radon and thoron survey in three municipalities).

Discussion

As described in the preceding chapters, an important number of projects and studies has been conducted over the last 20 years in Austria. These efforts served to quantify the radon problem and to find solutions. Radon prone areas were identified, research on the factors determining the indoor radon concentration resulted in a better understanding of the relevant building characteristics, efficient remedial and preventive measures were tested and established, extensive surveys in kindergartens, schools, town halls, water supplies and show caves and tourist mines led to a reduction of the radon risk of the concerned individuals. The knowledge gained was used for legislative matters, for the setup of Austrian Standards, for information material for the public. Finally, the Austrian Centre for Radon was established to better coordinate and concentrate the radon efforts.

But: What was the actual impact on the radon exposure in Austria, how many lives have been saved?

So far, the impact in terms of radon risk reduction was very little. Priority was given to provide help (i.e. information, measurement, mitigation) to those with a high individual radon risk (i.e. high radon levels at home or at work). However, only approximately 0.07 % of the 70,000 buildings which are expected to exceed the action level of 400 Bq/m³ for existing buildings have been mitigated so far! Even worse, due to missing legal requirements to implement preventive measures in new buildings (at least in radon prone areas) the number of radon resistant new buildings is negligible. Bearing in mind that the total building stock increased by over 70 % in the last 40 years in Austria this was clearly a missed opportunity to decrease the collective radon risk.

Finally, despite folders, brochures, websites, and a radon CD public awareness is still very low.

Future tasks

Following the arguments of the previous chapter and with a view to recent publications and drafts on the radon issue (WHO Radon Handbook 2009, ICRP Radon Statement November 2009, EU BSS draft, IAEA BSS draft), the prime future tasks can be identified as:

- There is the need of a binding and executable legislation together with clear assignment of responsibilites to federal and local authorities as this is a prerequisite for an effective radon policy
- Implementation of compulsory preventive measures in building codes as this is an essential and cost-effective measure to reduce the collective radon risk
- Surveys in town halls and other public buildings in all radon prone areas to raise radon awareness of so-called multipliers (local politicians, local administrations)
- Frequent information campaigns to raise awareness of the population
- Extensive measurement campaigns to identify the buildings with elevated radon levels; as a consequence this will require training of building professionals for mitigation
- Integration of the radon issue in school and academic education (builders, engineers, architects, physicians)
- Improvement of the existing radon potential map (further indoor measurements, correlation with geology)

The future tasks should be shaped in a national strategy which also explains its implementation with respect to competent authorities and responsibilities, required financial resources, and evaluation and monitoring of the radon situation.

Conclusions

20 years of substantial efforts led to a widespread knowledge with respect to radon prone areas, factors determining indoor radon levels, remedial and preventive measures, radon in schools, kindergartens, town halls, water supplies, and show caves in Austria. Austrian radon standards and various information material for the public were issued.

However, the impact of these efforts in terms of lives saved or reduction of the mean radon risk was very low so far.

Therefore, in order to convert this knowhow into reduced health risk a number of measures are required like radon specific legislation including the clear assignment of responsibilites, implementation of preventive measures in building codes, further surveys and measurement campaigns to identify high radon buildings, frequent information campaigns, training of builders, engineers, and architects, and the improvement of the existing radon potential map for Austria. A national radon strategy which includes above tasks has to be set up.

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Norway’s new national radon strategy

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Abstract
A multidisciplinary working group, including the Norwegian Radiation Protection Authority (NRPA), was initiated in May 2007 by the Norwegian Ministry of Health and Care Services to suggest a coordinated national effort to reduce radon exposure. Their findings were that since radon levels in Norwegian dwellings are log-normally distributed, most of the radon cancers are induced at low levels. They estimated that 70 % of the annual lung cancers in Norway induced by radon occur at levels below 200 Bq/m^3. A successful radon strategy therefore needs to reduce not only high but also moderate radon levels. Radon exposure occurs in all categories of buildings i.e., individual risk comes from the sum of exposures from different buildings; work and leisure. The working group’s findings lead to the Norwegian government publishing a new national strategy for radon in July 2009. The strategy seeks to reduce the sum of all exposures and is directed at all categories of buildings. The national strategy aims to achieve radon levels (a) that are as low as practically achievable and (b) that are below given maximum limits. It is divided into the following sections – radon in: land planning, construction of new buildings; existing dwellings, communities with extreme radon problems, public buildings (including schools and kindergartens) and in work places. The NRPA has been appointed by the Norwegian government to coordinate the implementation of the new national radon strategy in the period 2010–2014. Furthermore, the NRPA has changed its recommendations concerning radon, based on scientific evaluation of the current literature and in accordance with the new national strategy. These changes include ALARA and new action and maximum recommended radon levels of 100 Bq/m^3 and 200 Bq/m^3, respectively. This presentation will present the radon strategy, its justification and the planned implementation.

Introduction
Radon (Rn-222) is a radioactive noble gas produced continually in bedrock material and soils as part of the uranium-238 decay series. Both soil air and groundwater can therefore contain radon. It has a relatively short half-life (3.82 days) and is therefore potentially mobile before it undergoes radioactive decay. Generally, when discussing radon and health effects, one implicitly refers to α-emissions from radon gas and selected daughter nuclides, notably Po-218 and Po-214 (half-lives of approximately 3 minutes and 160 microseconds, respectively).

Increased mortality of metal miners working in the Erz Mountains, central Europe, due to unspecified respiratory illnesses was observed as early as in the 1500s. Recognition that lung cancer was to blame came in the late 1800s, with radon being proposed as the likely cause for the lung cancers in the 1920s. Today, radon is one of the most studied human carcinogens and was recognised as such by the International Agency for Research on Cancer in 1988[1], based on the 11 largest studies on miners (60 000 individuals) which have also been evaluated by the Biological Effects of Ionizing Radiation panel (BEIR VI, 1999).

Radon typically enters buildings from the underlying ground, especially during colder periods as heating a building creates a pressure gradient sucking soil air into the building via cracks and apertures in its foundation. Inhalation of radon and its decay products leads to the deposition of α-particles in the respiratory system. The α-particles have a high linear energy transfer and are quickly stopped by surrounding body tissue. This leads to lots of energy being deposited within very small areas and the inherent high relative biological effectiveness of α-emissions. The final endpoint from received doses is lung cancer; indeed radon is recognised as the second largest cause for lung cancer after active smoking. Strong evidence exists that a single α particle can cause complex clustering damage to a cells DNA and induce major genomic changes e.g. mutation (BEIR VI, 1999). The risk attributed to radon exposure is therefore assumed as a linear relationship with no threshold and no dose-rate dependence.

Several studies have more recently investigated indoor radon and lung cancer, notably three large pooled analyses (Lubin et al., 2004; Darby et al., 2005; Krewski et al., 2006). Importantly, the pooled analyses also made corrections for the smoking habits of study individuals. The pooled analyses show a linear relationship between radon exposure and the risk of developing lung cancer, also at low/moderate exposure levels i.e. applicable to the general population. The risk of radon-induced lung cancer increased with exposure and is proportional to the indoor radon concentration and the exposure time. Moreover, the pooled analyses confirmed there is no lowest threshold radon concentration where no risk occurs. The risk is greatest for smokers and former smokers, though never-smokers can also develop lung cancer as a result of radon exposure. Specifically, the pooled analyses have shown:

- the relative increase in lung cancer risk from a given radon exposure is the same for smokers and non-smokers
- the absolute radon risk is higher for smokers compared to non-smokers
- the majority of lung cancers induced by radon exposure are caused by a synergy effect between smoking and exposure to radon and these lung cancers could have been avoided if the individual either had not chosen to smoke or had not been exposed to radon.

In total, the studies of miners coupled with pooled analyses present strong scientific evidence that exposure to radon leads to an increased risk of lung cancer. Other health effects that have been associated to radon exposure include leukaemia in children, cardiovascular disease, stomach cancer, multiple sclerosis (MS) and skin cancer. However, such studies often rely on differing methodologies producing conflicting

[1] http://monographs.iarc.fr/ENG/Monographs/vol43/volume43.pdf

results and conclusions such that to date there is no scientific concensus for either proving or disproving such health effects are associated to radon exposure.

The Norwegian Experience

Increased focus on radon nationally and internationally led to The Norwegian Ministry of Health and Care Services initiating and leading a multi-sector working group in autumn 2007 to report on the challenges faced and available mitigation measures regarding radon (*arbeidsgruppen for samordnet innsats mot radon*). The working group consisted of delegates from the Ministry of the Environment, the Norwegian Labour Inspection Authority, the National Office of Building Technology and Administration, the State Housing Bank, the National Institute for Public Health, the Norwegian Directorate of Health and The City of Oslo; the Norwegian Radiation Protection Authority (NRPA) acted as secretariat.

Radon concentrations measured in Norwegian homes are approximately log-normally distributed; most Norwegians are exposed to low to moderate radon levels in the home (Fig. 1).

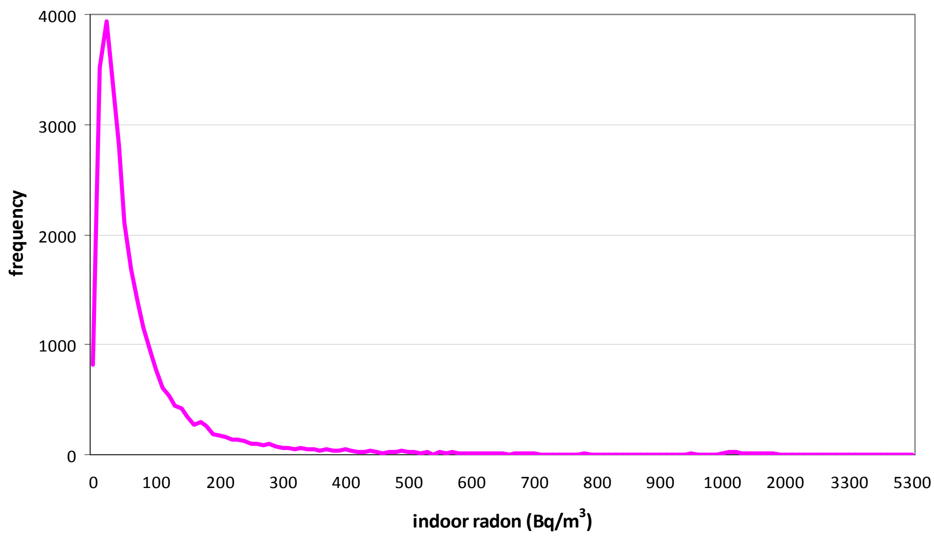

Fig. 1. Results from a national radon mapping campaign in 2000-2001. One radon measurement per dwelling, calculated as the annual average. Notice change of scale intervals on x-axis at indoor radon concentrations >1000 Bq/m^3; Total number of dwellings measured approximately 29 000. [StrålevernRapport 2001:6] available on www.nrpa.no (in Norwegian).

However, preliminary calculations based on the approach used by Darby et al., (2005) adjusted for the difference between the assumed average indoor radon concentrations in the United Kingdom and Norway (59 Bq/m^3 versus 88 Bq/m^3) have shown that approximately 300 people die each year in Norway from radon-induced lung cancer. Due to the distribution of indoor radon concentrations it has been estimated that approximately 70 % of these radon-induced lung cancer deaths occur at indoor radon concentrations below 200 Bq/m^3. To reduce the number of radon deaths substantially therefore requires that radon concentrations are reduced in all buildings, not just in the highest risk areas as all reduction of radon concentration and exposure time will have a favourable impact on health. Taking the population as a whole, health benefits from reducing indoor radon exposure can be considerable, even if only a modest reduction is achieved, so long as the reduction applies to a large number of buildings.

The multi-sector working group ended in the spring of 2009, when it formally submitted its report to the Ministry of Health and Social Care Services. The report and recommendations form the basis the Norway's new national strategy for the reduction of radon exposure in Norway, published 1 July 2009.

The national radon strategy

The national strategy is a tool for management and coordination of radon-prevention work in many sectors and shall be implemented in the five-year period 2009-2014. The strategic goals stated in the strategy are that the Norwegian government will:

- Work towards reducing radon levels in all types of building and premises to below the stated limits
- Contribute to reducing radon exposure in Norway as low as reasonably achievable.

As low as reasonably achievable implies that radon exposure should be reduced as much as reasonably possible, not only to a level just below the stated maximum limit. The choice of strategic goals is based on the knowledge that the risk from radon is proportional to exposure with no lower safe threshold value, such that all reduction of radon exposure will yield a health benefit. Radon can be present in all types of building and premises, and efficient prevention of radon risk therefore implies that radon levels must be reduced generally across society. The strategic goal of achieving as low radon levels as reasonably achievable is supplemented with legally binding limits where appropriate. This will ensure that the authorities have a basis for effective enforcement and compliance.

New indoor radon limits for certain types of building have been adopted (100 Bq/m^3, action required; 200 Bq/m^3, maximum) which are in line with the current recommendations from the World Health Organisation (WHO 2009), the Nordic Radiation Protection Authorities[2] and are supported by the NRPA.

Sub-strategies are proposed for work on:

- Radon in land planning
- Radon and the construction of new buildings
- Radon in existing dwellings
- Norwegian communities exposed to especially serious radon problems
- Radon in buildings and premises to which the general public are admitted
- Radon in the workplace

Stated goals and suggested measures for each sub-strategy are briefly described as follows:

[2] http://www.nrpa.no/dav/a4c7e5d25d.pdf

Radon in land planning

Here, the stated goal is:

- Radon must be emphasised in a systematic and sufficient way in all land planning.

A principal challenge for the work of local authorities in land planning in relation to radon will be to classify the land in relation to the radon hazard. Measures suggested in the national strategy include studying the relationship between radon hazard and geological conditions, as well as which requirements must be set for radon protection of buildings in relation to different degrees of radon hazard. A map-based tool for use in assessing radon hazard when planning land use at local and regional levels is suggested as well as establishing routines and systems that ensure that data concerning the building ground and geology, radon in household water supplies from drilled wells, building construction and radon in the indoor air are collected, both from the public and private sectors. These data should be made available for relevant local, regional and central authorities for administrative/management purposes. Identifying the occurrence of local areas with permeable surficial sediments/deposits and radium-rich bedrock is stipulated, as such areas are of importance with regard to radon levels in Norway.

Radon and the construction of new buildings

Here, the stated goal is:

- New buildings that are being constructed in Norway must have indoor radon concentrations that are as low as reasonably achievable and always less than 200 Bq/m^3.

This target can be achieved by setting specific requirements for anti-radon measures in all new buildings in the technical regulations pursuant to the Planning and Building Act. Specific requirements would include the use of radon barriers (e.g. a membrane), or the required ventilation of the ground below the building. It is difficult to estimate values for indoor radon concentrations in structures before they have been completed and taken into use. However, by setting requirements for preparing the buildings foundations such that it is possible to use active ventilation of the ground, it will be easier to carry out subsequent initiatives with ventilation/pressure modification to reduce the radon concentration to an acceptable level if the indoor radon levels are deemed too high in the finished building. Radon surveys should be carried out in all new buildings when they are taken into use. Specific measures include the possibility of establishing standardised methods/minimum requirements for radon-preventive measures to be used when constructing new buildings and premises. Supplementary guidance material will be necessary, generated by the National Office of Building Technology and Administration in collaboration with the NRPA. Such user-oriented guidelines that describe the regulatory framework and clarify the areas of responsibility for radon when constructing new buildings will be important for both builders and the local authorities supervising development.

Radon in existing dwellings

Here, the stated goal is:

- The proportion of dwellings with indoor radon concentrations exceeding 200 Bq/m^3 must be considerably reduced by 2020. The average indoor radon concentration must be reduced considerably by 2020, and a large proportion of dwellings must have achieved as low radon levels as reasonably achievable.

Different measures other than legal initiatives are the desired option to motivate and encourage homeowners to carry out radon surveys and technical preventive measures in their own homes. One condition for achieving the targets is that the radon-reducing building engineering methods are cost-efficient, safe and reliable. The development of better expertise and increased capacity for carrying out radon-reduction methods within the building industry will also be important. Specific suggested measures include undertaking systematic studies of radon in dwellings situated in all municipalities, the development of standardised radon-reducing measures for existing dwellings and generating campaigns to inform the population about radon: health risks, surveys and the implementation of remedial measures in dwellings. An additional proposal currently under review is the inclusion of radon measurements/status in home buyer reports.

Norwegian communities exposed to especially serious radon problems

Here, the stated goals are:

- All Norwegian communities in ”radon extreme areas” are mapped.
- Acceptable health conditions for the inhabitants of such communities are ensured through the introduction of necessary measures.

Extreme indoor radon concentrations can be measured in some buildings situated in specific areas in Norway. A large proportion of the buildings may have high radon values in such areas, including extreme values up to 50 000 Bq/m^3 in a few cases. Several such areas have been documented, but because the majority of Norwegian buildings have not yet been surveyed for radon, the possibility of finding other radon extreme areas during future surveys cannot be excluded. Such areas require special measures and follow-up, such as intensive information and survey campaigns when indications exist that a high radon hazard is present and assessing the need for individual assessment and medical follow-up, if necessary, of persons who have been exposed to very high radon concentrations over long periods.

Radon in buildings and premises to which the general public are admitted

Here, the stated goals are:

- The proportion of buildings with indoor radon concentrations above the maximum limit (200 Bq/m^3) is to be reduced considerably by 2020.
- The average indoor radon concentration is to be reduced considerably by 2020, and a large proportion of buildings have achieved as low radon levels as reasonably achievable.
- All schools and kindergartens have radon concentrations below the stated maximum limit.

The focus in this sub-strategy has been placed upon schools and kindergartens especially; the combination of obligatory presence and young individuals is deemed as a need for especially stringent requirements. The NRPA has completed an estimation of the costs involved in measuring all schools and kindergartens for radon and completing mitigation measures were appropriate (NRPA report, in press). The cost estimates were based upon the newly drafted legal limits for such buildings (100 Bq/m^3 action and 200 Bq/m^3 maximum), national statistics archive data and cost estimates received from recognised companies in the radon sector. Total costs were estimated as 50 MNOK for all kindergartens, primary and secondary schools [approximately 6.3 million €]. Other categories of public buildings will also be important however, such as hospitals, shopping malls, office buildings, hotels, restaurants, banks, commercial buildings, jails etc. Such buildings are often large and they contribute to the radon exposure of a large number of individuals. Those who are exposed to radon in such areas have no possibility of knowing that they have been exposed, or be in a position to reduce the exposure levels themselves.

Radon in the workplace

Here, the stated goal is:

- Norwegian workplaces shall have building and equipment conditions that ensure radon concentrations are suitable for a fully adequate working environment, based on consideration of employees' health, safety and the working environment.

Radon gas is the greatest source of exposure for ionising radiation amongst all Norwegian employees together. The working environment is already regulated with regard to other potential health risks such as asbestos and indoor climate, and initiatives outlined within this sub-strategy will also have a positive impact by reducing employees' exposure to radon. Together with the previous sub-strategy it is seen as important to study and devise standardised radon measurement techniques and anti-radon measures in different types of building as well as contributing to the development of knowledge about radon measurement techniques and anti-radon measures in large buildings generally.

Implementation of the national radon strategy

To implement the strategy's wide-ranging and ambitious goals the Ministry of Health and Social Care Services appointed the NRPA to establish a coordination group that would follow up the strategy. The coordination group includes participants from sectors that have policy instruments relevant to radon that will be able to suggest and produce radon mitigation measures. The group have met to discuss an overall plan for implementation as well as the establishment of specific working groups which will concentrate on different areas such as: land planning & radon mapping ; radon in existing dwellings; radon in the workplace and radon-related communication & information. This process is on-going.

Three specific pilot studies are also underway. The NRPA are funding a project working together with SINTEF Byggforsk[3] to establish suggested standard practice for a radon measurement protocol in large buildings, specifically schools. The results from this pilot study will enable a more widespread measurement campaign in schools during the 2010-2011 measurement period. The NRPA is also collaborating with the Geological Survey of Norway (NGU)[4] on a project designed to suggest a standardised method for measuring/classifying building raw materials (crushed stone and aggregates are first priority) with respect to their potential to cause elevated indoor radon concentrations, typically when used as fill materials around the foundation of new buildings. One of the goals from this project is to realise the establishment of a national system where radon information is available on-line from the producer, enabling consumers to base their decision on the potential radon risk. Lastly, the NRPA has initiated a pilot radon measurement study in dwellings that previously adopted radon mitigation methods during a 1999–2003 national subsidy campaign, to assess the long term effectiveness of different mitigation techniques used in Norwegian housing.

Regulatory amendments which will contribute to implementing the strategy

Changes have already entered into force 1 July 2009 in the Planning and Building Act (with regard to planning regulations) in which radon risk areas, or areas with potential risk, are to be given special consideration during land planning. Municipal authorities will now be required to include radon risk in risk and vulnerability analyses undertaken prior to new urban development. Furthermore, new radon requirements were adopted in the Planning and Building Act (with regard to building regulations) in March 2010. The new requirements include the introduction of legally binding indoor radon limits for new buildings (100 Bq/m^3, action required; 200 Bq/m^3, maximum). The new regulation also stipulates the use of anti-radon construction techniques, such as inter alia a membrane, and incorporating passive ventilation in the building foundations (which can be "activated" if necessary by installing a suction fan).

Another Norwegian regulation which is important regarding implementation of the strategy is currently in the process of being amended, namely the Radiation Protection Regulations. The new requirements include legally binding limits (100 Bq/m^3, action required; 200 Bq/m^3, maximum) for indoor radon in kindergartens, schools and rented accommodation, and are expected to be adopted in June this year, entering into force in 2014. The first of the pilot projects mentioned above is designed to help establish standard measurement techniques that will enable enforcement of these new radon limits.

Lastly, several other regulations are in the early stages of evaluating possible amendments with regard to radon, notably regulations concerning home buyers/sellers information requirements and radon in the workplace.

[3] http://www.sintef.no/Home/Building-and-Infrastructure/
[4] http://www.ngu.no/en-gb/

NRPA recommendations

The NRPA revised it's recommendations with regard to radon in 2009, the new recommendations are summarized as follows:

All buildings should have radon levels as low as reasonably achievable and within the recommended limits:

- Action Limit of 100 Bq/m^3
- As low levels as reasonably achievable – mitigation may also be relevant under the action limit
- Maximum Limit of 200 Bq/m^3

All buildings should be measured for radon regularly and always following modifications. Radon measurements should be performed as long-term measurements during the winter months using track-etch detectors. Radon mitigation measures in existing buildings should be source specific (aimed at identified radon sources) and seek to achieve as low radon levels as reasonably achievable.

Action Limit of 100 Bq/m^3
The action limit is defined as the threshold over which the NRPA recommends that radon mitigation measures are always initiated. If the annual mean indoor radon concentration is over the action limit, the NRPA recommend that effective radon mitigation measures are implemented as soon as possible to reduce radon levels. It is also recommended that radon measurements are repeated after mitigation, to ensure that the effect of measures is sufficient.

Maximum Limit of 200 Bq/m^3
The maximum limit is defined as the maximum indoor radon concentration that the NRPA considers should exist in all living rooms in all buildings. If the annual mean indoor radon concentration is over the maximum limit, it is recommended that (repeated) radon mitigation measures are implemented, followed up by control measurements, until indoor radon concentrations are as low as reasonably achievable and below the maximum limit.

The term action limit does not define a threshold where one can conclude that radon levels below it are "safe" levels, or where mitigation is not recommended nor has no purpose. If measurements reveal radon levels below the action limit, but where it is considered possible to achieve a substantial reduction of the levels through specific measures, such measures should be implemented to ensure that radon levels are as low as reasonably achievable.

Conclusion

Norway's new national radon strategy has set out wide-ranging and ambitious goals that will require a coordinated, systematic and sustained effort across several sectors in society. The NRPA is interested in sharing experience and keeping good contacts with other relevant authorities with regard to this work.

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National measurement database in radon research in Finland

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Abstract

STUK has carried out indoor radon measurements in Finnish dwellings since 1980s using alpha-track detectors. Our database contains measurements in 70 000 dwellings in detached houses, 17 000 dwellings in semi-detached and terraced houses and 5 000 flats. Residents are asked to fill in a two-page questionnaire form that contains questions on building characteristics, such as foundation type, ventilation type and radon prevention. Measurement activity is affected by location of residence, as people living in high-radon area are more likely to carry out radon measurements. Representative national and regional average concentrations were estimated by calculating the values in 1x1 km cells, and weighting each cell by the dwelling density. The calculated national average radon concentration in dwellings (excluding flats), 137 Bq/m^3, settles in between the representative values of 145 Bq/m^3 and 121 Bq/m^3 obtained in national random sampling surveys in 1990 and 2006, respectively. Municipal-specific radon map of Finland, as well as curves presenting average radon concentration by construction year, calculated using the dwelling-density weighting method, are presented. Until now, random sampling surveys have been necessary to obtain representative information on radon situation in Finland. The dwelling-density weighting approach opens up an alternative method. The database can be utilised, e.g., in radon prevention and mitigation studies.

Introduction

Radiation and Nuclear Safety Authority (STUK) has carried out indoor radon measurements in Finnish dwellings since 1980s using alpha-track detectors (Castren et al. 1992, Weltner et al. 2002). By the summer 2008, our national database contains measurements in 70 000 dwellings in detached houses, 17 000 dwellings in semi-detached and terraced houses and 5 000 flats. Nowadays most measurements are paid by the residents, but the material includes also measurements ordered by local authorities as well as those included in various STUK surveys and research projects.

Large indoor measurement databases can be used for radon mapping and identifying radon-prone areas (e.g. Miles, 1998). In Finland, a national Radon Atlas was published in 1997 (Voutilainen et al. 1997). An updated Radon Atlas will be published in 2010. On the other hand, national and regional radon statistics, such as average concentrations and percentage of houses exceeding the action level, are typically determined by representative sampling surveys. There have been two nationwide random sampling radon surveys in Finland with about 3 000 participants in each. The nationwide mean concentrations were 145 Bq/m^3 in houses and 82 Bq/m^3 in flats during the first survey 1990 - 1991 (Arvela et al. 1993). The respective values were 121 Bq/m^3 and 49 Bq/m^3 in the other survey in 2006 - 2007 (Mäkeläinen et al. 2009; 2010).

This paper investigates the possibility to calculate the national and regional radon statistics using the database material. At present STUK is carrying over 10 000 measurements per year. Utilising the data that is collected in radon measurements in any case would be a cost-effective means to update the Finnish radon statistics. The obvious problem is that the measurement material is highly unrepresentative. Residents in high-radon areas are more likely to perform the measurement, and also the regional campaigns are usually directed to risk areas. The bias is compensated by calculating the values for 1x1 km cells and weighting the values by the dwelling density in the cell. The results are compared to those obtained in the national random sampling surveys.

Material and methods

The radon measurements in the national database in Finland have been carried out using alpha-track detectors. The method has been described by Reisbacka (2010). The detector is posted to the resident together with a two-page questionnaire form that contains questions on building characteristics, such as construction year, foundation type, ventilation type and radon prevention.

The material of the present paper includes measurements in detached, semi-detached and terraced houses by the summer 2008. Apartments in blocks of flats are not considered.

The resident is instructed to locate the detector in the lowest residential floor, e.g. in the living room or bed room. Only measurements in actual living areas (not in cellars, etc.) of dwellings that were normally inhabited during the measurement were included in the material. Two or more detectors used simultaneously in a same dwelling is considered here as one measurement, with the average of the concentrations taken into account.

Measurements were included if the duration was at least 30 days and no more than 25% of the time had taken place outside the official measurement period, 1 November to 30 April. Annual average radon concentration was calculated using a seasonal correction factor of 0.85. The value was determined as the annual average vs. winter-time concentration ratio in the random sampling survey in 2006 (Mäkeläinen et al. 2009). However, also measurements longer than 272 days were included with the result interpreted as the annual average, without using the seasonal correction factor.

Measurements carried out in the same dwelling at different times were identified based on their address or, in case of insufficient address information, by the resident's statement in connection with the later measurement. A total of 87 457 different dwellings, i.e. 6% of all the dwellings in detached, semi-detached and terraced houses in Finland, were identified.

The location coordinates were determined by address using the StreetMap Finland road network material (ESRI Finland Oy), and data from the Population Register Centre and the National Land Survey of Finland. Some of the houses measured prior to 1997 were located from paper maps (Voutilainen et al. 1997). The positions of 75 270 dwellings are known accurately enough so that they could be located in a 1x1 km cell. The database contains 86 182 measurements in these dwellings (average 1.14 measurements per dwelling). There was on average 1.14 detectors used in each measurement.

The national and regional average concentrations as well as percentages of dwellings exceeding 200 Bq/m^3 were estimated by a dwelling-density weighting method. The values were calculated in 1x1 km cells and each cell was weighted by the number of dwellings (not including flats) in it. The dwelling density information was obtained from the Grid Database of Statistics Finland and is based on the situation on 31 December 2005. The grid data contains no detailed information on dwellings in cells that contain only one building. These cells include 1.3% of the Finnish population, and they were assumed to contain one dwelling, if there were inhabitants in the cell. A total of 1.4 million dwellings in 103 004 cells were included in the analysis. There was at least one measured dwelling in 21 258 cells. These measured cells included 73% of all the dwellings. The other cells were given the average concentration and percentage exceeding 200 Bq/m^3 from the nearest measured cell, or average of the values if there was more than one measured cell at the same distance. Each cell was included in the municipality where the centre of the cell is located.

The material of the present paper was compared to the 2 267 dwellings in detached, semi-detached and terraced houses completing the representative random sampling survey in 2006. The newest houses are underrepresented in the present material because the measurement is often made many years after the construction (Figure 1). The main difference in building characteristics is that the material contains clearly more hillside houses than the random sampling survey (Table 1). Other than that, there is no significant difference in the distributions of foundation types. This is assuming that the houses without a basement with an unknown foundation type do have the same percentage of slab-on-ground foundations as the other houses without a basement. There is no significant difference in ventilation types either (Table 2).

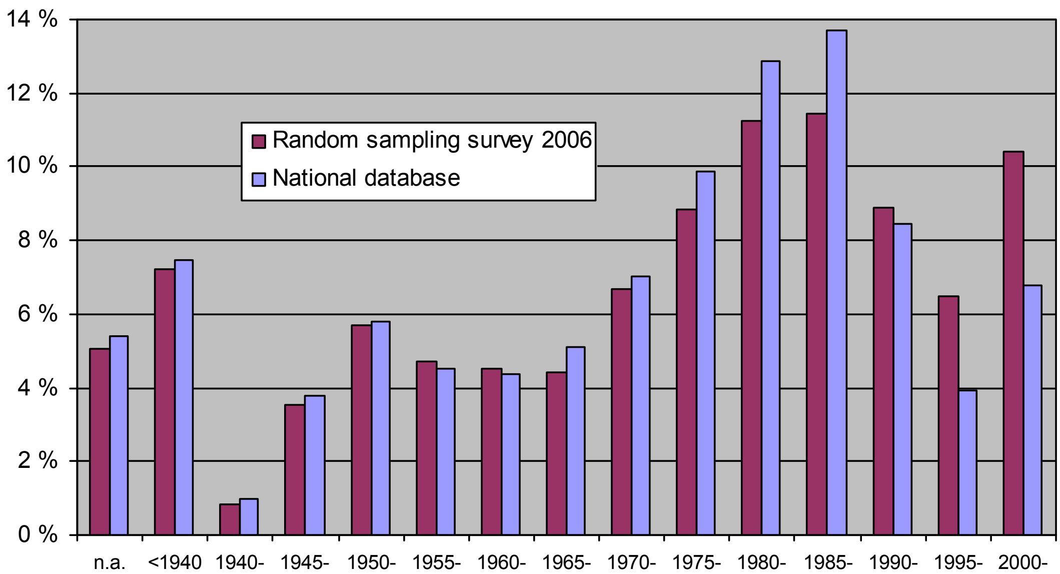

Figure 1. Dwellings in detached, semi-detached and terraced houses by construction year.

Table 1. Foundation types according to the questionnaire form filled in by the resident.

Foundation type	National database	Random sampling survey 2006
No basement, slab-on-ground foundation *	15% (39%)	33% (40%)
No basement, other type of foundation *	7% (18%)	18% (21%)
No basement, foundation type unknown	34%	10%
Cellar or partial cellar	22%	25%
Hillside house	21%	14%
Total (foudation type known)	52 179	2 060

*percentage in parentheses assuming the class *No basement, foundation type unknown* has the same *slab-on-ground* frequency than the other no basement-houses.

Table 2. Ventilation types according to the questionnaire form filled in by the resident.

Ventilation type	National database	Random sampling survey 2006
Natural	62%	58%
Mechanical exhaust	19%	19%
Mechanical supply and exhaust	19%	23%
Total (ventilation type known)	69 499	2 070

All in all, the comparison shows no indication that factors not related to the location of the house would cause significant distortion in the present material.

Results

The average radon concentration of the 75 270 dwellings included in the material is 227 Bq/m^3, if the first measurement in each dwelling is considered (Table 3). The estimated average concentration in all the 1,4 million dwellings in detached, semi-detached and terraced houses, determined by weighting the values calculated for 1 km^2 cells by the dwelling density, is 137 Bq/m^3.

There is 7 703 dwellings with multiple measurements. Typically, the latter measurement is carried out to verify the success of the radon mitigation, initiated by a high concentration in the first measurement. Therefore the average concentration is clearly lower (128 Bq/m^3), if the latest measurement in each dwelling is considered.

Table 3 shows also the results of the random sampling surveys. According to the survey of 2006, the average concentration in houses constructed after 1991 is not any lower as compared to that in older houses (Mäkeläinen et al., 2009). The difference between the surveys (145 Bq/m^3 in 1991 vs. 121 Bq/m^3 in 2006) is partly due to the higher outdoor temperature during the latter survey. Also the houses built before 1991 and having radon mitigation done after 1991 may have contributed to the difference.

The national average radon concentration and the percentage exceeding 200 Bq/m^3 obtained in the present work settle in between those from the two surveys.

Table 3. The average radon concentration and percentage of dwellings exceeding 200 Bq/m^3 in Finland (flats are not included).

Source	Number of measured dwellings	Average, Measured dwellings (Bq/m^3)	Average, All dwellings	>200 Bq/m^3, Measured dwellings (%)	>200 Bq/m^3, All dwellings
Database, highest value in each dwelling	75 270	233	139 [a]	31.5	16.9 [a]
Database, first meas. in each dwelling	75 270	227	137 [a]	31.0	16.6 [a]
Database, latest meas. in each dwelling	75 270	199	128 [a]	28.5	15.5 [a]
Database, lowest value in each dwelling	75 270	194	127 [a]	28.0	15.2 [a]
Random sampling survey 1990	2 096		145		17.9
Random sampling survey 2006	2 267		121		15.1

[a] Calculated by the dwelling-density weighting method.

There are 20 provinces in Finland having 9 000 – 230 000 dwellings each (flats not included). The weighting of the database material (first measurement in each dwelling) by dwelling density results in most cases in slightly higher radon values as compared to the random sampling survey in 2006 (Figure 2). Six provinces with the highest concentrations constitute a continuous high-radon area that is indicated in Figure 3.

Until now, the municipal-specific radon maps of Finland have presented the indoor radon situation concerning the measured dwellings (Figure 3a). On the other hand, the municipal radon averages calculated by the dwelling-density weighting method are shown in Figure 3b. The largest difference between the maps concerns the municipality of Kajaani. The average of the 713 measured dwellings is 405 Bq/m^3, and the estimate given by the dwelling-density weighting method is 120 Bq/m^3. The difference is due to a 3 km^2 local high-radon area that was extensively measured because of the high concentrations found there in the 1980s. This area includes only 5% of the 9 700 dwellings in detached, semi-detached and terraced houses in Kajaani, but as much as 45% of those measured.

The complete data of 75 270 dwellings was divided to smaller subsets to study how the number of measured dwellings affect the result obtained by the dwelling-density weighting method. A total of 376 subsets with 200 dwellings each were created. Any single dwelling was included in only one of the sets. Similarly, 250 sets of 300 dwellings, 150 sets of 500 dwellings etc. were made. The smaller the subset size, the higher the expectation value of the national average concentration calculated from the

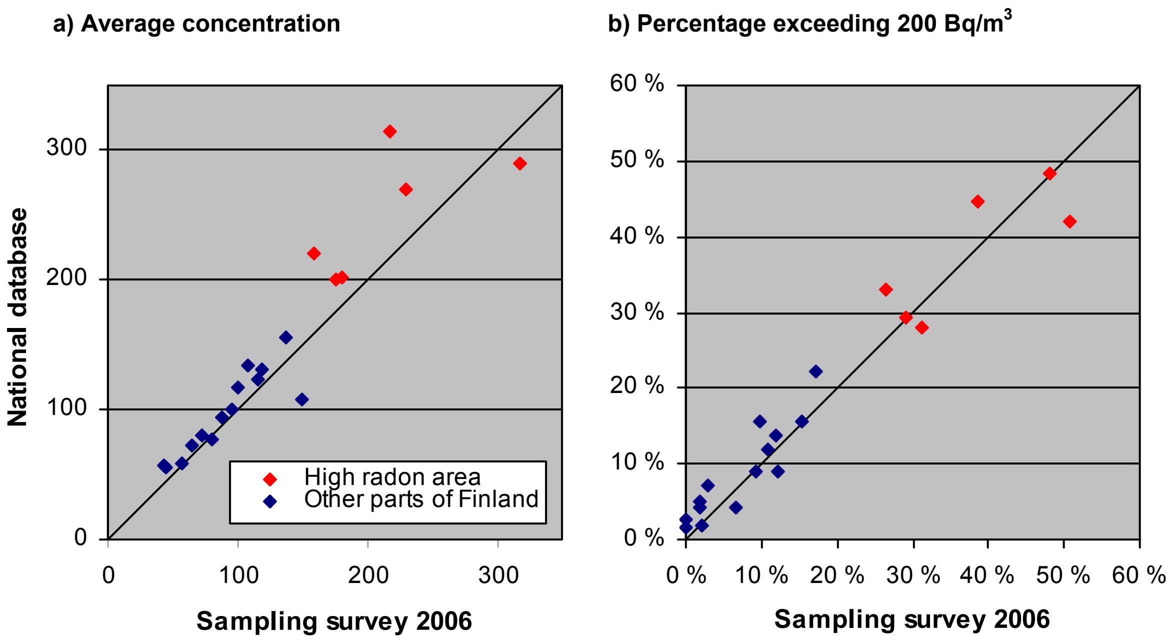

Figure 2. Radon in Finnish dwellings (not including flats). Values obtained from the national database by dwelling-density weighting method (first measurement in each dwelling) compared to the random sampling survey in 2006. Each point represents one of the 20 provinces of Finland.

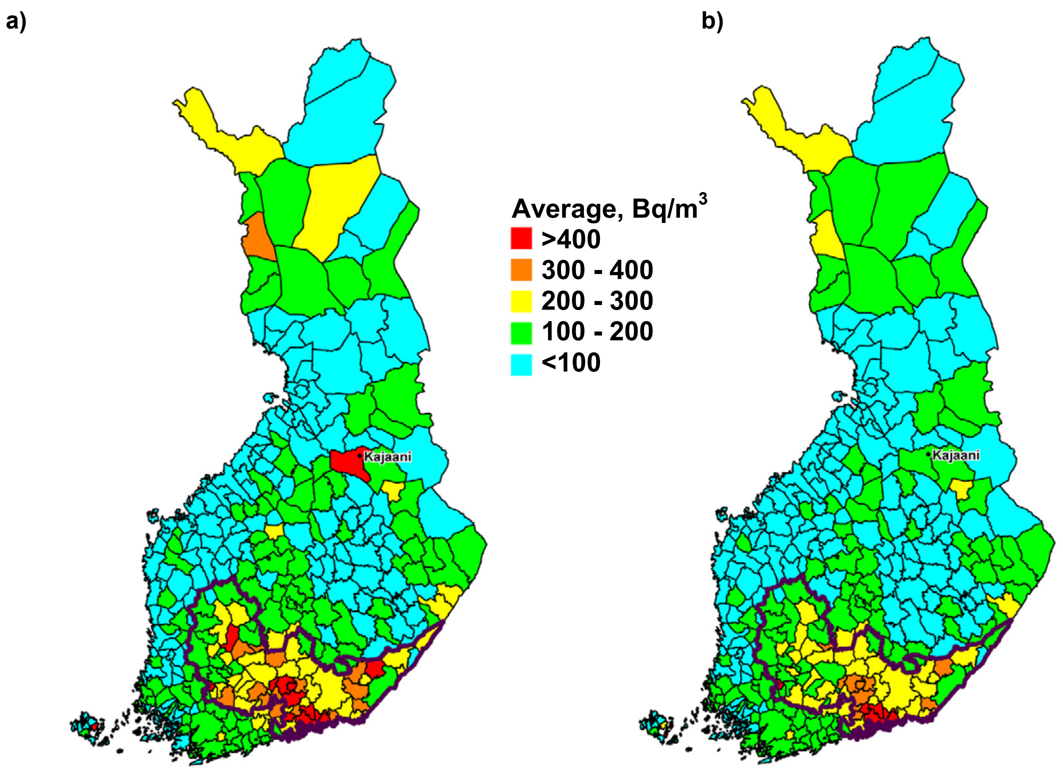

Figure 3. Radon by municipalities in dwellings (not including flats). a) Measured dwellings and b) All dwellings, estimated by the dwelling-density weighting method. First measurement in each dwelling is considered. High-radon area (six provinces with the highest average concentration) is separated by a violet line.

subset (Figure 4). The expectation value deviates less than 5% from the value 137 Bq/m^3 obtained using all the 75 270 dwellings, if at least 5 000 dwellings are included. In an extreme case of making the estimate from one dwelling only, the expectation value equals the average of the measured dwellings (227 Bq/m^3). The relative standard deviation among the concentration averages calculated using the different subsets is $3.6 \cdot n^{-0.6}$, as fitted in the data shown in Figure 4, where n is the number of dwellings used in the calculation. The fitted relative standard deviation is 0.4% for the complete data of 75 270 dwellings.

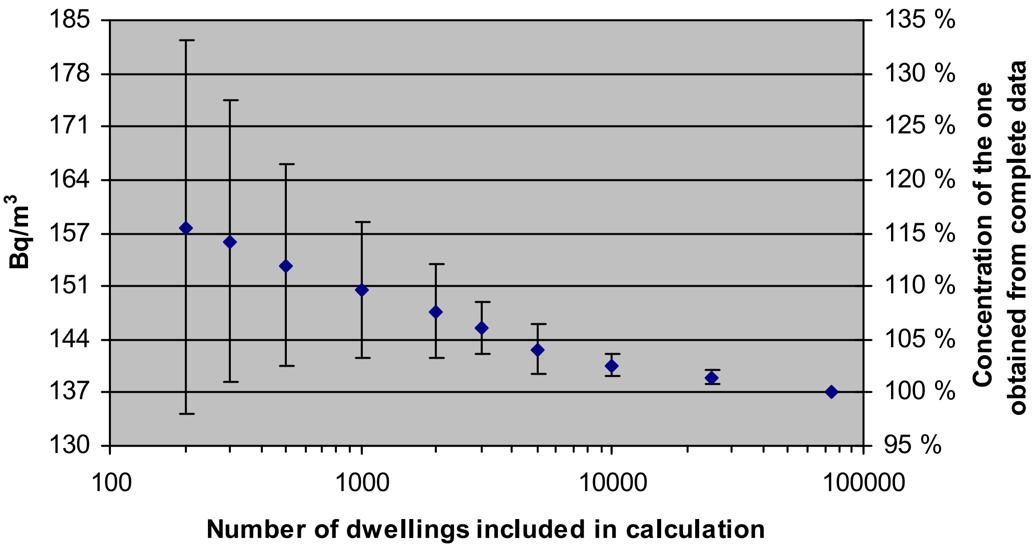

Figure 4. Average radon concentration in Finnish dwellings (not including flats) calculated by the dwelling-density weighting method. Uncertainty is given as ± one standard deviation. First measurement in each dwelling is considered.

The material contains 71 216 dwellings with a known year of construction, indicated by the resident. They were divided to sets of 1 000 (or 3 000) by the construction year. The dwellings with the same year were put in random order. The national radon average was calculated for each set using the weighting factors from the same grid data base (valid December 31, 2005) for all the sets. Thus, the analysis is simplified in a way that it does not consider the fact that houses are not built in the same 1 km^2 cells each year. The concentration trend of recently build houses is rather similar than the one by the random sampling survey in 2006 (Figure 5). The concentrations of older houses follow more closely the random sampling survey in 1990, being clearly higher as compared to the survey in 2006.

The oldest dwellings in the last set of 3 000 measurements were constructed 2002, that is 6 years before the latest measurements included in the material. The time gap between the construction year and the time when the estimate is available can be shortened by using smaller sets, with an expense of decreased accuracy. The oldest dwellings in the last group of 1 000 measurements were constructed 2005.

The national average concentrations calculated from these chronological datasets are 147 Bq/m^3 (1000 dwellings per set) and 143 Bq/m^3 (3 000 dwellings per set). These values are slightly lower respectively, than the corresponding expectation values 150 Bq/m^3 and 145 Bq/m^3 shown in Figure 4 for randomly chosen sets. Thus the chronological subsets overestimate the national average concentration (137 Bq/m^3) slightly less than the randomly chosen subsets.

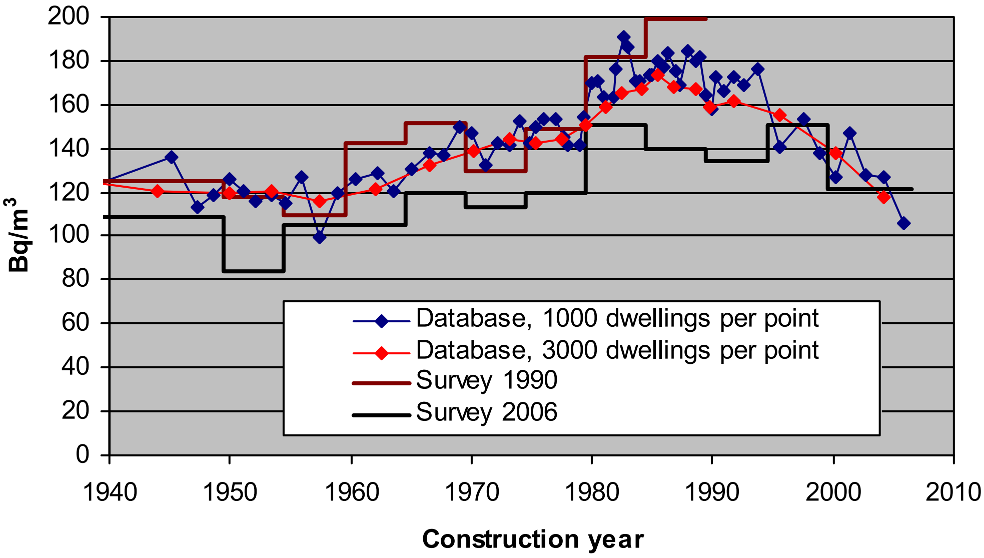

Figure 5. Average radon concentration in dwellings (not including flats) in Finland. The values calculated from the national database (first measurement in each dwelling) using the dwelling-density weighting method are compared to the results of the random sampling surveys in 1990 and 2006.

The average radon concentration of the Finnish houses was the highest in the 1980s and 1990s. The increasing trend before the 1980s was mainly due to the increased popularity of the slab-on-ground foundation type. On the other hand, three main reasons have been identified for the decrease in radon concentrations since the 1990s; i) increased popularity of the crawl-space foundation type, ii) mechanical supply and exhaust ventilation strategy is used in practically all the new houses and iii) radon-safe construction practices have become common (Mäkeläinen et al. 2009). Figure 6 shows, that the recent decreasing trend is the strongest in the high-radon area. This suggests that the radon-safe construction practices, being much more common in the high radon area, are a major contributor to the decreasing trend.

Discussion

The difference in radon concentration between the measured dwellings and the other dwellings seems to be caused mainly by factors related to the location of the house, such as soil gas radon content and soil type. These factors can be compensated for by the dwelling-density weighting method described in this paper. The number of measured dwellings in our database is large enough for producing municipality-specific results, although more work is needed to estimate the minimum amount of measurements needed for reliable results. The material can also be used, e.g., for monitoring radon situation in new houses by construction year. The sensitivity of the results to grid size needs to be studied. There is also a 0,25x0,25 km grid database that can be used, if the 1x1 km grid appears to be too coarse producing significant inaccuracy.

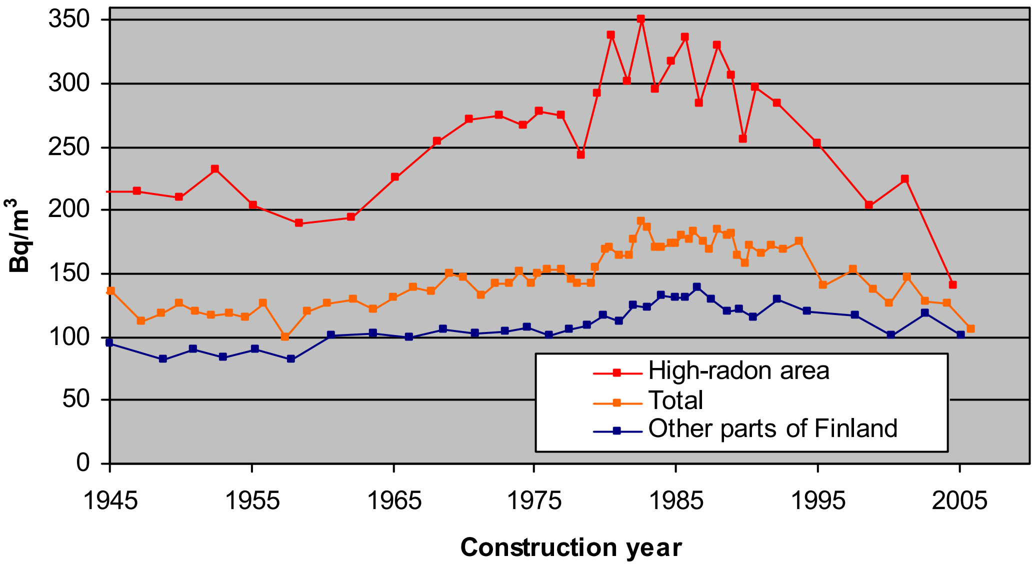

Figure 6. Average radon concentration in dwellings (not including flats) in Finland calculated from the national database (first measurement in each dwelling) using the dwelling-density weighting method. The high-radon area is indicated in Figure 3.

This paper considers only dwellings in detached, semi-detached and terraced houses, but it is important also to study blocks of flats. This is because the average radon concentration in the lowest floor flats with either soil or rock directly beneath them (no cellar etc.) is as high as in houses (Mäkeläinen et al. 2009). Since the number of measured flats is small (4 908), the radon statistics can not be produced with the same elaborateness as can be done in case of houses. The dwelling-density weighting approach requires some modifications when applied to the flats. Measurement is recommended especially for the lowest-floor apartments and consequently they are overrepresented in our material. Another factor causing overrepresentation of high concentrations is that a high value in one apartment often results in measuring also the other apartments in the same building.

Radon measurements are primarily intended to determine if mitigation is needed. Therefore residents are instructed to carry out the measurement in the lowest floor with living areas, because the concentration is usually higher there than in the upper floor. The abundance of lowest-floor measurements was not compensated for in this work. The effect of the radon detector location on the detected concentration needs to be analysed, if the concentration people actually are exposed to is to be accurately determined.

Conclusions

The regional and national average radon concentrations and percentages exceeding 200 Bq/m^3 were estimated by a dwelling-density weighting method using the measurement results in 75 270 dwellings with a known location. The results agree with those from national random sampling surveys with a reasonable accuracy. Until now, random sampling surveys have been necessary to obtain representative information on radon situation in Finland. The dwelling-density weighting approach opens up an alternative method. The new approach is cost-effective, as it utilises data that is collected in any case. The large number of measurements enables also presentation of municipality-specific radon situation.

Our future plans include utilisation of the national measurement database more widely in radon research. Measurement results, combined with the building characteristics data collected by a questionnaire form, can be utilised, e.g., in radon prevention and mitigation studies.

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Reducing lung cancer incidence by health campaigns: A review of the uptake, health benefits and cost effectiveness of Smoking Cessation and Radon Remediation Programmes in Northamptonshire, UK

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Abstract

The greatest risk factor for lung-cancer is smoking, the second largest factor being raised radon levels at home. Initiatives to stop smoking and reduce domestic radon levels have met with some success, but in both cases a significant proportion of those affected fail to respond. The two risk factors combine, so that those who smoke and live in a house with high radon levels are at higher risk than if exposed to only one of the two threats. There is the potential for combined public health campaigns to better target those affected.

Our group has studied both the smoking cessation and radon remediation programmes in Northamptonshire, UK, considering the costs and health benefits of each, and, using postal questionnaires, recording the demographics of participants. Our analysis suggests that the demographics of the two groups are significantly different. Those who remediate tend to be older, include fewer smokers and have fewer children.

The health benefits from stopping smoking are greater than those from radon remediation, and the smoking cessation programme costs less per lung-cancer averted. In addition, the continuing reduction in smoking prevalence in the UK will reduce the effectiveness of radon remediation campaigns.

This paper discusses the synergy between the programmes, and argues that there is merit in an integrated approach to these health campaigns, and in particular to extending smoking cessation programmes to include advice on reducing the risks from radon.

Introduction

Smoking is the most significant risk factor for lung-cancer. Since this became known, education and health campaigns have been conducted to reduce smoking prevalence. In

England, smoking prevalence has dropped from 45% in 1974 to 28% in 1998, declining further to 21% in 2008 (Dept. of Health, 2010).

Radon, a naturally-occurring radioactive gas, is the second most significant risk for lung-cancer after tobacco smoking. High levels of radon were first identified in uranium mines, but more recently, it has been established that significant levels are found in the built environment, and case-control studies have shown an associated increase in lung cancer in the public from radon in their homes (AGIR, 2009; BEIR VI, 1999; Darby et al., 2005).

Northamptonshire, a county in central England, is one of the areas in the UK where a significant proportion of homes have radon levels over the UK domestic Action Level of 200 Bq·m^{-3} (Bradley et al., 1997). Radon levels can be tested simply, and, if raised radon levels are found, remediation work, usually involving the introduction of a sump and attached pump to extract radon to outside and costing around £750 (around 825€), will reduce radon levels nearly always well below the Action Level. Over the last 18 years, campaigns to measure and reduce radon in homes have been implemented through the local councils' environmental health departments. Despite publicity, only around 40% of householders have tested radon levels in their home, and of those who discover raised levels, only 15% remediate their homes (Chow et al., 2007). Our group has studied the characteristics of those who remediate their homes, and has shown that they are older, have fewer children, and include fewer smokers than the general population (Denman et al., 2004).

The risks from radon and smoking are considered to be sub-multiplicative (BEIR VI, 1999), and so smokers, who are most at risk from radon, are not being targeted by current radon remediation campaigns. This led our group to consider the local smoking cessation initiatives, and whether these might be valuable in reducing radon-induced lung cancers. Our initial work showed that the smoking cessation programme in Northamptonshire has added value compared to cessation programmes in areas with lower radon levels. In addition, there is greater health benefit for a smoker living in a high-radon house from quitting smoking than from remediating the house and continuing smoking (Groves-Kirkby et al., 2008).

In England, smoking cessation programmes are commissioned by the local Primary Care Trust (PCT), part of the National Health Service (NHS), with the assistance of General Practitioners (GPs). Our group has recently studied the characteristics of those who join and participate in smoking cessation programmes.

This paper compares those who remediate and those who quit smoking, and considers the characteristics of each group, and their similarities and differences. It also compares the cost-effectiveness of each programme, and considers the impact of the steady reduction of smoking prevalence in England. These results can inform future public health initiatives to improve the response to both risks.

Material and methods

All houses in the radon remediation series were remediated by a single company following UK Radon Council good practice (The Radon Council Ltd, 1995). In early 2002, additional personal information to make the individual risk assessments was obtained by postal questionnaires sent to all the houses in the study. The questionnaire included questions about all of the residents in the house, including age, occupation, smoking habits and the time each spent inside the house on a recent day; detailed results from this study have been reported previously (Denman et al., 2004).

Participants of the Northamptonshire PCT Smoking Cessation Programme were contacted by telephone in 2007 to ask about their quit status. During the call, participants were invited to complete a written questionnaire, and if they agreed, this was posted to them with a stamped addressed envelope for return. The answers were entered into a bespoke Access database, using double entry and record comparison to ensure data accuracy.

The results from the smoking cessation programme were assessed by examining the bivariate relationships between the main outcome measure (whether the respondent indicated that they continued to have quit smoking), and the socio-demographic factor under consideration, using Chi-squared tests. The results from the remediators were compared to the smoking cessation scheme participants in a similar analysis.

A separate exercise analysed demographic data from 4,626 smokers registering with the Northamptonshire Smoking Cessation Service during 2004-2005 and setting personal quit-dates. Smoking status was validated at 4 weeks by carbon monoxide monitoring, and at 26 and 52 weeks by telephone questionnaire (unvalidated, self-reported). Using age, gender, smoking status and postcode of residence (surrogate for domestic radon concentration) derived from this cohort as input data for the European Community Radon Software (ECRS) tool (Degrange et al., 2000), estimates of the lung-cancers averted in the quitting cohort were generated, permitting preliminary analysis of the relative cost-effectiveness of radon remediation and smoking cessation.

Results

In the radon remediation series, 122 questionnaires were sent, and 73 householders replied (59.8%). The houses contained 162 occupants, an average of 2.22 per house (range 1 to 5). These included 138 adults and 24 children. 1 household did not give personal data on the occupants, leaving 72 houses with 160 occupants suitable for further analysis. Householders had been living in the house for an average of 16.2 years (range 0.8 to 45.9 years). 3 households (6 occupants) had moved in since remediation. 6 households contained smokers, 3 indicated that around 100 cigarettes a week were smoked in the house, 2 suggested 30 cigarettes, while one household smoked 4 cigars a week. In addition, one respondent indicated that only a young adult in their household smoked, but did so only outside the house. 10 of the 160 occupants (6.25%) indicated that they were current smokers. In 2 households, both adults smoked. The remaining 4 households who smoked contained 5 non-smokers who would be subject to passive smoking risk in additional to the radon risk.

In the smoking cessation series, some 482 4-week quitters were identified from the Northamptonshire PCT Stop Smoking Service. These smokers had successfully quit at 4 weeks, as assessed by the Department of Health criteria, during the period 1st July 2006 to 30th September 2006. The sample population had previously consented to be followed up at one year, as part of routine data collection to ascertain their current quit status, and were asked, during this telephone contact if they would provide some further information in a written questionnaire for research. Some 317 questionnaires were sent out. Completed questionnaires were received from 103 quitters (32%), 68 of whom confirmed that they had not smoked since, while 35 had relapsed. 77 lived with a partner, or parent, while 63 had children under 18 living at home, while a further 17 had children over 18 at home.

There are statistically significant differences in family size between remediators, quitters, and the national population. Remediators' family size is smaller, and more likely to be 2, less than that of the national population ($p < 0.001$), while the distribution of family sizes for quitters differs to that of both remediators ($p = 0.006$) and the national population ($p = 0.011$), with the quitters generally having larger family sizes.

The length of time that people had spent in their current house was also studied, and compared to National Statistics. The results are shown in Table 1. Separate data was obtained for <1 year and >40 years but the frequencies were too low for valid statistical analysis and they were therefore combined to form those shown in the Table. Although it appears that remediators have been in their current home longer than the national average, this is not statistically significant ($p = 0.110$). However, it is statistically significant that quitters are more likely to have been in their current house for a shorter time than both remediators and the national population ($p = 0.036$, and $p = 0.033$ respectively).

Table 1. Length of Time in Current Home.

Time in Current House	England, Home Owners 2001/2	Radon	Quitters
t < 3 years	2,755,000	5	18
3< t <5 years	1,424,000	6	15
5 < t <10 years	2,416,000	13	27
10 < t < 20 years	3,525,000	20	19
t < 20 years	4,174,000	26	24
Total	14,294,000	70	103

From the questionnaires, analysis of the ages of responders indicates that the remediators tend to be older than quitters, and for the radon remediators there is a clear peak in the age range 60 to 80 years, which is also seen in Figure 1. However, the difference between remediators and quitters is not statistically significant at the 5% level ($p = 0.185$).

The age distribution of all occupants of remediated houses and of houses with quitters is shown in Figure 1, together with a comparison with the age distribution of the Northamptonshire population. The differences in this case are highly statistically significant (all have $p < 0.001$). This is presumably related to the statistically different family sizes in each group, noted above.

The questionnaire sent to those on the smoking cessation programme also permitted comparisons between those who had remained quitters for a full year, and those who had relapsed since the previous assessment at 4 weeks. Table 2(a) shows that the continuing quitters were more likely to have children under 18 at home. This is statistically significant, with $p = 0.002$. Those who relapsed were more likely to have children over 18 at home, as in Table 2(b). This is also statistically significant ($p = 0.003$). Continuing quitters are also more likely to be living with a partner or parent, see Table 3, with this finding being significant at the 5% level ($p = 0.046$).

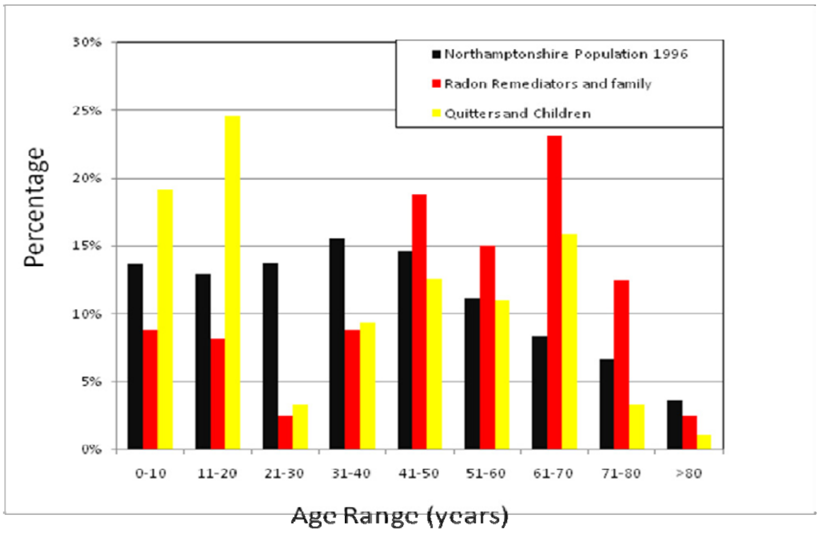

Fig. 1. Age Distribution of local population, occupants of remediated homes, and of houses with quitters.

Table 2. Association between quit status and having children at home (a) under 18 and (b) over 18.

Table 2(a)

Status	Children under 18 at home	
	Yes	No
Relapsed	14	21
Still not smoking	49	19
Total	63	40

Table 2(b)

Status	Children over 18 at home	
	Yes	No
Relapsed	11	24
Still not smoking	6	62
Total	17	86

Table 3. Association between quit status and living with partner or parent.

Status	Parent or Partner in home	
	Yes	No
Relapsed	22	13
Still not smoking	55	13
Total	77	26

Of the initial cohort of 4,626 quitters participating in the cost-effectiveness study, only 435 remained tobacco-free after 52 weeks. Using the ECRS software, loss of life-expectancy and excess lung-cancer incidence were modelled for each individual in the study, assuming that they: (i) continued smoking, (ii) quit smoking at the time of the study and (iii) had never smoked, in each case for zero radon exposure and the mean annual level in their home imputed from their postcode of residence. Preliminary results are summarised in Table 4. The average life-year saving among ex-smokers in the Northamptonshire radon environment is 1.21, being the difference between the average smoking-attributable life-year losses of 1.93 and 0.72 for smokers and ex-smokers respectively, compared with 1.10 in a radon-free environment.

Table 4. Cost-Effectiveness Comparison.

Smoking Cessation Study		Radon Remediation Study	
Population	462		127
Attributable Lung Cancers	55.22		6.84
Total cost (service+medication)	£1,156,515	Total remediation cost	£746,725
Total cost per quitter	£2,503		
Cost per life-year saved	£2,069		£8,047

Discussion

UK Smoking Cessation Programmes place emphasis on quitting at an early age, as the British Doctors Study (Doll et al., 2004) showed that those who quit successfully by 35 years avoid much of the excess mortality risk due to smoke, those who quit successfully by 50 avoid about half the risk, and those at 60 avoid one third the mortality risk. The age of the occupants when radon remediation is carried out is similarly important, as the degree of risk from radon depends on the total time spent exposed in the home together with the average radon level in the house prior to remediation.

In our study, an analysis of the ages of responders indicates that, although remediators appear to be older than both quitters and the Northamptonshire population, this difference is not statistically significant at the 5% level. However, the age distribution of quitters together with their partners and children has a statistically lower age profile than the population, which is in itself lower than remediators and their families. This contrasts with the relative risk to other occupants in the home, since the reduction in health risk when radon is reduced is similar for each member of the family, while for the smoker, the risk to other family members is the lesser one of passive smoking, assuming they do not smoke, or none, if the smoker smokes outside.

In addition, our study shows that remediators have smaller families than quitters, and have lived in their current homes longer. Therefore participants on the smoking cessation programmes represent a different target population from those who remediate, so there is a potential to extend the reach of a radon remediation initiative.

The questionnaire to quitters also asked about the quitters' knowledge of radon, and whether radon, amongst many other factors, played any role in the decision to quit. These results, reported elsewhere (Groves-Kirkby et al., 2008), showed that few respondents regarded radon as a significant decision factor, and that the direct health concerns as a result of smoking were much more significant as factors affecting the decision to quit.

Smoking cessation programmes have been in operation for longer than radon remediation programmes, and there has been extensive analysis of their efficacy, reflected in many more publications. As a result, smoking cessation programmes have evolved to improve their operation, and to target remaining smokers.

The age profile of UK smokers has changed significantly since 1970 (Dept. of Health, 2010), dropping significantly so that people aged 20 to 24 smoke more than any other age group, which is consistent with the age profile of quitters in our study. There is a considerable literature on other characteristics of smokers, and the relative success of different smoking cessation techniques, and the likelihood of relapse.

Solberg et al. (2007) studied the educational background of young adult smokers (age 18 to 24 years) in the US, and concluded that the level of interest in quitting, number of quit attempts, and relapse rates did not depend on educational level, although higher educational level was associated with a lower proportion of smokers. Macy et al. (2007), studying a similar group, showed that 33% of long-term quitters who had quit for over 1 year, had relapsed by 5 years. They were less likely to relapse if they were married to a non-smoker, had only one parent who smoked, or worked in a smoke-free building. Tucker et al. (2005), also studying young smokers and quitters (23 to 29 years), found that smokers were more likely to have higher rates of substance abuse, illegal activity, poor mental health and victimisation, but these factors were less relevant to quitting than social transitions and interpersonal factors, race, ethnicity, and health status. Other studies have shown that quitters were more likely to relapse if they had higher emotional distress, higher nicotine dependence, higher alcohol consumption, and more medical problems (Augustson et al., 2008), and increased post-quit anger (Patterson et al., 2008). However, Freund et al. (1992) report that recent hospitalisation and a diagnosis of heart disease increases the likelihood of cessation.

Mills et al. (2008) questioned smokers attending US Emergency Departments and found that 37.8% (441/1168) had children. They found that smokers with children were more interested in quitting, and more confident in doing so. Our study supports these findings, with 61.2% (63/103) of those entering the smoking cessation programme having children under 18 at home, rising to 72.1% (49/68) for continuing quitters at one year. Interestingly, our study shows that, if the children are over 18 then the converse is true – that the parent is more likely to have relapsed.

Smoking cessation programmes have developed over the years, but it is proving increasingly difficult to reduce smoking rates further, and Twigg et al. (2009) suggest that a heterogeneous range of different programmes will be required to reach ‘hard-to-engage’ populations. Bauld et al. (2008) compared two types of cessation programme in Glasgow – a group scheme with GP support, and a pharmacy-based scheme – and showed that the demographics of those recruited to each scheme was somewhat different, with a higher proportion of those aged 16 to 40 attending the pharmacy-based scheme. There was also a significantly higher proportion of people with children, whilst the group scheme had an increased percentage of females. The quit rate at 4 weeks was higher for the group scheme. They showed that those who were more socially deprived were less likely to quit, which concords with other studies, which also note that those in socially deprived areas are more likely to smoke (Twigg et al., 2009). Of the factors considered in our study, Bauld et al. (2008) found a significantly higher cessation rate in those with partners/spouses ($p = 0.046$), but only in the group scheme; whereas those with children had a significantly lower cessation rate ($p = 0.003$) but only in the pharmacy scheme. It should be noted that this study was conducted for 4-week quitters, whilst our study was of long-term (1-year) quitters, and many authors note that there is a large relapse rate over the first year. For example Zhou et al. (2008), in a major international study, note a relapse rate close to 80% in the first quarter, with 60% of the remainder relapsing in the next quarter. Zhou et al. also note that relapse rates vary between countries, with subjects in USA and Canada more likely to relapse than those in France, Spain and UK. Others more likely to relapse include those who have failed to quit before, those with cessation-related sleep disturbance, and with heightened anxiety.

Despite these concerns, smoking prevalence in England has steadily reduced over many years, and the Department of Health have published an initiative to reduce smoking prevalence to 10% in 2020 (Dept. of Health, 2010). This target is ambitious, but Canada and Victoria, Australia, both demonstrate falling prevalence rates and already have smoking prevalence below 20%.

Our study indicates that it would be possible to extend the response to the health risks of radon if quitters were provided with safety and remediation information during their interaction with smoking cessation programmes in areas affected by radon.

It should be noted, however, that quitting smoking also saves money, whilst the householder needs to spend money to remediate if raised levels are found. It is interesting to note that, in their questionnaire responses, quitters placed the cost issue as the 7th most significant factor in their decision to quit smoking – below the direct health risks, but above family pressure, and the knowledge of radon (Timson et al., in preparation).

Among smokers in Northamptonshire, individual and population mortality increases with smoking status and radon exposure, with smoking accounting for 5 to 8 times more life-years lost than radon. Quitting smoking is expected to reduce life-years lost to avoidable lung-cancer by 121 years per 100 quitters at a cost of £2,069 per life-year saved. If the same group were to remediate their homes while remaining smokers, the intervention would reduce life-years lost to lung-cancer by 21 years for every 100 remediators at a cost of £8,047 per life-year saved. For ex-smokers and non-smokers with comparable radon exposure, remediation would reduce life lost to lung-cancer by 10 years per 100 remediators and 3 years per 100 remediators respectively, at costs of £17,000 and £62,000 per life-year respectively.

The effect of smoking prevalence is important in radon remediation programmes. For example, AGIR (2009) estimate that, for an annual build of 200,000 new homes in the UK, installation of radon-proof membranes will currently avert 4.4 lung cancers per year. Applying the current relative risks for radon in smokers and non-smokers, this would drop to 2.5 lung cancers per year by 2020 if the UK achieves its 10% smoking rate target.

Conclusions

This study has confirmed that the higher motivation to quit in smokers with children at home, noted by Mills et al. (2008), converts into a higher percentage of smokers with children entering our smoking cessation programme initially and then a statistically significant higher proportion of such smokers continuing to quit at one year. Conversely, the presence of young adults over 18 in the home increases the likelihood of relapse before one year. The presence of a partner or parent in the home decreases the likelihood of relapse.

The study also indicates that those joining a smoking cessation programme differ considerably from remediators, and in particular have significantly larger families, while remediators have lived in the same house for a longer time. Thus they represent a different target group, who do not currently consider radon as a significant risk.

Finally, although the quantitative analysis of cost-effectiveness remains incomplete, it is apparent that in a radon affected area, the cost per life-year saved from a smoking cessation campaign may well be significantly less than that arising from a radon remediation campaign.

There is thus the potential to extend the response to the health risks of radon if provided with safety and remediation information during their interaction with the smoking cessation team in radon Affected Areas.

The approach of radon remediation programmes in England and Wales to date has been through local Environmental Health departments and targeted at remediating the home. However, this analysis suggests that there is scope for the National Health Service to take up this concern, to target smokers and to benefit from the greater experience of running smoking cessation programmes. In addition, the falling smoking rates suggests that the balance between smoking cessation programmes and radon remediation programmes in lower radon areas should be coordinated in order to achieve the best value for money, and highest benefits.

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A model describing indoor concentrations of thoron and its decay products

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Abstract

In the past, the radioactive noble gas thoron (^{220}Rn) was thought to occur only in negligible concentrations. But recently, increased thoron concentrations were found in traditional Chinese and Indian mud buildings, where their contribution to the inhalation dose of the dwellers can be in the order of magnitude of the dose from radon (^{222}Rn). Moreover, the WHO's Radon Handbook of 2009 advises dose reductions also with relatively moderate exposures, for which even average additional thoron concentrations can be crucial. Therefore, a model of the occurrence of thoron and its decay products indoors was developed.

The solution of the differential equations which describe the sources and sinks of the nuclides of the thoron decay chain yielded a theoretical relation between the concentrations of the nuclides in the unattached and the attached state, which are important for inhalation dosimetry. Several differences to existing radon models could be identified. Transfer coefficients occurring in the differential equations were determined experimentally. For this purpose various atmospheric conditions were adjusted in a small-scaled traditional mud dwelling, which had been erected in a laboratory at Helmholtz Zentrum München. Because of the prevalence of increased thoron concentrations in homes with mud as building material, special emphasis was given to study thoron exhalation from mud structures.

To validate the model, model predicted concentrations of thoron and its decay products of real Chinese and Indian mud dwellings were compared to measured concentrations in those rooms. The model makes possible to assess exposures of dwellers to thoron and – combined with a dose model – to calculate their inhalation doses in dependence on their living habits from easily measurable constructional properties of the dwelling. The model predicts significant exposures for residents of buildings with mud construction elements in other countries as well.

Introduction

Need for a thoron model

The radioactive noble gas radon and its decay products have been known as indoor health risks for several decades as they can cause a significant dose to the respiratory tract and other tissue. Only recently, the connection between the exposure to radon and an increased incidence of lung cancer with a linear dose-response relation even below 200 Bq m^{-3} was shown in epidemiological studies (Darby et al. 2005, Krewski et al. 2005). These findings, among others, induced the World Health Organization to advise measures for the mitigation of exposures also with relatively moderate concentrations of radon and its decay products (WHO 2009).

The radon isotope ^{220}Rn (thoron) occurs in most houses mainly from the building material as its half-life is only 55.6 s. For dwellings made of brick or concrete, it contributes to the inhalation dose of the dwellers in the order of 10–20% of the dose caused by all radon isotopes (UNSCEAR 2000). Therefore, previous studies focused on the exposure to the prevalent isotope ^{222}Rn (called radon in the following) whereas thoron was examined only rarely or as a side-aspect of radon measurements (e.g. Stranden 1980, Steinhäusler 1996). In the light of the proposed reduced reference levels (WHO 2009), the dose contribution by thoron gains importance for the total indoor exposure in such dwellings.

Moreover, in the last years occurrences of increased thoron concentrations have been discovered in dwellings made of unfired mineral building materials as adobe and mudbricks (Sreenath Reddy et al. 2004). The contribution of thoron and its decay products to the inhalation dose of people living in the traditional caves which are dug into the loamy soil of the Central-Chinese Loess Plateau has been found to be up to 50% (Shang et al. 2005).

Therefore a characterization of the influence of indoor parameters on the occurrence of thoron and its decay products is required. The model for the calculation of the concentrations of radon and its decay products in mines which was formulated by Jacobi (1972) and was adapted to arbitrary indoor atmospheres by several contributors (Knutson 1988, Porstendörfer 1994) can be used as a base. However, the decay chain of thoron differs from that of radon in several essential properties such as the decay modes and half-lives of the respective nuclides, which causes various significant changes in the model structure. Additionally, several transfer coefficients between the compartments of the thoron model cannot be adopted from the radon model but must be determined separately.

This study presents results from a self-contained indoor thoron compartment model (Meisenberg and Tschiersch 2010) illustrates the reasons for necessary differences to the radon model. Useful instructions for the metrology of thoron are deduced from the model predictions. Types of houses in which increased thoron concentrations can be expected are presented.

Relevant properties of thoron and its decay products

Thoron with its short half-life of 55.6 s cannot penetrate foundations which can usually be found in most modern buildings. On the other hand, a comparatively small thickness such as that of walls and ceilings can be sufficient to cause significant exhalations of thoron. Therefore the building material and thus a part of the building is the major source of indoor thoron. Hence, a thoron model must incorporate the exhalation of the gas and the influences on it.

The spatial distribution of thoron in a room is not homogeneous but features increased concentrations close to thoron exhaling surfaces. Therefore its concentration cannot be specified by a single value but must be characterized depending on the position in the room.

As with radon and its decay products, the dose after inhalation of the radioactive nuclides of the thoron decay chain is mainly caused by their alpha-particle emission. Thus the exposure to thoron can be assessed by the potential alpha-particle concentration *PAEC*, which is the total energy of the alpha-particles emitted by the decay products per volume. Therefore, a thoron model must specify the influence of indoor parameters to this quantity, which can differ from that on the activity concentration of the nuclides.

The contribution of the single decay products to the potential alpha-energy concentration of the thoron decay chain does not only depend on their position in the decay chain but also on their half-life. The long-lived ^{212}Pb ($T_{1/2}$ = 10.6 h) contributes almost the complete portion (91.3%) whereas the contribution of ^{212}Bi ($T_{1/2}$ = 1.01 h, 8.7%) is much smaller but can still be relevant. The contributions of ^{216}Po, ^{212}Po (because of their short half-lives) and ^{208}Tl (only beta-decay) are negligible.

The half-lives of the significant nuclides ^{212}Pb and ^{212}Bi are much longer than that of the nuclides in the radon decay chain. Thus, radioactive decay is less important as a sink whereas possible other sinks such as air exchange influence the concentrations of the nuclides more strongly.

Similar to those of radon, the decay products of thoron originate as single airborne atoms or ions (which cluster water molecules around them) but can also attach to aerosol particles or deposit onto surfaces. The exposure to unattached and attached decay products yields different contributions to the dose in the different parts of the respiratory tract whereas deposited decay products do not add to the exposure of people. Beside the different relevant nuclides, a thoron model must take into account these three states. Additionally, the activity size distribution of the attached decay products is another relevant input parameter for dose calculations.

With these properties, a suitable structure of a model for the indoor occurrence of thoron and its decay products is shown in Figure 1.

Material and methods

Experimental mudbrick room and real rooms

Increased concentrations of indoor thoron have been found in dwellings made of unfired mineral building material such as mud (Sreenath Reddy et al. 2004, Shang et al. 2008). For the determination of the transfer coefficients of the thoron model, a room was erected from mudbricks at Helmholtz Zentrum München. It features dimensions of l = 2.8 m, w = 1.5 m, and h = 1.8 m, a volume of 7.1 m^3, and a semi-cylindrical vault; thus it represents at a reduced scale a type of dwellings which is common on the Central-Chinese Loess Plateau. The mud of the inner plaster, which is commercially

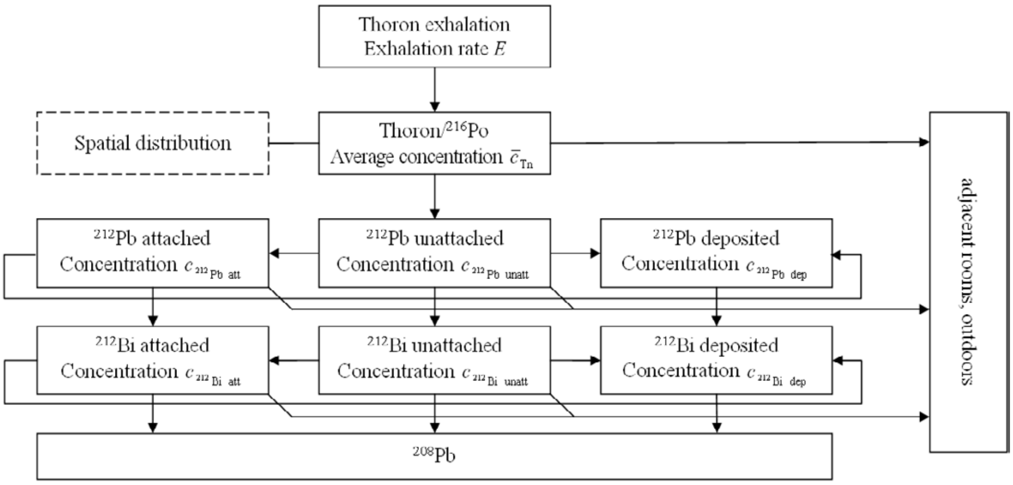

Fig. 1. Structure of the model describing the indoor concentration of thoron and its decay products.

available (Claytech, Germany), was mixed with thorium-rich granite to increase the exhalation of thoron for higher precision of short-time measurements; its specific ^{232}Th activity is about 50 Bq kg^{-1}.

For a validation of the measurement results from the experimental room, measurements were also performed in mudbrick dwellings on the Central-Chinese Loess Plateau and in northern India.

Measurement of thoron concentrations and exhalation rates

Thoron concentrations were measured with the common radon and thoron measurement device Rad-7 (Durridge, USA), which uses the electrostatic deposition of newly formed thoron decay products. It conveys air with a volume flow rate of about 1 L min^{-1} from the end of an inlet tube.

Thoron exhalation rates were measured by means of the equilibrium concentration which is reached in a closed accumulation chamber (Tuccimei et al. 2006). In contrast to radon, leakages and back-diffusion do not need to be taken into account because of the short half-life of thoron. Samples for which the exhalation rate was measured were sealed with thoron-proof lacquer on all sides but one.

Measurement of concentrations of thoron decay products

Working level monitors (commercially available from Tracerlab, Germany; Peter 1994) were used for the measurement of unattached and total thoron decay product concentrations. The decay products were deposited on a sampling substrate with an air flow rate of about 150 L h^{-1}. Their decay was measured online by alpha spectrometry.

For the deposition and measurement of the total thoron decay product concentration, a cellulose nitrate filter was used as the sampling substrate. A wire screen with a mesh size of 625 µm was applied as the sampling substrate for the measurement of unattached decay products (Meisenberg and Tschiersch 2009). With the given air flow rate, this screen features a negligible deposition probability for attached decay products whereas unattached decay products are sampled with an efficiency of about 40%.

Further measurement methods

Specific activities of ^{232}Th in solid samples were measured by gamma spectrometry using its gamma-emitting decay products in equilibrium with it.

Air exchange rates of rooms were measured by one-time release of CO_2 as a tracer gas to the indoor air. The concentration of the homogeneously spread CO_2 decreased by air exchange with clean ambient air. The rate of decrease is the air exchange rate.

Aerosol number concentrations were measured with a condensation particle counter (3022A, TSI, USA) with a lower cut-off diameter of 7 nm and an air flow rate of 18 L h^{-1}.

Results and Discussion

Exhalation of thoron from mud

The exhalation rate E of thoron was measured at various samples of mud such as bricks, plaster, and rammed earth. A dependency on the specific ^{232}Th activity A_{spec}, which can be approximated by a linear function, was found (Figure 2):

$$E = (0.031 \pm 0.003)\,\mathrm{kg\,m^{-2}\,s^{-1}} \cdot A_{spec}\,.$$

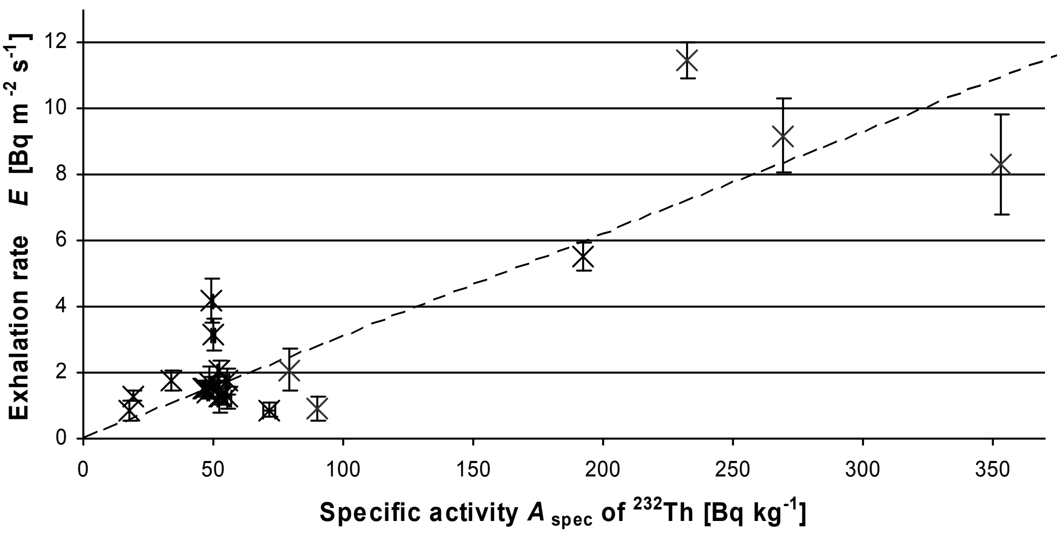

Fig. 2. Exhalation rate of various mud samples from Europe and Asia as a function of their specific ^{232}Th activity as well as the line of linear regression.

For the determination of the diffusion length of thoron in mud, the exhalation rate was measured at two samples from which a few millimeters of mud were removed between each measurement of the series. The exhalation rate remained constant until thicknesses of less than 2 cm were reached.

At several samples of mud, the exhalation rate after firing the sample was compared with that in the unfired state. For this purpose, the samples were fired at a temperature of 900 °C for three days. The exhalation rates were reduced on average to only 6.4 ± 5.1% of the initial value.

Although a distinct influence of the specific ^{232}Th activity in the mud on the exhalation of thoron was observed, even ordinary ^{232}Th activities can lead to increased concentrations of indoor thoron and its decay products: Samples of mudbricks and plaster from the Central-Chinese Loess Plateau featured specific activities of 53 ± 7 Bq kg^{-1} on average, whereas specific activities of 60 Bq kg^{-1} on average (with values up to 310 Bq kg^{-1}) were found in natural building stones from Europe (EC 1999).

The short diffusion length in the mud causes a significant thoron exhalation even from thin layers of mud as plasters. However, fired bricks and clinkers do not seem to be a relevant source of thoron; there, the pores are obstructed by the partly melted minerals.

Spatial distribution of thoron

In the experiment room, the concentration of thoron was measured at several distances from a thoron-exhaling wall. Several measurement series were performed at different air exchange rates. In all measurement series, a decrease of the concentration within a few centimeters was found; with increasing air exchange rates from 0.1 h^{-1} to 4.5 h^{-1} the slope of the decrease became smoother whereas with even higher air exchange rates the gradient increased again. With an air exchange rate of only 0.1 h^{-1} almost no thoron could be found in the middle of the room whereas with higher and still realistic air exchange rates concentrations of up to 15% of the concentration directly at the wall were observed (Figure 3).

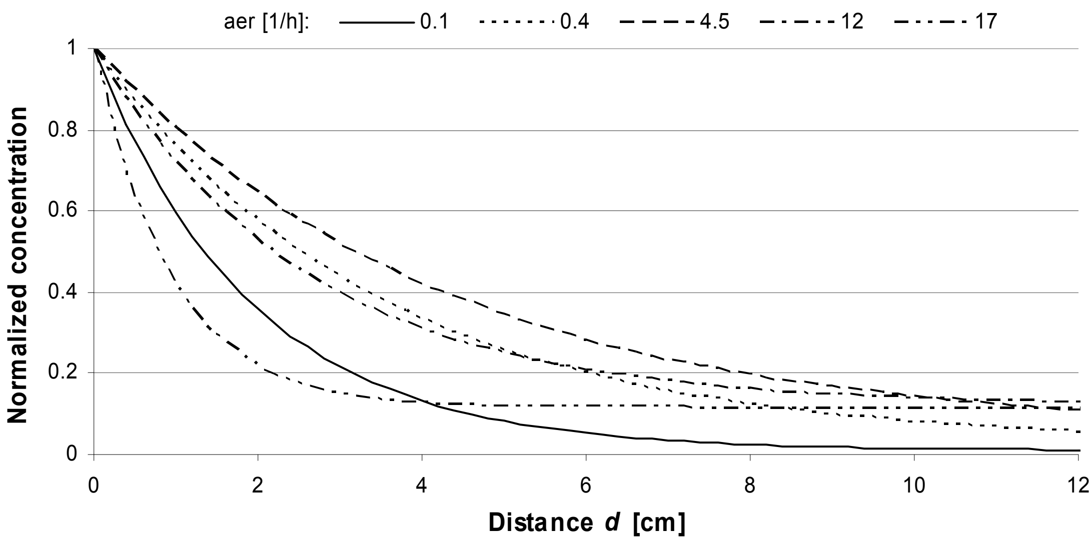

Fig. 3. Concentrations of indoor thoron (relative to the value directly at the wall) as a function of the distance from a thoron-exhaling wall at different air exchange rates aer.

The concentration profile at the lowest air exchange rate is governed by the diffusive transport of the gas. The Gaussian error function describes the spatial distribution of particles which are transported by diffusion if their source spans over one half-space as it is the case with thoron from a wall. When this function is fitted to the measured data, the diffusion length of thoron in air can be determined as 2.6 ± 0.2 cm.

The thoron concentration especially in the middle of the room is strongly influenced by the mixture of the air – in this case caused by air exchange.

Concentration of the thoron decay products

Concentrations of the thoron decay products were measured at different air exchange rates aer in the experimental room. Figure 4 presents the results by means of the equilibrium factor *F*, which is the concentration of the decay products weighted by their contribution to the *PAEC* relative to the average concentration of thoron.

The strong influence of the air exchange on the equilibrium factor and thus on the decay product concentration can be seen; this influence as well as that of possible further sinks is promoted by the small decay constant λ of 0.065 h^{-1} of ^{212}Pb as the most important decay product. The values obey a $\lambda/(\lambda+aer)$ relation. Usual air exchange rates in modern dwellings are in the range from below 3 to about 10 h^{-1} (i.e. *F* in the range from 0.02 to 0.006) with a trend to lower values of less than 1 h^{-1} in low-energy houses ($F = 0.06$). Typical values for the radon decay chain are in the range from 0.4 to 0.2 (Porstendörfer 1994).

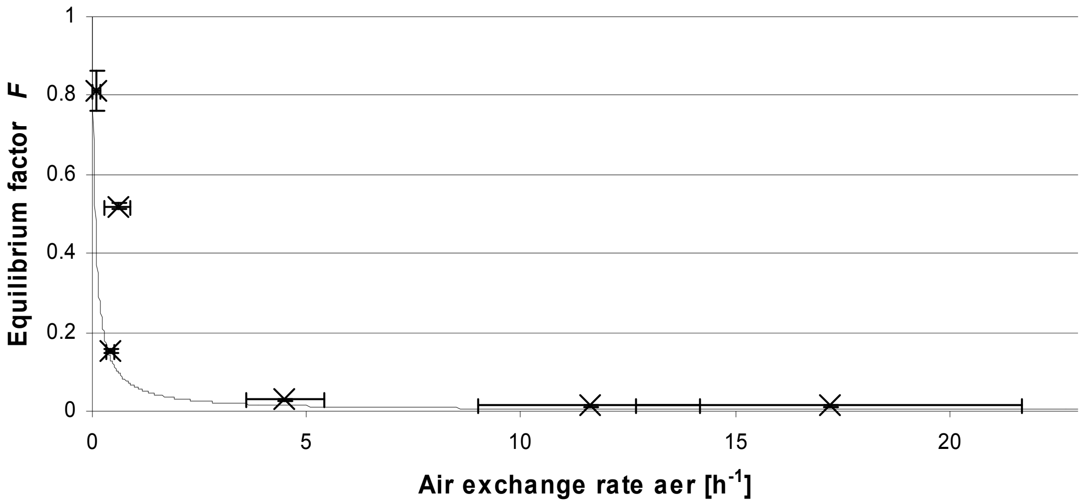

Fig. 4. Equilibrium factor of the thoron decay chain as a function of the air exchange rate: Measurement results from Chinese (X) and from Indian (◊) dwellings and fitted curve.

Unattached fraction of the decay products

The unattached fraction of the thoron decay products was measured in the experiment room at different aerosol concentrations as its most important influence (Figure 5). The air exchange rate during the measurements was 0.1 h^{-1}. Similar to the equilibrium factor, this parameter plays an important role for the unattached fraction because of the long half-lives of the thoron decay products.

The observed values are much smaller than those of the radon decay products with values between 0.05 and 0.20 for usual indoor conditions (aerosol concentrations between 1000 and 20000 cm^{-3}; Porstendörfer 1994). With higher air exchange rates, however, a greater unattached fraction can be expected because of the shorter lifetime of the decay products in the room.

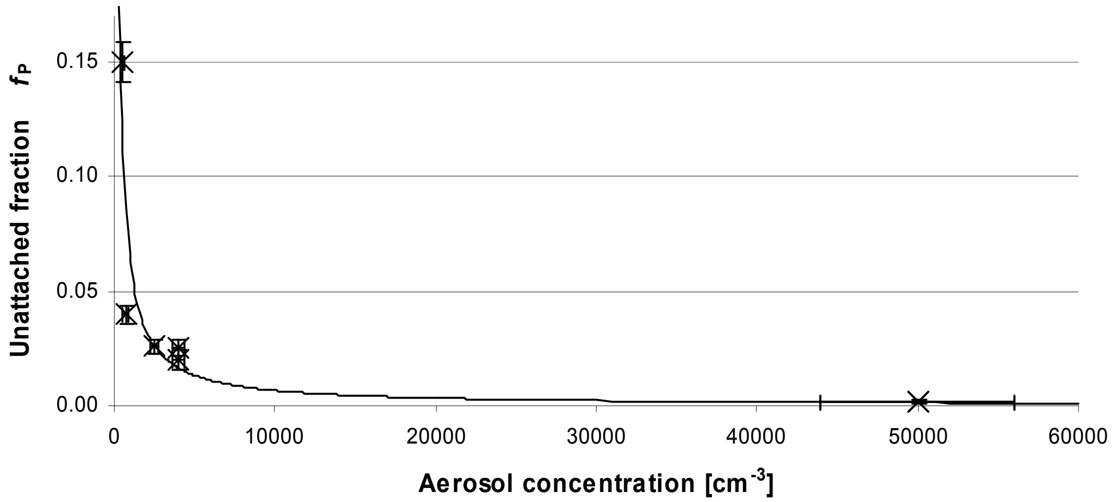

Fig. 5. Unattached fraction as a function of the aerosol concentration and fitted curve.

Conclusions

Consequences on the metrology of thoron exposures

For the assessment of the exposure of people to radioactive gases, the relevant quantities for the calculation of the inhalation dose must be measured. Both for radon and thoron, these are the concentrations of the airborne decay products. For radon, however, a single indoor measurement of the gas can usually provide a satisfying estimate of the inhalation dose.

For thoron, two reasons impede the use of gas measurements for the assessment of the exposure to the decay products: Firstly, the thoron concentration is not homogeneous and does not even feature a constant profile with different indoor atmospheric conditions. Thus, a single measurement at one position cannot give information about the average thoron concentration in the room. Secondly, air exchange plays such an important role as a sink of the long-lived thoron decay products that their concentrations can vary by about one order of magnitude with different air exchange rates. Thus, even the average thoron concentration is not a significant quantity for the decay product concentration if the air exchange rate is unknown.

Therefore, the measurement of the decay product concentration itself is essential. Separate measurements of the concentrations of ^{212}Pb and ^{212}Bi and of the unattached and the attached state are helpful but less important. The development of cheap and simple passive detectors for the decay products is desirable.

Calculation of inhalation doses

Recently, inhalation dose coefficients for the nuclides of the thoron decay chain were published (Li et al. 2008). With these values, their concentrations in mudbrick dwellings of the Central-Chinese loess plateau can yield annual effective doses in the range of up to 2 mSv if a daily indoor stay of 10 h is assumed. Even with a small aerosol concentration of 6000 cm^{-3}, more than half of the dose is contributed by attached ^{212}Pb. Unattached ^{212}Pb and, even less, attached ^{212}Bi add the remaining part.

A calculation of expected concentrations of thoron and its decay products in European mud dwellings shows that they can cause similar doses also there.

Indication for potentially increased thoron exposures

As increased concentrations of thoron and its decay products were found mostly in buildings of unfired mineral building material such as mud, a survey of thoron exposures should initially focus on such buildings. In Europe, unfired mud is a traditional building material for timber-framing buildings but also gains attention as an ecological material in modern indoor architecture. As overviews over the content of ^{232}Th in building materials exist (e.g. EC 1999), indication of areas of mud with high concentrations of ^{232}Th is given.

Air exchange is an effective sink of the long-lived thoron decay products. Therefore, rooms which feature small air exchange rates are prone to high concentrations of the decay products in particular. The refurbishment of traditional buildings aiming at reduced energy consumption might increase the number of rooms where both small air exchange rates and mud as the building material are present.

The concentrations of thoron (averaged over a room) and of the decay products are proportional to the ratio of the area of the thoron-exhaling surfaces to the room volume. Therefore, small rooms, in which this ratio is greater, are more susceptible to increased concentrations than large rooms.

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First results of measurements of equilibrium factors *F* and unattached fractions f_p of radon progeny in Czech dwellings

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Abstract

The unattached fraction of radon decay product clusters f_p and equilibrium factor F are dose relevant parameters in all dosimetric approaches to dose calculation. In the past, three year weekly continuous measurements of unattached and attached activities of radon daughter product and air exchange rate were carried out during heating season in thirty occupied typical Czech family houses.

The results indicated significantly different weekly averages of equilibrium factor F and f_p for houses located in towns compared to villages. Due to this fact, approximately 10% average increase of equivalent lung dose rate was estimated in the detriment in towns.

Average values of equilibrium factor F and f_p were 0.40 and 8.6% in urban houses and 0.32 and 10.7%, respectively in rural houses.

Based on measurements of mean values of f_p, average effective dose conversion coefficients per WLM were estimated at 15.0 mSv/WLM in urban houses and 15.9 mSv/WLM in rural houses, respectively.

Introduction

The unattached fraction of radon decay product clusters f_p and equilibrium factor F are dose relevant parameters in all dosimetric approaches to dose calculation. Direct measurements of f_p and F under different ambient conditions allow a comparison of influence of ambient conditions to relative dose contribution [4].

The results of sensitivity analysis applied to the classical Jacobi–Porstendörfer room model [5] generally describing dynamics of both unattached and attached short-lived radon progeny in the room indicate *ACH* (air exchange rate), deposition on surfaces q_f and attachment rate to aerosols X of unattached radon daughter product as the key quantities influencing most strongly the behaviour of observed values f_p and equilibrium factor F in a room. The most important role in processes affecting values of deposition q_f and attachment rate X plays, besides *ACH,* predominantly aerosol size distribution and total aerosol concentration.

In the past, three year weekly continuous measurements of unattached and attached activity concentration of each radon progeny and of *ACH* were performed during heating season in thirty occupied typical Czech family houses.

Primarily, the purpose of the present paper is to introduce the results of direct measurements of F and f_p under different living conditions and to estimate the influence of these conditions to dose contribution.

We also focused on the calculation of the key model parameters X, q_f and *ACH* in measured houses in order to independently estimate the most probable values of F and f_p by means of the room model and to compare them with those obtained from direct measurements.

Since there are well known practical and principal problems arising from long-term measurement of aerosols in occupied dwellings with any SMPC (Scanning mobility particle sizer) caused by evaporating of wide spread butyl alcohol saturator used in particle condensation nuclei counters into the room, aerosol particle size distribution and aerosol concentration were not systematically directly measured. Nevertheless, aerosol concentration Z could be estimated indirectly from measured values f_p by means of Porstendörfer's approximate formula ($f_p = 414/Z$) [3].

Material and methods

All measurements were conducted in thirty occupied typical Czech or two-storey family houses from October to March during heating season. The houses were chosen randomly and they were equipped with central heating system. Originally, we intended to divide them according to assumed different ambient living conditions from the aerosol concentration and its size distribution standpoint in structure as follows: towns – villages, smokers – non smokers. Unfortunately, after finishing all measurements we had to distinguish only basic living conditions according to the house location i.e. town – village and newly according to daily time period.

To detect the assumed influence of different occupant's indoor activities and outdoor changes (traffic density, weather conditions etc.) on behaviour of observed parameters f_p and F we divided each daily 24 hours taking measurement time period into "day" taking from 6 a.m. to 6 p.m. and into "night" taking from 6 p.m. to 6 a.m., respectively. Exposure duration took minimally one week in each house.

Measurement instruments

For measurement of unattached and attached activity concentration of each short-lived radon daughter progeny, we used continuous monitor Fritra 4 (SMM – Prague, CZ). Monitor Fritra 4 has built-in memory buffer allowing more than 4000 records taking each 120 minutes. This measuring interval included ten minutes sampling and 110 minutes taking computational procedure with implemented four alpha counting intervals providing minimum detectable activity (MDA) for each radon progeny in following ratio (^{218}Po: ^{214}Pb: ^{214}Bi) = (1: 0.6: 0.5) stepwise 40 Bq.m^{-3}, 12 Bq.m^{-3}, 15 Bq.m^{-3}. The MDA for measured radon concentration was estimated to about 15 Bq.m^{-3}. Unattached activity concentration of each radon progeny was measured from the screen with mesh 300 and cut-off diameter d_{50} = 4 nm and the attached activity concentration of each radon progeny from the Millipore filter type AA, 0.8 µm placed behind the screen.

The description of monitor Fritra 4 alone in more details and quality assurance from measurement of unattached and attached activity concentration for each radon progeny and from measurement indoor average aerosol concentration standpoint is given elsewhere [2].

Radon measurements were performed by means of well-known continuous monitor Alphaguard (Genitron, D) certified at the German reference chamber at the PTB in Braunschweig. During all measurements, we used its 60 minutes measuring diffusion mode. The both monitors were placed mostly in living rooms at each house.

Theoretical background

i) Equilibrium factor F and f_p as dose relevant parameters

The lung equivalent dose H from inhalation of short-lived radon progeny can be written by means of measured time integral of radon concentration as follows:

$$H = DFC \cdot \frac{T_{\exp}}{170} \cdot F \cdot \frac{a_{\mathrm{V}}}{3700} \tag{1}$$

where

DFC is the dose conversion factor [mSv/WLM],
F is the equilibrium factor,
a_V is the average radon concentration [Bq m^{-3}],
$T_{\exp}$ is the exposition duration [h].

According to calculations by Marsh and Birchall [4] the *DFC* in eq. (1) can be expressed as a linear function of unattached part of equilibrium equivalent concentration of radon progeny f_p as follows:

$$DFC(f_p) = 11.35 + 0.43\, f_p \tag{2}$$

with f_p expressed as the percentage of total PAEC.

Finally, considering eq. (1) and (2) and assuming “the same” breathing rate, the influence of two different ambient conditions to dose can be expressed in terms of relative change of equivalent dose rate H_1/H_0 as follows:

$$\frac{H_1}{H_0} = \frac{F_1 \cdot (k + f_{p,1})}{F_0 \cdot (k + f_{p,0})} \tag{3}$$

with subscripts 0 and 1 representing two different ambient conditions and k - representing the ratio 11.35/0.43 from eq.(2).

ii) Attachment rate X, deposition rate q_f and air exchange rate ACH

The attachment rate X and deposition rate q_f (plate-out) and ACH can be calculated in principle simultaneously from measured unattached and attached activity concentration of each radon daughter progeny by means of an algebraic inversion of the model [2] expressed for room in steady state as follows:

$$
\begin{aligned}
a_{\mathrm{j,f}} &= \lambda_{\mathrm{i\text{-}1}}(a_{\mathrm{j\text{-}1,f}} + R_{\mathrm{j\text{-}1}}a_{\mathrm{j\text{-}1,a}}) / (ACH + \lambda_{\mathrm{j}} + q_{\mathrm{f}} + X) \\
a_{\mathrm{j,a}} &= (ACH.a_{\mathrm{j,o}} + (1 - R_{\mathrm{j\text{-}1}})\lambda_{\mathrm{j}}a_{\mathrm{j\text{-}1,a}} + Xa_{\mathrm{j\text{-}1,f}}) / (ACH + \lambda_{\mathrm{j}} + q_{\mathrm{a}}) \\
a_{0,\mathrm{i}} &= (Q_{Rn} + ACHa_{0,o}) / (\lambda_0 + ACH)
\end{aligned} \tag{4}
$$

with initial conditions $a_{0,\mathrm{i}} = a_{0,i,\mathrm{f}}, a_{0,\mathrm{i,a}} = 0, a_{\mathrm{j,o,f}} = 0$

a_{j} - activity concentration of j-th radon daughter product [Bq m^{-3}]

λ_{j} - decay constant of j-th radon daughter product [h^{-1}]

R_{j} - fraction of recoiled atoms of j-th nuclide from aerosol particles

1$^{\mathrm{st}}$ subscript j = 0, 1, 2, 3, 4 stepwise ^{222}Rn, ^{218} Po, ^{214}Pb, ^{214} Bi, ^{214} Po

2$^{\mathrm{nd}}$ subscript i, o - denotes indoor/outdoor

3$^{\mathrm{rd}}$ subscript f, a - denotes unattached/attached activity j-th radon daughter product

Q_{Rn} - sum of radon entry rates into a house [Bq m^{-3} h^{-1}]

Attachment rate

Under realistic indoor conditions, the observed attachment rate X in a room in „steady state" can be approximately calculated only on the base of measurement of unattached and attached activity concentration ^{218}Po as follows [3].

$$X = \frac{a_{1\mathrm{ia}}}{a_{1\mathrm{if}}}\lambda_1 \tag{5}$$

where

$a_{1\mathrm{ia}}$, $a_{1\mathrm{if}}$ are attached and unattached concentrations, respectively ^{218}Po [Bq m^{-3}]

λ_1 is decay constant ^{218}Po [h^{-1}]

or in case of measurement the proper aerosol size distribution as follows:

$$X = \int_0^\infty \beta(d)\mathrm{Z}(d)\mathrm{d}d \tag{6}$$

where

$\beta(d)$ - attachment coefficient of attached radon progeny to aerosol as a function of particle diameter d [cm^3 h^{-1}]

Z(d) - aerosol size distribution [cm^{-3}]

According to [1] attachment coefficient $\beta(d)$ can be calculated as follows:

$$\beta(d) = \frac{2\pi D_0 d}{\frac{8D_0}{dv_0} + \frac{d}{2\delta_0}} \tag{7}$$

where

D_o is diffusion coefficient 6.8 10^{-2} cm^2 s^{-1}

v_o is mean thermal velocity 1.72 10^4 cm^{-1}

$\delta_o = d/2 + s_o$ with s_o = 4.9 10^{-6} cm is the mean free path of the unattached decay product clusters.

Plate-out

According to the model, the deposition rate q_f, of unattached daughter products can be calculated approximately under realistic indoor conditions ($ACH << X + \lambda_1$) as follows:

$$q_f = \lambda_1 \frac{a_0}{a_{1f}} - (\lambda_1 + X) \tag{8}$$

with all symbols as in previous text.

Air exchange rate

Under the assumptions of the model introduced by the set of eq. (4), the *ACH* can be calculated both by means of an algebraic inversion given previously and by means of numerical iterations applying the set of eq. (4).

On the basis of our previous experiments carried out in the radon chamber of NRPI (National Radiation Protection Institute), we developed the code calculating *ACH* in principle as follows:

Input data for the code were measured lower and upper limit of ratios activity concentrations stepwise Rn/^{218}Po, ^{214}Pb/^{218}Po, ^{214}Bi/^{218}Po and lower and upper limit estimates of *X*, q_f and *ACH*. The code provides stepwise outputs in a form of sets of required parameters *X*, q_f, *ACH* and recoil *R* fulfilling all input conditions. The most probable value of *ACH* was chosen from the set in which the values *X* and q_f matched the best to values *X* and q_f, calculated according to eq. (5) and eq. (8) respectively on the basis measured values of attached and unattached concentration of ^{218}Po. A large number of tests previously performed under the defined *ACH* in the NRPI radon chamber indicated overall agreement of calculated and true values of *ACH* up to acceptable 30%.

iii) Statistical evaluation

The differences between measurements were evaluated using the Student test and assuming log-normal distribution.

Results and discussion

The results of confirmatory data analysis from all measured houses (N=30) in the form of weekly geometric means (GM), corresponding standard deviation (GSD), 95% confidence interval (95% CI) for desired quantities of interest and p-value are summarized in Table 1 and Table 2.

In Table 1 are shown results of the analysis applied to measured data for compared time periods day – night. From Table 1 by means of p-values can be seen that all quantities of interest are not significantly different, except *ACH*.

The observed statistically significant approximate average 30% increase of *ACH* during day compared to night was not surprising. Due to this fact, a higher indoor aerosol concentration coming from outdoor air is assumed and implicitly higher attachment rate *X* (see eq. (6)) is expected and also measured.

The results of attachment rate *X* calculated according to eq. (5) can be approximated by their basic parameters (GM = 24 h^{-1}, GSD = 1.63) for time period day and (GM = 20 h^{-1}, GSD = 1.41) for time period night, respectively. These results agree fairly well with published data. According to [1], attachment rate *X* is ranging from 20 h^{-1} to 50 h^{-1} in poor ventilated houses (ACH < 0.5 h^{-1}) without additional aerosol sources.

Considering eq. (6) and the average attachment coefficient $\beta(d)$ = 0.005 cm^3 h^{-1} published by Porstendörfer in case of poor ventilated houses (ACH < 0.5 h^{-1}) without additional aerosol sources, we can expect average indoor aerosol concentration Z ≈ 4800 cm^{-3} during day and Z ≈ 4000 cm^{-3} during night, respectively.

The results of measured deposition rate q_f calculated according eq. (8) can be approximated by their basic parameters (GM = 23 h^{-1}, GSD = 1.48) for time period day and (GM = 22 h^{-1}, GSD = 1.35) for time period night, respectively. These measured results agree relatively well with published by other authors 40 h^{-1} [7], 10 h^{-1} [3], 30 h^{-1} [10].

According to already cited [1] typical published average values F and f_p in indoor air range between 0.2 - 0.4 and 0.05 - 0.20, respectively for "normal" aerosol conditions Z ≈ (1000 - 20 000) cm^{-3} with mean values of F = 0.3 and f_p = 0.096. These results are in good agreement with our measured results f_p (GM= 0.099, GSD= 1.58), F (GM = 0.35, GSD = 1.31) found for time period day and f_p (GM= 0.107, GSD= 1.47), F (GM = 0.36, GSD = 1.32) found for time period night.

In Table 2 are shown results of the analysis applied to measured data for compared locations town – village. From p-values in Table 2 can be seen that all quantity of interest are significantly different except deposition q_f and the fact that observed mean values F, f_p, X and q_f agree very well with the published data mentioned in previously.

In our opinion, the higher average value F and logically lower average value f_p in indoor air in towns compared to villages are caused mainly by higher average aerosol concentration probably in consequence of different indoor activities. This fact is indicated by dramatically higher values of attachment rate X in towns compared to villages and statistically significant lower average value *ACH* approximately about 30%.

If we consider the above mentioned average attachment coefficient $\beta(d)$ = 0.005 cm^3 h^{-1} published by Porstendörfer, eq. (6) and measured means of attachment rates X from Table 2, we are able to estimate average indoor aerosol concentration Z ≈ 5200 cm^{-3} in case of towns and Z ≈ 3600 cm^{-3} in case of villages. From Table 2 can be also seen that measured average value *ACH* is ranges around (0.3 - 0.4) h^{-1} in case of both towns and villages.

If we consider Porstendörfer's approximate formula f_p = 414/Z mentioned previously (see Introduction) and measured average values of f_p calculated for different observed conditions from Table 1 and Table 2, we can estimate average value of indoor aerosol concentration as follows:

- Z ≈ 4800 cm^{-3} in case of towns and *Z* ≈ 4000 cm^{-3} in case of villages respectively
- Z ≈ 4200 cm^{-3} for time period day and *Z* ≈ 3900 cm^{-3} for time period night.

These results indicate very good agreement of both approaches used to estimate average aerosol concentration *Z* i.e. by means of measured attachment rate X and f_p, respectively and also implicitly a good accuracy of measurements of unattached and attached activity concentrations of radon daughter product by means of monitor Fritra 4.

Since average value f_p in measured urban houses were 8.6 % and 10.6 % in rural houses then with respect to eq.(2) lung dose sensitivity factor DFC= 15.0 mSv/WLM should be applied in case of urban houses and DFC =15.9 mSv/WLM in case of rural houses, respectively. Further, according to [4], the lung dose sensitivity factor DFC contributes to more than 99% of the effective dose per WLM and therefore the above mentioned values of DFC can represent very well the effective dose conversion coefficient per unit WLM.

To verify the Jacobi–Porstendörfer model (see eq.4) from the prediction of *F* and f_p point of view, in Figs. 1 and 2 are illustrated results of comparison of measured values *F* and f_p with those calculated by means of the room model on the basis of input data, i.e. average values *X*, q_f and *ACH* (taken from Table 1 and Table 2) for investigated ambient conditions. For all calculations, average value of recoil R = 0.5 [1] was used. The results indicate a very good agreement between directly measured values of *F* and f_p and calculated by means of the room model.

Assuming "the same breathing rate" the influence of our different investigated ambient conditions (denoted 0 and 1 respectively) to relative changes of the equivalent lung dose rate H_1/H_0 is illustrated in Fig. 3.

In Fig. 3 is seen the average value of H_1/H_0 calculated for proper observed ambient conditions applying eq. (1-3) and combined standard uncertainty acquired by application of the Law of Propagation of Uncertainty to eq.(3).

About 10% significant relative increase of the equivalent lung dose rate in case of towns compared to villages was found. On the other hand, difference in contribution to relative lung dose rate in case of observed conditions day – night was not statistically significant.

Table 1. Results of confirmatory analysis of weekly average values f_p, F X, q_f and *ACH* over time periods day and night.

Quantity (units)	Period	GM	GSD	95% CI		p- value
f_p	Day	0.099	1.58	0.04	0.247	
	Night	0.107	1.47	0.05	0.231	p = 0.494
F	Day	0.35	1.31	0.20	0.60	
	Night	0.36	1.32	0.21	0.63	p = 0.533
X	Day	24	1.63	9	64	
(h^{-1})	Night	20	1.41	10	40	p = 0.130
q_f	Day	23	1.48	11	50	
(h^{-1})	Night	22	1.35	12	40	p = 0.653
ACH	Day	0.41	1.59	0.16	1.04	
(h^{-1})	Night	0.30	1.82	0.09	0.99	p = 0.024

F, f_p denote equilibrium factor and unattached fraction of equivalent equilibrium concentration, respectively
X, q_f denote attachment rate and deposition rate of unattached radon progeny
ACH denotes air exchange rate.

Table 2. Results of confirmatory analysis weekly average values of *fp*, F X, q_f and *ACH* for location town and village.

Quantity (units)	Location	GM	GSD	95% CI		p- value
f_p	Town	0.086	1.58	0.04	0.247	
	Village	0.107	1.40	0.05	0.231	p = 0.003
F	Town	0.40	1.25	0.26	0.63	
	Village	0.32	1.31	0.19	0.55	p = 0.002
X	Town	26	1.61	10	67	
(h^{-1})	Village	18	1.37	10	34	p = 0.001
qf	Town	23	1.41	12	46	
(h^{-1})	Village	22	1.42	11	44	p = 0.661
ACH	Town	0.30	1.91	0.08	1.09	
(h^{-1})	Village	0.40	1.56	0.16	0.97	p = 0.046

The quantities f_p, *F*, *X*, q_f and *ACH* have the same meaning as in previous Tab.1.

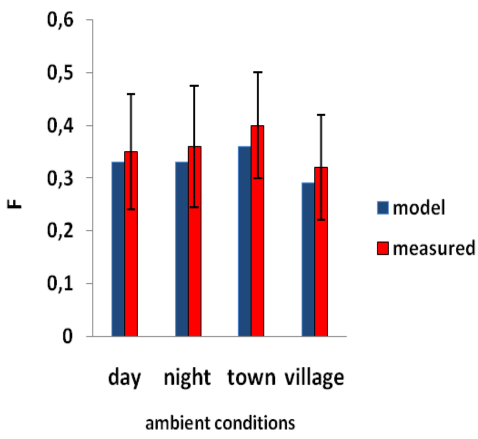

Fig. 1. Results of comparison of equilibrium factor *F* calculated via the room model and directly measured, respectively (with GSD).

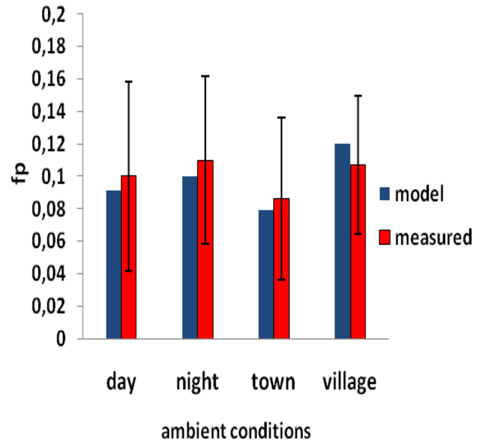

Fig. 2. Results of comparison of f_p calculated via the room model and directly measured, (with GSD)

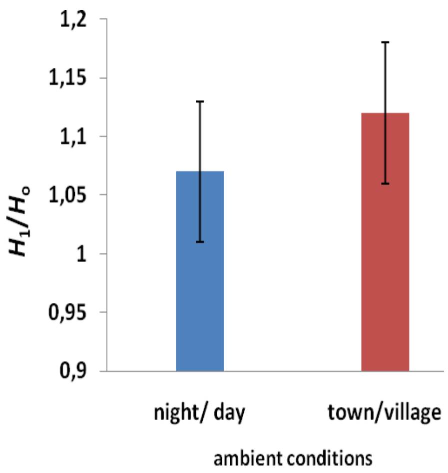

Fig. 3. Average relative contribution to lung equivalent dose rate H_1/H_o under different ambient conditions and the corresponding combined standard uncertainty.

Conclusions

The results of weekly continuous measurements of radon concentration and unattached and attached concentrations of each short-lived radon progeny in thirty randomly chosen Czech family houses during heating season indicated statistically significant difference of weekly averages equilibrium factor F a f_p for urban houses compared to rural. The log-normal distribution of weekly means f_p and F is represented in case of towns as follows: f_p (GM= 0.086, GSD = 1.58), F (GM = 0.40, GSD = 1.25) and f_p (GM = 0.107, GSD= 1.40), F (GM = 0.32, GSD = 1.31), respectively in case of villages. Due to this fact approximately 10% average increase of equivalent lung dose rate was calculated in the detriment of towns.

The difference in mean values f_p and F during observed time period day - night was not statistically significant.

The effective dose conversion coefficients per WLM were estimated to 15.0 mSv/WLM for towns and 15.9 mSv/WLM for villages, respectively.

Both key parameters of the Jacobi-Porstendörfer room model, i.e. attachment rate X and plate -out q_f of unattached radon progeny ranged from 10 h^{-1} to 60 h^{-1} with the mean around (20 – 25) h^{-1}.

The air exchange rate ranged from 0.1 h^{-1} to 1 h^{-1} with the mean approximately around (0.3 – 0.4) h^{-1} and its statistically significant average 30% decrease during the night compared to day was observed.

The average indoor aerosol concentration was significantly higher in towns ($Z \approx$ 5000 cm^{-3}) compared to villages ($Z \approx 4000$ cm^{-3}). On the other hand, difference between mean indoor aerosol concentration during observed time period day ($Z \approx 4200$ cm^{-3}) was not statistically significant in comparison to night time period ($Z \approx 3900$ cm^{-3}).

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A comparison of one and three month radon measurements in Ireland

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Abstract

Indoor radon concentrations are subject to high temporal variation that can make short term measurements of radon unreliable and difficult to interpret. Oscillations in radon concentrations can be smoothed by carrying out long-term measurements using CR-39 alpha track detectors, with a typical measurement period of three months or more. Short term radon measurements can be unreliable when used to predict how average long term radon concentrations within a building compare with the Reference Level. Ideally, the measurements would be carried out over a full year as this would provide the most accurate result. However, householders are often reluctant to wait so long for results and detectors left in place for so long are commonly lost. Furthermore, there is evidence to suggest that the sensitivity of a detector may change over such a long period of time, increasing the uncertainty associated with the result. Currently in Ireland, radon measurements are carried out over a minimum period of three months. This is a compromise between the need for a long exposure period to provide a reliable estimate of the long-term mean radon concentration and the problems associated with year-long measurements. The Radiological Protection Institute of Ireland (RPII) have recommended that radon be included in the conveyancing process but recognise that a 3 month measurement period may create too much of a delay when householders are likely to require a radon result in a shorter timeframe. This project compares the results of one month measurements with three month measurements using CR-39 alpha track detectors. Statistical analysis of the results from 662 homes around Ireland are reported.

Introduction

Radon is a naturally occurring radioactive gas which originates from the decay of uranium present in rocks and soils. When radon surfaces in the open air, it is quickly diluted to harmless concentrations, but when it enters an enclosed space, such as a house or any other building, it can sometimes accumulate to relatively high concentrations. In air, radon decays quickly to produce radioactive particles that, when inhaled, are deposited in the airways and in the lungs. These result in a radiation dose which, over a long period of time, may lead to lung cancer. International Agency for

Research on Cancer (IARC) has classified radon as a class 1 carcinogen (IARC, 1988). Worldwide exposure to radon in homes is linked to 3% to 14% of lung cancers, depending on the average radon concentration in the country and method of calculation. (WHO, 2009). In Ireland the average indoor radon concentration is 89 Bq/m^3 which is the eight highest in the world. (Fennell *et al*., 2002, WHO, 2009). When risk estimates for indoor radon are applied to Ireland some radon can be linked to some 10 to 15% of all cases of lung cancer in Ireland and it is the second most significant cause of lung cancer after smoking (RPII and NCRI 2005).

Over 91,000 homes in Ireland are predicted to have radon concentrations above the national Reference Level of 200 Bq/m^3. This equates to over 7% of the national housing stock. (Fennell *et al.,* 2002). In 2006, the RPII recommended that radon measurements, and where necessary, remediation at the time of sale and/or purchase of buildings is an appropriate mechanism to reduce the collective dose from radon in Ireland. (RPII, 2006). There are currently some 4000 radon measurements carried out in Ireland each year and there are 1.9 million dwellings in the country. (Department of Environment, Heritage and Local Government (DEHLG), 2008, RPII, 2009). Therefore it could take over 400 years for the present housing stock to be measured and for those houses predicted to have high radon levels to be identified. In 2007, the number of second-hand homes purchased was almost 38,000 (DEHLG, 2007) therefore if a radon measurement was linked to the conveyancing of houses each house in the country could be tested for radon within about 40 years.

Current radon measurement practice in the RPII is to use CR-39 alpha track detectors over an exposure period of at least three months in conjunction with the application of a seasonal correction factor (Colgan *et al.,* 2008; RPII, 2008). Normal practice is to send two detectors with each measurement pack; one to be placed in a bedroom and one in a living area (RPII, 2008). The average of both rooms is seasonally adjusted and it is this seasonally adjusted average which is compared to the Reference Level. One potential difficulty with linking radon to conveyancing is the need for a three month measurement period as householders are often unwilling to wait so long. Conseqently some countries have introduced shorter ”screeening” measurements for use during conveyancing. (Ibrahimi and Miles, 2009). Extrapolating from shorter measurements to an annual average introduces uncertainties into the result and these will normally be much larger than the uncertainties caused by counting and calibration errors on the detectors [Miles, 2001]. Therefore it is prudent only to use short term screening measurement in selected circumstances, such as conveyancing, and only when the relationship between the short term measurements and the standard long term measurement is known. This study was, therefore carried out to determine this relationship and to investigate possibility of using a one-month “screening” measurement during conveyancing to determine whether a house is above or below the Reference Level.

Material and methods

This project was run in two phases. Phase 1 was carried out during 2008. The RPII contacted 102 customers who had recently applied for a standard radon measurement and invited them to instead participate in this study free of charge. All of those contacted agreed to participate and the households were distributed all over Ireland.

Initial analysis of the detectors revealed that 89% of the three month detectors had radon levels less than 200 Bq/m^3. The dataset did not contain sufficient data for higher radon concentrations therefore phase 2 of the project sought to obtain data on the relationship between three month observations and one month observations at higher radon concentrations. The RPII contacted 840 customers who had previously measured radon levels above 200 Bq/m^3 in their homes. Those contacted were from areas all over Ireland.

In order to make the reply process as easy as possible, the invitation letter had a response slip at the bottom for people who wished to participate, with spaces to add a contact phone number, an email address and a signature. Also enclosed was a freepost envelope to return the response to the RPII. 612 people replied agreeing to participate in the project (a response rate of 73%). This response rate was unexpectedly high. Experience in the UK of response rates to radon surveys has been in the range of 20 – 40%, depending on the areas being targeted (Kendall *et al.,* 2005). The high response rate for this research could be due to how easy it was to respond to the letter.

For both phases of the project, measurement packs were sent to all volunteers. Each measurement pack contained:

- Personalised letter: thanking for participation and setting out instructions for detector placement and return
- Two short-term detectors (labelled: Bedroom 1M; Living area 1M)
- Two long-term detectors (labelled: Bedroom 3M; Living area 3M)
- 1 month record sheet: to record the dates the detectors were placed and removed for the 1 month period
- 3 month record sheet: to record the dates the detectors were placed and removed for the 3 month period
- 2 freepost return envelopes

After the one and three month periods had elapsed, each participant was reminded to return the two relevantly labelled detectors and record sheet. This was done firstly by a phone call; if there was no answer, they were sent either an email or letter instead.

The RPII offers a commercial radon measurement service using alpha track detectors comprising a two part polypropylene holder and a CR-39 (poly allyl diglycol carbonate) detection plastic. Details of detector processing are given in Fennell *et al.,* 2002. The standard radon measurement service offered by the RPII (3 month radon measurement) is subject to stringent quality control procedures and is accredited to ISO 17025 by the Irish National Accreditation Board. Although the one month measurements would not be accredited, they were subjected to the same stringent quality control procedures during this project.

Results

As already described the measurement procedure provided four radon concentration results for each household: living area 1 month, bedroom 1 month, living area 3 month and bedroom 3 month. Mean values for each household were calculated for the one month results and the three month results.

To ensure a high level of confidence in the results, both phases of the study covered all counties of Ireland and contained both high and low radon concentrations. Although the measurement regime was designed to ensure that detectors were in place for set periods of one month and three months, detectors were returned from some households with longer measurement periods. It was decided that observations such as these could mislead the estimation process. In addition for some households useable measurements were not obtained for all four detectors. After eliminating those households with measurement periods greater than one and three months and those with fewer than four results, results from a total of 622 homes were available for analysis. The data consisted of 93 households in Phase 1 and 529 households in Phase 2.

As mentioned the analysis of the Phase 1 data proved to be inconclusive so we will report on two data sets. One formed with all data from Phase 1 and Phase 2. The second formed from merging data from Phase 1 and Phase 2 but with what were deemed to be extreme values that is >9000 Bq/m^3, removed.

A linear regression model was fitted to the data to determine if a useful relationship could be established between one month and three month measurements. It is known that radon concentrations can be modeled well by the lognormal distribution if they have been corrected by subtracting a background average value of outdoor radon concentration and that predictions using this assumption are more accurate than models based on untransformed measurements (Gunby *et al.,* 1993, Fennell *et al.,* 2002 and Murphy and Organo, 2008). Fennell *et al.* (2002) reported 6 Bq/m^3 as being representative of the average outdoor radon concentration in Ireland. Consequently all subsequent modelling was performed on radon levels in which the background level of 6 Bq/m^3 was firstly subtracted from the radon levels before the levels were log-transformed to provide a better fit to a normal distribution.

1.1 Regression analysis

1.1.1 All data from both phases of the project

The merged data set comprised all of the data from phase 1 and phase 2 of the project 622 observations, 529 from phase 2 and 93 from phase 1. As was previously described the measurements in phase 2 were deliberately chosen from higher radon areas. Figure 3.1 plots the background corrected log-transformed results from the one month radon measurements against the log-transformed three month radon measurements for this merged data set. As can be seen in this plot, where the phase 1 measurements are clustered on the bottom left corner of the graph, there was an increased variance among the one month measurements in phase 1 compared to phase 2. However, even with this added variability it is clear from Figure 3.1 that there is still a strong linear association between the one month measurements and the corresponding three month measurements. Indeed the correlation coefficient for this merged data set is r = 0.935.

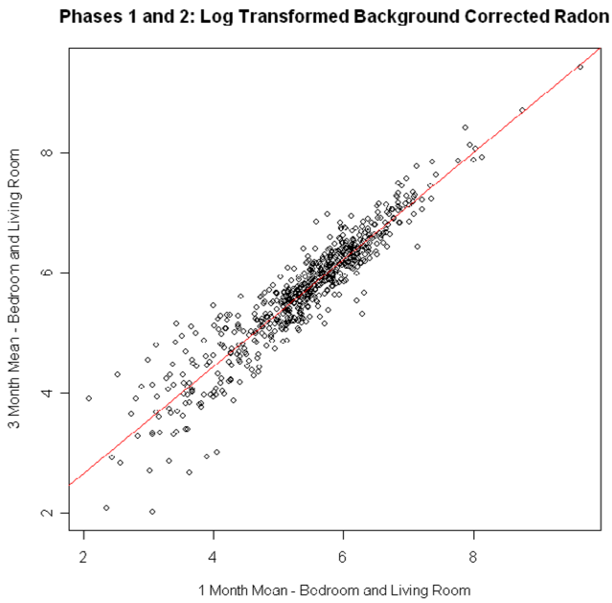


Figure 3.1. Log-transformed household mean 1 month radon concentration vs. log-transformed household mean 3 month radon concentration [all data from both phases of the project].

Table 3.1 describes the fit of the linear regression model that was fitted to this data set. The ANOVA F test P-value which is less than 2.2 x 10^{-16} and the t-tests of the significance of the regression parameters (P-value < 2.0 x 10^{-16}) all indicate that the simple linear regression model is a good fit to the data. The coefficient of determination is R^2 = 87.41% indicating that approximately 87% of the variation in the one month measurements can be explained by the linear association between the one month measurements and the three month measurements.

Table 3.1. Simple linear regression analysis of log-transformed household mean 1 month radon concentration vs. log-transformed household mean 3 month radon concentration [all data from both phases of the project].

	Parameter Estimate	Standard Error	T-statistic	P-Value
Intercept	0.881	0.074	11.86	<2.00x 10^{-16}
Slope	0.891	0.014	65.62	<2.00x 10^{-16}
Source	DF	Sum Sq	Mean Sq	4306.3 <2.2x 10^{-16}
Log 3 month	1	523.62	523.62	
Error	620	75.39	0.12	
Total	621	599.01		
	R-Sq = 87.41%	R-Sq(adj) = 87.39%		

Figure 3.1a contains a Histogram and a QQ plot of the residuals from these plots. While the histogram shows that the residuals are symmetrically distributed, the QQ-plot displays a certain deviation from normality at the two ends of the plot indicating that the regression residuals are somewhat more dispersed than would be expected from a normal distribution. Having said this, a Kolmogorov-Smirnov test of normality on these

residuals gave a p-value of 0.085 thus failing to reject the null hypothesis of normality for the residuals at significance level 0.05.

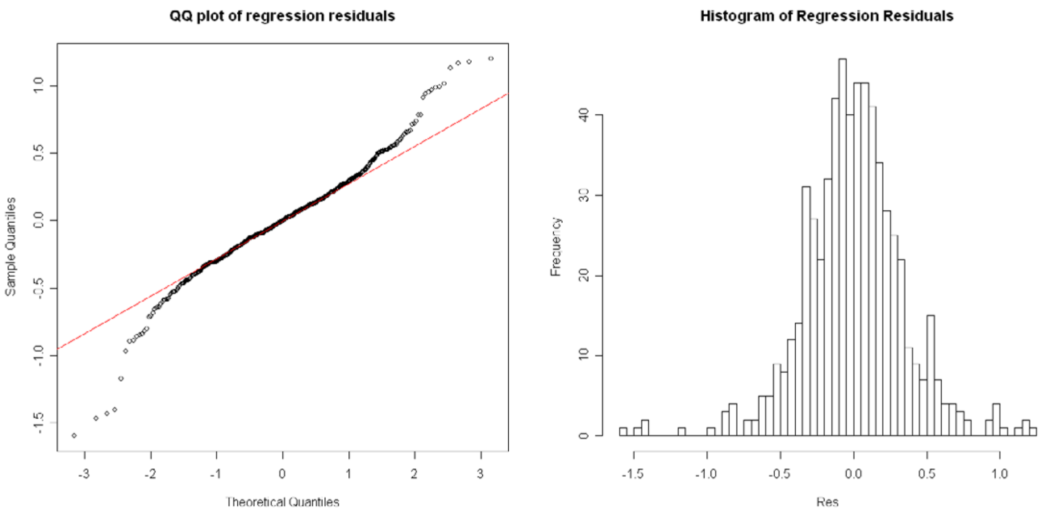


Figure 3.1a. Residuals from the Log-transformed household mean 1 month radon concentration vs. log-transformed household mean 3 month radon concentration [all data from phase 1 and 2 of the project.

1.1.2 Data from both phases of the project with two extreme results (greater than 9,000 Bq/m^3) removed

Two very extreme results (>9,000 Bq/m^3) were removed and the merged data set reanalysed to examine whether these two results had acted as influential observations. Figure plots the background corrected log-transformed results from the one month radon measurements against the background corrected log-transformed three month radon measurements. The correlation coefficient was computed to be 0.932.

Table 3.2 describes the fit of the linear regression model that was fitted to this data set. The ANOVA F test P-value which is less than 2.2 x 10 $^{-16}$ and the t-tests of the significance of the regression parameters (P-value < 2.0 x 10 $^{-16}$) all indicate that the simple linear regression model is a good fit to the data. The coefficient of determination is R^2 = 86.9% indicating that approximately 87% of the variation in the one month measurements can be explained by the linear association between the one month measurements and the three month measurements.

The parameter estimates changed slightly when the two extreme observations were removed indicating that these observations did not unduly influence the model fit.

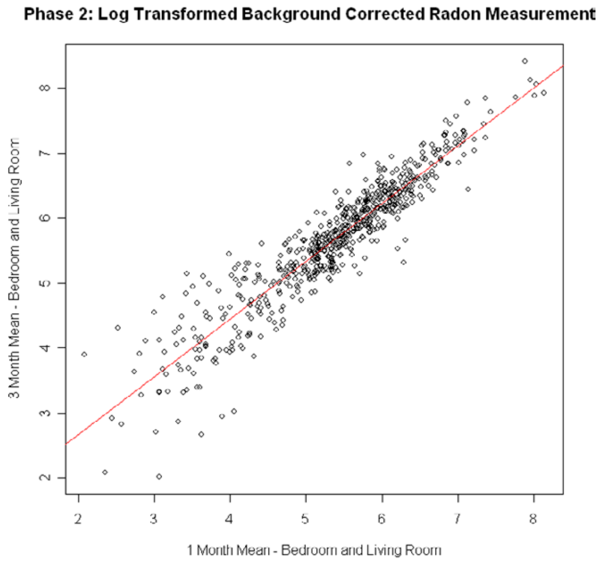
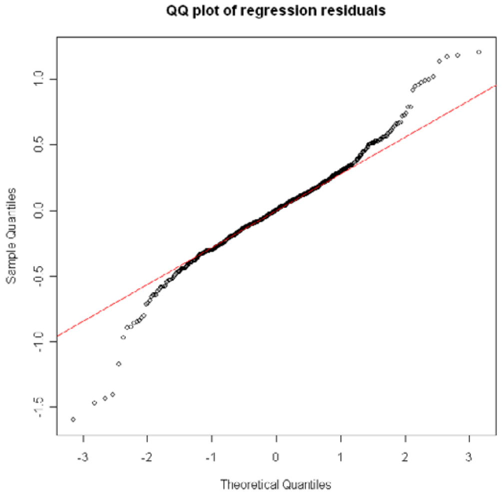


Figure 3.2. Log-transformed household mean 1 month radon concentration vs. log-transformed household mean 3 month radon concentration [data from both phases of the project with two extreme results (> 9,000 Bq/m^3) removed].

Table 3.2. Simple linear regression analysis of log-transformed household mean 1 month radon concentration vs. log-transformed household mean 3 month radon concentration [data from both phases of the project with two extreme outliers removed].

	Parameter Estimate	Standard Error	T-statistic	P-Value
Intercept	0.880	0.076	11.59	<2.00x 10^{-16}
Slope	0.891	0.014	64.04	<2.00x 10^{-16}
	1.10	0.00502	(1.08, 1.13)	1.10
Source	DF	Sum Sq	Mean Sq	F-statistic P-value
Log 3 month	1	500.25	500.25	4101 <2.2x 10^{-16}
Error	618	75.39	0.12	
Total	619	575.64		
	R-Sq = 86.9%	R-Sq(adj) = 86.88%		

Figure 3.2a contains a Histogram and a QQ plot of the residuals from these plots. The removal of two observations would not greatly influence these residuals and indeed the histogram and the QQ-plot are almost identical to those found on the full merged data set again indicating that the regression residuals are somewhat more dispersed than would be expected from a normal distribution. But once again, a Kolmogorov-Smirnov test of normality on these residuals gave a p-value of 0.087 thus failing to reject the null hypothesis of normality for the residuals at significance level 0.05.



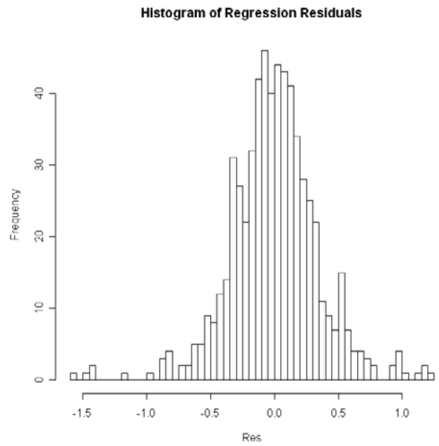


Figure 3.2a. Residuals from the Log-transformed household mean 1 month radon concentration vs. log-transformed household mean 3 month radon concentration [Data from phase 1 and 2 of the project with two extreme observations removed].

1.2 Setting upper and lower threshold levels

The regression models which were fit to the data sets enable us to predict a three month radon concentration on the basis of a one month measurement. We have seen that the regression models fit well so these predictions can be viewed as reliable. The predictions are, however, just point estimates and as such are subject to uncertainty. We can quantify this uncertainty by producing confidence intervals rather than point estimates. The aim of this modelling is to establish two separate bounds on the one month measurement. The first is a lower threshold such that any one month measurement that is below this threshold can be confidently viewed as indicating that a three month measurement taken on the same house would be lower than the 200 Bq/m^3 Irish National Reference Level. The second threshold that must be computed is an upper threshold such that a one month measurement exceeding this threshold would produce a three month measurement exceeding 200 Bq/m^3. The lower and upper thresholds can be computed from the regression model using standard 95% one sided prediction intervals for the background corrected log transformed radon levels which are exponentiated and have had the background of 6 Bq/m^3 added back. Hence to find the threshold levels we solve the following inequalities for the upper threshold (X_U) and the lower threshold (X_L).

$$200 < 6 + \exp\left(\hat{\beta}_0 + \hat{\beta}_1 \log(X_U - 6) - t_{95\%} s \sqrt{1 + \frac{1}{n} + \frac{(\log(X_U - 6) - \overline{X})^2}{SS_{XX}}} \right)$$

$$200 > 6 + \exp\left(\hat{\beta}_0 + \hat{\beta}_1 \log(X_L - 6) + t_{95\%} s \sqrt{1 + \frac{1}{n} + \frac{(\log(X_L - 6) - \overline{X})^2}{SS_{XX}}} \right)$$

where X_T are the upper or lower threshold levels, $\hat{\beta}_0$ and $\hat{\beta}_1$ are the estimates of the intercept and slope from the regression, s is the estimate of the standard deviation of the error terms in the regression model, n is the sample size, $t_{95\%}$ is a one sided 95% critical value of the t distribution $\overline{X}$ is the mean of the background corrected one month measurements and $SS_{xx} = \sum (x - \bar{x})^2 = s_x^2(n-1)$ is the sum of squares of the background corrected one month measurements.

Table 3.3 summarises the lower and upper thresholds (Bq/m^3) of the equivocal result region for the two the datasets described above. Outside of these ranges, one month radon concentrations derived from the datasets can be regarded as definitive indicators of mean annual levels below or above the Irish National Reference Level of 200 Bq/m^3 with 95% confidence. Results falling between these thresholds are regarded as equivocal, and would require a measurement for a three month period in order to estimate a more accurate mean annual level.

Table 3.3 Upper and lower threshold levels (95% CI) for one month radon measurements using alpha track detectors.

Dataset used for analysis	95% Confidence level	
	Lower threshold (Bq/m^3)	Upper threshold (Bq/m^3)
All data from both phases of the project	89	234
Data from both phases with 2 extreme results (>9,000 Bq/m^3) removed	89	235

Discussion

Many international studies have investigated the possibility of reducing the standard radon measurement period with alpha track detectors from three months to a one month period or less (Quindos *et al.,* 1992, Bowring, 1992, Klotz *et al.*, 1993, Chen, 2003, Groves-Kirkby *et al.,* 2006, Ruano-Ravina *et al.,* 2008 and Ibrahimi and Miles, 2009). However, none of these studies were based in Ireland and so cannot be assumed as directly applicable. Climatic conditions, as well as seasonal and dwelling-occupancy factors influence the amount of radon that may emanate from the ground and enter buildings (Sundal *et al.,* 2007). These reasons pointed to the need for an indigenous study covering the whole of the country of Ireland.

Groves-Kirkby *et al.* (2006) found threshold values of 109 - 478 Bq/m^3 for one month alpha track detectors. However, the study had only 18 data-points for the one month measurements and the authors warned that the one and three month results in their study should be treated with caution due to the low number of data-points (Groves-Kirkby *et al.,* 2006). Ibrahimi and Miles (2009) investigated the accuracy of two-week radon measurements using alpha track detectors and found threshold levels of 170 - 300 Bq/m^3. However, their results were based on measurements made in 59 homes in the UK. Although the research suggest a tighter range of threshold levels than what is currently in use for the screening detector service in the UK (current threshold levels are 100 - 400 Bq/m^3), (Ibrahimi and Miles, 2009). In order to ensure a high level of confidence in the results of this study, it covered all counties of Ireland and had a sufficient spread of both high and low radon concentrations. A total of 622 short and long term measurements were available for analysis.

Conclusions

At time of writing this report a formal recommendation has yet to be made on how this study will be used to support the RPII's aim to link radon to the conveyancing of properties in Ireland (RPII, 2006). While the results summarised in table 3.3 are encouraging further work is needed. Before being in a position to recommend a threshold range the RPII will take account of new seasonal correction factors for Ireland. (Burke *et al.,* in press). In addition, the RPII are likely to recommend conservative round numbers for the threshold band.

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Radon hazard evaluation based on measurements of indoor air and gamma measurement surveys in combination with bedrock and drift geology mapping

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Abstract

During the last 10 – 15 years, The Norwegian Radiation Protection Authority has carried out several surveys of radon in indoor air. These data have been useful for the overall evaluation of radon risks at the local level, and they have been used both for estimations of average radon exposure and, to a certain extent, to identify radon-prone areas. According to the new radon strategy (2009) in Norway, the local authorities will need to consider the radon hazard in connection with area planning, and if necessary impose restrictions or local regulations regarding new building constructions in very high risk areas. The Geological Survey of Norway and the Norwegian Radiation Protection Authority have established a combined approach to mapping of radon in Norway, based on direct measurements of radon in indoor air, bedrock and drift geology, and the mapping of uranium/radium in the ground using airborne gamma-measurement surveys. This work resulted in the release of overview radon awareness maps covering the most densely populated areas in Norway around the capital city Oslo. The performance of the hazard evaluations was tested in an area centered around the municipality of Gran, and was shown to be able to enclose most of the high risk areas. Areas already known to be associated with elevated radon hazards were confirmed, and additional potential high risk areas not yet confirmed by indoor radon measurements were identified. The Norwegian Geological Surveys have recently carried out new air-borne gamma measurement surveys in areas south-west of Oslo. Corresponding datasets on indoor radon are available, and includes areas known to be strongly affected by radon. A test of the combined approach in this area will be independent of the earlier studies, and will be used for additional testing and refinement of the method that has been established. The results of this work so far will be presented.

Indoor radon and construction practices in Finnish homes from the 20th to the 21st century

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Abstract

Indoor radon is the second leading cause of lung cancer in Europe. In order to reduce the exposure to radon, it is necessary to understand the structural features of buildings that affect radon concentrations. These features vary with the year of completion of the building.

A population-based random sample study was performed in the homes of 2,882 people. Radon measurements were made using alpha track detectors. Information on the structure of the dwellings was collected using a questionnaire filled in by the residents.

The weighted national mean of indoor radon concentration in single-family and row houses was 121 Bq/m^3 and in blocks of flats 49 Bq/m^3. The national mean was 96 Bq/m^3.

According to our study, the radon concentrations in new single-family and row houses began to rise after the 1960s, and continued to rise until steadying during the 1980s and 1990s, decreasing thereafter to the level of the 1970s. These changes can be explained by changes in building methods.

Before 1965, the prevailing house type was one with a basement with no direct entrance into the living area. This solution reduced the radon entry into the living area. Thereafter, a slab-on-ground foundation, which promotes radon-bearing air entering the house, became the most common foundation. A more radon-resistant foundation, crawl space was popular during 1950s. Its popularity decreased thereafter and increased again after 2000.

Since the year 2000, mechanical supply and exhaust ventilation has been installed in most new dwellings. This reduces the entry of radon due to smaller pressure differences. In addition, the use of radon prevention techniques in new building has become more common because of new building codes.

The changes in the Finnish construction practice from the 20th to the 21st century have had clear consequences, which can be seen in indoor radon concentrations. After a slow awakening to radon awareness in the 1980s, research, guidance and legislation have resulted in the current downward trend in indoor radon concentrations.

Introduction

Indoor radon exposure increases the risk of lung cancer in a residential environment (Darby et al. 2005). The most important reason for high radon concentrations in dwellings is the radon-bearing soil air that flows into the building. Gaps between the slab and foundation walls, permeable materials used in foundations and a high underpressure increase radon concentration. Indoor radon exposure can be reduced by identifying and mitigating dwellings with high concentrations, and by radon prevention techniques in new building.

Regulations concerning indoor radon for the health protection and building control authorities in Finland are working reasonably well and are on a good level on the world scale. According to Decree 944/92 by the Ministry of Social Affairs and Health, the indoor radon concentration should not exceed 400 Bq/m^3, whereas new buildings should be built so that 200 Bq/m^3 is not exceeded (Ministry of Social Affairs and Health, 1992). In addition, Part D2 of the National Building Code of Finland (Indoor climate and ventilation of buildings) sets 200 Bq/m^3 as a reference value for the planning of new buildings (Ministry of Environment, 2004). Since 2004, according to Regulation B3, radon should be taken into account in new building throughout the country (Ministry of Environment, 2004). A radon technical plan should be included in the building licence application. This plan can only be exceptionally ignored when building in areas where exceeding 200 Bq/m^3 seldom occurs.

Material and methods

The results of this study are based on the random sample study by STUK, in which 6,000 Finns were selected using simple random sampling from the Finnish population registry (Mäkeläinen et al. 2009).

The radon measurements were made using alpha track detectors in two half-year periods, the first from April to November 2006 and the second from November 2006 to April 2007. Information on dwelling construction was collected using questionnaires filled in by the inhabitants. Fifty-seven per cent of the selected population accepted the call to participate, and 48 per cent completed a valid measurement; 58 per cent (2,267) of the residents living in houses (single-family and row houses) and 31 per cent (599) of those living in blocks of flats completed valid measurements.

The annual mean of radon concentration for each dwelling was calculated as an arithmetic mean weighted by the lengths of the two measurement periods. The final national statistics were calculated by weighting the annual means using the number of inhabitants of houses and flats in each province.

Results

Annual mean of radon concentration

The weighted average for annual means of radon concentration was 96 Bq/m^3 for all dwellings, 121 Bq/m^3 for houses and 49 Bq/m^3 for flats. The medians were 56 Bq/m^3, 75 Bq/m^3 and 36 Bq/m^3 respectively. Radon concentration notably varies regionally. Häme and South-Eastern Finland (the provinces of Itä-Uusimaa, Päijät-Häme, Kymenlaakso, Kanta-Häme, Pirkanmaa and South Karelia) constitute a continuous high radon area in which a quarter of all Finnish house inhabitants reside. In this area the mean radon concentration in houses was 201 Bq/m^3 and in the rest of Finland 96 Bq/m^3.

Effect of the year of completion of building on the radon concentration

Radon concentrations in new buildings are connected to the construction practices of their time. They were at their lowest in houses built during the 1950s (Fig.1). During the 1960s and 70s the radon concentrations started to rise, being almost double at the beginning of the 1980s compared with houses built during the late 50s. Thereafter, the radon concentrations in new houses have decreased, and since the year 2000 the radon concentrations in new houses has been at the same level as in houses built in the 60s and 70s.

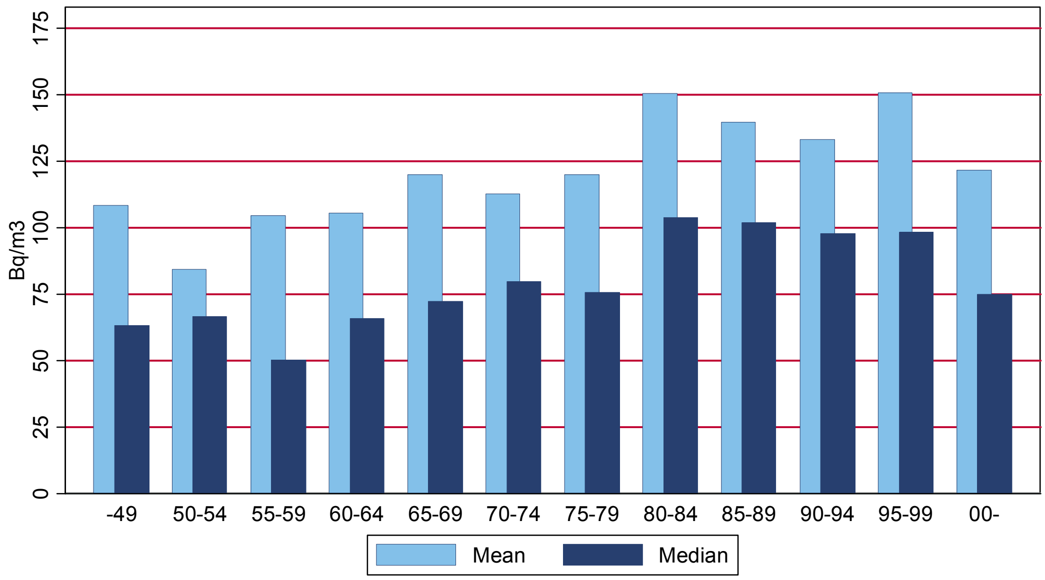

Fig. 1. Means and medians of annual average radon concentration in houses according to the year of completion of the building.

Effect of foundation type on radon concentration

Houses were divided into three types according to the ground contact of the foundations. Both hillside houses and houses with a cellar have basement walls backing onto the ground. In a hillside house, one wall is on ground level and the wall facing it is at least partly backing onto the ground. In houses with a cellar, all walls are ground-contact walls. The third type is a house without a basement that mostly has a slab-on-ground or crawl space foundation. The inhabitants selected the figure most like their house type in the questionnaire.

Until the beginning of the 1960s the most common type was a house with a cellar (Fig. 2). Thereafter, houses without a basement became most common. Since the 70s, a hillside house has been more common than a traditional house with a cellar.

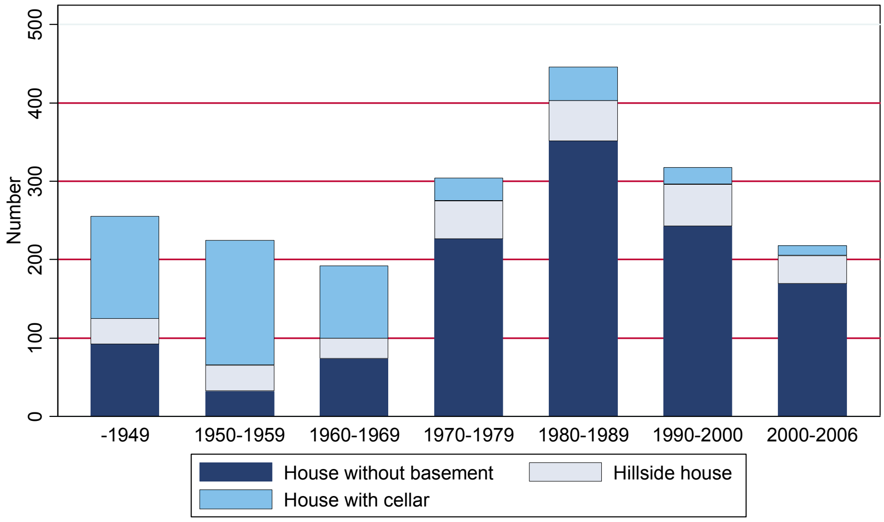

Fig. 2. Numbers of house types in the study data according to the year of completion of building.

The highest radon concentrations are in houses with a basement (hillside houses and houses with a cellar) built in the 1990s and later, having medians of 156 and 152 Bq/m^3 (Table 1). Concerning houses completed before the 1990s, hillside houses have higher concentrations than other types (median 92 Bq/m^3), but houses with a cellar are the lowest (median 66 Bq/m^3). On the other hand, radon concentrations in houses without a basement are almost at the same level regardless of building year (medians 77 and 78 Bq/m^3).

Table 1. Type of house and radon concentration.

House type	All houses*			Built before 1990			Built in 1990-2006		
	Count	Mean Bq/m^3	Median Bq/m^3	Count	Mean Bq/m^3	Median Bq/m^3	Count	Mean Bq/m^3	Median Bq/m^3
House without a basement	1253	119	76	775	118	77	412	117	78
Hillside house	288	156	102	193	143	92	89	191	156
House with a cellar	519	117	66	454	111	66	35	183	152

*Including those with building year missing

In houses with a basement, walls backing onto the ground increase radon leaks into the dwelling without proper sealing. If the staircase to the lowest floor is open, radon-bearing air from downstairs mixes with the air in the rest of the dwelling. On the other hand, if the entrance to the basement is normally closed or the entrance to the lowest floor is only from the outside, radon-bearing air cannot so easily enter the upper floors. This is the case in the older houses with a basement, where only 22 per cent had an open staircase (Table 2). Another factor increasing radon concentrations in new houses with a basement is probably the higher proportion of light-weight concrete blocks used as ground-contact wall material: 71 per cent in the newer houses compared with 16 per cent in the older ones with a basement. In the new houses, the basement floor is also more often used as a living area: 57 per cent in the newer houses and 26 per cent in the older ones. The residents were asked to carry out the measurement on the lowest floor with a living area as the concentration there is usually the highest. This also results in higher concentrations in the new houses.

Table 2. Houses with a basement: Radon and entrance type from the upper floors to the basement level.

Entrance to the lowest floor	All houses with a basement*			Built before 1990			Built in 1990-2006		
	Number	%	Median Bq/m^3	Number	%	Median Bq/m^3	Number	%	Median Bq/m^3
Open staircase	200	28	117	127	22	103	68	64	163
Staircase and door	390	55	65	350	61	65	19	18	179
Only from outside	119	17	78	95	17	76	20	19	117
All	709	100	78	572	100	71	107	100	157

House without basement and radon

A house without a basement has been the most common foundation type in houses since the 1960s. The two most used foundation types in this house type are slab-on-ground and crawl space (Fig. 3). When slab-on-ground laid between the foundation walls is used, radon-bearing soil air gets in through the gap between the foundation wall and the slab unless the gap and feed-throughs are sealed. The crawl space foundation is fairly radon-resistant if radon leaks are stopped by sealing the foundation joints and feed-throughs, and ventilation of the base floor is arranged.

Crawl space as a foundation type was very common before 1960 (Fig. 3). Its popularity decreased thereafter, but, according our data, since the year 2000 it has been getting more popular than ever since the 1950s. Whereas the percentage of crawl space houses was only 4 per cent during the 1970s, it has risen to approx. 20 per cent in houses built after 2000.

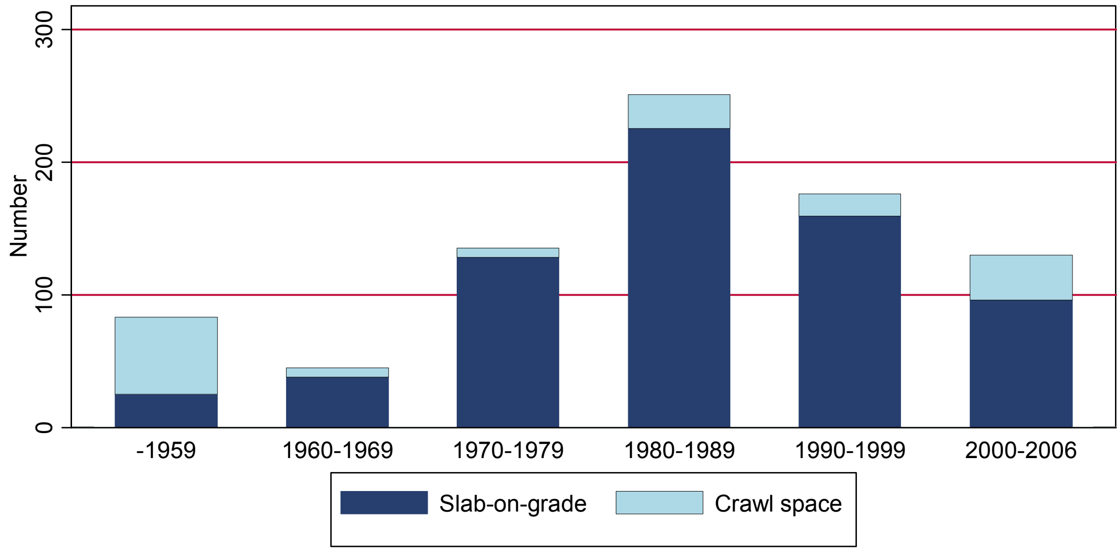

Fig 3. Numbers of slab-on-ground and crawl space foundations in houses according to the year of completion of building.

Differences in radon concentrations between these types are clearly seen (Table 3). Radon concentrations are higher in houses with slab-on-ground foundations built both before and after 1990.

Table 3. Foundation types slab-on-ground and crawl space in houses without a basement.

Foundation type	All houses with a basement			Built before 1990			Built in 1990-2006		
	Count	Mean Bq/m^3	Median Bq/m^3	Count	Mean Bq/m^3	Median Bq/m^3	Count	Mean Bq/m^3	Median Bq/m^3
Slab-on-ground laid between foundation walls	688	132	94	415	134	96	251	130	93
Crawl space	150	61	44	96	59	48	49	60	38

Cast concrete was a clearly more radon-resistant option (medians 75 Bq/m^3 and 58 Bq/m^3) than light-weight concrete blocks (medians 136 Bq/m^3 and 106 Bq/m^3) in both older and newer houses. Light-weight concrete blocks are very permeable, and when used as building material for foundation walls, radon-bearing soil air can easily flow through them from the soil into the dwelling. In houses with slab-on-ground foundations built before 1990, 31 per cent had light-weight concrete blocks used in the foundation walls, but for those built thereafter, the percentage was already 66 (Table 4). Radon concentrations were higher in the older than in the newer houses regardless of the material of the foundation walls. However, because the proportion of foundation walls made of light-weight concrete blocks is smaller in the older houses than in the new ones, the radon concentration in all the houses with slab-on-ground has not changed much (medians 96 Bq/m^3 and 93 Bq/m^3, Table 3).

Table 4. Material of foundation walls and radon in houses with slab-on-ground foundations.

Material of foundation walls	Built before 1990			Built in 1990-2006		
	Number, proportion (%)	Mean Bq/m3	Median Bq/m3	Number, proportion (%)	Mean Bq/m3	Median Bq/m3
Cast concrete	237 (69%)	114	75	74 (34%)	91	58
Light-weight concrete blocks	105 (31%)	158	136	146 (66%)	150	106

Ventilation type and radon

Ventilation types were compared in houses without a basement with slab-on-ground foundations because the group is large enough and does not have temporal confounding factors, as was seen in the last chapter. The data shows that mechanical exhaust ventilation is the most disadvantageous ventilation type with regard to radon (Table 5). This ventilation type increases the underpressure of the dwelling and radon leaks from the soil. The situation in the houses with natural ventilation is only slightly better. The mechanical supply and exhaust ventilation is best with regard to radon. This was as expected: a well-adjusted supply and exhaust system ensures a sufficient air exchange rate, and reduces the underpressure and, consequently, radon leaks into the building.

Mechanical supply and exhaust ventilation has become more popular in new houses. In single-family houses, the change to this system started in the 1980s. In row houses the change has been slower. These changes are one cause of the reduction in radon concentrations observed since 1990 (Table 4).

Table 5. Ventilation type and radon concentration in houses without a basement with slab-on-ground laid between foundation walls.

	All houses			Built before 1990			Built in 1990-2006		
Ventilation type	Count	Mean Bq/m^3	Median Bq/m^3	Count	Mean Bq/m^3	Median Bq/m^3	Count	Mean Bq/m^3	Median Bq/m^3
Natural	284	143	97	244	140	98	28	183	104
Mechanical exhaust	141	148	115	68	136	104	69	163	137
Mechanical supply and exhaust	230	110	76	78	118	81	147	104	75

Radon prevention in new building

Guidance on radon prevention in new building has been available since 1994 (Ministry of Environment, 1994). This first guide was replaced by the RT Building Information File published in 2003 (Building Information Ltd., 2003). The guides recommend sealing the joints between the foundation wall and floor slab and walls backing onto the soil with bitumen felt, sealing the feed-throughs, and installing radon piping as a preparatory measure. The goal is a low radon concentration using only passive sealing. If, however, the radon concentration in the house is measured as more than 200 Bq/m^3, activation of the radon piping by installing a fan on the roof is needed. Activating the fan sucks radon-bearing air from the soil under the slab.

Radon piping has been installed in 64 per cent of houses with slab-on-ground foundations built after 1995 in the high radon areas of Häme and South-Eastern Finland. The percentage of similar houses in the rest of Finland was 14. Sealing of the foundation constructions was not so common: 18 per cent in the high radon area and 14 per cent in the rest of Finland. Both installing radon piping and sealing of foundation constructions were performed in the high radon area and the rest of Finland in 11 and 4 per cent of dwellings respectively.

Radon prevention is clearly more common in the high radon area. The prevention measures showed some reduction in radon concentration, especially in the high radon area, but the numbers of dwellings were too small for conclusions.

Conclusions

Radon concentrations in new houses have increased from the 1950s to the 1980s, and have decreased thereafter, especially since the year 2000. The recent decrease is caused by several reasons, one being the growing percentage of houses with crawl space foundations. Crawl space is a radon-resistant construction if the flow of radon-bearing air into the dwelling is stopped by sealing the base floor and feed-throughs, and by taking care of the ventilation of the crawl space.

Mechanical supply and exhaust ventilation has become the most common ventilation system in new houses, and this has also reduced radon concentrations. The third cause of the reduction in radon concentrations in new dwellings is the use of radon prevention in new building, which has become more popular, especially in high radon areas. The installation of radon piping systems is common in high radon areas; sealing of ground-contact constructions has not been used as often as was expected.

Although radon concentrations in new buildings have decreased, there are also factors that are against the downward trend. Houses with a basement have high radon concentrations. In both hillside houses and houses with a cellar, the proportion of dwellings having living areas on the basement floor has increased, and the entrance from the upper floor to the basement floor is more and more often an open staircase. In addition, the growing use of permeable light-weight concrete blocks as a building material for foundation and basement walls instead of cast concrete has increased radon concentrations.

Part 3 of the National Building code of Finland, which concerns the foundations, came into force in the year 2004. The code requires radon prevention in all buildings. The radon concentration in the completed building should be below 200 Bq/m^3. When the provisions are widely implemented, the reduction in radon concentrations in new buildings is expected to continue.

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Geochemical case study of sediment samples from Olkiluoto, Finland – assessment of radon emission

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Abstract

In order to define the naturally-occurring radioactive materials that are source of radon in natural environments, a comprehensive analytical (geochemical, physical and chemical) methodology was employed to study sediment samples from overburden glacial formation in Olkiluoto (W Finland). Techniques such as gamma-spectrometry, emanation measurements, sequential chemical extraction, scanning electron microscopy (SEM), electron microprobe analyses (EMPA) and inductively-coupled plasma mass spectrometry (ICP-MS) were used to determine the potential source of radon. Samples are representing a typical Finnish red-ox profile formed on glacial sediments deposited around 9,000 years ago. In Olkiluoto samples activity concentration of ^{238}U, ^{226}Ra and ^{228}Ra show no significant trend in a function of depth secular equilibrium between ^{238}U and ^{226}Ra occurred only in the samples collected from the reductive zone. In the oxidized horizon higher radium than uranium activity concentration was measured. In this horizon the $^{226}Ra/^{238}U$ is 1.4. In Th decay chain secular equilibrium occurred between ^{232}Th and ^{228}Ra. Radon productions and emanation factors are very low, varying between 0.01 – 0.03 Bq/(kg h) and 0.04 – 0.08, respectively. The uppermost sample collected from the oxidized horizon has the highest radon production and emanation factor. This sample has the highest ^{226}Ra content (52 Bq/kg), as well. The sequential extraction revealed that in the oxidized zone more exchangeable Ra is present than in the reductive zone. In general, more radium was leached (ca. 20% of the total) easily than uranium and thorium, except in the deepest sample, where 30% of the total uranium was leached easily. Monazite, xenotime, zircons, apatite and U-Th-silicates were identified in the samples as the main sources of uranium, thorium and radium. These minerals were partly weathered in the uppermost horizons and mainly fresh in the deeper ones.

The radon situation in Swedish dwellings based on recent measurements

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Abstract

Gammadata has been measuring radon levels in Sweden and abroad since the late 1980´s and is one of the largest suppliers of CR-39 based radon detectors in Europe. Our extensive database includes several hundreds of thousands of measurements. This makes it possible for us to derive several conclusions of the radon situation in Sweden. We present radon levels measured in single family dwellings and their dependence on several factors such as the type of ventilation and foundation of the measured dwellings.

Introduction

Due to the relatively large problem with high radon levels in dwellings in Sweden an examination of which factors increase or decrease the risk for radon levels above the action level is of interest. In the past years radon measurements in dwellings have been available for the general public through offers sent out to all house owners within several municipalities making the selection of measurement locations much more random than previously. This increases the possibility of using data from a measurement database to draw general conclusions concerning radon levels.

We have selected to examine single family dwellings comparing ventilation, foundation, and construction year. The data used is based on the information provided by the person performing the measurement. For single family dwellings we assume that the majority of the information concerning the building is correct.

Since a large portion of the measurements in our database are incomplete concerning information about the building certain combinations of construction year and foundation and ventilation yield very few data. Those results containing fewer than 30 measurement points have been removed from this examination in order to reduce the risk of single measurements of extreme values having a large effect on the result.

In Sweden we have three sources of radon in dwellings, the ground, the building material and the water. Water is the least common source and hence we disregard its effect on the radon level indoors in our analysis. The building material that can provide radon levels above the action level is a light concrete which is based on alum-shale. This material was widely used in the 50´s through the 70´s. In the 50´s and 60´s as much as half the buildings constructed contained some amount of alum-shale concrete (G.A. Swedjemark), The production of this material was stopped 1975 but houses built in the years following can also contain it. Since this material is common only in Sweden we have based our analysis solely on those measurements where there is a notation that the dwelling does not contain alum-shale.

Construction year of the building

Examining the radon levels in single family dwellings can be used as a general quality measurement of the different house constructions in different eras. To examine this we divide into 10 year periods thereby assuming no major reformations in house constructions were made during each time period. A clear relationship between the year a building is constructed and the radon level is evident, see table 1:

Table 1. radon levels in private dwellings depending on year of construction.

Year of construction	Number measured	Average radon level	Highest measured level	Percentage above 200 Bq/m^3
1800-1899	4263	149	6759	16,6
1900-1919	8235	150	>12200	17,7
1920-1929	7826	154	6510	19,5
1930-1939	12949	142	8410	18,5
1940-1949	15174	157	8240	23,5
1950-1959	24866	177	8910	30,0
1960-1969	43229	197	16220	34,5
1970-1979	45412	172	>24500	28,3
1980-1989	21426	111	17050	13,8
1990-1999	10963	71	10960	6,5
2000-2008	12132	59	5050	4,7

In table 1 it is clear that the period ranging from 1950-1979 an increase in the radon level measured is noted compared to previous years. This time period also coincides with an increase in building rate of houses.

Foundation and ventilation

The question arises whether certain combinations of constructions produced in certain time periods are more likely to experience radon problems than others. The house constructions selected for examination includes both the foundation of the house as well as the ventilation type. The four most common foundation types in Sweden are as follows; basement, crawlspace, slab, and split level house, in no particular order. The selected ventilation types are as follows; natural ventilation, mechanical exhaust, and mechanical supply and exhaust air ventilation (most often used together with heat recovery).

In general we find that houses containing basements or split levels in combination with natural ventilation are responsible for the highest radon levels.

Looking at natural ventilation and the radon level dependence for different years and different house foundations it is clear that split level dwellings tend to have the highest radon levels, see figure 1. Until the 1970´s the basement was constructed to contain garage, storage areas, laundry rooms, furnaces etc and were therefore not used as living space. In an annual average calculation only living areas are included. Therefore, in dwellings with the original construction high levels of radon can be present in the basement which are ventilated away before reaching the living space upstairs as it is normally separated from the basement by a door. This is most likely the reason why houses containing basements tend to have lower levels of radon compared to split-level houses where living areas are placed partially underground (C. Boox)

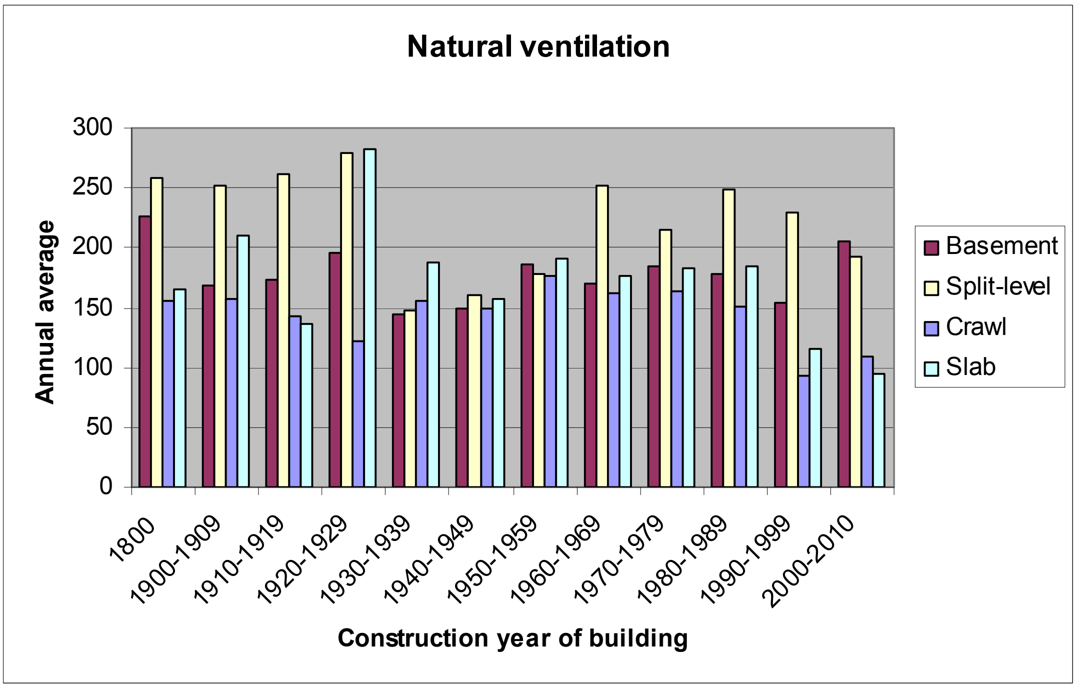

Fig. 1. Annual average radon levels measured in dwellings with natural ventilation which are constructed in the eighteen hundreds until present day and the dependence on house foundation.

For dwellings up until the past 10 years, crawl space tends to have lower levels then the rest. An interesting feature is that slabs as well as crawlspace constructions manufactured in the past 20 years show a marked improvement of radon levels as compared to older constructions and when compared to basements and split-levels of all years. It is disappointing that the same level of improvement is not seen in modern basements and split-levels. Another interesting feature is the increase in radon levels seen in slab constructions of the 1920´s. This may in part be due to the concrete used at this time period, a lack of funds after the first world war tended to increase the dilution of concrete with fillings such as rock which in effect made the material more porous. Also, in the beginning of the 1900´s concrete slabs were not of the same type of construction as today, the main function at those times was to have a more practical floor rather than no floor at all so the slabs were quite thin and not insulated from the ground below.

The same comparison for mechanical exhaust and mechanical supply and exhaust is shown in figures 2, and 3 below.

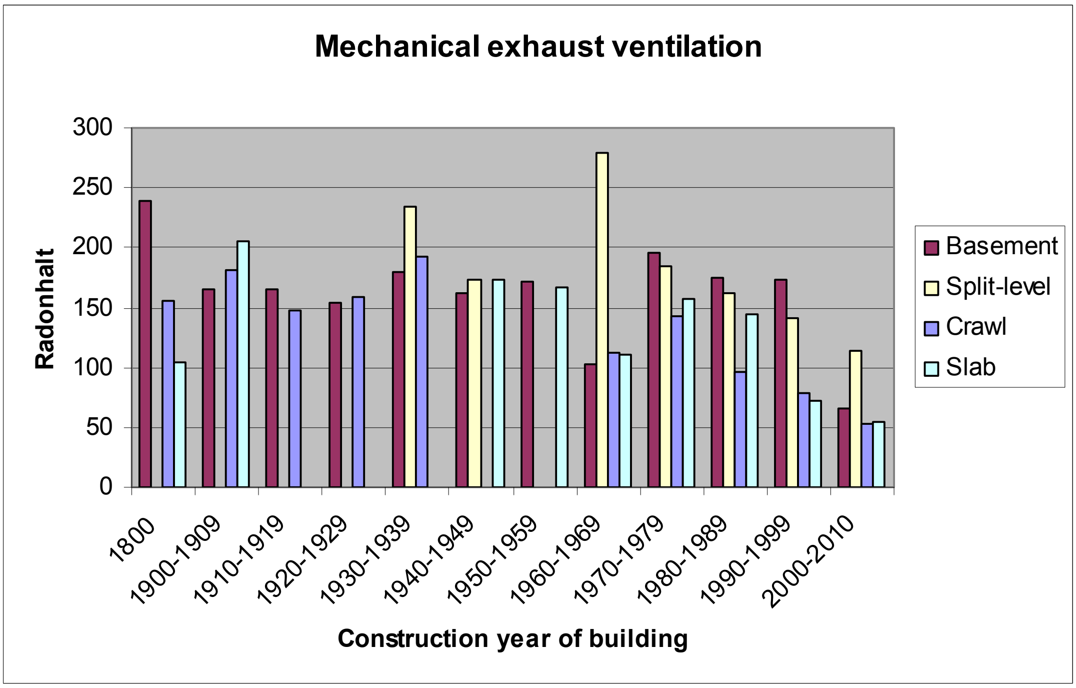

Fig. 2. Annual average radon levels measured in dwellings with mechanical exhaust ventilation which are constructed in the eighteen hundreds until present day and the dependence on house foundation.

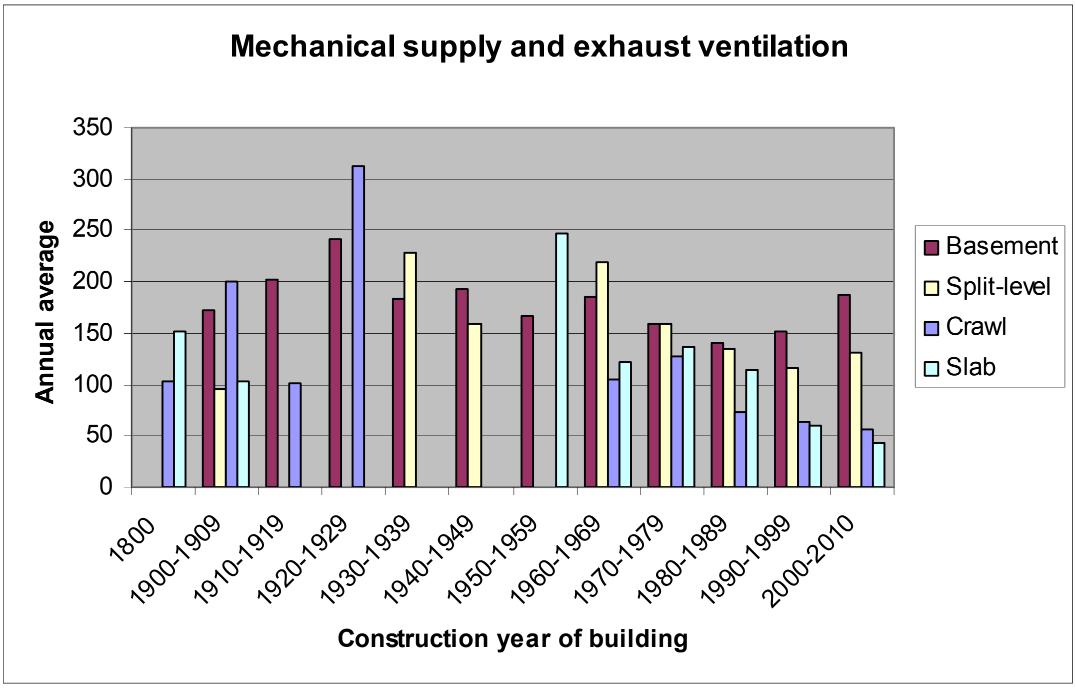

Fig. 3. Annual average radon levels measured in dwellings with mechanical supply and exhaust ventilation which are constructed in the eighteen hundreds until present day and the dependence on house foundation.

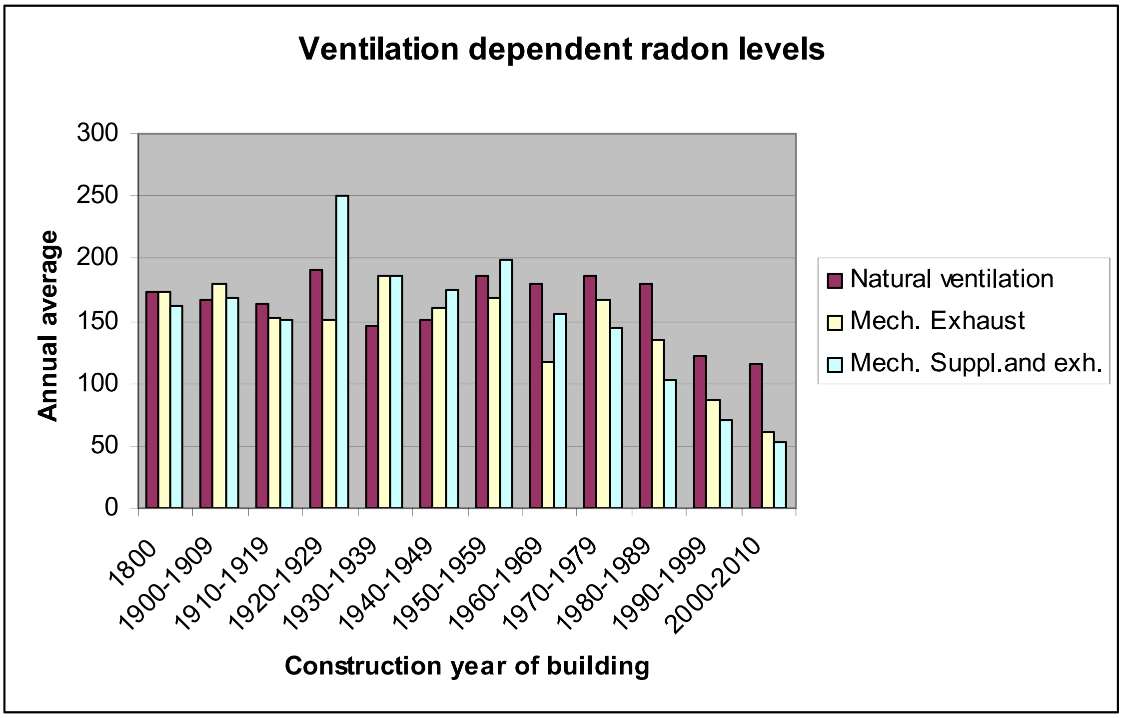

Fig 4. Radon levels in single family dwellings depending on type of ventilation and building construction year.

Mechanically ventilated dwellings have seen a clear improvement in radon levels starting in house constructions in the 1980´s up until the present. Part of the reason may be an increased use of mechanical ventilation in the original building construction. Mechanical exhaust has been used in Sweden starting in the 1930´s while mechanical supply and exhaust air ventilation with heat recovery started being used in the 1970´s (H. Adolphson). For natural ventilation, improvements are seen in buildings from the 1990´s and onward. For mechanical exhaust and supply ventilation this means that in buildings manufactured previous to 1970´s the system was installed afterwards. Although this installation can be assumed to have led to an improved ventilation in the dwelling it has no apparent great effect on the radon level as compared to natural ventilation in these data. In several cases it is evident that mechanical ventilation rather has the opposite effect on the radon level which supports the general notion that using mechanical ventilation does not automatically mean low radon levels (something which our customers informs us of regularly as well). However this does not mean that installation of mechanical ventilation for radon reducing purposes is not effective. As mechanical ventilation is used as a form of remediation it may well be that several of the dwellings included in this data have reduced from an even higher radon.

Another reason that natural ventilation may have a disadvantage over mechanical is that during the energy crisis of the 1970´s the general population was encouraged to reduce their energy consumption by improving insulation in their houses. This is well known to have caused an increase in radon levels (B. Clavensjö) as well as a reduction in ventilation. Therefore it is possible that part of the reason why natural ventilation tends to have higher radon levels is that it no longer is working according to the original construction because the amount of fresh air intake has been reduced.

Conclusions

A modern building with a slab or crawlspace construction together with mechanical ventilation is the best construction according to this study in order to minimize radon levels in single family dwellings. The new WHO recommendations are that radon levels in dwellings should be below 100 Bq/m3. When examining figure 4 above it is noteworthy that the modern buildings with natural ventilation in this study tend to have an average above the WHO recommendation. We therefore propose that mechanical ventilation in combination with slab or crawlspace should be recommended for future construction of single family dwellings

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Radon concentrations in newly built homes in Norway

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Abstract

Since 1997 requirements on radon in new buildings have been included in building regulations and guidelines in Norway. A large survey of radon in dwellings were carried out in Norway in the period 2000-2001 and the percentage of homes with radon concentration above 200 Bq/m^3 was estimated to 9 % (Strand et al. 2001). These homes were mainly built before the 1997 Building Regulations but little has been known about the effects of the regulations and guidelines on the newly built homes.

During the winter season 2008 the Norwegian Radiation Protection Authority carried out a nation-wide survey including 750 randomly selected dwellings built in the period 2000-2007 to investigate to what extent radon is still a problem in the new housing stock. The survey showed that the 1997 Building Regulations in force at the time of the survey, and the guidelines supporting it, had very limited effect on the radon concentration in the newly built dwellings.

Previous to this study it was assumed that the awareness on radon affected areas could be higher in municipalities strongly affected by this problem. Therefore, another survey was also carried out in seven selected municipalities where extensive radon problems had been found in earlier studies. In two municipalities, 50-60 % of the new dwellings had annual mean radon concentrations above 200 Bq/m^3. In one of the municipalities, all 10 newly built homes had radon concentrations below 200 Bq/m^3. In the remaining 4 municipalities, the fraction of radon concentrations above 200 Bq/m^3 was in the range 10-30 %. The conclusions were that the requirements in the 1997 Building Regulations and supporting guidelines had very little effect on radon concentrations in newly built homes.

New building regulations have been adopted after the finishing of this study and will take effect from 1 July 2010. The new building regulations include specific requirements for the use of radon barriers and preparing the building foundations such that it is possible to actuate active ventilation of the ground, if necessary, after finishing the building.

Introduction

The Norwegian Radiation Protection Authority (NRPA) has carried out several radon surveys during the last twenty years and radon is far the largest contributor to annual radiation dose to the public in Norway. There is no known threshold concentration below which radon exposure represent no risk of developing lung cancer. Exposure to radon in homes has been estimated to cause approximately 300 lung cancer deaths each year in Norway.

Radon gas from the ground enters homes through cracks and openings in the foundation of the building. In July 1997 the building regulations in Norway for the first time required planning of building constructions to avoid enhanced health risk by contaminants in the ground, which included radon. A maximum level for indoor radon concentrations was not given in the 1997 regulations. However, 200 Bq/m^3 was the recommended maximum level in the guidelines supporting the regulations, issued by The National Office of Building Technology and Administration and supported by NRPA.

In 2008 two surveys were carried out by NRPA in order to get knowledge about the radon levels in newly built homes, constructed in the period 2000-2007. In the nation-wide survey, the homeowners were randomly selected. In the other survey all relevant homeowners in seven municipalities, known to have extensive radon problems, got an invitation to take part.

The goal of the surveys was to explore whether the radon concentrations in recently built homes are low, and below the maximum level of 200 Bq/m^3, as recommended in the guidelines. Further, the goal of one of the surveys was to study radon levels in newly built homes in municipalities where previous surveys had shown extensive radon problems.

Material and methods

Indoor radon concentrations can vary considerably due to changes in weather conditions and ventilation rates etc. In Norway, long-term integrated radon measurements during heating season are preferred for assessing the annual average radon concentration. The NRPA recommend a measurement period of at least two months and all results should be seasonally adjusted.

Before building a new home a planning permission is required. Addresses for all estates given planning permission from the year 2000 to the start of the project in the autumn 2007 were ordered from the Norwegian land register. Almost 280 000 addresses were received. Planning permission is also required before reconstruction of homes and construction of other buildings, e.g. tool or car sheds. For this reason a lot of homes constructed outside the period of interest (2000-2007) were also measured. These measurements had to be rejected from the study later in the process.

For practical reasons only addresses which was identical to the homeowners address was included in the surveys, excluding homes hired out.

For the nation-wide survey addresses were randomly selected. For the municipal survey all addresses of newly built homes were selected. The municipalities of Drangedal, Grane, Nes (in the county of Buskerud), Skjåk, Tana, Ullensvang and Ulvik were included in the survey (figure 1).

Fig. 1. The seven selected municipalities with extensive radon problems included in the 2008 municipal survey.

In February 2008, an invitation was sent to approximately 2750 homeowners with an offer to participate in the two surveys; 2500 homeowners in the nation-wide survey and 250 in the municipal survey. Enclosed to the letter of invitation the homeowner could find two CR-39 etched track radon detectors, full instructions for placement of the detectors, a questionnaire and a pre-paid envelope for the return of detectors.

The homeowners were asked to place one detector in the main living area and one in a bedroom. Further they were asked to fill in the questionnaire which should be returned together with the detectors. The questionnaire included detailed information about the point of measurements, building construction, ventilation, etc. The data in this study will be further analysed and published at www.nrpa.no in a NRPA report in English.

The overall response rate in the two surveys was approximately 49 %. However, approximately 36 % of the measurements had to be rejected because the homes were constructed outside the period of interest (2000-2007).

Results and discussion

Nation-wide survey

Measurements from a total of 758 randomly selected homes constructed in the period 2000-2007 were used in the nation-wide survey. The results from the survey are given in table 1.

The mean annual radon concentration for all categories of homes constructed in the period 2000-2007 is estimated to 66 Bq/m^3. 8 % of the homes have at least one measurement exceeding the recommended upper level of radon concentration of 200

Bq/m^3. Taking the uncertainties of the measurements into account this is practically the same result, 9 %, as estimated for the Norwegian housing stock mainly built before the introduction of radon requirements in the 1997 Building Regulations.

The mean annual radon concentration is known to vary for different categories of homes. The results for the different categories for this survey are given in table 1 showing that detached homes have got the highest mean annual radon concentration, and the highest percentage of homes above 200 Bq/m^3. The measured terraced homes have lower mean radon concentration than the measured detached homes.

Horizontally separated dwellings in two-family homes also showed high mean annual radon concentration and a high percentage above 200 Bq/m^3, but based on the result of only 24 homes.

Blocks of flats or multifamily homes have got the lowest mean annual radon concentration, 33 Bq/m^3, and 1 % of the homes above recommended maximum level of 200 Bq/m^3. The reason for this is that most of the flats do not have direct contact with the ground.

Table 1. Results from the 2008 nation-wide survey of homes constructed in the period 2000-2007.

Building category	Number of homes	Mean (Bq/m^3)	Maximum (Bq/m^3)	Homes above 200 Bq/m^3 (%)[1]
All categories	758	66	2670	8
Detached homes	320	90	2670	12
Terraced homes	139	66	1254	9
Horizontally separated dwellings in two-family homes	24	71	782	8
Blocks of flats or multifamily homes	236	33	368	1
Unknown	39	59	823	8

[1]Homes with at least one of the two measurements above 200 Bq/m^3
[2]Annual mean radon concentration

Municipal survey

Measurements from a total of 102 homes constructed in the period 2000-2007 were used in the municipal survey. The results from the survey are given in table 2.

Table 2. Results from the 2008 municipal survey of homes constructed in the period 2000-2007.

Municipality	Number of homes measured	Mean (Bq/m^3) (2)	Maximum (Bq/m^3) (2)	Homes above 200 Bq/m^3 (%)(1)
Drangedal	24	352	1900	58
Grane	10	43	150	0
Nes (Buskerud)	15	167	510	60
Skjåk	20	131	900	20
Tana	12	63	220	16
Ulvik	5	129	330	20
Ullensvang	16	100	420	25

[1] Homes with at least one of the two measurements above 200 Bq/m^3
[2] Annual mean radon concentration

The seven municipalities were known to have extensive radon problems. The mean annual radon concentration for all categories of homes is estimated to 166 Bq/m^3. In this survey, the mean annual radon concentrations were varying between 43 Bq/m^3 and 352 Bq/m^3 for the seven municipalities. All measurements in the municipal survey were made in detached homes (100 homes) or terraced homes (only 2 homes).

For comparison the mean annual radon concentrations for the seven municipalities are shown in the bar graph (figure 2) together with two bars representing all homes and the detached homes, respectively, in the nation-wide survey.

In the municipalities Drangedal and Nes (Buskerud), 58 % and 60 % of the homes, respectively, at least one of the two measurements in the home indicated an annual mean radon concentration above 200 Bq/m^3. However, in Grane no homes were identified as being above the reference level given in the guidelines to the building regulations.

For comparison the percentages of homes with at least one measurement above 200 Bq/m^3 in the seven municipalities are shown in the bar graph (figure 3) together with two bars representing all homes and the detached homes, respectively, in the nation-wide survey.

Further analysis have to be done to get a better understanding of why new homes still have high radon concentrations in areas well known to have extensive radon problems. However, the total results indicate that the building regulations have not had the wanted impact on the radon problem in the majority of the high risk municipalities investigated.

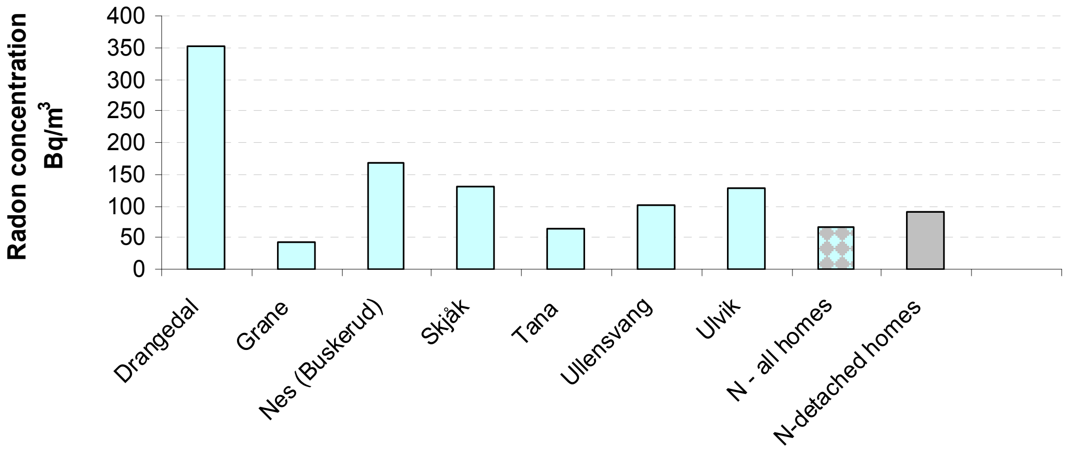

Fig. 2. The mean annual radon concentration in the seven municipalities compared with the mean radon concentration in all the homes and all the detached homes in the nation-wide survey.

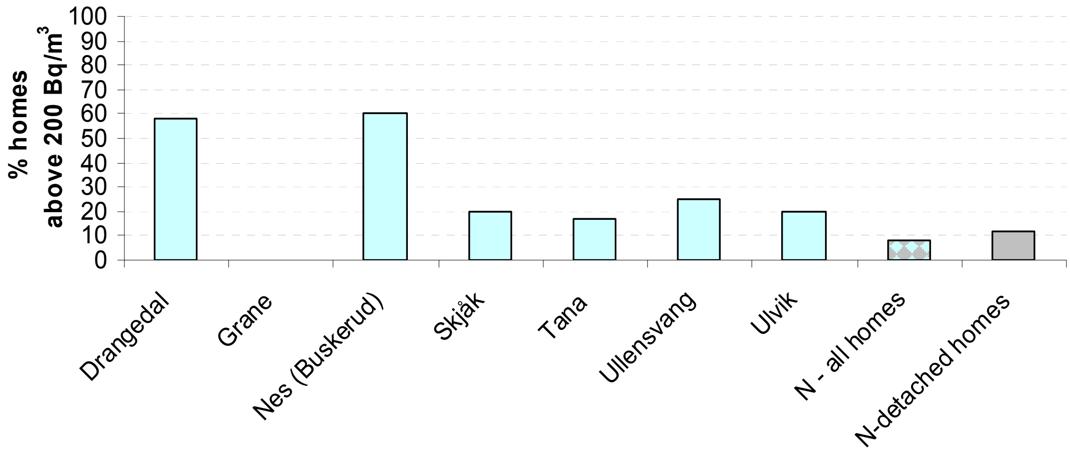

Fig. 3. The percentage of homes with at least one measurement above the maximum upper level of 200 Bq/m^3 compared with the percentage in all the homes and all the detached homes in the nation-wide survey.

Conclusions

Both the nation-wide and the municipal surveys carried out in this study indicate that a significant portion of the newly built homes in Norway have radon concentrations above the recommended maximum level of 200 Bq/m^3. The building regulations, which came into force in July 1997, do not seem to have had any particular influence on the radon concentration in the Norwegian housing stock.

New building regulations have been adopted after the finishing of this study and will take effect from 1 July 2010. The new building regulations include specific requirements for the use of radon barriers and preparing the building foundations such that it is possible to actuate ventilation of the ground, after finishing the building. The new requirements include the introduction of legally binding indoor radon limits for new buildings (100 Bq/m^3, action level; 200 Bq/m^3, maximum). Because the requirements in the new regulations are more specific than the 1997-regulations it will hopefully keep the radon concentrations below the upper maximum level of 200 Bq/m^3 and further reduce the mean annual radon concentration in the coming new housing stock.

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Short-term measurements of radon concentration in dwellings

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Abstract

Radon concentrations in dwellings are known to vary from day to day depending on weather conditions and occupant lifestyle. Consequently, short-term measurements (STM) of a few days are generally less representative of the annual mean radon concentration than long-term measurements (LTM) performed over several months. On the other hand, STM may be used as screening techniques for testing radon levels. The aim of this study was to develop a screening method based on STM to provide fast and reliable indications of radon levels in dwellings.

Radon concentrations were measured in 184 dwellings in Switzerland over a 3 month period using alpha-track detectors (Radtrak). Together with LTM, STM were performed at 3 time points, i.e. at the beginning, the middle and the end of the 3 month period, using activated charcoal canisters (Picorad) exposed during one day. A statistical comparative analysis of LTM and STM in those dwellings was performed to derive the screening method.

Results obtained with STM overestimated, on average, by 35% those obtained with LTM, with a standard deviation of 65%. This was explained by the reduced ventilation applied during STM. The variability of STM at the three different time points was characterized by a standard deviation of 24%. Based on those results, the following screening method was proposed considering an action level of 400 Bq/m^3:

(a) if results of STM are less than 110 Bq/m^3, the annual mean radon concentration is below 400 Bq/m^3 with a confidence level of 95%;
(b) if results of STM are above 970 Bq/m^3, the annual mean radon concentration exceeds 400 Bq/m^3 with a confidence level of 95%;
(c) if results of STM range between 110 and 970 Bq/m^3, LTM are required to determine whether the dwelling is above or below 400 Bq/m^3.

This method was validated for several dwellings. We conclude that STM provide valuable information on radon levels in dwellings to determine if further LTM are necessary. In Switzerland, this STM screening method is primarily used in real estate transactions.

Introduction

Radon exposure is the main component of the radiation dose to the Swiss population and the leading cause of lung cancer after smoking. As part of preventive healthcare, the Radiological Protection Ordinance set in 1994 a yearly average limit of 1000 Bq/m^3 and introduced a guidance value of 400 Bq/m^3 for the concentration of radon gas in Swiss dwellings. A major objective of the ongoing national radon programme is then to identify dwellings where radon concentration exceed the legal limit of 1000 Bq/m^3 and ensure that remediations are performed. To determine radon concentrations, the Swiss Federal Office of Public Health (FOPH) recommends measurements over a 3 month period using dosimeters, during the heating period. Most of the usual dosimeters like alpha-track detectors and electret ionization chambers are used for such long-term measurements (LTM). In Switzerland, those dosimeters are available from approved measuring laboratories.

Since radon concentration fluctuates from day to day depending on atmospheric conditions and occupant lifestyle, short-term measurements (STM) of a few days are less likely than LTM over several months to represent the yearly average radon concentration in dwellings (U.S. Environmental Protection Agency, 2009). Nevertheless, if results are needed quickly, it is widely accepted that STM may be used as screening techniques for a rapid initial testing of radon levels, especially if radon concentration is likely to be significantly higher than the national average value. For instance, there is often limited time to deal with radon in case of real estate transaction and people are asking more and more about radon levels before they buy or rent a home.

The aim of this study was to develop a screening method based on STM to provide fast and reliable indications of radon levels in dwellings. For this purpose, we first aimed at testing STM as compared to LTM that reflect a better average radon concentration. A strategy of screening measurement based on the reliability of STM was then derived and assessed.

Material and methods

As part of a measurement campaign, radon concentrations were measured in 184 dwellings of the canton of Vaud in South West of Switzerland over a 3 month period using alpha-track detectors (Radtrak). The canton is located by the Lake Geneva and at the foot of the Jura Mountain where elevated indoor radon levels were observed. Together with LTM, single STM were performed at 3 time points, i.e. at the beginning, the middle and the end of the 3 month period, using activated charcoal canisters (Picorad) exposed during one day. Each single STM was performed simultaneously with two Picorad detectors in order to assess their reproducibility. Detectors were always placed in the lowest inhabited area in contact with normal breathing air. Areas with a high air circulation rate were avoided (doors, windows). A statistical comparative analysis of LTM and STM in those dwellings was performed.

In order to assess whether electret detectors (E-Perm SST) can be used alternatively to activated charcoal canisters for STM, both types of detectors were exposed simultaneously in a home-made radon chamber during 1, 2 and 3 days. Measurements were limited to a period of 3 days to avoid saturation of activated charcoal canisters. The obtained radon concentrations were then normalized to the reference concentration in the radon chamber measured with an AlphaGuard monitor traceable to the international standard. Moreover, the same experiment was conducted in situ, in a house with high radon levels (~ 1500 Bq/m^3). Performances of both activated charcoal canisters (Picorad) and electret detectors (E-Perm SST) were compared.

Results

Radon concentrations measured with LTM varied between a few Bq/m^3 and 900 Bq/m^3, with an average of 62 Bq/m^3. For STM, the average radon concentration was 85 Bq/m^3. A good reproducibility was found for activated charcoal canisters as showed by the standard deviation of 13% obtained between responses of both dosimeters exposed simultaneously at the same position during a single STM. The dispersion of the three STM performed with both activated charcoal canisters was 24%. Indeed, the Gaussian distribution of the ratio of each single STM to the mean of the three STM given in Fig. 1 is characterized by a mean of 1.00 and a standard deviation of 0.24. This dispersion characterizes the temporal variability of radon concentration and represents the reliability limit of STM.

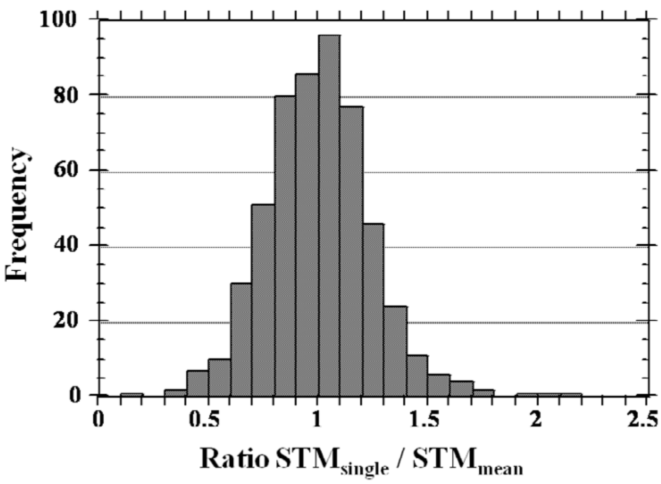

Fig. 1. Distribution of the ratios between the single STM and the mean of three STM (beginning, middle and end of the period). Results of each single STM correspond to the average of radon concentrations measured simultaneously with two activated charcoal canisters.

The distribution of the ratio between each single STM and the LTM was considered Gaussian with a mean (μ) of 1.35 and a standard deviation (σ) of 0.65 (see Fig. 2). The overestimation of 35% obtained with activated charcoal canisters can be explained by ventilation restrictions imposed before and during the STM. Based on this Gaussian distribution (μ = 1.35 and σ = 0.65), it is possible to calculate the probability P that the LTM exceeds a given action level AL using the radon concentration C obtained from STM. The following expressions are then derived:

$$5\% = P\left(\frac{C}{AL} < (\mu - 1.64 \cdot \sigma)\right) = P(C < 0.28 \cdot AL)$$

$$95\% = P\left(\frac{C}{AL} < (\mu + 1.64 \cdot \sigma)\right) = P(C < 2.42 \cdot AL)$$

Thus, considering a given action level AL, the probability that the LTM is below L_{INF}, defined as 0.28·AL, is of 5%. In other words, the probability that the LTM is above L_{INF} is of 95%. Similarly, there is a probability of 5% that the LTM is higher than L_{SUP}, defined as 2.42·AL. Values of L_{INF} and L_{SUP} are given in Table 1 for three different action levels.

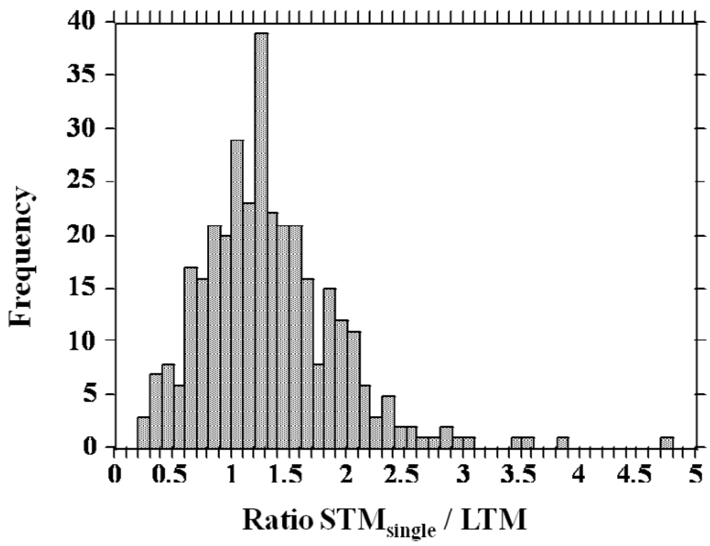

Fig. 2. Distribution of the ratios between the single STM and the LTM.

Table 1. Values of L_{INF} and L_{SUP} according to different action levels.

Action Level [Bq/m^3]	L_{INF} [Bq/m^3]	L_{SUP} [Bq/m^3]
200	60	480
400	110	970
1‘000	280	2’420

Results obtained with electret detectors (E-Perm SST) and activated charcoal canisters (Picorad) are compared in Fig. 3 for both measurement conditions, i.e. in the radon chamber and in the existing house. Each point corresponds to the average radon concentration measured with two dosimeters normalized to the reference concentration obtained with the AlphaGuard monitor during the measurement period (1, 2, 3 or 7 days). A very good agreement was observed for electret detectors, always within ± 10% except for the one-day measurement in the radon chamber (-12%). A relatively good agreement was also found for activated charcoal canisters with a maximum deviation of -22% for the one-day measurement in the radon chamber and +38% for the three-day in situ measurement.

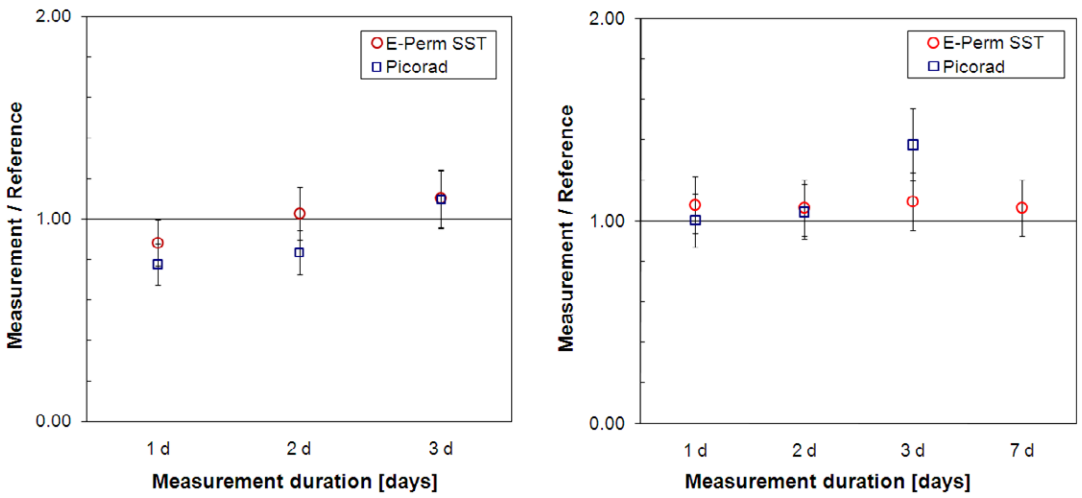

Fig. 3. Comparison of radon measurements with electret (E-Perm SST) and activated charcoal (Picorad) detectors. Measurements were performed in a radon chamber (left) and in an existing house (right) with a radon concentration of about 1500 Bq/m^3. Each measurement corresponds to the average of radon concentrations measured simultaneously with two detectors.

Discussion

Based on these results, we propose a screening method based on STM to provide fast and reliable indications of radon levels in dwellings. For the measurements, windows and doors should be kept closed at least 12 hours before starting the measurement and during the whole measurement period. The recommended duration of measurement is 2 days using activated charcoal canisters (Picorad) and at least 2 days using electret detectors (E-Perm type SST). Note that electret detectors can be quickly evaluated immediately at the end of a measurement and thus are somewhat more appropriate than activated charcoal canisters for STM. Results of STM are then interpreted as follows, for a given action level (values of L_{INF} and L_{SUP} are given in Table 1):

(a) if the result of STM is less than L_{INF}, the yearly average radon concentration is below the action level with a confidence level of 95%;

(b) conversely, if the result of STM is above L_{SUP}, the yearly average radon concentration exceeds the action level with a confidence level of 95%;

(c) while no additional measurements are required in both cases (a) and (b), if STM range between L_{INF} and L_{SUP}, LTM are necessary to determine whether the radon concentration in the dwelling is below or above the action level.

In the UK, a similar study was conducted by the Department for Environment, Food and Rural Affairs (Phillips P.S et al., 2004). STM were based on a period of one week using charcoal detectors, electret detectors and alpha-track detectors. Results of STM were also compared with 3-month measurements performed with alpha-track detectors. For an action level of 200 Bq/m^3, the DEFRA reported similar values of L_{INF} and L_{SUP} as those showed in Table 1, i.e. 68 and 522 Bq/m^3 when activated charcoal canisters were used and 59 and 667 Bq/m^3 using electret detectors.

Our screening method was validated for several dwellings in Switzerland. In our practice, STM are mainly used in the framework of real estate transactions.

Conclusions

Short-term measurements (STM) are appropriate for a rapid screening test of radon levels in dwellings. In this study, a method based on STM, i.e. two-day measurements using electret detectors or activated charcoal canisters, has been developed to determine, with a confidence level of 95%, if an action level (in Bq/m^3) has been exceeded or not and if further long-term measurements are required. In Switzerland, STM are essentially being used in the framework of real estate transactions.

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Estimating the health benefits of progeny extraction units as a means of reducing exposure to radon

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Abstract

Radon exposure to the general public can be reduced by preventing entry of radon gas into buildings using a passive radon-proof membrane or an active sump and pump system. However, a significant majority of the radiation dose delivered is from the decay products of radon rather than from the gas itself. These decay products (also referred to as progeny) are present in indoor air, with an equilibrium factor – a measure of the ratio of progeny to radon gas – of between 0.4 to 0.5. As a result, systems which extract radon progeny from the air by filtering have been promoted as means of reducing exposure to the general population.

The European Community Radon Software (ECRS) offers a means of estimating lung-cancer risk associated with an individual's exposure to radon, and includes the possibility of estimating the health risk from different proportions of radon gas and its progeny by varying the value of the Equilibrium Factor. This software was used to estimate the health benefits associated with reduced decay products in differing concentrations of radon gas. The results were compared to health benefits expected if the risk was reduced by the standard method of reducing the radon gas concentration below the Action Level, which in the UK is 200 Bq·m^{-3} for domestic properties.

These calculations showed that there is the potential for efficient extraction units to provide the necessary dose and risk reduction where initial average radon gas concentrations are up to 800 Bq·m^{-3}. However, above 1000 Bq·m^{-3}, such systems cannot reduce the health risk sufficiently to reach levels comparable to those resulting from radon gas reduction to below the Action Level.

Introduction

The naturally-occurring radioactive gas, radon, is the second most significant risk for lung cancer after tobacco smoking. High levels of radon were first identified in uranium mines but, more recently, it has been established that significant levels are found in the built environment, and case-control studies have shown an associated increase in lung-cancer in the public from radon in their homes (BEIR VI 1999; Darby et al. 2005; AGIR 2009).

Radon decays into other radioactive elements, and two of these, Polonium-218 and Polonium-214, have been shown to deliver a significant proportion of the radiation dose received by occupants of domestic premises. In a sealed system, radon and its progeny exist in secular equilibrium. In the domestic environment, however, secular equilibrium is never maintained, because progeny are continuously being removed from indoor air by surface deposition and ventilation. The degree of disequilibrium between the radon gas and the progeny is represented by the Equilibrium Factor (EF), which is defined as the ratio of the potential alpha energy concentration (PAEC) of the actual progeny mixture to the PAEC of progeny in secular equilibrium with the radon gas. Thus EF is between 0 and 1, and within domestic buildings is typically in the range 0.4 to 0.5. Although they are intrinsically solid materials, radon progeny can exist in suspension in air, either unattached, or attached to aerosol particles. Porstendörfer (1984) has discussed these processes, including plate-out of particles and the influences on the ratio between radon and progeny.

Although technical solutions to the radon problem in domestic properties have been largely based on prevention of ingress of radon gas from the soil under the influence of the climatically-dependent pressure difference between the interior of the dwelling and the external environment, there has recently been renewed interest in the physical removal of radon progeny by direct filtering of the internal air, thereby reducing the equilibrium factor between radon gas and its solid progeny.

A number of studies by our own group (Marley et al. 1998) and others have confirmed that filters can indeed remove radon progeny from air (Ogorodnikov et al. 1962; Busigin et al. 1980) and their efficacy as respiratory filters in mining has been studied (Wake et al. 1992).

Offermann et al. (1985) reported that an air-cleaner used to control the indoor concentration of respirable particles and radon progeny, especially if fitted with a high efficiency particulate air filter (HEPA-filter), was effective in removing radon progeny, a finding confirmed by Li and Hopke (1991), who reported that air cleaners effectively reduce the median dose due to radon and its progeny, and that air-filtration is more effective than electronic air cleaning. However, while air cleaning can be effective in reducing total radon progeny, concentrations of unattached radon progeny, a dominant factor affecting the dose conversion factor, can increase with air-cleaning. Henschel (1994) reviewed techniques for the purpose of reducing indoor radon concentration, commenting that the effects of air-cleaners on health risk were unclear due to the increase in the unattached fraction. Tokonami et al. (2003) noted that use of an air-cleaner enhances the dose conversion factor critically, since the unattached fraction increases significantly due to aerosol removal. Most recently, Kranrod et al. (2009) showed experimentally that radon progeny dose reduction of the order of 50% was feasible by use of a well-configured air-cleaner.

These uncertainties notwithstanding, there are several devices on the market specifically claiming to reduce the risk from radon[1,2]. However, the US Environmental Protection Agency (EPA) does not recommend radon remediation using filters (EPA 2010).

[1] http://www.nrpltd.com/radon_gas_filter.php
[2] http://www.airpurifiersandfilters.com/radon-air-purifiers.php

Material and methods

The European Community Radon Software (ECRS) tool (Degrange et al., 2000) performs lung-cancer risk calculations specific to European populations for individual or collective radon exposure profiles. Specifically, the software is capable of generating a range of individual risk-related estimates, including reduced life expectancy and expected age at death, for subjects whose age, sex, smoking habits and domestic radon exposure are known, and can furthermore take into account the equilibrium factor applicable to the environment being investigated. ECRS has therefore been used in the present study to consider whether deploying EF-reducing filtration techniques in homes can produce a significant health benefit to occupants.

In addition to basic demographic, radon exposure and smoking status input parameters, ECRS permits user control of Equilibrium Factor and aerosol parameters. Two Equilibrium Factor tables are provided, attributable to ICRP 65 (ICRP, 1994) and the UK National Radiological Protection Board (NRPB) respectively. Whereas the ICRP parameters, the system default, do not vary with room type, the NRPB data addresses the possible impact of ventilation rate and air smoke content on the equilibrium factor. For the present study, a derivative of the ICRP dataset was used, with identical values of Equilibrium Factor being used throughout the house for each case modeled.

ECRS modeling addresses additional aerosol parameters, representing the overall radon progeny activity-size distribution by a sum of lognormal distributions (modes). For each of these modes, unattached progeny and three attached modes (nucleation, accumulation and coarse), values are given for fraction of total PAEC, dispersion (i.e. geometric standard deviation), and particle size in terms of Activity Median Aerodynamic Diameter (AMAD) for attached particles and Activity Median Thermodynamic Diameter (AMTD) for unattached particles. Two data models are provided, a default Generic House, where the parameters do not vary with room type, and a generalized model, which considers the possible impact of ventilation rate and air smoke content on the aerosol parameters and provides specific room-by-room data. The Generic House option was used in the present analysis.

For the present study, analysis was based on 40-year-old, Non-Smoking and Smoking Males and Females exposed to radon levels of 0, 200, 600 and 1000 $Bq{\cdot}m^{-3}$, for Equilibrium Factors ranging from 1.0 down to 0.02. Using the results from this analysis, iso-risk plots were generated, identifying the effect of reduction of Equilibrium Factor on individual excess lung-cancer risk. The calculated risks of contracting lung cancer were then compared to the risk at a radon level of 200 $Bq{\cdot}m^{-3}$ (the UK domestic Action Level) with an Equilibrium Factor of 0.5.

Results

At all levels of Equilibrium Factor and for both sexes, excess lung-cancer risk is linearly proportional ($R^2 > 0.99$) to radon exposure over the range 0 - 1000 $Bq{\cdot}m^{-3}$ (five times the UK Action Level). Specific Risk factors (incremental risk attributable to 1 $Bq{\cdot}m^{-3}$ exposure) for 40-year-old Non-Smoking Males and Females are themselves linear in Equilibrium Factor (Figure 1), as is loss of life expectancy consequent on radon exposure (Figure 2).

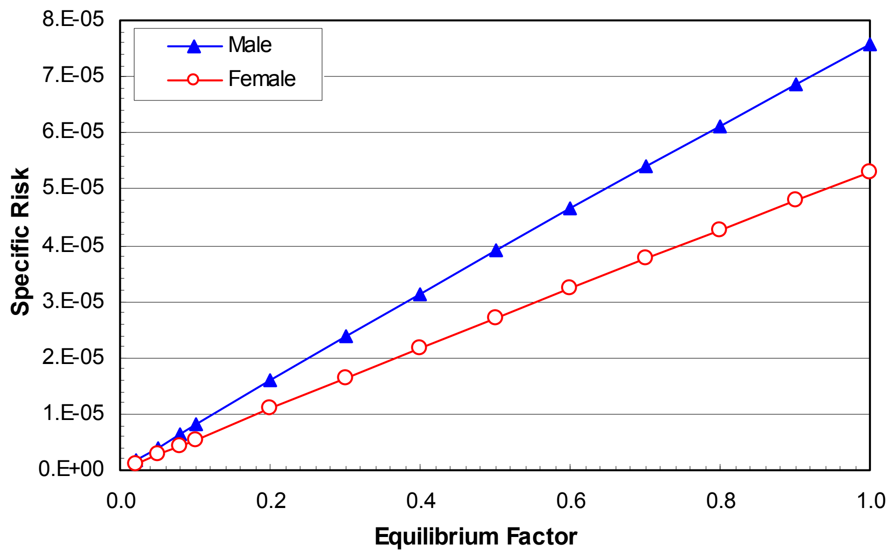


Fig. 1. Variation of excess risk per unit Bq.m^{-3} radon concentration, linear in radon over the range 0 – 1000 Bq.m^{-3}, with Equilibrium Factor for 40-year-old non-smoking Males and Females.

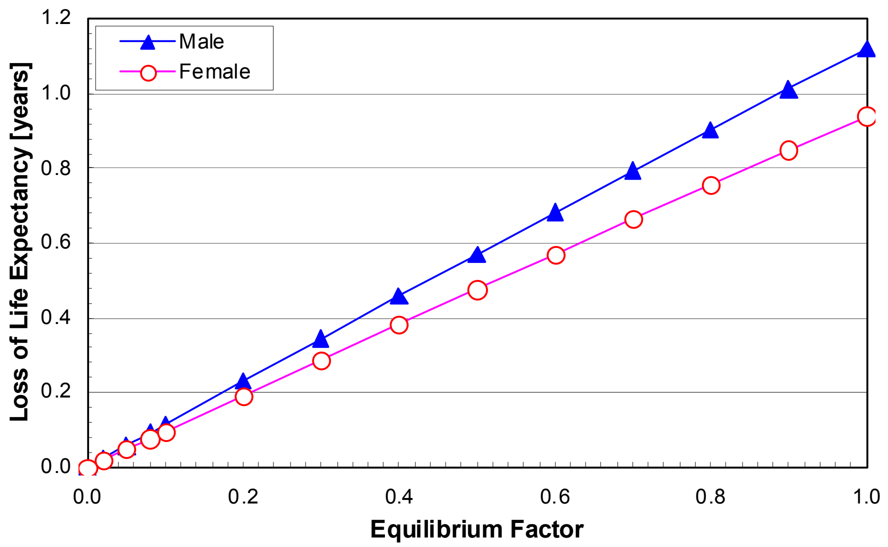
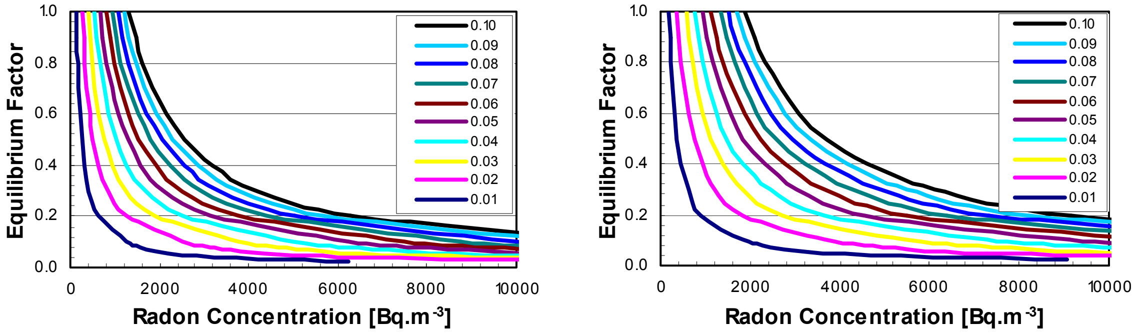


Fig. 2. Loss of Life Expectancy Attributable to Radon Exposure of 1000 Bq.m^{-3} for 40-year-old non-smoking Males and Females.

Excess risk represents the additional risk of contracting lung-cancer directly attributable to radon exposure. This is a complex function of age, sex, radon exposure level and duration, equilibrium factor and smoking status. By way of example, Figure 3 shows the set of iso-risk plots in the exposure-level/equilibrium factor domain, generated for 40 year-old non-smoking males and females with whole-life exposure to the indicated radon concentration levels, spanning the range of excess risks from 0.01 to 0.1 (1% - 10%). Similar families of curves can be generated for individuals of other ages, of different smoking status, and with any desired combination of radon exposure and home occupancy. These all have the same rectangular hyperbolic analytical form, the product of radon concentration and equilibrium factor being constant for any given level of excess risk.



(a) Males **(b) Females**

Fig. 3. Iso-Risk Plots for 40-year-old Non-Smokers, Excess Lung-Cancer Risk as parameter.

Discussion

At an exposure level of 200 Bq·m^{-3} with an Equilibrium Factor of 0.5, the current UK Action Level case, the excess risks of contracting lung-cancer are 0.008 and 0.0055 for Males and Females respectively. To facilitate further analysis and discussion, Figure 4 reports the 0.008 risk curve for Male subjects over the typically encountered range of domestic radon concentration levels. A similar curve can be generated for Females, in this case with a risk parameter of 0.0055.

As the figure shows, the iso-risk curves in the EF-radon domain take the analytical form of rectangular hyperbolæ (in this case, characterised by *EF•radon* = 106), a fact readily verified by inspection. Thus, if the Equilibrium Factor in a 200 Bq m^{-3} environment is reduced by a factor of 5 to 0.10, a reduced excess lung-cancer risk situation obtains at all radon levels from zero up to 1000 Bq m^{-3}.

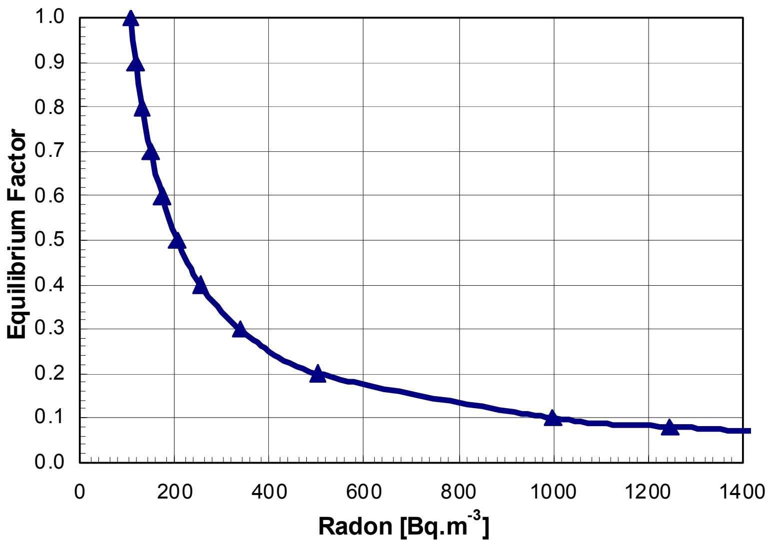


Fig. 4. Iso-Risk Plot (Excess Risk = 0.008) for Males in 200 Bq.m^{-3}, EF = 0.5 Environment.

Curling et al. (1990a) developed an optimization model for filter thickness, solidity, and fibre diameter to minimize inhalation dose from radon decay products, based on modified forms of the Porstendorfer-Jacobi (Porstendorfer, 1984) room model and the Jacobi-Eisfeld (Jacobi and Eisfeld, 1980) lung-dose model. The resulting optimal design, a thin filter of low solidity and large fibre diameter, confirmed that significant reduction in the dose rate could be achieved using a filter system. While the theoretical model predicted 80% reduction in the dose rate, with the inherent assumption of movement of 230 room volumes per hour through the fan, initial experimental trials (Curling et al. 1990b) achieved 50% reduction.

Kranrod et al. (2009) confirmed by measurement that an air-cleaner equipped with an HEPA filter and a deodorizing activated carbon filter preferentially removed the attached fraction of radon progeny from the room air, reducing EF to around 71% of its un-filtered value. Because of the differing contributions to the risk from radon gas, attached progeny and unattached progeny, the 71% reduction in Equilibrium Factor achieved in this way again resulted in a 50% reduction in dose. In the case studied, the initial radon gas level was around 300 $Bq \cdot m^{-3}$ and the risk to occupants was reduced below that at the Action Level of 200 $Bq \cdot m^{-3}$.

Examples in the literature of the Equilibrium Factors achieved in rooms where such filtration systems were operational are limited. Marley et al. (1998) measured F=0.17 in a UK hospital operating theatre, while Li and Hopke (1991) found F=0.134 in a US house. Reduction of the Equilibrium Factor to around this level in any room where the average radon gas level is below 800 Bq m^{-3} would reduce the risk to occupants sufficiently. If an equilibrium factor of 0.1 could be achieved, then the upper limit would be 1000 Bq m^{-3}, and so this could be regarded as an upper limit for such systems.

The performance of air-cleaners will depend on the air volume passing through the filter, and therefore will depend on the size of the room, the air throughput capability of the system and the characteristics of the filter. Thus it would be expected that an air-cleaner will not achieve as significant a reduction in progeny in a larger room. In addition, the movement of radon gas around the building needs to be considered. Additional units may be required in other rooms, if the radon gas moves slowly enough between rooms to decay and create new decay products, or if the source of radon gas and decay products is in another part of the building and reaches the living room and bedrooms independently.

For this reason, such units cannot be installed in any arbitrary room within a dwelling and be expected to achieve significant health benefits throughout the home. The performance of commercial units needs to be tested in standard conditions to ensure that a sufficient reduction in Equilibrium Factor is achieved. Furthermore, when deployed in a home, the performance needs to be checked. Standard track-etch devices only measure radon gas, and so there will be a need to develop simple assessment devices which measure both radon gas and progeny if these sort of units are to be deployed widely.

Conclusions

The theoretical calculations demonstrated in this paper demonstrate that air filtration units can reduce occupants' risk below the limit implied by the Action Level, provided that the radon gas level in the building is moderately raised above the Action Level. At increasing radon levels the air filtration units must be more efficient to achieve the required benefit, and above 1000 Bq·m^{-3}, the unit cannot reduce risk sufficiently to reach the target.

Thus such units may have their place in remediating homes with radon levels up to 800 Bq·m^{-3}, and may also be of value in homes remediated by other means where the level has been brought down, but not below the Action Level. Kranrod et al. (2009) have confirmed that satisfactory performance of such units can be achieved, at least at 300 Bq·m^{-3}.

Several practical issues remain before such units can be widely adopted. Firstly the performance of commercial units needs to be evaluated to assess how far they can reduce the Equilibrium Factor in a range of room sizes. Secondly, a simple monitoring system that can measure both radon gas, and progeny, and is comparable to the simplicity of the track etch systems for radon gas, must be developed, so that the health benefit from such units can be proven.

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Indoor radon anomalies and correlation with geodynamic events: A case study in the Etnean area

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Abstract

More than 50% of the total absorbed dose from natural radiation sources is attributed to radon gas and its daughters (UNSCEAR, 1993). In the last decades, an increasing regulatory and scientific interest concern radon indoor monitoring, because radon and its daughters produces a risk of lung cancer, by inhalation of high concentrations for a long time. The indoor radon transportation is due principally by two different mechanisms: diffusion from soil and building materials and convection flow. The diffusion mechanism depends on the radium concentration in soil and building materials, as well as on the diffusion coefficient; convective flows are generated by a pressure difference between the inside and the outside of a building. Many external factors can also influence the diffusion process: rainfall, freezing and increasing atmospheric pressure decrease the exhalation rate, while increasing temperature can increase it. In this study temperature, pressure and humidity are monitored together with radon concentrations.

In 2006, for a period of four months between winter and spring, both in soil and indoor radon gas, and meteorological parameters, have been monitored continuously in two buildings at St. Venerina, situated on the low eastern flank of the volcano. This area is crossed by the Timpe Fault System (TFS), that are composed of faults mainly trending towards NNW–SSE and NW-SE. The indoor radon anomalies have been studied trough statistical methodology and they have been correlated with the geodynamic events of volcano Etna, recorded during the investigated period. Its values seems to show variations not due to local meteorology, diffusion or convection mechanisms, instead the predominant effect seems to be the geodynamics of the Etnean area. This study is an example that shows how in the active volcano-tectonic areas, the indoor-radon monitoring, based only on seasonality and floor numbers, is not exhaustive to establish the indoor radon levels, and how radon accumulation may depends on several and complex effects, each one able to generate anomalies of its concentration.

Introduction

During his life, man is constantly exposed to many ionizing radiation sources of natural origin, the most important of whose are ^{40}K and the radionuclides produced by the radioactive families of Uranium and Thorium. These elements have been produced at the time of Earth's creation and their decay products are widely distributed in soil, water and air. Of particular interest from the scientific point of view is the presence of radon gas in the air, in particular the isotope ^{222}Rn, produced by alpha decay of ^{226}Ra, which belongs to the decays chain of ^{238}U. Radon is a noble gas, odorless, colorless and tasteless, seven times heavier than air. Since it is chemically inert, it is characterized by great mobility. The World Health Organization (WHO) through the International Agency for Research on Cancer, included the as a group 1 carcinogen - in a list of 75 substances. The new document that the WHO has recently presented about has raised the level of concentration at the national reference value of 100 Bq/m^3, with an express recommendation to not exceed the limit value 300 Bq/m^3 (WHO HANDBOOK ON INDOOR RADON A PUBLIC HEALTH PERSPECTIVE - 2009), these values should be useful to the national authority to enact the appropriate construction standards. Exposure to radon in dwellings due to the presence of Uranium in soil and building materials is one of the major problems of radio-protection today.

^{222}Rn studies and monitoring showed that this geo-gas can reveal a variety of interesting features, as well as permit a depth comprehension of the Earth's crust.

Relatively recent is a new line of research, that is foretell geodynamic events through the sudden fluctuations of its concentration, the so-called ***radon anomalies***, which can be correlated with the geodynamics of the investigated area (La Delfa et al., 2007; Planinic et al., 2005; Vizzini et al., 2006-2007; Zmazek et al., 2005). The link existing between radon in soil anomalies and seismic events, has been explained by the so-called ***pore-collapse model*** (J.-P. Toutain et al., 1999).

The site

The town of St. Venerina (figure 1) is located on the lower eastern slope of volcano Etna. All the area is exclusively composed of volcanic soils, which refer to lava flows of different ages. From the structural point of view, the territory is characterized, as the entire eastern slope of Mt. Etna, by fault systems with prevailing directions NE-SW and NW-SE, which affect all the area. Some of these tectonic dislocations are well evident by the presence of sharp uneven terrain, the largest of which are called "Timpe".

Based on historical data, some of these faults, are defined “seismogenic”, because they are responsible for seismic activities, both in the recent and historical periods. The seismic activity that is originated from these faults, usually affects relatively shallow crustal levels ($h \leq 7$ km), with generally modest magnitude of earthquakes. However in periods of volcanic eruptions the related seismic events can reach the highest values recorded in all the Etnean area. What previously said was particularly evident in the well-known disaster occurred in October 29, 2002, during one of the most intense eruptions of Mount Etna, when a seismic swarm, and one great earthquake of magnitude 4.4 (Richter scale) and intensity VIII (MCS scale), with epicenter in St. Venerina, was responsible for considerable damages to persons and dwellings.

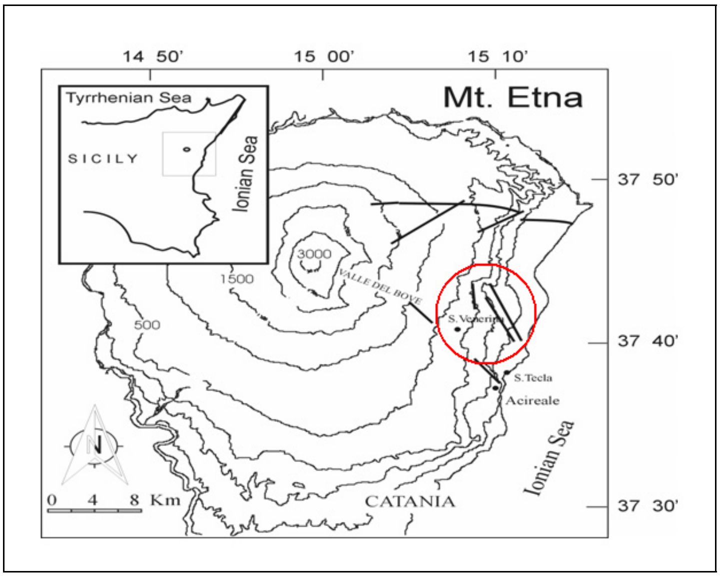

Fig. 1. Map of Mt. Etna and Si.Ge.T. active faults (red circle).

Material and methods

The main contribution of the in soil radon concentrations is due to the emanation process from soil or fractures in the ground. Materials with high Uranium and Thorium contents and which are located near the Earth's surface, can produce a high radon exhalation rate. In this context, Sicily is an island with different specific geological characteristics, offers a variety of lithotypes and therefore of features for building materials. The rocks and soils in all the Etnean area, are characterized by igneous rocks, which have the most Uranium and Thorium contents among the other lithological varieties of the island (metamorphic and sedimentary rocks). Therefore it was considered appropriate to effect a characterization of some rock samples, using gamma spectroscopy, to measure the primordial radionuclides contents. This study, as well as of academic nature, has practical implications on the human exposure to environmental radioactivity, since the rocks sampled in the monitoring sites are those processed and used for the construction of buildings for residential purposes, and this is particularly evident in all the Etnean area and at St. Venerina in particular, where the large part of dwellings are built with volcanic rocks and ash. The samples analysis has been effected revealing the γ-ray emitted by the radionuclides originating from ^{238}U and ^{232}Th decay chains and from ^{40}K, using a HPGe detector with efficiency of 40% and electric cooling. The measurements of radionuclides concentrations are used to determine both the indoor radon concentration and the gamma dose rate (Rizzo et al., 2001).

A benchmark for the radiological risk assessing due to external gamma radiation, is the ***Radium equivalent activity***. The meaning of the Radium equivalent activity is to have a single number or index that can describe the *γ-output* which is originated from different mixtures of Uranium (or Radio), Thorium and ^{40}K in a material. The Radium equivalent activity can be calculated as follows (Beretka and Mathew, 1985):

$$Ra(eq) = A(Ra) + 1.43A(Th) + 0.077A(K)$$

where:

A (Ra) is the ^{226}Ra activity (which is the same as ^{238}U) in Bq/kg

A (Th) is the ^{232}Th activity in Bq/kg
A (K) is the ^{40}K activity in Bq/kg

This equation is based on the assumption that 370 Bq/kg of ^{238}U, 259 Bq/kg of ^{232}Th and 4810 Bq/kg of ^{40}K produce the same dose range. Moreover, the Ra(eq) value of 370 Bq/kg is equal to the annual dose equivalent of 1.5 mSv per year, which represents the highest absorbed dose by a person in a year, caused by the exposure to natural radioactivity coming from building materials. The Radio equivalent activities obtained in the St. Venerina area, oscillate between 221,32 Bq/kg and 246,74 Bq/kg, with an average value of 236,01 Bq/kg, that is well below the threshold limit of 370 Bq/kg.

The Uranium and Thorium contents in rocks in the territory of St. Venerina, don't explain the in soil radon concentrations, more than 20000 Bq/m^3, which were recorded in previous studies (La Delfa et al., 2009; Vizzini et al., 2006-2007), and even more the indoor radon concentration fluctuations, from 2 to 5 times than the seasonal average concentrations, which seems to not be linked with seasonality. The radioactivity based on the primordial radionuclides content is indeed a constant value, characteristic of the lithology of the site. Based on these considerations, a punctual investigation in the territory of St. Venerina has been effected, near tectonic structures in the north part of the town, within the so-called geostructural system of Timpe (Si.Ge.T.), that is a faults system oriented NNW-SSE (red circle in figure 1).

The study of seismicity and the possible correlation with radon concentration fluctuations was carried out during the period between 12th January to 12th April 2006.

This temporal range was conveniently chosen as significant for these studies because a change in the seismic and volcanic styles of the volcano has been detected, while the meteorological parameters, both for indoor and in soil radon monitoring, had not drastic changes of their mean values in all the investigated period.

The in soil radon concentrations collected, showed fluctuations which, as known from the literature, are attributable both to changes in meteorological parameters, and to changes due to the dynamics of the volcano (Planinìc et al., 2005; Shi Yuchun and Xu Yingfeng, 1994; Zmazek et al., 2005). From the radon gas concentrations and the meteorological parameter data (temperature (°C), pressure (mBar) and relative humidity (%)), recorded every 10 minutes from the active monitoring device AlphaGUARD (GENITRON), the daily average values have been calculated. The in soil experimental data of St. Venerina have been collected in order to identify radon anomalies, precursors of seismic events related to the volcano Etna activity.

The results obtained, which are published in La Delfa et al., 2007, showed that there is a close correlation between in soil radon fluctuations and the geodynamic activity of the volcano. This conclusion is even more reliable because during the monitoring period there haven't been drastic changes in the meteorological parameters, which could affect the radon fluctuations of purely geodynamic nature. From these reasons, it became possible to assume that both the in soil and indoor radon concentration fluctuations, recorded and presented in this study, may be associated to the geodynamic activity of the volcano.

Simultaneously with in soil radon monitoring, indoor measurements have been effected, through another AlphaGUARD in modality of natural gaseous spread, put in a confined little room located at the ground floor of an ancient uninhabited building, lacking of windows, and located at a distance of about 7 meters from the in-soil operating instrument. There is a 15-20 cm thickness of concrete separating the room from the soil; the building walls are constituted of lava blocks. This building has been chosen because can be considered as representative of the historical centre of the town.

Indoor radon measurements at St.Venerina have been carried out also in a new building (reinforced concrete) from 12th January to 14th April 2006. In particular measurements have been performed in a basement of one edifice far about 50 meters from the site where in-soil active monitor was operating. It was built with the most current seismic criteria after the famous earthquake of magnitude 4.4 (Richter scale) and intensity VIII (MCS scale) in October 29, 2002, previously described. This dwelling can be considered as representative of the new neighborhoods of the town. In this new site indoor radon has been acquired using ORTEC Charcoal Canisters (CC) technique, following the EPA protocol (Gray and Windham, 1987).

The CC, preliminary weighed, had to be exposed, uncovered, in the confined place for 48 hours. The canister gamma activity (the activity of ^{214}Bi and ^{214}Pb radon daughters) was measured by means of NaI (Tl) scintillation spectrometer. The typical time between the end of the exposition and the beginning of the measurements was about four hours, which is the time needed to reach the transient equilibrium between parent and γ-emitted daughters. The concentration values expressed in Bq/m^3 were determined through the spectral analysis, exposure time and calibration factor known. Humidity sensors, used for all the investigation period to verify the reliability of the CC methodology (Gervino et al., 2004), have ensured that the humidity was always below 70% throughout the monitoring period.

Results and discussion

Both in soil and indoor radon measurements are shown in figure 2 a, b:

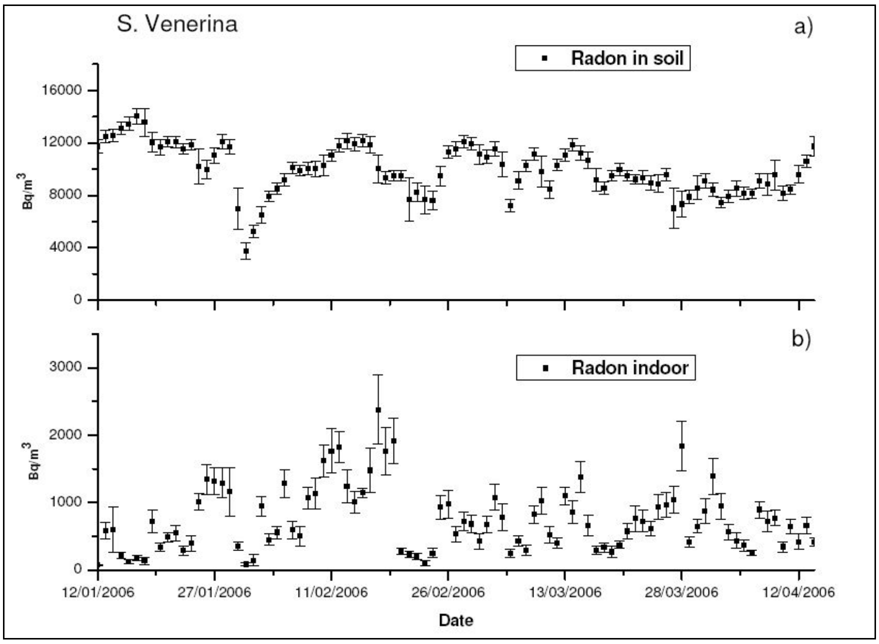

Fig. 2. Both in soil and indoor (old building) radon concentrations.

As it's possible to see, the two trends seems strictly correlated. The indoor meteorological parameters have been also recorded for all the considered period. They are shown in figure 3 as normalized cumulative curves:

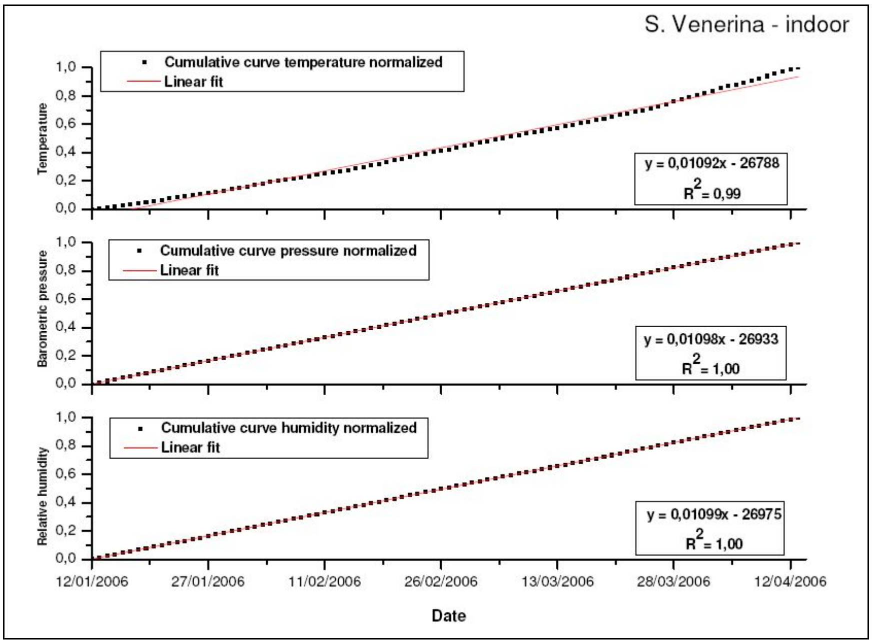

Fig. 3. Normalized cumulative curves of the indoor meteorological parameters.

During the monitoring period, no remarkable variations of meteorological parameters have been observed ($R^2 \geq 0.99$). As it's possible to see, only at the end of the investigation period and at the same time, a little temperature increase and a little humidity decrease have been noticed. These gradual variations are typical of the following spring arrival.

A ***cross-correlation*** analysis was carried out. It is a standard method of estimating the degree to which two series are correlated. Considering the two series the cross correlation r at delay d is defined as the following function:

$$r(d) = \frac{\sum_i [(x(i) - mx) \cdot (y(i-d) - my)]}{\sqrt{\sum_i (x(i) - mx)^2} \sqrt{\sum_i (y(i-d) - my)^2}}$$

Where the two series are indicated with *x(i)* and *y(i)* respectively; *mx* and *my* are the arithmetic means of the corresponding series; $d = 1, 2, ..., N-1$ is the delay in days (the two series represent daily trends).

Result of the cross-correlation is shown in figure 4:

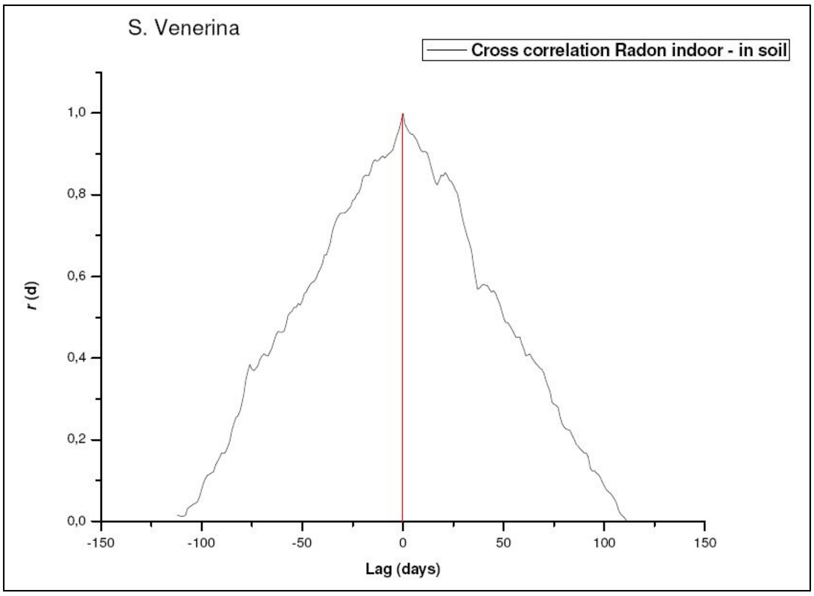

Fig. 4. Normalized cross-correlation between in soil and indoor (old building) concentrations.

As it's possible to see, the maximum correlation is for *d*=0, indicating that indoor and soil radon concentrations have the maximum correlation without delay. Therefore, in old buildings, radon quickly migrates from soil to indoor, reaching values even exceeding those recommended.

Both the indoor radon concentration measurements, that is those detected through the active device AlphaGuard, and those detected through CC, are shown in figure 5 a, b:

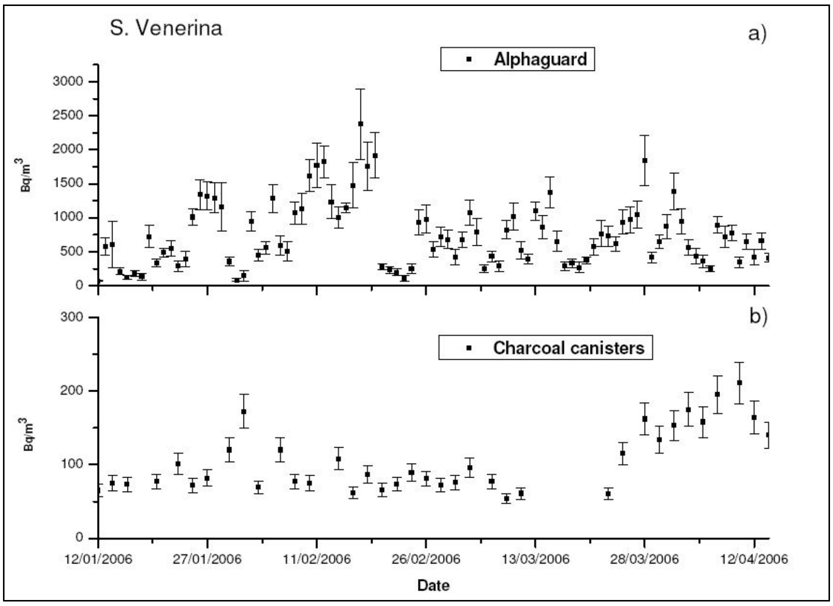

Fig. 5. Both the indoor radon concentration measurements: a) old building, through active device AlphaGUARD; b) new building, through CC detectors.

As previously made for the old building, a cross correlation analysis was carried out. The result obtained is shown in figure 6:

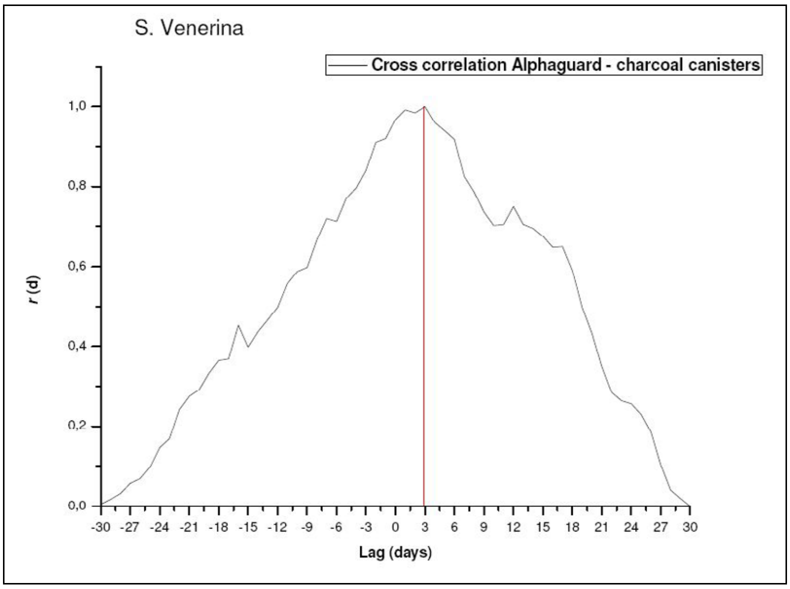

Fig. 6. Normalized cross-correlation between both indoor radon concentrations.

Though one might expect that the new building is able to effectively disperse the radon gas from the subsurface surrounding the building, again in this case, the two trends show a maximum correlation, with a shift of three days, that is inside the radon half-life.

Conclusions

The analysis of rock samples collected throughout the Etna area, have showed that the concentration of primordial radionuclides produce a Radio equivalent activities within the maximum value of 370 Bq/m^3. The values obtained represent the contribution of natural background radiation due to the lithology of the area. These values don't explain both in soil and indoor radon concentration fluctuations recorded in previous studies and, in particular, in the monitoring period considered in this study. It was verified by La Delfa et al., 2007 that the in soil radon fluctuations, showed in figure 2a, are closely linked to geodynamic of the volcano Etna, and this result is even more reliable, because the weather conditions had not undergone drastic changes for all the time interval considered.

Simultaneously with the in soil radon monitoring, indoor radon concentrations in two different types of dwellings have been monitored (figures 2b and 5b), together with the indoor meteorological parameters (figure 3). The indoor weather parameters have stable trends throughout the monitoring period. The cross-correlation analysis have pointed out that the indoor radon concentrations measured, both in the old and in the new building, show the same fluctuations of the in soil radon concentrations. While in the old building, the maximum correlation is without temporal delay, in the new buiding has been obtained a temporal delay of 3 days, however inside the radon half-life. These results show that in active volcano-tectonic areas such as St. Venerina, the geodynamics of the area is strictly correlated also with the indoor radon concentrations, and it seems to be the main external factor of indoor accumulation, in periods of stable weather conditions.

This study wants to point out that in geodynamic settings such as the Etnean area, the indoor radon monitorings (effected to estimate the population exposure to radon gas), which are usually made following a criterion exclusively based on seasonality and floor numbers, should be made considering also the geodynamics of the area of interest, possibly extending the analysis in periods of significant seismic and volcanic activities.

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Influence of environmental factor on radon emanation coefficient

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Abstract

Radon gas is generated from radium in soil and rocks. Radon emanation coefficient is regarded as an important factor affecting the generation of radon from soil and rocks and it depends on various environmental factors such as soil particle size, temperature and moisture saturation. In this study, radon emanation coefficient of Japanese soil samples is measured by accumulation method. We have examined experimentally the dependency of radon emanation coefficient on different soil particle size (from < 106 ƒÊm to > 1000 ƒÊm), temperature (5, 25, 40, 55 ‹C) and moisture saturation (from 0 to 0.9).

Evaluation of radon risk using GIS (Geographic Information System) techniques in the Central Hungarian Region

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Abstract

A new technique of Wattananikorn et al. (2008) for radon potential mapping based on airborne uranium concentration maps was tested in the Central Hungarian Region. Airborne uranium maps (Tyhomirov, 1965-66 digitalized by Boros, 2009) covering eastern parts of the area were processed and analyzed by standard GIS tools. Soil gas radon concentrations were estimated (Wattananikorn et al., 2008) as well as prospective indoor risk values (Kemski et al., 2001). After the spatial analysis of these data with population maps, the results show different radon risk categories at all settlements (Kohli et al., 1997). GIS analyses were validated by in situ soil gas radon measurements and laboratory analyses of soil samples, as well as radon concentration data in indoor areas. Enough good results were gained proving that tested radon potential estimation method is applicable mainly for selecting settlements needed extended research.

Introduction and objective

Radon (^{222}Rn) is responsible for 48% of the total annual effective dose from natural sources of a man living in average environment (UNSCEAR, 2000). The inhalation of this radioactive isotope and its short lived daughters may cause lung cancer potentially (e.g. Bochicchio, 2008 and references therein).

Generally, mapping of radon risk is based on in-situ soil gas concentration (e.g. Gates & Gundersen, 1992; Kemski et al., 2001) or indoor concentration measurements (e.g. Minda et al., 2009). Both methods are expensive and time-consuming to carry out on extended areas but a crude estimation method for radon risk also exists.

Our main aim was modifying and testing this cost-effective and concise procedure which was published recently (Wattananikorn et al., 2008). Other objective was the classification of all settlements into categories based on the results of tested radon risk mapping/estimation method for gaining which settlements need more extended research.

The studied area

Central Hungarian Region is the most populated area and cultural, educational, financial and political centre of Hungary (Fig. 1A). Hence the idea to predict potentially health deteriorating impacts in the area is widely acceptable. An airborne uranium concentration map (Tyhomirov, 1965-66) of the northern part of the region is available (Fig. 1B) from the middle of the 60's. The digitalized version of this map (Boros, 2009) means the base of the present work.

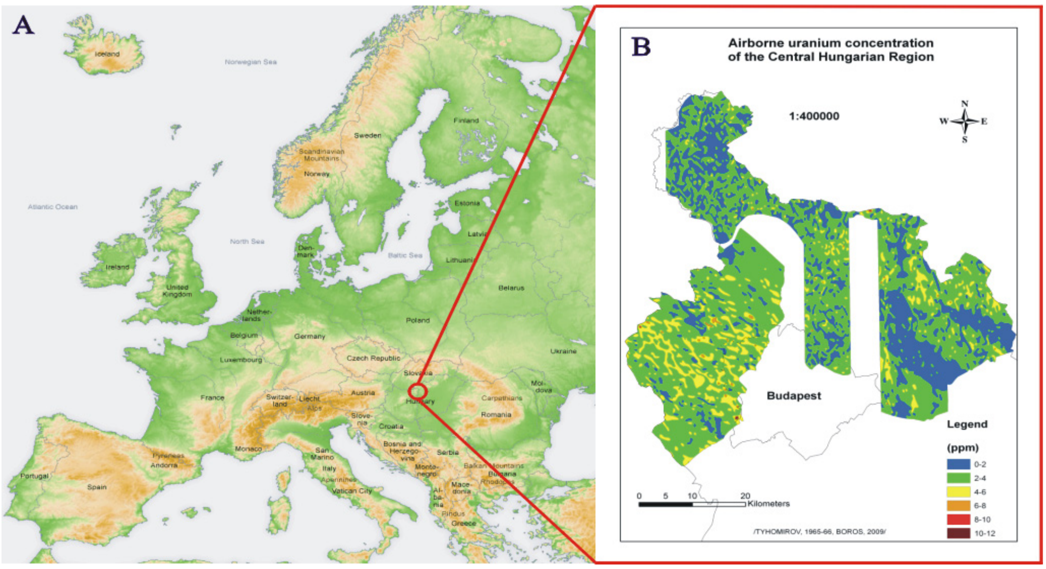

Fig. 1A. Location of Central Hungarian Region in Europe and 1B. Airborne uranium concentration map on the northern part (TYHOMIROV, 1965-66 digitalized by BOROS, 2009).

Methods

Basis of present work is GIS (Geographic Information System) analysis of different databases. The applied software is the ArcGIS 9.3, which can be used for making vector-based maps with geostatistic correction. It also shows the correlation between evaluated and measured databases originated from the following methods (1-4. methods). The complex data processing makes possible testing the estimation of soil gas radon concentration as well as indoor values.

Evaluation of data

1. Soil gas radon concentration was evaluated based on the airborne uranium concentration map of the studied area (Fig. 1B) (Tyhomirov, 1965-66 digitalized by Boros, 2009) and *Eq. 1.* which was published by Wattananikorn et al. (2008).

$$Eq.\ 1.:\ N = 1.92\ u,$$

where N is the predicted soil radon concentration in kBq/m^3 and u is the airborne equivalent uranium in ppm. It is based on measurements in Thailand at 14 areas of 7 test sites of three basins which are covered mainly by Quaternary alluvial and terrace deposits. Correlation between soil radon concentration and airborne equivalent uranium was determined considering the standard deviation of measured values. Hence the upper limit of soil radon concentration can be estimated by *Eq. 1.*. We assumed that this equitation with little changes can be valid in Central Hungarian Region too, although differences in rock type occur. Since differences in permeability values were neglected only a crude estimation can be presented.

2. From the calculated soil gas radon concentration distribution prospective indoor risk values were estimated using *Eq. 2.* (Kemski et al., 2001; Wattananikorn et al., 2008).

$$\textit{Eq. 2.: Indoor radon} = 1.91 \textit{ Soil radon} - 3.3,$$

where indoor radon is in Bq/m^3 and soil radon is in kBq/m^3. Its basis is studying of relationship between maximum ground floor indoor radon and soil radon concentrations in Germany. However, construction features and the age of houses should be also taken into account in further work.

After a spatial and geostatistical analysis of estimated indoor data the results show different categories at all settlements on the settlement map (after Kohli et al., 1997). Most hazardous settlements and distribution type can be studied averaging the values in polygons of settlements' areas (from AGROTOPO digital database) and weighting them by their size. More details on 1. and 2. methods can be found in the references.

Method testing

3. Data on constructed soil gas radon concentration map was compared by data at 28 sites of two types of direct measurements in interest of testing the estimation method based on airborne uranium concentration data. These two types of methods signed by a and b can be found below.

a, In-situ soil gas radon concentration measurements were done using RAD7 radon detector and soil probe at an average depth of 75 cm (according to Gates & Gundersen, 1992).

b, Soil gas radon concentrations were evaluated based on radon mass exhalation rate (similar like Sakoda, 2008) and density, porosity measurements in laboratory by using Eq. 3. (Szabó, K. Zs., 2009).

$$\textit{Eq. 3.: Soil radon} = 100\, E_{mass}\, \rho \,/\, 1000\, p\, \lambda,$$

where soil radon is in kBq/m^3, E_{mass} is the radon mass exhalation rate in Bq/kg/s, ρ is the density of soil sample in kg/m^3, p is the porosity in %, λ is the decay constant of radon in 1/s. A parallel work is supporting the possibility of using this method for soil gas radon concentration estimations (Szabó, Zs., 2009).

4. Indoor radon concentration data were also tested by a few monthly long measurements with Radosys track etched detectors.

Results and discussion

Soil gas radon concentration

Estimated soil gas radon concentration map (1. method) and values of test measurements (3a. and 3b. methods) are presented on Fig. 2.

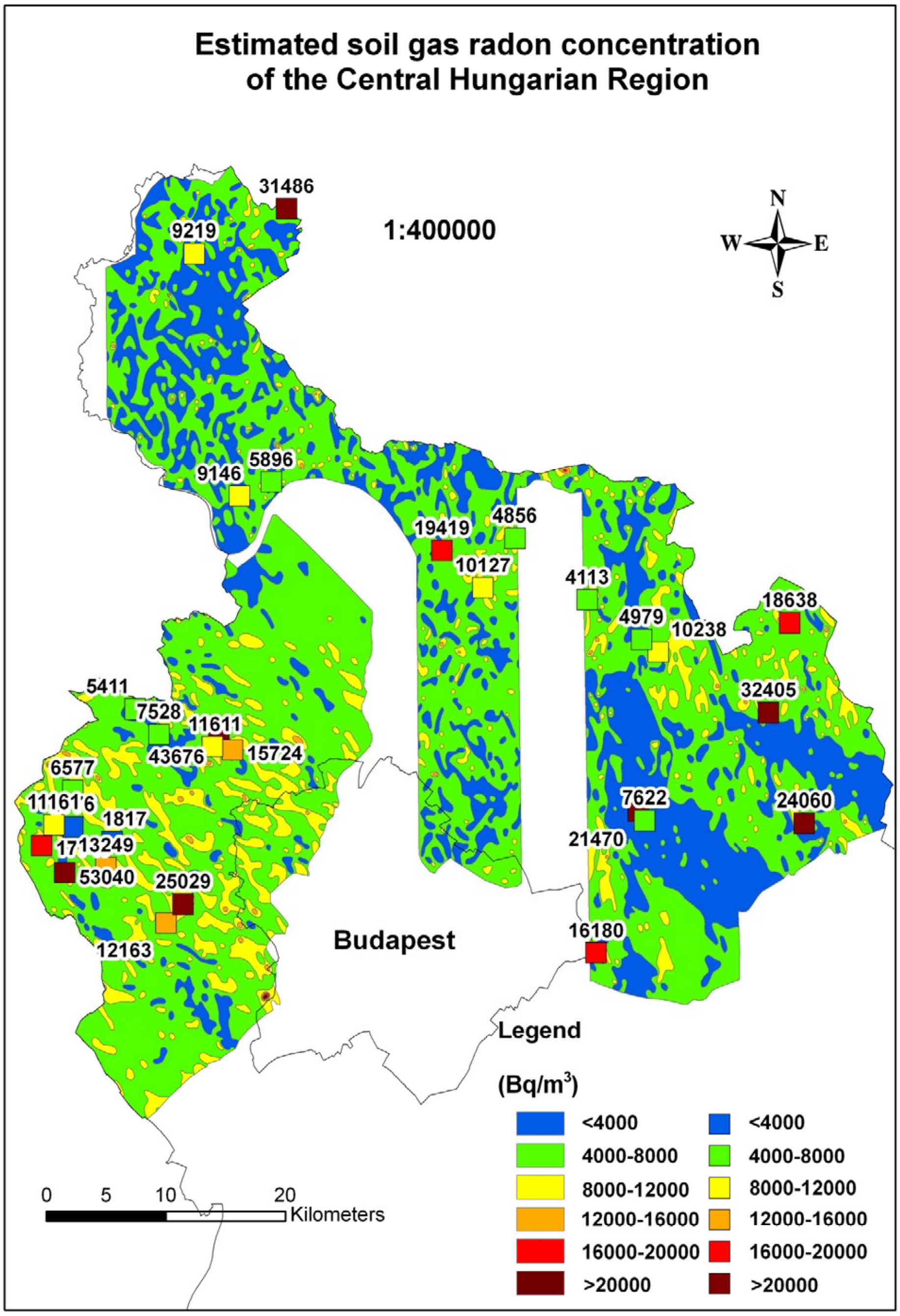

Fig. 2. Constructed soil gas radon concentration map showing the testing points.

The map (Fig. 2.) indicates that the high risk areas are mainly on the limestone and dolomite hills on west side of the region. These areas having higher estimated values of radon concentration should be first explored direct measurements to obtain answer whether they have high risk due to radon. However results of direct measurements show that other areas also can have higher values (e.g. Pesti Plane, Gödöllői hills).

Correlation between estimated and measured values is 0.6 - 0.7 indicating that calculation based on airborne uranium concentration ignores a lot of affecting parameters but can be applied as a crude estimation. However, note that local anomalies cannot be determined.

These results also suggest that a new equation (*Eq. 4.*) should be used instead of *Eq. 1.* in this area (different geological relation) because of the occurred underestimation presumably originated from the differences between permeability values.

$$Eq.\ 4.: N = 2.12\ u,$$

where N is the predicted soil radon concentration in kBq/m^3 and u is airborne equivalent uranium in ppm. Using this equitation most calculated soil gas radon concentration values will be closer to the measured values giving the best estimation.

Indoor radon concentration

Estimated indoor radon risks were calculated for each settlement (2. method). Results are summarized in a histogram on Fig. 3. Number of columns was selected as the *ln[number of settlements (n=95)]*.

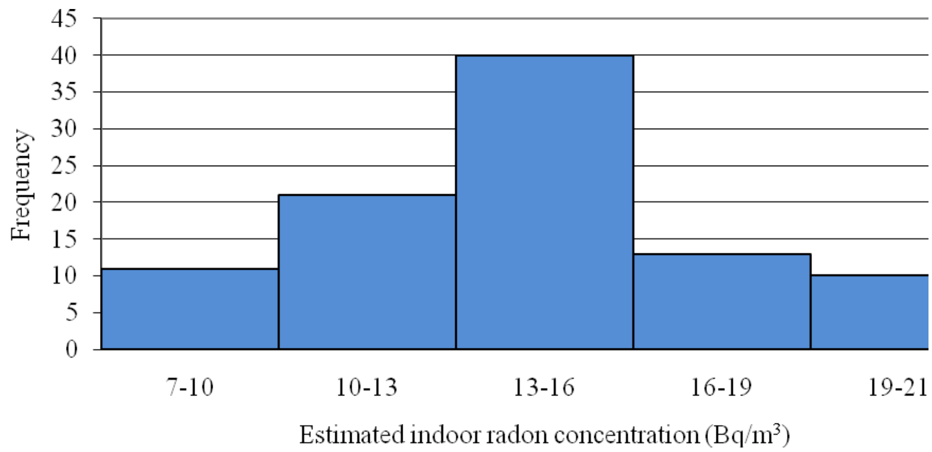

Fig. 3. Histogram of evaluated indoor radon concentrations in dwellings averaged by settlements (n=95).

Values of estimated indoor radon concentration are very low. The procedure has a probability to underestimate indoor radon here as indoor test measurements also indicate (4. method). More direct measurements are necessary to build up a new equitation instead of *Eq. 2.*.

Results originated from estimation indicate a normal distribution (Fig. 3.). This mainly refers to that average airborne uranium concentrations below settlements have a normal distribution and there are not significant average values for one settlement in the Central Hungarian Region. It is again emphasized that it is necessary to determine the realtionship between soil gas and indoor radon concentration at this area too, hence further measurements are needed.

Conclusions

1. A cost-effective method for radon risk estimation based on airborne uranium maps was modified and tested with direct measurements in the Central Hungarian Region.
2. Based on the evaluation high risk areas are mainly on the west side of the region.
3. Since local anomalies cannot be predicted, a correlation of 0.6-0.7 between estimated and measured soil gas radon values is quite reasonable. The calculation based on airborne uranium concentration ignores a lot of affecting parameters but can be applied as a crude estimation. A new equitation (*Eq. 4.*) was also built up for this area.
4. Based on the constructed soil gas radon concentration map, it is possible to select settlements for extended research.
5. Values of estimated indoor radon concentrations are very low. The procedure has a probability to underestimate indoor radon.
6. Average airborne uranium concentrations below settlements have a normal distribution.
7. Further testing and corrections of the procedure are needed.
8. GIS techniques are available and well applicable in radon risk estimation.

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Deposition and clearance of inhaled radon progenies in the central airways

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Abstract
Most of lung cancers of former uranium miners developed in the central airways. Current computational fluid dynamics calculations indicate high local deposition densities in this region. However, cellular burdens of radon progenies deposited in the deeper regions of the lung and cleared up by the mucociliary escalator may also contribute to the health effects in the large central airways. The surface of airway generations increases with airway generation number, thus, it looks a reasonable supposition that the dose of the deeply deposited and up clearing radon progenies may significantly contribute to the total dose in the large airways.

In this work, the deposition distribution of inhaled radon progenies in the respiratory system was computed by the newest version of the stochastic lung model. A clearance model was developed to simulate the up clearing fractions of attached and unattached radon progenies. The stochastic lung model was applied also to compute the surface of the airways. Finally, the surface activity densities of the primarily deposited and upcleared fractions have been calculated and compared at airway generation level for different breathing patterns.

The main input data of the clearance model are the followings: deposition data, velocity of the mucus in each airway generation, surface area of the airways and the half lives of radon progenies.

Based on the results, the radiation burden per unit surface area in the central airways is the highest in the first few airway generations mostly due to the deeply deposited and up cleared radon progenies. Comparing the amounts of the primary deposited and the up cleared radon progenies in the large airways one can conclude that the burden of the up cleared fraction is higher than that of the primary deposition. The results suggest that a possible reason for the preferential occurrence of radon induced lung cancer in the central airways may be, besides the primary deposition, the contribution of the up clearing radon progenies.

Introduction

Large scale histopathological studies have demonstrated that radon induced lung carcinomas of former uranium miners occurred predominantly in the large bronchi, especially in airway generations 3-5. Computational fluid dynamics calculations yielded high primary deposition density values in this region of the thoracic airways. At the same time, the dose contribution of the deeply deposited and up clearing radon progenies in these large bronchial airways is usually neglected. By the increase of generation number the total surface of airway generations highly increases and the deposition efficiency does not decrease remarkably with generation number, thus the dose contribution of the up clearing radon progenies may be significant in the large airways.

The objective of this study is to compute the generation number specific primarily deposited and the up cleared particle fractions to establish their relative contribution to the resultant surface activity in the large bronchi.

Methods

In the present work, the primary deposition distributions of the inhaled radon progenies were computed in the whole respiratory system by the newest version of stochastic lung deposition model (SLM) originally developed by Koblinger and Hofmann (1990). Particle deposition simulations were performed at four different breathing conditions characteristic of sleeping, sitting, light exercise and heavy exercise physical activities. Functional residual capacity, tidal volume, and breathing cycle times characteristic of an adult man during the four types of activities were derived from ICRP66 (1994) publication.

A new bronchial clearance model has been elaborated to simulate the up clearing fractions of particles deposited in the deeper airway generations. Particles trapped by the high viscosity mucus were assumed to move together with the mucus layer. Mucus velocity values were derived from the available literature. Based on the work of Chopra (1979), the tracheal mucus velocity was supposed to be 1.5 cm/min. Mucus velocity in each generation was 2/3 of the mucus velocity in the preceding higher order airway generation. Based on the ICRP66 data, the halftimes of the transport of the deposited attached radon progenies from bronchial airways to blood is 10 hours. This mechanism was also built into the model.

Figure 1 presents the velocity of the mucus as a function of generation number. In this study only the deposition of ^{214}Pb and the clearance of ^{214}Pb and ^{214}Bi isotopes were simulated, where ^{214}Bi is the decay product of the primarily deposited ^{214}Pb isotope. The applied particle diameter is considered to be 200 nm which is the most frequent aerodynamic diameter in the number distribution of the environment aerosols (Haninger, 1997). The unattached fraction of ^{214}Pb is negligible (Marsh et al. 2008). Regarding attached radon progenies the number distribution is the relevant because usually zero or one radon progeny attaches to an aerosol particle. The ^{214}Pb and ^{214}Bi radioisotopes decay and result in ^{214}Po, which is alpha active with a very short half-life. Finally, the ratio of the up cleared and primarily deposited activity densities has been calculated in each airway generation at the above mentioned four breathing patterns. The main input data of the clearance model are the deposition data, the velocity-distribution of mucus, the airway lengths and the half-lives of radon progenies. The average lengths and the surfaces of the bronchial airways (Table 1) were taken from the newest version of stochastic lung model (Balásházy et al. 2009).

Table 1. The average lengths and total surface areas of the airways in airway generations 1-20 based on the newest version of the stochastic lung model.

Generation number	Length (cm)	Surface (cm^2)	Generation number	Length (cm)	Surface (cm^2)
1		68.88	11		277.6
2		22.35	12		358.5
3		21.21	13		459.3
4		28.85	14		581.1
5		43.36	15		726.8
6		56.4	16		931.5
7		80.28	17		1144
8		117.5	18		1301
9		159.2	19		1408
10		215.7	20		1446

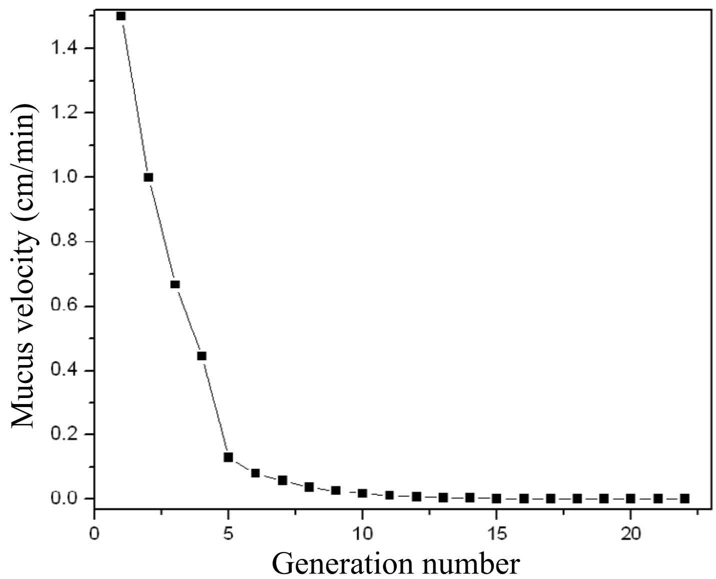

Fig. 1. Mucus velocity as a function of airway generation number.

Results

Figures 2–5 summarise the results of the surface decay density distribution computations. The upper panels refer to the primary deposition, while the lower ones depict the decay density distribution of the up clearing ^{214}Pb and ^{214}Bi isotopes at sleeping, sitting, light exercise and heavy exercise breathing conditions, respectively. The number of selected inhaled particles was always 10^5 at the computation of the deposition fraction distributions.

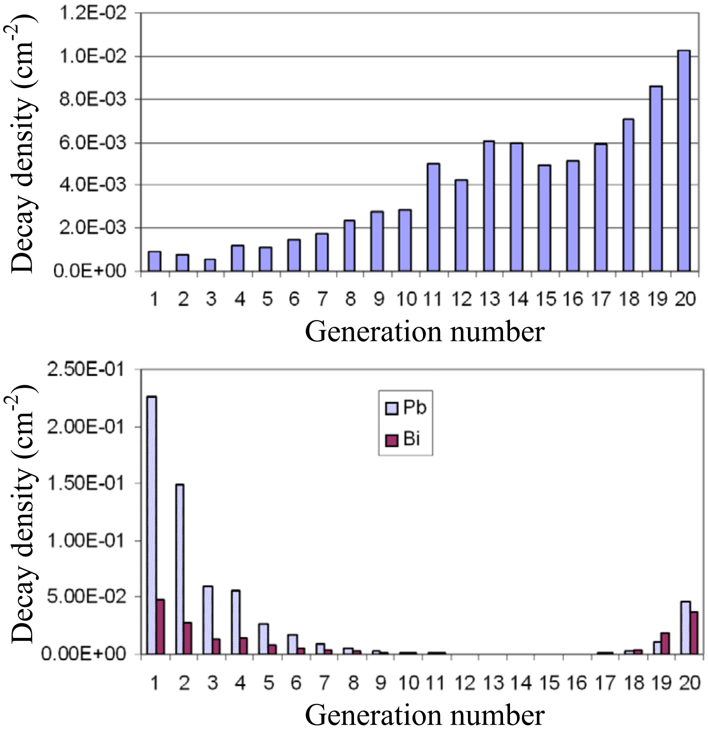

Fig. 2. Decay densities as a function of airway generation number in case of the primarily deposited (upper panel) and up cleared (bottom panel) radon progenies during sleeping.

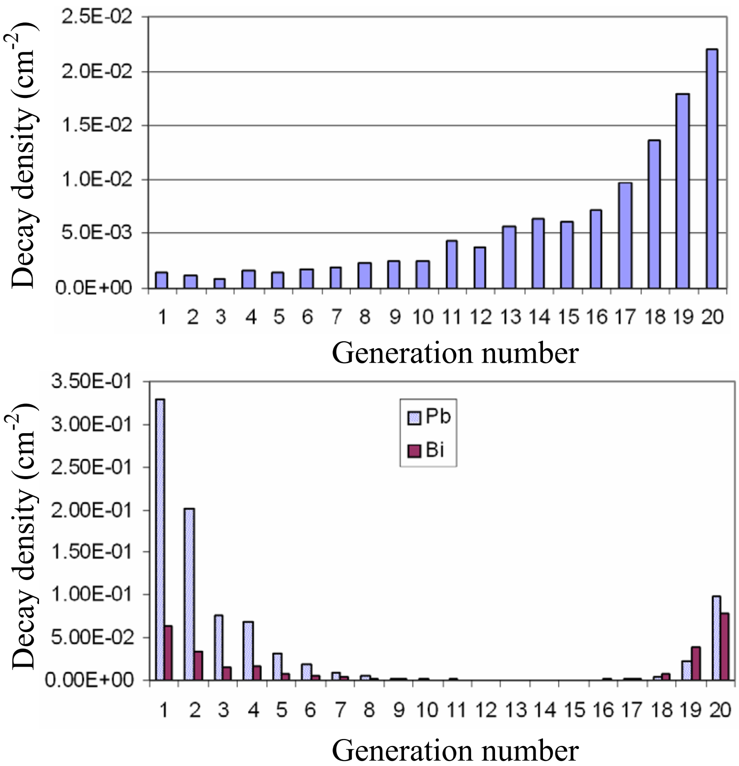

Fig. 3. Decay densities as a function of airway generation number in case of the primarily deposited (upper panel) and up cleared (bottom panel) radon progenies during sitting.

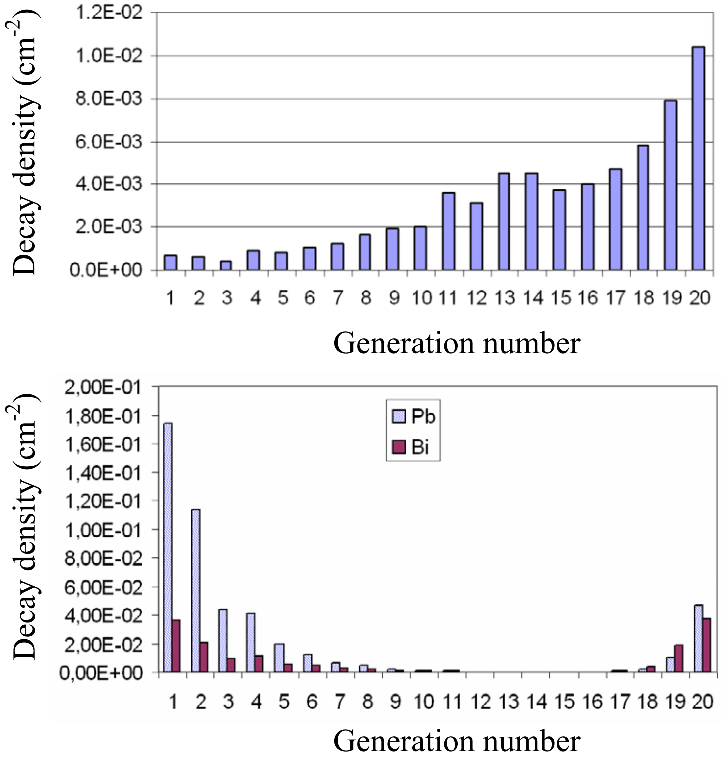

Fig. 4. Decay densities as a function of airway generation number in case of primarily deposited (upper panel) and up cleared (bottom panel) radon progenies during light physical exercise.

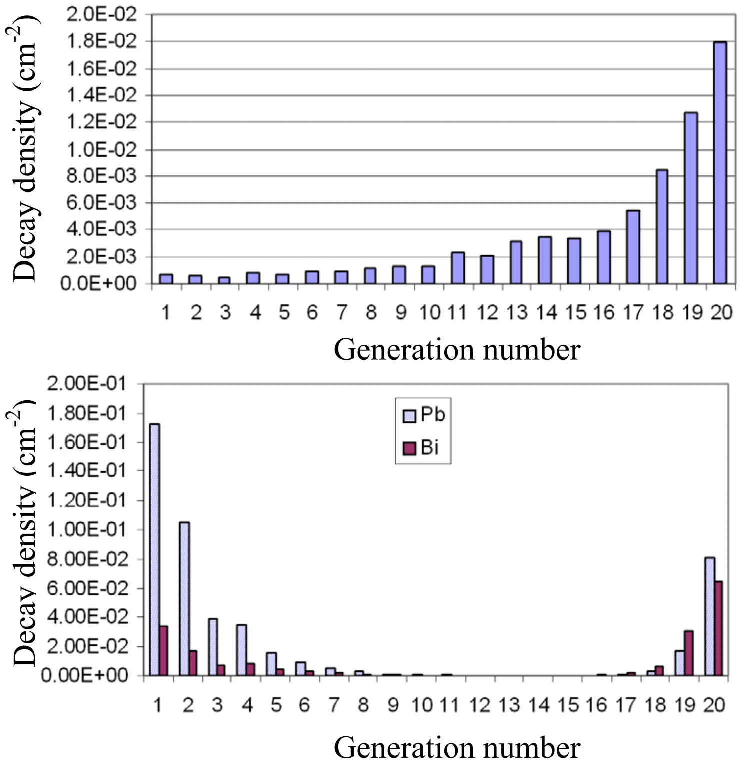

Fig. 5. Decay densities as a function of airway generation number in case of primarily deposited (upper panel) and up cleared (bottom panel) radon progenies during heavy physical exercise.

The figures demonstrate that the decay density due to the primary deposition increases with the increase of the generation number for each type of physical activity.

The decay density originating from the up cleared ^{214}Pb and ^{214}Bi isotopes is significant in the first few generations and slightly in the peripherial regions. This tendency holds for all the studied physical activities. The cumulated decay density, provided by the primary deposition and the up clearing fraction, is the highest in the first few generations, that is, in the large bronchi where most of the radon induced preneoplastic and neoplastic lesions were found. Comparing the relative contributions of the primarily deposited and up cleared isotopes it can be stated that in the most exposed region (large bronchi) the up cleared fraction is determinant. This aspect is usually not considered in current computational dosimetry and risk modelling. Present computations may provide useful input for such models and thus contribute to a quantitatively better estimation of radiation exposures and the related health consequences. Since only about 1 % of the total number of radon progenies is not attached to the aerosols in the mine environment and 5-6 % in homes (BEIR VI, 1999), in the current approach the unattached fraction has been neglected.

Conclusions

Based on the present results, in the central airways, the radiation burden of the deeply deposited and up clearing radon progenies can be up to two orders of magnitude higher than the burden of the primarily deposited fraction in this airways. Physical activity,althogh affects the deposition and the related clearance fractions, does not seem to influence the above tendency. The results demonstrate that one of the reasons of the site specific radon induced lung cancer may be the dose contributions of the deeply deposited and up clearing radon progenies.

In our next publication we will comput the deposition and alpha burdens of all the short lived radon progenies (unattached: ^{218}Po; attached: ^{218}Po, ^{214}Pb, ^{214}Bi).

Acknowledgements

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Reducing the risk to military personnel from radon gas

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Abstract

Radon is attributable to approximately 1500 deaths a year in the UK alone. It is a naturally occurring radioactive gas that derives from uranium found in rocks and soils, and is therefore a natural environmental hazard. Employers have a legal obligation to assess the radon health risk for all their employees. Recent improvements in the UK radon data set and developing work on the risk of radon gas has made this a priority issue for the UK Ministry of Defence (MOD). The Defence Science and Technology Laboratory (Dstl) were given the challenging task of developing a strategy to address this issue and to implement a monitoring protocol to cover the whole of the MOD estate (thousands of locations) throughout the UK and overseas. The development of this involved:

– Working with and/or obtaining information from the Health Protection Agency and the Health and Safety Executive.
– Utilising the most up-to-date guidance from the Building Research Establishment (BRE) and academia to understand the most appropriate and effective remediation techniques.

The radon monitoring program is now well under way covering thousands of workplaces, with the aim of protecting the health of military personnel now and in to the future. A number of examples are given, including the unusual example of 'working' in caves, where radon levels may be extremely high. A three year research project has been undertaken assessing the radon doses to military personnel in caves in the North Pennines currently used for adventure training activities. The findings from this work not only aid in protecting the personnel directly involved, but also have the potential to inform risks in future caving activities. The data set collected will be invaluable for future theatres of operation and also potential litigation work. Most importantly, the remediation work undertaken will reduce the health risk to MOD employees across the UK and overseas.

An assessment of radon exposure to CR-39 alpha track detectors during postal transit in Ireland

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Abstract
In many countries, including Ireland, radon measurements are carried out by sending radon detectors to and from customers through the standard postal system. In Ireland, radon detectors are left in place for at least 3 months before the customer returns them by post. The detectors used are CR-39 alpha track detectors and they are sent in standard A5 envelopes with no radon-proof membranes or seals. To date, this has been done on the premiss that the level of radon detected during postal transit was insignificant compared with the much longer period in situ in the customers' premises. This study investigates the contribution made by levels of radon in the Irish postal system to detectors in transit. Thirty one volunteers, located throughout the country were identified for participation in the project. Practically every county in Ireland was represented. Two CR-39 detectors were sent to each volunteer in standard issue envelopes. On receipt of the envelopes, the volunteers immediately posted them back using a similar envelope. Detectors posted close to or during weekends are likely to be in the postal system the longest. For this reason the Radiological Protection Institute of Ireland always issues detectors before Friday. However, there is no control over when customers return them. Detectors were posted to volunteers on a Thursday afternoon to ensure the transit time coincided with at least one weekend. Results from this study conclude that there is no significant radon exposure to Cr-39 alpha track detectors during normal postal transit in Ireland.

Introduction
Radon-222 commonly referred to as radon, is a naturally occurring radioactive gas, formed by the radioactive decay of radium-226, a component of the uranium-238 decay series. Uranium is present in soils and rock. Radon is the largest contributor to the annual radiation dose received by the Irish public, accounting for approximately 60% of all radiation exposure in Ireland (Colgan *et al.,* 2008). Radon has been classified as a class 1 carcinogen by the International Agency for Research on Cancer (IARC 1988, 2001). Inhaling radon and its progeny can lead to lung cancer. It is estimated that in Ireland 150-200 people a year die from radon induced lung cancer (RPII & NCRI, 2005).

Ireland's relatively high radon concentrations were identified during the National Radon Survey (NRS) which was carried out between 1992 and 1999 (Fennell *et al.,* 2002). An average indoor radon concentration of 89 Bq/m^3 was calculated for Ireland, based on measurements made in 11,319 randomly selected homes over a 12 month period. A recent survey by the World Health Organisation (WHO) found this national average to be the eighth highest in the world (WHO, 2009).

Current radon measurement practice in the Radiological Protection Institute of Ireland (RPII) is to use CR-39 alpha track detectors over an exposure period of at least three months in conjunction with the application of a seasonal correction factor (Colgan *et al.,* 2008; RPII, 2008). Normal practice is to send two detectors with each measurement pack; one to be placed in a bedroom and one in a living area (RPII, 2008). These radon detectors are sent to and received from customers through the standard postal system. They are sent in standard A5 envelopes with no radon-proof membranes or seals.

The RPII has made a recommendation that radon measurements, and where necessary, remediation at the time of sale and/or purchase of buildings is an appropiate mechanism to reduce the collective dose from radon in Ireland. However, it recognises that it may not always be possible to complete a three month measurement during the conveyancing process. Hence the RPII has undertaken research on the use of shorter screening measurements for these circumstances (Rochford *et al.*, IRPA 2010). With a shorter measurement period a contribution from radon in the postal system could contribute to and influence the final result.

Thus, this research was undertaken to investigate the contribution made by levels of radon in the postal system to detectors in transit.

Material and methods

Thirty-one volunteers, located throughout the country participated in the project. Twenty three of the twenty six counties in Ireland were represented.

Fig. 1. Location of the volunteers for this study.

The detectors were issued and posted out in accordance with normal procedures. However, on this occassion on receipt of the detectors the volunteer was requested to return them immediately to RPII.

When posting detectors to its customers the RPII avoids weekend and public holidays so as to minimise the time the detectors are in transit. However, there is no control over when customers post back the detectors, which could include weekends. To simulate this, detectors were posted from RPII to volunteers on a Thursday afternoon to ensure maximum time in the postal system.

When the detectors were received at RPII they were subject to the normal practice and procedures in use in the laboratory.

The Radon Measurement Service is accredited by the Irish National Accreditation Board (INAB) to the International Organization for Standardisation and International Electrotechnical Commission (2005) standard ISO/IEC 17025. The scope of accreditation covers radon concentration measurement in air using passive CR-39 alpha track etch detectors for an exposure time of three to twelve months and a minimum detectable concentration of 10 Bq/m^3.

Quality control procedures are carried out on all new CR-39 detectors received in the laboratory: 10% of the unexposed detectors are etched. The subsequent counting determines the background track density ($tracks/cm^2$). If the average background track density of detectors from a sheet (approximately 100 CR-39 detectors) is less than 50 $tracks/cm^2$ the sheet is released for use and if this is greater than 50 $tracks/cm^2$ the sheet is discarded. The average track density for a number of detector sheets accepted for use in the RPII is presented in figure 2. In addition, 5% of the new detectors are exposed in a secondary standard radon chamber. The radon source maintains a radon concentration of 3000 Bq m^{-3}. This is continuously monitored by an ATMOS 12dpx, which is calibrated annually by the manufacturer using a working standard that is traceable to the National Institute of Standards and Technology (NIST), USA.

Detectors are chemically etched in 6.25 M sodium hydroxide at 75 ^{0}C for 8 hours, resulting in uniform tracks ranging from 240 to 340 mm^2. Following the chemical etch, the detectors are measured using the laboratory's Image Analysis System (Quantimet).

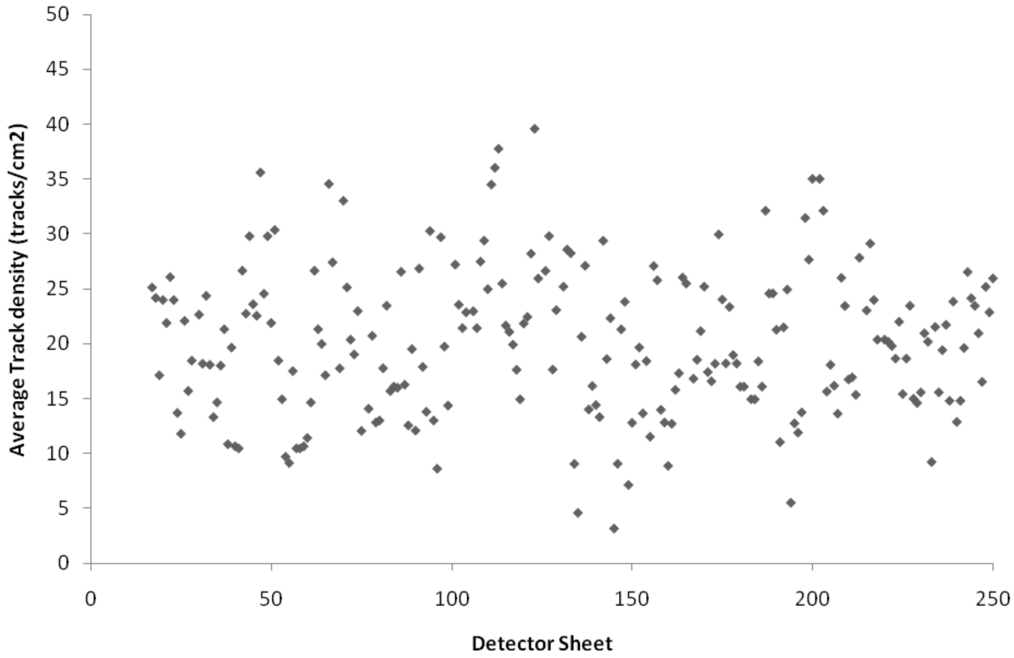

Fig. 2. Average track density of many detectors used by RPII.

Results

It is evident from the data presented in figure 2 that there is considerable variability in background tracks among detectors. The average track density in the set of sheets illustrated in this figure range from 3 to 40 with an average of 21 tracks/cm^2.

The average track density for the two detectors received from each volunteer was calculated and is presented in figure 3.

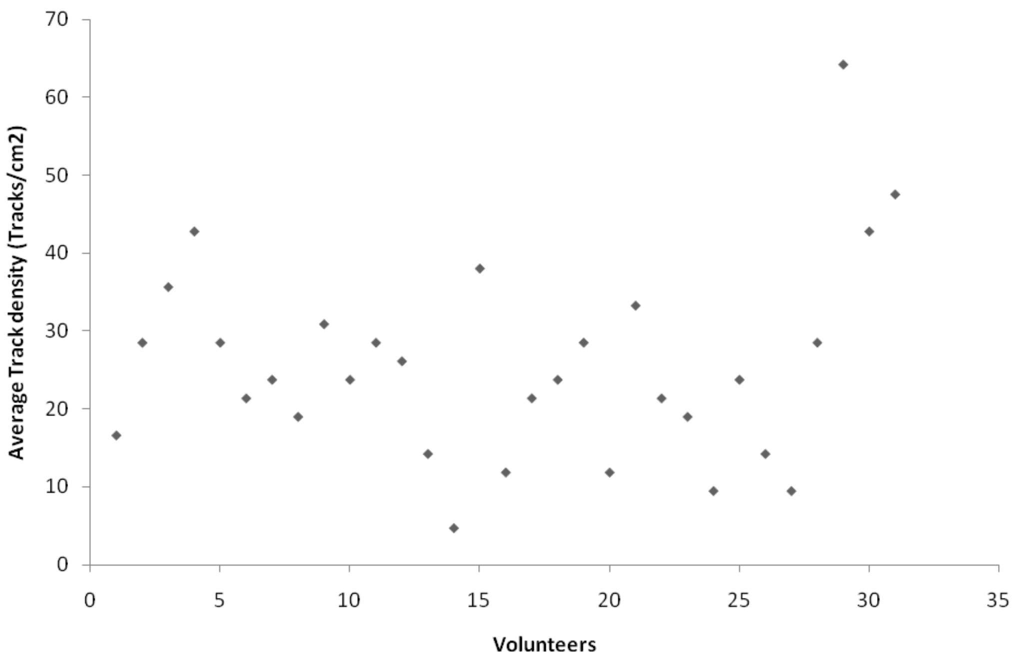

Fig. 3. Average track density of the two detectors.

With the exception of one set of detectors, the average track density for the detectors returned by the volunteers is below 50 tracks/cm^2, the upper limit for background track density at RPII. The range of values is between 5 and 64 tracks/cm^2.

The highest average track density, 64 tracks/cm^2, is from a volunteer address in a county in the west of Ireland. These detectors were in the postal system for six days. This, however, was not the longest transit time.

Two other addresses from this county were also included in this study. One of these recorded the lowest average track density at 5 tracks/cm^2. A similar transit time was recorded for the third detector set from this county which recorded a track density of 29 tracks/cm^2.

The shortest time between detector dispatch and return to RPII for detectors in this study was four days and the longest was seven days. The average time spent in the postal system being five days.

Comparing figures 2 and 3 it can be concluded that there is no significant difference between the detector background track density accepted at RPII and the track density on detectors after a transit time of four to seven days in the postal system.

Conclusions

This study was undertaken to investigate the contribution made by levels of radon in the Irish postal system to detectors in transit.

The result of this study involving thirty one volunteer addresses across the country demonstrates that there is insignificant radon exposure to Cr-39 alpha track detectors during normal postal transit in Ireland.

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Radon areal distribution in Campania region (Italy) inferred from a geostatistic analysis

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Abstract

The territory of Campania region is one the most rich of Radon of Italy and presents prone areas of indoor radon because of the characteristics of the subsoil and of the building materials of local origin which is prevalently volcanic. The national survey on indoor radon classifies this region with a mean concentration higher of 30% than national mean value, but other regional surveys on selected building underline the presence of numerous sites where indoor radon levels are very high. In last fifteen years, our work-group has investigated radon levels within dwellings, schools and workplaces. The principal aim of the present work is to summarize the results of about 700 measurements by using a geostatistic approach which has allowed the realization of a Radon distribution map of the region. Each result is the mean value of an entire year and has been performed by the SSNTDs with LR115 films. Data have been normalized respect to the building material and floor location and, then, located on the map. The lognormal distribution of data has allowed the application of geostatistic techniques using spatial interpolators that not require an uniform distribution of data. The analysis has been performed by the Arcview software which allow the selection of the best variogram and furnishes also the map of the variance of results. Different maps have been extracted from different normalization of data. The maps well describe the real situation according to the experimental results.

Radon mapping strategy in the Republic of Moldova

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Abstract

Radon is a chemically inert radioactive gas. It is formed by the natural radioactive decay of uranium in rock, soil, and water. Naturally existing, low levels of uranium occur widely in Earth's crust. Radon is responsible for the majority of the mean public exposure to ionizing radiations. Constant exposure to high concentration of radon gas may cause lung cancer. Radon gas from natural sources can accumulate in buildings, especially in confined areas such as basements. The radon concentrations in a building are dependent on the concentration of radium in subjacent ground and surrounding soil, the geological bed rock, the radioactivity of building materials, the ventilation conditions, the meteorological conditions and human activities, also. The results of radon concentrations monitoring in the air samples which have been collected in different buildings placed on the territory of the Republic of Moldova during the period of time since 1991 till 2008 years are given in the paper. Investigations have related, that the ^{222}Rn concentrations (92,0...179,1 Bq/m^3) in most cases do not exceed a maximum permissible level. The establishment of ^{222}Rn concentrations in the air samples collected from Cricova storage of wine undergrounds, Chisinau underground galleries, and Milestii Mici, some mines from Orhei, traced out the values of concentrations (200...1800 Bq/m^3) which exceeded the maximum permissible level. Only 2,1% among all investigated houses in territory of Republic of Moldova exceeds the level of 200/Bq/m^3 for radon concentration. The results require the need of radon concentrations monitoring carrying out in dynamics, with the subsequent elaboration of the radon concentrations maps [1].

Reference

Bahnarel I., CoreÅ£chi L., Thomas Streil et. al. Evaluation of exposure risk to radon in the conditions of Republic of Moldova // Scientific Journal of the Academy of sciencis of Moldova Republic, 2007, nr. 4, p. 317–325.

Radon survey in a village close to a former uranium mine (S-W Hungary)

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Abstract

The studied former uranium mine is located at the village of Kovagoszolos (S-Hungary). In the area Permian sandstone layers were found to be rich in uranium for mining, which lasted for 40 years. Kovagoszolos has ~350 family houses. At different distances from the surface projection of a mining tunnel radon concentration was measured in 120 houses with passive detector. The average value of the radon concentration is 483 Bq/m^3. Significantly higher radon concentrations (average 667 Bq/m^3) were measured in houses within 150 m from the surface projection of the mining tunnel, compared with the houses further than the 300 m belt (average 291 Bq/m^3). Nine houses were selected for detailed study: short term indoor radon concentration measurement (RAD 7, AlphaGuard); physical (^{226}Ra content, radon exhalation) and geochemical study on collected soil samples from house gardens. From the nine studied houses, in four particular ones the radon concentration was above the recommended level. The radium content in all cases was higher than the world average value. In the studied soil samples the finest fraction is dominated, potential radon source minerals were considered as Fe-oxides and hydroxides. Among the four chosen houses in one house an extrem radon anomaly was recognized. The indoor radon concentration was measured for one month (by active and passive detectors). Outdoor the gamma dose rate, soil gas radon concentration, radon exhalation and air radon concentration were also determined in the surrounded area of the house. In the center of detected radon anomaly (an area 5 m in diameter) the radon concentration of the outdoor air was 1660 Bq/m^3, the gamma dose rate on the soil surface 380 nGy/h, the radon exhalation 20167 mBq/m^{2}s and the soil gas radon concentration was > 2000 kBq/m^3. Based on these results, it is clear that the local geology plays as important role in the development of a radon anomaly as the distance from the surface projection of the mining tunnel.

Radon survey based on home stored CDs/DVDs

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Abstract

This work presents results from a pilot radon survey based on retrospective measurements by CDs/DVDs. A short description of the CD method for retrospective measurements of integrated radon concentrations is given. The results of the survey are presented, demonstrating that the method allows measurements representative for the risk assessment with uncertainty better than 20 %. The survey includes private homes and public buildings. A large percent of the measurements were organized remotely using disks sent by mail, which clearly demonstrates the serious potential of the method for large-scale radon surveys.

Introduction

At present radon (^{222}Rn) is considered the second most important cause for lung cancer after smoking (WHO, 2009). When the contemporary risk from radon is assessed, it is necessary to reconstruct the historical exposure of the subjects. This goal is targeted by the retrospective methods. The majority of them are based on measurements of the long-lived radon progeny ^{210}Pb/^{210}Po implanted in hard surfaces (Samuelsson, 1988) or trapped in spongy materials (Oberstedt and Vanmarcke, 1996). The first approach gives results that are highly dependent on the complex behaviour of the short-lived radon progenies in the air, while the second involves elaborate and costly radiochemical procedures. A principally different approach for retrospective measurement of radon by compact disks (CDs) was proposed (Pressyanov et al. 1999, 2001). Since then substantial experimental and theoretical work has been devoted to tests of the CD method (Pressyanov 2009, 2010; Pressyanov et al. 2001, 2003, 2004). The aim of this work is to present results from a pilot study organized in Bulgaria that demonstrates the practical potential of the method for large-scale radon surveys.

Description of the CD method

The CD method is based on the remarkable ability of the material from which CDs and DVDs are made to absorb ^{222}Rn. This material is the same type of polycarbonate from which some track-etch detectors for alpha particles (Makrofol®, Lexan®) are produced. The basic idea of the method is to combine the absorption ability and the track-etch properties of the material of the disks. When a disk is exposed to air, ^{222}Rn from the air is absorbed in the polycarbonate (Fig 1.). The alpha particles emitted by the absorbed ^{222}Rn and its short-lived progenies form latent tracks inside the disk. The number of those tracks is proportional to the activity concentration of ^{222}Rn in the air, integrated over the exposure time. However, near the surface, tracks are also formed by ^{222}Rn progenies existent in the air or plated-out on the disk's surface. In such way, a "noise" signal is formed, which is hard to correlate to the activity concentration of ^{222}Rn (due to the complex behaviour of radon progenies in the air). To avoid the "noise" signal, tracks are developed at depths greater than 79 µm below the disk surface. This depth is enough to ensure that no tracks are formed by progenies of ^{222}Rn (and ^{220}Rn) plated-out on the surface or present in the air (Pressyanov et al. 2000).

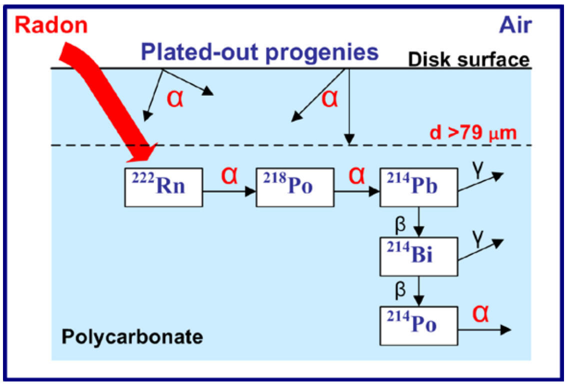


Fig. 1. Principle of the CD method - radon is absorbed in the polycarbonate and the tracks at d ≥ 79 µm are only due to the absorbed ^{222}Rn and its progenies.

At a depth greater than 79 µm below the disk surface the net track density $\boldsymbol{n_0}$ (number of tracks per unit area) is only due to radon absorbed in the polycarbonate and is proportional to the activity concentration of radon in the air $\boldsymbol{A_V}$, integrated over the exposure time $\boldsymbol{T_{exp}}$:

$$n_0 = CF \int_{o}^{T_{\exp}} A_V \, dt \qquad (1)$$

where ***CF*** is a calibration factor. Experimental studies (Pressyanov 2010; Pressyanov et al. 2000, 2003, 2004) have shown that for a given depth of etching the following factors have negligible influence on ***CF***:

- ^{222}Rn progenies in the air or plated-out on the disk surface;
- ^{220}Rn and its progenies;
- ^{210}Po, generated inside the polycarbonate by absorbed radon for reasonable exposure times;
- fading and aging of the polycarbonate material;
- storage of the disks in enclosed (but not hermetic) space, like jewel cases or cabinets;
- cigarette smoke and dust;
- atmospheric pressure in the range 57 – 148 kPa;
- humidity in the range 0 – 100% RH.

In addition, theoretical estimates indicate that no effect on the track density could be expected from the laser light used to write and read data from the disks (Pressyanov 2010).

The only factor known so far to have significant influence on the calibration factor is the ambient temperature. However, as shown in (Pressyanov et al. 2003), a correction for the ambient temperature could be made after the exposure. If such a correction is not applied, the expected bias of the result is within 12% for etching at 80 µm. It is estimated for the interval 10 ÷ 30^0 C and for reference temperature of 20^0 C (Pressyanov 2010). Considering that the average room temperature varies in an even narrower interval, the bias for disks exposed indoors should be even smaller.

The applicable range of the method for 10 years exposure time is from a few $Bq.m^{-3}$ to over 20 $kBq.m^{-3}$ (Mitev et al. 2010; Pressyanov et al. 2003, 2008). The upper range is reached by etching at higher depths, where the track density is smaller. On the other hand, a disk exposed as shortly as one year might give statistically significant signal above 30 $Bq.m^{-3}$. The method is applicable for the whole range of activity concentrations of ^{222}Rn indoors that are of practical interest.

Principle steps of the CD method

Exposure

It is assumed that the exposure continues from the day the disk is obtained by its owner till the day the disk is processed in the laboratory. Any tracks formed before the owner obtains the disk are considered “background”. To estimate an average value for the background track density in CDs and DVDs a large number of new disks were etched. They were bought from different places and were of different brands. The individual background track densities varied from 2 to 40 $tr.cm^{-2}$. The average background track density was estimated at 6,3 $tr.cm^{-2}$ with standard deviation 2,3 $tr.cm^{-2}$.

In order to make retrospective measurements which are representative for the risk from exposure to radon, the analysed disks have to be more than 5 years old. Our experience shows, that in most households disks which are at least 10 years old and whose age could be determined with accuracy better than 1 year are found. However, in case “old” disks are not available, disks that are less than 5 years old are analysed. Such disks could also detect an existing radon problem, but would have limited value for estimates of the long-term past exposure.

Gathering disks

A major advantage of the CD method over other methods for retrospective radon measurements is the fact that no visit to the studied homes is necessary. Disks could be sent by mail, which allows large-scale surveys to be organized very efficiently.

Calibration

Another advantageous feature of the method is the possibility for *a posteriori* calibration. This means that the individual ***CF*** of each disk could be determined after it is taken for analysis (i.e. after the exposure). To do this two pieces are cut from the disk (usually two quarters). One of them is processed right away and the “original” track density in the disk ***n*** is found. The other piece is additionally exposed to known activity concentration of radon. Then ***CF*** is determined as:

$$CF = \frac{n_C - n}{I_V} = \frac{\Delta n}{I_V} \tag{2}$$

where $\boldsymbol{n_C}$ is the track density in the additionally exposed piece and $\boldsymbol{I_V}$ is the integrated activity concentration to which it was exposed. Note that $\boldsymbol{n}$ and $\boldsymbol{n_C}$ should be determined for the same depth.

In our laboratory the calibration is performed using the set-up shown in Fig 2. (including 50 L hermetic box, certified ^{226}Ra source and AlphaGuard® radon monitor). At initial activity concentration of 2 $MBq.m^{-3}$ 10 hours of exposure are enough to provide an increase in the track density of over 400 $tr.cm^{-2}$. After the exposure, the disks are left to degass in radon-free atmosphere for about 4 weeks, so that all absorbed radon could decay or desorb. Further the two pieces are processed together in an identical way.

Any bias due to difference in the properties of the disks is avoided by *a posteriori* calibration. Our experience shows that even disks of the same brand might have somewhat different ***CFs***. The average ***CF*** determined for 34 disks is 0,020 ± 0,03 $cm^{-2}/kBq.h.m^{-3}$. The biggest difference we have observed between the individual ***CF*** of a disk and this value is –45%. That is an argument in favour of the *a posteriori* calibration of each studied disk. The procedure increases the time needed to obtain the final result, but is quite simple and could be applied simultaneously on many disks.

(a) **(b)**

Fig. 2. Set-up used for calibration of CDs. a) Tracebility is provided by a certified ^{226}Ra source and AlphaGuard® radon monitor. (b) The exposure is carried out in 50 L hermetic box.

Pre-etching and electrochemical etching

A surface layer is removed from each disk, so that the tracks at a certain depth below the surface (d ≥ 79 μm) could be developed. This is done by chemical pre-etching (CPE) using an aqueous solution of 52% KOH (m/v) and 40% methanol (m/v) at 30^0 C. At this regime the bulk-etch velocity is about 1 $\mu m.min^{-1}$. The depth of the removed layer is determined by measuring the disk thickness before and after the CPE by micrometer.

The tracks at the studied depth are revealed by electrochemical etching (ECE). The ECE process is performed at effective electric field of 3.0 $kV\ mm^{-1}$ with frequency of 6 kHz. The temperature is 25^0 C. The etching solution is a mixture of ethanol with

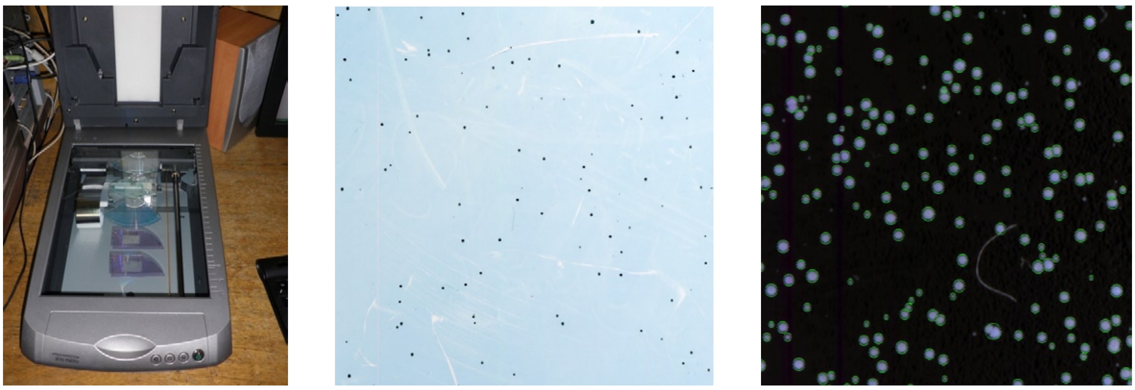
6N KOH solution with 1:4 volume ratio (Vanmarcke and Janssens 1986). The process is preceded by a 30 min pre-etching with the same solution. After this the electric field is applied for 3 hours. The equipment available in our laboratory (shown in Fig. 3.) allows us to process about 30 disk quarters daily.

(a) (b)

Fig. 3. Set-up used for CPE and ECE. a) Disks prepared for etching. (b) Injecting the etching solution (front) and ECE in progress (back).

Track counting

The tracks are counted automatically after the disk is scanned by a commercial computer scanner (Tsankov et al. 2005) as shown in Fig.4(a). An example of a scanned image is shown in Fig. 4(b). A computer program (DGTrack) developed specifically for counting ECE tracks in CDs/DVDs is used to process the image (Mitev et al. 2010). The program has the ability to discard artefacts (like scratches that are common in used disks) and to separate overlapping tracks (as can be seen in Fig.4(c)). Scanning and counting 30 disks by DGTrack takes up to 60 minutes and most of this time is consumed by scanning (Mitev et al. 2010).



(a) (b) (c)

Fig. 4. (a) Scanning CD/DVD pieces for automatic counting of tracks. (b) Scanned image of the etched area of a CD. (c) ECE tracks in a CD counted by DGTrack (Mitev et al. 2010).

Estimating radon activity concentration

After the track density at a certain depth in the disk $\boldsymbol{n}$ is determined along with the corresponding $\boldsymbol{CF}$, the average radon activity concentration for the period for which the disk was stored is found as:

$$A_V = \frac{n_0}{CF.T_{\exp}} = \frac{n - n_b}{n_C - n} \frac{I_V}{T_{\exp}} \tag{3}$$

where $\boldsymbol{T_{exp}}$ is the length of this period (i.e., the age of the disk) and $\boldsymbol{n_b}$ is the background track density. The relative uncertainty of $\boldsymbol{A_V}$, $\boldsymbol{\delta(A_V)}$ is given by:

$$\delta(A_V) = \sqrt{\delta^2(stat) + \delta^2(I_V) + \delta^2(T_{\exp}) + \delta^2(temp) + \delta^2(d)} \tag{4}$$

where the first term accounts for the statistical uncertainty in the track densities, and $\boldsymbol{\delta(I_V)}$ and $\boldsymbol{\delta(T_{exp})}$ are the relative uncertainties of $\boldsymbol{I_V}$ and $\boldsymbol{T_{exp}}$ respectively. The term $\boldsymbol{\delta(temp)}$ accounts for any difference between the temperature of exposure of the original disk and the temperature during the calibration exposure. The term $\boldsymbol{\delta(d)}$ accounts for the uncertainty in the depths of etching – this uncertainty might mask small differences in the depths at which the two pieces are etched. The last two terms in Eq(4) are evaluated theoretically, using a model given in (Pressyanov 2009). The typical and extreme values of the components of the uncertainty of $\boldsymbol{A_V}$ are shown in Table 1.

As it could be seen, in most cases the average radon activity concentration could be determined with uncertainty better than 20%. Higher uncertainty is usually due to the high relative uncertainty of the exposure time when one or two year-old disks are used.

Table 1. Typical and maximum values of the components in the relative uncertainty of radon activity concentration determined by retrospective measurements with CDs/DVDs.

Relative uncertainty	Typical, %	Maximum, %	Notes
δ(stat)	5	10	-
δ(I_V)	5	10	-
δ(T_{exp})	10	50	Maximum – for short exposure times
δ(temp)	12	33	For T = 10 ÷ 30^0 C (Typical at 80 μm and Maximum at 120 μm)
δ(d)	5	10	(Typical at depth uncertainty ±3 μm and Maximum at ±5 μm)
Total (***δ(A_V)***)	18	62	-

Results

A blind test of the method was performed in 2009 in the frames of an international intercomparison of passive radon detectors organised by the Health Protection Agency (HPA), UK. Compact disks were exposed at HPA and were sent to us by mail. A good agreement between the referent values and the results obtained by the CD method was found within the stated uncertainties. The results confirmed that the method could detect elevated radon levels even for short exposure times.

A pilot survey started in Bulgaria aiming to test the potential of the method for large-scale measurements. Around 200 disks were collected from 88 buildings. The disks were provided by their owners and varied in brands and age (from 1 to 15 years). The regions from which disks were collected include the cities Sofia and Varna, the town of Eleshnitsa and the region of the town of Bouchovo. In the last two regions uranium mining and milling had been in progress in the past and they are considered areas with high radon risk. The distribution of the activity concentrations in the studied buildings in these regions are shown in Fig.5(a). Although the number of buildings is small, it is clear that in many of them radon poses a problem. For comparison, the distribution of the activity concentrations in the studied buildings situated in Sofia, Varna and some other locations is presented in Fig.5(b). The obtained average is close to the average reported by UNSCEAR (UNSCEAR 2000). The averages and the medians for the studied buildings in the 3 regions, from which most of the disks were collected, are given in Table 3. The 95% confidence interval (CI) for the median was determined using non-parametric statistical analysis (Conover 1980). Moreover, it should be noted that CDs were taken from some of the homes in Eleshnitsa in which prospective measurements had been made in the past. As shown in (Pressyanov et al. 2010) the results obtained by CDs correlate very well with those obtained by the independent past measurements.

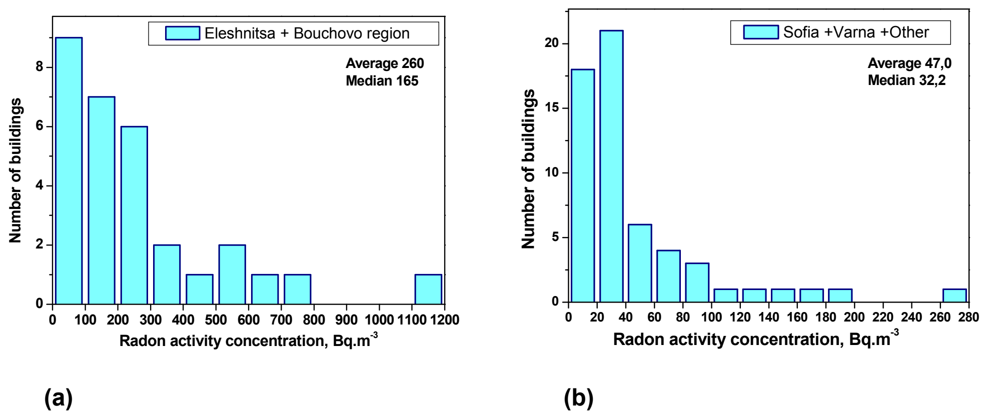


Fig. 5. Distribution of the activity concentration of radon obtained by measurements with CDs/DVDs in buildings in (a) risk areas (Eleshnitsa and in the region of Bouchovo); (b) areas considered "normal" (Sofia, Varna and some other locations).

Table 2. Values of the average and median of the activity concentrations of radon in buildings in three towns in Bulgaria obtained by measurements with CDs/DVDs.

Area	Number of buildings	Radon activity concentration , Bq.m^{-3}	
		Average	Median (95% CI)
Sofia	24	41,1	29,5 (19,0; 38,7)
Varna	17	38,5	17,3 (13,8; 29,7)
Eleshnitsa	22	166,1	123,4 (78,4; 210,0)

Most of the disks used in the survey (75) had been stored in homes, while the rest of them had been stored in different public buildings – four schools, a kindergarten, a hotel and several workplaces. Elevated radon concentrations were found in the kindergarten (about 500 Bq.m^{-3}), which had since been successfully remediated by an active suction system. An interesting result was obtained in one of the other buildings (Fig.6). The analysis of a 3,5 year-old disk (CD1) lead to an estimate for the radon activity concentration about two times higher than the analysis of a 1,9 year-old disk (CD2). It turned out that before the second disk was bought, passive mitigation of the building had been installed. Subtracting the integrated activity concentration estimated by CD2 from the one estimated by CD1 allowed us to determine the average radon activity concentration for the period before the remediation (Fig.6). In this way, using CDs we determined a reduction factor of 3,7 ± 1,4 (at the level of 1σ). For comparison, conventional measurements showed about 3 times reduction in radon concentrations after the remediation. The example illustrates the ability of the CD method to roughly detect substantial changes in radon concentrations. In addition, the same approach could be used to exclude the last few years, which do not contribute to the present risk. This application of the CD method is useful for case-control studies and will be explored further.

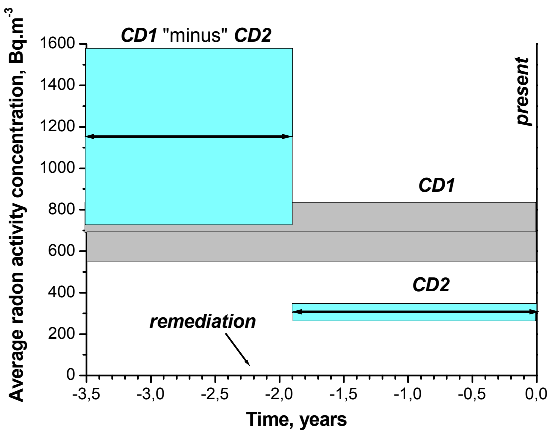


Fig. 6. Estimate of the average radon concentration for a past period by using two disks of different age. Remediation of the building has been made before CD2 was bought. The filled regions mark the estimated concentrations ± 1σ.

Conclusion

The conducted survey, however small, demonstrates that the CD method:

- allows precise retrospective measurement of the activity concentration of ^{222}Rn with uncertainty potentially better than 20 %;
- allows individual calibration of each disk after the exposure;
- could be widely applied – CDs and DVDs are available in homes, work places, schools, etc.;
- could provide results representative for risk estimation – in many households disks as old as 10 ÷ 15 years are available;

- pinpoints buildings with radon problem even with disks less than one year-old;
- could detect significant changes in the activity concentration of ^{222}Rn in the past by etching disks of different age;
- allows to monitor the individual exposure of the occupants even when they change residencies often;
- allows to organise measuring campaigns remotely by asking the participants to send disks by mail;
- is fast, simple and inexpensive.

The above features of the CD method make it appropriate for large-scale radon surveys. The method gives results that are representative for the risk from exposure to radon and could prove useful in case-control studies.

Acknowledgements

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Radon measurement method with passive alpha track detector at STUK, Finland

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Abstract

In Finland, Radiation and Nuclear Safety Authority (STUK) has performed approximately 5 000 – 20 000 indoor radon measurements per year. The detector material is polycarbonate and the holder model is STUK's own. After at least a two-month measurement in the heating period, the polycarbonate films are electrochemically etched in order to see the alpha tracks. Tracks are calculated with image processing and analysis system. Research and Environmental Surveillance of STUK is an accredited testing laboratory T167 (EN ISO/IEC 17025) by the decision of the Finnish Accreditation Service (FINAS). Airborne radon concentration is one of the accredited fields of testing performed at STUK and verified by FINAS. STUK's Health Risks and Radon safety laboratory also makes comparison measurements every second year with primary standard of Physikalisch-Technische Bundesanstalt (PTB), the national metrology institute laboratory in Germany. STUK's Health Risks and Radon Safety laboratory sends its reference measuring device to PTB for calibration. This is to conserve the measurement standard at STUK.

Introduction

The first passive radon detector that STUK used was STUK's own making. It was an open Kodak LR-115 detector, used from 1980 to 1984 (Mäkeläinen, 1984). In 1984 STUK started to use closed passive radon detector. The detector holder was made from clear plastic and the detector was a 300 - µm thick Makrofol polycarbonate film. The other side of the detector was clear and the other side was matt (Mäkeläinen, 1986). In 1989 the detector holder material was replaced with an electric conductive (carbon fibre reinforcement) ABS plastic and 250-µm thick polycarbonate was used as the detector.

Material and methods

The present day passive detector is used for dwellings and workplaces. The detector holder is STUK's own design, closed and with a filter, 17 mm high and 45 mm in diameter. It is made from ABS plastic including 10% carbon fibre. There are six holes in the holder cover, and below the cover there is a paper filter disc, Whatman / Schleicher & Shuell 589/3, diameter 45 mm. STUK's detector is shown in Figure 1.

Figure 1. STUK's detector.

The detector is made of 250 - µm thick polycarbonate sheet, Bayer Makrofol DE 1-1 CC. Both sides of the detector are clear. STUK gets the film directly from factory in 610 x 915 mm sheets, clear cover plastic on one side and green cover plastic on the other side. Finnish company die-cuts the Makrofol to keep the clear cover plastic as one piece and the green one will be removed just before packing it into the detector holder. Cover plastic prevents the detectors from scratching. The background was three tracks per cm^2 for the new detector and today it is five tracks per cm^2. STUK performs control tests for background in every three months. Roughly one out of 5 000 detectors breaks down because of high voltage. Also the track density can vary in one or two per 5 000 detectors so that they must be rejected.

In the detector holder there is a 19-µm thick aluminium-metallized sheet of polyester, Hostaphan RNK 19. The surface resistance for the aluminium coating is 2 ohm/m^2, and the thickness of the aluminium coating is less than 0,05 µm. Inside the holder steel spring keeps the parts in their places. The detector holders, detectors and questionnaires are identified using bar code labels. The bar code type Interlieved 2/5 is suitable for small detectors like in our case.

The pre-etching time is 1 h 12 min and the electrochemical etching time is 2 h 22 min. Etching temperature is 36 ºC, frequency 2 kHz and field 28.8 kV/cm. The etching solution is potassium hydroxide, KOH and ethanol. For the KOH solution we take 385 ml saturated potassium hydroxide and 620 ml distilled water, and compound it with ethanol 1:1. On the other side of the detector, we use sodium chloride solution, 61 g NaOH in 1 l water, as a conductive liquid. Control unit takes care from etching times. Fluke 87 True RMS Multimeter stores ranges of voltage, average, maximum and minimum. Etching chambers in constant temperature cabinet are shown in picture 2.

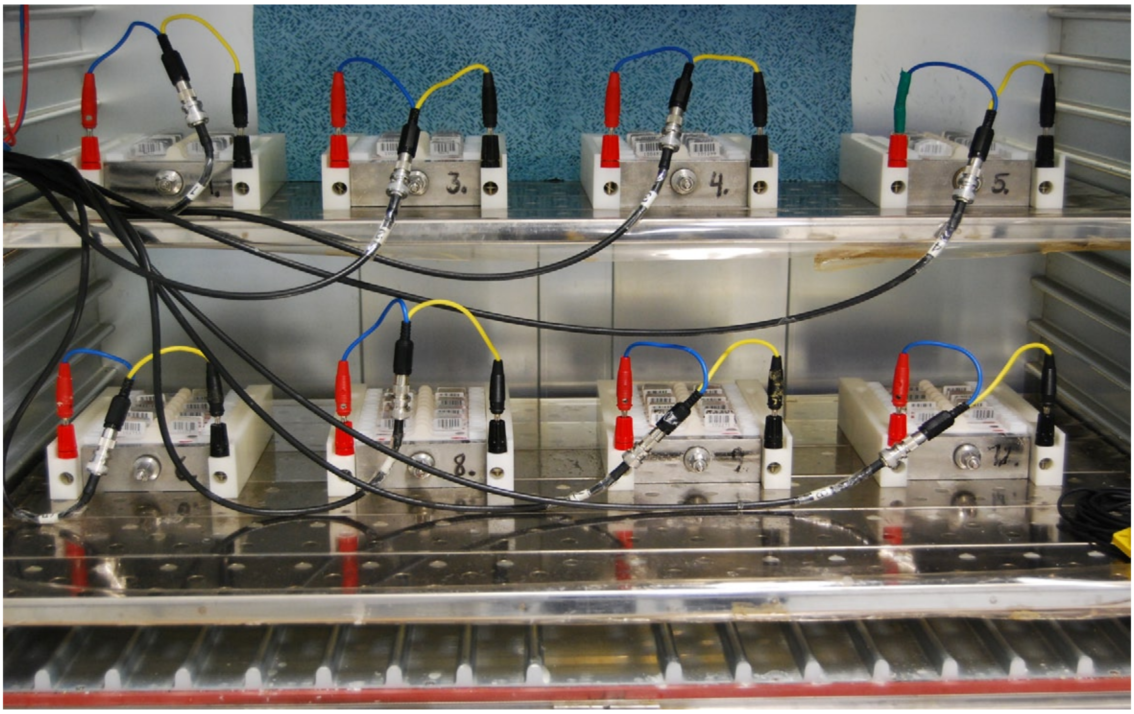

Figure 2. Etching chambers in constant temperature cabinet.

Track counting is made with Leica image analyze system. The software is Qwin, the microscope Mz6 and the camera DC 300. The counting area used is 1.52 cm². We take one picture from the detector and the program will divide it into 16 sectors for calculation. If the amount of tracks differs significantly in more than three sectors, the system rejects the detector for hand control. STUK makes every other month verification to the counting system by re-counting the same test detectors.

The main testing system is to use test detectors. 300 - 600 test detectors are exposed to the same radon concentration. At least every twentieth detector is a test detector and they will go through every stage daily.

Laboratory information system RAMI includes the customer register and the measurement register. RAMI takes care of result calculations, result letters and invoices.

Research and Environmental Surveillance of STUK is an accredited testing laboratory T167 (EN ISO/IEC 17025) by the decision of the Finnish Accreditation Service (FINAS). Airborne radon concentration is one of the accredited fields of testing performed at STUK and verified by FINAS. STUK participates in intercomparisons of passive radon detectors. STUK's Health Risks and Radon Safety laboratory also makes comparison measurements every second year with primary standard of Physikalisch-Technische Bundesanstalt (PTB), the national metrology institute laboratory in Germany. STUK sends its reference measuring device, Alpha Guard, to PTB for calibration. This is to conserve the secondary measurement standard at STUK.

In Finland, Radiation and Nuclear Safety Authority has performed approximately 5 000 – 20 000 indoor radon measurements per year. Amount of annual radon measurements is shown in figure 3.

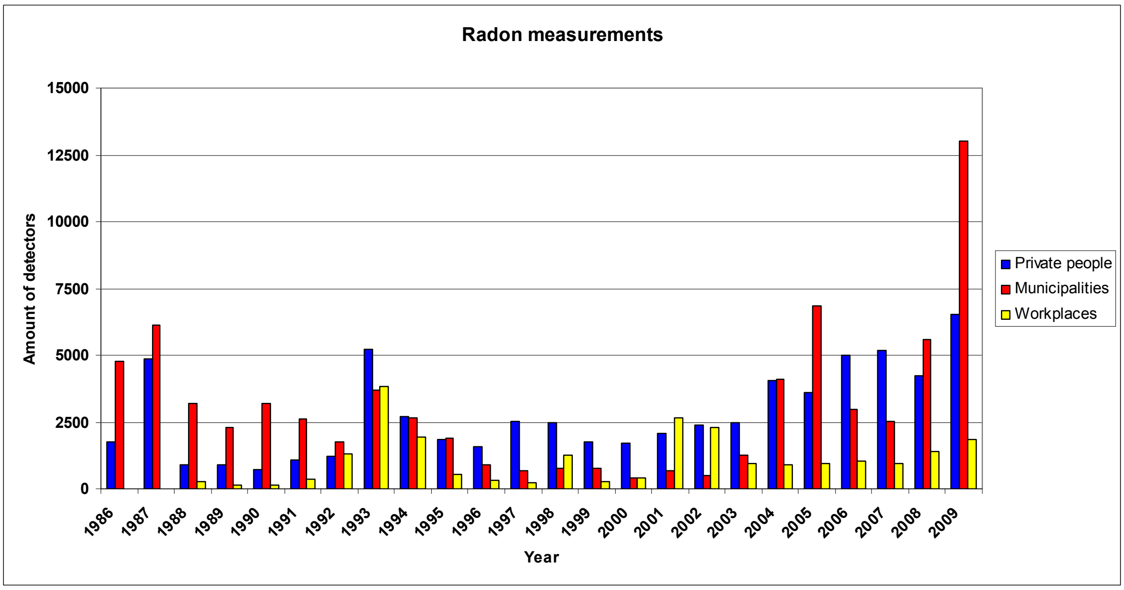

Figure 3. Amount of annual radon measurements in Finland.

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Results obtained in the measurement of Rn-222 with the Romanian standard system

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Abstract
This paper presents the following results: (i) Realization of the metallic radon standard system; (ii) Quantitative extraction of radon from the radium source; (iii) Absolute standardization by liquid scintillation counting and relative measurements of radon activity, by using the HPGe gamma-ray spectrometry method and the reentrant ionization chamber. The final purpose of the work is to calibrate adequately the secondary standard systems, for using them in the measurement of the working standards, glass vials containing radon gas standard, in order to assure the traceability. A method for modeling the radon transport in various matrices was also elaborated, with reference to the simulation of the radon detectors containing active charcoal. It will be validated experimentally, with radon gas standards.

Introduction
The necessity to create a primary Romanian ^{222}Rn standard system and the general concept of such a system to be established at IFIN-HH, Radionuclide Metrology Laboratory were presented at IRPA 12 Congress by Sahagia et al. (Sahagia et al. 2008). A preliminary variant of a glass circuit was realized and used for the extraction of radon. It contains a vacuum system for the circulation and recovery of radon and a Pylon solid source of ^{226}Ra, http://www.pylonelectronics.com/pylonradioactive.php, type RN-1025-250, flow-through, with the activity: $A_{\mathrm{Ra\text{-}226}}$= (259 ± 10)kBq, on 21 August 2008, traceable to the NIST, USA. The method for absolute standardization of ^{222}Rn in equilibrium with its daughters by liquid scintillation counting (LSC) was first used by a common French-Romanian team at the Laboratoire National Henri Becquerel (LNHB), France, such as presented by Cassette et al.(Cassette et al. 2006). The method was developed at IFIN-HH, with some original aspects, referring to the correction for the decay of the very short half life daughter ^{214}Po during the extendable type dead time

of the LS counter. The results obtained in the absolute standardization of radon were published by Sahagia et al.(Sahagia et al. 2009a)

This paper presents the following results:(i) Realization of the final metallic system, provided with all the necessary control gauges and the preparation of sources by the quantitative extraction of radon from the radium source. (ii) Absolute standardization by LSC and relative measurements of radon activity, by using the HPGe gamma–ray spectrometry method and the CENTRONIC IG12/20A ionization chamber. The final purpose of the work will be to calibrate adequately these secondary standard systems for using them in the measurement of the working standards, glass vials containing gas radon, prepared with the radon facility. These standards will be used for the calibration of equipments of various laboratories involved in radon measurement. A method for modeling of the radon transport in various matrices was also elaborated. One part of it refers to the simulation of the radon detectors containing active charcoal, a particular case of a purely diffusive transport in a medium with constant saturation. It will be validated experimentally, with radon gas standards produced with the standard radon system.

Material and methods

1. Presentation of the system and the preparation of sources

The new variant of the system was constructed from stainless steel and contains all the necessary gauges for the control of the pressure inside the circuit. It allows the extraction of large activities of radon from the Pylon source. At the same time it can be used in several modes of work, as it contains two ways of access to the circuit: (i) the concomitant extraction in two recipients; (ii) the extraction of radon in a single recipient and the transfer into another. Both these types of operation were intended to prepare the final (iii) mode, consisting of the sequential extraction and quantitative transfer from a recipient to another, in order to establish the traceability chain. This mode of operation is treated in an internal IFIN-HH Report (Report, 2009). Photograph 1 presents the two variants of installations

Fig. 1 (a) Glass system.

Fig.1(b) Metallic system.

Several sources were prepared in glass ampoules or vials, in both modes of work, (i) and (ii).The pair of sources 9 - 10 were prepared in the manner (i), and the pairs 13-14, 15-16 and 17-18 were prepared in the manner (ii). The sources 9,13,15 and 17 contained the radon as gas, while the sources 10, 14, 16 and 18 were prepared by adsorption/ dissolving of radon in liquid scintillator (LS) type OPTIFLUOR O, a Perkin Elmer product. The recovery of radon in the recipients was done at the 77 K temperature, by using Dewar vessels with liquid nitrogen. The pressure in the metallic variant of system was fully controlled and allowed for the preparation of high activity sources by the recovery of the main part of radon exhaled by the Pylon source. Figure 2 represents both types of radon ampoules, flame sealed.

Fig.2. Flame sealed ampoules: left – LS; right - gas.

All the operations were performed in a radionuclide fume cupboard, coupled to the ventilation circuit. The radon concentration in the laboratory was permanently measured with a monitor, Radon Scout, SARAD GmbH, Germany product, coupled to a personal computer. Figure 3 shows the registrations for a 24 h period of the radon concentration in the laboratory, during the preparation of sources.

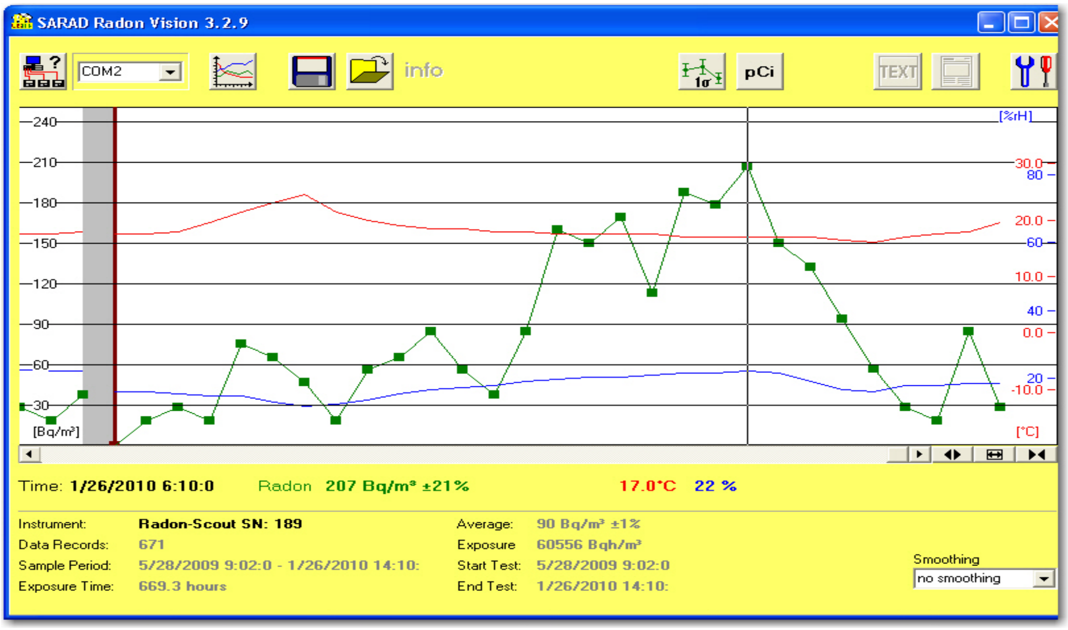

Fig. 3. Radon concentration in the laboratory.

Such as it can be seen from the figure, the maximum concentration values are registered during the night time, when the laboratory ventilation is not working; the marked, 210 Bq/m^3, is at 6 h in the morning. During the working time the values are less than 90 Bq/m^3, showing that no contamination occurs.

2 . The measurement of activity

The measurements were performed by the absolute LSC method with the ampoules prepared in LS, and relatively, by the gamma–ray spectrometry using a high resolution HPGe system and by the measurement in a CENTRONIC IG12/20A well type ionization chamber, with all types of recipients.

2.1. Measurement by the LSC method

The method was described in the above cited papers. The method consists basically from the measurement of the whole sequence of short lived nuclides of the ^{222}Rn decay chain in a LS counter. The main correction is due to the disintegration of ^{214}Po during the dead time of the LS counter. The formula for the calculation of the ^{222}Rn from the counting rate R_0 is:

$$R_0 = 5.0242 \cdot A_{\mathrm{Rn222}} \tag{1}$$

In relation (1), R_0 represents the corrected counting rate for the ^{214}Po decay. The approximate relation for this correction is:

$$R = R_0\left[1 - 0.0008576\,\tau\left(1 + \frac{\tau}{2}(\rho - \lambda)\right)\right] \tag{2}$$

In this relation, R is the registered counting rate, corrected for background; τ is the fixed dead time of the counter, in μs, $\lambda = 0.00427$ (μs)$^{-1}$ is the decay constant of ^{214}Po and ρ is the true counting rate.

A method based on the precise determination of the fixed counter dead time τ = (40.07±0.06) μs and the repetition of measurements for several days, with the extrapolation to the source preparation time was applied, such as described by Sahagia et al.(Sahagia et al. 2009a). The extrapolation is correct, and the data are correctly used, when the linear extrapolation equation is accomplished:

$$[R_0] = [R_0]_0\, 2^{-\frac{t}{T_{1/2}}} \quad or \quad \log[R_0] = \log[R_0]_0 - \frac{\log 2}{T_{1/2}}t \tag{3}$$

$[R_0]_0$ is the extrapolated counting rate, corresponding to the source preparation time, and $T_{1/2} = (3.8232 \pm 0.0008)$ d is the half life of ^{222}Rn according to Monographie BIPM-5. (Bé et al.2008). Contrary to the previous measurements, when low ^{222}Rn activities were extracted from the Pylon source (a maximum 1572 Bq at the preparation time), in this case higher activities were extracted and had to be measured. It was necessary to wait for longer periods the decay of radon, in order to be measured. As the extrapolation was extended on longer time intervals, more measurements and checks of the correct extrapolations were necessary.

2.2 The measurement by gamma-ray spectrometry

The measurements were performed with gas and LS recipients. A gamma-ray spectrometer, provided with a high efficiency HPGe detector was used. The sources were measured at a distance of 44.2 cm from the detector surface, in order to diminish the influence of the recipient geometry. The measurements were made on the two main photopeaks: 352 keV (Pb-214) and 609 keV (Bi-214). The emission intensities for the two quanta, $I,(\gamma s^{-1})/Bq$ were respectively: 0.3560(7) and 0.4549(19) (according to Bé et al 2008). Various models for calculation of efficiency, leaving from point source approximation and the GESPECOR programme (GESPECOR, 2007) were used. More details are given in the Internal Report IFIN-HH, 2009.

2.3 Measurements with the ionization chamber

The laboratory disposes of a well type ionization chamber type CENTRONIC IG12/20A, provided with an electrometer type Keithley 6517A, which was calibrated for a number of 18 different radionuclides. The results were validated during our participation in key and supplementary comparisons, Sahagia et al.(Sahagia et a. 2009b). The calibration factors were expressed in terms of the ratio between the ionization current and the activity of the measured sources.The measurements were performed with both types of recipients by registering the obtained ionization current. In the case of the gas ampoules, the measurements based on the HPGe were taken as reference. In the case of the LS ampoules, both methods of activity measurement, LSC and HPGe, were used. The measured activities were considered as the arithmetic mean of the two photopeaks' results.

3. Modelling methods

A conceptual and mathematical model describing the migration of the radon generated from the radium existing in the soil was elaborated. The experience gained in various modeling situations, such as presented by Toro et al. (Tora et al. 2008) was used. The radon transport is produced first of all due to the diffusion which occurs as a consequence of the radon concentration difference between the two phases: gas and liquid. Leakage of the transport fluids, air and water, drives to an advective - dispersive transport. The description of this biphasic emanation process is complicated also by the physic-chemical properties of the porous medium; more equations must be written both for the transport and leakage, coupled by the degree of saturation and physic-chemical processes. The modelling consists in several types of activities: (i) Definition of a conceptual model, containing the work hypotheses; (ii) Elaboration of a mathematical model, containing the sets of equations which describe the physical processes; (iii) Elaboration of the numerical model for solving the equations. A particular case of the model application is the simulation of the transport of radon inside the detectors containing active charcoal, a pure diffusive transport in the medium with a constant saturation. The calculation results follow to be compared with those experimentally obtained with the use of radon gas standard vials.

Results

The radon prepared sources activity measurement results, obtained by using the three methods, are presented in Tables 1, 2 and 3.

1. Measurement of activity by the LSC method

A number of 4 LS ampoules were measured and the results, in terms of total decay chain activity $[R_0]_0$, radon activity, $[A_{Rn\text{-}222}]_0$ on the source preparation time and the optimum calculated half life, $T_{½}$, are presented in Table 1.

Table 1. LSC measurement results.

Amp. no	Time from the preparation date, d / no of points	$Y_0 = A_0 + B_0\, t$, $\tau = 40.07$ μs
10	(20.8403 – 28.9062)/ 5	Y_0 = (5.7411 ± 0.0074) – (0.07793 ± 0.00028) t $[R_0]_0$= (550 883 ± 9 475) Bq; $T_{1/2}$ = (3.8628 ± 0.0138) d $[A_{Rn222}]_0$= (109 646 ± 1 886) Bq, k=1
14	(8.8125 - 14.8611)/ 5	Y_0 = (5.0762 ± 0.0087) – (0.07868 ± 0.00014) t $[R_0]_0$= (119 182 ± 2 395) Bq; $T_{1/2}$ = (3.814 8± 0.036) d $[A_{Rn222}]_0$= (23 721 ± 477) Bq, k=1
16	(19.0556 - 26.0208)/ 2	Y_0 = (5.7725 ± 0.0066) – (0.07868 ± 0.00014) t $[R_0]_0$= (592 243 ± 8 943) Bq; $T_{1/2}$ = (3.8274 ± 0.036)d A_{Rn222} = (117 878 ± 1 780) Bq, k=1
18	(12.9410 – 30.000)/ 3	Y_0 = (5.4486 ± 0.0053) – (0.07855 ± 0.00068) t $[R_0]_0$ = (280 931 ± 3 425) Bq; $T_{1/2}$ = (3.8232 ± 0.033) d $[A_{Rn222}]_0$ = (55 916 ± 671) Bq, k=1

2. Measurement of activity by the HPGe gamma-ray spectrometry

A number of 2 gas and 2 LS ampoules were measured and the results are presented in Table 2.

Table 2. Results of measurement by the HPGe Gamma-ray spectrometry.

Recipient No./ Measure ment	Photo peak,	Nγ, s^{-1}, on reference time	ε, $s^{-1}/(\gamma s^{-1})$ /model	Activity on reference time , Bq	A_{Bi}/A_{Pb}	^{222}Rn, activity, Bq, on reference time
1	2	3	4	5	6	7
9/ horizontal	352 keV 609 keV	13.231 ± 0.071 10.459 ± 0.061	0.000364/point 0.000200/point	^{214}Pb:102100 ^{214}Bi: 114952	1.126	101547 ± 2132 113915 ± 2392
13/ horizontal	352 keV 609 keV	24.98 ± 0.135 19.810 ± 0.114	0.000368/point 0.000217/point	^{214}Pb:190675 ^{214}Bi :200682	1.052	189657 ± 3983 198906 ± 4177
10/ horizontal	352 keV 609 keV	12.263 ± 0.066 10.036 ±0.058	0.000364/point 0.000200/point	^{214}Pb: 94630 ^{214}Bi: 110313	1.166	94117 ± 1946 109318 ± 2296
10/ vertical	352 keV 609 keV	10.872 ± 0.059 8.897 ± 0.052	0.000275 GESPECOR 0.000174 GESPECOR	^{214}Pb:111052 ^{214}Bi :112403	1.012	110450 ± 1767 111389 ± 1782
14/ vertical	352 keV 609 keV	2.330 ± 0.023 1.890 ± 0.019	0.000275 GESPECOR 0.000174 GESPECOR	^{214}Pb: 23800 ^{214}Bi: 23878	1.0033	23671 ± 426 23663 ± 426

2. Measurement of activity by the ionization chamber

Two gas and four LS sources were measured. The results are presented in Tables 3 and 4.

Table 3. Measurement of gas ampoules with the ionisation chamber.

Amp no.	I, pA, on the reference time	Photopeak energy	Rn-222 Activity, MBq, HPGe Determined, Table 2	F, pA/MBq, HPGe based
1	2	3	4	5
9	5.831 ± 0.044	352 609	0.10155 ± 0.0021 0.11392 ± 0.0024	57.42 ± 1.28 51.18 ± 1.13
13	10.877 ± 0.083	352 609	0.18966 ± 0.0040 0.19891 ± 0.0042	57.34 ± 1.26 54.68 ± 1.20

Table 4. Measurement of the LS sources with the ionization chamber.

Amp no	I, pA, on reference time	Activity, MBq, LSC determined, Table 1	F , pA/MBq, LSC	Activity, MBq, HPGe determined, Table 2	F, pA/MBq, HPGe	Difference Col (6 - 4)/ Col 4
1	2	3	4	5	6	
10	5.617± 0.039	0.10965 ± 0.00189	51.23± 0.94	0.11092 ± 0.00177	50.64± 0.88	-1.15%
14	1.145± 0.034	0.023721 ± 0.000477	48.26± 1.74	0.02367 ± 0.00043	48.38± 1.69	+0.25%
16	5.963± 0.042	0.11788 ± 0.00178	50.59± 0.86	-	-	-
18	2.814± 0.028	0.055916 ± 0.000671	50.33± 0.81	-	-	-
		Weighted mean values and standard deviations :	50.49±0.65		50.17±1.13	-0.63%

Discussion

The measurement results, such as presented in Tables 1, 2, 3 and 4 reveal the followings.

Regarding the absolute, liquid scintillation method, it is the primary method, to be taken into account for the realisation of a primary radon standard. The LSC method assured a good precision of standardization, combined standard uncertainty situated between 1% and 2%, even in the case of these high activity sources. The calculated radon half life values from the extrapolation procedure are in agreement with the theoretical value $T_{1/2} = (3.8232 \pm 0.0008)$ d.

Regarding the HPGe gamma-ray spectrometry,Table 2 results, one may comment. The theoretical ratio of the ^{214}Pb and ^{214}Bi activities is $A_{Bi} / A_{Pb} = 1.00363$. The determined ratio values, column 6, is much higher for the point source approximations in efficiency calculation; the ^{222}Rn activity values, column 7, differ significantly from a photopeak to another and consequently the model is not adequate for efficiency evaluation. The GESPECOR variant applicable only to the LS sources is much better, as the values agree and also they agree with the values measured absolutely by the LSC (sources 10 and 14). This program is not applicable to the gas recipients, maybe because the volume distribution of the ^{214}Pb and ^{214}Bi inside these sources is not known. The conclusion is that the only possibility to determine the activity of gaseous sources is to achieve a quantitative transfer of radon from the gas to LS recipients, to measure them by the LSC method and to establish the traceability chain.

Table 3 and Table 4, ionization chamber measurement results discussion. The calibration factors obtained with 2 gas ampoules, leaving from the gamma-ray spectrometry measurements are not consistent in terms of their combined standard uncertainties. On the contrary, for a number of 4 different LS sources the calibration factors agree, once as individual results with reference to the LSC method, and twice as compared with those determined taking as reference the activity measured by the HPGe gamma ray spectrometry method, based on the use of the GESPECOR program for efficiency calculation. This result has the same conclusion, that the main problem is to develop the method for the transmission of activity unit from the LSC method to the gaseous sources, in order to assure the traceability chain.

Conclusions

- The radon system was used to prepare recipients containing radon extracted from a radium source, both as gas and dissolved in liquid scintillator.
- The activity of all radon recipients was measured by various methods: the LS ampoules were measured absolutely by the LSC method; all recipients, gas and LS, were measured relatively by using the gamma-ray spectrometry and the reentrant ionization chamber.
- The comparison of results revealed that in the case of the LS ampoules a model for gamma–ray spectrometry efficiency calculation is adequate, such as the agreement of results demonstrated, while for the gas recipients no model can be used.
- The general conclusion is that a traceability scheme for the transfer of activity unit from primary to secondary standards is needed.

Acknowledgement

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Estimating lung cancer risk due to radon exposure in the radon-prone areas of Belgium

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Abstract

Radon exposure in Belgium is particularly pronounced in the southern part of the country, characterized by a sub-surface composed of highly deformed and fractured (black) shale, schist and quartzite of the Ardenne massif. A national indoor radon measurement campaign (1995-2000) showed that all of the high radon risk areas (where more than 5% of the measured buildings exceed the current Belgian action-level of 400 Bq/m³) were situated within the Ardenne massif, affecting a population of about 380 thousand. For this reason, detailed information, measurement and prevention campaigns have been organized for the local population and municipal authorities. Whereas the national average indoor radon concentration is about 50 Bq/m³, this average increases to about 130 Bq/m³ in the high risk areas. Here, 13% of the houses exceed the action level, affecting more than 50 000 people, and 4% of the houses exceed 800 Bq/m³, affecting more than 15 000 people. In 33% of the dwellings, the design level for new buildings (200 Bq/m³) is exceeded. According to the risk estimates from international epidemiological studies, about 18% of the occurring lung-cancers in the high risk areas (~43 cases on 240 per year) would be due to radon exposure. About 38% of the radon-induced lung-cancers (LC) would occur in the population exposed to more than 400 Bq/m³ (about 30 lung-cancers per year). Comparison of these theoretical values with actual LC statistics of the Belgian Cancer Registry shows a good match between the total number of annual LC in the high risk areas (239 calculated to 240 observed in the period 2004-2005). The correlation between LC incidence rate and average radon concentration however is obscured by the high number of influencing factors (migration, age-distribution, life-habits,…) and the relatively limited population and LC incidence rate. Radon campaigns aim at stimulating house owners and building responsibles to mitigate the radon affected buildings and to apply preventive measures in new buildings. In the high risk areas, preventive reduction of radon exposure to below 200 Bq/m³ should lead to a reduction of the LC incidence rate with 7%.

Introduction

For the general population, radon is usually the most prevailing source of radiation exposure in the indoor environment (ICRP 60, 1990). The link between radon and lung cancer has been first recognised through miner cohort epidemiological studies (BEIR

IV, 1988; BEIR VI, 1998; UNSCEAR 2000). More recent case-control studies highlight the linear no-threshold relation between lung cancer risk and indoor radon-concentration in dwellings (Darby et al., 2005). In order to efficiently manage the radon exposure, most European countries have adopted a radon action plan, setting out the criteria, strategies and practical aspects of radon controlling activities. The Belgian radiation protection regulation (ARBIS, 2001) foresees the control of radon exposure in workplaces and in dwellings in radon prone areas. A national radon measurement campaign during the 1990's highlighted the occurrence of radon prone areas (defined at that time as municipalities where more than 1% of the dwellings exceed the action level of 400 Bq/m³) in the southern part of the country (Poffijn and Vanmarcke, 1990; Zhu et al., 1998). Ongoing indoor radon measurements led to the classification of the territory into radon regions (Fig. 1) on a national scale and into radon classes on a local scale (Fig. 4). Within the radon prone areas, a specific region shows a high risk of having high indoor radon concentrations. This is the high risk area (HRA, Radon region 2 on figure 1), where more than 5% of dwellings exceed the action level. It is characterised by the occurrence of highly deformed metamorphic rocks of Lower Palaeozoic age. The HRA affects a population of about 380 thousand. If the national action level would be brought to 200 Bq/m³, the population living in the HRA would increase to more than 1.1 million. The measurement campaigns also allowed assessing the radon exposure of the Belgian population (Table 1).

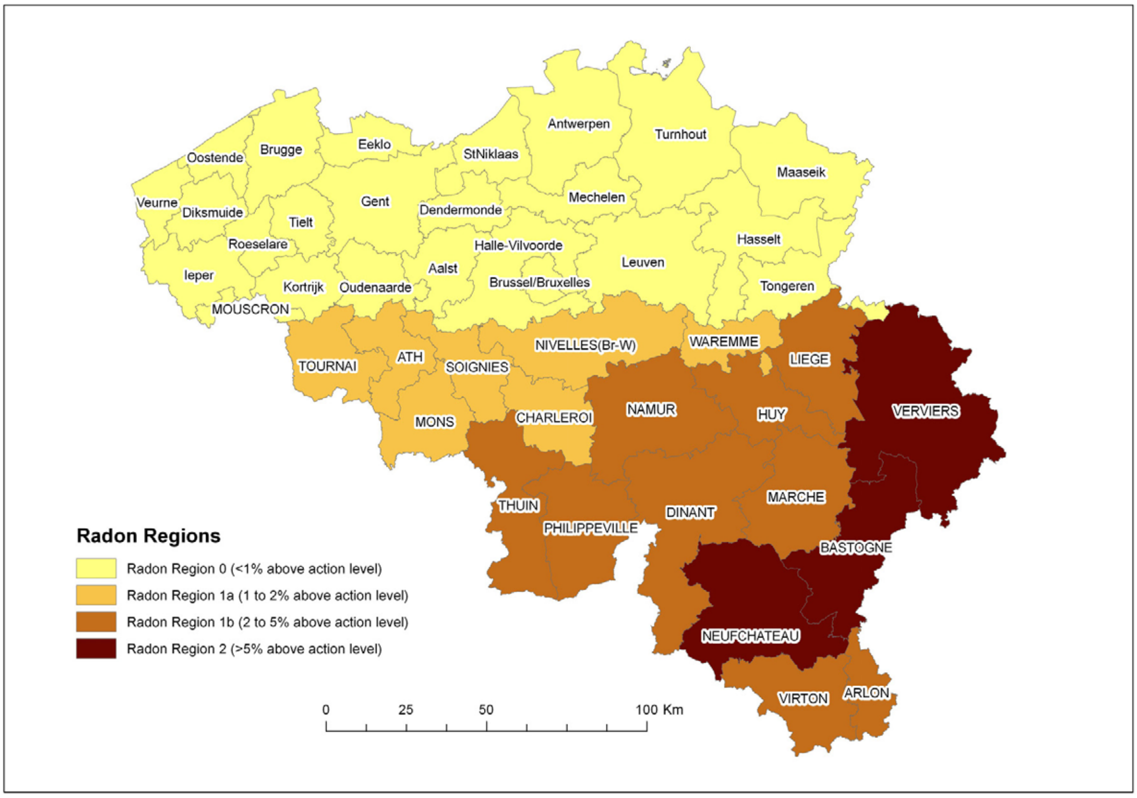

Fig. 1. Radon distribution per district.

Table 1. Statistics for the indoor exposure of the Belgian population. GM: Geometric mean. GSD geometric standard deviation. Radon concentrations are in Bq/m³.

	Population	dwellings	Median	GM	GSD	Max	% >100	% >200	% >400	% >800
Belgium	10584534	5043023	52	59	1.7	4204	11.0	2.2	0.4	0.1
Wallonia	3435879	1570265	69	84	2.0	4204	25.0	4.4	1.3	0.3
Flanders	6117440	2928158	38	44	1.2	70	5.0	0.8	0.0	0.0
Brussels	1031215	544601	38	44	1.2	120	5.0	0.8	0.0	0.0
HRA	376568	166000	127	137	2.5	5500	43.0	33.0	13.1	4.3

Material and methods

The Belgian Cancer Registry containes LC incidence data on a municipal scale starting from the year 2004. The most recent data concern 2005. These data show the variability in LC incidence rate between men and women as well as some regional variation. Table 2 shows the annual LC incidence rate for the period 2004-2005 in Belgium. For this period, about 7000 LC per year have been registered.

Table 2. Observed annual LC incidence rate in Belgium in the period 2004-2005.

Region	population	men	women	total
Belgium	10584534	5361.5	1539.5	6901.0
Wallonia	3435879	1744.0	521.5	2265.5
Flanders	6117440	3213.5	844.0	4057.5
Brussels	1031215	404.0	174.0	578.0
HRA	376568	180.0	61.0	241.0

Table 3. Observed annual LC incidence rate in the HRA in the period 2004-2005.

District	men	women	total
Verviers	127.5	46.5	174.0
Bastogne	21.5	6.0	27.5
Neufchâteau	31.0	8.5	39.5
Total	**180**	**61**	**241**

The linear increase without threshold of the relative risk with indoor radon concentration has been estimated in various recent studies (Baysson et al., 2004; Darby et al., 2005). In this paper, the value of 16% increase of relative risk per trench of 100 Bq/m³ has been used following Darby et al., 2005. This leads to the estimation of LC incidence rate based on the average (GM) radon concentration for each region (Table 4). The obtained values show a good correlation between the observed values (table 2) and calculated LC occurrence. This estimation uses a smoker population of 1/3, which is justified taking into account the importance of the long term exposure of the population.

Table 4. Estimated annual LC incidence rate in Belgium for smokers (S) and non-smokers (NS). Lifetime LC risk (0.16% increase per Bq/m³)* expressed per thousand. Lifetime =70 y.

	LC risk NS (promille)	LC risk S (Promille)	LC NS	LC S	total	Without radon	due to radon
Belgium	4.487	110.534	452	5571	6023	5504	520 (9%)
Wallonia	4.651	114.574	152	1875	2027	1787	240 (12%)
Flanders	4.389	108.110	256	3149	3405	3181	224 (7%)
Brussels	4.389	108.110	42	521	563	536	38 (7%)
HRA	4.999	123.139	18	221	239	196	43 (18%)
No radon*	4.100	101.000					

**After Darby et al.. 2005*

A more refined estimate of LC occurrence can be obtained taking into account the variability of the radon exposure within the HRA. This estimate gives a slightly higher LC incidence rate.

Table 5. Estimated annual LC incidence rate in the HRA for smokers (S) and non-smokers (NS).

Exposed to (Bq/m³)	Ref used (Bq/m³)	Population in the HRA	annual LC				
			NS	S	total	without radon	due to radon
>800	800	18828	2	20	22	10	12 (56%)
400 to 800	400	30125	2	24	25	16	10 (39%)
200 to 400	200	86611	4	54	59	45	14 (24%)
<200	100	241004	11	133	144	124	20 (14%)
total		**376568**	**19**	**232**	**250**	**194**	**56 (23%)**

Results

The number of calculated LC (Table 4) corresponds well to the number of observed LC (Table 2). On the national level this would mean that about 8% of the LC incidence rate would be due to radon. whereas this incidence rate due to radon increases to 18% in the HRA. The calculated incidence rate within the HRA depends largely on the exposure class, where for the highest (lifetime) exposures of more then 800 Bq/m³ 56% of the LC incidence rate could be attributed to radon.

When looking at the scatter plot of lung cancer incidence rate and radon concentration per district, no statistical correlation can be observed. For the data per municipality, the LC incidence rates for the observed period are so low that no correlation whatsoever can be observed. When looking at the (age standardised) LC incidence rate among women, however, a vague trend can be observed (Fig. 2).

Figure 3 shows the age standardised LC incidence rate among women per district. This figure shows that the HRA contain some of the highest incidence rates in the country, except for one district (District *Marche*) with very low LC incidence rates.

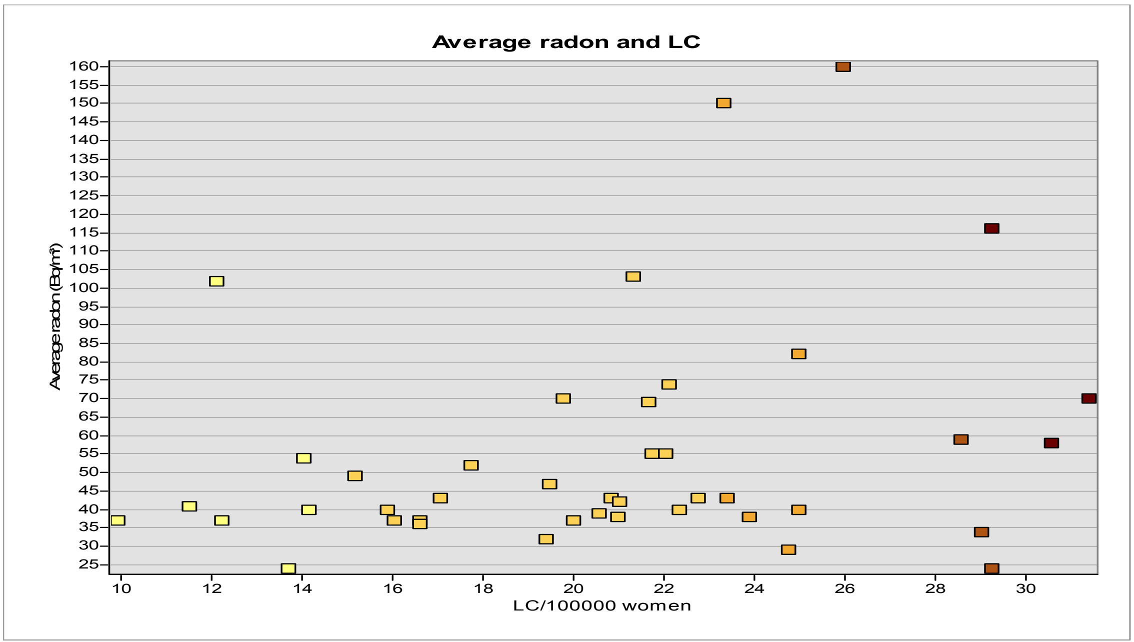

Fig. 2. Scatter plot of the average (GM) radon concentration per district versus LC incidence rate among women.

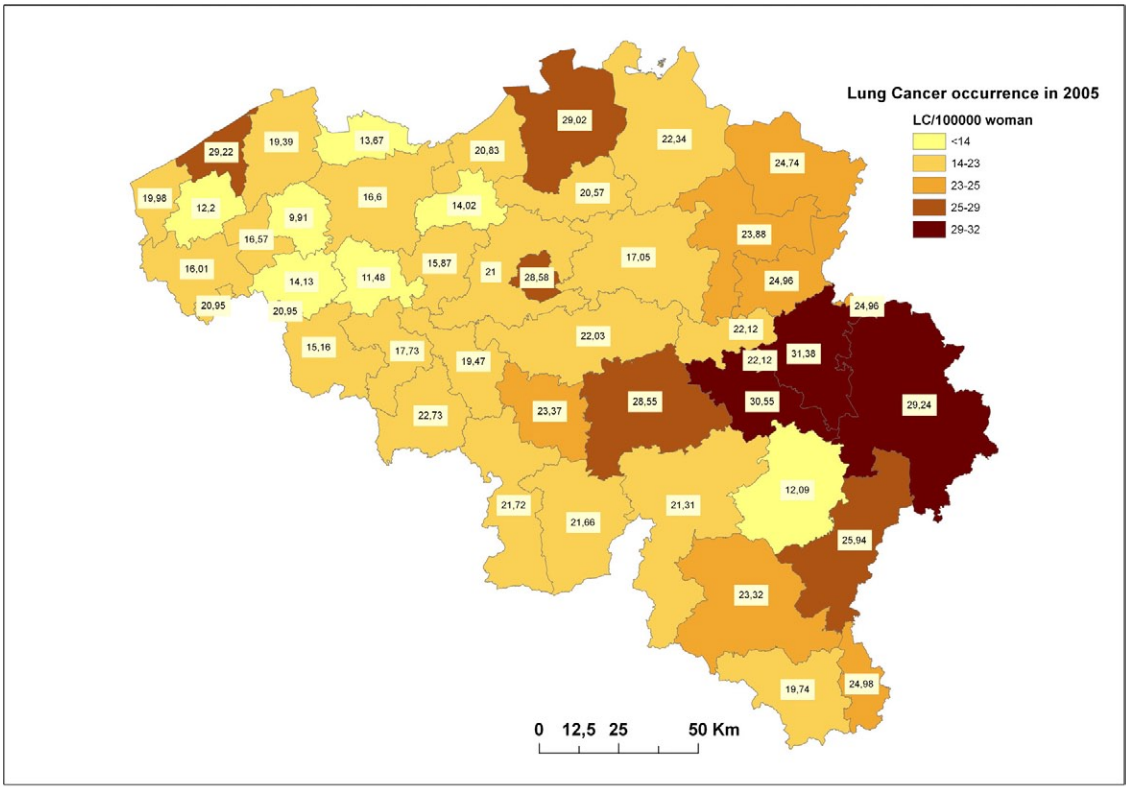

Fig. 3. Distribution of age standardised LC incidence rate among women per district.

Figure 4 shows the radon distribution per municipality on the Belgian territory. This figure shows that, in the case of the district 'Marche', partly covered by the HRA, only 4 municipalities, representing only 22% (12368) of the population of the total district (53123) actually are within the HRA.

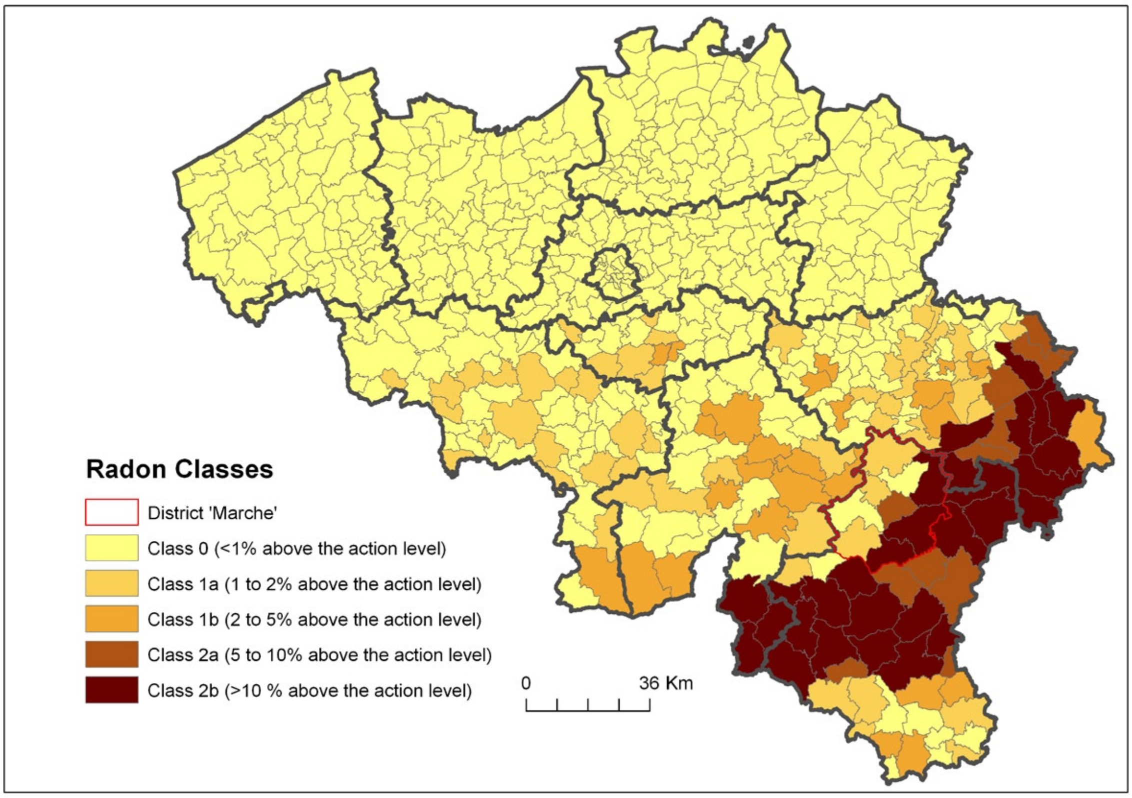

Fig. 4. Radon distribution per municipality.

Discussion

The LC incidence for 2004-2005 contained in the National Cancer Registry correlates well with the data calculated from the radon risk studies. Comparing the observed number of lung cancers to the calculated number of lung cancers, there is a better match for the high risk areas then for the other areas. Part of the explanation could be that much more radon measurements have been made in the HRA than in the less affected regions. improving the accuracy of the calculated LC incidence values.

A slight trend can be observed when looking at the distribution of radon on the territory and LC incidence rate for women per district. A remarkable deviation from this general trend (district 'Marche' on fig. 4) could be due to the discrepancy between the district boundaries and the HRA limits. Refining the resolution of the LC incidence rate data to the municipality scale, however, leads to too few cases to allow statistical correlation. A possible trend in the LC incidence rate data among man, from the other hand, seems not at all suggested by the data. Smoking habits and occupational exposures might be a reason for this.

Conclusions

A reoccurring question among the population in the high risk areas concerns the link between regional variations in LC incidence rate and increased indoor radon concentrations in their region. The current preliminary study investigates the risk estimates based on the recently published international epidemiological studies of radon and LC, applied to the high radon risk areas in Belgium, and compares them to the observed number of LC in this region as available from the Belgian Cancer Registry. Although the Belgian Cancer Register contains until today only data for the years 2004-2005, some preliminary conclusions can be drawn from the present analysis.

LC incidence rate correlates well with LC estimates taking into account radon, based on the most recent international epidemiological studies (Baysson et al., 2004; Darby et al., 2005). For areas with a better estimation of radon concentration distribution, the fit between the observed and calculated LC incidence rate is better. This confirms the quality of the used epidemiological studies and their ability to estimate the influence of radon on the general LC incidence rate and hence to allow a risk reduction policy based on the epidemiological findings.

This preliminary study indicates that the correlation between the regional variation of LC incidence and the regional variation in radon concentration is statistically weak and remains until now unproven. This is partly due to the very low number of cases, and due to the high uncertainty about other influencing factors, such as smoking/living habits and migration of inhabitants. Once the Cancer Registry data set will be extended and contain the LC incidence rate data of most recent years, a statistical study could possibly reveal this correlation, as is suggested by the weak trend already observed for LC incidence rate among women.

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Estimation of seasonal correction factors – a new model for indoor radon levels in Irish homes

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Abstract

Indoor radon concentrations have been shown to vary considerably with season. Many countries account for this by applying a correction factor to radon measurements of less than one year. To date, Irish radon measurement services have used correction factors based on data collected during a UK national survey of radon in 2000 homes in the 1980s. In the absence of similar data for Ireland at the time, these were considered suitable for use due to the similarities between the climate, house types and lifestyles in both countries. In order to better estimate the long term radon concentration in Irish homes from measurement results, a dataset comprising measurements from 5640 Irish homes was analysed and used to derive a set of correction factors specifically for Ireland. These were generated by means of Fourier decomposition analysis and the new correction factors compared, using 95% confidence intervals, to those derived in the UK using the same analysis and to those currently in use in Ireland. In both cases, differences were found between 10 of the 12 monthly seasonal correction factors. The results of this analysis will be given along with an overview of the methods used.

Introduction

1.1. Background

Radon is a naturally occurring radioactive gas which originates from the decay of uranium present in rocks and soils. When radon surfaces in the open air, it is quickly diluted to harmless concentrations, but when it enters an enclosed space, such as a house or any other building, it can sometimes accumulate to relatively high concentrations.

In air, radon decays quickly to produce radioactive particles that, when inhaled, are deposited in the airways and in the lungs. These result in a radiation dose which, over a long period of time, may lead to lung cancer. Exposure to radon in homes is linked to between 150 and 200 lung cancer deaths each year in Ireland. This represents 10 to 15% of all cases of lung cancer in Ireland and it is the second most significant cause of lung cancer after smoking (RPII and NCRI 2005).

Indoor radon concentrations can vary significantly due to a large number of factors which include the local geology, soil permeability, building and lifestyle characteristics and climate (Gunby et al 1993). Ireland's relatively high indoor radon concentrations were established during the National Radon Survey which was carried out between 1992 and 1999 by the Radiological Protection Institute of Ireland (RPII 2002). An average radon concentration of 89 Bq/m^3 was calculated for Ireland, based on the 11,319 homes measured during this survey. This national average has been shown in a recent survey by the World Health Organisation to be the eighth highest concentration in the world (WHO 2009). Since 1989 the Radiological Protection Institute of Ireland (RPII) has worked to raise awareness of this issue and provides a measurement service to encourage home-owners to measure and, where necessary, remediate their homes.

To measure indoor radon concentrations the RPII use a passive alpha track-etch detector which consists of a two-part polypropylene holder and a CR-39 (polyallyl diglycol carbonate) detection element (RPII 1992). The annual average radon gas concentration for a home is determined using two detectors; the first is placed in the main living area and the second placed in the main bedroom for at least three months. This three month measurement is then converted to an annual average radon concentration by applying a seasonal correction factor. The seasonally adjusted annual average radon concentration for the home may then be compared with the National Reference Level of 200 Bq/m^3.

1.2. Seasonal correction factors

The temperature difference between indoors and outdoors combined with the effect of wind on a building result in a lower atmospheric pressure indoors relative to that in the ground. This results in a pressure-driven flow of radon from the ground into the building, once an entry route exists (Wrixon et al 1988). Consequently, external changes in temperature between seasons can result in significant variation in indoor radon concentrations. For this reason it is important to apply seasonal correction factors to radon measurements of less than one year. In Ireland, the seasonal correction factor applied to any three month measurement is the mean of the three individual monthly correction factors for the measurement period (RPII 1994). These were derived from data collated for the UK national survey by Wrixon et al (1988). As mentioned in Organo and Murphy (2007), the suitability of these Irish factors has not been assessed until now, as it was assumed that the seasonal characteristics of both the UK and Ireland are similar.

The use of seasonal correction factors is a complex issue and has been much debated in the literature. Miles (2001) has shown that, for homes of "atypical" construction, factors such as wind speed and direction may have more influence on radon levels than the external temperature. He pointed out that a significant minority of homes (maybe 10 to 20% in the UK) have no substantial seasonal variation or followed a different seasonal pattern to that for typical homes. However, he showed that for a "typical" home, i.e., a home that responds to the weather in a manner typical to many Northern European homes, the application of seasonal correction factors improved the accuracy of estimates of the annual average radon concentration. Denman et al (2007) agreed that such correction factors improve the accuracy of predictions of long-term risks for measurements of at least three months.

It has been shown that one of the most significant effects on indoor radon concentrations is rock type (Gunby et al 1993) and that the type of geology can impact on seasonal variation. For example, it has been shown that radon-rich sedimentary geologies show a high degree of seasonal variation, while radon-rich igneous geologies result in relatively constant radon concentrations (Groves-Kirkby et al 2009). Consequently, a number of studies advise against the use of a single seasonal correction factor for a wide range of geologies. Pinel et al (1995) initially raised concern regarding this while more recently, Gillmore et al (2005) also advised caution in applying a single set of seasonal correction factors to regions with complex geologies. Pinel et al's study calculated correction factors using a series of measurements made in southwest England. However, despite the very different geology in this part of the country and different time-scales over which the measurements were made, the correction factors derived agreed closely with the original correction factors based on nationwide data (Wrixon et al 1988).

Due to this concern regarding the application of general seasonal correction factors to specific regions, a number of studies have derived regional seasonal correction factors. A study by Denman et al (2004) in an area with uniform geology showed that the seasonal variation observed did not match the "average" seasonal correction factors derived by the National Radiological Protection Board (NRPB) for use across the UK (Wrixon et al 1988). This study was carried out on a small sample size (34 homes) and over a limited geographical area and consequently, it is not surprising that the correction factors differ from those derived by the NRPB. The UK Childhood Cancer Study Investigators (2000) also derived region specific seasonal correction factors and concluded that the use of these improved the accuracy of their data. However, other studies have derived regional seasonal correction factors and found either that there was no statistical difference between these correction factors and the "average" correction factors (Grainger et al 2000) or that the differences were minor (Baysson et al 2003).

On balance, it is clear that the use of seasonal correction factors will result in a better estimate of the long term indoor radon concentration. On this basis, a collaborative study was carried out between the RPII and the School of Mathematical Sciences at University College Dublin to derive seasonal correction factors specifically for Ireland.

Material and methods

The statistical methodology used to estimate seasonal correction factors for Ireland is based on that described in Pinel et al. (1995). This makes use of the lognormality of background-corrected indoor radon concentrations and describes the data using Fourier decomposition. The data is transformed appropriately and a linearization is used to maintain approximately normal errors for least squares regression. The Fourier parameters are estimated by least squares regression and used to compute the appropriate seasonal correction factors.

This methodology results in two equations. The first is used to calculate “three month seasonal correction factors” which may be applied to correct radon measurements of three months (equation 1):

$$\hat{f}_i = \frac{3\sum_{k=1}^{12}\hat{m}_k}{12\sum_{k=i-1}^{i+1}\hat{m}_k} \qquad (1)$$

The second equation is used to calculate “one month seasonal correction factors” and may also be applied to a radon measurements of greater than three months. The mean of the individual monthly correction factors for the measurement period is calculated and applied to the radon result.

$$\hat{f}_i^{(1)} = \frac{\sum_{i=1}^{12}\hat{m}_k}{12\hat{m}_i} \qquad (2)$$

Application of this method to 5,640 background-corrected, lognormally distributed data from the RPII’s home measurement database results in two sets of seasonal correction factors for use with future radon measurements in Irish homes. This statistical methodology is described in full detail by Burke et al. (2010).

Results

3.1. Comparison with UK factors

Table 1 reports the three-month seasonal correction factors (equation 1) calculated for Ireland using the Fourier decomposition analysis alongside the factors calculated by Pinel et al. (1995) for the UK using the same analysis. It should be noted that the convention adopted in reporting the seasonal correction factors means that the UK factors reported below are in fact the reciprocals of those presented by Pinel et al.

A simulation study was carried out – simulating 10,000 datasets, each consisting of monthly data with 1,000 observations in each of the twelve months. This allowed the calculation of the empirical 95% confidence intervals for the Irish factors.

Table 1. New Irish seasonal correction factors compared to UK factors.

Measurement Month (i)	New Irish Factors	New Irish Factors: Standard Error	New Irish Factors: 95% Confidence Interval	UK factors (Pinel *et al*, 1995)
1	1.14*	0.00627	(1.12, 1.17)	1.35
2	1.14*	0.00633	(1.12, 1.17)	1.27
3	1.10	0.00502	(1.08, 1.13)	1.10
4	1.04*	0.00220	(1.02, 1.07)	0.91
5	0.97*	0.00217	(0.94, 0.99)	0.75
6	0.89*	0.00728	(0.87, 0.92)	0.65
7	0.86*	0.01106	(0.83, 0.88)	0.64
8	0.86*	0.01123	(0.83, 0.88)	0.74
9	0.89	0.00769	(0.87, 0.92)	0.89
10	0.96*	0.00259	(0.93, 0.99)	1.09
11	1.04*	0.00189	(1.01, 1.06)	1.25
12	1.10*	0.00485	(1.08, 1.13)	1.35

Note:
Correction factors apply to a three month period, centred on measurement month i
A significant difference between factors is indicated with *
The standard errors were calculated using the Delta method (Pawitan 2001)

It is clear that all of the factors are significantly different with the exception of the factors associated with the third and ninth months (March and September). These months correspond to the time points close to where the graph of the two sets of factors intersect (Figure 1).

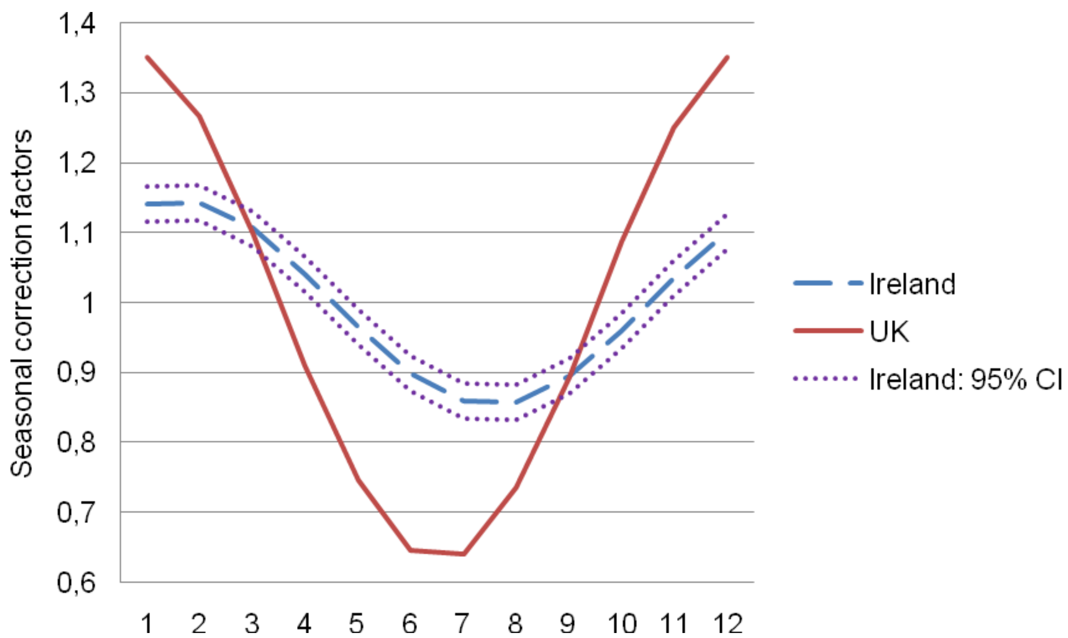

Fig. 1. Comparison of new Irish seasonal correction factors with UK factors for months 1 - 12

3.2. Comparison with seasonal correction factors currently used in Ireland

The seasonal correction factors currently in use in Ireland (RPII 1994) are one month factors based on the Wrixon et al. moving average analysis of UK data. The seasonal correction factor that is presently applied to any three month measurement is the mean of the three individual one month seasonal correction factors for the measurement period. These are given in Table 2 where they are compared with the one month seasonal correction factors that have been calculated in this study using equation (2).

The Irish seasonal correction factors calculated in this study are significantly different from the current factors in ten of the twelve months (Table 2). The two sets of factors associated with measurement months 4 and 10 (April and October) are not found to be significantly different (Figure 2).

Ideally, in order to fully validate the newly calculated seasonal correction factors, we would compare seasonally corrected observations to true annual measurements for each dwelling. Unfortunately, since this was not a designed experiment, we do not have access to both short term and corresponding annual measurements on a house-to-house basis and this validation analysis cannot be done. However, we can provide some evidence to support the robustness of the new Irish seasonal correction factors. In particular, we have examined the year-to-year variation of the factors by analysing data from two three-year subgroups (1999-2001 and 2003-2005) and found that the pattern of the seasonal correction factors remains stable across the years.

To illustrate the application of these factors in practise, an example may be useful for the reader. Consider a January measurement of 130 Bq/m^3. The RPII estimate the total measurement uncertainty to be 27% (Hanley et al 2008). The 95% confidence interval on the measurement is therefore (94.9 Bq/m^3, 165.1 Bq/m^3). The January factor (1.16) with corresponding 95% confidence interval (1.14, 1.20) is then applied to each end of the measurement confidence interval (94.9 Bq/m^3, 165.1 Bq/m^3). This allows us to create a 95% confidence interval for the seasonally adjusted value (108.19 Bq/m^3, 198.12 Bq/m^3).

Table 2. Comparison of new and current seasonal correction factors for Ireland.

Measurement Month	New Irish Factors	New Irish Factors: 95% Confidence Interval	Current Irish Factors
1	1.16*	(1.14, 1.20)	1.35
2	1.16*	(1.14, 1.20)	1.30
3	1.12*	(1.10, 1.16)	1.20
4	1.05	(1.02, 1.08)	1.05
5	0.96*	(0.93, 0.99)	0.90
6	0.89*	(0.85, 0.91)	0.75
7	0.85*	(0.80, 0.86)	0.60
8	0.84*	(0.80, 0.86)	0.65
9	0.88*	(0.85, 0.90)	0.80
10	0.96	(0.92, 0.98)	0.95
11	1.04*	(1.01, 1.07)	1.10
12	1.11*	(1.10, 1.15)	1.20

Note:
Correction factors apply to a one month period
A significant difference between factors is indicated with *

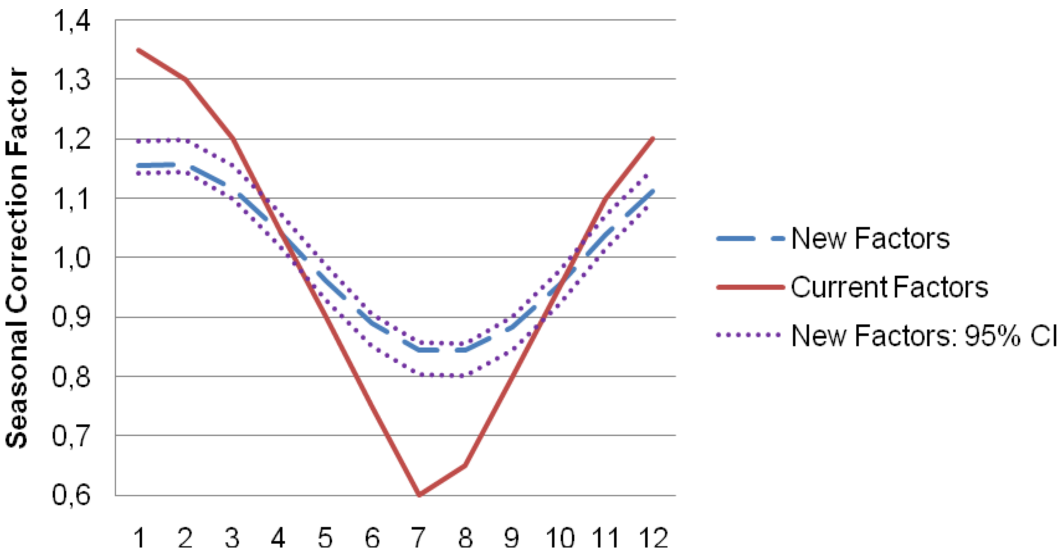

Fig. 2. Comparison of new and current one-month seasonal correction factors for Ireland for months 1 to 12.

Discussion

The "three month" correction factors have been generated from three-month data so ideally, if all measurements were exactly three months duration, these correction factors would be applied to future radon measurements. However, in reality, many measurements exceed three months, making use of "three month" correction factors impractical. In this paper we also have estimated a set of "one month" correction factors. For measurement periods differing from three months, the use of the mean of the individual one month correction factors for the measurement period is more practical and flexible. Use of these factors has shown that there is negligible difference in the correction factor that is applied, whichever set of correction factors are chosen.

Both the "one month" and the "three month" correction factors differ from the equivalent factors derived by Pinel et al. and Wrixon et al. There may be a number of reasons for this observed difference. It has been shown that monthly mean radon concentrations correlate strongly with the monthly mean outdoor temperature during the measurements (Miles 1998). Overall, there are larger extremes of temperature in the UK, most likely due to the climate moderating influence of the Atlantic Ocean on Ireland. This may account for some of the observed differences in seasonal correction factors. As discussed earlier, seasonal variation in indoor radon concentrations may also be affected by other external factors such as local geology, the structure of the dwelling or climate and may vary from year to year. In addition, it may not be appropriate to apply correction factors for measurements of less than three months, to homes of atypical construction or those built on unusual geology. Analysis, to investigate the effects that these external factors may have on the seasonal correction factors derived, is currently ongoing in University College Dublin.

Conclusions

The main aim of this paper was to produce, for the first time, a set of seasonal correction factors for indoor radon levels in Ireland derived from Irish data. Prior to this the factors used in Ireland were based on UK data.

The new set of seasonal correction factors for Ireland is calculated using a Fourier decomposition procedure recommended by Pinel et al. (1995) as an improvement on the Wrixon et al. (1998) procedure. The new seasonal correction factors for Ireland are statistically significantly different from the current factors in use in Ireland and the UK.

As discussed, we are aware that there are other external factors that influence the seasonality of radon emissions. However, the set of factors reported here is that which best summarizes the position for a typical Irish home and their use will provide a better estimate of long-term radon concentrations in Irish homes.

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Upgrade of the NIRS radon chamber – generation of aerosol-attached radon progeny

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Abstract

A condensation monodisperse aerosol generator has been used in NIRS radon-aerosol chamber; aerosol particles are generated by the evaporation–condensation method and supplied into the chamber through the sampling port. Carnauba wax is used as the aerosol material. This study describs the generation and measurement techniques for radon and its progeny, in order to make available an accurate calibration facility for radon concentration and to allow investigations covering fundamental questions of physics with regard to the behavior of radon and its progeny as a function of the environmental parameters. Furthermore, we try to control the monodisperse aerosol with a differential mobility analyzer.

The study of behaviour of radon and its decay products in the low layer of the troposphere

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Abstract

Radon and its decay products are often used like tracers of dynamic processes in the atmosphere. They are used to study climatic phenomena and to produce climatic models scores of the time.

The source of radon in the atmosphere is the bedrock. The transport of radon from soil could be characterized by radon flux; UNSCEAR (2000) presents the estimation of the worldwide average of radon flux 0.016 Bq·m^{-2}·s^{-1}. The value of radon flux in one place isn't in the time constant - it is influenced by three fundamental parameters: the concentration of radon in bedrock, the soil moisture and permeability of soil. Methods of determination of radon flux are very sensitive on interpretation of gained results in the dependence on the atmospheric condition for individual measurement.

The concentration of radon and its decay products isn't in the low layers of the atmosphere constant - it is strongly influenced by dynamics of the atmosphere. The main factors which influence the concentration of decay products are dry and wet depositions. Generally it is possible to observe differences between concentrations in the day and in the night and in cold and warm parts of the year. The mean concentration of radon outdoor are about 10 Bq·m^{-3} according to UNSCEAR (2000).

The research in this area was supported by project of SUJ 200402, investigation of a vertical and horizontal distribution of radon and its decay products in the atmosphere was realized among others.

Dose

Only a minority of mankind could be characterized as workers of category A (B) and a negligible number of workers is employed in waterworks and other NORM workplaces; therefore, we take into account the remaining men and women.

It is estimated [1], that the inhalation of radon decay products (RnDP) represents an essential part of human irradiation (dose).

The "radon" influence greatly differs for individual states. Our detectors for radon measurement were applied in Kuwait and the concentrations were many times lower than European results. Values in ships cruising to the South Pole were also one one-hundredth of the values measured in Central Europe.

Not only do thousands of kilometers or miles play a role, an Austrian village Umhausen is split due to different geological structures in two radon levels.

Czech Masiff

A very small country in Central Europe, the Czech Republic – CZ, "suffers" due to an influence of tectonic processes ending 5.10^6 years ago. Many years after (1985) it was organized representative research led to a mean of indoor EER (energy equivalent radon concentration connected with RnDP) of 55 Bq·m^{-3}. It could be transformed to a year's average of radon concentration of 140 Bq·m^{-3}; probably the largest mean in the world.

The method of this estimation was very simple – an application of a solid state nuclear track detector (SSNTD): Kodak LR115 it this wave "bare" without the diffusion chamber. The choice of application in dwellings was governed by electricity consumers list.

The Task

The question:
When values of radon in dwellings are so high in Central Europe, what happens to the concentration outside? The indoor concentration is influenced not only by bad insulation of houses, but mostly by intensive radon flux from the Earth's surface.
(It has been published that radon concentration outdoor in Masiff Central in France is sometimes more than 1000 Bq·m^{-3}!)

Therefore, this is one of the reasons why the Czech State Office for Nuclear Safety (SONS) supported our outdoor measurement (the project SUJ 200402).

For this reason we developed methods for the estimation of instantaneous values of EER ("grab sampling" – called BUHS), pseudocontinuous measurement (TS96), and integral month mean estimation (STANTOM).

The output – distance

The results of measurement were interesting in some cases:

The distribution of instantaneous EER was estimated in heights of 0.3 m to 20 m as homogenous (measurement on a crane, on a view-tower). After the application of SSNTD, the radon concentration is seen as decreasing - see results after SSNTD application in diffusion chambers in heights of up to 40 m.

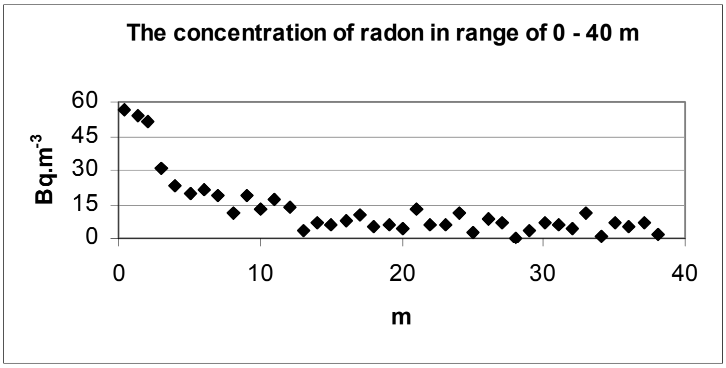

Fig.1. The concentration of radon in range of 0 – 40 m.

The distribution in the dimension of thousands of kilometers does not differ fundamentally – in Braunschweig (western Germany), the morning values were lower to compared to values in our country (the other points), but not dramatically:

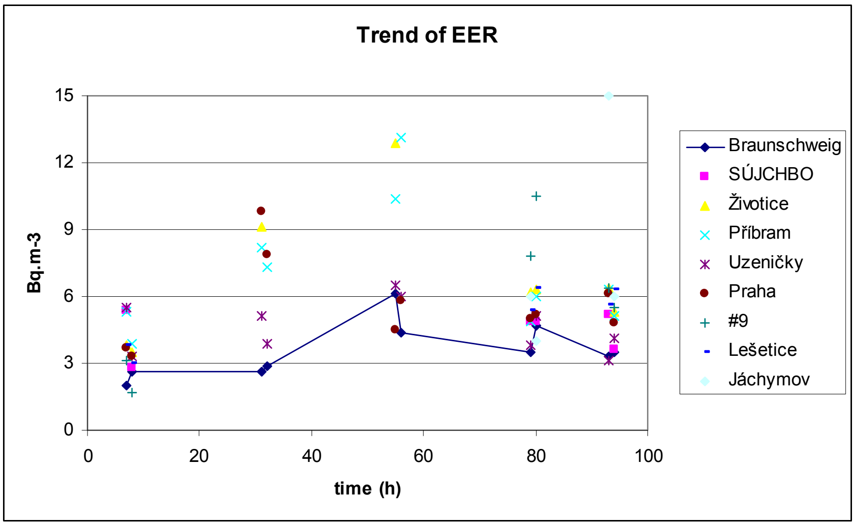

Fig.2. The trend of EER.

We didn't have an opportunity to measure in the Bering strait as our Japanese colleagues, but by chance we measured in Indianopolis (USA), on the coast of Finland and in central Romania. In most cases, the values and tendencies were similar. Of course, in the coastal regions the concentrations were lower due to negligible radon flux from the sea's surface. This hypothesis was supported by a more frequent number of measurements realized by the Dutch, when the influence of wind direction was seen as very important. The North Sea as well as other bodies of water are surely a big anti-source of radon.

We realized measurements around some radon sources or around absent source: old dump tailings originated in the Marie-Sklodowska era and ended up in the 1950s; the largest Czech lake; sludge beds near a reproducing plant. In these cases, no influences of radon sources or their absence were observed. Surprisingly, the mixing of air is probably a prevailing process. There is only one exceptional case – the neighborhood of active uranium mine exhaust, where radon (EER) is higher.

The output – time

Time (which is not always money) plays a role. The daily trend has been published many times. Only in the case of valleys (e.g. in Spain [2]), the tendencies are not typical. In all the cases of our location, there were only two systems observed:

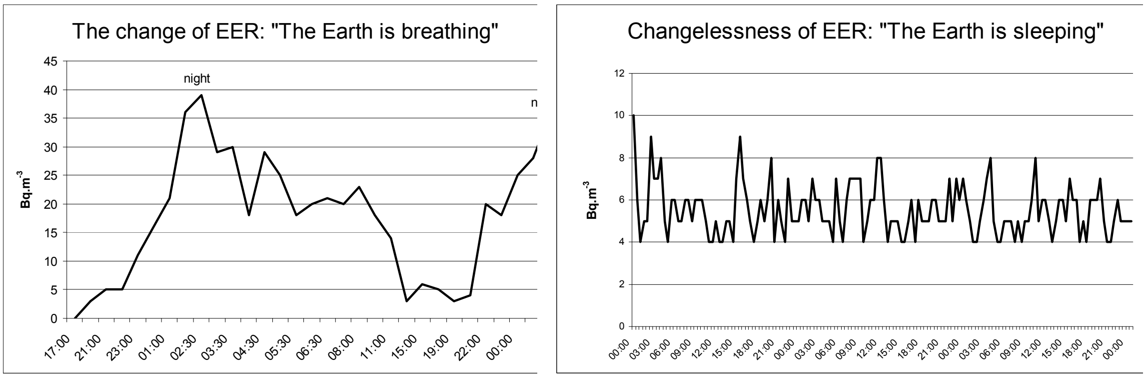

Fig.3 Different behavior of the Earth.

Appendix 1

Other radionuclide concentrations were measured, but the accuracy here is not sufficient for estimations with sufficient uncertainty:

- Concentration of thoron (^{220}Rn) daughter products is less than 0,3 Bq·m^{-3}
- Concentration of long-lived alpha-emitting radionuclides. The mean result published by the State Institute for Radiation Protection – SURO [3] after a huge volume sampling is 0.5 mBq·m^{-3}, and our results are lower - 0,2 mBq·m^{-3}.

Preliminary results of F (ratio of EER and radon concentration) show to values near to 0.3. The ratio of RnDP unattached to aerosol to total RnDP is approximately 0,1.

Appendix 2

There were about 200 000 miners employed in the uranium industry and the registration of their lung cancer incidence and their intake was relatively accurate. This is a reason for including Czech data into metastudies

The described differences between parts of our country also allow us to realise epidemiological studies between inhabitants.

Appendix 3

In 1985 the Organization for Economic Cooperation and Development organized a world-wide comparison to define reference laboratories [4]. As far as we know, more of them do not yet exist (the Australian ARL Yallambie is the only one). One can see again some international intercomparison tendencies. One of them [5] showed us that our measurement could be compared to most results of European laboratories.

Appendix 4 – opinion

Theories exist that radon outdoor is possible to estimated after gamma dose measurement. Our experiences showed us that the correlations between measurable units (gamma dose - radium concentration – radon in soil – radon flux – radon outdoor – EER outdoor) are very weak. These hypotheses were supported by many field measurements.

Appendix 5 – time spent indoors

In our country only 0.03 of inhabitants are involved in agriculture and forestry.

Are there countries around the world where the whole population does not stare at the TV most of evening? After our sociological mini-research, the time spent outdoors in the CZ is 0.07 - lower than the global mean published by ICRP.

The problematic question is the duration of stay in cars and trucks. After commented research, an individual spends 0.05 in vehicles. Of course, due to air-condition, the EER is reduced by factor 2.

Conclusion

It is not surprising, that the intake (and therefore the contribution to dose) for an inhabitant outdoor is not very important.

Taking into account the usual transformation 6 nSv/ ($Bq \cdot h \cdot m^{-3}$) – [6], the dose for inhabitants being indoor is 3 mSv per year and for being outdoors only 0.015 mSv per year. The facts of lower radon concentration of an afternoon were included into the calculation and other sources were not impacted.

Another important fraction of the effective dose is of course the photon irradiation - 1 mSv per year is a rough estimation of this part of the effective dose. We did not see any essential difference between the situation outside and inside buildings.

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Dosimetric calculations for uranium miners exposed to radon gas, radon progeny and long-lived radionuclides

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Abstract

In support of epidemiological studies involving pooling of data for German, French and Czech uranium miner studies, the individual annual doses from the year of first employment to 1999 were required. The annual absorbed doses to regions of the lung, red bone marrow, liver and kidney of each individual miner within the cohorts were calculated. The annual absorbed doses arising from the alpha radiation alone were calculated as well as the absorbed doses arising from the mixture of alpha, beta and gamma radiation. The doses arising from exposure to radon progeny, and to the long-lived radionuclides in the uranium ore dust have been calculated with biokinetic and dosimetric models developed by the International Commission on Radiological Protection. The organ doses arising from the inhalation of radon gas alone have been evaluated using published dose coefficients that were calculated with a specific pharmacokinetic model. The external gamma dose has also been measured and is included in the calculation of the total absorbed dose to specific organs. In this paper the methodology and the parameter values chosen for different exposure scenarios are presented as well as the results of organ dose calculations.

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Doses of *In Utero* and postnatal exposure to the Techa river offspring cohort

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Abstract
This paper presents estimates of doses from external exposure and intake of radionuclides for a cohort of about 20,000 offspring, whose parents were exposed in the early 1950s from discharges of liquid radioactive wastes into the Techa River (Southern Urals, Russia). Bone-seeking ^{90}Sr was the main contributor to the internal haemopoietic tissue doses received *in utero* and during infancy following maternal ingestion of the radionuclide before or during pregnancy and breastfeeding as well as postnatal exposure from direct ingestion. To provide reliable estimates of doses from ^{90}Sr, biokinetic models developed by the International Commission on Radiological Protection in Publications 88 and 95 were specifically adapted to the Techa River populations to describe strontium transfer to the foetus and breast milk. These models were validated by direct measurements of ^{90}Sr in the foetal skeleton, breast milk and maternal skeleton. To refine dosimetric model for the *in utero* and infant periods, a series of hybrid computational phantoms were developed for the 10, 20, and 30-week foetus and a newborn, from computed tomography (CT) and magnetic resonance imaging of preserved foetal specimens of similar gestational ages and a 6-day neonatal female cadaver. MicroCT imaging of autopsy specimens of the newborn skeleton allowed consideration of the 3D microstructure of marrow cavities and bone trabeculae across the newborn skeleton. Features of *in utero* haemopoiesis were considered and taken into account in dose estimates. Doses from postnatal exposure were evaluated with the Techa River Dosimetry System TRDS-2009D, taking account of ingestion of $^{89,90}Sr$ in breast milk and ^{137}Cs intake in cows' milk. Improved assessments of *in utero* and postnatal doses will be used in epidemiological studies to evaluate risks from chronic exposures in early life.

This work has been funded by EC (contract FP6-516478) and the US DoE Office of Health Programs.

Application of the new ICRP reference computational phantoms to internal dosimetry

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Abstract

In nuclear medicine, radiopharmaceuticals are incorporated into the body and distributed through biokinetic processes. Thus, each organ can become a source of radiation delivering a fraction of emitted energy in tissues. Therefore, internal radiation dose must be calculated accurately and realistically. Until now, the absorbed doses were derived from standard mathematical phantoms, in which different regions are defined by complex equations. An alternative class of anatomical models called voxel phantoms was then developed to offer realistic geometries of the human anatomy.

In this context, the International Commission on Radiological Protection (ICRP) has chosen to adopt voxel models to represent the reference adult. Consequently, the Specific Absorbed Fractions (SAFs), first step allowing the absorbed dose calculations, has to be re-evaluated for the new ICRP reference phantoms.

SAFs were thus evaluated for monoenergetic photon and electron sources using OEDIPE software and MCNPX Monte-Carlo code. The SAFs values were then validated by a comparison with those obtained by the German Center for Environmental Health (HelmholtzZentrum München) using EGSnrc Monte Carlo code.

The results show a general agreement for photons and high-energy electrons with differences lower than 8% between both laboratories. Nevertheless, significant differences were found for electrons at lower energy.

Given these results, the impact of the new ICRP reference phantoms on the absorbed doses per incorporated activity to patients from radiopharmaceuticals was estimated. Absorbed doses were calculated for the [^{18}F] FDG and [^{99m}Tc] ECD in 25 organs for the new ICRP reference computational phantoms. The dose estimates are overall similar to those of the mathematical phantom except for some particular organs (urinary bladder or uterus) where observed differences can be up to a factor of 2.

Introduction

In nuclear medicine, internal radiation dosimetry specifically deals with the deposition of radiation energy in tissue due to the incorporation of a radionuclide within the body. However, unlike external radiation dose (which can be measured), internal radiation dose must be calculated accurately and realistically either for diagnostic or therapeutic applications.

Until now, the method to calculate absorbed dose was based on the MIRD formalism and standard mathematical phantoms, which different regions are defined by complex equations. Then, through the evolution of computers' power, an alternative class of anatomical models called voxel phantoms was developed. They are based upon three-dimensional images techniques such as magnetic resonance imaging or computed tomography and provide detailed information about human anatomy.

Therefore, the International Commission on Radiological Protection (ICRP) has chosen to adopt voxel models, based on realistic images, to represent the reference adult. Consequently, the different dosimetric parameters must be reevaluated. Prior to the determination of absorbed doses, the Specific Absorbed Fractions (SAFs) need to be calculated for the new ICRP reference phantoms.

In this context, a fruitful collaboration between the French institute of radiological protection and nuclear safety and the German center for environmental health allowed the evaluation and the comparison of the SAFs for monoenergetic photon and electron sources using the Monte-Carlo codes MCNPX and EGSnrc, respectively.

Material and methods

1. The new ICRP reference phantoms

Both dose assessments due to radionuclides and SAF calculations for monoenergetic photons and electrons were performed using the ICRP reference adult male computational phantom (RCP-AM) and reference adult female computational phantom (RCP-AF) (Zankl et al. 2007, ICRP 2009). These models developed by the HelmholtzZentrum München are based on existing voxel phantoms (REX and REGINA) derived from CT tomographic data of real individuals. Each of RCP-AM and RCP-AF has 141 organs and tissues whose dimensions and masses are consistent with the ICRP reference anatomical and physiological parameters of ICRP publication 89 (ICRP 2002).

2. OEDIPE

The software OEDIPE was used in order to perform the calculations. OEDIPE is the French acronym for "tool for personalised internal dose assessment". It was developed at IRSN on the basis of a friendly graphic interface developed using the Interactive Data Language (ITT Visual Information Solutions, USA) (Chiavassa et al. 2005, 2006). The main purpose of this software is to estimate, in case of internal contamination or administration of radiopharmaceuticals, the personalised dose delivered by associating voxel phantoms and the MCNPX Monte-Carlo particle transport calculation code (LANL, USA) (Waters 2002). It can provide either the dose distribution at the organs' level or at the voxel level.

3. SAFs calculation

The RCP-AM and RCP-AF phantoms were integrated in OEDIPE to create automatically the MCNPX input file. Monoenergetic photon and electron sources were uniformly distributed in the source organs. In all the simulations, 5 million histories were generated and 15 energies of emission were chosen ranging from 0.01 to 10 MeV. The transport of secondary particles was taken into account. The tally *F8, energy deposit tally, was used to score absorbed energy in selected target organs. The SAF values were then derived for each source and target region using equation 1:

$$SAF(r_k \leftarrow r_h) = \frac{E_k / E_h}{m} \qquad (1)$$

Where r_h is a source organ, r_k is a target organ, E_k is the energy absorbed in r_k, E_h is the energy emitted from r_h and m is the mass of the target organ.

SAFs for the different source and target organs and for both, photons and electrons, were calculated and compared with those of the HMGU using the EGSnrc Monte Carlo code (Kawrakow and Rogers 2003). A comparison was also made between photon SAFs calculated using the new ICRP voxelized reference phantoms and those previously calculated for the ORNL male and female stylized phantoms. Finally, electron SAFs were compared to the former assumptions used by the ICRP and MIRD committees (Snyder et al. 1978, ICRP 1978). Indeed, for electrons, if the source and target regions are the same, the radiation is assumed to be entirely in the source organ. If the source and target regions are different, the specific absorbed fraction is reduced to 0. And in the particular case of walled organs, it is assumed that the Φ_i is equal to half the mass of the source organ.

4. Dose assessment

As it will be discussed further in the result section, the use of the new ICRP voxelized phantoms had led to differences in the photon and electron SAFs. Thus, it was interesting to study if the same influence could be observed for the dosimetry of radiopharmaceuticals.

When incorporated radionuclides are considered, biokinetic processes cause a distribution of the substance in the body, and, consequently, several organs of the body become source organs.

For a given radionuclide, the absorbed dose D_{rk} to a target region r_k from radiation emitted from a source region r_h is defined by the MIRD Committee (Loevinger et al 1991) and calculated using the following equation:

$$D_{rk} = \sum_h \tilde{A}_h \, S\,(r_k \leftarrow r_h) \qquad (2)$$

Where $\tilde{A}_h$ is the cumulated activity in source region (mCi h or MBq s) and equal to the total number of nuclear transformation in r_h; $S\,(r_k \leftarrow r_h)$ is the S factor which is defined by the following equation:

$$S = \frac{\sum_i k\,(n_i\, E_i\, \phi_i)}{m} \qquad (3)$$

In the MIRD equation 3, the factor k is a proportionality constant (rad g/mCi h MeV or Gy kg/MBq s MeV); n_i= number of particles with energy E_i emitted per nuclear transition; E_i=energy per particle (MeV); ϕ_i= fraction of energy absorbed in the target; **m**=mass of target region (g or kg).

In our present work, the organ absorbed doses were evaluated using OEDIPE software. OEDIPE takes into account the spectra data of several radionuclides derived from the publication 38 of the ICRP. The former assumptions used for electrons were not therefore considered.

Using OEDIPE, the sources of radiations were uniformly distributed in the organs of the voxel phantoms proportionally to the cumulated activity of the radionuclide, published in the publication 106 of the ICRP. An input file is then automatically created and is ready to be run with MCNPX. After the Monte Carlo calculation, the output file is processed by OEDIPE to obtain the absorbed doses per unit activity (mGy/MBq). Knowing the initial activity incorporated, absorbed doses are consequently obtained for 25 selected regions. Effective dose E was calculated from the estimated organ doses and tissue weight-factors, according to the formula set out in CIPR 60.

The absorbed and effective doses were calculated for 2 commonly used radiopharmaceuticals: the [^{18}F] FDG and [^{99m}Tc] ECD according to the following procedure:

For a contamination due to the administration of the [^{18}F] FDG, the activity is distributed in the pre-defined source organs: lungs, heart wall, brain and the liver as described in the publication 106 of the ICRP. The remaining activity is uniformly distributed in the other organs and tissues. Our definition for “Other organs and tissues” excludes the colon contents, the stomach contents, the small intestine contents, the gall bladder contents, the urinary bladder contents, the air inside the body and the skin at top and bottom, in addition to the defined regions.

The same assumption is also done for the [^{99m}Tc] ECD concerning the distribution of radioactivity in the “Other organs and tissues”.

Absorbed and effective doses calculated for the 2 radiopharmaceuticals using voxel phantoms were finally compared to those assessed using the mathematical phantoms.

Results and discussions

1. SAFs calculation

a. Photon SAFs

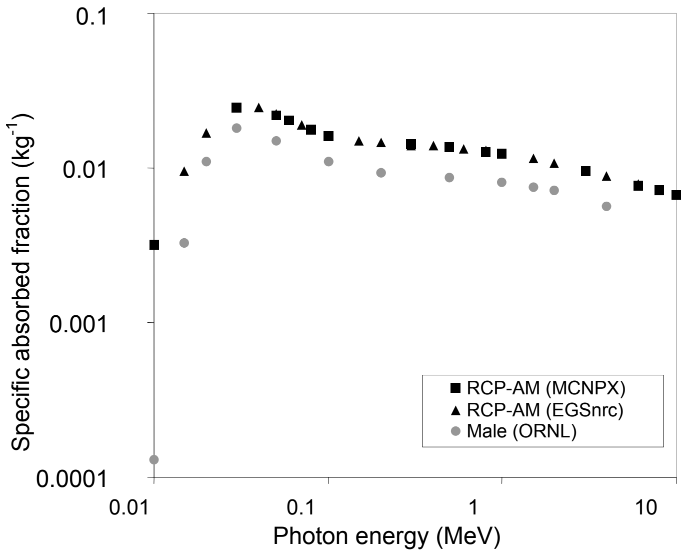
The source organs chosen are the lungs, the thyroid and the liver. The colon wall, the stomach wall, the liver and the breasts were selected as target organs. Some selected examples are given to illustrate our results.

An overall good agreement was obtained between SAF values assessed by MCNPX and EGSnrc. Figure 1 shows that, for photon sources in the liver, SAFs in the lungs agreed within 4 % between the two Monte Carlo codes. Similar results are obtained for all the sources and targets studied.

The SAF (Lungs ← Liver) values estimated by MCNPX and EGSnrc using voxel phantoms are also compared in figure 1 with those calculated for ORNL phantoms. Figure 1a shows that SAFs calculated in the new reference male computational phantom diverge from those calculated by Cristy and Eckerman (Cristy and Eckerman 1987) for mathematical models. Indeed, for energies below 300 keV, the ratio between the SAFs calculated for voxel and mathematical phantoms can reach 25. For higher energies, this maximum ratio is reduced to 1.5. Over the entire energy range, the results show that SAFs calculated with mathematical models are underestimated, especially at low energies. As it is shown in figure 2b, MCNPX and EGSnrc estimations for the female reference computational phantom are also different from those of the stylized model of Stabin (Stabin et al.1995). The difference can reach a factor of 18.3 at 10 keV and is reduced to 1.2 at 100 keV.

The observed discrepancies are due to the different shapes and distances between the organs of the mathematical and voxel phantoms, whose influence is predominant at low energies. Indeed, the interorgan distances are larger in the mathematical phantoms than in reality, due to the simplified organ shapes of the stylized phantoms that were required owing to the limited computational capacities available at the time when these models have been developed.

(a)



(b)

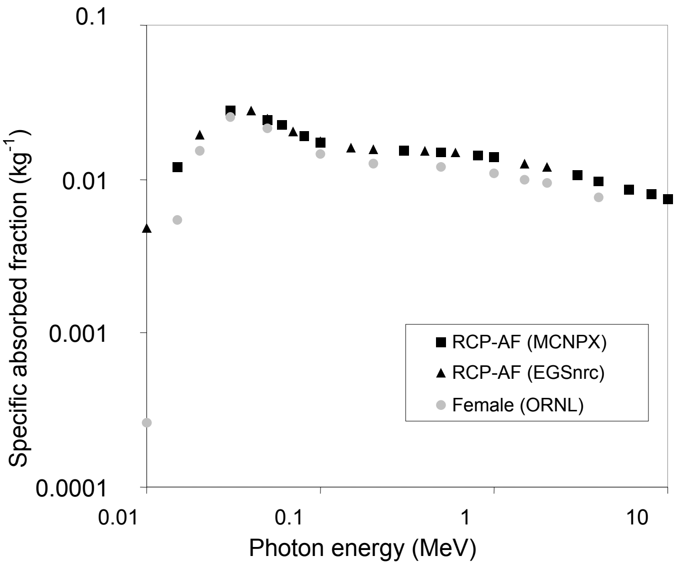


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Fig. 1. Photon SAFs (Lungs ←Liver) for the adult (a) male and (b) female voxel (MCNPX / EGSnrc) and stylized (ORNL) phantoms.

b. Electron SAFs

Figure 2 illustrates the comparison of the electron SAFs calculated at HMGU and IRSN for RCP-AM and RCP-AF in the case where the source organ is the lungs and the target is the breasts. A good agreement is obtained for energies higher than 500 keV. Nevertheless, for distant source and target regions, SAFs calculated by the HMGU and IRSN can differ by several orders of magnitude for energy electrons lower than 500 keV. In some cases, variations exceed 100 % at 50 keV and 60 keV. The discrepancies between the SAFs for the low energy electrons are mainly attributed to the different low-energy electron transport algorithms used by EGSnrc and MCNPX due to high Monte-Carlo statistical uncertainties for energies deposited in distant organs.

Even though electrons are considered as weakly penetrating radiation, another simulated result shows that irradiation of adjacent regions cannot be always neglected. Indeed, in figure 2, when former ICRP and MIRD approximations were applied, SAFs (Breasts ← Lungs) were supposed to be equal to 0 over the full energy range. However, Monte Carlo simulation demonstrates that, for energies ranging between 6 MeV and 10 MeV, SAFs would not be negligible in the reference computational phantoms (about 0.02 for the male phantom and 0.034 for the female phantom at 10 MeV).

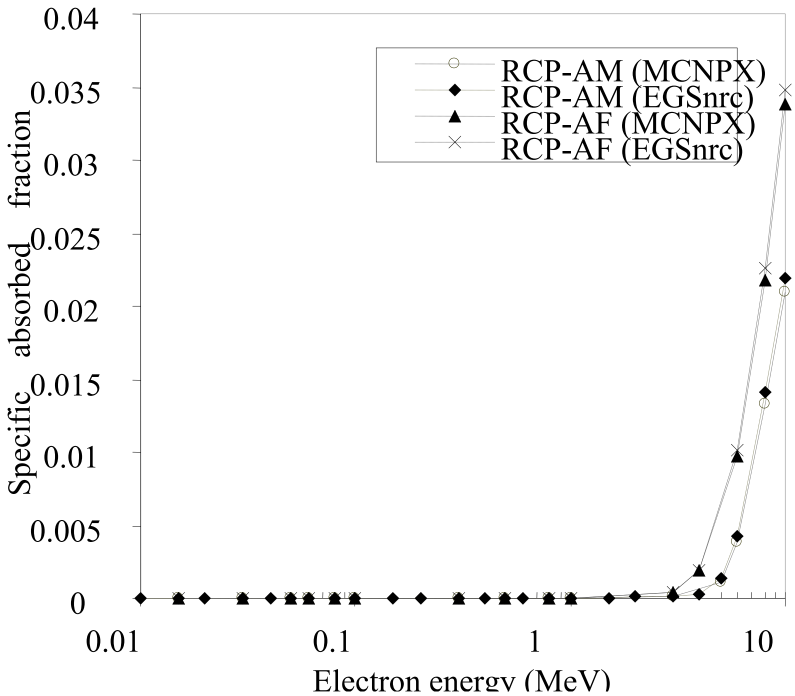

Fig. 2. Electron SAFs (Breasts <- Lungs) for the adult reference computational phantoms. With the former MIRD and ICRP approximation of non-penetrating electrons, this would always be 0.

2. Dose assessment

As an example of the results, figure 3 shows comparative values of some organ-absorbed doses and effective dose for the adult computational and mathematical reference phantoms for the [^{18}F] FDG radiopharmaceutical. Absorbed doses evaluated using the new ICRP reference phantoms were quite similar to those calculated in the ICRP 106 using mathematical models. Nonetheless differences were noticed for some particular organs. Indeed, the dose absorbed by the bladder of the mathematical phantom is higher by a factor 2.4 to the dose calculated for the male voxel phantom and by a factor 2.3 to the one evaluated for the female voxel phantom.

The same observation is valid for the absorbed doses due to the incorporation of [^{99m}Tc] ECD. In fact, the dose absorbed the urinary bladder shows significant differences by a factor 2 between the absorbed doses calculated for the mathematical models and those calculated for the adult computational phantoms.

The discrepancies observed in the absorbed dose values are mainly due to differences in the geometry and organ distances between the mathematical and the voxel phantoms and to the approximations previously used by the MIRD committee.

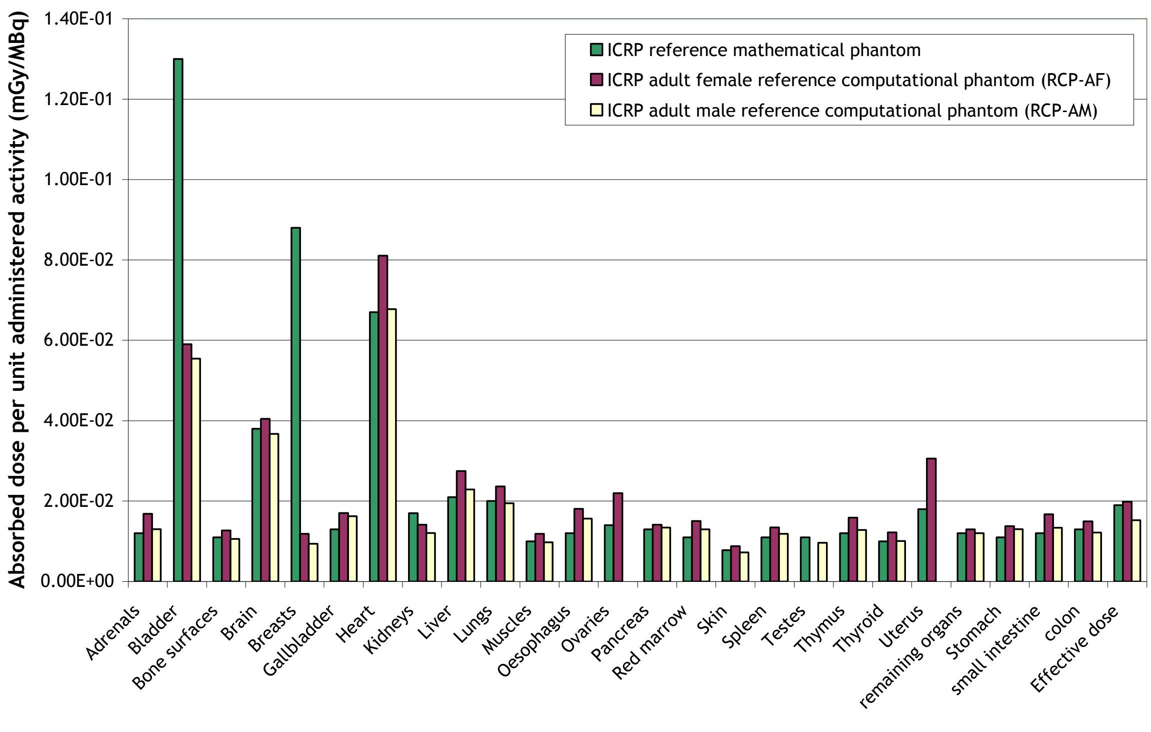

Fig. 3. Organ-absorbed doses per unit activity administered for [^{18}F] FDG.

Conclusions

The aim of this study was to show the effect of the new reference computational phantom recently adopted by the ICRP in internal dosimetry. First, a comparison between photon and electron SAF values calculated at the HMGU and IRSN was achieved for the male and female reference voxel phantoms with two different Monte-Carlo codes (EGSnrc and MCNPX respectively). Photon and high energy electron SAFs evaluated for many source-target organ pairs agree well between the 2 institutes. However, larger differences are noticed for low energy electrons (<500 keV) due to different physical models and algorithms used in the Monte Carlo codes. Furthermore, absorbed doses per unit administered activity were calculated for both [^{18}F] FDG and [^{99m}Tc] ECD in 24 defined organs and the remaining ones for the new ICRP reference computational phantoms. The dose estimates are overall similar to those of the mathematical phantom. However, differences are found between doses absorbed in particular organs. Indeed, for the urinary bladder, the absorbed doses published in the ICRP publication 106 are almost 2 times higher than the dose evaluated for the voxelized phantoms. Differences are also noticed for the uterus where the absorbed dose calculated using the computational phantom is 70 % higher than the dose calculated using mathematical models. The reasons of such discrepancies may be due to the different geometries of the organs in the 2 phantoms and to the approximations used in the past for mathematical phantoms. Careful investigations will help to explain the observed discrepancy.

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The guidelines for internal dose assessment after “malevolent use” incidents presented in TMT Handbook

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Abstract

The main objectives of internal contamination monitoring after a malevolent use incident are to quantify absorbed doses to organs for people who may be subject to deterministic health effects because of very high exposures, and to quantify committed effective doses for people with lower levels of internal contamination. Rapid assessments of monitoring measurements are likely to be needed for two reasons. First, urgent decisions on follow-up actions may have to be taken (e.g. on therapeutic treatment of life-threatening radiation exposures, or on the need for more accurate monitoring to allow decisions on decorporation measures to be made). Second, assessments of internal dose may be needed to support treatment and decorporation measures and to provide information to individuals and others on potential effects on health. The simple action level scheme proposed in TMT Handbook (published in 2009) is described. The action levels are related to dose, but are expressed in terms of measured quantities so that direct comparison with monitoring results can be made, allowing rapid decisions to be made on follow-up actions. The use of the simple “look-up” tables provided in TMT Handbook for quantitative assessment of internal doses is also described These tables allow the direct determination from monitoring measurements of both committed effective dose and absorbed doses to lungs, red bone marrow and the colon. The biokinetic and dosimetric model parameter values assumed for the calculations of the data contained in these tables are discussed. Dose assessment look-up tables are a potentially useful resource, particularly during the early stages of the response to an incident, but users should be aware of their limitations. The potential uncertainties in assessed dose arising from the use of default values for model parameters such as inhaled particle size, absorption type and gastro-intestinal uptake factor are discussed.

Improving ingestion dose modelling for the ARGOS and RODOS decision support systems: A Nordic Initiative

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Abstract

A Nordic work group under the NKS-B activity PARDNOR has revised the input parameters in the ECOSYS model that is incorporated for ingestion dose modelling in the ARGOS and RODOS decision support systems. The new parameterisation takes into account recent measurement data, and targets the model for use in Nordic preparedness. The importance of some of the revisions is illustrated.

Introduction

Most European countries have integrated either the ARGOS or the RODOS decision support system in their preparedness against nuclear accidents. The ingestion dose module in both of these two systems is based on the ECOSYS model, which was developed shortly after the Chernobyl accident. ECOSYS default parameters (see, e.g., Müller & Pröhl, 1993) are generally applied in dose calculations without much consideration of their applicability and representativeness for the particular case. However, the default values reflect southern Bavarian conditions with respect to for instance dietary habits, food import patterns, farm animal feeding regimes and crop growth seasons, and are far from representative of for instance Nordic conditions. Another problem with the ECOSYS default data is that much of the generic data describing the processes of mobility of contaminants is rather old, and does not reflect the current state of knowledge, e.g., following the host of post-Chernobyl investigations. This affects the quality of parameters such as deposition velocities for different types of particles, leaching rates, fixation rates, desorption rates, resuspension enrichment factors, natural weathering rates, transfer factors and biological half-lives for farm animals. It is also problematic that crop transfer factors have so far not been linked to a soil classification. Many of these shortcomings may greatly impinge on ingestion doses, and the parameters have therefore been revised by a Nordic work group under the NKS-B activity PARDNOR. This paper gives examples demonstrating the importance of both generic and location-specific parameter changes.

Methods and results

The text below gives a few examples of the work that was carried out to improve the quality of some generic parameters used in the ECOSYS model, as well as to accommodate specific Nordic values for location (country or region) specific parameters.

Dietary habits

Dietary habits can vary widely between countries, depending on, e.g., the climate and local tradition. For inclusion in the ECOSYS model (and thereby in ARGOS and RODOS) it is essential to obtain recent location-specific data for the consumption pattern, so that proper estimates can be made of the doses received by a local population in the event of an incident leading to contamination in food production systems. A survey was first made in all the Nordic countries to assess which data would be available for different age groups on the dietary composition. In-line with the outcome of this and to address as wide a range of discrete age groups as possible, thereby also complying with the methodology used in the ECOSYS model, it was decided to consistently focus on four age groups: infants and young children 1-4 y, teenagers <15 y, young adults (ca. 30 y), and more senior adults (ca. 60 y). For all countries, except for the Faroe Islands, fairly recent survey data are available for such age groups. For some countries separate data for each of the two genders are available. In some cases gender deviations of some significance have been noted, but such differences are not readily accommodated in the framework of the ECOSYS system.

Figure 1 shows a comparison of the consumption of wheat and rye flour in the different Nordic countries. Here it is seen that a typical Danish adult consumes about 3.3 times as much rye flour as does the typical Norwegian adult. Such traditional differences in diet are important to take into account in optimising efforts to reduce doses after a contaminating incident. However, as this relationship is 4.3 for the senior adults, but only 2.9 for teenagers, this particular difference between Norwegian and Danish diets seems to be declining, stressing the need for recent dietary data for the modelling. Fairly recent dietary data is available for most Nordic countries. The exception is the Faroe Islands, where the latest, and not very detailed dietary survey was conducted in 1981-82. Anyway, in the Faroe Islands and Iceland, these grain products are imported from abroad. It is also seen from Figure 1 that due to differences in climate, in the northernmost countries (Norway and Finland), practically all wheat used for consumption by humans is spring wheat, whereas in Denmark, Sweden and Germany, the majority of the wheat produced/consumed is winter wheat. This means that if a contamination occurs in the spring, the wheat plants will in Finland and

Norway have undergone very little development, whereas in the more southern countries, where sowing took place already in the previous autumn, and the warmer weather would lead to a more rapid plant development, the wheat plants could be quite mature, and receive a comparatively considerably larger contaminant deposition. This would greatly affect wheat flour contamination levels in the first year harvest.

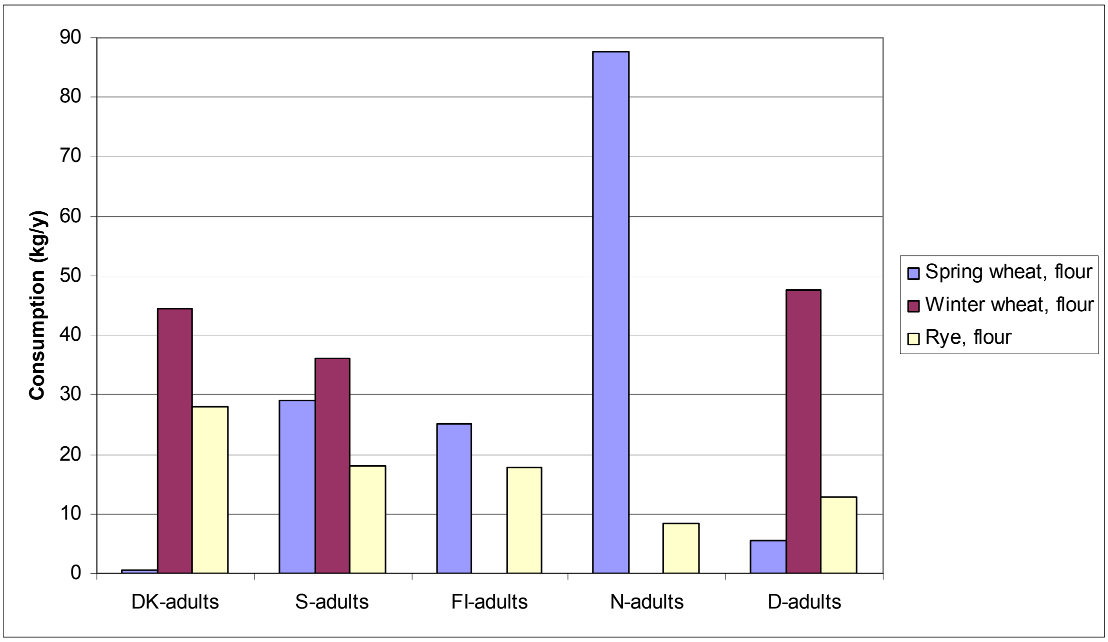

Fig. 1. Consumption of wheat and rye flour in the Nordic countries, compared with the German ECOSYS defaults (average figures for adults - ca. 30 y). DK=Denmark; S=Sweden; FI=Finland; N=Norway; D=German defaults.

Figure 2 shows a comparison of the consumption of potatoes, leafy vegetables and root vegetables in these countries. Also here there are great differences, as it would seem that the average Faroese adult consumes about 3 times as large amounts of potatoes as the average Icelander. Since the best available Faroese data is rather old (for these particular products from an estimate made in 1962), this difference should be considered with some caution. However, a new, unpublished rough estimate of the average for all age groups would be of the order of 70 kg per year (Bjarnason, 2007), indicating that this consumption is still comparatively very high. It is also seen that Danes consume relatively large amounts of root vegetables (particularly compared with Swedes), whereas Swedes and Germans seem to consume much larger quantities of leafy vegetables than do the inhabitants of the other considered countries. It should of course here be noted that consumption of directly contaminated leafy vegetables can give very high early phase doses.

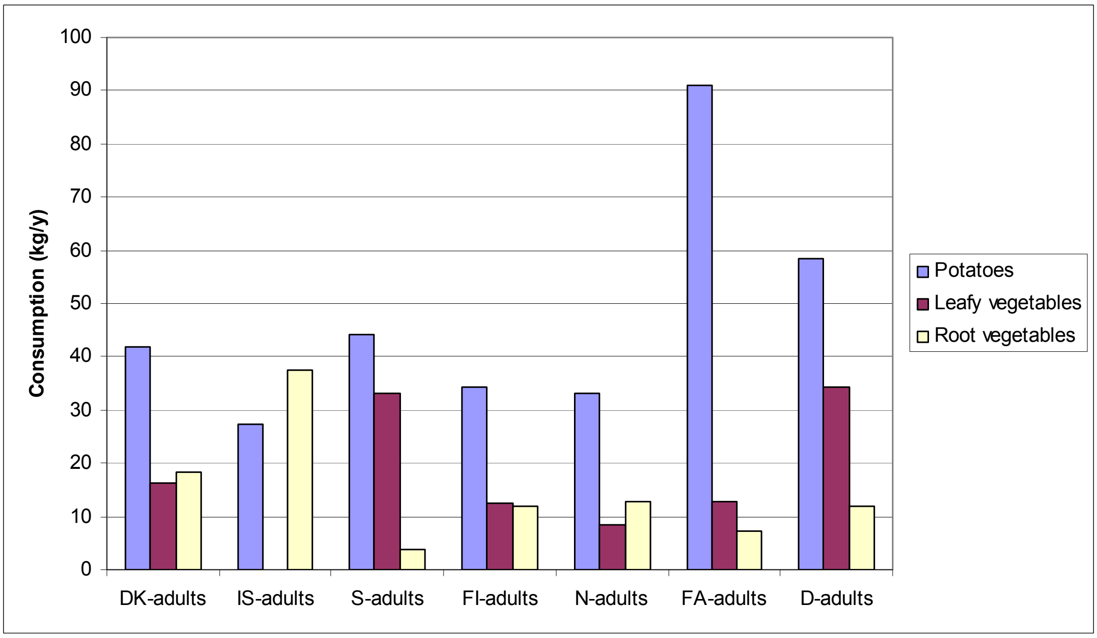

Fig. 2. Consumption of potatoes, leafy vegetables and root vegetables in the Nordic countries, compared with the German ECOSYS defaults (average figures for adults - ca. 30 y). DK=Denmark; IS=Iceland; S=Sweden; FI=Finland; N=Norway; FA= Faeroe Islands; D=German defaults.

Leaf area index

The leaf area index (LAI) is the factor in ECOSYS which determines the state of growth of all vegetation. This can have great bearing on the deposition (dry and wet) to crops, and thus also greatly influence ingestion dose in general. Since LAI depends strongly on the climate (soil temperature variation), it is essential to identify LAI data that adequately represent the location to be modelled. For instance, the Danish Institute of Agricultural Sciences has, based on large amounts of measurement data, developed a simple empirical model / data set describing the seasonal variation of LAI for a number of different crops (Plauborg & Olesen, 1991; Olesen, 2006). The key variables are here the sowing time, soil temperature and harvest time. Values are given for normal and low fertilisation status. The model may thus be applied for different climates, and should be applicable also for other Nordic conditions, where the variation over the year in soil temperature is known. An example of a temperature sum and LAI data set is given in Table 1 for spring barley, with normal fertilisation level. TSum represents a summation over days multiplied by the average daily values of temperature (degree days). Based on the time-variation of TSum at a given location, a date can be associated with each TSum in the table below.

To test the general validity of the model for description of LAI development in Nordic areas, an annual soil temperature variation measurement dataset was obtained from the Swedish University of Agricultural Sciences in Uppsala (SLU, 2008; Kyllmar & Johnsson, 2006). Compared with a Danish data set for the location Herfølge, the Swedish temperatures are over the entire year some 1-5 °C lower. The results of applying the soil temperature data from Uppsala in the Danish model for barley can be compared with measured values of LAI for barley in Uppsala, at two different times of the year: the 18th of June and the 13th of July, 1995 (Thorgeirsson & Søgaard, 1999). Fig, 3 shows the development of LAI over a season, modelled on the background of the Danish dataset and the soil temperature data reported for Uppsala, assuming respectively normal and low fertilisation status. As can be seen, the model curves are in good agreement with the measured LAI data for the Uppsala location (denoted by the triangles in Fig.3), indicating that the Danish model dataset may also be applied for other localities with (slightly) different climates. However, it should be noted that in the northernmost areas of the Nordic countries, crop sort varieties are sometimes specially selected to give rapid development, to make the most of the short growing season. Data for the soil temperature development over the year were compiled for localities in all Nordic countries.

Table 1. Relationship for spring barley at normal fertilisation level between TSum and LAI-total, according to the Danish empirical model.

	TSum	LAI-total
Sowing time	0	0
	110	0
	210	0.41
	310	1.16
	410	2.53
	509	5
	1180	5
	1590	2
After harvest	0	0.3

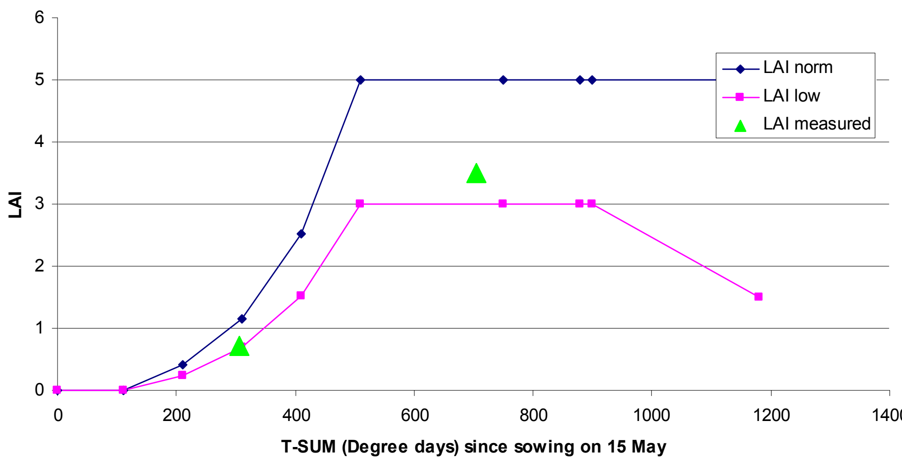

Fig.3. Seasonal variation in LAI for barley at Uppsala, Sweden. Model curves compared with two measured data points.

Fixation rates

In ECOSYS, fixation rate is a generic parameter governing the loss of bioavailability of the contaminants due to strong attachment of radionuclides particularly to clay minerals. Fixation half-lives are generally (with the exception of caesium and strontium) set to very high values (mostly 1000 years), to reflect the low tendency of fixation in soil of most radionuclides considered in ECOSYS. This seems reasonable, and dose estimates over 50 years (the longest period over which accumulated ingestion dose can be estimated in the excel version of ECOSYS) would not be affected much by refining these parameter estimates. Specifically for caesium, the selective fixation in clay minerals gives a much shorter fixation half-life. Here, the ECOSYS default value is 8.7 years. Like other ECOSYS parameters, this is based on old assessments, not adequately taking into account the many assessments made after the Chernobyl accident, in which the physicochemical forms of the contaminants are representative of those that can be expected in connection with a future large nuclear power plant accident.

Applying a modified version of the sequential extraction technique described by Tessier et al. (1979), it was suggested by Oughton et al. (1990) that the 'strongly fixed' fraction of a contaminant in soil could be defined as the part of the contamination that can not be extracted with the most inert solutions, but requires addition of strong acid to go into solution. Such modified Tessier extractions have also been applied by other workers, and it has become a standard technique for sequential extractions to assess the mobility in soil and sediments of contaminants like ^{137}Cs and ^{90}Sr. If it is assumed that the mobile (easily extractable) fraction decreases exponentially over time, sequential extractions carried out by various workers at different times after the contamination from the Chernobyl accident took place (e.g., Salbu et al., 1994; Andersson & Roed, 1994; Bunzl et al., 1997; Shand et al., 1994) on soil samples from Nordic, Ukrainian, Russian and other European areas with different characteristics, suggest that radiocaesium fixation half-lives would typically range between about 1.3 and 2.7 years for most soils with significant mineral phases, and some 4-5 years for very sandy or organic soils.

Consider an ECOSYS example scenario, where dry deposition of ^{137}Cs occurs on the 1st of January, and diets, import fractions and feeding regimes reflect Danish conditions, whereas the rest of the parameters are ECOSYS defaults. If the fixation half-life for caesium is here changed from 8.7 years to 2 years, ECOSYS would estimate the influence on ingestion dose components as shown in Table 2.

Table 2. Percentage change in ingestion dose contributions to adults integrated over resp. 2 and 50 years, by setting the ^{137}Cs fixation half-life to 2 years (scenario described above).

	Winter wheat flour	Fruits	Cow's milk	Beef (cow)	Total
2 years	18.5 %	23.7 %	0.5 %	0.5 %	0.5 %
50 years	67.6 %	69.9 %	7.7 %	8.0 %	9.5 %

The figures in Table 2 show that particularly some of the long term dose contribution estimates could be rather far off if a fixation half-life value of 8.7 years is used for a mineral soil.

The data for sequential extractions of strontium is sparser, but investigations by Oughton et al. (1990), Oughton et al. (1992) and Salbu et al. (1994) suggest a fixation half-life of respectively 20 years, 11 years and 23 years for soils with mineral content. This is in good agreement with the default fixation half-life of 20 years applied in the ECOSYS model.

Conclusions

A review of the ECOSYS model and its default parameters identified a number of points where elaboration is deemed necessary before ECOSYS should be applied for Nordic emergency decision support. For each of the Nordic countries, a dataset has been established describing the typical diets for four different age groups, ranging from young children to senior adults. Large dietary differences were found (also due to different food production conditions in the different Nordic countries), which could influence ECOSYS consumption dose estimates considerably. The applicability of a simple Danish empirical model for calculation of leaf area index as a function of soil temperature was positively tested for use for Swedish conditions (Uppsala soil temperatures) by comparison with a few reported measurements of leaf area index and corresponding soil temperature sums. Representative soil temperature sum values were collected for different locations in the Nordic countries, which could be applied to improve ECOSYS leaf area index parameterisation. The ECOSYS default value of the rate of radiocaesium fixation in soil was compared with a new estimate, taking into account more recent experimental data. The difference in this value could have considerable influence on long term consumption dose estimates. Also a large number of other ECOSYS input parameters have been examined, and improvements and targeting of these for Nordic conditions have been suggested by the same Nordic work group (see, e.g., Hansen et al., 2010)

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Ce-doped SiO_2 optical fibres for real-time dosimetry of external radiotherapy beams

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Abstract

The introduction of new technologies in radiation therapy aimed at improving the dose distribution to the tumour volume, calls for the development of innovative devices able to assure the quality of radiation beams and in-vivo real time dosimetry measurements.

This work describes the use of a novel glass composite fibre dosimeter for the characterization of different types of radiotherapy beams. The dosimeter is based on the radioluminescence emission of a small portion of SiO_2 fibre doped with cerium prepared by means of the sol gel and powder in tube techniques. The doped portion is connected by fusion splicing to a commercial optical fibre which is optically coupled to a photomultiplier tube operating in photon counting mode. The signal is processed by a circuitry and software properly designed. In order to subtract the Cerenkov contribution to the signal response a double optical fibre geometry is adopted.

The system was tested by using conventional radiotherapy photon and electron beams, as well as stereotactic fields generated by a 6 MV linear accelerator equipped with cone stereotactic collimators. Characterisation measurements were also performed using 138 MeV proton pencil beams at the Center for Proton Therapy of the Paul Scherrer Institute (PSI), using the discrete spot scanning technique.

The system showed interesting dosimetric properties in terms of sensitivity, reproducibility, linearity, energy and angular dependence. Moreover, the small dimension of the fibre and the possibility to perform real time measurements of the dose rate, proved to be particularly useful for a reliable evaluation of the output factors of small stereotactic fields and for resolving different scanning sequences of proton pencil beams.

Introduction

The development of new technologies for external radiation therapy has been impressively fast during the last few years. Intensity Modulated Radiation Therapy (IMRT) has become the standard methodology in many countries; the number of tomotherapy machines operating thought the world is continuously increasing as well as the centres and facilities dedicated to hadrontherapy. In fact, the introduction of novel concepts for radiation beam delivery can lead to an improvement of the treatment outcome by means of a dose distribution which more strictly conforms the tumour volume.

These progresses call for a corresponding improvement of instruments and methodologies employed for the maintenance of a satisfactory quality assurance system. Furthermore, the possibility to perform prompt checks of the delivered dose through in-vivo dosimetry measurements become more and more important when treatments involving dose escalation to target volume are planned.

This work reviews the characteristics of a novel radioluminescent (RL) dosimeter based on a glass composite optical fibre. The system was tested with photon and electron beams used in conventional radiotherapy, as well as with small stereotactic fields. The luminescent and dosimetric properties of the fibre were investigated also with proton beams through an active scanning irradiation regime.

The small dimensions, together with the possibility to perform real-time measurements of the dose, make the system an useful tool for assuring the quality of different types of radiation beams and for in-vivo dosimetry measurements.

Material and methods

The dosimetric system

The core of the dosimetric system consists in a portion of silica optical fibre doped with cerium, obtained by combining the sol-gel and powder in tube techniques. The synthesis procedure, developed at the Department of Materials Science of the University of Milano Bicocca and explained in detail elsewhere (Chiodini et al. 2003; Vedda et al. 2004), is here briefly summarised.

The sol solution was obtained with Tetraethoxysilane (TEOS) and Ce (III) nitrate as precursors in a solution of water and ethanol. The sol was placed in a thermostatic chamber at a temperature of 40 °C to activate the sol–gel transition. Xerogel powder was obtained by rapid drying in rotating evaporator and by further grinding in agate mortar to improve the grain size uniformity. Slow sintering up to 1100 °C in a quartz chamber gave the final glass powder, which was then treated in vacuum at higher temperature (1500–1600 °C) for some minutes in order to improve the rare-earth scintillation efficiency and eventually remove the OH content excess. The glass powder was finally introduced in a vacuum-sealed quartz tube (10^{-4}-10^{-5} mbar) and heated in a furnace at 2100 °C, applying a proper pulling speed to obtain the portion of doped fibre of the required diameter (range: 100- 200 μm). This portion was connected by fusion splicing (Starlite S.r.l, Italy) with a commercial optical fibre, which was optically coupled (Fraen Corporation S.r.l., Italy) to a photomultiplier tube (PMT) (Hamamatsu, R7400 P) operating in photon counting mode. The signal was processed by an acquisition unit properly designed (EL.SE S.r.l., Italy), mounting a secondary PMT for the subtraction of the Cerenkov light produced in an undoped fibre stretched parallel to the doped one. A picture of a prototype of the acquisition unit with the fibre sensor is shown in Fig. 1.

Fig. 1. Picture of a prototype of the acquisition unit (ELSE S.r.l., Italy) with the RL fibre.

The characterisation tests

Detailed descriptions of the set up and measurements performed to characterise the dosimetric system under different irradiation conditions has been reported elsewhere (Mones et al. 2006 and 2008; Veronese et al. 2010). Briefly, all the tests performed with radiotherapy photon and electron beams where performed at the Radiotherapy Department of the Hospital of Novara (Italy), using two medical linear accelerators: a Clinac 2100 CD and a DBX (Varian, USA) equipped with cone stereotactic collimators. The main properties investigated with these machines were:

- reproducibility and stability of the RL signal,
- dose response,
- dose-rate dependence,
- energy dependence,
- angular dependence,

Moreover, depth-dose profiles (PDDs), transverse profiles (TPs) and output factors (OFs) measurements of small beams were performed.

Irradiations with proton beams were carried out at the Center for Proton Therapy of the Paul Scherrer Institute (PSI), using a scanned proton pencil beam. Basic properties like reproducibility and dose response were tested also with protons; afterwards, the measurements were focussed on the capability of the system to resolve different scanning sequences and to measure the PDD of a mono-energetic proton beam.

Results and discussion

Fig. 2 shows typical RL signals versus irradiation time, and their reproducibility under the same experimental conditions. Typically, a reproducibility better than 1% can be obtained. The height of each curve (expressed in counts per seconds, cps) is proportional to the dose rate; the integrated area (corresponding to the total counts registered during the irradiation) is proportional to the dose. Consequently, the intensity of the RL signal increases with increasing the dose rate, as shown in Fig 3, but, for a fixed dose, the total counts remain almost constant, indicating that the dosimeter enables a reliable evaluation of the absorbed dose, independently of the dose rate used.

A linear increase of the total counts with increasing dose was observed, independently of the type and energy of the radiation beam. An example of dose-response curve is given in Fig 4. Moreover, the detector proved to be particularly suitable for the measure of PDDs and TPs, as shown in Fig. 5 and 6 respectively. In particular, the small dimensions of the doped fibre enable the correct assessment of the OF of small fields, like those employed in stereotactic radiosurgery. Fig. 7 shows the OF of circular fields measured with the doped fibre, compared to the results obtained using a micro-ionization chamber (Exradin A16, collecting volume 0.007 cm3*)*. A satisfactory agreement between the two sets of data was obtained, with a maximum discrepancy of the order of 1% in correspondence of the smallest field of 12.5 mm diameter.

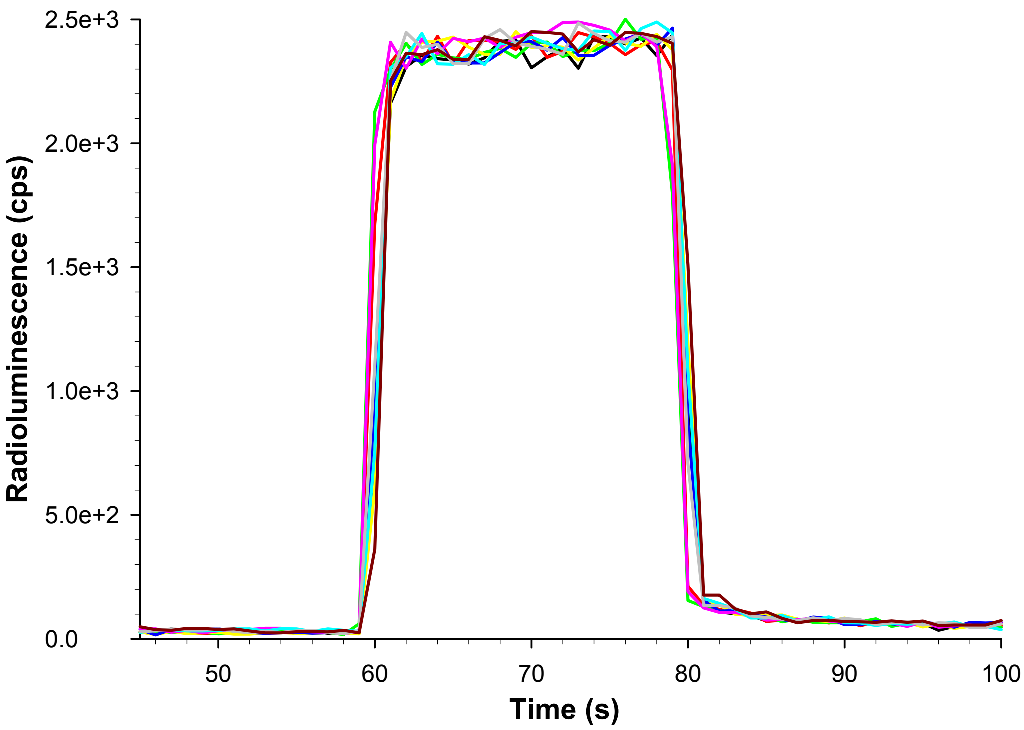

Fig. 2. Series of RL signals versus irradiation time measured with the doped fibre irradiated with 6 MV photon beams.

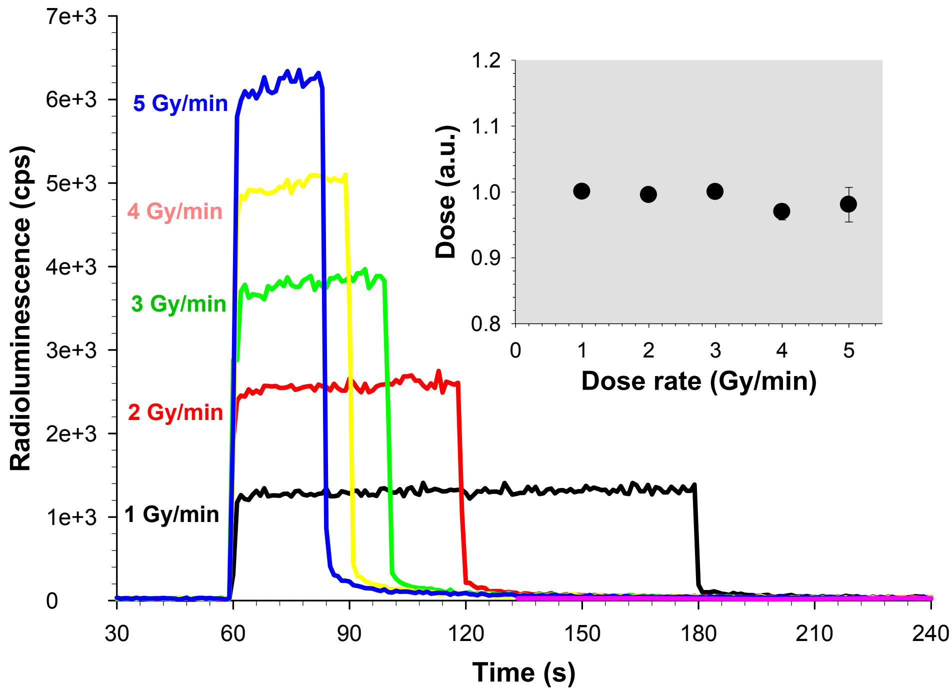

Fig. 3. RL signals versus irradiation time measured with the doped fibre irradiated with 6 MV photon beams, using different dose rates. The integrated areas proved to be independent of the dose rate as shown in the inset, where the measured dose is plotted versus the corresponding dose rate.

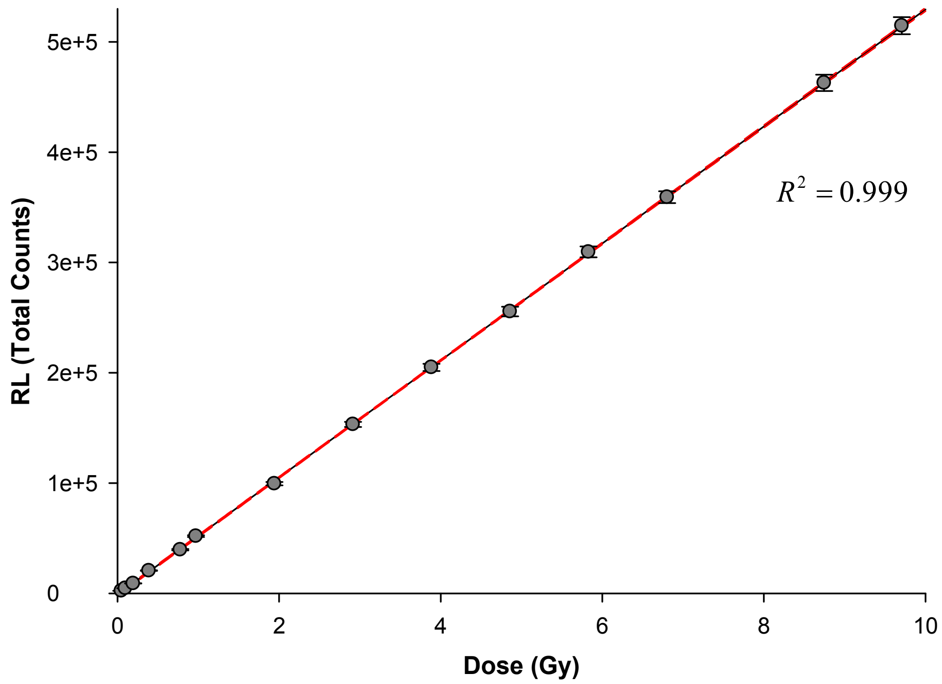

Fig. 4. Example of linear dose response of the fibre dosimeter.

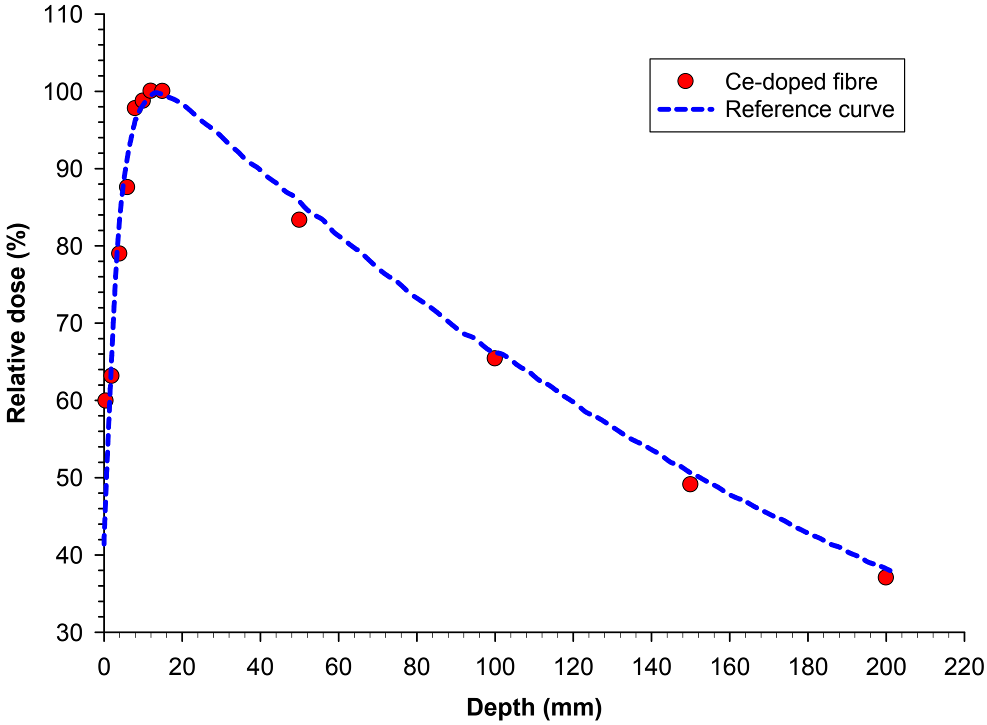

Fig. 5. Depth dose profile of a 6 MV photon beam in water measured with the doped fibre and compared with the reference curve obtained with a diode.

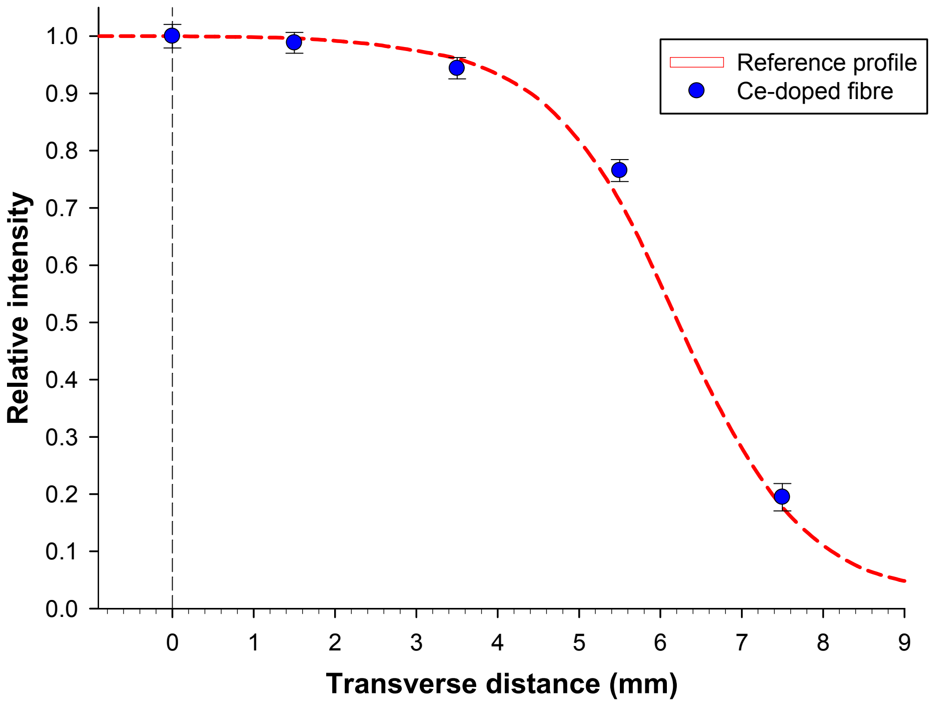

Fig. 6. Transverse profile of stereotactic field (12.5 mm diameter collimator) measured with the doped fibre and compared with the reference curve obtained with a diode.

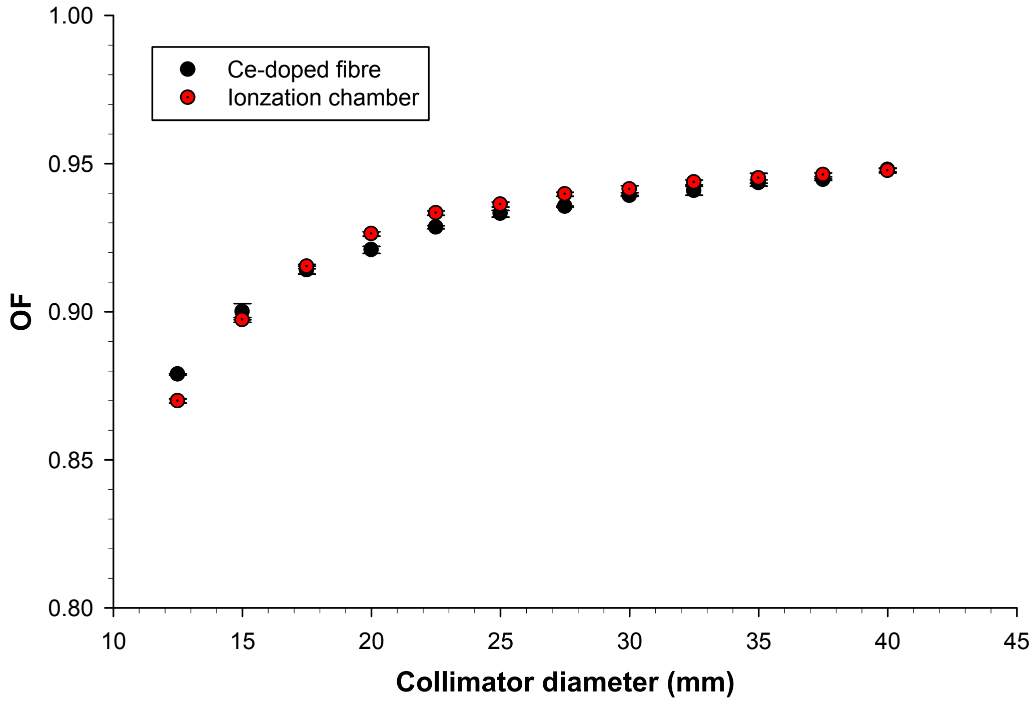

Fig. 7. Output factors of stereotactic fields measured with the doped fibre and compared with OF obtained using a micro-ionization chamber.

An interesting property of the system observed during the tests performed with proton beams is the possibility to discriminate the dose contributions due to the various spots composing the irradiation scanning sequence. An example of signal obtained by irradiating the doped fibre placed in the centre of a cubic irradiation volume (6 cm side) is shown in figure 8, where the modulation of the signal intensity reflects the temporal and spatial characteristics of the scanning sequence. Further tests attested the capability of the system to assess the total dose in one point independently of the sequence and spot duration used to deliver such dose.

As many other scintillators (Safai et al. 2004), also silica doped fibres suffer of a progressive decrease of the RL efficiency with increasing the Linear Energy Transfer (LET) of protons. This quenching effect is evident by comparing the PDD of a 138 MeV proton beam measured with the RL detector, with the corresponding curve obtained using a standard ionization chamber. The results of this analysis is shown in figure 9, where an underestimation of the dose of approximately 10% can be observed at the depth of the Bragg peak.

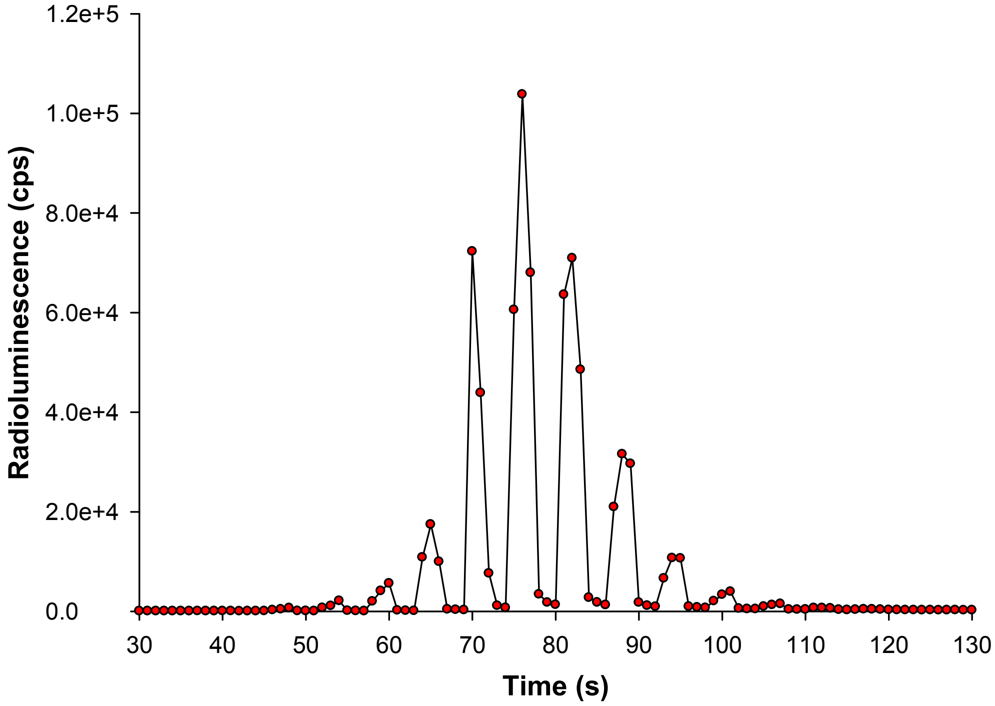

Fig. 8. Example of RL signal versus irradiation time measured with the doped fibre irradiated with a 138 MeV proton beam using the spot scanning technique.

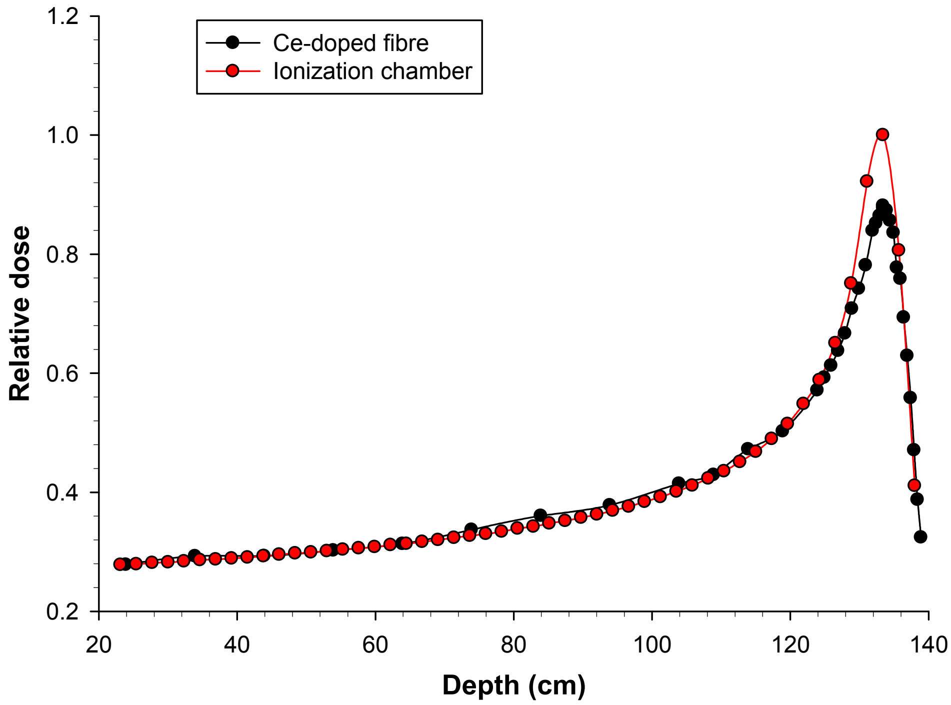

Fig. 9. Depth dose profile of a 138 MeV proton beam in water measured with the doped fibre and compared with the reference curve obtained with an ionization chamber.

Conclusions

The results of the various tests performed using different radiation beams attested the promising properties of the dosimetric system. On the basis of these outcomes, a pre-industrialization agreement was signed between the inventors of the system and EL.SE S.r.l., an Italian Company specialized in the development and sale of radiation technology devices for medical applications.

Acknowledgments

This work was supported by Regione Lombardia in the frame of the project “New scintillating glasses” (NSG) and by the Institute of Nuclear Physics (INFN) in the frame of the project “ Dosimetry by luminescence in medical and environmental exposures” (DOLMEEN).

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Guidelines for the use of active personal dosemeters in interventional radiology

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Abstract

The optimization of the use of active personal dosemeters (APDs) in interventional radiology (IR) is performed by one of the work packages of the ORAMED project, a Collaborative Project (2008 – 2011) supported by the European Commission within its 7th Framework Program. Due to the specificity of the X-ray fields used in IR (low energies, pulsed fields), the current technology of APDs is inadequate, notably in terms of energy detection threshold and dose rate response. The work presented in this paper consists in establishing guidelines related to the use of APDs in IR. The guidelines are divided in three parts. The first part presents the real radiation field characteristics encountered in IR. The dose rate in the direct field at the level of the table (i.e. above the patient) ranges from 2 to 360 Gy.h-1. The dose rate in the scattered beam at the level of the operator ranges from 5.10 – 3 to around 10 Gy.h-1.Finally, the energy of the scattered spectra ranges from 20 to 100 keV. The second part describes the results of tests performed in laboratory conditions and in hospitals. All APDs present a linear dose response and most of them a satisfactory response at low energies (down to 20 keV). However, some present dose rate and angular responses that do not always fulfil the ISO 61526 standard requirements. Tests in pulsed mode show that limitations of several APDs are mostly due to high dose rates rather than to pulse frequency. This point was confirmed by tests in hospitals. The last part of the guidelines proposes advices in the choice of the APD type and some methods to correct the response of the devices depending on the IR procedure type.

Radiation protection in radiotherapy: primary standard dosimetry of high-energy photon beams in Austria

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Abstract

The Austrian absorbed dose to water primary standard is a graphite-calorimeter. It was developed by the Federal Office of Metrology and Surveying (BEV) in cooperation with the Research Centers Seibersdorf. The BEV is the National Metrology Institute (NMI) of Austria. The graphite-calorimeter is a Domen-type calorimeter and in operation since 1983. The realization of the unit absorbed dose to water is based upon absorbed dose to graphite measurements. The absorbed dose conversion is done by two independent methods based on the photon fluence scaling theorem. The graphite-calorimeter was originally designated for determination of absorbed dose to water in ^{60}Co gamma ray beams. The progress in radiation therapy within the recent years required for extension of the graphite-calorimeter application range to enable primary standard dosimetry of high-energy photon beams. The development of the primary standard is based upon Monte Carlo simulations with PENELOPE code and measurements with the graphite calorimeter and ionization chambers. Furthermore the graphite calorimeter and its corresponding components had to undergo a refurbishment and modernization process. This paper presents the results of the energy range and application enhancement of the primary standard. The determined correction and conversion factors for high-energy photon beams and a detailed uncertainty budget of the primary standard is given. Furthermore the re-evaluated correction factors for ^{60}Co gamma ray beams are presented. To validate the results the BEV participated the international key comparison for absorbed dose to water in ^{60}Co gamma radiation at the Bureau International des Poids et Mesures (BIPM). Moreover the BEV planned and coordinates the EURAMET Project 1021 intended for direct comparison of absorbed dose to water primary standards. The accomplishment of the BEV high energy calorimetry project was promoted by the Physico-technical Testing Service (PTP), which is an entity of the BEV.

Introduction and overview

The Austrian absorbed dose to water primary standard is a graphite-calorimeter. It was developed by the Federal Office of Metrology and Surveying (BEV - Bundesamt für Eich- und Vermessungswesen) in cooperation with the Research Centers Seibersdorf. The BEV is the National Metrology Institute (NMI) of Austria. The graphite calorimeter is a Domen-type calorimeter and is in operation since 1983. The realization of the unit of absorbed dose to water is based upon absorbed dose to graphite measurements. The graphite calorimeter is designed for quasi-adiabatic and quasi-isothermal mode of operation. The absorbed dose conversion is done by two methods based on the photon-fluence scaling theorem. These methods are: conversion by calculation and conversion with an ionisation chamber.

In Austria the calibration and verification of therapy dosemeters is done in ^{60}Co gamma radiation. The applied secondary standards, e.g. used in hospitals, are traceable to the primary standard graphite calorimeter. The conversion of the ^{60}Co chamber factor to the chamber factor for high-energy photons and high-energy electrons uses the code of practice of the Austrian standard ÖNORM S 5234-3. The improvement of the absorbed dose to water primary standard directly leads to an improvement of quality assurance measurements in Linac radiotherapy, i.e. directly related to the radiation protection of patient. Further the accurate knowledge of the applied dose is a main factor influencing the success of a radiotherapy and therefore from great importance for the treatment planning.

The graphite calorimeter was originally designated for determination of absorbed dose to water in ^{60}Co gamma ray beams. The progress in radiation therapy within the recent years required the extension of the graphite-calorimeter application range to enable primary standard dosimetry of high-energy photon and electron beams. The accomplishment of the BEV high-energy calorimetry project was promoted by the Physico-technical Testing Service (PTP), which is an entity of the BEV.

To enable the graphite calorimeter for primary standard dosimetry of high-energy photon and electron beams a set of beam quality dependent conversion and correction factors was required. Additionally the graphite calorimeter, the graphite phantom and all corresponding components had to be adapted to the measurement requirements for high-energy photon beams. This was done within a refurbishment and modernization process intended to ensure the quality and reliability of the primary standard and to maintain the primary standard at an international level. The correction and conversion factors were obtained via Monte Carlo simulations with PENELOPE code, measurements - with the graphite calorimeter and ionization chambers - and with the use of published factors.

First of all measurements and simulation studies for the estimation of correction factors were carried out for ^{60}Co gamma rays to achieve a well-founded basis. This lead to the re-evaluation of the BEV absorbed dose rate to water reference value for ^{60}Co gamma ray beams. To validate those results the BEV participated the international key comparison for absorbed dose to water in ^{60}Co gamma radiation at the Bureau International des Poids et Mesures (BIPM). Subsequently measurements and Monte Carlo studies for selected high-energy photon beam qualities were performed.

To achieve beam quality specific correction factors it was necessary to consider the radiation field characteristics of irradiation facilities used for measurements. This included the Monte Carlo modelling of the BEV ^{60}Co teletherapy unit and of Varian Linac treatment heads. Thereby photon energy spectra were determined. These spectra constitute the basis of beam models used for the graphite calorimeter specific Monte Carlo simulations to obtain the required application specific correction factors. For high-energy electron beams concepts were developed for the realization of primary standard dosimetry. Nevertheless the practical implementation requires adaptations of the graphite phantom.

A confirmation of the energy range and application enhancement of the graphite calorimeter and thus of the implemented correction and conversion factors is done in the framework of the project 1021 of the European Association of National Metrology Institutes (EURAMET). This project is intended for the direct comparison of primary standards for absorbed dose to water in ^{60}Co and high-energy photon beams. The participating NMI's are: BEV, the Federal Office of Metrology (METAS - the NMI of Switzerland) and the Physikalisch-Technische Bundesanstalt (PTB - the NMI of Germany). For the comparison exercise the BEV transported the graphite calorimeter primary standard to METAS and PTB for operation in the accelerator radiation fields.

The advance of the graphite calorimeter provides the methodical fundamentals to enable the BEV for the accomplishment of primary standard dosimetry of high-energy photon and electron beams.

Results published

Information on the accomplishment of primary standard dosimetry of high-energy photon beams in Austria can be found in the following papers and documents:

Baumgartner A. Primary standard dosimetry of high-energy photon and electron beams. Ph.D. thesis (in German), Vienna University of Technology, (2010)

Baumgartner A., Hranitzky C., Stadtmann H., Maringer F.J. Determination of photon fluence spectra from a ^{60}Co therapy unit based on PENELOPE and MCNP simulations. Radiation Measurements, under review.

Baumgartner A., Steurer A., Maringer F.J. Simulation of photon energy spectra from Varian 2100C and 2300C/D Linacs: simplified estimates with PENELOPE Monte Carlo models. Applied Radiation and Isotopes 67 (2009) 2007–2012

Baumgartner A., Steurer A., Tiefenböck W., Gabris F., Maringer F.J. Correction factors of an absorbed dose primary standard graphite calorimeter in ^{60}Co gamma ray beams. Radiation Protection Dosimetry, under review.

Baumgartner A., Steurer A., Maringer F.J. Energy range and application enhancement of the BEV absorbed dose graphite calorimeter: Primary standard dosimetry of high-energy photon beams. Metrologia, under review.

Kessler C., Allisy-Roberts P.J., Steurer A., Baumgartner A., Tiefenboeck W., Gabris F. Comparison of the standards for absorbed dose to water of the BEV, Austria and the BIPM for ^{60}Co gamma radiation. Metrologia 47 Tech. Suppl. 06017 (2010)

ETAM
Method for Exposure Conditions Reconstruction

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Abstract
Laboratory for personal dosimetry in the Serbian Institute of Occupational Health, Belgrade, is using LiF TLD nearly twenty years. Equipped with Harshaw TLD Reader Model 6600 (12^{th} serial unit), REMSMENU software and Computerised Glow Curve Deconvolution (CGCD - DOS version) has prepared calibration for elapsed time estimation.

Recognizing importance of more information in personal dosimetry practice, Laboratory developed criteria for estimation exposure conditions named **ETAM – Exposure Time Assessment Method.**

ETAM is based on LiF fading and difference between low temperature peaks half-life. The main values are calculated using CGCD Programme which gives: *FOM* (Figure of Merit), *peak2*, *peak3* and *peak5*. Then using laboratory's calibration curves is possible to calculate elapsed time T_2 and T_3. Ratio *peak2/peak5* and *peak3/peak5* are linear functions of logarithms of elapsed time T_2 and T_3 respectively. So, even for one LiF crystal estimation is based upon two sets of parameters. One TLD for personal dosimetry usually has two or more LiF crystals.

ETAM recognizes three types of exposure conditions: one-shot exposure (O), repeated exposure (R) and continuous exposure (C).

It is crucial to analyse together all main values. For example: if deconvolution *FOM* is too high it does not mean that method is inapplicable. Contrary, it could be argument that exposure was not type O. ETAM criteria is discussed in the paper supported with numerous illustrations from real practice results. Laboratory has about 2000 users monthly. They are working in different exposure conditions, but mainly type C and R.

Introduction
Serbian Institute of Occupational Health (SIOH) "Dr Dragomir Karajovic" is the newest name (from 2009) of the institution established in 1953 with the aim to develop occupational medicine, to investigate all occupational hazards, radiological as well. Since 1959 the Institute was organized in five centres. The Radiation Protection Centre has three departments: Dosimetry, Radioecology and Medical Department.

Dosimetry Department (DD) consists of two parts: sector for radiation protection measures and Laboratory for Personal Dosimetry (LPD). Very often personal dosimetry is performed for users who engage DD for radiation protection measures. So, plenty of data from both sides: from “in situ” measurements and corresponding personal dosimetry results is big advantage in the case when it is necessary to prepare a special report for user (comments for doses above investigated levels or limits).

LPD has used different techniques during fifty years: film dosimetry until early 80’s; then TLD - MgB_4O_7, Toledo 654 Reader; and Harshaw TLD Reader Model 6600, LiF crystal, from January 1992. LPD has more then 2000 users with monthly frequency.

ETAM – Exposure Time Assessmnet Method

Exposure Time Assessment Method (ETAM) has been developed for analyzing TLD, type 100 (LiF:Mg,Ti), for benefit of users, occupationally exposed workers. Lately, it has been applied to environmental dosimetry with considerable results (O. T. Marinkovic, V. M. Spasic Jokic, 2008).

A new equipment from 1992

LPD has got Harshaw TLD Reader Model 6600, 12000 dosimeters (card with two LiF crystals) and DOS version of Harshaw Computerized Glow Curve Deconvolution Programme (CGCD). Being 12th serial unit the Reader is controlled by REMSMENU software. Next new TLD cards were not bought until 2007.

LPD has participated in two national TLD intercomparasions before 2000 and EURADOS IC2008. The first international participation was very successful. All LPD’s reported results: Hp(10) and Hp(0.07) at EURADOS IC2008 were excellent.

Recognizing low level of personal dosimetry reputation among occupationally exposed workers prescription for elapsed time estimation in User’s Manual (HARSHAW/FILTROL, 1988) was rated as very useful tool. First calibration for elapsed time estimation according CGCD was prepared in 1995 (O. Marinkovic, 1995).

Very high Hp(10) was appearing from time to time. Elapsed time estimation for one-shot irradiation is a part of special report. Sometimes, special report prevented misused of intentional irradiation; sometimes helped to detect unknown irradiation. Undoubtedly, using CGCD helped LPD to upraise common credence.

Collecting experience based on hundreds analyzing glow curves results are systemizing to support a method for recognizing one-shot, repeated or continuously exposure of TLD.

LiF Crystal

Elapsed time estimation is based on LiF five temperature peaks below 300^{o}C. First one has so short half-life that is of no importance to speak about. Peaks 2, 3 and 4 have half-lives of the order of several hours and a few months respectively. Peak five has long half-life (~ 80 years) and is used as stable.

Read-out cycle keeps recommended time temperature profile. Crystal is heated with constant rate. Output signal is registered in 200 channels during heating. Plotting glow curve we can see different shape due to elapsed time between irradiation and read-out moments. Figure 1. shows an example for very short and very long elapsed time after irradiation.

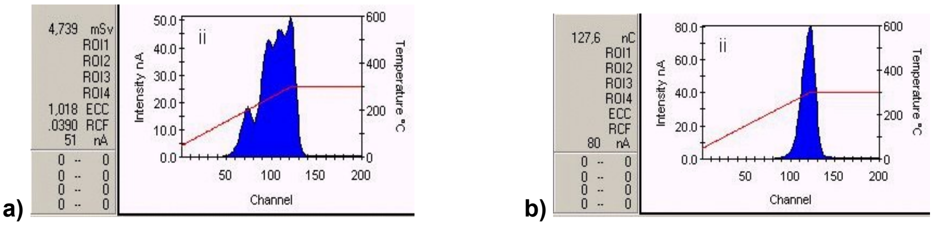

Figure 1. Glow curve shape: a) short elapsed time; b) long elapsed time after TLD irradiation.

Calibration for elapsed time estimation

Two hundreds and fifty crystals were irradiated and read-out after different elapsed time (O. Marinkovic, et al. 2001). Verification is performed from time to time with a few dosimeters in different range.

Peak area ratios *peak2/peak5* and *peak3/peak5* are functions of elapsed time *T*. Linear regression parameters are given in equations (1) and (2).

$$peak2/peak5 = 0.22 - 0.075 \times LogT_2 \quad (1)$$

$$peak3/peak5 = 0.88 - 0.26 \times LogT_3 \quad (2)$$

T_2 and T_3 theoreticaly represent the same elapsed time *T* for one-shot exposure. ETAM analyses these values for repeated and continuos exposure.

Here is an example (Figure 2.) of glow curve deconvolution for crystal read-out prolonged enough after irradiation to loose peak2 completely.

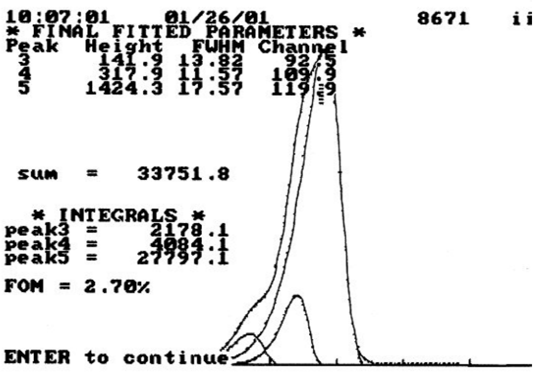

Figure 2. A glow curve deconvolution result.

Limited interval for application (1) and (2) are respectively 857 days and 2424 days. According experience from routine practice monthly read-out cycle is preferred.

ETAM parameters

REMSMENU has option for creation file selected.crv from primary files (DAT, I1, I2, I3). CGCD is applied to selected.crv and gives result visible at Figure 2:

– First line: reading time (10:07:01), reading date (01/26/01), TLD bar-code with chip's identification (8671 ii);

- Second line: Title ∗FINAL FITTED PARAMETERS∗;
- Third line: subtitle for four columns – Peak, Height, FWHM, Channel;
- Next lines: results for deconvoluted peaks respectively (values peak2 would be detected if elapsed time was not more then 500 hours)
- Sum: TLD signal after background subtraction.
  Note that ″background″ is ″system″ background (crystal+photomultiplier background); not consequence of natural radiation;
- Another title ∗INTEGRALS∗: there are written peaks area due to CGCD result;
- FOM – Figure of Merit in percents.

Final fitted parameters are acceptable or not acceptable for ETAM. If results cannot be accepted, CGCD should be repeated with corrections.

An example for not accepted first deconvolution results is given in Table 1.

Table 1. CGCD unacceptable results.

Bar-code & date (mm/yy)	FOM (%)	Sum (r)	Peaks			Sum (v)
			Height	Channel	Integral	
5652 (ii) 05/99	21.60	476.3	Peak 2			956.4
			-1.676	95.8	0.000	
			Peak 3			
			3.805	148.0	183.8	
			Peak 4			
			-8.007	125.7	0.000	
			Peak 5			
			13.33	142.3	772.6	
12657 (ii) 07/08	14.86	119.9	Peak 2			221.266
			0.360	118.3	6.066	
			Peak 3			
			-3.956	93.7	0.000	
			Peak 4			
			4.081	97.0	103.2	
			Peak 5			
			3.687	150.4	112.0	

Peak height is relevant for ETAM. Negative peak height clearly indicates faulty deconvolution.

Peaks detection **Channels** are prompt criteria for deconvolution quality. From peak2 to peak5 detection channels upraise. If two peak channels are not successive (for example from Table 1, 5652ii: peak 3 was detected at channel 148.0, but peak 4 at channel 125.7), we have indication that deconvolution is not useful.

CGCD gives TLD signal after background subtraction - "**Sum**". This value is parameter for dose calculation (depended on calibration factor). Could be used also for deconvolution appraisal for ETAM purposes meanness what are reported unites. We mark this value with "**Sum (r)**" - real. A total peaks area is marked with "**Sum (v)**" which can be named "the area of **virtual** glow curve". Difference between Sum (r) and Sum (v) is a measure for correspondence real glow curve area and virtual glow curve area.

FOM – Figure of Merit - is a measure of the accuracy of deconvolution. Composite glow curve (above mentioned as virtual) is reconstructed adding the peaks areas and compared with the original curve. Recommendation in CGCD User's Manual for one-shot elapsed time estimation is that FOM higher then 2% represent uncertain deconvolution. For ETAM criteria even FOM =17.87% is acceptable (see TLD 9006iii; Table 3) if other results are clear for analyzing. In this case irradiation was not one-shot but happened in repeated portions.

ETAM – Three types of dosimeter irradiation

Speaking about workers' exposing condition we are talking about dosimeter exposing condition in fact. Usually, exposing is devided into portions during working time.

ETAM recognizes three types of exposing: one-shot, type **"O"** ; repeated, type **"R"** and continouslly exposure, type **"C"**.

Table 2. Examples for exposure type "O".

Bar-code & date (mm/yy)	FOM (%)	Sum (r)	Peaks			Sum (v)
			Height	Channel	Integral	
2321 (iii) 04/95	5.67	642.0	**Peak 2**			658.52
			0.277	90.5	11.99	
			Peak 3			
			1.969	112.2	33.23	
			Peak 4			
			8.417	134.7	166.3	
			Peak 5			
			22.15	147.3	447.0	
5621 (iii) 12/97	2.21	5102.5	**Peak 2**			5190.5
			4.689	89.2	80.05	
			Peak 3			
			38.65	116.8	713.1	
			Peak 4			
			63.07	135.0	1308.1	
			Peak 5			
			137.9	150.1	3088.8	

Table 2. Continued.

8393 (ii) **05/99**	4.02	594.7	**Peak 2**			591.52
			5.421	92.3	105.0	
			Peak 3			
			5.601	119.7	134.0	
			Peak 4			
			4.378	135.3	76.82	
			Peak 5			
			10.25	152.4	275.7	
4237 (ii) **02/01**	3.10	1449.6	**Peak 2**			1471.9
			93.0	74.2	175.6	
			Peak 3			
			121.9	100.8	244.2	
			Peak 4			
			166.4	115.2	341.2	
			Peak 5			
			321.0	134.3	710.9	
3184 (iii) **03/06**	2.08	219.2	**Peak 2**			217.06
			2.583	76.4	42.58	
			Peak 3			
			2.059	97.7	38.48	
			Peak 4			
			1.883	109.4	32.80	
			Peak 5			
			4.004	124.9	103.2	
6519 (ii) **03/07**	2.05	3514.4	**Peak 2**			3542.1
			16.09	72.9	241.3	
			Peak 3			
			39.93	94.3	618.0	
			Peak 4			
			49.31	106.7	933.6	
			Peak 5			
			90.37	121.5	1749.2	

CGCD results for these types of irradiation are given in Tables 2, 3. and 4. Data of reading-out dosimeters are recorded and is keeping with all relevant documentation. Here was not important to put full date. Only month and year is written in the same column with dosimeter identification number (barcode added with chip's position sign – crystal ID).

A LiF crystal is exposed at least to natural ionizing radiation between two read-out cycles. Additional irradiation comes in portions: extremely short (in digital radiography) or long lasting during interventional diagnostic. Accidental exposure is an example of one-shot exposure.

Results presented in Table 2. are type "O". TLD was irradiated in LPD and elapsed time is well known. Notice that FOM in these cases are from 2% to 6%.

Table 3. Examples for exposure type "R".

Bar-code & date (mm/yy)	FOM (%)	Sum (r)	Peaks: Height	Peaks: Channel	Peaks: Integral	Sum (v)
9006 (ii) 02/06	5.16	78.38	**Peak 2**			81.387
			-	-	-	
			Peak 3			
			0.303	96.1	4.530	
			Peak 4			
			0.557	114.4	6.087	
			Peak 5			
			3.361	123.4	70.77	
9006 (iii) 02/06	4.51	124.7	**Peak 2**			129.525
			-	-	-	
			Peak 3			
			0.512	95.6	7.590	
			Peak 4			
			0.850	113.6	9.935	
			Peak 5			
			5.381	122.5	112.0	
9006 (ii) 01/10	17.87	74.31	**Peak 2**			89.163
			-	-	-	
			Peak 3			
			-0.582	10.3	0.000	
			Peak 4			
			0.098	75.7	2.033	
			Peak 5			
			3.508	122.5	87.13	
9006 (ili) 01/10	4.91	102.0	**Peak 2**			104.38
			-	-	-	
			Peak 3			
			0.397	95.7	6.042	
			Peak 4			
			0.740	114.8	7.308	
			Peak 5			
			4.241	128.3	91.04	

TLD 9006 (Table 3.) is regular personal dosimeter belonging to a cardiologist for many years. Dosimeter is worn above apron. There are a lot of similar examples related to working conditions where TLD (and worker) is exposed several times to high dose in interventional diagnostic during one month.

Important differences between "R" and "C" type of irradiation is peak 2. In the Table 4. there are results for dosimeters exposed continouslly to low dose rate between two read-out cycles. Peak 2 is detected here. Repeated exposure enlarge peak 3 more then peak 2. It is the most important difference between types "R" and "C".

Table 4. Examples for exposure type "C".

Bar-code & date (mm/yy)	FOM (%)	Sum (r)	Peaks			Sum (v)
			Height	Channel	Integral	
10345 (ii) 01/10	9.79	6.371	**Peak 2**			6.587
			0.007	71.9	0.170	
			Peak 3			
			0.046	94.4	0.704	
			Peak 4			
			0.059	110.2	0.731	
			Peak 5			
			0.225	120.8	4.982	
4848 (ii) 01/10	8.80	7.225	**Peak 2**			7.436
			0.008	71.4	0.209	
			Peak 3			
			0.054	94.4	0.950	
			Peak 4			
			0.058	109.3	0.891	
			Peak 5			
			0.244	120.4	5.386	
6631 (ii) 01/10	7.76	9.120	**Peak 2**			9.256
			0.007	72.1	0.199	
			Peak 3			
			0.041	95.6	0.684	
			Peak 4			
			0.071	113.6	1.080	
			Peak 5			
			0.337	123.2	7.373	

Determination of irradiation type

Calibration equations (1) and (2) enable two results for elapsed time: T_2 and T_3 for one LiF crystal. LPD operates with dosimeters (personal and environmental) which have two LiF crystals.

Elapsed time estimation usually is based upon four results: $T_{2(ii)}$, $T_{2(iii)}$,, $T_{3(ii)}$ and $T_{3(iii)}$. A few results calculated with equations (1) and (2) are given in Table 5.

Table 5. Elapsed time calculation.

crystal ID	peak2	peak3	peak5	T_2 (h)	T_3 (h)
		type "O"			
8486 (ii)	13,98	99,83	290,00	195	115
8486 (iii)	15,34	78,99	248,60	129	145
5632 (ii)	7,70	73,37	355,10	441	389
5632 (iii)	5,38	77,27	290,20	485	229
		type "R"			
9006 (ii)	0,00	4,53	70,77	857	1375
9006 (iii)	0,00	7,59	112,00	857	1330
		type "C"			
10345 (ii)	0,17	0,70	4,98	301	694
4848 (ii)	0,21	0,95	5,39	261	508
6631 (iii)	0,20	0,68	7,37	374	1066

Discussion

ETAM is based on results obtained with CGCD Programme. LPD has many data to support ETAM. Here are ones selected by criteria to clearly demonstrate method applied during fifteen years.

There are three types of dosimeters exposing: one-shot, repeated and continouslly.

Table 5. - Type « O »

$T_{2(ii)}$, $T_{2(iii)}$,, $T_{3(ii)}$ and $T_{3(iii)}$ are compatible. Elapsed time can be calculated and uncertainty can be expressed separately for peaks:

TLD 8486: $T_2 = (162 \pm 33)$ h; $T_3 = (130 \pm 15)$ h or finally $T = (150 \pm 40)$ h.
TLD 5632: $T_2 = (463 \pm 22)$ h; $T_3 = (309 \pm 80)$ h or finally $T = (390 \pm 80)$ h.

Table 5. - Type « R »

$T_{2(ii)}$, $T_{2(iii)}$,, $T_{3(ii)}$ and $T_{3(iii)}$ are incompatible.

Result for peak2 can be explained as one-shot exposure 857 hours before heating. But, result from peak3 tell us that time of exposure (if it was one-shot) happened before dosimeter have been distributed to worker. So, conclusion is that dosimeter was irradiated in separately portions during using period. Peak2 and peak3 half-lives are shorter then peak5 one. So, first two decreased with time while peak5 increased.

Table 5. - Type « C »

Taking into account that FOM is very good in the case of one-shot exposure T_2 and T_3 have to determine elapsed time with uncertainty less than 50%. Here we have distinction T_2 and T_3 of about 100%, 200% or even 300%.

Conclusions

ETAM is powerful tool for analyzing working conditions in occupational exposure as well as in environmental monitoring.

Disadvantage is time consume. Recognizing benefit in very important cases it justifies spent time for applying ETAM.

ETAM should be upgraded. Huge experience and keeping records from fifteen years can support ETAM with more analysis.

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Neutron and photon response of a CMOS pixel detector for a future electronic dosimeter

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Abstract
The RaMsEs group of the IPHC in Strasbourg is working on a new compact device for active neutron dosimetry. The electronic part of the detector is a CMOS active pixel sensor based on MIMOSA5 technology, originally designed for tracking in particle physics. The thin active layer of these sensors leads to a low sensitivity to γ-photon background, which is an attractive feature in neutron fields. Measurements have been carried out with various radioactive sources: α-particles from ^{241}Am for calibration in counting rate and photons from ^{241}Am (59 keV) and ^{60}Co (1.17 MeV and 1.33 MeV). When the device is exposed to the fast neutrons of a ^{241}AmBe source, we obtain good discrimination between recoil protons from fast neutrons and the electron background.

Introduction
Active neutron dosimetry has become mandatory since 1995 for people receiving more than 20 mSv per year (international norm IEC 1323). About 60 000 workers are affected by this policy in Europe, mainly those in nuclear power plants and in medical physics (Queinnec 2008).

Since the RaMsEs group is concerned by dosimetric applications, we aim at developping a new real-time dosimeter based on CMOS technology. Successful tests have already been performed to estimate the response of the sensor to fast neutrons (Trocmé et al. 2008) at the LMDN (Laboratoire de Métrologie des Neutrons), IRSN, Cadarache, France.

As neutron environments usually consist of mixed γ/n fields, new experiments have been carried out to test the response of the sensor to γ-rays and to determine a threshold for removing the photon background. The sensor was exposed to photon sources and to AmBe, a mixed n/γ source. All experimental results have been compared with Monte Carlo simulations performed with MCNPX 2.6.0f using the LA150 libraries for neutrons (Hendricks et al. 2008).

Material and methods

Neutrons can generate charged particles through various elastic or inelastic processes, and the choice of a converter material (in composition and thickness) depends on the energy of incident neutrons.

Converter

As this work involves only fast neutrons, we chose a hydrogen-rich polyethylene radiator to benefit from a high (n,p) elastic scattering cross section (> 1 barn below 10 MeV). This $(CH_2)_n$ converter has a 900 µm thickness, such as to optimize the number of recoil protons produced by the AmBe source. This optimal thickness produces proton equilibrium, as shown by Monte-Carlo simulation with MCNPX 2.5.0 (Trocmé et al. 2008).

Sensor

The CMOS chip, called MIMOSA5 (Minimum Ionizing MOS Active pixel sensor), has been described elsewhere (Turchetta et al. 2001). It is made of four independent matrices each of 512×512 pixels, with each pixel having an area of 17×17 µm². One sensitive matrix has 0.75 cm² area.

A standard sensor consists of several layers. The "front side" is a 6 µm-thick SiO_2 layer in which is contained the readout electronics. The sensitive area is the epitaxial layer of 14 µm thick pure silicon grown on a silicon substrate of 300 µm. For our dosimetry application, a thinned down sensor has been used, meaning that the substrate has been chemically removed. In this chip, the epitaxial layer has a thickness of 10-12 µm covered by a passivation layer of about 100 nm SiO_2 (Deptuch et al. 2005). In our experiments, the chip is back-illuminated: the incident particles impinge directly onto the passivation layer. This chip configuration allows to detect low-energy protons and α-particles which otherwise would partially be stopped by the 6 µm-thick SiO_2 layer that now lies downward the epitaxial layer.

When a charged particle crosses the sensor, electron-hole pairs are created along the path of the particle in the epitaxial layer, and the electrons are collected by pure thermal diffusion on the pixels (Turchetta et al. 2001). The initial charge is spread over many neighbouring pixels, defining a "cluster". To determine if the cluster is selected or not as a real event, a cut on the signal-to-noise ration ($S/N > 5$) is applied on the pixel of maximum amplitude known as the seed pixel.

Acquisition system

The sensor is bonded to a PCB (Printed Circuit Board). The chip delivers four analog outputs (one per matrix) which are received by a motherboard driven by a FPGA (Field-Programmable Gate Array) and a four-channel 10-bit ADC/pixel (Analog to Digital Converter). The acquisition PC includes a standard digital card (National Instruments PCI-6534) which sends instructions and treats the data signals from the motherboard. The acquisition program is coded in C.

A complete reading of one matrix is called a frame. A reset signal is applied on all pixels after two consecutive frames to suppress the important contribution to electronic noise from the leakage current. In order to cancel the pedestal (the pixel charge without any detection), a frame-to-frame subtraction is operated (Correlated Double Sampling)

(Turchetta 2001). The reading frequency is 5 MHz and one pixel is read every 200 ns. The readout time of one full frame is about 50 ms ($512\times512/5.10^6s^{-1}$) and the data flow reaches 10 MB/s.

Results for α-particles

To calibrate the sensor in counting rate, α-measurements were carried out with a ^{241}Am source, emitting on average 5.45 MeV α-particles with an activity of 22.3 kBq (± 10%) in the forward direction (2π). These α-particles generate large clusters with a total charge well above the electronic noise (Fig.1).

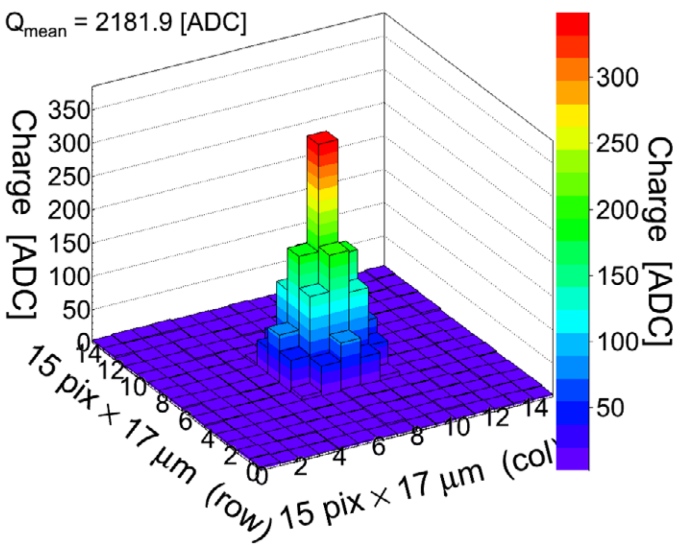

Fig.1. Mean α-cluster seen in the sensor for the ^{241}Am and at 2.3 cm from the source.

The source was placed at four distances. Our experimental results (Table 1) and simulations show excellent agreement, leading to detection efficiency for α-particles close to 100%.

Table 1. Summary of the results for ^{241}Am α-measurements confronted with simulation. Uncertainties on experimentally detected particles are statistical only.

Source-sensor distance (cm)	Detected α/min (experimental)	Detected α/min (simulation)	α detection efficiency (%)
1.8 ± 0.2	52220 ± 229	52740 ± 230	99.0 ± 0.6
2.3 ± 0.2	34872 ± 187	34980 ± 187	99.7 ± 0.8
2.8 ± 0.2	23914 ± 155	24240 ± 156	98.7 ± 0.9
3.6 ± 0.2	13646 ± 117	13800 ± 117	98.9 ± 1.2

Results for photons

As the photon background is a major concern in all neutron fields, experiments were also performed with γ-ray sources to determine the response of the sensor to photons.

Unshielded ^{241}Am source

Our point-like ^{241}Am source of (11.35 ± 0.57) µCi activity is sealed to absorb alphas and electrons. The main photon rays of this source are X-rays of 13.9 (13 %), 17 (19 %) and 21.2 keV (5 %), and γ-rays of 26.3 (2 %) and 59.5 keV (36 %) (in brackets the respective intensities) (LNHB 2007). To remove the X-rays of the ^{241}Am, a 3mm thick aluminium absorber can be inserted between the sensor and the source.

A first set of measurements was carried out without X-ray shielding. Knowing that the polyethylene converter itself generates electrons, we started without converter. As usual, the fired pixels are grouped into clusters, but the electrons generate lower charge than α-particles. In this case, the applied cuts to distinguish signal from noise lead to a significant loss of signal.

The response of the sensor can be simulated with MCNPX, and according to Table 2, a count rate of 1667 events/min are expected. The calculation assumes that a photon, once converted in the epitaxial layer, is necessarily detected. This cannot be the case as the signal generated by low energy electrons (or by electrons whose range in the sensitive layer is short) will be lost in the electronic noise.

Table 2. Response to ^{241}Am photons (without X-ray shielding): experimental and simulated events as a function of cut in deposited energy E_{dep}. Uncertainties are statistical only. Electrons produced in the aluminium absorber are included in the simulation results. The sensor was placed 3 cm from the source.

Cut on deposited energy	γ detected/min - simulation	γ detected/min – experimental
No cut	1667 ± 41	446 ± 21
E_{dep} > 10 keV	1472 ± 38	
E_{dep} > 15 keV	662 ± 26	
E_{dep} > 20 keV	100 ± 10	

Table 2 shows the effect of three different cuts on the detected energy, to find which simulated cut fits experiment. The simulation shows that retaining more than 20 keV deposited energy would underestimate the number of photons detected. Obviously, our sensor has a sensitivity which goes down to X-ray energies. According to this, we conclude that the lower detection limit of this detection chain (DAQ+analysis) is between 15 keV and 20 keV.

Shielded ^{241}Am source

To check our sensitivity to low energy photons, a 3 mm thick aluminium absorber was inserted between the source and the sensor. The thickness is a compromise between reducing the X-ray signal while retaining enough intensity for the γ-rays.

The counting rate obtained without absorber (to 3 cm from the source) is 446 γ/min whereas we find 70 γ/min when the absorber is used. It decreases significantly with shielding. Knowing that a 60 keV photon crosses our absorber with 80% probability, this is a clear demonstration that X photons are detected in our sensor and that an efficient shielding will remove them.

^{60}Co source

As mentioned earlier, the polyethylene neutron-to-proton converter is also a source of electrons. Simulations have been carried out to determine the effect of the 900 µm converter on the electron contamination in a neutron experiment.

According to simulation results shown on Fig. 2, the γ-ray contamination will be highest for incident energy between 1 and 2 MeV. To test this result, experiments with the two γ-rays 1.17 and 1.33 MeV of (0.43 ± 0.02) µCi ^{60}Co have been performed with and without converter (distance source-sensor: 1cm). The photon response is estimated by the ratio γ-rays detected/γ-rays emitted. The results are summarized on Table 3.

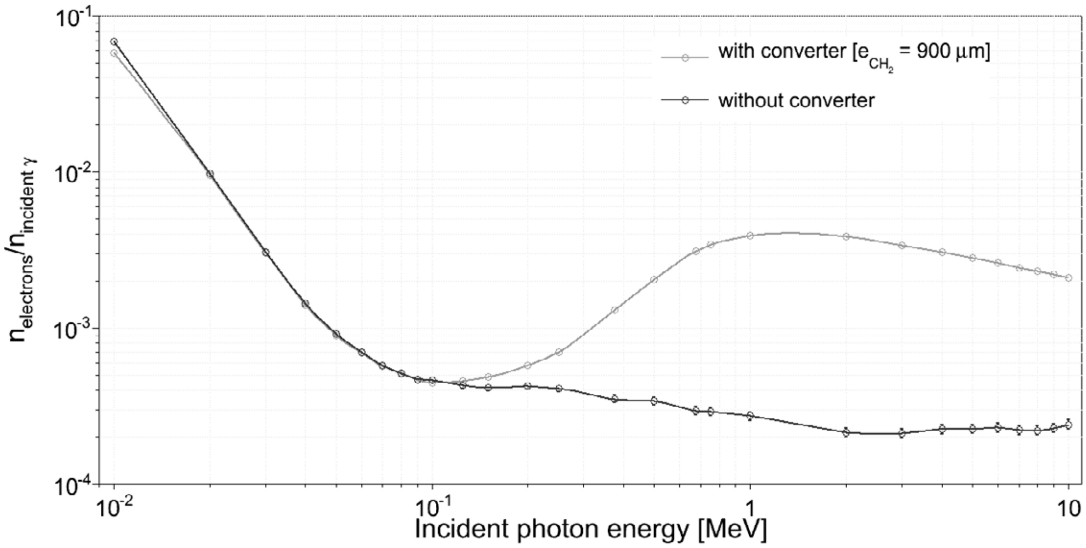

Fig. 2. Effect of the converter on the number of secondary electrons able to generate a signal charge in the epitaxial layer (simulation MCNPX).

For the ^{60}Co source without converter, the experimental response of our detector is of about 0.03 %, in good agreement with the simulation. The agreement is worse when the polyethylene converter is present, because of the secondary electrons generated by the $(CH_2)_n$ converter. However, the resulting energy deposited by electrons in the sensor is about 5 keV (Fig. 3), which is far below the detection limit of our electronics (as has been suggested by the ^{241}Am experiments). With a simple 15 keV cut on deposited energy in the simulation (Table 3), a good agreement is obtained between simulated and experimental efficiency. This is consistent with a sensitivity threshold of the sensor around 15 keV, as shown by previous ^{241}Am experiments.

We conclude that in a mixed field of neutrons and MeV photons, the additional photon effect of the converter will remain negligible.

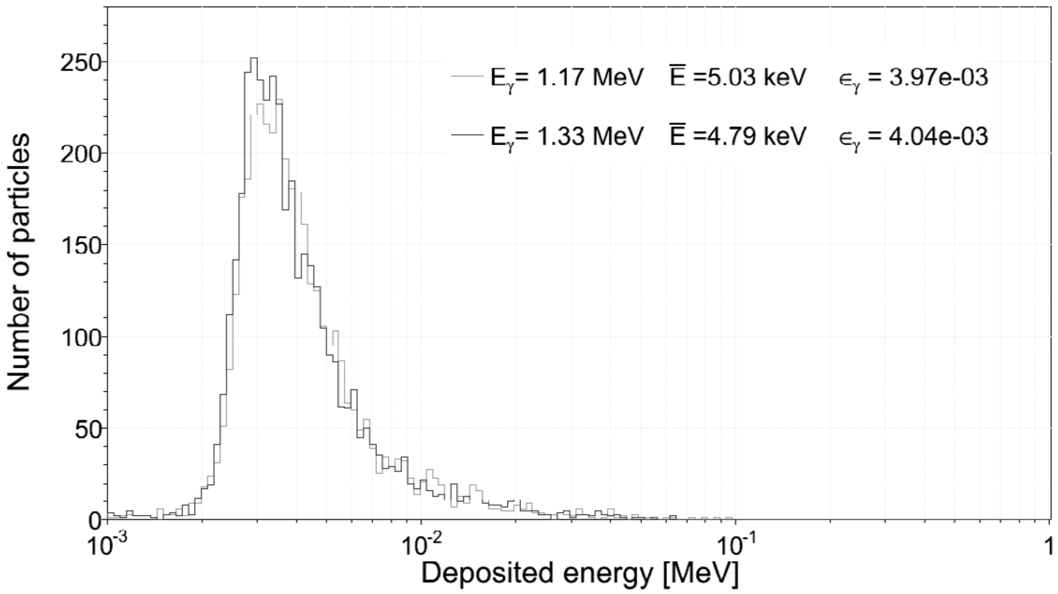

Fig. 3. Deposited energy distribution obtained by simulation for a ^{60}Co source with a 900 µm polyethylene converter (number of emitted photons : 10^6). The mean deposited energy is given by $\bar{E}$ and the detection efficiency for each γ-ray by ε_γ.

Table 3. Measured and simulated photon response for ^{60}Co. The results for ^{241}Am photons are added for comparison.

Source - configuration	Experiment	Simulation	Simulation with cut on energy (E_{dep} > 15 keV)
^{60}Co	(0.28 ± 0.02).10^{-3}	(0.71± 0.02).10^{-3}	(0.30 ± 0.02).10^{-3}
^{60}Co + $(CH_2)_n$ converter	(0.26 ± 0.01).10^{-3}	(8.03 ± 0.10).10^{-3}	(0.25 ± 0.01).10^{-3}
^{241}Am	(2.27 ± 0.05).10^{-3}	(7.76 ± 0.09).10^{-3}	(2.97 ± 0.02).10^{-3}
^{241}Am + Al screen	(0.29 ± 0.01).10^{-3}	(0.34 ± 0.06).10^{-3}	(0.25 ± 0.01).10^{-3}

Results for fast neutrons

To check that the recoil proton signal of a fast neutron source is easily distinguishable from photon signal, AmBe experiments have been performed. The neutron fluence rate of the source has been determined with Bonner Sphere Spectrometry by the University of Barcelona: (2.24 ± 0.10).10^6 s^{-1} (Domingo et al. 2009). According to MCNPX, our AmBe source generates two distinct populations in the sensor corresponding to 4.438 MeV γ-rays, which originate from the AmBe (Liu et al. 2007) and a much higher signal coming from the recoil protons from the converter (Fig. 4).

AmBe results and discussion

Fig. 5 shows the charge distribution in pixel clusters measured for a 30-min exposure at 15 cm from the source. We notice indeed two populations of particles, the first one peaked at low values of charge and the other one widely extended at much higher charge. In the measured distribution however (Fig. 5), the separation of the two populations is less than expected.

As already stated, the electron/hole transport mechanism in our silicon detector (which is not depleted) results in an incomplete charge collection (Turchetta et al. 2001). This implies a large spreading of the recorded signal, which is what we observe on Fig. 5. According to simulation (Fig. 4) a simple charge threshold should completely suppress the photon-background but the real situation (Fig. 5) is more complicated, since the photon signal overlaps slightly the proton distribution and vice-versa.

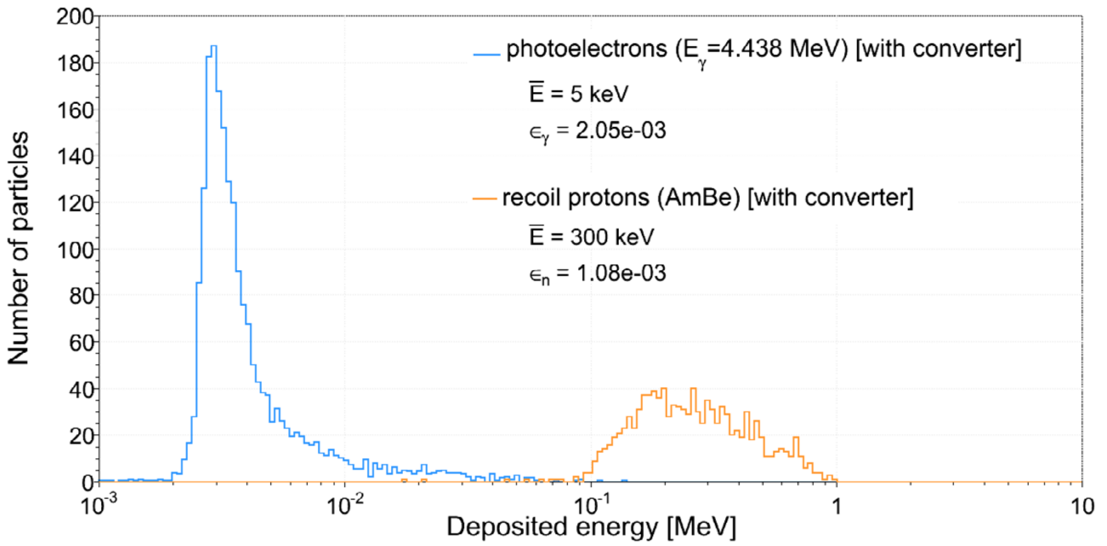

Fig. 4. Deposited energy distribution calculated with MCNPX for AmBe source. Photons are simulated as monoenergetic γ-ray of 4.438 MeV, with an intensity of 0.57 to the neutron flux (Liu et al. 2007). The mean deposited energy is given by $\bar{E}$ keV.

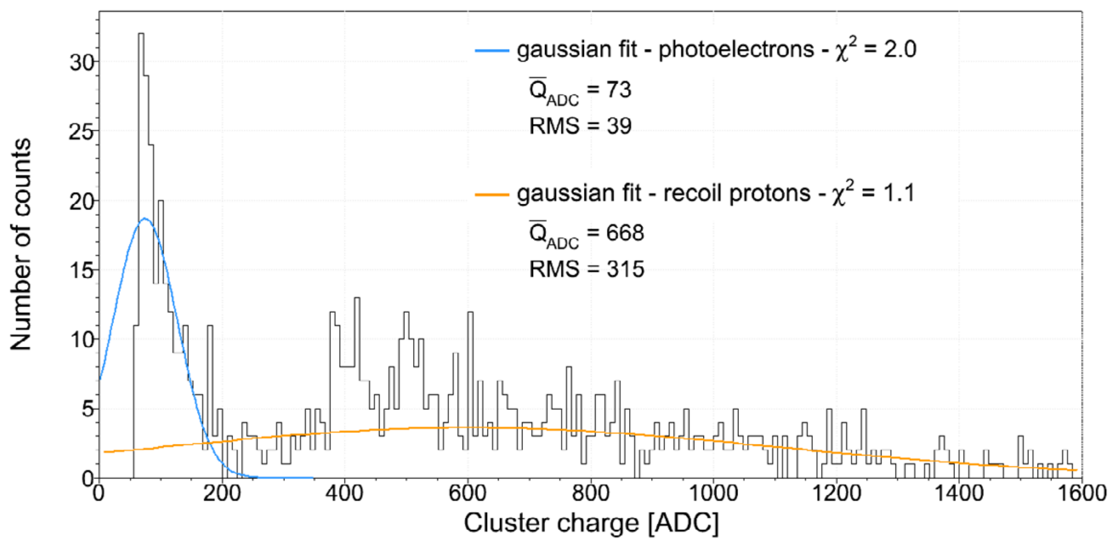

Fig. 5. Cluster charge distribution measured with a 30-min exposure at 15 cm from the ^{241}Am-Be source. Two gaussian functions were fitted on the two populations to estimate the number of recoil protons hidden in the photon peak and the number of photons that pollute the proton signal.

Proton clusters have high charge but also a high multiplicity (defined as the number of hitted pixels in a cluster) in each cluster, whereas secondary electron clusters have between 1 and 3 pixels. If we chose to cut the signal to 200 ADC, we remove the main part of the photon peak while keeping enough proton signal. To estimate the number of (low energy) protons lost in the photon peak, gaussian fits on the two populations of Fig. 5 allow an extrapolation for each population. Other parameterizations to fit the two populations have been tested. These other functions do not significantly change the conclusions.

To take into account the photon contamination in the proton peak, we define the signal purity p as $p = 1 - \frac{n_\gamma}{n_{hits}}$. For this fit, we can choose to cut the signal below 200 ADC counts leading to a high 99% purity (and a low 10 % ratio of lost protons).

We finally determine the absolute detection efficiency of our sensor as the ratio of the number of detected protons (i.e. number of counts above 200 ADC + number of protons lost in the electrons peak) and the number of incident neutrons. This leads us to a high efficiency $\varepsilon_{corrected} = (0.91 \pm 0.03).10^{-3}$ for AmBe neutrons whereas the simulated value is $\varepsilon_{MCNPX} = (1.08 \pm 0.33).10^{-3}$. Both efficiencies are in good agreement.

Conclusions

A test of a CMOS pixel sensor for AmBe neutrons (mixed with a possible photon contamination) has been successfully performed.

The sensitivity threshold of the sensor was checked with a ^{241}Am γ-source, which gave a threshold around 15 keV.

The X-ray response of the sensor is easily eliminated by an appropriate aluminium shielding.

A high γ-transparency of the sensor is demonstrated by its response to ^{60}Co photons, and the additional contribution of the polyethylene radiator itself to the electron background is negligible.

An efficient response of the sensor of nearly 10^{-3} is measured for fast neutrons from an AmBe source. This good sensitivity to neutrons is obtained with a signal almost free of photons. With an appropriate ADC cut of 200 for AmBe, we can remove the most part of electrons and obtain a low sensitivity to photon-background.

The next step will be to study the sensor response to thermal and intermediate neutrons with ^{10}B and ^{6}Li converters.

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Development of approaches for realistic retrospective evaluation of doses of selected cases of internal contamination

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Abstract

The work was focused on estimation of the intake and committed effective dose from internal contamination with americium 241. The various models of americium bio-kinetic were described primarily. Mathematical background of the intake estimation was expressed together with the test of the quality of the model fit to measured data. The uncertainty of the data was estimated in order to obtain reasonable estimate of the intake. The extensive uncertainty analysis of the in vivo measurement of the head and related assessment of the total activity in the skeleton was performed. For that issue was build a voxel phantom of the human head. The most of the factors affecting the measurement of the activity retained in the skull were characterized and its magnitude was quantified. The uncertainties for bioassay quantities, urine and stool samples, were estimated also. The intakes and doses of all subjects were assessed, but the default parameters of the americium model had to be changed. The discrepancy between the model and the data was discussed.

Introduction

The paper resumes achievements and work presented in the thesis of the same name. Estimation of the committed effective dose for seven worker contaminated with ^{241}Am during late 70'a 80's in the Czech Republic was main goal of the work. The workers follow-up started in 1995 in National Radiation Protection Institute (NRPI) in Prague and thus long term data were available. The difficulty of the task grounded in a fact that data start about more than four thousand days since intake, hence no information on intake route and pattern. Secondly, data did not match with prediction from standard model for americium published in ICRP 67 (ICRP 1993).

Material and methods

Subjects

The subject group consists of six men and one woman. Follow-up starts in 1995 while regular data for all seven subjects began in 1998. Subjects processed AmO_2 in 70's and 80's and their internal contamination were find on a whole body counter equipped with

NaI(Tl) scintillation crystals. There were no special calibration for Americium in this time and thus no early data are available, but the whole body measurements were used for time of intake estimation. All subjects were retired or did not work with americium anymore when follow-up started.

Bioassay data

Daily urine samples for all workers and daily stool sample for five subjects were collected. Samples of excreta were analyzed by the means of radiochemical separation followed by alpha spectrometry in NRPI. The combined uncertainty of the measured values was known and the data no need any special treatment, because alpha spectroscopy method is quite standard. More over NRPI laboratory regularly undergo PROCORAD comparison in order to ensure procedure quality (Filgas et al. 1998).

Estimation of the skeletal activity

The skeleton activity estimates were given on the beginning of the work; however there was suspicion on data quality. Fact that preliminary estimates of intake, based on skeleton data, significantly overestimates intakes based on bioassay data was main reason. An effort were undertaken in order to improve data quality. The original data on skeleton activity of ^{241}Am had been determined by equation 1

$$A_{skel} = \frac{N}{Y \times t \times \eta \times q}, \qquad \text{Eqn. (1)}$$

where A_{skel} is skeleton activity in Bq, N is detector counts (net peak area) from the skull measurement, Y is yield for gamma ray with energy 59.6 keV (^{241}Am), t is time of the measurement in seconds (typically 1800 seconds), η is detection efficiency in counts per emitted photon (CPP) and q is skull to skeleton activity ratio used to recalculate skull activity to the whole skeleton. Some terms from equation one, like Y, t and N, were known well and thus could not caused significant error. Detection efficiency of the skull measurement and skull to skeleton ratio were studied therefore.

NRPI whole body counter, equipped with two LEGe detectors, had been calibrated by a four head phantoms in the past (Malátová et al. 2000). Skull measurement geometry used in NRPI is shown in figure 1 (left).


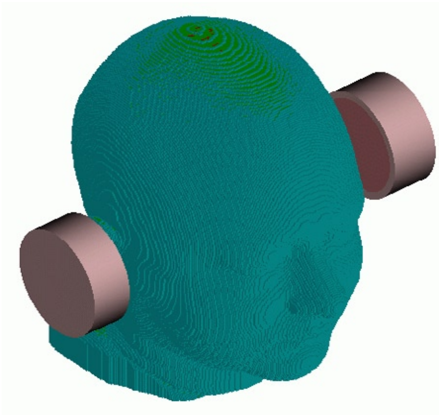

Fig. 1. Geometry of the measurement in NRPI: reality (left - property of NRPI), simulation with a voxel phantom (right)

Detector efficiency estimated from the phantoms ranged from 0.0055 to 0.122 CPP. The lowest value, belonged to the most realistic phantom BPAM-001 (Hickman et al.1988) borrowed from United States Transuranium and Uranium Registries (USTUR) was used for the skull activity calculations. There was supposition that observed spread of the detection efficiencies is caused by the different size of the phantoms. A Monte Carlo (MC) simulation with simplified measurement geometry and realistic detector models were performed in order to examine the hypothesis. Concentric spheres were used as substitute of real heads. Results of the simulations are shown in figure 2 (top), where head size (on x axis) was described by single parameter called head mean radius. The radius was calculated as arithmetic mean of three skull perimeters (Vrba 2007).

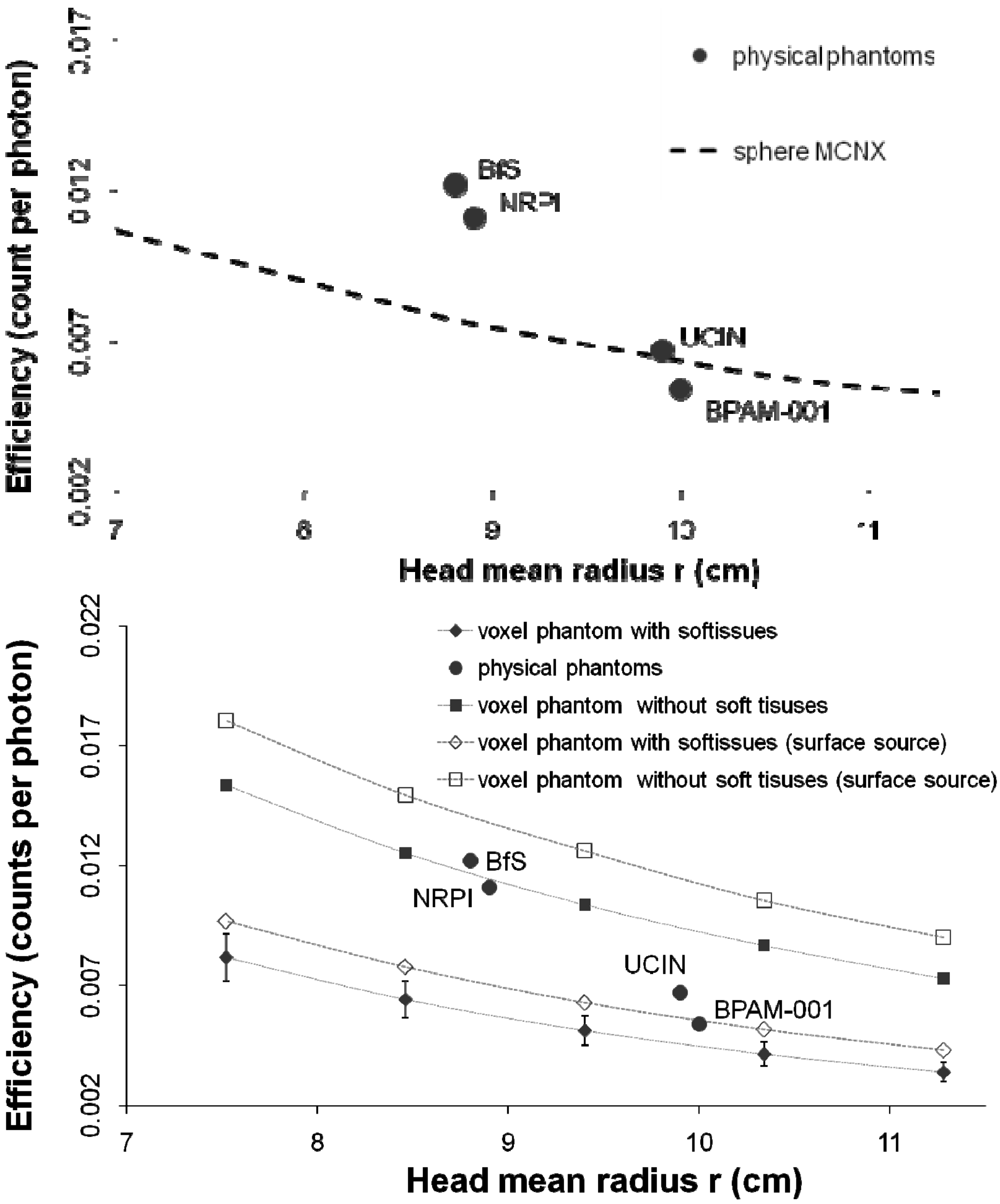

Fig. 2. Comparison of detection efficiencies of physical phantoms with Monte Carlo simulations: Sphere approximation (top), voxel phantom (bottom)

The decision to build a head voxel phantom and simulate measured geometry more accurately was made because the results were not conclusive. The detail description of the voxel phantom (sketch in figure 1 left) production and most of the results was published previously (Vrba 2007). Important study information is summarized in figure 2 (bottom) and the following list of the findings:

1) Detection efficiency depends on head size but not in extent shown by physical phantoms (visible from figure 2)
2) The main cause of detection efficiency discrepancy among physical phantoms is source distribution and insufficient inner filling of the cranial cavity.
3) Simulation with a voxel phantom can be better choice for calibration of detection system when physical calibration phantoms suffer from inaccuracy.
4) MC is capable to estimate the most important sources of uncertainty of ^{241}Am skull activity.

There was introduced head size dependent calibration curve, based on MC data, which respects mentioned results. Skull to skeleton activity ratio, originally based on skull to skeleton mass ratio and assumption on ^{241}Am homogenous distribution in skeleton, was revised in order to accord with USTUR whole body cases (Lynch et al. 1998). Neck vertebrae influence number of detected counts and thus correction coefficient were established. Mentioned changes are described by equation 2.

$$A_{skel} = \frac{N}{Y \times t \times \eta(R) \times q \times v}, \qquad \text{Eqn. (2)}$$

where $\eta(R)$ is head size dependent detection efficiency, v is vertebrae correction factor and the meaning of other symbols is same as in equation 1. Innovated method of the skeleton activity determination leads to activities even higher (for the most of the subjects) than the original ones, but the data credibility was improved.

Data uncertainty

Proper uncertainty assessment is crucial for correct data interpretation and thus the uncertainty of different datasets was determined. The system of scattering factors, introduced in IDEAS guidelines (Doerfler et al. 2006), was used in order to describe the uncertainty. The total uncertainty of any measurement involved in the intake assessment process can by approximated by lognormal distribution while natural logarithm of the Scattering Factors (SF) is equal to relative standard deviation of the distribution.

The total uncertainty for bioassay samples was derived from the data directly according methodology described in paper by Marsh and coworkers (Marsh et al. 2007). The SF for urine data was 1.36 and 1.77 for feces. There were included following uncertainties in the total uncertainty estimate of skeleton activity: reproducibility of detector position, counting error, calibration errors (uncertainty of Monte Carlo modeling), spectra evaluation, activity distribution in the skull bones, neck vertebrae correction, skull to skeleton ratio and effect of head size to the measurement. Scattering factors values for particular sources are summarized in table 1.

Americium models

Biokinetic models for americium were studied and compared in a theoretical part of the thesis. The ICRP 67 (ICRP,1993) model was chosen for evaluation because committed effective dose is bind with the ICRP biokinetic model.

Table 1. Uncertainty sources assumed for skeleton activity.

Uncertainty source	SF value	method of assessment, note
density and tissue composition of the voxel phantom	1.03	MC study with changed density within ±10%
statistical - calculation MCNPX	1.005	from MCNPX output file
cross -section library MCNPX	1.05	Guess
micro distribution in head	1.01	MC study simulating inhomogeneous distribution according USTUR data
neck vertebrae correction	1.05	correction uncertainty equals to correction factor
number of counts	1.1	Poisson error 10%
spectra evaluation	1.08	blind test – trials with one real measurement spectra, software and manual methods
detector position	1.13	dislocation of the detector centre within -2 to 2 cm in plane parallel to temporal bone and -1 to 1 cm perpendicularly
skull to skeleton ratio	1.16	inefficient statistic estimate of ratios for USTUR whole body cases
head size (known/unknown)	1.14/1.28	inefficient statistic estimate and data from MC study
Total (known/unknown head size)	1.32/1.37	

Fitting method and test of the fit quality

The maximum likelihood method supported by IDEAS guidelines were chosen, replacing weighted least square method. The quality of the fits was controlled by chi-square statistic test (one-side) and by autocorrelation coefficient proposed by Puncher et al. (Puncher et al. 2007). Lognormal character of the data to fit deviations was checked with Kolmogor-Smirnov test. All test were performed with significance level $\alpha = 0.05$.

Intake assumption

The exact time of intake and the intake pattern was unknown for all subjects. It was presumed that americium aerosol with AMAD = 5 μm and lung absorption class M was inhaled in a single intake. Such assumption was made in order to have conservative estimates and reduce complexity (to avoid too many assuming parameters). It has to be mention that the choice has no significant effect on data to model relation because earliest data staring 4000 days since assumed intake and are not influenced by the way of the intake any more.

Results

The first step of the internal dose evaluation is intake evaluation. This was done be a fitting of the data by default model setting (inhalation, class M, AMAD = 5μm). Each data set (urine, feces, skeleton) was fitted solo and also all together, when scattering factors were used as the weights. Chi-square and autocorrelation statistic were calculated and tested subsequently together with Kolmogor-Smirnov lognormality test. Results are summarized in table 2.

Table 2. Intake estimates with default americium model (inhalation, class M, AMAD 5 μm)

Subject ID	Data	Intake (Bq)	Statistic for all data fit*		
			$\chi 2$ [a]	ρ [b]	KS[c]
PP	skeleton	13108	fail	fail	fail
	urine	5220	fail	fail	fail
	feces	3963	pass	fail	fail
	all	6821	fail	n/a[d]	fail
SS	skeleton	9725	fail	fail	fail
	urine	3335	fail	fail	fail
	feces	2815	pass	fail	fail
	all	4612	fail	n/a	fail
HV	skeleton	20922	fail	fail	fail
	urine	8685	fail	fail	fail
	feces	7907	pass	fail	fail
	all	11157	fail	n/a	fail

Subject ID	Data	Intake (Bq)	Statistic for all data fit*		
			$\chi 2$ [a]	ρ [b]	KS[c]
FK	skeleton	10800	fail	fail	fail
	Urine	4000	fail	fail	fail
	feces	6259	pass	fail	pass
	All	5959	fail	n/a	Fail
PS	skeleton	58922	fail	n/a	fail
	urine	20387	fail	pass	pass
	All	29870	fail	n/a	pass
PV	skeleton	161195	fail	fail	fail
	urine	38738	fail	fail	fail
	All	59408	fail	n/a	fail
JH	skeleton	302887	fail	fail	fail
	urine	39200	fail	fail	fail
	feces	35670	fail	fail	fail
	All	85619	fail	n/a	fail

[*]test results for null hypothesis (that fit is right) while the intake was based on all data sets evaluated together, [a] chi-square test, [b] autocorrelation test, [c] Kolmogorov-Smirnov test, [d] not applicable or too few data points.

Intakes based on different data namely skeleton and bioassay differs about factor 2 to 9, but intakes based on urine and feces are in a reasonable agreement when data uncertainties are taken in account. Given results had not been acceptable and model modification has to be done in order to fit data in more plausible way. The modification of the material specific parameters will not bring any improvement, because such changes affect just early data. Modification of person specific parameters, such as skeleton liver uptake ratio or excretion rates was only possibility. Usage of person specific parameters results in a dose (person specific dose), which is no more consistent with the ICPR meaning of the effective dose. The analysis and justification of the following approach is given in the discussion.

The uptake ratio for skeleton and liver was a fist choice. The ratio was determined from the skeleton retention data and feces measurement (feces data partially depends on liver activity) because direct liver retention was unknown. This modification improves intakes agreement for skeleton and faces data but did not improve match with intake based on urine set. Adaptation of transfer rate from the blood directly to the urinary bladder was performed in order to match intakes based on skeleton and feces. Transfer rate from blood to small intestine has to be changed for the subject JH because skeleton retention was higher than fecal excretion and liver to skeleton ratio were not sufficient to reduce discrepancy between intakes. Newly estimated intakes from all data, with the model parameters are given in table 3.

Table 3. Intake estimates, personal dose and model parameters

Quantity	Subject ID						
	HV	SS	FK	PP	JH	PV	PS
transfer rate from blood to liver (d^{-1})	3.95	1.86	7.66	2.32	0.23	3.25[c]	3.25[c]
transfer rate from blood to skeleton (d^{-1})[a]	7.32	8.36	5.46	8.13	9.17	7.66[c]	7.66[c]
transfer rate from blood to urinary bladder (d^{-1})	1.05	0.95	0.65	1.19	0.12	0.4	0.81
transfer rate from blood to small intestine (d^{-1})					0.15		
Intake (kBq)	13.2	5.7	7.5	7.9	141.6	89.3	34.2
conversion coefficient[e] ($Sv \times kBq^{-1}$)[d]	0.0337	0.0347	0.0342	0.0336	0.0393	0.0367	0.0349
Person specific dose (Sv)[b]	0.44	0.2	0.26	0.27	5.56	3.28	1.19

[a] cortical and trabecular bone, [b] calculated as committed effective dose but with person specific parameters, [c] mean value of cases with feces data (PV and PS has no feces data), d conversion coefficient respecting modified model behavior. [e] recalculated conversion coefficient respecting changes of biokinetic.

Some statistical tests failed even for adjusted model; predominantly for urine data. An effort was undertaken in order to find out explanation.

The relative values, i.e. ratios of values from different datasets (urine, feces, skeleton) were used in order to normalize measured data. Such approach allows combining data from individual cases; however one degree of freedom is lost. A more stable dataset (activity in skeleton, excreted activity with urine), with lower scattering factor (excluding systematic part of uncertainty) was used for denominator. Figure 3 provides comparison of the measurements and the model predictions. It can be seen clearly, that feces to urine ratio is predicted properly, however both ratios with skeleton in denominator show significant difference.

Another way of comparison of the observed data with model predictions was that skeleton data of each case were fitted with a single exponential function and the exponents were compared with those predicted by the ICRP 67 americium model in the time interval from 5000 to 10000 days since the intake. Experimentally obtained exponents ranged from -7.44×10^{-5} to -2.45×10^{-6} with geometric mean -2.47×10^{-5}, which differs from the value of -4.07×10^{-5} from the ICRP model. When the same approach as for skeleton was applied for time course of excreted activity by urine and feces, the agreement of trends was poorer. The geometric mean of real data exponents is -3.14×10^{-4} for urine and -2.32×10^{-4} for feces, whereas the ones from standard ICRP 67 model are -1.02×10^{-4} and -1.08×10^{-4} respectively.

Discussion

The most questionable issue of the thesis was model modification. There are two serious questions: How can be interpreted intake estimates obtained from the improper fit? and What is credibility of the intakes or doses obtained from the person specific model? The reasoning of the author lead to conclusion that intakes with poor statistical measures has no any scientific value, and thus model has to be changed or not used.

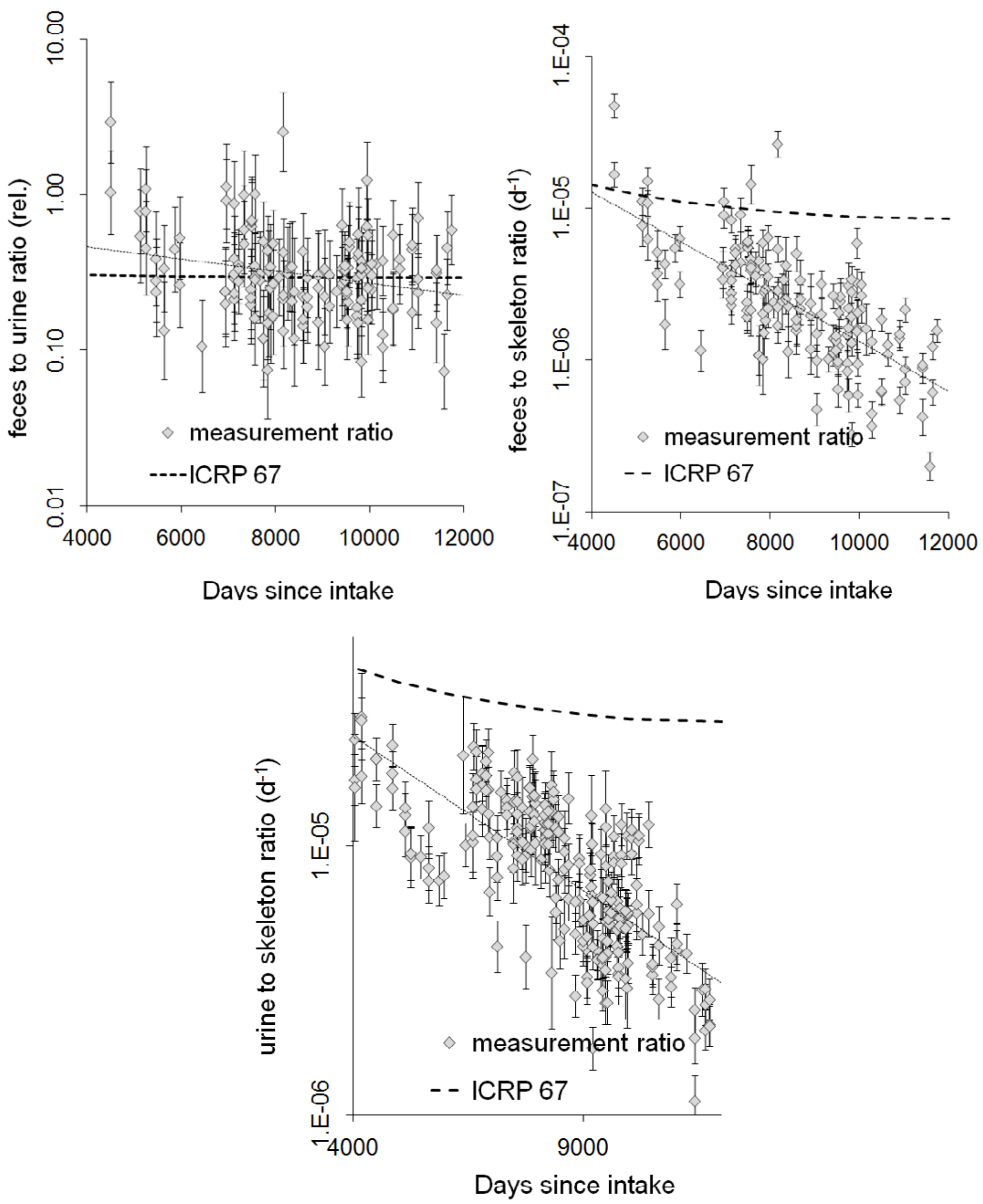

Fig. 3. Comparison of data and ICPR model prediction with basic setting.

The ICPR 67 model for americium consist of 14 compartments and together with the model of human alimentary tract represent quite complex system with many degrees of freedom, compare to three available data sets.

Ratio between skeleton and liver is quite variable base on autopsy data. Model assumes that 80% of the activity reaching blood is transfer to skeleton (30 %) and liver (50%). Increase of the skeletal portion (of these 80%) leads to reduction of activity excreted by feces and higher dose to bone surface. Thus overall dose to a subject will be higher. Adjustment of skeleton to liver ratio has no effect on bone metabolism (redistribution among bone surface, volume and marrow) which is based on studies with laboratory animals and has to be kept unchanged. Lower liver uptake can be explained by a radiation liver damage which may lead to rerelease of the liver activity to the blood. This issue was observed at high americium intakes per body mass of beagle dogs (Luciani et al. 2006).

Modification of urine excretion is more problematic. Reduction of excreted activity by urine could be adjusted by a change of soft tissue retention or directly by kidney to urinary blade or blood to urinary bladder transfer. There were no data on soft tissues and kidney thus blood to urinary bladder transfers was modified. The intervention does not change proportionality among kidney, soft tissue liver or skeleton i.e. does not affect model balance. Reduction of transfer rate from the blood to the small intestine, applied in JH case, has similar effect, i.e. does not significantly affect ratio among tissues and does not reduce overall dose too.

Author is aware that presented modification overestimate, compare to the standard model setting, dose and has an effect on early excretion of the americium in the model and thus used setting has neither general validity nor robustness. However, model adjustments improve significantly fit qualities to the level where can be accepted.

The trend of bioassay data was not imitated by the modified model. Insufficient data (no information on liver, kidney and soft tissue) does not permit more serious intervention to the model. Any trial in that way will be scientifically unjustified without data.

Conclusions

Every step of internal dose assessment was examined and relevant approach was used in the thesis. Application of MC technique for in vivo calibration and uncertainty assessment greatly improved skeletal data quality and provide valuable information on problematic issues among physical calibration phantom (namely USTUR BPAM-001 and BfS skull phantom). Estimated uncertainties of all data types enable better evaluation (proper weighting in fitting procedure) of the intakes as well as checking of the ICRP 67 americium model validity. Maximum likelihood fitting procedure together with statistical tests provided robust framework for intake estimation. Standard ICPR 67 model predictions between 5000 and 10000 days since intake were not able fit observed data of all subjects in acceptable manner. Person specific modification of the model lead to acceptable fits but not for urine data sets. Analysis of all cases together showed up that urine to feces ratio is predicted very well. Time trend of bioassay to skeleton ratios is quite constant in the model, while real data trends drop more rapidly. Detail analyses of the data trends imply that model underestimate retention in the skeleton, which was estimated from data to be about 77 years (mean half-time) over given time period. Time dependent reduction of exceed activity via bioassay is more rapid in the reality, i.e. excretion is lessen with the increasing time, than in the model. Person specific doses were determined however results are not only way to solution. All subject data and general findings were presented in 2009 on International Conference on the Health Effects of Incorporated Radionuclides and accepted for publication in Journal of the Health Physics Society (Malátová, in press).

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Radiation protection dosimetry using recombination chambers at radiotherapy facilities

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Abstract

Radiation protection measurements in radiotherapy departments concern either patients or medical personnel and technical personnel. In some rare cases radiation doses to other persons (carers, comforters and visitors), should also be evaluated. Patient protection involves in-phantom measurements, which are performed first of all for treatment planning but can also be used for determination of dose equivalents in patient's organs outside the treatment field. Occupational radiation protection is based on determination of ambient dose equivalent H*(10) at workplaces and on individual monitoring of exposed persons. In the paper we will discuss only the radiation protection dosimetry.

Usually, free-air ionization chambers are used for X-ray measurements at electron accelerators and neutron rem-meters with appropriate moderators for determination of neutron dose in case of radiotherapy with protons and heavy ions. The paper presents additional possibilities offered by use of recombination chambers and measuring methods based on initial recombination of ions. The main advantage is that the recombination methods can provide information about parameters of radiation quality (radiation quality factor, neutron-to-gamma dose ratio, distribution of absorbed dose versus LET).

The paper shows four examples how the chambers can be used at different radiotherapy facilities:

1) Measurements of H*(10) and neutron dose along the patient table at 15 MV X-ray therapy (Oncology Centre, Warsaw, Poland). The measurements were performed in the cabin, in the maze, in the control room and in the corridor at the entrance to the maze with slightly opened door;
2) Measurements of H*(10) and evaluation of radiation quality factor near a phantom irradiated by 170 MeV protons in the proton therapy treatment room of the Dubna phasotron (Russia);
3) Characterisation of mixed radiation field inside and outside therapy cave for eye proton therapy (60 MeV proton cyclotron, Kraków, Poland);
4) Measurements of H*(10) behind shielding of heavy ion therapy facility (GSI, Darmstadt, Germany).

Introduction

In general there are two types of radiotherapy facilities: electron/X-ray and hadron. In the paper we will discuss only the radiation protection dosimetry at those facilities because the dosimetry for radiotherapy is a substantially different subject (values of dose rates in the irradiated organ and outside shielding differ up to a billion times). For the radiation protection of the patient, personnel and other persons e.g. comforters, it is necessary to estimate the equivalent dose in the patient's organs or in appropriate phantoms, and the ambient dose equivalent values inside and outside the treatment room. This obviously should be taken into account when selecting an appropriate detector.

Radiotherapy departments are usually equipped with dosimetric detectors for X-ray and gamma radiations. Sometimes, there are also neutron dose-equivalent-meters (rem-meters), especially at hadron therapy facilities. The additional use of a recombination chamber can be advantageous, because this chamber is sensitive to all kinds of radiation and provides information on radiation quality.

Material and methods

Recombination chambers and recombination methods

Recombination chambers are high-pressure ionization chambers, designed in such a way that, for a certain range of gas pressure and dose rates, initial recombination of ions occurs when the chamber operates at polarizing voltages below saturation, and is much greater than volume recombination. This phenomenon is used for the determination of dosimetric quantities dependent on radiation quality, using a relationship between the initial recombination efficiency and local ion density (in ionised gas).

Electrodes of the chambers are mostly made with a tissue-equivalent material. Gas in the chamber usually contains hydrocarbons; however, hydrogen-free chambers are also used as detectors with very low sensitivity to high-LET radiation.

Measurements of H*(10) were mostly performed with a large recombination chamber of the REM-2 type. This is a cylindrical, parallel-plate ionization chamber with 25 tissue-equivalent electrodes, volume of 1800 cm^3, mass of 6 kg and effective wall thickness of about 2 g/cm^2. The chamber of the same design, but with Al electrodes and filled with CO_2 (2.8 MPa), is denoted as GW2. The tissue-equivalent chamber filled with BF_3 is referred as BOR-3.

All recombination methods are based on the comparison of ion collection efficiency at a given polarizing voltage U, in the investigated radiation field, and the ion collection efficiency at the same voltage U in the reference gamma radiation field (in our measurements, this was the radiation field of a ^{137}Cs source). Therefore, either the whole saturation curve should be determined or a number of points on this curve (at least two), depending on the dosimetric quantities to be determined and on the desired accuracy.

About 30 types of recombination chambers and about 20 recombination methods have been developed until now (Zielczynski and Golnik 2000, Golnik 1996, Zielczynski et al. 2007); however, only a few of them were used at RT facilities. Among them, there are:

(1) The method based on the determination of the recombination index of radiation quality (RIQ, Q_R) (Zielczynski and Golnik 1994). In this method, two values of the polarizing voltage should be applied consecutively to the chamber electrodes in order to determine the absorbed dose and to estimate the radiation quality factor. One voltage should be high enough to ensure that the chamber operates close to saturation. At the second voltage, specific for certain chambers, initial recombination occurs and the ion collection efficiency is measured.

(2) The extrapolation recombination method (ERM) (Zielczynski 2004). Measurements of the ionization current at four to seven voltages are needed in order to determine separately the low- and high-LET components of the absorbed dose.

(3) The recombination microdosimetry method (RMM) (Golnik 1995 and Golnik 1996). Here, about 20 polarizing voltages should be applied in order to obtain a crude distribution of the absorbed dose versus restricted LET. When RMM is used for the determination of only two components of the absorbed dose, associated with low- and high-LET particles, this method becomes similar to ERM.

Difficulties associated with measurements

(1) When recombination chambers operate also at low voltages (several volts), the charge memory effect may occur, i.e. the measured ionization current depends on history of the chamber's operation (Zielczynski and Golnik 2000). The uncertainty caused by the charge memory effect can be practically eliminated by performing measurements at both polarities of voltage applied and averaging the measured values of ionization current.

(2) Cables connecting the chamber with registration units are usually long. This is, above all, inconvenient; but also, long electrometric cables can cause additional noise therefore diminishing the accuracy of the measurements.

(3) Precise monitoring of facility radiation filed is needed, especially when sequential measurements of the ionization current at different voltages are performed.

Results

The examples shown here were selected mainly with the intention to present the measurements of such parameters which could be determined with recombination chambers, while they were not measurable neither with free-air ionization chambers nor with standard neutron rem-meters.

Measurements at the treatment room at a 15 MV medical accelerator at the Maria Curie Oncology Centre in Warsaw (Poland)

Several measurement series were performed at the 15 MV Varian Clinac 2300C/D accelerator at the Oncology Centre in Warsaw. H*(10) was measured also in the treatment room and in the control room using a BOR-3 chamber placed in a 1 kg paraffin moderator. The chamber was designed so that its response (the ratio of the current at appropriate voltage to H*(10)) was nearly the same for gamma radiation and for neutrons of the energy range similar to the neutron energy spectrum at medical accelerators.

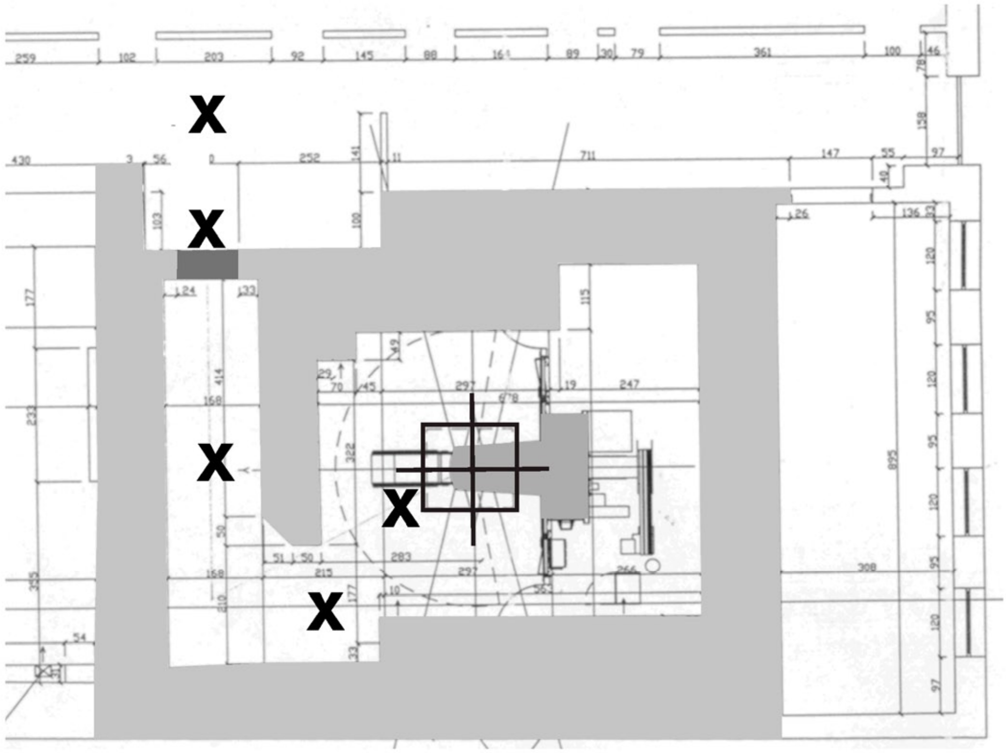

Fig. 1. The measurement points at 15 MV Varian at the Oncology Centre in Warsaw: by the patient table, in the maze, in the corridor (slightly opened door).

The following values of H*(10) have been obtained for a 4×4 cm^2 irradiation field, and a dose rate of 1 Gy/min at the isocentre: 33 mSv/h at 1 m from the isocentre (contribution of neutrons to the dose equivalent was found to be equal to 34%), 8 mSv/h at 3 m, 0.8 mSv/h in the maze and 1,3 μSv/h in the control room.

The measurements with recombination chambers have been performed also in the corridor behind the heavy shielding door (opened, partly opened and closed). This situation simulates the radiotherapy session in case of children irradiated and desired possibility of quick intervention of medical staff inside the treatment room (mentioned shielding door opens to slow).

The measurements were performed using the REM-2 chamber. During the measurements accelerator was operated continuously. The absorbed dose rate in the isocentre was 2.4 Gy/min. The total ambient dose equivalent rate at the entrance to the maze with door opened was H*(10) = 76 μSv/h, the ambient radiation quality factor Q*(10) was 2.6 Sv/Gy. The contribution of gamma radiation to H*(10) was determined by the hydrogen-free (neutron insensitive) recombination chamber GW2 was approximately 33% The value was H_γ*(10) = 24 μSv/h and neutron rem-meter Studsvik 2200 showed H_n*(10) = 51 μSv/h.

In the corridor with free access for people, the total ambient dose equivalent was 3 times smaller H*(10) = 25 μSv/h. When the shielding door was partly closed the value decreased to the value 2.7 μSv/h. Closing the door completely causes decrease of the value below 0.3 μSv/h).

Proton therapy facility of the Joint Institute for Nuclear Research in Dubna (Russia)

The measurements were performed at the Dubna phasotron using a 170 MeV proton beam with the intensity of 1.7 μA. The beam was directed onto the water phantom, and the Bragg peak was slightly broadened (2 cm at 90% of the maximum absorbed dose). The REM-2 chamber was used for the determination of H*(10) in the treatment room (Gryzinski et al. 2009).

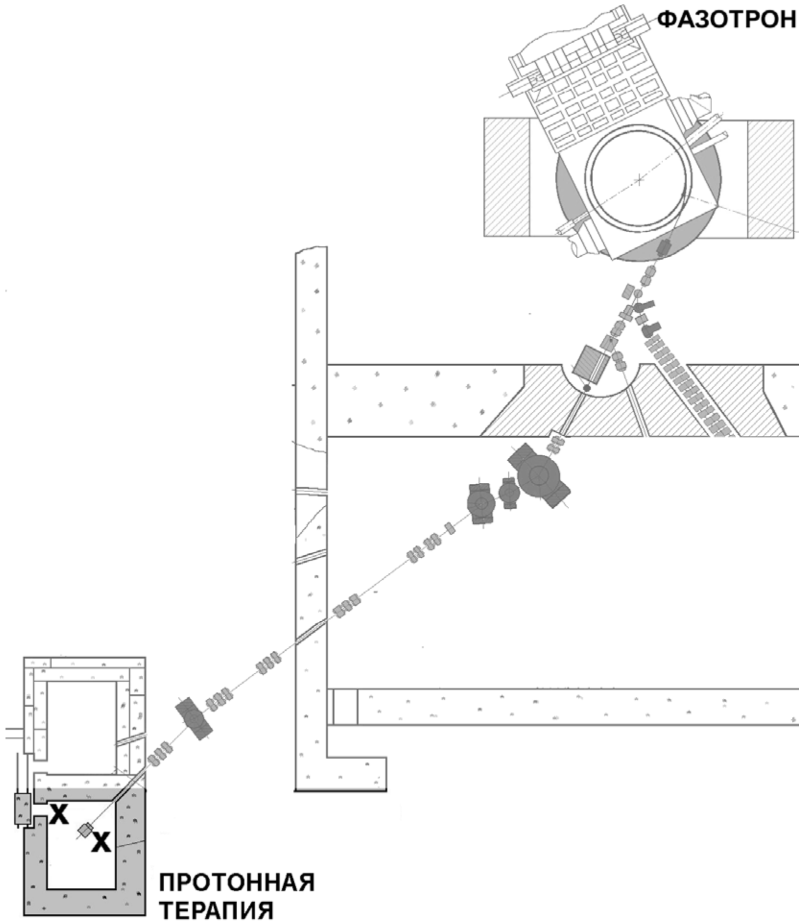

Fig. 1. The sketch of treatment room at 170 MeV proton therapy in Dubna.

A large recombination chamber of REM-2 type has been used for determination of the ambient dose equivalent, H*(10) and the ambient quality factors Q*(10) at different distances from the water phantom irradiated by 170 MeV protons – the water phantom represents the patient in the middle of the cabin. Measurement points represent places of eventually standing comforter: by the patient (at the distance 0.5 m and near the door at the distance 2.2 m from the isocentre). At the distance of 0.5 m, the ratio H*(10)/D was equal to 0.05 mSv/Gy, where D is the absorbed dose in the broadened Bragg peak. It shows that in specific cases it is possible to consider a bystander near the patient. It was shown that H*(10) was approximately inversely proportional to the distance from the phantom. The radiation quality factor Q*(10) was equal 3.5 ± 0.7 Sv/Gy at the distances from 0.5 m to 3.5 m.

The treatment room for proton therapy of the eye, Kraków (Poland)

An eye phantom was irradiated with a 40 MeV proton beam at 10% of the nominal intensity for the treatment. The total ambient dose equivalent rate (neutron + gamma), measured with an REM-2 chamber at the distance of 2.5 m from the phantom (point inside the treatment room P1), was H*(10) = 220 µSv/h, i.e. the value 10% higher than the reading of the neutron rem-meter NM2B of Nuclear Enterprises Ltd. The contribution of the gamma radiation, determined by a GW2 chamber, was about 5% of the total H*(10). The ambient quality factor Q*(10) was 7.7 Sv/Gy.

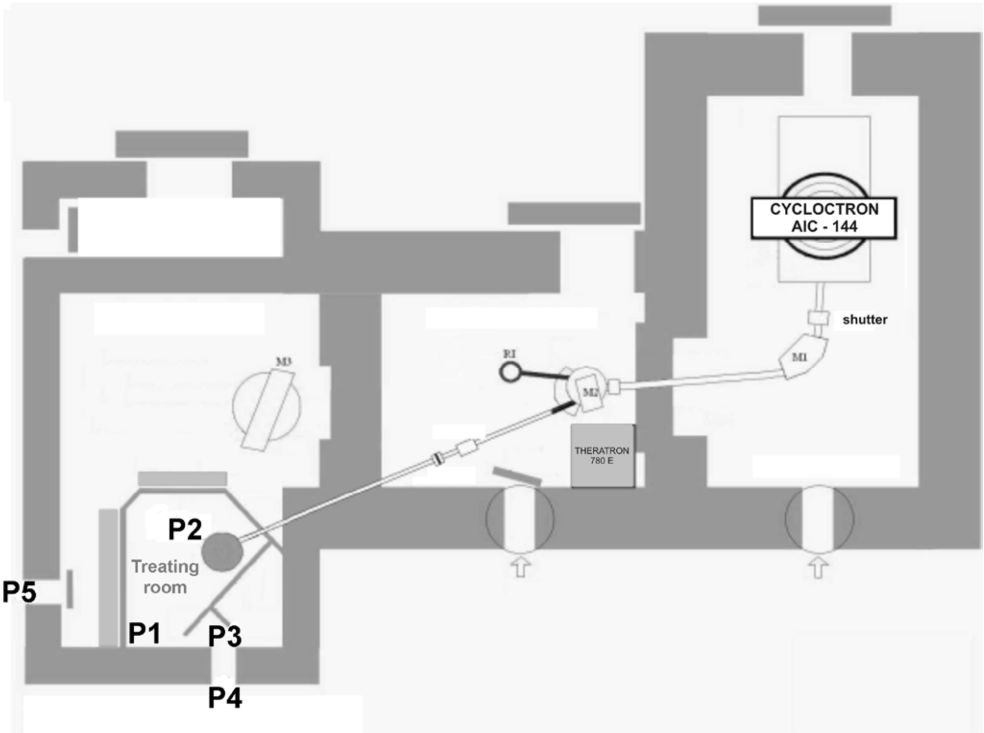

Fig. 1. The situational sketch of AIC-144 cyclotron in Kraków.

The measurements made inside the treatment room and in two points in the corridor (one near the entrance door P4 and the second on the line of accelerator beam P5) resulted in H*(10) 2 and 1.2 µSv/h, respectively. In the maze, close to the entrance door P3, the value was equal 18 µSv/h (40 MeV proton beam at 10% of the nominal intensity).

After shifting the energy to 57 MeV of protons in the facility in Kraków to its final form and intensity for eye therapy the values of H*(10) in those points are: 0.8 mSv/h at the distance 0.5 m (P2) and in the corridor 15 µSv/h (P4) and 10 µSv/h (P5). The contribution of gamma radiation was approximately 4% inside and 6% outside the treatment room.

Discussed possibility of a bystander near the patient for facility in Dubna could also be taken into account here. The ratio H*(10)/D for such person stayed during the treatment nearby patient (P2) was only 0.005 mSv/Gy, where D is the absorbed dose in the broadened Bragg peak (absorbed dose will not exceed 1 mSv for the treatment).

Outside shielding of a heavy ion therapy facility at GSI Darmstadt (Germany)

A beam of carbon ions with energy of 4.8 GeV was directed onto an Al target. Measurements of H*(10) were performed at the point denoted as OC-10, behind a concrete shield, using an REM-2 chamber, a hydrogen-free chamber GW2, and a BOR-3 chamber filled with BF_3 . The measurements gave the value of H*(10) = 26 µSv/h, which was about 1.5 times higher than the values measured with standard rem-meters. The contribution of gamma radiation to H*(10) was approximately 7%. The value of the ambient radiation quality factor Q*(10) was 4.7 Sv/Gy at the point of measurements.

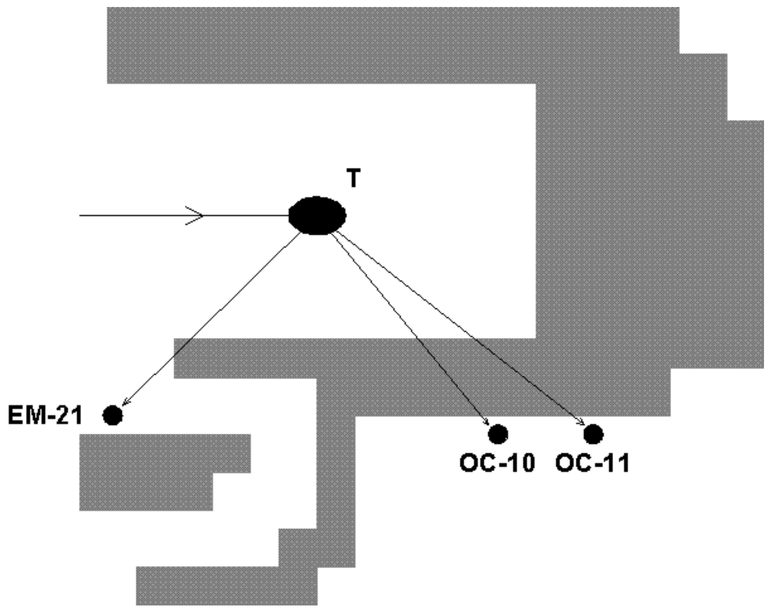

Fig. 1. The situational sketch of GSI ion therapy beam and concrete shielding. Results presented in this paper concern the point OC-10.

Conclusions

The measurements performed with recombination chambers at different radiotherapy facilities generating mixed radiation fields confirmed that the use of recombination chambers was very useful for the determination of the ambient dose equivalent and of other parameters of interest for radiation protection. The main advantage is that recombination chambers provide information on radiation quality, and it is possible to match the design of the chamber to the measure conditions.

The use of recombination chambers is especially advantageous in high-energy hadron therapy, because recombination chambers are sensitive to all kinds of radiation, including high-energy neutrons. The value of H*(10) measured with recombination chambers at high-energy facilities (outside the beam) is 10 to 50% higher than the values resulting from the measurements with standard rem-meters. There are two main reasons for such a difference: the first, that neutron rem-meters are insensitive to radiation other than neutrons, and the second, that they usually underestimate the ambient dose equivalent due to high-energy neutrons.

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Experimental monitoring of Ozone production in a PET cyclotron facility

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Abstract

In this paper Ozone production from radiolysis process (Kanda, 2001, 2005) due to radiation fields around a biomedical cyclotron has been studied. Ozone is known to be the most toxic gas produced by ionizing radiation around particle accelerators (Swanson, 1980), (Holloway et al., 1980) , (NCRP, 2005).

In order to evaluate Ozone concentration in working environment, two measurement campaign, with short term and long term, were performed, using passive diffusion dosimeters. Considering the concentration's limits issued by ACGIH (American Conference of Governmental Industrial Hygienists) as reference values, the results of measurement campaigns demonstrated that Ozone is typically present at levels of < 0.1 ppm and does not represent a significant risk factor for workers around a biomedical cyclotron.

Introduction

Intense radiation fields can produce Ozone in ambient air around particle accelerators, due to radiolysis process. This toxic gas is produced in the interaction of radiation with major air components, Oxygen and Nitrogen; this process is considered to be relatively independent from the type of radiation, so that it is expected that gamma, beta and also neutron radiation are effective Ozone generators (Swanson, 1980).

Ozone is a very reactive molecule. It can cause damages to the respiratory system if inhaled in ambient air concentration about 1 ppm; higher concentrations can cause also pulmonary edema. Moreover, it is known that there are relevant individual differences in response to Ozone inhalation (CDC, 2005). Being then Ozone presence a potential risk for operators at accelerator facilities (Holloway et al., 1980), an evaluation of its concentration in a specific work environment becomes a relevant component of health and safety assessment.

The number of cyclotron installation in the world is strongly increased, in the last 10 years, due to the success of Positron Emission Tomography (PET); in these installations, the principal risk factor for workers is clearly related to radiation fields;

nevertheless, concurring risk factors should be considered as well. Among these factors, Ozone needs particular attention; however, few data have been published as regards Ozone production in biomedical cyclotron installations.

In this work we tested the use of simple and inexpensive passive Ozone monitoring dosimeters, in order to assess their functionality in an accelerator facility and to obtain and estimate of Ozone concentration in the vault of a PET cyclotron and nearby laboratories.

Material and methods

In order to evaluate Ozone production at our installation, we perform a series of experimental measurements using a model of diffusion badge dosimeter (ChromAir, K&M environmental, AFC International Inc., La Motte IN). This is a direct reading, passive, diffusion dosimeter based on the indigo dye discoloration method (Bader, 1982).

The principle of operation of the diffusion dosimeters is shown in Fig. 1 .

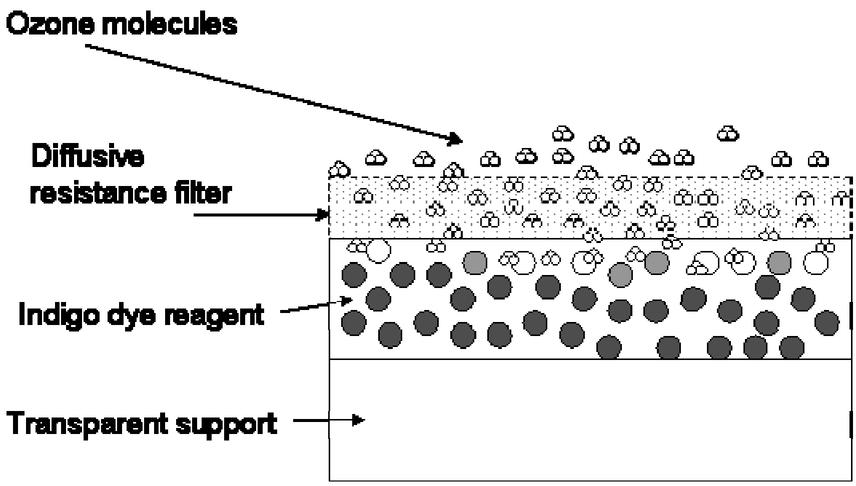

Fig. 1. Illustration of a section of the sensitive part of a dosimeter; the Ozone molecules diffuse gradually through diffusive resistances of different thickness. As they react with the dye (potassium indigo trisulfonate) this is decolored from blue to white.

Each dosimeter has six measurements cells, in contact with ambient air through diffusive resistance filters of increasing thickness; this produces a graduation of the exposure in each cell, that allows for a stepwise measurement on scale of integrated exposure (e.g. the product of concentration by exposure time) from 0.08 to 1.8 ppm·hour.

This type of device is simple to use and inexpensive, but its major advantage is that it is completely not influenced nor damaged by the intense radiation fields produced during irradiations; this makes possible to estimate the Ozone generation not only in working environments, but also inside the cyclotron bunker. The dosimeters are factory calibrated; reading is made by visual inspection and comparison with the background color in the scale reported in the back side of each badge (Fig. 2).

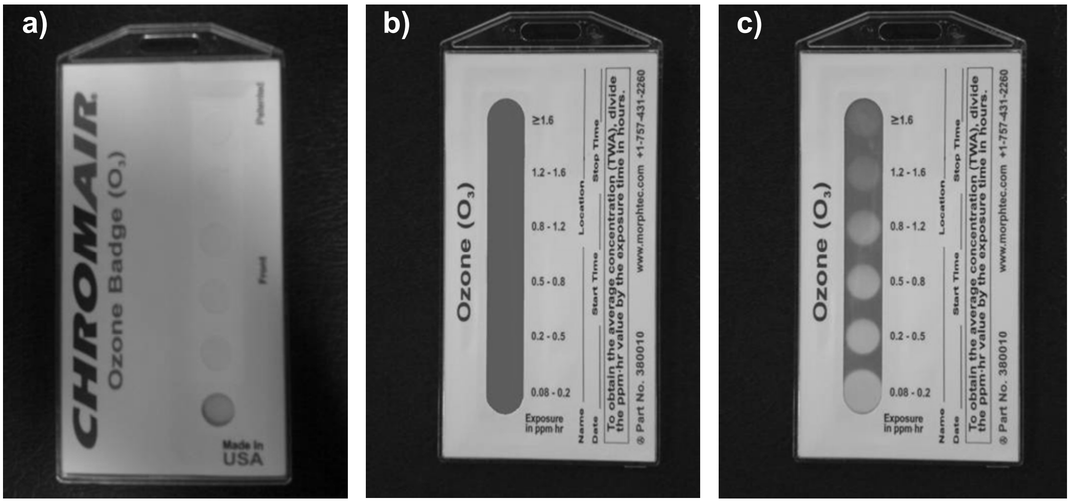

Fig. 1. a) front side of a dosimeter badge; b) Blank dosimeter backside; c) Dosimeter back side: decoloration due to Ozone absorption is evident.

As previously said, dosimeters provide measurement in ppm·hour; to set out the average concentration during the time of exposure, the reader has to pick the higher exposure cell, that is the higher in order with a color variation, and divide the corresponding value by the sampling time in hours.

On the other hand, these dosimeters are only moderately accurate; for our visual direct reading approach, measurement uncertainty has been evaluated as the half width of each measurement step between cell and cell. This means an uncertainty of the order of 40% at the lower level of concentration of 0.08 ppm·hour; in the central range of measurement, at a level of about 1 ppm·hour the uncertainty is evaluated at about 20 %; this is considered in line with the type of device and acceptable for the aim of this work.

Measurements were performed during routine irradiations with our GE PETtrace cyclotron, accelerating H- ions at 16.5 MeV on H218O liquid target for 18F- production (Marengo et al., 2008). The cyclotron in our site is "naked", e.g. not self-shielded, so that the total mass of air irradiated is relevant; the total volume of the vault is 120 m^3.

Diffusion dosimeters were placed in significant positions: inside the cyclotron bunker, in the radiopharmacy laboratory and, to assess the Ozone background, in adjacent rooms in which leakage radiation is very limited and outside the PET facility. The dosimeters positions were chosen in accordance with previous works, where radiation field at our site were characterized (Gallerani, 2008). Dosimeters were placed at an height of 1.5 m from the floor.

Two different series of measurement were preformed: a) limited time exposures, with the ventilation system off during normal and prolonged irradiation cycles and, b) long time exposures, with the ventilation system on the bunker working at the normal rate of air changes.

Limited time exposure were made either by exposing the dosimeters during typical 1 hour bombardment, with ventilation off, the removing the dosimeters, packing them in a sealed plastic bag and storing at 4°C until new exposure. A total time of 7.6 hours of bombardment were sampled in this way; furthermore, a single prolonged irradiation of 3.7 hours was performed.

One dosimeter was placed in air at a distance of 100 cm from the $H_2{}^{18}O$ target, at an angle of 45° from the beam direction. In the geometry of installation of the PETtrace cyclotron, this position represents the reference point for dose measurements. A second dosimeter was placed inside the bunker, on one of the peripheral walls at 180° form the beam direction and at a distance of 350 cm from the target.

During these measurements a "blank" dosimeter was positioned outside of the bunker, in the electronics technical room at a distance of 300 cm from the bunker shielded door. This dosimeter was left exposed to ambient air during the whole time interval of measurement (223 hours).

In long time sampling, with room ventilation on at normal air exchange rate of 10 vol/hour, two dosimeter were again placed in the bunker, one at the reference position at 100 cm form the target and the second on the peripheral wall at 180° at a distance of 350 cm from the target. Other dosimeters were placed in the electronics technical room, in the radiopharmacy laboratory (two positions), in front of the two most used synthesis hot cells, at a distance of 250 cm form the surface of the bunker wall and about 100 cm from the front of the hot cells. Furthermore, a "blank" dosimeter was placed in the exterior of the PET centre building, to sample environmental Ozone outside of the facility. Two long term sampling, each for a time of 240 hours of continuous exposure were performed. During each of these samplings, the effective bombardment time was of 20 hours, while during the remaining time the cyclotron was in standby mode.Body text, first paragraph after the heading.

Results

During all the experimental measures, no sign of damage was observed at a visual inspection on all the 30 dosimeters used. Activation was negligible, even for the dosimeters placed at the reference position at 100 cm from the target, were the secondary neutron flux is relatively intense.

In the short time measurements performed only during bombardment time, with room ventilation off, none of the dosimeters showed any discoloration, even in the first cell. Ozone exposure was lower than the minimum detectable quantity. Being the lower level of measurement of the dosimeters at 0.08 ppm·hour, the upper limit of concentration was obtained by dividing the detection limit by exposure time. Table I reports the results of these measurements; it has to be recalled that the "blank" dosimeter in this series of sampling was exposed during all the campaign (223 hours); The results of long term samplings of 220 hours, with normal room air ventilation, are shown in Table II.

In these measurements, all the detectors being exposed for a long time, thus to increased amounts of Ozone, showed discoloration up to the third / fourth cell. However, average concentrations resulted to be very low and in agreement with the minimum detectable concentration evaluated for short term measurements.

All the average concentration measured were less than 0.01 ppm and, in particular, there was no significant difference, in consideration of measurement uncertainties, between the values obtained in the working areas and in environment air outside of the facility. In long term irradiations with ventilation off ($^{18}F^-$ target loaded with $H_2{}^{18}O$, 4 hours irradiation at 40 μA), the concentration measured was less than 2.10^{-2} ppm.

Table 1. Summary results of the short term measurements with room ventilation off.

Measurement position	Average concentration (ppm)	Uncertainty (ppm)
Inside the bunker, at reference position at 100 cm from the target	< 0.01	0.008
Inside the bunker, peripheral wall	< 0.01	0.008
Electronics technical room	0.002	0.0007

Table 2. Summary results of the long term measurements with room ventilation on.

Measurement position	Average concentration (ppm)	Uncertainty (ppm)
Inside the bunker, at reference position at 100 cm from the target	0.006	0.0008
Inside the bunker, peripheral wall	0.005	0.0008
Electronics technical room	0.003	0.0006
Radiopharmacy lab position 1	0.005	0.0008
Radiopharmacy lab position 2	0.005	0.0008
Exterior of the facility	0.006	0.0008

Conclusions

The American Conference of Governmental Industrial Hygienists (ACGIH) indicates exposure limit values, TLV (Threshold Limit Value), representing the concentration levels in ambient air to which workers may be exposed without adverse effects for health (ACGIH, 2001, 2009). Concentration's limits issued by ACGIH were assumed as reference values for comparison of our results (CTIPLL-ISPESL, 2006).

The most important quantity to be considered is the TLV-TWA, set to 0.1 ppm for Ozone. TLV-TWA is the Time Weighted Average limit, that represents the value of concentration to which workers are exposed in continuous, long time working conditions.

It's important to pay attention to the fact that TLV does not represent a border line between danger and safety, but they are only indicative values for comparison in the sake of optimization .

The main result of our work is that all our experimental measures showed that Ozone concentration in a busy PET facility with a non self-shielded cyclotron, are at least one order of magnitude less than the TLV, not only in the radiopharmaceutical laboratory area, but also inside the cyclotron vault.

According to our results, if an assessment of Ozone production in a PET cyclotron facility still worth careful consideration, it is confirmed not to be a relevant risk factor, even in conditions of limited air exchange rate. This conclusion can guide choices in term of the needs for monitoring. In particular, on the basis of our results, provision for installed systems for continuous sampling or investment for dedicated active monitoring equipment should be considered not justified.

Instead, passive, easy to use and relatively inexpensive dosimeters like the ones we tested in this study, proved to be adequate and, taking into account complete insensitivity to radiation fields, they can be considered a useful tool for periodic monitoring campaigns aimed to confirm working conditions.

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Reproducibility assessment for a new neutron dose evaluation system

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Abstract
The Paul Scherrer Institut (PSI) is investigating a new neutron dose evaluation system manufactured by TASL. This system takes images of the etched tracks in CR-39 detectors by a high magnification microscope, which are then analysed by a software algorithm. Each measured track is characterised by a multitude of separate parameters, which are then used in the neutron exposure algorithm to enable optimum noise discrimination, sensitivity calibration and dose calculation. However, before the system can be used in the routine dosimetry service, it has to be tested for its intrinsic properties, such as reproducibility, linearity, stability etc. In this paper we focus on the reproducibility of the evaluation process. Three sets of detectors, i.e. non-irradiated detectors as well as detectors irradiated with 3 mSv and 6 mSv, were analyzed for reproducibility study throughout the duration of 10 weeks. Interesting behaviour was observed, which will be used as input for potential improvements in the treatment of the detectors and for possible adaptations of the evaluation algorithm by the manufacturer.

Introduction
The Paul Scherrer Institut in Switzerland has been accredited by the Swiss Accreditation Service (SAS) for measuring the personal neutron dose equivalent with chemical etched CR-39 (PADC, Poly Allyl Diglycol Carbonate) detectors since 1998. Since then the track counting has been performed with the Autoscan 60 reader, where a “light-in-the-detector’s side” technique is used, which causes the etched tracks or pits in the detector to be seen as points of light and can hence be counted (Fiechtner and Wernli 1999). However, the Autoscan 60 has the disadvantage that it is no longer manufactured and a direct consequence of this is the shortage of spare components. For this reason the PSI is investigating a new evaluation system manufactured by TASL. However, before the system can be used in the routine dosimetry service, it has to be tested for its intrinsic properties, such as reproducibility, linearity, stability etc. In this paper we focus on the reproducibility of the evaluation process.

Material and methods

CR-39 detectors

The study was carried out with CR-39 detectors from the manufacturer TASL. For measuring the neutron dose the CR-39 detector is packed into a dosemeter holder (Figure 1), made of hydrogenous material (about 10% hydrogen). The neutron dosemeter is sensitive to neutrons with energies above ~200 keV.

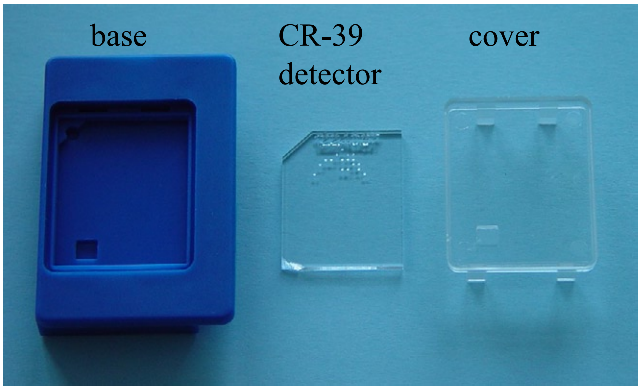

Fig. 1. CR-39 detector and dosemeter holder (base and cover).

Etching procedure and evaluation

Before the CR-39 detector can be evaluated, it has to undergo a chemical etching process. According to the recommendations by the manufacturer, the detectors are etched for 2 h 50 min with 6.25 N sodium hydroxide at 85°C. The etching process is terminated via a neutralisation for 15 min in a weak hydrochloric acid solution (0.1 N HCl) and a cleaning with hot and cold distilled water. After this process the detectors are evaluated on the evaluation system by TASL. This system acquires images of the etched tracks by a high magnification microscope, which are then analysed by a software algorithm. Each measured track is characterised by a multitude of separate parameters, such as covering size, shape, position, optical density, noise discrimination or quality of measurement. These parameters are then used in the neutron exposure algorithm to enable optimum noise discrimination, sensitivity calibration and dose calculation.

Measurements and Results

Reproducibility testing was performed with a set of 30 detectors consisting of 10 non-irradiated detectors, 10 detectors exposed to a personal neutron dose equivalent of 3 mSv and 10 detectors to a personal neutron dose equivalent of 6 mSv. The calibration was performed in an Am-Be reference field at the calibration laboratory of PSI. All detectors were analyzed with the TASL-system 16 times over the course of two months. Initially a scan was carried out every second day and afterwards with an increasing time interval but always at least once per week. The readings of each sub-set over time are shown in Figure 2-4. From the figures following observations can be made:

(a) The non-irradiated detectors have a mean background dose equivalent of 0.03 mSv. The standard deviation is 0.06 mSv. Because of the internal dose algorithm the background values can also be slightly negative. Over the period of testing the analyzed set shows a constant pattern.

(b) The set of detectors irradiated with a reference dose equivalent of 3 mSv show a mean dose equivalent of 3.6 ± 0.8 mSv over all readings during the complete time period. The measured dose increases over time from a mean value of 3 ± 0.2 mSv at scan 1 (averaged over detector 1 to 10) to a mean value of 3.8 ± 0.8 mSv at scan 16. From time to time outliers (scan 5, 8, 15) are observed, which were not identified by the software as a problem.

(c) For the set of the detectors irradiated with a reference dose equivalent of 6 mSv the average dose equivalent of 6.1 ± 1.0 mSv was obtained over all readings during the complete time period. The measured dose increases over time from a mean value of 6 ± 0.6 mSv at scan 1 (averaged over detector 1 to 10) to a mean value to 6.6 ± 0.6 mSv at scan 16. Detector 9 shows a strange behaviour in the middle of the testing period, but this effect was indicated as “rejected” by the evaluation software. The variance of the readings is smaller for the higher reference dose of 6 mSv than for 3 mSv.

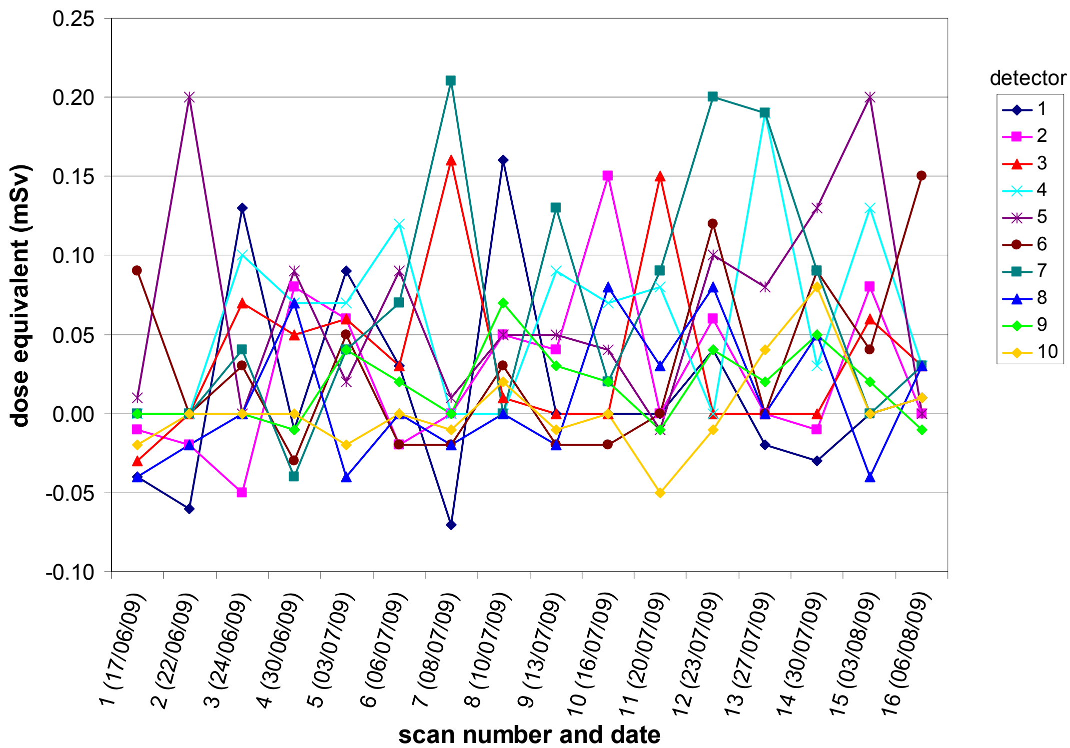

Fig. 2. Readings of 10 non-irradiated detectors versus scan date.

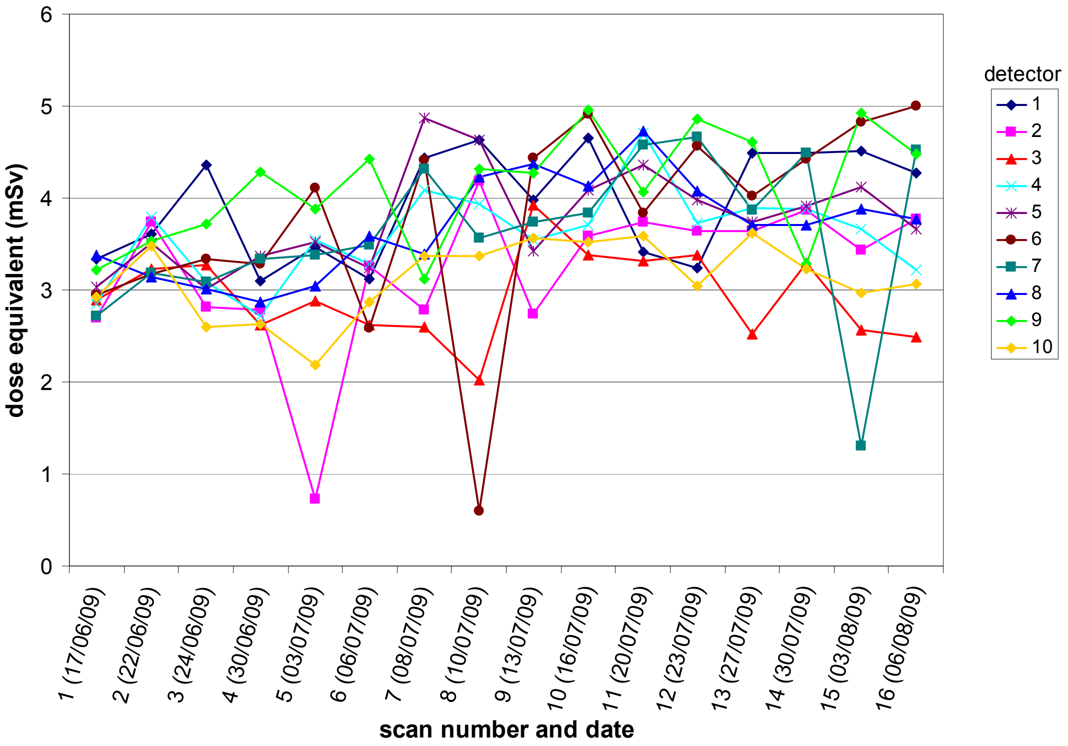

Fig. 3. Readings of 10 detectors irradiated with 3 mSv versus scan date.

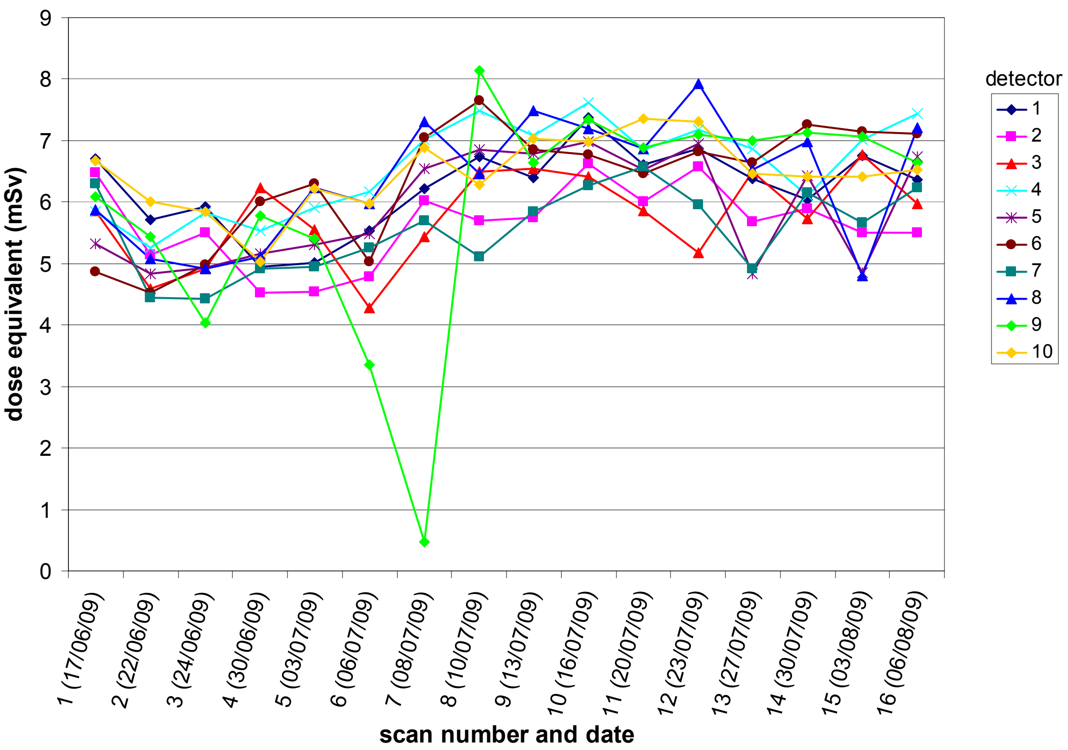

Fig. 4. Readings of 10 detectors irradiated with 6 mSv versus scan date.

Discussion and conclusions

The analysis of the non-irradiated detectors shows a stable behaviour over the testing period. In case of the irradiated detectors, a time effect could be observed. Especially the detectors irradiated with 6 mSv appear to have lower doses during the first few scans before settling down to a higher dose value. In addition, the study shows that some detector's scans have unexpectedly low values. Only in case of the higher reference dose, this effect was indicated as a problem by the evaluation software and rescanning was advised. The other outliers were not identified by the software algorithm.

The study indicates that there is a strong need for a subsequent reproducibility study to collect more data for a better interpretation of the behaviour of irradiated detectors. Moreover, linearity tests and an intensive collaboration with the software developers are foreseen for eventual adaptations of the neutron dose algorithm.

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Simulations of radiation fields of a photon and a fast neutron calibration facility

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Abstract
At the Karlsruhe Institute of Technology (KIT) (formerly: Forschungszentrum Karlsruhe (FZK)) exists a calibration laboratory for dosimetry and radiation protection purposes. The laboratory provides different radiation fields such as photons stemming from Cs-137 sources and X-ray devices as well as fast neutrons from a Cf-252 source.

The reference fields, which are mainly used for calibrations of active and passive radiation protection monitoring instruments, are characterized according to ISO calibration procedures. In order to verify that radiation field qualities are according to international standards not only measurements but also simulations were performed. In this way suitable conditions for research and calibration purposes are guaranteed. Additionally the variability of the fields (including backscattering) which is difficult to impossible to measure can be determined by simulations.

The simulations were carried out with the Monte Carlo Codes MCNP5 and PENELOPE2008.

Introduction
In the field of radiation protection dosimeters and dose rate meters are used in many areas such as individual monitoring or environmental monitoring. The corresponding measuring instruments are calibrated in reference radiation fields defined by a national or international standard. The proof that the fields comply with the regulations of ISO standards can be provided by measurements and/or radiation transport simulations. But measurements are often too complex and laborious so that simulations are used which can be verified by few relevant measurements.

The calibration laboratory at KIT provides different photon and neutron radiation fields for dosimetry and calibration purposes. Among other things the calibration laboratory possess a Cs-137 radiation facility consisting of four Cs-137 sources with different source strengths and an X-ray irradiation facility consisting of a low and a high energy X-ray tube. The photon irradiation devices are installed in a concrete bunker, while a Cf-252 neutron radiation facility is located in a wooden hall.

For research and quality management purposes simulation models of the different calibration facilities were created and first simulations to investigate the properties of the radiation fields of these facilities were performed.

Examples of investigations are:

- The determination of the influence of backscattering on the dose conversion for the Cs-137 irradiation facility
- The determination of X-ray spectra for the X-ray irradiation facility and the comparison with reference spectra
- A dose estimation at and around the Cf-252 neutron irradiation facility

The determination of the influence of backscattering on the dose conversion for the Cs-137 irradiation facility

Dosimeters for individual and environment monitoring indicate the dose (or dose rate) in personal dose equivalent or ambient dose equivalent. However, air kerma is the radiation quantity often used for calibrated ionization chambers (secondary standards) to determine the reference photon dose at a calibration point. Therefore conversion coefficients from e.g. air kerma to ambient dose equivalent are needed.

For a proper characterization of the Cs-137 irradiation field it is essential to know the uncertainty of the dose conversion. The uncertainty is mainly caused by backscattered photons who lead to a difference between the real Cs-137 energy spectrum at the point of calibration and the theoretical ideal energy spectrum defined in the international standard ISO 4037-1 (ISO 4037-1 1996). This difference could lead to non-negligible uncertainties when applying a dose conversion calculation.

The ISO 4037-1 requires an uncertainty of 2 % for the dose conversion, when no better estimation is known. An irradiation place specific estimation for the dose conversion uncertainty was obtained from simulations with the radiation transport program MCNP5 (X-5 Monte Carlo Team 2003).

The determination of X-ray spectra for the X-ray radiation facility and the comparison with reference fields

The energy distribution of X-ray radiation fields is dependent on the construction of the X-ray tube and the surrounding area. Variations of the energy distribution can lead to fundamental dose variations and therefore wrong calibrations. For that reason one has to prove that the energy distribution of the calibration fields conforms to the reference energy distribution given by ISO 4037-1.

Simulations with the radiation transport program PENELOPE2008 (Salvat et al. 2008) were performed to determine the theoretical energy spectra produced by the two X-ray tubes of the calibration facility. The simulation model is based on the technical drawings of the X-ray tubes. The simulated energy spectra were compared to energy spectra which were measured at the reference radiation fields of the Physikalisch-Technische Bundesanstalt (PTB). Mean energies as well as the full-width-at-half-maxima (FWHM) of the spectra were determined.

The dose estimation for fundamental measuring points of the Cf-252 neutron radiation facility

The calibration laboratory possesses a Cf-252 neutron irradiation facility. The neutron source is now 7 years old and according to the half-life of 2.7 years the source activity becomes quite low. Therefore it is aimed to buy a new Cf-252 source with a higher activity. Due to the short life-time it is advantageous to order the highest possible source activity with respect to the given dose limits at and around the facility.

Simulations with the radiation transport program MCNP5 were performed to estimate the neutron dose at different measuring points inside and outside of the calibration laboratory. By comparison of the calculated doses with measured ones the used simulation model could be verified and the highest tolerable activity for a new source could be estimated.

Material and methods

Simulation models for the radiation facilities were created and simulations were performed either by using the radiation transport program PENELOPE2008 or MCNP5. MCNP5 was used for the Cs-137 photon source and the Cf-252 neutron source. PENELOPE2008 was used for the X-ray irradiation facility, because it is optimized for simulating electrons and photons in the low energy range of X-rays. While MCNP5 uses class I algorithms, PENELOPE2008 allows to apply a class II algorithm.

Cs-137 source

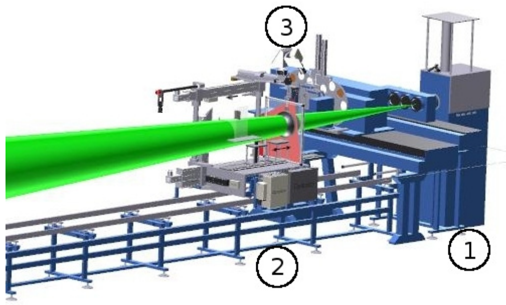

Figure 1. Drawing of the photon irradiation facility inside the concrete bunker (Liedtke 2008): Shown are the shielded storage place for the Cs-137 sources (1), the slide system to place dosimeters/devices at different distances on the central axis of the photon beam (2), in green an example for a collimated Cs-137 beam, and the two X-ray tubes which can be moved to the central beam axis instead of the Cs-137 sources (3).

A simplified model of the photon irradiation facility was created. Figure 1 shows the drawing on the facility inside the concrete bunker. Four Cs-137 sources with different source strengths are located in a shielded storage place and can be moved out in a system which provides a collimated beam. A slide system provides the possibility to place dosimeters/devices at different distances on the central axis of the Cs-137 beam. For the simulations the surrounding concrete walls and floors were also considered.

X-ray device

Models of the two different tubes were realized according to their construction specifications (see fig. 1 for the location of the devices). The simulation of the X-Ray fields originates in the process of electrons impinging on the anode of the X-ray tube. The simulations include the influence of self-absorption and the effect of the exit window as well as the individual absorbers required for N-Series standards after ISO-4037-1. In figure 2 a "model"-tube is displayed.

To determine the spectra of the fields, a „virtual“ detector was placed on the central beam axis at one meter distance from the anode. This geometry complies with the measured reference spectra from PTB (Böttcher Büermann 2009).

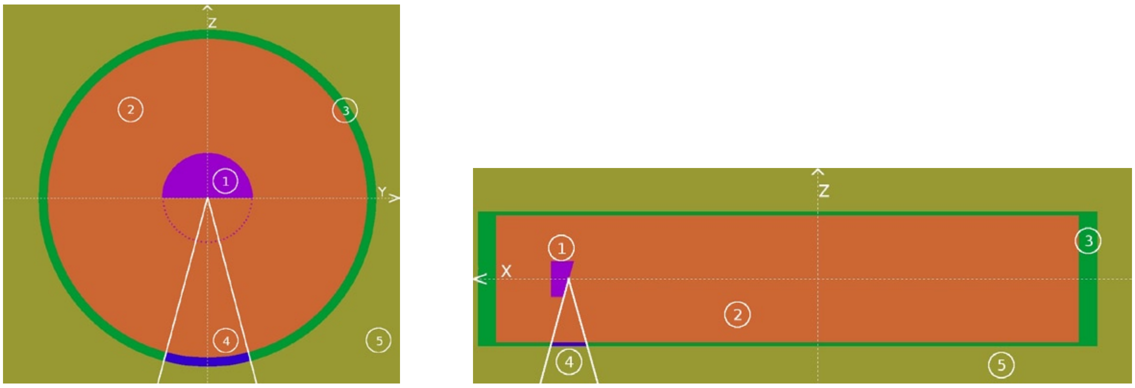

Figure 2. Simulation model of the low energy X-ray tube in top (left) and side view (right) (Harendorf II 2009): Shown is the tilted anode (1), the vacuum (2) in the steel cylinder (3), the exit window (4) and the surrounding air space (5).

Cf-252 source

The Cf-252 source is stored in a shielded place in the ground and can be moved out in a flexible tube to the irradiation place. Driven out in irradiation position, the source is placed in a thin aluminium cylinder. This cylinder together with the source inside has a PTB-reference calibration for the neutron flux. To realize almost free-in-air conditions the device is installed in a wooden irradiation hall and usually mounted 3-4 m above the ground.

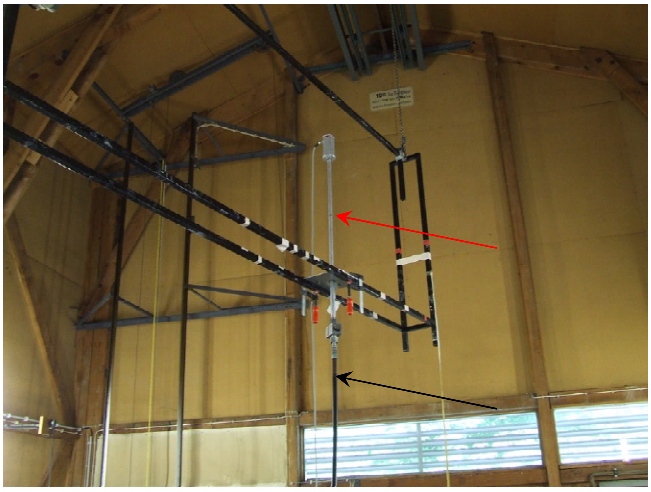

Figure 3. The irradiation device in the wooden irradiation hall: The red arrow indicates the position of the moved out Cf-252 source in a thin aluminium cylinder. The source can be moved to this place in a flexible tube from a shielded storage place (black arrow). The black frame of a provisional hoist system and parts of the wooden hall are visible too.

Results

Cs-137 source

Simulations were performed to investigate the uncertainties of the conversion from the photon fluence into air kerma and into ambient dose equivalent, respectively. Scattered photons in the irradiation facility yield the main contribution to the uncertainty of this conversion procedure. From both conversions finally an uncertainty of the conversion from air kerma into the ambient dose equivalent was deduced. The result for this uncertainty amounts to 1.34 % (Harrendorf I 09).

However this value for the uncertainty is mainly limited by the statistics of the simulations. To obtain the result 7.5 x 10^8 source photons were simulated with MCNP5. With the used complex geometry of the irradiation device including walls and floors the simulation took almost a week to finish.

X-ray source

In figure 4 a comparison of simulated and measured X-ray spectra is exemplary shown for a tube potential of 40 kV (Harrendorf II 2009). Plotted is the "PENELOPE" named spectrum of the simulation and the two spectra measured at PTB (PTB 150 kV, PTB 300 kV). The simulated and measured spectra agree quite well.

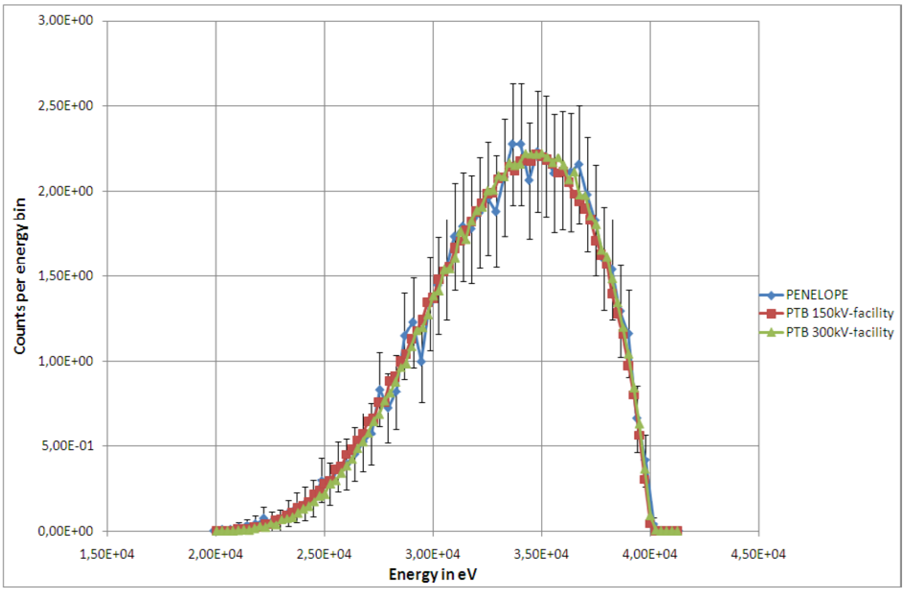

Figure 4. X-ray spectra for an X-ray tube potential of 40 kV (Harrendorf II 2009, Böttcher Büermann 2009). The error bars belong to the simulation.

Table 1 shows the mean energy E_{SIM} of the simulated spectra and the mean energy E_{ISO} for the specific tube potential according to ISO 4037-1. ΔE represents their percent difference.

According to ISO 4037-1 a calibration laboratory has to prove that the deviation is less than 3 % for tube potential ≥ 30 kV and 5% for tube potential < 30 kV. For all tube potentials shown in table 1 the difference is less than 3 % and therefore the requirements of the ISO 4037-1 are fulfilled.

Furthermore the resolution RSIM (FWHM) of the simulated spectra is shown in table 1. For comparison the specified resolution RISO of ISO 4037-1 is listed together with the difference of both resolutions ΔR. The ISO 4037-1 requires that the resolution of the spectra differs less than 15 % for a tube potential below 30 kV and 10 % for a potential ≥ 30 kV. This is fulfilled for all tube potentials shown in table 1.

Table 1. Results for the mean energy and resolution of the X-ray spectra (Harrendorf II 2009): Shown is the mean energy for the simulated spectra E_{sim}, the mean energy given by ISO 4037-1 E_{ISO} as well as their percent difference ΔE. Furthermore the resolution (FWHM) of the simulated spectra R_{SIM}, the required resolution R_{ISO} after ISO 4037-1 and the their difference ΔR is given.

Radiation quality	Tube potential	E_{SIM}	E_{ISO}	ΔE	R_{SIM}	R_{ISO}	ΔR
N-15	15 kV	12355 eV	12000 eV	2,96 %	29,01 %	33,00 %	12,10 %
N-20	20 kV	16335 eV	16000 eV	2,10 %	32,02 %	34,00 %	5,83 %
N-25	25 kV	20395 eV	20000 eV	1,98 %	31,29 %	33,00 %	5,17 %
N-30	30 kV	24715 eV	24000 eV	2,98 %	29,03 %	32,00 %	9,28 %
N-40	40 kV	33342 eV	33000 eV	1,04 %	30,68 %	30,00 %	2,28 %
N-60	60 kV	47940 eV	48000 eV	0,13 %	36,70 %	36,00 %	1,94 %
N-80	80 kV	65249 eV	65000 eV	0,38 %	32,59 %	32,00 %	1,85 %
N-100	100 kV	83403 eV	83000 eV	0,49 %	29,21 %	28,00 %	4,32 %
N-120	120 kV	100500 eV	100000 eV	0,50 %	28,78 %	27,00 %	6,60 %
N-150	150 kV	114879 eV	118000 eV	2,64 %	37,95 %	37,00 %	2,58 %
N-200	200 kV	162513 eV	164000 eV	0,91 %	28,02 %	30,00 %	6,61 %
N-250	250 kV	205993 eV	208000 eV	0,97 %	27,34 %	28,00 %	2,34 %
N-300	300 kV	249713 eV	250000 eV	0,11 %	26,70 %	27,00 %	1,13 %

Cf-252 source

Measurements and simulations were performed in and outside the wooden calibration hall. The position of the measurements and simulations of a dose distribution are displayed in figure 5. The simulated dose distribution inside the hall is indicated by different colours from red (high dose) to dark blue (low dose). The simulations and measurements were performed for positions below the source. Therefore the simulations show a reduced dose close to the source position due to shielding effects of the holding structure and flexible tube.

Figure 6 shows the comparison of the dose rate for both the measurements and simulations for the positions shown in fig. 5. There is a good agreement between the measured and simulated results. The differences of ≤ 20% are covered by the uncertainty of the LB 6411 neutron dose rate detector (Berthold Technologies 1995).

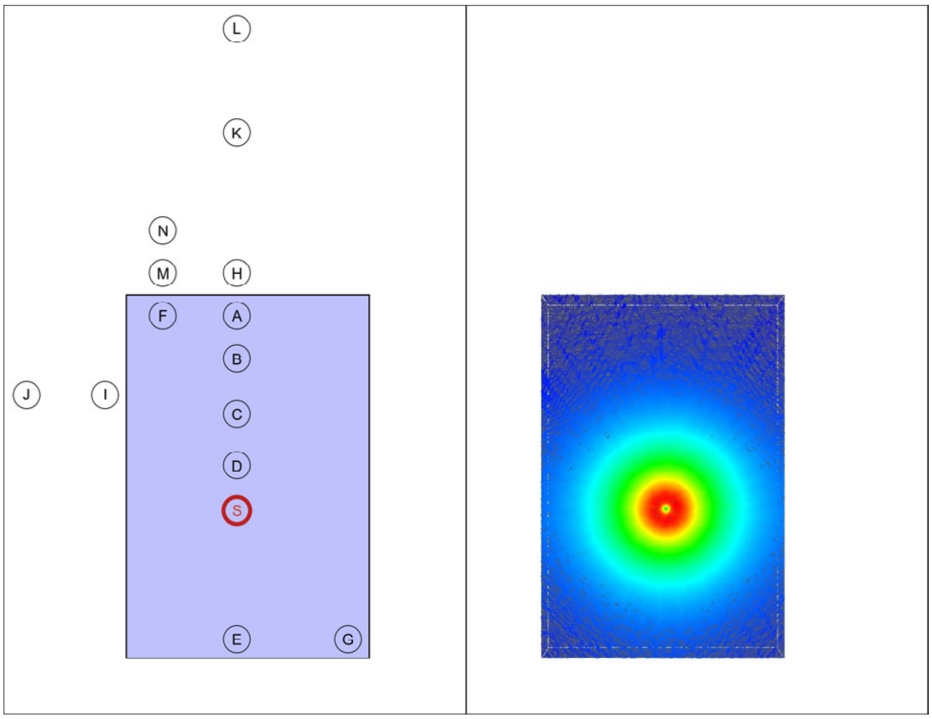

Figure 5. Left: Location of the measuring points for the determination of the neutron dose rate: the outline of the wooden hall is shown in blue, the source (S) in red and the different measuring positions are labelled with (A-N). Right: simulated dose distribution inside the hall.

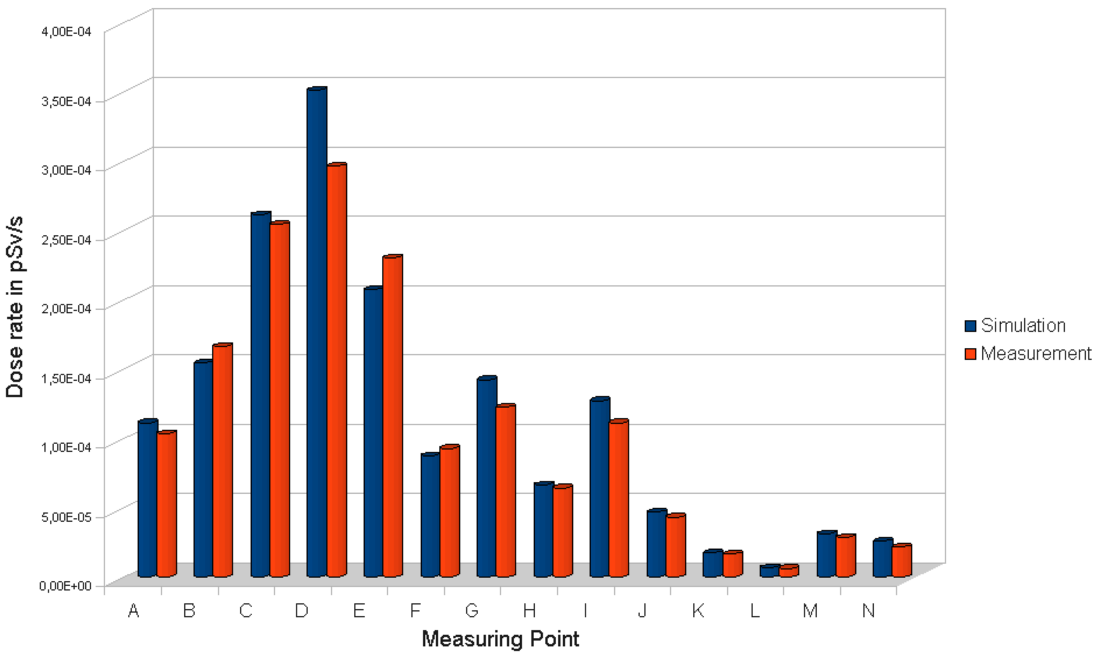

Figure 6. Comparison of the neutron dose rate of the simulations and measurements (Harrendorf 2008): shown is the ambient neutron dose rate in pSv/s normalized to one neutron emitted by the simulated source. The position of the different points are indicated in figure 5.

Discussion

The first simulations for radiation fields of the three different devices in the KIT calibration laboratory show good results. All simulation data contribute to the goal to describe the radiation fields in a suitable way. Usually simulations have to verified by measurements to be reliable.

For the Cs-137 irradiation facility a smaller uncertainty than given by ISO 4037-1 could be estimated. The uncertainty is mainly influenced by the amount of backscattered photons in the radiation field. For the simulations also the uncertainty of the statistics plays an important role. May be our simulation results could be improved with an increased number of source photons.

For the X-ray radiation facility the investigated spectra (Narrow-spectrum series) fulfil the following requirements of ISO 4037-1:

- Within an uncertainty of 5% the simulated X-ray spectra agree with those of the PTB reference fields
- The deviation of the mean energy of the simulated spectra are within the ISO limits
- The resolution of the spectra is conform to the ISO specifications

The Cf-252 irradiation facility can be well described by simulations. Hence detailed dose estimations e.g. for radiation protection purposes can be obtained in this way.

Conclusions

A future goal is to employ simulations for quality management purposes. To determine or verify the properties of a radiation field with measurements alone is often too time-consuming and complicated. The measurement of the X-ray spectra and to measure the amount of backscattered photons in a radiation field are examples. Once a verified simulation model for a specific irradiation device exists, simulations should be able to describe the radiation fields properly. Therefore reliable simulations could limit measurements to a few key-measurements verifying the simulations and at the same time the device properties. For example the half-value layer for an X-ray irradiation device could be measured only for one or two radiation qualities indicating the correct working of the device. Moreover, the simulations could indicate the most sensitive measurements for this purpose.

For quality assurance regular repeated controls of the radiation fields are mandatory. The employment of simulations as quoted above would help to reduce the effort for this procedure.

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TL and OSL techniques for calibration of $^{90}Sr+^{90}Y$ clinical applicators

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Abstract
The $^{90}Sr+^{90}Y$ clinical applicators are beta sources used in brachytherapy in the treatment of superficial injuries of eyes and skin. According to international recommendations, these applicators must be calibrated periodically. Thermoluminescent dosimetry (TLD) may be used for the calibration of these sources. However, the optically stimulated luminescence (OSL) technique has been already demonstrated as useful for beta dosimetry. In this work, thermoluminescent pellets of $CaSO_4$:Dy and Al_2O_3:C nanodot OSL detectors were utilized for the calibration of some clinical applicators. Both kinds of dosimeters were exposed to different dermatological applicators for the calibration procedure. The results obtained from both luminescent techniques were compared, and agreement was achieved.

Introduction
The thermoluminescent phenomenon is a luminescent dosimetric technique that has been used for many applications for several decades. Due to its advantages (Olko 2010), it presents applications in a large field of research, including individual, environmental and spatial dosimetry. Furthermore, the use of thermoluminescent dosimeters (TLDs) has been applied with success in the calibration and dosimetry of $^{90}Sr+^{90}Y$ sources used in brachytherapy, called clinical applicators (Oliveira and Caldas 2007; Holmes et al. 2009; Soares et al. 2009).

According to Holmes et al. (2009) and to international recommendations (IAEA 2002; ICRU 2004), clinical applicators should be calibrated to assure: that a quality control program has been applied, that the sources are used correctly and that the treatments of the patients are consistent. Furthermore, it is necessary that each source has a calibration certificate traceable to a primary standard whenever possible (Holmes et al. 2009).

TLDs are recommended for the calibration of beta radiation sources (Oliveira and Caldas 2007; Soares et al. 2009) due to their small dimensions and easy handling. $CaSO_4$:Dy + Teflon pellets were studied and presented efficiency in the detection of beta radiation of $^{90}Sr+^{90}Y$ sources (Campos and Lima 1987; Oliveira and Caldas 2007).

The optically stimulated luminescence (OSL) technique has been actively utilized as a dosimetry method for archaeological and geological dating. In the present days, the OSL technique has shown usefulness for personal dosimetry too (Botter-Jensen et al. 2003).

The OSL phenomenon is based on the luminescence emitted from an irradiated insulator or semiconductor as a result of light stimulation.

Since the development of this technique, its use is restricted because of the lack of a good luminescent material presenting high sensitivity to radiation and optical stimulation efficiency, a low atomic effective number and adequate fading characteristics (Akselrod and McKeever 1999). The Al_2O_3:C detector is a very good luminescent material for OSL dosimetry, because it presents excellent dosimetric characteristics. It was the first material introduced commercially for personal monitoring based on an OSL reader system from Landauer (McKeever et al. 2004).

The OSL technique has several advantages over the thermoluminescence (TL) technique: the readout method is optical, requiring no heating of the samples; the measurement is less destructive and potentially more sensitive than TL; and the response may be evaluated several times on the same sample (Akselrod and McKeever 1999; McKeever 2001; Akselrod 2000).

The objective of this work was to compare the absorbed dose rates obtained as result of the calibration of $^{90}Sr+^{90}Y$ dermatological applicators using two luminescence techniques: thermoluminescence and optically stimulated luminescence.

Materials and methods

Initially, the TL and OSL detectors were studied in relation to the response reproducibility, and the lower detection limits were determined. For the reproducibility study of the $CaSO_4$:Dy dosimeters, the pellets were exposed to a $^{90}Sr+^{90}Y$ standard source (1850 MBq, 1981), Buchler GmbH & Co., model BSS1, Germany, at 11 cm (dose of 1 Gy). For the reproducibility study of the OSL dosimeters, the samples were irradiated with a $^{90}Sr+^{90}Y$ standard source (460 MBq, 2004), Isotrak, model BSS2, Germany, at 30 cm (dose of 6 mGy).

In this work, a $^{90}Sr+^{90}Y$ clinical applicator (called NIST applicator), calibrated at the primary standard laboratory of the National Institute of Standards and Technology, USA, was utilized as reference system in the calibration of three $^{90}Sr+^{90}Y$ plane dermatological applicators. Two of them (A1 and A3) have calibration certificates from Amersham, and A2 applicator does not have any calibration certificate. The main characteristics of these applicators can be observed in Table 1.

For this work, $CaSO_4$:Dy thin pellets, with dimensions of 0.2 mm of thickness and 6.0 mm of diameter, were utilized. These pellets were mixed with Teflon, and they were produced at the Dosimetric Materials Laboratory of IPEN.

The TLDs were exposed to the clinical applicators at null distance between source and dosimeter. The TL measurements were taken immediately after each irradiation, using a Harshaw TLD Reader, model 3500, with a linear heating rate of 10ºC.s^{-1} and a constant flux of N_2 of 5.0 l.min^{-1}. The light emission was integrated in the temperature interval between 180ºC and 350ºC. After the irradiations, the pellets were thermally treated at 300ºC during 3 hours, for reutilization.

Table 1. Characteristics of the $^{90}Sr+^{90}Y$ NIST, A1, A2 and A3 dermatological applicators.

$^{90}Sr+^{90}Y$ applicator	Manufacturer and model	Nominal absorbed dose rate (Gy/s)	Calibrated by	Calibration date
NIST	Atlantic Research Corporation / B-1 S/N 233	0.40 ± 0.02	NIST	28.01.2003
A1	Amersham / SIQ 18	0.056 ± 0.011	Amersham	08.11.1968
A2	No information	——	——	——
A3	Amersham / SIQ 21	0.053*	Amersham	17.09.1986

*No uncertainty provided in its calibration certificate

Fig. 1. $^{90}Sr+^{90}Y$ NIST applicator (plane/dermatological source) used in this work as reference in the calibration of the other applicators.

The Landauer Al_2O_3:C nanodot dosimeter is a layer of Al_2O_3:C sandwiched between two layers of polyester for a total of 0.3 mm of thickness and 7.0 mm of diameter. These detectors were exposed to the three applicators under the same conditions that the TLDs. The OSL dosimeters were optically treated at 26x10^3 lux during one hour prior each utilization. A Delta OHM radiometer, model D09721, LUX LP 9021PHOT sensor, was utilized to determine the light level. The Landauer microStar reader and software were utilized to evaluate the OSL detector response. The measurements were taken immediately after irradiation.

Results

Initially, the TL and OSL samples were studied in relation to their response reproducibility and lower detection limit. Dose-response curves were obtained for both materials using the NIST applicator and, finally, the absorbed dose rates were obtained for the A1, A2 and A3 applicators.

Reproducibility study of the dosimeters

The reproducibility of the TL response of $CaSO_4$:Dy samples was obtained after five series of irradiations (1 Gy), measurements and thermal treatments. The maximum perceptual deviation obtained was equal to 7.1 %, and the associated uncertainty was 8.7 %.

The reproducibility of the OSL response, using ten Al_2O_3:C detectors, ten times irradiated (6 mGy), measured and optically treated, was already determined as 4.9 % (Pinto and Caldas 2009), and the associated uncertainty was 2.8 %.

Lower detection limit

The lower detection limit was obtained by the variability of the TL response of non-irradiated $CaSO_4$:Dy samples. The limit obtained for the TLDs pellets was 56 µGy, presenting the same order of magnitude of the results obtained for this material ($CaSO_4$:Dy) by Campos and Lima (1987).

In the case of OSL detectors, the lower detection limit was determined graphically. The Al_2O_3:C were irradiated with the source of $^{90}Sr+^{90}Y$ of the BSS2 beta secondary standard system in a dose interval from 0.05 to 10 mGy. A dose-response curve was obtained and presented linear behaviour between 0.2 and 10 Gy.

The lower detection limit was obtained extrapolating the dose-response curve of 0.05 to $1x10^{-2}$ mGy (Figure 2), to compare it with the detection limit presented by the manufacturer (Landauer), of 0.2 mGy to beta radiation. In this work, the value obtained was 0.13 mGy.

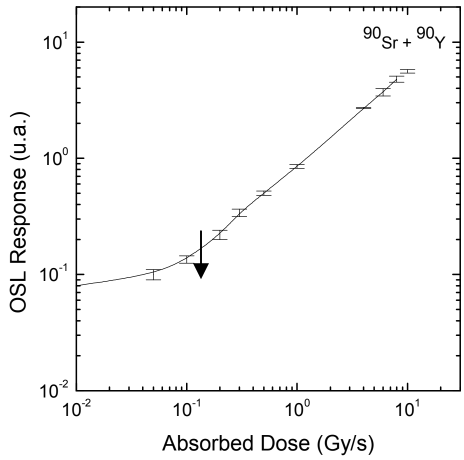

Fig. 2. Dose-response curve of the Al_2O_3:C detectors to $^{90}Sr+^{90}Y$ (BSS2), using the OSL technique, and utilized in the determination of the lower detection limit of the OSL response of Al_2O_3:C nanodot samples.

Dose-response curves of NIST applicator

The TL response of the $CaSO_4$:Dy samples was obtained in relation to absorbed dose in the air, irradiating the pellets with the NIST applicator in a dose interval from 5 to 20 Gy, at the null distance between dosimeter and source. The dose-response curve obtained can be observed in Figure 3. The TL pellets presented the expected response in the tested dose interval, with linear behaviour up to 10 Gy.

A dose-response curve was obtained for the OSL detectors (Figure 4) using the NIST applicator, under the same conditions of TLDs, but in a dose interval from 3 to 10 Gy. A linear behaviour can be observed.

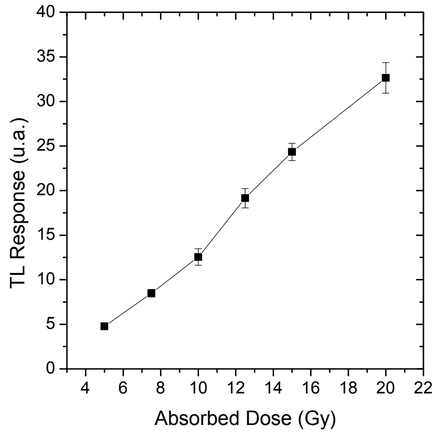

Fig. 3. Dose-response curve of the $CaSO_4$:Dy thin pellets, irradiated with the reference NIST applicator ($^{90}Sr+^{90}Y$).

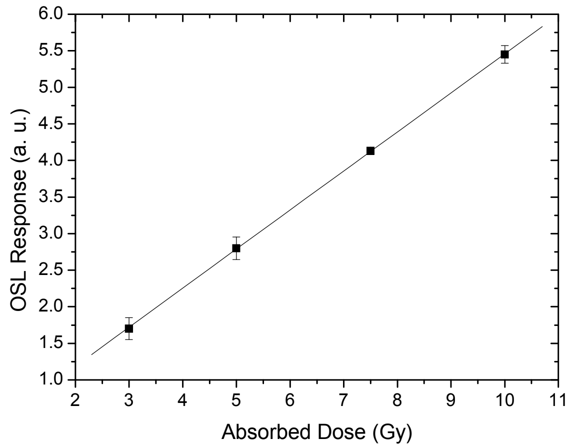

Fig. 4. Dose-response curve obtained of the Al_2O_3:C detectors, irradiated with the reference NIST applicator ($^{90}Sr+^{90}Y$).

Calibration of the $^{90}Sr+^{90}Y$ dermatological applicators

Using the dose-response curves of TL and OSL detectors, it was possible to calibrate the A1, A2 and A3 clinical applicators.

The $CaSO_4$:Dy thin dosimeters and the Al_2O_3:C detectors were exposed to each one of the clinical applicators during the irradiation time interval of 330, 330 and 300s, respectively for the A1, A2 and A3 applicators. These time intervals were chosen according to the activity of each source. The null distance between pellets and clinical applicators was always used during these irradiations.

The absorbed dose rates were determined and the results can be observed in Table 2.

Table 2. Absorbed dose rates obtained in this work, using the TL and OSL techniques, of the clinical applicators, in comparison with those provided in their calibration certificates.

$^{90}Sr+^{90}Y$ applicator	Absorbed dose rate (Gy/s)		
	Certificate	TL technique	OSL technique
A1	0.0213 ± 0.0043	0.0254 ± 0.0051	0.0275 ± 0.0055
A2	No certificate	0.0294 ± 0.0060	0.0336 ± 0.0067
A3	0.0299 ± 0.0060	0.0342 ± 0.0068	0.0386 ± 0.0077

The maximum relative uncertainty for the TL measurements was equal to 9.6 % for the A1 applicator. In the case of the OSL measurements, the maximum relative uncertainty was equal to 1.7 % for the A2 applicator.

Discussion

The same procedure was adopted for the calibration of $^{90}Sr+^{90}Y$ clinical applicators using two techniques: TL and OSL. The usefulness of the TL detectors in the calibration of these applicators was verified in a previous work (Antonio and Caldas 2009), but the TL measurements in this work were taken in another TL reader (Harshaw Nuclear System, model 2000 A/B). These results, using different models of TL readers, are very similar (maximum difference of 16 % for the A3 applicator) demonstrating the efficiency of this kind of calibration method.

Conclusions

The reproducibility study of both TL and OSL detectors showed satisfactory results.

The absorbed dose rates obtained using TL dosimeters of $CaSO_4$:Dy and OSL detectors of Al_2O_3:C, in the calibration of the $^{90}Sr+^{90}Y$ clinical applicators, in comparison with the values provided in their calibration certificates, presented a maximum difference of 22 % (for the OSL technique and A1 and A3 applicators) between techniques and respective certificates, and of -12 % (A2 applicator) also between the two techniques.

It can be concluded that the OSL technique is as effective as the TL technique for the calibration of dermatological clinical applicators used in brachytherapy procedures.

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Air kerma standard and measurement comparison for HDR ^{192}Ir brachytherapy sources calibrations

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Abstract

This paper describes the Institute of Nuclear Energy Research (INER, Taiwan) established the reference air kerma rate (RAKR) calibration standard for measurement of the HDR ^{192}Ir brachytherapy source strength. An bilateral comparison has been made in the RAKR standards for HDR ^{192}Ir brachytherapy sources at the INER and PTB (Germany) and the measurement difference was within the overall standard uncertainty and showed a good agreement of the two calibration standard systems established at the INER and the PTB. The accuracy in RAKR determination has increased the use of robust well-type chambers designed for brachytherapy. Besides, INER also designed a set of portable measuring device and worked with 20 domestic hospitals to organize an on-site measurement comparison program exploring the status of HDR ^{192}Ir brachytherapy source strength determination in Taiwan. A comparison result was present the ratios of RAKR with vendor values, as determined by INER and hospitals from the program. The ratios fall in all cases within the ±3% guaranteed by the vendors for a coverage factor of k=2 or at 95% confidence level.

Introduction

The isotope ^{192}Ir is frequently used in high dose rate (HDR) brachytherapy sources. The source strength is to be determined in terms of the reference air kerma rate (RAKR) (ICRU 1997, IAEA 2002) before such a source can be used in clinical practice. The RAKR is the air kerma rate to air, in air, at a reference distance of $d_{ref} = 1$ m, corrected for attenuation and scattering and refers to the quantity as determined along the perpendicular bisector of the source (commonly shaped as a cylinder). Compared to techniques for free-in-air RAKR measurement using small ionization chambers, the use of well-type chambers at hospitals is highly recommended. The objective of this research is to fabricate spherical ionization chambers as the INER's calibration standard for RAKR measurement of the HDR ^{192}Ir brachytherapy source strength to provide dosimetry traceability and well-type chambers calibration services in Taiwan. To verify the accuracy of the HDR ^{192}Ir RAKR standard, a HDR ^{192}Ir brachytherapy source calibration comparison has been made at the National Radiation Standard Laboratory of INER and PTB (Germany) in 2009.

In addition, ^{192}Ir sources have a half-life of approximately 74 days and hospitals renew their HDR sources with intervals of around 3 months. Each source comes with a certificate on which the vendor states its strength. An independent check of this value performed by the hospital is widely recommended as an important part of a quality assurance program (Baltas et al. 1999). To investigate the status of HDR ^{192}Ir brachytherapy source strength determination in the afterloading units in Taiwan, measurements at the hospitals are carried out by the authors using calibrated equipment from the National Radiation Standard Laboratory of INER. The source strengths are compared to values specified by vendors and those measured by the physicists at the hospitals using their routines and the same source.

Material and methods

The INER uses a Nucletron microSelectron HDR Classic brachytherapy unit fitted with the 'Classic' source, part number 096.001, manufactured by Mallinckrodt Medical B V (The Netherlands). The average photon energy for HDR ^{192}Ir brachytherapy source is close to 0.4 MeV (Douysset et al. 2008). Fig. 1 shows a schematic diagram of the HDR ^{192}Ir brachytherapy source simulated in this work. The enclosure of the radioactive material consists of a cylindrical stainless steel AISI 316L capsule (length: 5.0 mm, radial thickness: 250 μm) which is sealed by laser welding. The ^{192}Ir is contained in the capsule as a metallic ^{192}Ir cylinder (length: 3.5 mm, diameter: 0.6 mm). The stainless steel capsule is welded to a metal plug and a 1500 mm long flexible stainless steel AISI 316 cable. The other end of the capsule is welded to a steel pin (tail). The identification of the source is engraved on the long side of the tail. The nominal initial activity of the source is between 370 GBq and 550 GBq. The photon intensity and spectrum for the source at the RAKR determination point was calculated using MCNP code of version 5 (X-5 Monte Carlo team 2003). In the calculation the geometry model of the HDR ^{192}Ir source was constructed and simulated in detail. The energy cut-off settings for the photon and electron transport were both 0.001 MeV. Photons were sampled uniformly in the core and were scored at the side of the HDR ^{192}Ir source

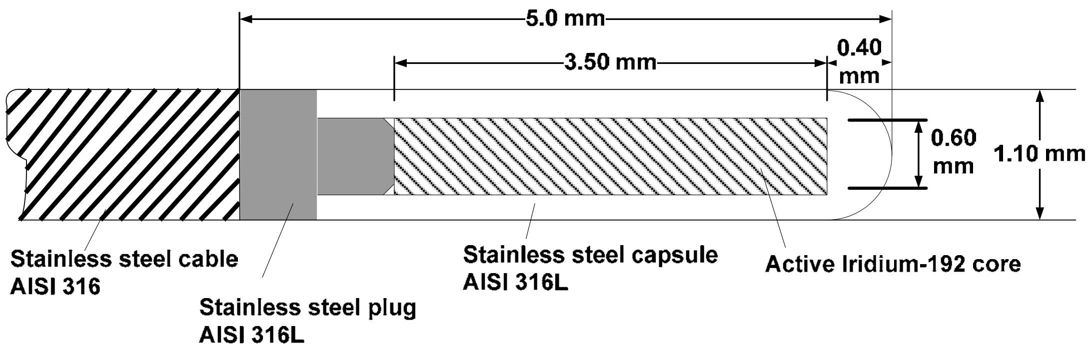

Fig. 1. Geometric model of the Nucletron mircoSelectron HDR ^{192}Ir Classic source.

INER designed and fabricated two spherical polarity ionization chambers which were numbered respectively A5SP4 SN: 001 and SN: 002 as the calibration standard for HDR ^{192}Ir brachytherapy source RAKR measurement. The contributions to the air kerma calibration factor for the different photon energies of the ^{192}Ir spectrum have to be weighted over the energy response of the ionization chambers. The method used at INER is to determine the calibration factors for the ionization chamber at ten different photon energies (ISO narrow-spectrum series, ^{137}Cs and ^{60}Co) ranging from 48 to 1250 keV. The energy response curve is plotted as a function of the photon energy. The correction factors associated with the standard chamber were determined by measurements and calculations. The RAKR can be calculated generally using the following expression:

$$\dot{K} = Q \times k_i \times k_{T,P} \times k_s \times k_a \times k_e \qquad (1)$$

where

$\dot{K}$ is the RAKR at the chosen reference time,

Q is the displayed ionization current (nA) on the electrometer corrected for leakage,

k_i denotes the average interpolated calibration factor of the chambers for the ^{192}Ir spectrum

$k_{T,P}$ is the temperature and pressure correction

$k_a\, k_s$ is the combined air attenuation and scatter correction which corrects the measured current for air attenuation and scatter between the source and the point of measurement

k_e is the calibration factor for the electrometer, thermometer and pressure meter

Besides, we decided to compare the calibration for HDR ^{192}Ir brachytherapy sources using the recommended well-type chamber calibration procedure. INER designed a set of portable measuring device and worked with 20 domestic hospitals using 20 afterloading units to organize an on-site measurement comparison program on source strength determination for HDR ^{192}Ir brachytherapy. Measurements at the hospitals are carried out by the INER using calibrated equipment. The calibrated well-type chamber, PTW 33004 was available. The electrometer, thermometer and pressure meter were calibrated against reference equipment at INER. The source strengths are compared to values specified by venders and those measured by the physicists at the hospitals using their routines and the same source. All participants employ their own well-type chambers traceable to INER and fulfill recommendations for use in brachytherapy.

Results and discussion

The photon spectrum at the RAKR measurement point was estimated by assuming the photon-emission probabilities for ^{192}Ir decay and calculating the attenuated spectrum reaching the measurement point. The calculation took into account the attenuation along all photon paths through the various materials by integrating over all source points in the cylindrical core. Fig. 2 shows the ^{192}Ir source photon spectrum calculated in this research, with which the spectrum for the similar field from the independent Monte Carlo calculations of Borg and Rogers (1999) shows a good agreement.

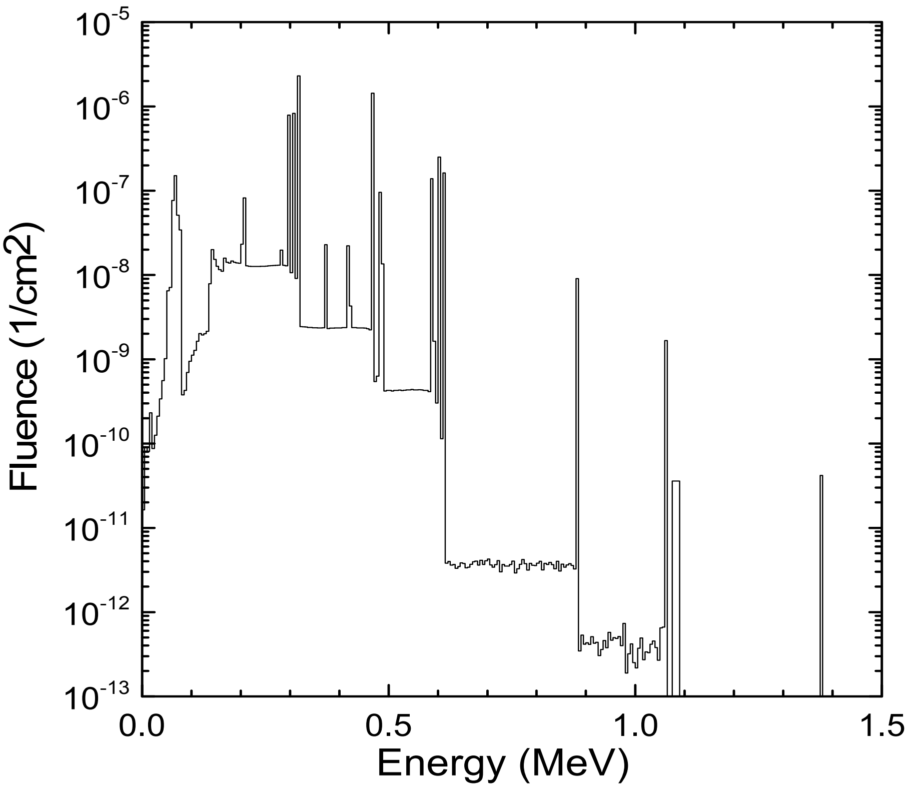

Fig. 2. Fluence spectrum for the Nucletron mircoSelectron HDR ^{192}Ir Classic source at the RAKR measurement point.

With ISO narrow spectrum series x-rays, ^{137}Cs and ^{60}Co beam qualities, we calibrated the INER-made spherical chambers (A5SP4 SN:001 and SN:002) and got the calibration curves of the chambers against the photon energy as Fig. 3. According to Fig. 3, the chamber calibration factors for various energies of the ^{192}Ir spectrum were found. By weighing the calibration factors with the relative intensity of each spectrum line, the calibration factors of the two spherical chambers against ^{192}Ir were obtained as 9.99×10^5 and 9.95×10^5 Gy C^{-1}, respectively.

From Eq. (1), it is seen that there were three correction factors needed for the RAKR calibration of the HDR ^{192}Ir brachytherapy source including the temperature/pressure correction factor ($k_{T,P}$), the room scattering correction factor (k_s) and the air attenuation correction factor (k_a) at 1 m distance from the source. They are described as the following:

- When the environmental conditions are different from reference conditions (22 □ and 101.325 kPa) during calibrations of chambers or sources, it is necessary to have temperature and pressure corrections.
- The shadow-shield technique (Bueermann et al. 1994) was used to evaluate the impact of scatter toward chamber measurement signal. It was performed that the

positions of source, shadow shield and the chamber were kept at least 1 m above the floor as well as 1 m away from the wall. The lead shields of 7 cm in diameter and 10 cm thick were placed between the chambers and the source to stop the primary radiation from the source getting to the chambers. By this shadow-shield technique, the average value of the dose fraction caused by scatter radiation of the two chambers was 6.19%. Therefore the scatter correction factor, k_s=1-0.0619 = 0.9381.

- Through the mass attenuation coefficients of various photon energies in the ^{192}Ir spectrum weighed with the relative intensity of each spectrum line and air density, the air attenuation at 1 m was expressed as $e^{-\mu x}$=0.987, so the correction factor, k_a = 1.0136.

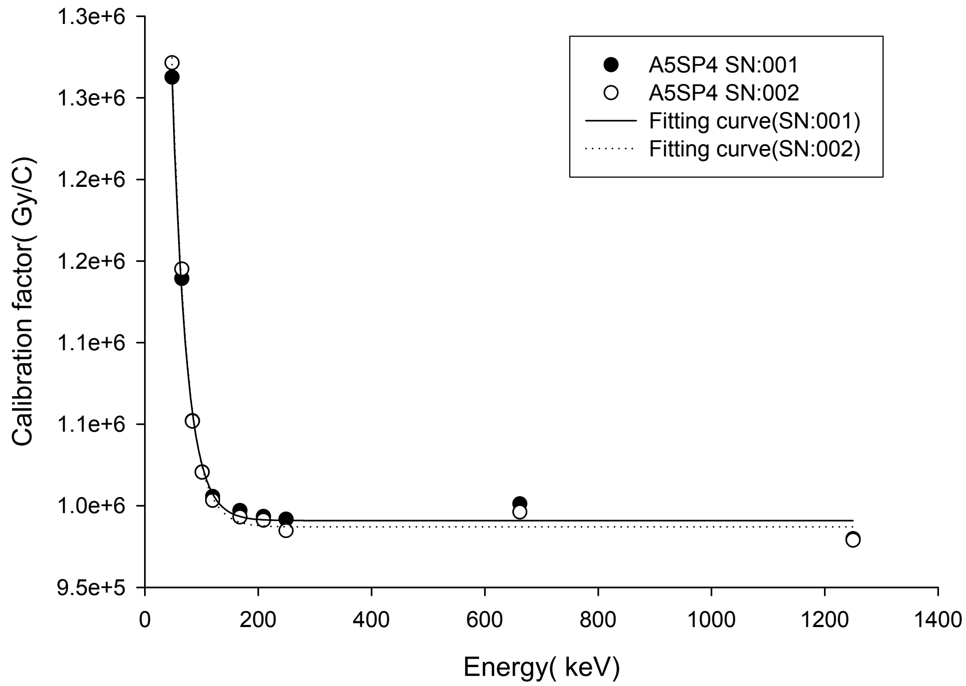

Fig. 3. The calibration factors of INER-made spherical chambers as a function of the mean energy of the radiation qualities.

INER adopted the method recommended by the ISO/IEC Guide 98-3 (2008) to calculate the RAKR measurement uncertainty and the result was 2.28% (k=2). INER used the ^{192}Ir source which had been calibrated by PTB (expanded uncertainty 2.5%, k=2) to make comparisons on RAKR measurements and learned that the difference of both sides was about 1.26% which was less than RAKR measurement combined uncertainty. It was verified that the measurement results of the ^{192}Ir RAKR standard system established by INER agreed with that of PTB.

The uncertainty of the source strength as stated by the vendors is ±3% at 95% or k=2. Fig. 4 presents ratios of RAKR with vendor values, as determined by INER and hospitals. The type of each source is specified in Table 1. The ratios fall in all cases within the ±3% guaranteed by the vendors at k=2, irrespective of the source type for each hospital included here. With the analyses of the measurement comparison program results, we could evaluate the variations in this ratio can therefore serve as an early indicator on problems in some part of system; either with the equipment or the measurement technique at the hospital or the vendor laboratory.

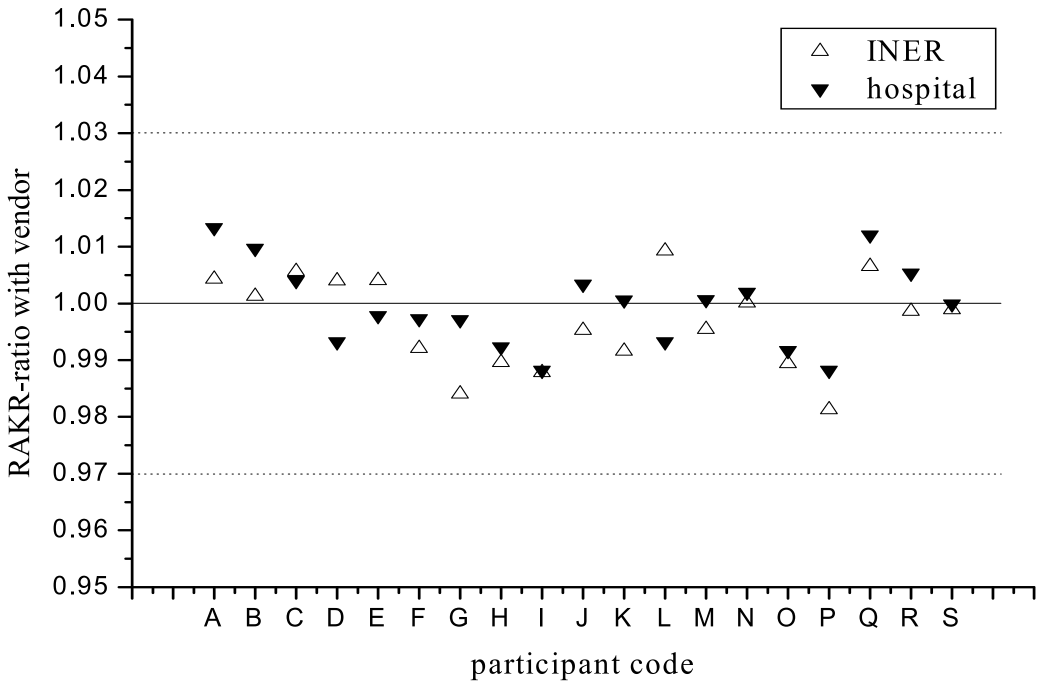

Fig. 4. Ratios of INER- and hospital-determined RAKR values, respectively, to the correcponding values on vendor certificates.

Conclusions

The technique for the dosimetry calibration laboratories to determine RAKR of HDR ^{192}Ir brachytherapy sources used in this study is simple and effective. The comparison of RAKR standards for HDR ^{192}Ir source between INER and PTB showed that the two national metrology institutes (NMIs) were in good agreement. The results indicated that the INER could provide accurate brachytherapy dosimetry traceability and well-type chambers calibration services Besides, a draft guide for quality assurance to institutions performing the HDR ^{192}Ir sources calibration was suggested based on the sources strength on-site measurement program results. If an institution determines ratios of RAKR with vendor values and differed more than 3%, that institution should review the data for calculations and measurements. With the analyses of the on-site measurement data, we could evaluate the feasibility to establish verification and audit technologies under the support of the competent authority to recognize the accuracy of brachytherapy source strength in hospitals and secure the treatment quality of patients.

Table 1. Specification of source types for the measured sources, participant code as to correspond with the results on RAKR ratios in Fig. 4.

Participant code	Source type
A	Nucletron mHDR-Classic
B	Nucletron mHDR-Classic
C	Nucletron mHDR-Classic
D	Nucletron mHDR-Classic
E	Nucletron mHDR-Classic
F	Nucletron mHDR-Classic
G	Nucletron mHDR-V2
H	Nucletron mHDR-V2
I	VariSource HDR-VS2000
J	Nucletron mHDR-Classic
K	VariSource HDR-VS2000
L	GammaMed HDR-12i
M	GammaMed HDR-12i
N	Nucletron mHDR-Classic
O	Nucletron mHDR-Classic
P	Nucletron mHDR-Classic
Q	Nucletron mHDR-Classic
R	Nucletron mHDR-Classic
S	Nucletron mHDR-Classic
A	Nucletron mHDR-Classic

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Instadose – Web based system for reading and monitoring dose through Internet access

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Abstract

Instadose™ is a small, rugged dosimeter based on patented direct ion storage technology and is NVLAP accredited. This breakthrough technology provides radiation workers with a more precise measurement of radiation dose and includes accurate long-term exposure tracking. A built-in memory chip stores each user's identity via an embedded unique serial code that is assigned to the user. Instadose allows users to have the flexibility to view their radiation dose at any time from any computer with internet access. Readings via a PC are enabled by a USB compatible detector. When a user wishes to obtain a reading they simply log-in to their account, plug-in instadose to a USB port and on the homepage click-on the computer image with the message "Plug in your instadose device now and click here to get an instant reading". The accumulated dose stored on instadose is processed through a proprietary algorithm. This fully automated transfer of data minimizes the chance of human error and misidentification. Once complete a graphical representation of the current dose will load on the screen. Users can also view their cumulative dose level by clicking "View Cumulative Dose." A variety of reports are available for download through AMP (Account Management Program)." The reports include:

– Radiation Exposure Summary Report
– History Detail Report
– Who Has Not Read Their Device.

Radiation qualities for patient dosimetry in x-ray imaging: from calibrations to clinical use

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Abstract

The response of an ideal dosemeter does not have energy dependence and same calibration coefficient can be used for different x-ray spectra. In calibration practice, the energy dependence should be studied by using different radiation qualities. Standard RQR and RQR-M radiation qualities (IEC 61267: 2005), intended to represent the x-ray beam incident on the patient in radiographic, fluoroscopic, dental and mammography examinations, are generally used for calibration in laboratories. They differ from the ones used in patient imaging and they do not cover the total HVL range of clinical use. The calibration coefficients need to be converted to the actual clinical radiation qualities by interpolation using appropriate specifiers of the energy spectrum. For reference class ionization chambers the response depends rather smoothly on the radiation energy, and the half-value layer (HVL) is often sufficient for specifying the radiation quality. However, this may not be the case for dosemeters with a strong energy dependence. In this study, calibration coefficients of different dosemeter types were determined using the standard radiation qualities and several other radiation qualities that are typically used in medical x-ray imaging. The HVL was used as the specifier for the energy distribution of the x-ray beam. The accuracy of air kerma measurements at different radiation qualities was investigated, especially with respect to the interpolations between the radiation qualities used in calibration procedures in order to derive the calibration coefficients for the radiation qualities in clinical practice. KAP meters and semiconductor detectors are examples of dosemeters, which have a strong energy dependence. For accurate measurements, it is not sufficient to make the interpolation based on HVL alone. With KAP meters the difference between calibration coefficients for standard and clinical radiation qualities with same HVL may exceed 15%.

Ambient dose equivalent measurements at IMRT medical accelerator

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Abstract

A dose equivalent meter based on a recombination principle has been used for measurements of ambient dose equivalent in mixed (gamma + neutrons) radiation fields outside the irradiation fields of linear medical accelerator. The main detector was a recombination chamber, i.e. tissue-equivalent, high-pressure ionization chamber operating under conditions of local recombination of ions. The second ionization chamber served as a monitor of the beam intensity. The use of recombination chambers makes it possible to determine the total absorbed dose, which is proportional to the saturation current, and recombination index of radiation quality, Q_4, which can be used as an approximation of the radiation quality factor. Ambient dose equivalent H*(10) can be well approximated by the product of the ambient absorbed dose D*(10) and Q_4. The automatically controlled measuring cycle includes determination of the ionization current at four different polarizing voltages, which are sequentially applied to the recombination chamber electrodes. Then, the readings are normalized to the monitor chamber readings and Q_4 values and ambient dose equivalent are calculated taking into account the relationship between the initial recombination of ions and radiation quality factor. Tests were performed at the 15 MV Varian Clinac 2300C/D accelerator and at the accelerator configured for IMRT treatment. The measurements confirmed that the ambient dose equivalent of mixed radiation in clinical conditions could be determined with accuracy of about 20%.

Quality mammography essential in detecting breast cancer

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Abstract
Between 1991 and 2007 breast cancer incidence increased 3.4 times with the Romanian women, the trend continuing to grow. Knowing that mammography can detect early breast cancer and that this is the key to successful long-term control of diseases and good prognoses, the present study aims at evaluating the influence of quality control in order to detect small changes in breast tissue.

There have been assessed five mammography labs (A, B, C, D, E), of which only A had partially implemented the system of quality control. The measurements performed include the evaluation of X-ray units, image receptors, image processors, patient dose and image quality. Mean glandular doses were calculated the way the American College of Radiology does (1999).

The assessment pointed out the following differences between labs B, C, D, E, vs. A: (1) the kVp reproducibility was good but the accuracy differs by more than ±5% from the nominal kVp setting (81.2% in E vs. 1.2% in A); (2) the developer temperature was up to 10^0C under the value specified by the film manufacturer; (3) the automatic exposure control was not used; (4) the number of visible test objects on the phantom image was unacceptably small in the B, C D, E; (5) the radiation beams and the mean glandular doses are higher or lower than guidance levels recommended for a high quality mammogram ($p<0.001$).

Quality control does not remove the problems but it can detect them before the clinical results are seriously affected. Mammography is the most exacting radiological examination and that’s why it is paramount to apply the quality control in all the labs.

Introduction
In the breast imaging, mammography is the safest method for detecting and managing breast cancer being associated with a false negative rate of 10-20%, according to the patient’s age, and the only method that facilitates detection of micro-calcifications.

The main objectives of a quality control programme are to improve diagnostic accuracy without unnecessary radiation, and to minimise the costs (Faulkner 2004). Such a programme may identify the problems before patients’ health is affected. The smallest modification in equipment functioning or film processing can seriously affect the image quality and the breast dose.

Starting with 2002, although the Romanian legislation has included performance criteria to be fulfilled by mammography labs (Order no. 285 2002), there still is no programme of quality ensuring and quality control. However, there are 10 pilot centres that took part in a programme of disseminating quality control activities organized in 2005 with the help of the American International Health Alliance and coordinated by Romanian Society of Breast Imaging. Such educational programmes were carried out in other countries (Vassileva et al. 2005), too, and they had three main objectives: quality control promotion, operator training with simple instruments and procedures, and operator training in quality control data analysis. Each centre received a special kit at the end of the programme so that they could carry out the film processing tests, compression and image quality tests.

As breast cancer incidence grew 3.4 times in Romania between 1991 and 2007, the present study aims at evaluating the influence of quality control in order to detect small changes in breast tissue and also attempts at evaluating some mammography labs in order to see whether they are efficient or deficient, and whether they correspond to the standards of quality control.

Material and methods

Five mammography labs, in five county hospitals have been checked. Only one has partially implemented the system of quality control.

The infrastructure:

- Hospital A - Senograph DMR – quality control partially implemented;
- Hospital B - Senograph DMR – without quality control;
- Hospital C - Senograph DMR – without quality control;
- Hospital D - Senograph DMR – without quality control;
- Hospital E - Mammomat 3000 – without quality control.

The following system components have been checked: the tube voltage (the kVp accuracy and reproducibility, the beam quality assessment), the film processing, and the system properties (the image quality and the average glandular dose).

In order to measure the high voltage accuracy and reproducibility of the radiation beam, the multi-functional device for testing radiological systems of the RMI – 242 type was used, that has a flat ionization chamber, calibrated for mammography.

There have been checked the highest, lowest and most commonly used clinical kVp. The kVp delivered by the X-ray generator was compared with the measured values.

The kVp accuracy was determined by comparing the average measured values with the values of the preset nominal kVp.

The kVp reproducibility was determined by calculating the coefficient of variation for each kVp setting.

The image quality was checked using the standard phantom accredited for mammography, of the Nuclear Associates 18-220 type, which simulates a compressed breast of 4.2 cm thick consisting of 50 % glandular tissue and 50 % adipose tissue. The phantom has an acrylic support and a wax insertion which contains appropriate details (6 fibres, 5 groups of specks and 5 masses) ranging from visible to invisible on the mammography image. The American College of Radiology (ACR) minimum score is 4 visible fibers, 3 speck groups and 3 masses (Hendrick et al. 1999).

This procedure is the most important in mammography quality checking as it verifies the functioning of all the imagistic chain components except for those regarding breast positioning and patient movement.

The phantom was exposed to the parameters used for the standard breast and the film was processed. The evaluation was made by counting the number of visible test objects on the mammography image.

In order to check the density difference (DD), the phantom was attached an acrylic disc (1 cm in diameter and 4 mm thick) which was placed a little under the first two largest fibres so that it shouldn't hide details.

After exposing the phantom to the parameters used for the standard breast and after processing the film, the background optical density (OD) in the geometric centre of the phantom was measured and the density difference between the inner values of the acrylic disc and outer values, right or left, perpendicular on the anode – cathode axis was computed.

The beam quality assessment was performed by measuring the half–value layer (HVL). For this reason 0.1 mm aluminium alloy sheets of 99.9 % purity (alloy type 1145) were used.

The entrance surface dose in the phantom has also been measured with the RMI – 242 device, by positioning the detector in the X-ray field besides the phantom, centred 4 cm in from the chest wall edge of the image receptor and with the centre of the chamber level at the same level with the phantom surface (Hendrick et al. 1999).

The average glandular dose for the standard breast was estimated out of the phantom entrance surface measured dose to which the conversion factors used by Faulkner K in RC-4 entitled Mammography Screening, at the 11^{th} Radioprotection Congress, Spain, 2004 were applied:

$$AGD = K_{air} \cdot p \cdot g \cdot s$$

where: p converts incident air kerma for perspex phantom to that of the standard breast (*table 1*);

g converts incident air kerma for standard breast to mean glandular dose;

s spectral conversion factor (*table 2*)

Table 1. The conversion factors p and g (Faulkner 2004).

HVL (mmAl)	p	g (mGy/mGy)
0.25	1.12	0.155
0.30	1.10	0.183
0.35	1.10	0.208
0.40	1.09	0.232
0.45	1.09	0.258
0.50	1.09	0.285
0.55	1.07	0.311
0.60	1.06	0.339

Table 2. The spectral conversion factors (Faulkner 2004).

Spectre	s
Mo/Mo	1.000
Mo/Rh	1.017
Rh/Rh	1.061
Rh/Al	1.044
W/Rh	1.042

Results

The kVp accuracy and reproducibility were within the action limits except for Hospital E, where accuracy was above 5 % (*table 3*).

The measured value for kVp in hospital E is almost double compared to the recommended value (the mammography device hasn't been checked for three years).

Table 3. The kVp accuracy and reproducibility.

Hospital	mAs	kVp		Accuracy (%)	Reproducibility (CV) (%)
		Nominal setting (kVp)	Average measured (kVp)		
A	117	26	27.07	4.1	0.49
	200	28	28.9	3.2	0.45
	31	25	26.13	4.5	0.51
B	63	27	27.3	1.1	0.58
	80	29	29.35	1.2	0.9
C	45	25	25.3	1.52	0.44
	63	28	28.5	1.76	0.9
	145	30	30.5	1.67	0.63
D	33	25	25.4	3.2	0.42
	117	27	27.76	2.81	0.41
	140	29	30.18	4.1	0.76
E	32	25	25.68	2.72	0.4
	18	28	39.1	39.64	1.5
	22	31	56.17	81.2	1.6
	16	26	33.3	28.1	1.8
			Action limit *:	> ±5%	> 2%

* Mammography Quality Control, Radiologist's Manual, Radiologic Technologist's Manual, Medical Physicist's Manual (Hendrick et al. 1999)

Due to these high values of tension, the AEC cannot be used in Hospital E. The technician chooses the exposure parameters according to the breast size and to what he feels while touching it, and sets very small vlues for the mAs product no matter what the breast size is (*table 3*).

The measured HVL was higher than the superior recommended limit [kVp/100 + 0.03 < HVL < kVp/100 + 0.12 for Mo/Mo or 0.19 for Mo/Rh in units of mm aluminum, with the compression paddle (Hendrick et al. 1999)], in all hospitals except for Hospital A (*table 4*).

Table 4. The half-value layer (HVL).

Hospital	Parameters				HVL (mmAl)	Recommended limits* (mmAl)	
	kVp	Target	Filter	mAs		HVL_{min}	HVL_{max}
A	26	Mo	Mo	117	0.31±0.02	0.29	0.38
B	27	Mo	Mo	63	0.4±0.02	0.30	0.37
C	28	Mo	Mo	63	0.45±0.02	0.31	0.38
D	27	Mo	Mo	117	0.41±0.02	0.30	0.37
E	28 (39.1)	Mo	Rh	18	0.6±0.03	0.31 (0,42)	0.47 (0,58)

* Mammography Quality Control, Radiologist's Manual, Radiologic Technologist's Manual, Medical Physicist's Manual (Hendrick et al. 1999)

In all five hospitals we measured the optical density of the glandular tissue area on the mammograms carried out that day. Except for Hospital A, where the optical density was 1.6 OD, in all the other hospitals the optical density was either lower than 1 OD (in B, C, E) or higher than 2 OD in Hospital D.

The first noticed thing on the film imaging the phantom was the poor quality of the image(especially the Mediphot type).

Figure 1 presents an ideal image of the test objects of the standard phantom and also one of the images we obtained in one of the hospitals.

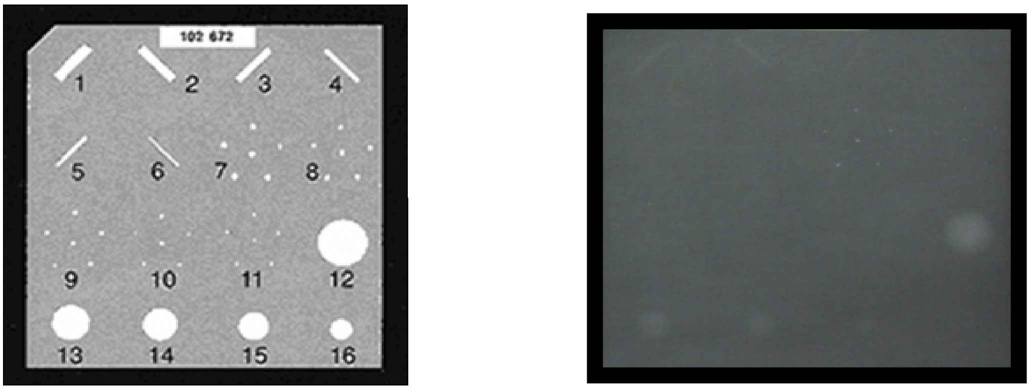

Fig. 1. The ideal image of ACR accreditation phantom (Maalej 2005) and the image we obtained at one of the hospitals.

All the labs were penalized because the films presented artefacts under the shape of fibres and specks so, C and D got the score of zero for the speck groups after artefact deduction.

The number of visible test objects on the phantom image was unacceptable in C, D, and E (*see table 5*).

Table 5. The image quality evaluation.

Hospital	Film	Density Diference (OD)	Background Optical Density (OD)	Number of visible objects *		
				Fibres	Speck groups	Masses
A	Kodak	0.42±0.02	1.66±0.09	5	4	5
B	Foma	0.35±0.05	1.4±0.1	4	3	3.5
C	Mediphot	0.30±0.04	1.23±0.05	4	0	3
D	Mediphot	0.51±0.03	2.3±0.1	4	0	3.5
E	Foma	0.29±0.02	1.06±0.08	4	2	1
Action limit *:		**0.4±0.05**	**<1.4±0.2 >2±0.2**	**4**	**3**	**3**

* Mammography Quality Control, Radiologist's Manual, Radiologic Technologist's Manual, Medical Physicist's Manual (Hendrick et al. 1999)

We can also notice in *table 5* that the background optical density was lower than the recommended values in C and E, and higher in D affecting thus the image quality.

The values obtained for the difference of density were considered unacceptable in C, D, and E, being either too low or too high in comparison to the recommended values.

The conversion factors recommended by ACR could not be used in order to estimate the mean glandular dose for the standard breast because, for the Mo/Mo target-filter combination, at such HVL values, they simply do not exist (Hendrick et al. 1999).

The entrance surface air kerma (ESAK) varied on a broad range, and differed from hospital to hospital and even for the same type of device. The AGD estimated for the standard breast were comparable to the reference levels only in the lab which had implemented the system of quality control (*figure 2*).

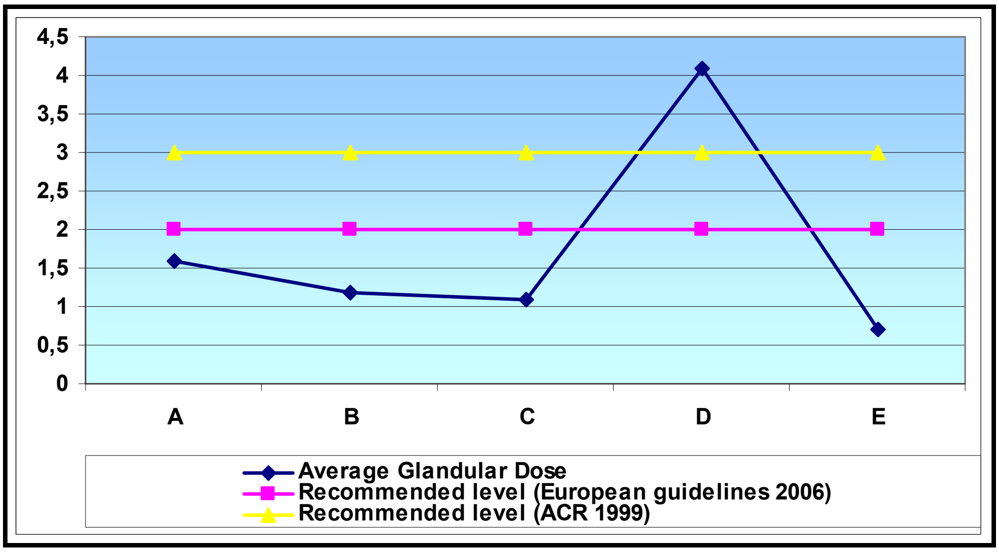

Fig. 2. The average glandular dose for the satndard breast at the five hospitals compared to the recommended levels.

Table 6 shows that the average glandular dose for the standard breast is obviously under the reference levels in Hospitals B, C and E, and over in D (p-value<0.001).

Table 5. The average glandular dose for standard breast.

Hospital	HVL (mmAl)	ESAK (mGy)	Conversion Factors			AGD (mGy)
			p	g	s	
A	0.31±0.02	7.9 ± 0.84	1.10	0.183	1	1.59±0.17
B	0.4±0.02	4.71 ± 0.67	1.09	0.232	1	1.19±0.14
C	0.45±0.02	3.88 ± 0.5	1.09	0.258	1	1.09±0.14
D	0.41±0.02	16.2 ± 1.58	1.09	0.232	1	4.1±0.4
E	0.68±0.03	1.94 ± 0.27	1.06	0.339	1.017	0.71±0.1
					Recommended level:	2*(3**)

* European guidelines for quality assurance in breast screening and diagnosis (Perry 2006)

** Mammography Quality Control, Radiologist's Manual, Radiologic Technologist's Manual, Medical Physicist's Manual (Hendrick et al. 1999)

Discussion

Except for hospital A, the technologists were not familiar with the quality control procedures, with the recommended frequency for performing any of the routine quality control procedures and the recommended maximum dose for a mammogram. The activity in these labs is reduced to performing the mammography and its interpretation by the radiologist.

It is well known that a mammogram with an acceptable contrast can be obtained only using an adequate X-ray beam, a firm compression, the grid when necessary, and a correct processing. From this point of view, the results of our research emphasised the presence of some major problems in B, C, D, and E laboratories.

In three out of five hospitals (B, C, and E) they don't use the Automatic Optimization Parameters (AOP), the exposure parameters being set out approximately according to the breast size or to what has been noticed while palpating. Radiologists consider that the image obtained using AOP is worse than their own eyes and experience, a possible fact as long as they have no control over the equipments which are not set by specialists when the film producer changes and they increase the contrast as the developing solutions grow older.

In all five hospitals they use one single type of high contrast film, irrespective of the breast size or its density. The films used are produced by some firms while the developing solutions by other firms due to the deficient acquisition system and public auction, which lead to acquiring the cheapest product.

The measured background optical density was lower or higher than the recommended values which also influences the image quality, as the films used in film-screen mammography lose their contrast at too low or too high optical densities. The glandular tissue, where the cancer starts from, generates the lowest optical density on the film. If the optical density is too low or too high, tracing the low contrast injuries in the glandular tissue is compromised. In order to guarantee the exposure of glandular tissues to sufficiently high optical densities, adjusting the automatic exposure control should have obtained optical densities for the phantom between 1.4 and 2.0 OD, which happened only in hospital A.

The contrast was also affected by the very low temperatures of the developing substances compared to those recommended by the producers. The developer temperature was smaller in B, C, and E, between 1.5 and 10 ^{0}C when the recommended action limits by the ACR are ±0.3 ^{0}C of the value specified by the film manufacturer. In fact, no hospital presented a technical checking report for the developing devices. What's more, hospital E processes the films together with the other diagnose films.

All these shortcomings that led to a decrease of the contrast are probably balanced, partially, by the very high compression (almost double in the three mentioned hospitals).

All the hospitals use only grids though it is known that using them increases the in-taken dose, and it is recommended not to use them for small breasts (the operators don't know how to remove the grid or if this is possible).

Except for hospital E, all the other hospitals use very rarely or not at all rhodium filters, recommended for very dense breasts.

Radiologists and mammography technologists should be familiar with the numerous artefacts that may create pseudo lesions or mask true abnormalities (Hogge et al. 1999).

The presence of artefacts was attributed to the poor film quality, to the dust, and/or to the processing way (roller, developer was mixed with the fixer, concentration of the developer, chemistry replenishment).

Quality image assessment underlined a clear difference between lab A and the other labs under scrutiny. The number of visible test objects on the phantom image was unacceptably small in the other hospitals, a fact that makes us consider that the benefit of the patient exposed to radiations in these labs was minimum, while the risk of not detecting the illness maximum.

Though the measured HVL were higher than the recommended superior limits, this fact does not infringe any standard, but a high HVL leads to a contrast decrease (Order no. 285 2002, Hendrick et al. 1999).

The estimated glandular average doses for the standard breast were either too low or too high, and it's a known fact that both high and low doses lead to missing minute details, diminish the early diagnose possibilities of cancer and increase the risk of inducing breast cancer through repeated irradiation (Law et al. 2002).

The maintenance firms that should ensure the service also raise problems, as they neither have specialists nor do they wish to sign contracts with hospitals, because these have only one mammography device which renders displacement unprofitable (in A and B hospitals more auctions were necessary until a contract could be concluded).

The technical checking is the only one at this particular moment which can bring information about the way the equipment works.

Conclusions

To sum up, the results indicated that the level of equipment performance is correlated with the image quality, which image quality estimated by radiologists is correlated with the phantom score. At the same time, the coincidence in clinical mammography reports is not correlated with equipment performance but seems to depend on the radiologists' experience.

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$CaSO_4$:Dy TL response for photons energies between 33 keV to 15 MeV

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Abstract

In the last decade radiotherapy has been the object of a technological revolution because of the new techniques developed such as Three Dimensional-Conformal Radiation Therapy and Intensity Modulated Radiation Therapy (IMRT). The clinical use of these techniques requires a complete knowledge of the imaging of the volumes to be treated and also the dosimetry of the clinical electron and photons beams that are used in those cases. Thermoluminescence (TL) or thermally stimulated luminescence has been actively developed in the past years due to its reliability, sensitivity and commercial availability and is currently in use with LiF: Mg, Ti (TLD-100) commercial dosimeters in the dosimetric quality assurance of the output of therapy machines which must be verified routinely. This work proposes the use of $CaSO_4$:Dy sintered discs as an alternative to LiF commercial dosimeters in the radiation therapy dosimetry of photons beams, studying the photon energy dependence response with energies ranging from 33 keV to 15 MeV using different phantoms such as water filled phantom, a PMMA and a solid water phantom. $CaSO_4$:Dy was chosen because is one of the most useful and sensitive thermoluminescent dosimetric material for radiation dosimetry, and in the form of sintered discs are very suitable for applications requiring a large number of measurements. The thermoluminescence dosimetry reader 3500 (Harshaw Model) was employed for the readout of the irradiated dosimeters. Results on reproducibility, radiation dose response and energy dependence show the possibility of their use as an alternative dosemeter of clinical photon beams.

Development and validation of a low dose simulation algorithm for computed tomography

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Abstract

Purpose: The purpose of this study was to develop and validate software for facilitating observer studies on the effect of radiation exposure on the diagnostic value of computed tomography (CT).

Methods: A low dose simulator was developed which adds Gaussian noise to the CT raw data with an automatically derived standard deviation according to the desired dose level. A validation was performed with two phantoms: a cylindrical test object and an anthropomorphic phantom. Images of both were acquired at different dose levels by changing the tube current of the acquisition (500 mA to 20 mA in five steps). Additionally, low dose simulations were performed from 500 mA downwards to 20 mA in the same steps. Noise was measured within the cylindrical test object and in the anthropomorphic phantom. This was performed in actual and simulated images. Finally, noise power spectra (NPS) of actual and simulated images were measured in water.

Results: The low dose simulator yielded similar image quality compared with actual low dose acquisitions. Mean difference in noise over all comparisons between actual and simulated images was 5.7 ± 4.6% for the cylindrical test object and 3.3 ± 2.6% for the anthropomorphic phantom. NPS measurements showed comparable shape and intensity.

Conclusion: The developed low dose simulator creates images that accurately represent the image quality of acquisitions at lower dose levels and is suitable for application in clinical studies. The major practical benefit of the algorithm is that it can be retrospectively applied to CT to optimize acquisition protocols in order to reduce the radiation dose.

Basic data for radiotherapy: Stopping of protons in liquid water

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Abstract

Protons have been used for years in external beam radiotherapy to treat patients with malignant tumours. Obviously, for successful treatments, the knowledge of the absorbed dose to the tumour and to the surrounding tissues is of paramount importance.

In proton radiotherapy, the dosimetry is usually carried out relative to the ^{60}Co beam, in terms of absorbed dose to water. As this dose is determined with the aid of an air-filled ionisation chamber, the ratio of air and water stopping powers for protons needs to be known with a high accuracy. However, the experimental data are scarce (or non-existent, depending on energy); these stopping powers are usually calculated with the refined Bethe-Bloch formula. Recent investigations in the low-energy region (below a few MeV) by other authors have shed doubts on the accuracy of the widely-used stopping power data for water. In addition to the proton therapy, an accurate knowledge of proton stopping is a prerequisite for a reliable neutron dosimetry. Dosimetry of proton and heavy ion exposure is also of importance in space missions and other branches of radiation dosimetry.

In this contribution we describe the measurement programme for determining the proton, and heavy ion, stopping in liquid water at energies below 30 MeV per nucleon; this is the region where the stopping power models typically have the largest uncertainties. Our method is based on a time-of-flight transmission technique using microchannel plate and semiconductor detectors. The major challenge with thin liquid targets in vacuum is to maintain the target as uniform as required by the desired experimental accuracy. Therefore, a special emphasis is placed on the construction and the characterisation of the liquid target. The methods for target thickness determination using mechanical and optical means are presented, together with preliminary results from the actual stopping power experiment.

Fast neutron detection by ALNOR type dosemeters with MCP-N thermoluminescence pellets

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Abstract
MCP-N thermoluminescence pellets with ALNOR type individual dosemeters were used to measure the fast neutrons component ot the ^{241}Am-Be source. The differences betwen TL signal of the pellets covered by the Al filter and without filter during irradiation were observed. The point of work was to find out the rison of observed differences in TL signal. Two possibilities, as activation of the Al filter and anisotrophy of the pellets were taked into consideration. The calculations that have been carried out show that the activities of radionuclides deponed in the Al filter after 600 s irradiation were to less to had influence to incerase the TL signal of the covered pellets. Olny the obtained dependence of TL signal from the pellets mass and high anisotrophy in consistance could explain the observed differences.

Introduction
Correct individual monitoring of the workers exposed to the neutrons is still one of unsolved problem in radiological protection. According to ICRP Report 103 (ICRP 1991) the neutrons at energy from 100 keV to 2 MeV have the highiest value of radiation weighting factor w_R, equal to 20 and equal to 10 for the neutrons at energy from 2 MeV to 20 MeV. In general the neutrons are regard as the most dangerouse type of radiation in the light of radiation protection purpose. The neutrons sources are mainly used in the calibration facielities and reaserch units. Today the most ofen exploit isotropic sources are Am-Be, Cf and Pu-Be. The neutrons emited by mentioned sources were mainly in the range of energy from 0,5 MeV to 10 MeV. The ideal dosemeter should to fulfil two main requirements, it should be able to detect the neutrons in wide range of energy from thermal to fast ones and it should have reasonably low neutron flux density detection threshold. In addition the dosemeter should to be unsesitive for gamma or should give opportunity to discrimination the gamma component. Courently for neutron individual monitoring a combinaton of properly designed albedo TLDs (Nikodemova et al. 1992) or bubble detectors (Apfel, Lo 1989) were mainly involved. In this paper some initial results for new individual neutron dosemeter concept were presented. The point of idea is to applicate in dosemeter two process. First relide on activation of the Al filter of the dosemeter and the secound relide on measurement of gamma component of the activation products using thermoluminescence material. Some

investigations to use the activation of Al for neutrons measurements by TLD were made before (Sants et al. 2006).

The following sections will present initial results which suggest possibility of realization of the new neutron dosemeter concept. In addition some consiterations about barriers and appear problems will be present.

Material

Filter and TL material

For all measurements presented in this work a MCP-N thermoluminescent material was used. MCP-N is a thermoluminescence material based on LiF with the enrichment 7,5% ^{6}Li and 92,5% ^{7}Li, doped with Mg, Cu and P, the material was in form of pellets of 1mm thickness and 4,5mm in diameter. Read out of the pellets was carried out with a RADOS RE-1 reader. It will be show that the pellets are characterised by high anisotropy of mass and consistence. During irradiation pellets were placed in ALNOR type dosemeters which had three 1mm thickness and 5mm in diameter Al filters and one possition without any cover. Each dosemeters were equipped in two pellets, one covered by filter and one in bared position.

Irradiation facielity

Irradiation was carried out in Central Laboratory for Radiological Protection (CLOR) using ^{241}Am/Be calibration source at activity 185 GBq. According to source certificate the neutron emmision of the source amount to 1,1 x 10^{7} n/s. Irradiations were made in four different distances from the source 0,8m; 1m; 1,2m and 1,4m.

Method

Basically the work was divided for two steps: calculation and measurement. Calculations cover a counts of the neutrons flux densities at dependence of distance from the source and counts of the activities of the activation products according to equiation (1), whereas activity of two isotopes ^{27}Mg and ^{24}Na as a results of the nuclear reaction respectively (n,p) and (n,α) were taked into consideration.

$$A = \int_{E=2}^{E=11} S(E)\sigma(E)\phi N(1-e^{-\lambda t})dE \qquad (1)$$

where *S(E)* is a percentage part in the spectrum of the neutrons at energy *E*, *σ(E)* is a cross section of the neutron at energy *E*, *φ* is a neutrons flux density, *N* is a nuber of Al nucleus in the filter, *λ* is a decay contant and *t* is a time of irradiation.

For calculations a Mathematica softawre was involved, wherease a input datas as cross sections, delay decais and energy spectrum of the Am-Be source were taked respectively from IAEA Evaluated Nuclear Data File (http://www-nds.iaea.org/exfor/endf.htm), LBNL Isotopes Project – LUNDS Universitet (http://ie.lbl.gov/toi.htm) and IAEA Technical Reports Series No. 252 (IAEA 1985).

Experimental part covers a measurements of TL signals of the pellets covered and bared during irradiations by neutrons beam. Time of irradiation was set to 600s. For irradiations four dosemeters were prepared, each of them includes two pellets which the TL signal registered just after annealin was in range (*c-σ, c+σ*) where *c* is an average TL signal from 40 pellets and *σ* is a standard deviation.

In addition to provide the dependence of the intensity of TL signal from mass of the pellets theirs mass measurements and TL signal registration after 1mSv irradiation using ^{137}Cs source were carried out.

Results and discussion

Read out of the pellets irradiated by neutrons shows systematic increase of TL signal from pellets covers by Al filter during irradiation, results for four distances are presented in Figure 1. To veryfi supposed influence of gamma radiation emmited by ^{27}Mg and ^{24}Na deponed in the dosemeter filter the adecvate calculations were carried out, results of them and values of neutrons flux densities at four distances are shown in Table 1. It was shown that the activities of activation products can not have influence on the differences in TL signals. Mass measurement of 19 pellets gives results from 33,72 mg to 37,34 mg. There is no clear dependence betwen TL signal and tha pellets mass, Figure 2. Results of mass measurements testify to quite high composition anisotrophy of the pellets. For example the TL signal for a group of pellets which mass is around 35,7 mg (marked in Figure 2) oscillate from 0,93 to 0,97.

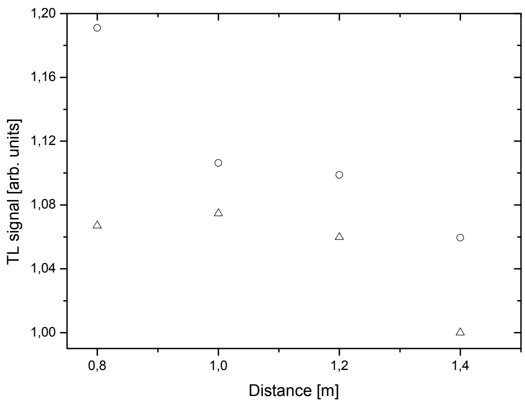

Fig. 1. Differences in TL signal from pellets covered (o) and bared (Δ) during irradiation at four distances.

The ratio of TL signal at the distances 0,8; 1; 1,2; 1,4m are respectively 1,12; 1,03; 1,04; 1,06.

Table 1. Neutron flux density and activities of the activation product. Time of activation 600s.

Distance [cm]	Neutron flux density [n/cm^{2}s]	Activity of ^{27}Mg [Bq]	Activity of ^{24}Na [Bq]
80	142	0,083	0,037
100	91	0,053	0,024
120	63	0,037	0,016
140	46	0,027	0,012

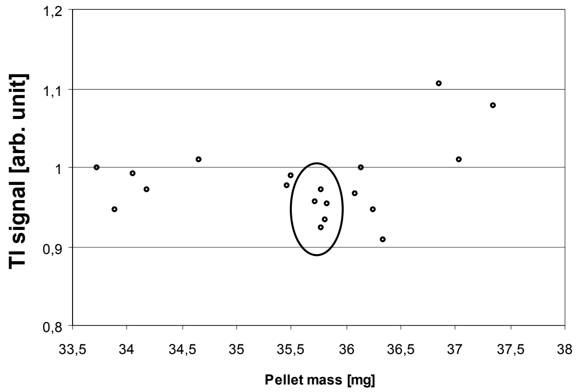

Fig. 2. Dependence of TL signal from the mass of pellet.

Conclusions

The concept to measure fast neutrons component indirectly by thermoluminescence materials which register the gamma ray of the activation products seems to be possible to realization. The main requirements that need to be fulfil are enought mass of the activated material and the very carefully prepared calibration of the TL material. One dosemeter should consist two or more TL materials which sensitivities are in very good agreement.

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Spectrophotometric response of the Fricke gel dosimeter developed at IPEN for clinical electron beams

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Abstract

Fricke gel dosimeter is one of the gel dosimeters that have been widely studied for application to clinical dosimetry. This work aims to evaluate the performance of the Fricke gel dosimeter developed at IPEN, prepared with 270 Bloom gelatine from porcine skin made in Brazil, for clinical electron beams from a VARIAN® electron linear accelerator in the energy range from 6 to 16 MeV to reference depth, using a water phantom and spectrophotometry technique. The following parameters were studied: color change, intra and inter batches reproducibility, dose response, dose rate and angular dependent response and response stability in function of storage time under two different conditions: 1) refrigeration and light protected and 2) room temperature and environment light. The excellent results obtained in this study indicate that the studied FXG solution can be a useful option for quality control of treatments of superficial tumours, using clinical electron beams.

Introduction

Different techniques and dosimetric materials have been widely studied because of the increasing demand for efficiency and safety in radiotherapy and radiosurgery treatments. The gel dosimetry has been highlighted because dosimeters can be shaped in three dimensions (3D) and different sizes and shapes so that they become very useful for complex dosimetry techniques such as intensity-modulated radiation therapy (IMRT) (Ibbott 2006).

Among the currently studied gel dosimeters is the Fricke gel dosimeter which is based on the oxidation of ferrous (Fe^{2+}) to ferric (Fe^{3+}) ions by action of ionizing radiation. Its response can be measured by optical absorption (OA) spectrophotometry and magnetic resonance imaging (MRI) evaluation techniques (Gore et al. 1984).

In this work the performance of the Fricke xylenol gel (FXG) dosimeter developed at IPEN, prepared with 270 Bloom gelatine from porcine skin made in Brazil, for clinical electron beams from a VARIAN® electron linear accelerator in energy range from 6 to 16 MeV to reference depth, using a water phantom and OA spectrophotometry technique was evaluated. Different parameters were studied namely: color change; intra and inter-batches reproducibility; dose response, dose rate and angular dependent response and response stability in function of storage time under two different conditions: 1) refrigeration and light protected and 2) room temperature and environment light.

Materials and methods

FXG solutions preparation

Different batches of Fricke gel solution using 5% by weight of 270 Bloom gelatine from porcine skin (GELITA®), ultra-pure water (ELGA® model PURELAB Option Q DV 25 water purifier), 50 mM of sulphuric acid (H_2SO_4), 1 mM of sodium chloride (NaCl), 1 mM of ferrous ammonium sulphate hexahydrate or Mohr's salt [$Fe(NH_4)_2(SO_4)_2 \cdot 6H_2O$] and 0.1 mM of xylenol orange ($C_{31}H_{28}N_2Na_4O_{13}S$) were prepared (Olsson et al. 1989). The chemical reagents (MERCK®) are of analytical grade.

The Fricke gel samples were conditioned in polymethyl methacrylate (PMMA) cuvettes (SARSTEDT®), with two parallel optical faces, dimensions of 10 x 10 x 45 mm^3 and optical path length of 10 mm and were individually sealed with polyvinyl chloride (PVC) film. The samples were stored under refrigeration ((4 ± 1) °C) and light protected during about 12 h (Olsson et al. 1989) after preparation and maintained 30 min at room temperature and light protected before irradiation.

FXG samples irradiation

The Fricke gel solution samples were irradiated in the reading cuvettes using a water phantom. Each three samples set of Fricke gel solution was packed with PVC film (Fig. 1) in order to avoid contact of the FXG solution with water. A 40 x 40 x 40 cm^3 MEDINTEC® water phantom filled with tri-distilled water was used to samples irradiation (Fig. 2).

Fig. 1. Three samples set of FXG solution packed with PVC film.

The samples were irradiated with clinical electron beams using a VARIAN® clinical electrons linear accelerator model CLINAC 2100C (Fig. 2) of the Radiotherapy Unit of the Cancer Centre of the Hospital Israelita Albert Einstein (HIAE), with electron energies between 6 and 16 MeV, absorbed doses from 0.05 to 21 Gy, dose rates from 80 to 400 cGy/min, irradiation angles between 0º and 180º, using a radiation field size of 10 x 10 cm^2 and different reference depth (0.6 to 2.0 cm/water) to ensure the maximum dose in the centre of each FXG sample. The temperature, humidity and atmospheric pressure of the irradiation room were maintained at 21° C, 55% and 697 mmHg, respectively, during the irradiations.

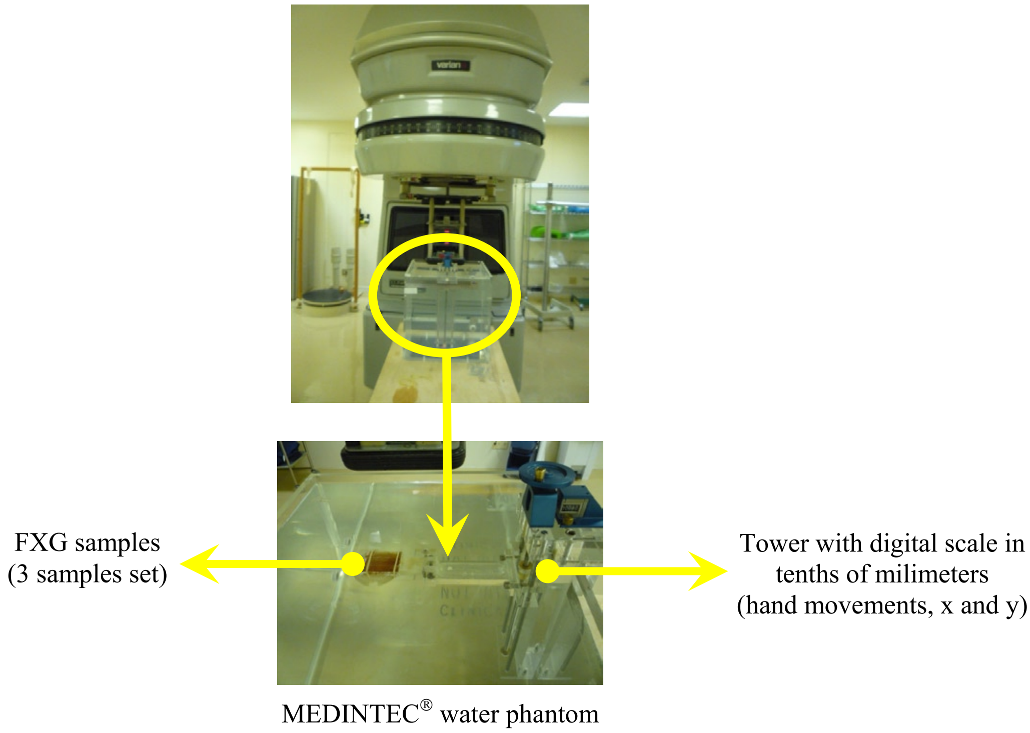

Fig. 2. Experimental set up for FXG samples irradiation in VARIAN® linear accelerator (CLINAC 2100C) using a water phantom.

OA evaluation

The OA spectrophotometry evaluation technique was used in this study. The measurements were performed using SHIMADZU® spectrophotometer model UV-2101PC (High Doses Laboratory of IPEN) immediately after dosimetric solution preparation and about 30 min after irradiation. The dosimetric wavelength was determined for each irradiated sample analyzed. This wavelength was used to obtain the absorbance values of the irradiated samples.

The following parameters were studied: color change (using an E.M.B.® lightbox model PRENDOGRAV of IPEN); intra and inter-batches reproducibility; dose response, dose rate and angular dependent response and response stability in function of storage time under two different conditions: 1) refrigeration and light protected and 2) room temperature and environment light.

The presented spectrophotometric responses correspond to the average of absorbance values of three samples and the error bars the standard deviations of the mean (type B uncertainties were not considered). The background value (non-irradiated samples) was subtracted from all absorbance values.

Results and discussion

Colour change

The colour change of the Fricke gel solution (in PMMA cuvettes) non-irradiated and irradiated with clinical electron beams with doses from 0.05 to 21 Gy, 12 MeV and 400 cGy/min is presented in Fig. 3. The samples present colour range from yellow-gold (non-irradiated solution) to violet (21 Gy) and the same colours are presented by FXG solutions irradiated with the others electron energies and dose rates studied.

Fig. 3. Colour range presented by Fricke gel solution non-irradiated and irradiated with doses between 0.05 and 21 Gy.

Intra and inter-batches reproducibility

To evaluate the intra-batch reproducibility of the FXG solution irradiated with clinical electron beams (1 Gy and 16 MeV) measurements of 5 sample sets subjected to the same experimental conditions were performed (Cavinato et al. 2010).

Five batches of 100 mL of FXG solution were prepared in the same day and under same experimental conditions in order to evaluate the inter-batches reproducibility of the dosimetric solution studied. These solutions were also irradiated with clinical electron beams (1 Gy and 16 MeV) and the optical measurements of samples were performed.

The intra and inter-batches reproducibility obtained is better than ± 3% (Cavinato et al. 2010) and ± 2%, respectively.

Dose response

The optical absorption spectra and spectrophotometric dose-response curve obtained of the FXG solution irradiated with clinical electron beams (12 MeV and dose range from 0.05 to 21 Gy) is presented in Fig. 4.

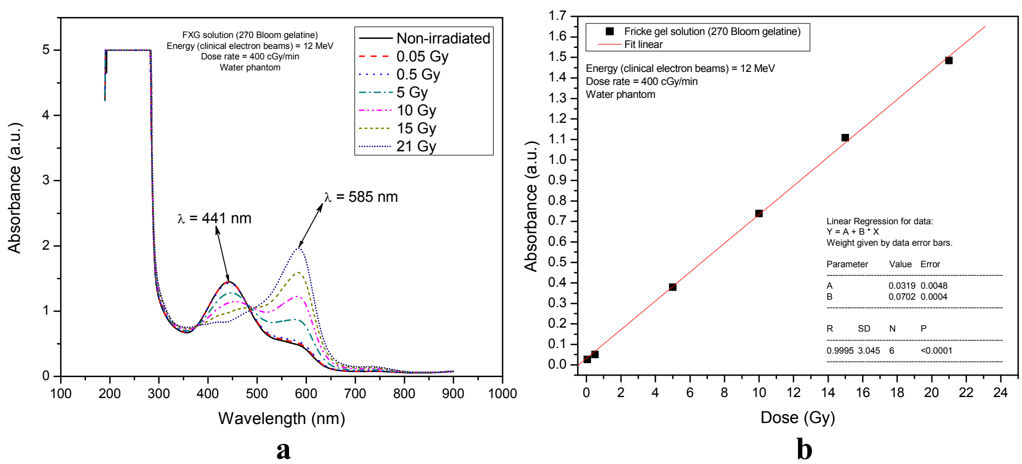

Fig. 4. Optical absorption spectra of FXG samples non-irradiated and irradiated with clinical electron beam (a) and dose-response curve of the FXG solution (b).

The solution prepared with 270 Bloom gelatine presents two absorption bands, as expected: one at 441 nm (Fe^{2+}) and other at 585 nm (Fe^{3+}) generated by induced oxidation of Fe^{2+} ions by ionizing radiation. It is observed intensification of absorbance values of the band at 585 nm (Bero et al. 2001) with increasing radiation dose while the absorption band at 441 nm tends to disappear (Fig. 4a) depending on the dose.

The Fricke gel solution presents a linear response over the dose range studied (Fig. 4b).

Dose rate

The dose rate dependent response (relative to higher absorbance value) of the FXG solution irradiated with 12 MeV, 10 Gy and dose rates from 80 to 400 cGy/min is presented in Fig. 5. The dose rate dependent response obtained is better than ± 3%.

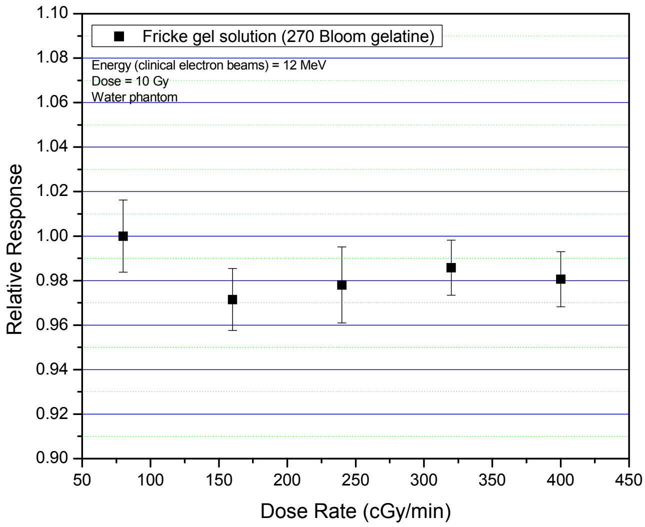

Fig. 5. Dose rate dependent response of the FXG solution irradiated with clinical electron beams.

Angular response

The angular dependent response (relative to higher absorbance value) of the Fricke gel solution irradiated with 6 and 12 MeV, 10 Gy and irradiation angles from 0º to 180º is presented in Fig. 6. The angular dependent response obtained for electron energies of 6 and 12 MeV is less than 1%.

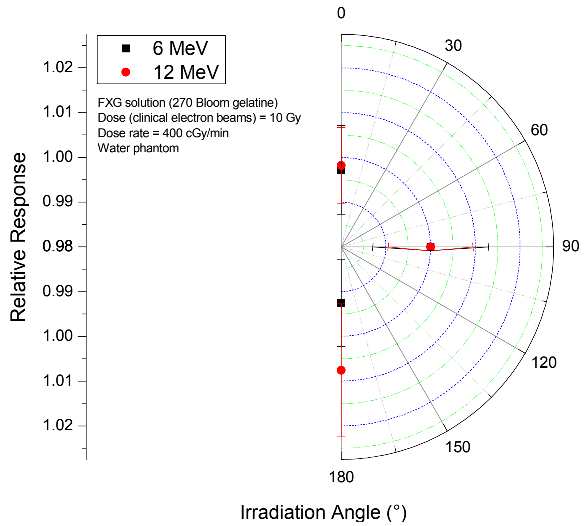

Fig. 6. Angular dependent response of the Fricke gel solution irradiated with clinical electron beams.

Response stability

The response stability of the FXG solution non-irradiated and irradiated (16 MeV and 1 Gy) in function of storage time under the conditions 1 and 2 studied (refrigeration and light protected, room temperature and environment light, respectively) is presented in Fig. 7.

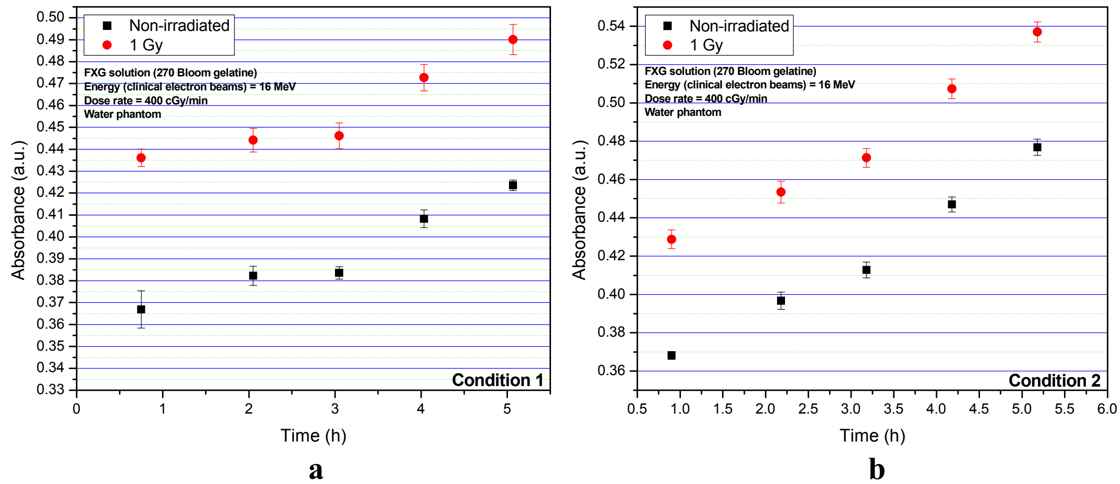

Fig. 7. Response stability of the Fricke gel solution non-irradiated and irradiated (clinical electron beams) in function of storage time under conditions 1 (a) and 2 (b).

It can be observed that in both storage conditions occurs intensification of absorbance values over time. However, for the non-irradiated and irradiated solution, there is a variation of the spectrophotometric response of less than 5% for conditions 1 (up to 4 hours after irradiation) and 2 (up to 3 hours after irradiation). Thus, the dosimetric samples should always be kept under refrigeration and light protected.

The energy dependent response previously studied (Cavinato et al. 2010) is better than ± 10% for the nominal energy of 9 MeV in the energy range studied. For energies greater than ≅ 12 MeV no energy dependent spectrophotometric response is observed.

Conclusions

The Fricke gel dosimeter developed at IPEN prepared with 270 Bloom gelatine produced in Brazil provides excellent results when irradiated with different energies, absorbed doses and dose rates of clinical electron beams. The results obtained in this study complement previous work using clinical electron beams (Cavinato et al. 2010) and indicate that the studied FXG solution can be a useful option for quality control of treatments of superficial tumours, using clinical electron beams. The obtained results also indicative the viability of employ this dosimeter in electron 3D dosimetry.

Acknowledgements

The authors are grateful to the HIAE to allow the samples irradiations in CLINAC 2100C accelerator and CAPES and CNPq by financial support.

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Dosimetric properties of agate stones

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Abstract
Brazilian stones have been studied for use in gamma high-dose dosimetry. Agate samples were studied in relation to their TL dosimetric properties, in this work. The TL curves obtained due to absorbed doses of 50 Gy up to 10 kGy exhibit two peaks around 150^{0}C and 210^{0}C. The minimum detection limits, dose-response curves and reproducibility of the TL response were obtained. The results indicate the possibility of use of these agate samples as radiation detectors for gamma high-doses, using the TL technique.

Introduction
Quality control programs in radiation dosimetry play an important role. Radiation processing presents several advantages such as food preservation, sterilisation of pharmaceutical and medical products and treatment of various materials. McLaughlin et al described several kinds of high-dose dosimeters, discussing their advantages and disadvantages. At the Radiation Metrology Laboratory of IPEN, Brazil, different stones have been studied for application in high-dose dosimetry. Amethyst [Rocha et al, 2003], quartz (Santos et al, 2001 and Navaro et al, 2002), topaz [Sousa et al, 2002] and jasper [Teixeira and Caldas, 2007] have already shown their usefulness for gamma dosimetry, using the thermoluminescent technique (TL).

Agate is a variety of chalcedony, a form of quartz, in which the color appears in bands or concentric zones. The bands may be of different colors or also of a uniform tone. The samples for this study were prepared from four different types of agate stones: yellow, moss green, gray and purple. The objective of the work was to characterize agate stone samples as detectors for gamma high dose radiation.

Materials and methods
Brazilian agate samples were studied in the work. All samples were initially cleaned, pulverized, and grain diameters between 0.075 and 0.180 mm were obtained. To facilitate easy handling, sintered agate pellets were prepared at the Laboratory for Production of Dosimetric Materials, IPEN, using Teflon as binder, and the parts were mixed in the ratio 2 (Teflon):1 (powdered sample). For the sintering process, the samples were thermally treated at 300^{o}C for 30 min followed by 400^{o}C for 1.5h.

The thermal treatment for reutilization of the material was established as 300°C for 1h. The irradiations of the samples were performed using a Gamma Cell-220 System of ^{60}Co (dose rate of 2.18 kGy/h), for doses from 50 Gy up to 10 kGy. The irradiations were made at ambient temperature; to guarantee the occurrence of electronic equilibrium during the irradiations, the samples were fixed between 3.5 mm thick polymethyl meth-acrylate plates (Lucite). The TL measurements were taken using a Harshaw Chemical Co. reader, model 2000 A/B, and the data acquisition was realized using a virtual instrument (ADC-212), Pico Technology Ltd., and a personal computer.

Results and discussion

Dosimetric properties such as lower detection limits, reproducibility and dose-response curves were obtained in this work. Figure 1 shows the thermoluminescent glow curves of the agate samples: gray, purple, moss green and yellow, taken 1h after their irradiation with 10 kGy. All samples present two main TL peaks, one around 150^0C and the second about 210^0C (dosimetric peak).

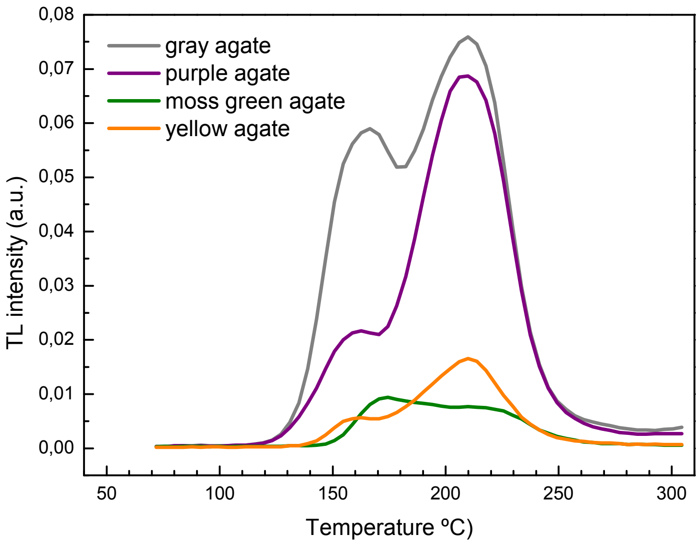

Fig. 1. TL glow curve of an agate-Teflon pellet irradiated with 10 kGy (^{60}Co)

To remove the first TL peak (150^0C) of the agate sample glow curves, thermal treatments at 130^0C during different time intervals: 5 min, 10 min, 15 min, 30 min and 60 min were tested. All thermal treatments presented the same results: peak 1 was removed and the thermal treatment of 130^0C/5min was chosen as adequate (Figure 2). All agate samples (gray, purple, moss green and yellow) presented the same results.

The TL response of the agate sintered pellets as a function of the absorbed dose was measured from 50 Gy up to 10 kGy (Figure 3). The standard deviation values were less than 3.2%. The lowest detectable value was determined studying the variation of the signal obtained by the reading of non-irradiated samples. The lowest detectable value was 4 mGy, 1.5 Gy, 1.0 Gy and 0.2 Gy for gray, purple, moss green and yellow agate samples, respectively. Figure 3 shows that the dose response is linear up to 5 kGy in case of gray agate samples and supralinear for the other samples.

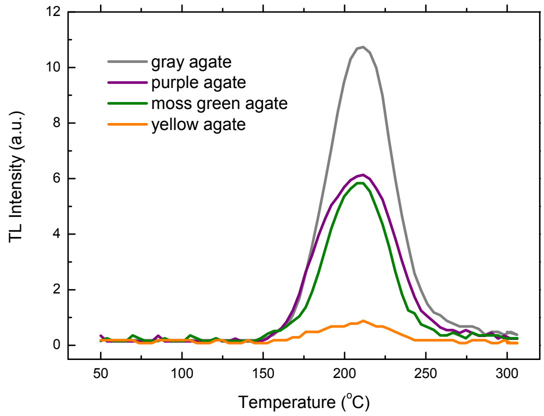

Fig. 2. TL glow curves of agate samples irradiated with 5 kGy (^{60}Co),after thermal treatments of 130^{0}C/5min.

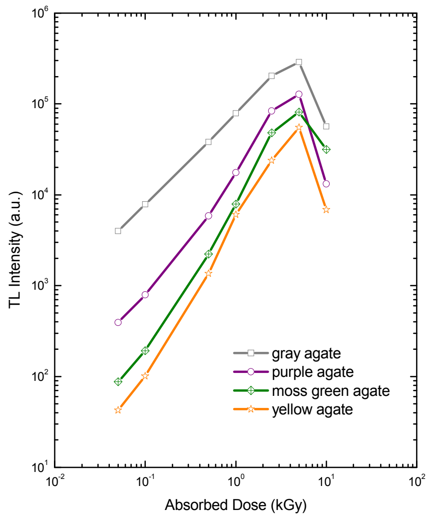

Fig. 3. Dose-response curves of agate samples for ^{60}Co radiation. Measurements were taken after the irradiation and the thermal treatment at 130^{0}C/5min.

Conclusions

The results obtained on the main dosimetric characteristics (reproducibility, dose response up to 5 kGy and lower detection limits) show that agate samples can be applied in high-dose dosimetry. The advantages of this kind of material are: found in Brazilian natural mines, easy to handle and very low cost. Agate samples may be applied for dosimetry in the main radiation process of pasteurization, processes of water purification, sterilization and disinfestations of products, among others.

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Characterization of polycarbonate dosimeter for gamma-radiation dosimetry

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Abstract

A simple and inexpensive polymeric material whose special properties can be determinate, quantified and related to absorbed dose was studied in this work. Commercial polycarbonate (PC) is a new type of film detector that suffers yellowing upon radiation exposure and the color change was used as dosimetric property to measure the absorbed doses. The dosimeter consists of a piece of polycarbonate film of dimensions 3 x 1 cm^2 and 3 mm thick and the spectrophotometry was the investigation technique used. PC films were irradiated with gamma doses between 1 and 150 kGy from a ^{60}Co Gammacell source. The absorbance was measure on a spectrophotometer Shimadzu UV-2101PC. The main dosimetric characteristics studied were: pre- and post-irradiation stability, dose – response, environmental conditions influence and response dependence with dose rate. PC films are easy to prepare and to analyze. The influence of environmental conditions was observed and must be corrected. Polycarbonate dosimeters presents good stability and reproducibility and linear behavior in the dose range studied indicating that the dosimetric characteristics are suitable to determine high gamma doses.

Introduction

Some polymeric materials present changes in their properties due to the interaction of ionizing radiation that can be related with the absorbed dose [2]. Radiation induced chemical reactions results in rearrangements and/or formation of new bonds and the main effects are scission of main chain (degradation) or cross-linking, which are simultaneous and competing processes. To be applied in radiation dosimetry polymer films must present preferentially linear response in the dose range to be measured. It is necessary to determine its dosimetric properties such as lower and upper limits of useful dose range; dose rate dependent response; stability before and after irradiation and environmental conditions effects [1,4,5]. Polycarbonates are usually applied in neutron and alpha particles detection using nuclear tracks detection technique nowadays they have been studied as a dosimeter to measure gamma-rays doses [6]. The radiation induced main chain scission of PC and produces phenoxy radical responsible for the yellowing [4,5]. PC films are easy to prepare and to analyze and inexpensive. The influence of environmental conditions was observed and must be corrected in the studied material. Polycarbonate dosimeters presents linear behavior in the dose range studied and indicates that the dosimetric characteristics of are suitable to determine high gamma doses. In this work samples of commercial polycarbonate were investigated to be applied as dosimetric material evaluating the radiation dose response to ^{60}Co gamma radiation.

Material and methods

Samples of commercially available polycarbonate with and without protection against UV radiation manufactured by "Policarbonatos do Brasil" (3 x 1 x 0.3 cm^3) were irradiated free in air, under electronic equilibrium conditions and at room temperature with doses between 1 and 150 kGy and dose rate of 2.60 kGy/h using a ^{60}Co gamma radiation source.

The radiation induced color change was the basis of measuring absorbed doses. Optical absorption measurements of non irradiated and irradiated samples were taken in a Shimadzu spectrophotometer UV-2101PC at wavelength ranging between 190 and 900 nm.

The lower dose detection limit was obtained by the absorbance values of the maximum wavelength of at least 60 non irradiated detectors and the upper limit was determinate by external limitations.

All presented results are the average of 3 measures and the error bars the standard deviation of the mean. The observed standard deviation of the mean for all applied doses was better than 96 %.

Results

The typical non irradiated PC spectrum was obtained and presents differences between PC with and without protection against UV radiation damage as showed in Fig. 1.

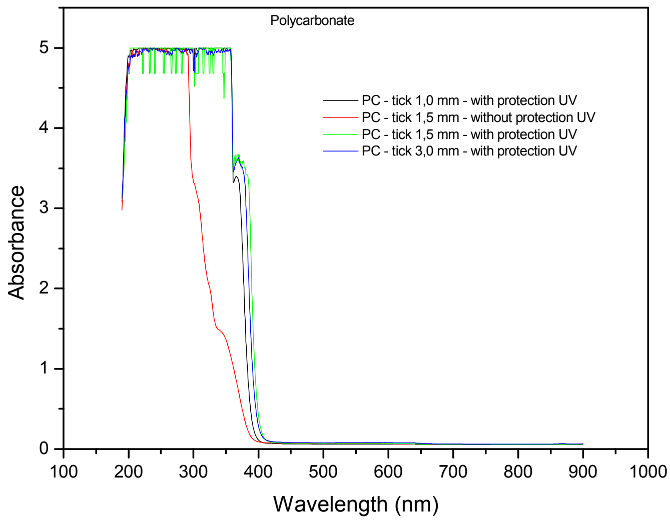

Fig. 1. Non irradiated polycarbonate samples absorption spectra.

The optical response stability before irradiation is better than 95% when PC samples are exposed to normal ambient conditions (20°C and 60% humidity), however, after irradiation the samples must be maintained at low temperature to avoid losses of response, approximately 30% after 48 hours at room temperature and 8% if maintained at 5°C and light protected.

The lower detection limit was obtained taking three times the value of the standard deviation of measurements of at least 60 non-irradiated PC samples, expressed in units of absorbed dose. The dose value of 1 kGy was obtained [7].

After irradiation PC spectra presents an absorption band centred in 412 nm, Fig. 2, that was used to evaluate the optical response.

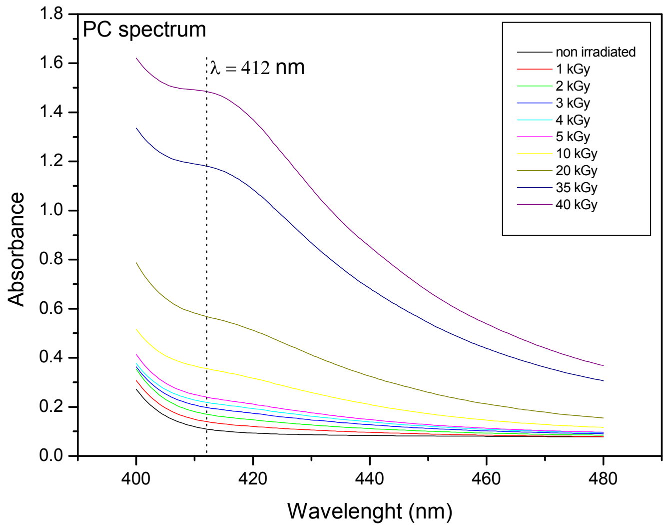

Fig. 2. Non irradiated and irradiated PC absorption spectra.

The change color of PC samples originally transparent to yellow dark when are irradiated with different gamma radiation doses can be seen in the Fig. 3

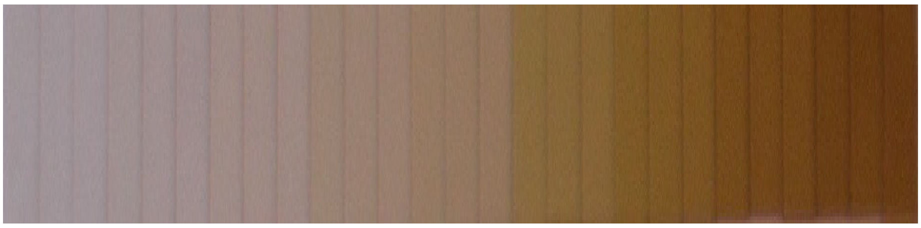

Fig. 3. Color change of PC samples irradiated with gamma doses between 1 and 150 kGy.

Experimental results show that the response reproducibility is better than 96%.

The dose rate dependence was significant for low doses and reading corrections must be made because this dependence can introduce large errors in the dose measures [3].

The absorption intensity of the 412 nm peak increase linearly as a function of absorbed dose as showed in the Fig. 4 to gamma radiation in the dose range between 1 and 150 kGy.

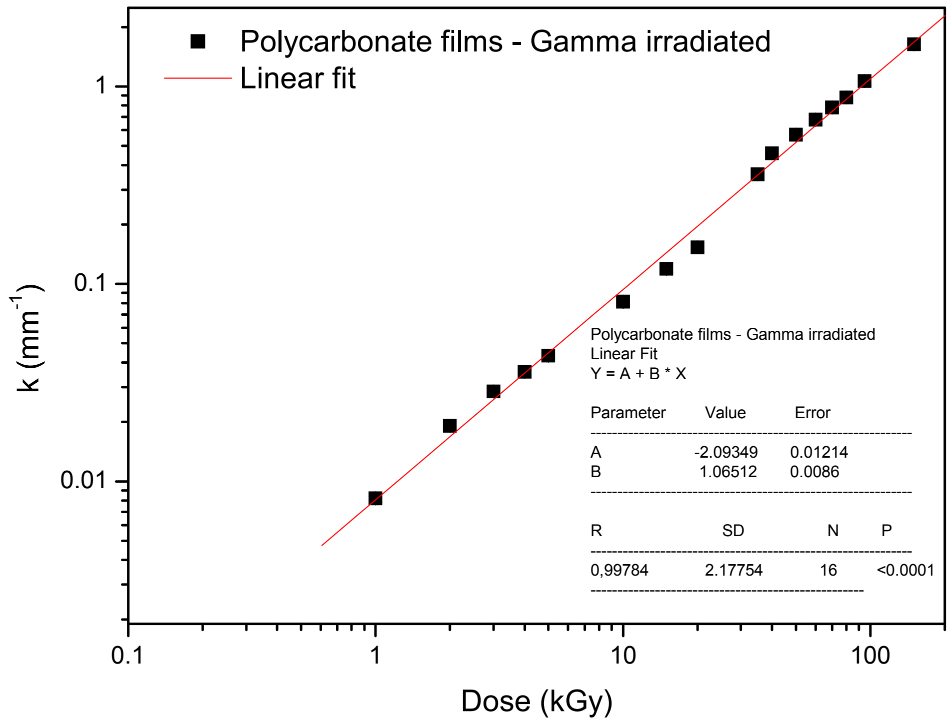

Fig. 4. Dose response curve of polycarbonate films irradiated with ^{60}Co gamma radiation, λ = 412 nm.

Conclusions

The results obtained indicate that PC can be used as an alternative dosimetric material. The advantages of the use of PC detectors in high-dose dosimetry are the reduced size, very low cost, easy handling and the fact that the absorbance analysis is very simple. The storage conditions doesn't affect the non-irradiated detector response, however, after irradiation the detector must be kept under refrigeration to preserve its optical response and reading must be done up to 24 hours after irradiation.
The dose response curves obtained show that this dosimeter can be used in an useful dose range between 1 and 150 kGy.

Acknowledgements

The authors wish to thank FAPESP (Fundação de Amparo a Pesquisa no Estado de São Paulo), CAPES (Coordenação de Aperfeiçoamento de Pessoal de Nível Superior) and CNPq (Conselho Nacional de Desenvolvimento Científico e Tecnológico) for the financial support.

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Pellets of oyster shell for high dose dosimetry: Preliminary study

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Abstract
Several kinds of silicates have demonstrated their usefulness as radiation detectors for high dose dosimetry. In this work, oyster shell powdered samples were tested for high doses using the thermoluminescence (TL) and thermally stimulated exoelectron emission (TSEE) techniques. Pellets were obtained from the oyster shell powder mixed with Teflon followed by a sinterization process of thermal treatments of 300°C/30min and 400°C/1.5h. The TL glow curves showed two peaks at 110 and 220°C, and in the TSEE case, two peaks at 120 and 180°C. The main dosimetric properties were studied: reproducibility, dose-response curves and lower detection limits. The preliminary results showed the potential use of the studied material as radiation detectors.

Introduction
The mean objective of the study of new materials is to identify and to overcome the limits imposed, naturally, for their physics and chemical properties (McLaughin *et al*, 1989). The research of new materials applied in dosimetry and other activities that involve ionizing radiations began as early as 1663 when Robert Boyler reported his observations to the Royal Society in London (Becker, 1973). Thereafter several researchers have published results about new materials. In the radiation metrology research group at IPEN some works have been carried out with silicates. Caldas (1989), Caldas and Souza (1991), Teixeira (2004) studied the dosimetric properties of Brazilian and imported glasses applied to gamma and electron radiations. The dosimetric properties of jade and inossilicates were carried out by Melo et al (2004; 2008). They studied the dose-response in the range of 10Gy to 20kGy through the thermoluminescent (TL) technique, verifying the linear behaviour between 10Gy and 1kGy, and the sublinear behaviour between 100Gy and 20kGy, using the thermally stimulated exoelectron emission technique.

Other materials as the calcium-based phosphors have been studied and showed good dosimetric properties. Since the 60's these materials have been studied by several researchers. Among them are Medlin (1968) and Nambi and Mitra (1979), that used the TL technique, to study these materials; they verified that calcites are strong emitters of TL; the dose response is linear up to 2 kGy. Bapat and Nambi (1975) showed that the calcite response becomes sublinear at higher doses before reaching a maximum intensity at a dose of about 20kGy, and the TL emission spectra consist of a broad band with maximum at 630µm.

The components that form the oyster shell are $CaCO_3$, CaO, SiO_2, MgO, Al_2O_3, SrO, P_2O_5, Na_2O and SO_3 (Yoon *et al*, 2003, Freire *et al*, 2009). In addition to all these compounds, there are trace amounts of Mn^{2+}, Pb^{2+}, Ce^{3-}, $(UO_2)^{2+}$, Dy^{3+}, Sm^{3+}, Tb^{3+}, Nd^{+3}, Eu^{3+} that are responsible for the luminescence of calcite and for a radiation-induced center of violet emission (Marfunin, 1979). Although the oyster shell is formed by many compounds, the $CaCO_3$ (calcium carbonate) corresponds to approximately 96% of material that forms it. Calcium carbonate is found in two polymorphic forms, calcite (rhombohedral) and aragonite (orthorhombic) (Ikeya, 1993). The aragonite is only meta-stable under ambient conditions, transforming to calcite on heating. A reconstructive aragonite to calcite transformation takes place in the 300 – 400°C range in biogenic samples (Balmain e*t al*, 1999). Due thermal treatment the luminescence in $CaCO_3$, that is due to band-to-band recombination, will blueshift or enhance the higher energy part of the spectra (Medeiros *et al*, 2006).

TL signal is present in two polymorphic forms and caused by presence of Mn^{2+}. This presence of Mn^{++} results in glow peaks at 180° and 250°C in aragonite (Medlin, 1963), strong glow peaks at 80° - 100°C and weaker ones at 150°-175°, 215°-230°C and 320° - 330°C in calcite (Sunta,1985).

The objective of the present work was to study of the dosimetric properties of gamma induced centers in marine oyster shell using the TL and TSEE techniques.

Materials and methods

The materials studied in this work, known as oyster shells (Figure 1), usually used for decoration, were obtained in a market of Recife city originating from the northwest coast of Brazil. The pellets were prepared mixing powered oyster shell, with diameter between 177µm and 74µm, with powered Teflon in a 1:2 proportion.

Fig. 1. a) Bracelet made of natural oyster shell; b) pellets of oyster shell powder with Teflon and c) powder of oyster shell on an aluminium foil.

The oyster shells were powered and sieved retaining the 177µm – 74µm size fraction. The samples were thermally treated at 300ºC/1h and kept in dark until the irradiation and the TL and TSEE measurements were performed. After the treatments the pellets were irradiated at room temperature using a ^{60}Co panoramic source of the Centre for Radiation Technology (IPEN), and a ^{60}Co radiotherapy source of the Centre for Radiation Metrology. The pellets were irradiated with doses in the range from 5Gy to 10kGy. All measurements were taken at room temperature, and the pellets were sandwiched in Lucite plates with 3.5mm thickness to guarantee the electronic equilibrium during the irradiations. The TL measurements were taken using a Harshaw TL reader model 2000 A/B in the range from 50° to 300°C and the TSEE measurements were taken using a home-made system developed at the Centre for Radiation Metrology in the range from 30° to 300°C. All TL and TSEE measurements were obtained with linear heating rates of 10ºC/s in nitrogen flux and P10 flux, respectively.

Results

The TSEE measurements show peaks appearing at approximately 120°C and 180°C (Figure 1a). The TL emission curve, presented in Figure 1b, shows two characteristic peaks at 110°C and 220°C.

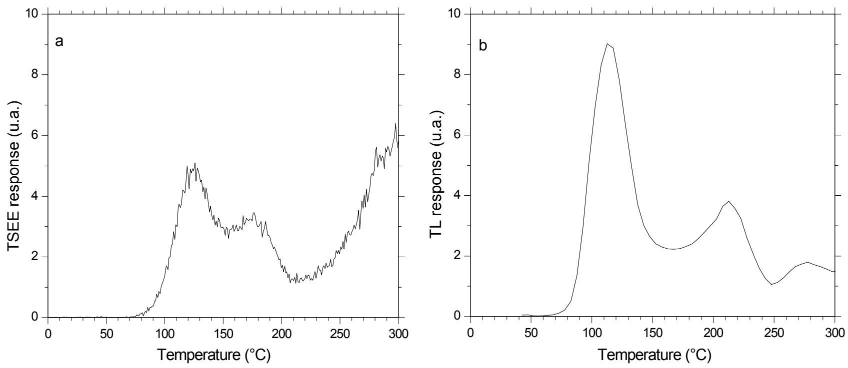

Fig. 1. TSEE(a) and TL(b) curves of the oyster shell in pellets, irradiated to 400Gy at room temperature, using a ^{60}Co source.

Figure 2 shows the dose-response curves of oyster shell pellets irradiated (^{60}Co) at room temperature over a range of 5Gy to 10kGy. The curve shows the increasing behaviour in function of the absorbed dose. The maximum standard deviation for the dose response curves was 3.5% for TSEE measurements (Figure 2a) and 4.5% of the TL measurements (Figure 2b).

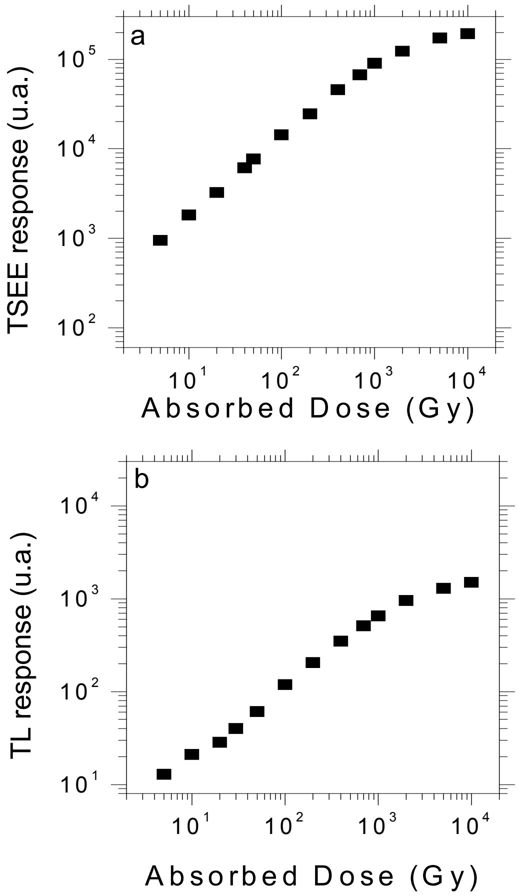

Fig. 2. TSEE (a) and TL (b) dose-response curves of the oyster shell in pellets, irradiated at room temperature, using a ^{60}Co source.

Reproducibility tests were carried out in samples irradiated to 5Gy (^{60}Co) at room temperature. The thermal treatment used before each irradiation of the samples was 300°C/1h. The maximum standard deviation, for irradiated samples, was 4.2% for TL measurements and 2.4% for TSEE measurements.

The lower detection limits were determined by variability studies of the TL and TSEE responses of samples treated at 300°C/1h and not irradiated. The value obtained using the detection of triple standard deviation of the samples was 500mGy.

Discussion

The results obtained show that oyster shells present possibility of use in high dose dosimetry. Oyster shells are biogenic materials in which the main minerals constituent is calcium carbonate. Observations carried out by Carmichael *et al* (1994) on the morphology of the oyster shells conclude that there is a great variation in the extent and type of mineralization. However, in studies carried out by Parker and Balmain *et al* (1999), they showed that a reconstructive aragonite to calcite transformation takes place in the 300 – 400°C. In the present work the pellets were made of powered oyster shell mixed with polytetrafluoroethylene and treated 400°C/1.5h, a procedure that probably transforms the aragonite in calcite.

The TL curves of calcium-based phosphors show two characteristic peaks at 175°C and 240°C (Anderle *et al*, 1998; Medlin, 1963; Nambi and Mitra, 1978) using a heating rate of 4.75°C/s. In the present work, the heating rate was 10°C/s that may have caused the dislocations of the TL peaks to 110°C and 220°C. The TSEE emission curve showed two peaks at 120°C and 180°C

The response of TL measurements shows a behaviour that agrees with that published by Bapat and Nambi (1975).

Conclusions

The good preliminary results of the studies with oyster shell samples in this work show the possibility of their use for high-dose dosimetry.

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Design and assembly of a simple monitor ionization chamber

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Abstract

At the Instituto de Pesquisas Energéticas e Nucleares, Brazil, a new monitor ionization chamber was designed, developed and characterized. This monitor is a parallel-plate type ionization chamber, and it has two sensitive volumes formed by three conductive foils. The chamber body is made of PMMA disks that are positioned outside the primary radiation beams. The pre-operational tests and the response stability tests were performed. The pre-operational tests, as saturation curve, ion collection efficiency, polarity effect and response linearity showed very good results. The response stability tests (repeatability and long-term stability) also showed satisfactory results, within international recommendations. Besides the good results, this ionization chamber is very simple to manufacture, and all materials used in this project are of low-cost and commercially available.

Introduction

Ionization chambers are the most used dosimeters. They are simple and they can be used for many purposes, depending on their material, volume and design.

A special kind of ionization chamber is the transmission chamber that is used as a monitor chamber for X-radiation beams. This kind of ionization chamber has some particular characteristics, such as a large sensitive volume which may cover the entire radiation beam section and is made of materials which are transparent to the X-rays (IAEA 2007).

In a Calibration Laboratory, the transmission ionization chamber can be used as a reference instrument since its response shows good stability and the calibration factor is transferred from a standard device (IAEA 2007). In this case, the transmission chamber may be calibrated periodically.

In the Calibration Laboratory of IPEN, two monitor chambers have recently been constructed (Yoshizumi and Caldas 2010a, 2010b). These ionization chambers have a ring-shaped design, and they measure only the radiation penumbra. They showed usefulness according to international recommendations.

Materials and methods

A simple transmission ionization chamber was designed and assembled at the Calibration Laboratory of IPEN. This ionization chamber has two sensitive volumes formed by two aluminized polyester foils and a graphite coated polyester foil. The chamber body is made of PMMA and the electrodes are connected to co-axial cables.

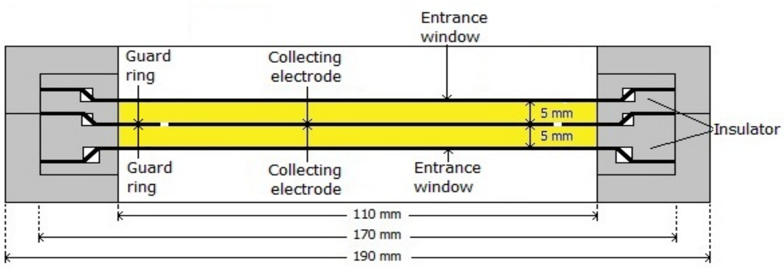
An electrometer, *Physikalisch-Technische Werkstätten* (PTW), model UNIDOS was used as a measuring assembly.

The characterization tests were performed using an X-ray unit, Pantak/Seifert, model ISOVOLT 160HS. Diagnostic radiology beam qualities, defined by IEC 1267 (1994), established in this equipment, were utilized in this work. The air kerma rates were measured using a secondary standard ionization chamber, PTW, model 77334. This standard ionization chamber calibration factors are traceable to the German primary standard laboratory *Physikalisch-Technische Bundesanstalt*, PTB.

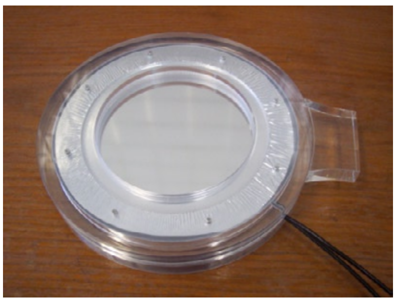
The response stability tests were performed using a $Sr^{90}+Y^{90}$ check source, PTW, model 8921.

Results

The transmission chamber developed at IPEN has two entrance windows of aluminized polyester foils of superficial density of 1.87 mg.cm^{-2}. The collecting electrodes and the guard rings are made of a graphite coated polyester foil. The graphite coating was made using a commercial lubricant spray Aerodag G, Acheson. A ring-shaped mask was made to separate the collecting electrode and the guard ring when spraying. A diagram and photography of the transmission chamber can be seen in Figure 1. The total sensitive volume of the chamber is about 63.5 cm^{3}.



(a)



(b)

Fig. 1. (a) Diagram and (b) photograph of the transmission ionization chamber.

Saturation curve, ion collection efficiency and polarity effect

The saturation curve was determined using the radiation beam quality RQA5 at a distance of 30 cm from the tube focal spot. The voltage applied to the sensitive volume of the chamber was increased in steps of ±50 V, covering a range from -400 V to +400 V. The results can be seen in Figure 2. The chosen operating voltage was -300V.

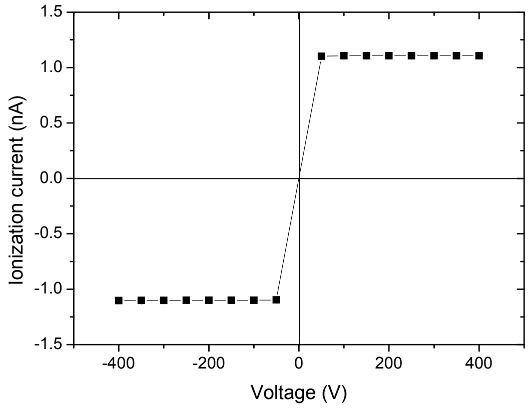


Fig. 2. Saturation curve of the transmission ionization chamber. The uncertainties associated to the measurements are negligible and they do not appear in the graph.

From the results obtained in the saturation curve test, the ion collection efficiency and the polarity effect of the chamber could be determined.

The ion collection efficiency was calculated using the two-voltage method (IAEA 2000):

$$k_s = \frac{\left(\frac{V_1}{V_2}\right)^2 - 1}{\left(\frac{V_1}{V_2}\right)^2 - \left(\frac{M_1}{M_2}\right)},$$

where $V_1 = \pm 300$ V and $V_2 = \pm 150$ V and M_i is the chamber response obtained when V_i is applied. For both voltages, the ion collection efficiency was higher than 99%, i.e., no ion recombination occurs in the sensitive volume.

The polarity effect measures the variation between the responses obtained for the same voltage but of opposite signals. For all tested voltages, the maximum variation was 0.66%, which is within the recommended limit of 1% (IEC 1997).

Response linearity

The chamber response linearity test was evaluated using the X-ray unit Pantak/Seifert and the diagnostic radiology quality RQA3. In this test, the ionization current applied to the tube was varied from 0.5 mA to 40.0 mA, covering a range of air kerma rate from 1 mGy.min^{-1} to 88 mGy.min^{-1}. As can be seen in Figure 3, the transmission chamber presents a linear response for the air kerma rate range studied. A linear regression was obtained that resulted in a correlation factor R of 0.99997.

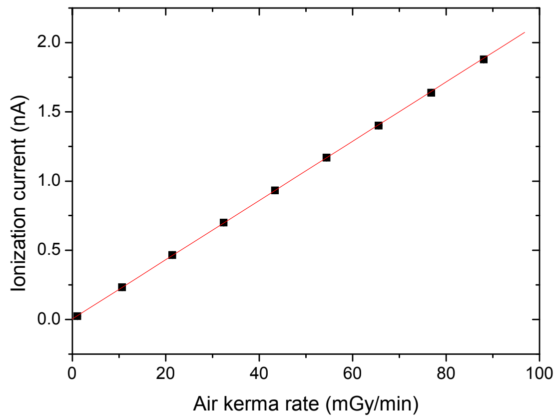


Fig. 3. Response linearity of the transmission ionization chamber. The uncertainties associated to the measurements are negligible and they do not appear in the graph.

Response stability tests

To evaluate the stability of the chamber response, four tests were performed: stabilization time, leakage current, repeatability and long-term stability. A PMMA holder was used to position the check source in front of the entrance window of the chamber.

For the stabilization time test, the check source was positioned in front of the transmission chamber, and then the chamber was connected to the electrometer, in the operating voltage of -300 V. The ionization current was measured ten times, 15 min, 30 min, 45 min and 1 h after the chamber connecting. The mean values obtained for each measurement time are shown in Table 1.

Table 1. Mean values of the ionization current obtained at different times from the application of the operating voltage to the transmission chamber.

Time (minutes)	Mean value of the ionization current (pA)
15	-100.9 ± 0.8
30	-101.2 ± 1.6
45	-101.1 ± 2.8
60	-101.2 ± 0.8

The ionization current obtained 15 minutes after switching on the chamber is 99.7% of the 1 h stabilization current. This result is within the recommended limits of ±2% of response variation (IEC 1997).

The leakage current was measured before and after irradiation. The charge was collected during 20 minutes with no radiation source present. The maximum leakage current was lower than 1.0% of the ionization current obtained with the minimum air kerma rate studied (using the beta check source), therefore within the recommended limit of 5.0% (IEC 1997).

For the repeatability test, ten consecutive measurements of the collected charge accumulated during 15 s were performed. The standard deviation of these measurements shall not be greater than 3% of the mean value according to the IEC 61674 publication (IEC 1997). This test was performed seven times in different days and the maximum variation obtained was 0.2%, therefore within the recommended limit.

From the mean values determined in the repeatability test, the long-term stability of the transmission chamber can be evaluated. Figure 4 shows the result of the seven tests performed.

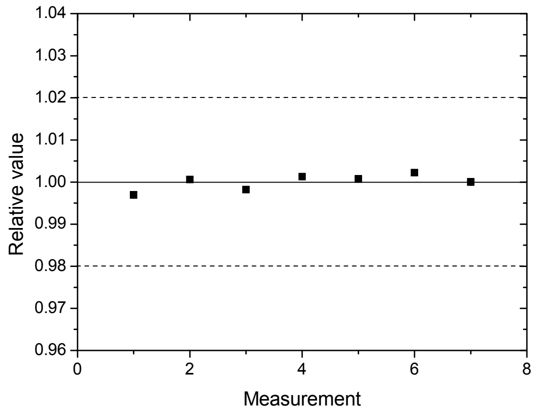


Fig. 4. Long-term stability of the transmission ionization chamber. The dashed lines represent the recommended limits. The uncertainties associated to the measurements are negligible and they do not appear in the graph.

From the results shown in Figure 4, the maximum variation of the chamber response in a period of 10 days was 0.3%. The IEC publication recommends a maximum variation of ±2% for the long-term stability test (IEC 1997), so the chamber response is stable according to this recommendation.

Conclusions

A simple transmission ionization chamber, using commercially available materials, was assembled. The characteristics and the response stability of the home-made transmission ionization chamber were evaluated. All performed tests resulted in good agreement to international recommendations. This chamber represents a good alternative since it is simple, easy to construct, and uses low-cost materials.

Ackowledgements

The authors are grateful to Conselho Nacional de Desenvolvimento Científico e Tecnológico (CNPq), Coordenação de Aperfeiçoamento de Pessoal de Nível Superior (CAPES), Fundação de Amparo à Pesquisa do Estado de São Paulo (FAPESP) and Ministério de Ciência e Tecnologia (INCT for Radiation Metrology in Medicine), Brazil, for the partial financial support.

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The use of TL dosemeters in HZE radiation fields

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Abstract
Astronauts in space are exposed to a radiation environment that can have deleterious consequences. To carry out an accurate dosimetry for the radiation protection of astronauts and cosmonauts engaged in long-term space flights, we measured efficiencies of Thermoluminescence detectors (TLDs) exposed to high-energy heavy ions accelerated at the NASA Space Radiation Laboratory (NSRL) at the Brookhaven National Laboratory (Upton, NY). The achieved knowledge will be applied to the experimentally measured dose onboard the International Space Station (ISS).

LiF:Mg, Ti (TLD-100), ^{6}LiF:Mg, Ti (TLD-600) and ^{7}LiF:Mg, Ti (TLD-700) were exposed at the NSRL to iron, protons and oxygen high energy beams. The preliminary results indicate that efficiency of TLDs is reduced for iron more that for protons and oxygen.

Introduction
The radiation hazard for the astronauts consists of protons and electrons trapped in the Earth magnetic field, proton emitted in the course of solar particles events (SPE), and protons and the energetic nuclei of other elements (HZE particles) that constitute galactic cosmic rays (GCR). It is necessary to accumulate new data on dosimetric and microdosimetric characteristics of onboard spacecraft radiation fields and passive dosimeter systems could help to fulfil this task [Reitz, 1994]. In particular, Thermoluminescence detectors (TLDs) are widely used in space dosimetry because of their small size, resistance to environmental conditions and the possibility of accumulating data for long periods of time without external power. The main purpose of these ground-based measurements was to acquire TL-efficiency to apply this knowledge to the determination of dose equivalent from space radiation. Since the energy range of space radiation ranges up to a few GeV/n and its spectral composition extends from protons to Fe nuclei, evaluation of TL-efficiency is crucial to the performance of TLDs.

The relative TL-efficiency, $\eta(E)$, describes quantitatively the efficiency of conversion of radiation energy absorbed in the detector into TL light, as compared with efficiency of such conversion after with standard radiation (γ radiation):

$$\eta(E) = \frac{TL(E)/D(E)}{TL(\gamma)/D(\gamma)}$$

Our interest is to determine TL efficiencies, because it depends not only upon the LET but also on the ion charged.

The scientific reason for the change in the TL-efficiency with increasing ionization density of the particle relates to the dose deposition profiles from high charged particles. A slowing down high charged particle deposits its energy to an overhelming extent by secondary electrons (δ electrons) created in the slowing down process. Depending on the charge and the velocity of the ion, the energies and ranges of these secondary electrons vary greatly [Berger et al, 2008].

Material and methods

We used TL phosphor, LiF:Mg, Ti: TLD-100, TLD 600 and TLD 700 (size 3.2 x 3.2 x 0.89 mm, see figure 1). Table 1 summarizes the laboratory procedures, such as heating cycles, reference radiation, TL reader etc., for all types of TLDs.

Table 1. Parameters of laboratory procedures.

Reference radiation	Photons 6MeV
TL reader	Harshaw 3500
Heating rate	5°C/s
Maximum temperature	400°C
Annealing cycle	400°C (1h), 100°C (2h)
Cooling rate	Slow
Calibration	Single chip

The reference radiation for the evaluation of TL-efficiency was a 6 MeV photon beam obtained at LINAC of “Istituto per la Cura dei Tumori di Napoli, Fondazione Pascale”.

Although GCR consist of 85% protons, 14% helium, and 1% heavier particles (iron 0,02% only), HZE nuclei are very important in space radiation dosimetry because highly charged and very densely ionizing. As a consequence, even though the number of HZE particles is relatively small, they have a significant biological impact that is comparable to that of protons. For this reason, TLDs were exposed to 1 GeV/nucleon iron beam, 1 GeV proton beam and 400 MeV/nucleon oxygen beam accelerated at the NASA Space Radiation Laboratory (NSRL) at the Brookhaven National Laboratory (Upton, NY). The size of the beam was about 15x15 cm and the disuniformity in this area below 5%. The dose was measured by a Far West thimble chamber. Details of dosimetry at NSRL are described in the reference [Lowenstein et al, 2007]. Until now, the maximum beam energy of charged particles for whom TLDs were exposed to evaluate TL-efficiency was 500 MeV/nucleon, at the Heavy Ion Medical Accelerator (HIMAC) of the National Institute of Radiological Sciences (NIRS) in Chiba, Japan [Yasuda et al, 2006].

Results

Post-exposition reading of TLDs was performed by an Harshaw model 3500 manual TL reader. In table 2 the preliminary results of TL-efficiencies for all types of TLDs exposed to different beams are reported.

Table 2. TL-efficiencies for different charged particle beams and different TLDs.

TLD type	Charged particle beam	Tl-efficiency
TLD 100	Protons 1 GeV/nucleon	0,58±0,05
TLD 100	Iron-ion beam 1 GeV/nucleon	0,30±0,03
TLD 100	Oxigen-ion beam 400 MeV/nucleon	0,51±0,05
TLD 600	Protons 1 GeV/nucleon	0,76±0,07
TLD 600	Iron-ion beam 1 GeV/nucleon	0,40±0,04
TLD 600	Oxigen-ion beam 400 MeV/nucleon	0,68±0,06
TLD 700	Protons 1 GeV/nucleon	0,73±0,07
TLD 700	Iron-ion beam 1 GeV/nucleon	0,35±0,04
TLD 700	Oxigen-ion beam 400 MeV/nucleon	0,71±0,07

These TL-efficiencies values suggest that the dose equivalent from space radiation for astronauts, calculated with calibration factor obtained exposing TLDs at 6 MeV photon beam was underestimated [Pugliese et al, 2008]. Consequentely, more accurate determinations of this parameter have to be performed, also taking into account the real spectrum of the radiation field in the space.

Conclusions

These preliminary results indicate that it is necessary to investigate more about TL-efficiency exposed to high energy charged particle beams to obtain the correct dose equivalent for astronauts in particular in view of planning future missions to the Moon and Mars. As expected, TL-efficiency of detectors exposed to iron beam is lower respect TL-efficiency obtained for protons beam at the same energy.

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Intercomparison of various dosimetry systems for routine individual monitoring

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Abstract

According to the studies of EURADOS (www.eurados.org) in addition to film dosimeters thermoluminescent (TLD) and recently radiophotoluminescent (RPL) dosimeter systems are the most widely used for individual monitoring. The situation in personnel dosimetry is permanently changing. For example in the last years because of introducing RPL dosimetry in Europe, the number of film controlled workers was reduced about 50%. In order to reach a high international standard in the application of these dosimeters for routine individual monitoring, there is a need for intercomparisons. Therefore an intercomparison was organised with the participation of six dosimetry services from six countries.

The objectives of the study were to compare and to improve laboratory calibrations, to compare the results of different dosimetry systems at the same irradiation conditions. The task of the participating laboratories was to irradiate their own dosimetry system used for routine individual dosimetry together with RPL dosimeters received from Chiyoda Technol Corporation with a dose $H_p(10) = 0.1$ mSv by narrow spectra X-ray beams of radiation quality N-60. The RPL (GD-450) dosimeters after irradiation were evaluated by Chiyoda Technol Corporation.

The results of this intercomparison indicated that the RPL and the other passive solid state detectors are very suitable for individual monitoring. The establishment of this international intercomparison allowed a broad exchange of experience. As a consequence, the calibration and measuring procedures used in particular laboratories, as well as the quality of the dosimeter systems employed could be improved step by step. We expect that the results of this investigation may encourage the participating services to improve the dosimetry methods and procedures.

Uncertainty of in vivo assessment of actinides activity in human skeleton

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Abstract

In vivo assessment of the actinides activity in the skeleton is valuable source of information on internal contamination of the human subjects. Two different methods, for uncertainty determination, were applied on in-vivo based assessment of skeleton actinides activity performed in National Radiation Protection Institute in Prague. Both methods provide quite comparable results why overall uncertainty depends on measures activity and ranges from 1.23 to 1.65 when expressed in scattering factors.

Introduction

In vivo assessment of the actinides activity in the skeleton is valuable source of information on internal contamination of the human subjects. The proper estimation of the overall uncertainty is highly important, however not simple task because there are various sources of uncertainties. Many of them are connected with a measurement but some of them, like skeletal or bone activity distribution need to introduce assumptions about them. Combining of the uncertainties can by difficult when uncertainty from one source is influenced by other one and thus common rule of uncertainty propagation for independent sources is not applicable.

The presented work tries to estimate overall uncertainty of the method used in National Radiation Protection Institute in Prague for assessing of ^{241}Am activity in the skeleton. Primal step of the process is measurement with two LEGe detectors placed 3 cm of the temporal bones of the skull (Malátová et al. 2000). The recalculation of the skull activity to whole skeleton is done after measurement spectra analysis and skull activity assessment by a ratio.

Material and methods

Estimation of the uncertainty from the long term measurement of subjects with old internal contamination (distanced from the time of intake) can provide some information on stochastic part of the process, like detector placement. Such approach is useful, but it does not carry out any information on systematic errors which could by of mayor terms of overall uncertainty. The most important sources of systematic uncertainty are differences of the real head and calibration phantom in term of activity distribution, size and attenuation, skull to skeleton activity ratio (used for recalculation of measured activity to total skeletal content) and spectra evaluation. Other uncertainty sources, like neck vertebrae, can affect measured counts and thus has to be considered too.

Two methods for uncertainty estimation were used in the paper. First one, IDEAS methodology, recognizes two type of the uncertainty: counting error which follows Poisson distribution (A type uncertainly) and other uncertainties which can by described by log-normal distribution and its standard deviation (called scattering factor –SF). Combining of these two types is possible as shown by Eqn. 1 when type A error is less that third of total uncertainty or relative Poisson uncertainty is less than 30% (Doerfler et al. 2006).

$$SF_i = \exp\sqrt{[\ln(SF_A)]^2 + [\ln(SF_B)]^2} \qquad \text{Eqn. 1}$$

Second method use Mote Carlo approach. All influencing parameters, like activity of the skull, number of the emitted photons during measurement period, detector position etc., are sampled from considered ranges and distributions. Total or marginal uncertainty is obtained from statistical evaluation of aggregated representations with given condition. In the other word: the MC method simulates the estimation process many times and from the sum of results builds distribution (histogram) and estimates its mean and deviation.

Uncertainty

The estimation of the different uncertainty for IDEAS method was done previously in a thesis (Vrba 2007a) and considered values and related assumption are summarized in table 1 side by side with values from Malátová (Malátová et al. 2000).

Table 1. Scattering factors assumed for IDEAS methodology estimate of total uncertainty.

Uncertainty source	Current study		Malátová[a] SF
	method of assessment – note	SF	
voxel phantom composition	MC study density variation ±10%	1.03	Not Applicable
statistical – MCNPX	from MCNPX output file	1.01	
cross -section library MCNPX	Guess	1.05	
micro distribution in head	MC study with inhomogeneous distribution according (Lynch 1998)	1.01	1.35
neck vertebrae correction	a conservative estimate	1.05	N/A
number of counts	Poisson error 10%	1.1	1.1
spectra evaluation	trials with a real spectra, software and manual methods	1.08	N/A
detector position	dislocation of the detector centre: -2 to 2 cm parallel to temporal bone and -1 to 1 cm perpendicularly	1.13	N/A
head to phantom size	inefficient statistic estimate, data obtained from MC study	1.14[c] 1.28[d]	1.38[b]
skull to skeleton ratio	inefficient statistic estimate from the literature	1.16	1.35
Total - known head		**1.32[c]**	**N/A**
Total - unknown head size		**1.37[d]**	**1.72**

a - equivalent value, b - single calibration constant based on physical phantom, c - calibration function with known head size, d - single calibration constant based on voxel phantom. (Vrba 2007b)

The algorithm of the MC uncertainty evaluation is described in figure 1 where sampling of the random variable is expressed by square brackets. The assumptions on variables from unknown distributions were same as for IDEAS approach, if applicable, or chosen in a conservative way.

Displacement of detectors, expressed by a centre of their windows, were sampled uniformly in region from -2 to 2 cm in plane parallel to the temporal bone and from -1 to 1 cm perpendicularly. Uniform distribution of the activity in the skull ranged from 3 to 100 Bq was chosen in order to cover activity interval from minimal detectable activity (MDA) to the region where the uncertainty from the counting is marginal.

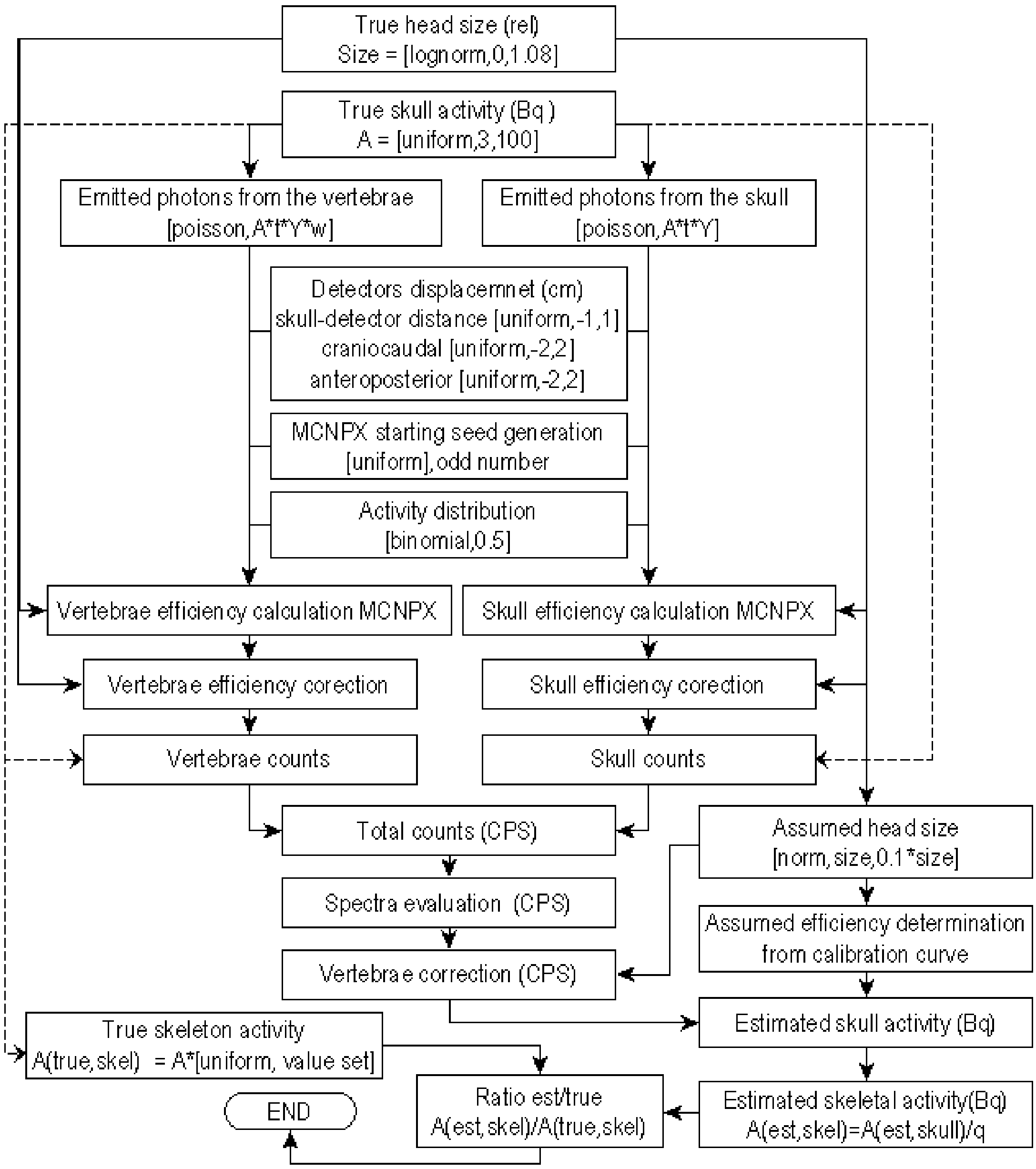

Fig. 1. Monte Carlo algorithm used to estimate uncertainty of americium skeleton activity assessment.

Skull size comes from lognormal distribution with SF 1.08 when maximal and minimal head size were limited according known head size (equivalent to hat-size 44 and 67 cm respectively). Finally skull to skeleton activity ratio was taken from uniform discrete distribution containing four known values based on USTUR whole body cases (Lynch et al. 1988). The distribution of the activity in the bones changed in time, due to bone remodeling and activity redistribution, and is quite unknown. The binomial distribution with equal probability of volume and surface activity placement was anticipated in order to eliminate underestimation of the effect.

There were not taken in account uncertainty from limited description of particle transport in MCNPX and applied cross-section libraries since there are no date on it and the expected uncertainty is small. The limitation of the phantom model was not covered either because modeling is rather difficult. The calculation of the registered photons with energy of 59.6 keV in the detectors was performed via MCNPX ver. 2.5 or 2.6, when counts from the skull and vertebrae were determined in separate simulations. The correction of the count in the bin representing total absorption had to be done because scaling of the phantom voxel side changes outer layers thickness covering the skull. Total number of MCNPX simulations pair (skull, vertebrae) was 10000 in order to have reasonable statistic even in a case when subset whit specific parameters are assumed.

Relative error of spectra evaluation is a function of the counts intercepted in the detectors. Dependency of the error was estimated by a power function which fits graph data points where gross peak area was on X axis and relative errors of the measurements on the Y axis. The sampling of spectra uncertainty takes corrected number of counts, determines relative error from the function and uses it as a parameter of the random number generator (lognormal distribution).

Results

Overall uncertainty estimated with IDEAS methodology, expressed by the scattering factor, is 1.32 if head size is known or 1.37 when head size is unknown. MC method estimates overall uncertainly almost of the same magnitude SF=1.35, when calculation cover all parameters in their full ranges and detection efficiency is corrected on head size. There is slight difference in total uncertainty when only surface (SF = 1.31) or volume (SF = 1.35) distribution in the bones tissue is concerned. When constant detection efficiency is assumed (like in the past) overall uncertainty increase to SF=1.42. Uncertainty (SF) as the function of the skull activity and source distribution is given in table 2.

Table 2. Uncertainty of skeleton activity expressed in SF.

Activity (Bq)		Considered source		
rage	mean	Surface	volume	both*
3-20	11.8	1.54	1.65	1.61
50-100	74.4	1.23	1.24	1.27
3-100	51.6	1.31	1.35	1.35

* Surface and volume sources represented with same probability

The MC approach evaluates uncertainty due neck vertebrae in more accurate way than simple assumption given in table 1. Neck vertebrae add a signal in detectors and shifts activity estimate to higher one if there is no correction. The estimated uncertainty due to vertebrae correction is about SF = 1.025 when an effect of activity distribution in bones can by neglected.

The approximation of the overall uncertainty distribution by lognormal one, proposed in IDES methodology, looks very reasonable because all MC histograms exhibit such character as is demonstrated in figure 2.

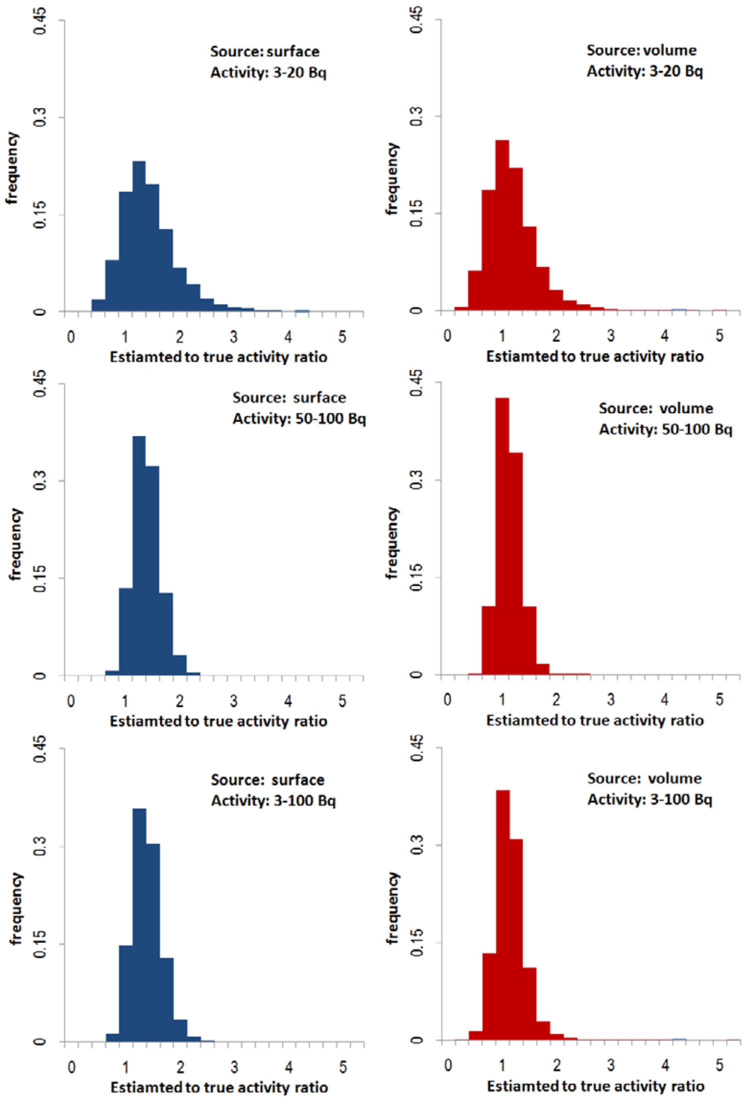

Fig. 2. Histogram of estimated to true activity ratio for different source distributions and skull activities.

Discussion and conclusion

Results of MC and IDEAS method are in reasonable agreement in a term of SF value, with regards to different approach. Distributions of the assessed activity to true one from MC approach exhibit lognormal character which is in coherence with IDEAS method. The MC estimate of total uncertainty does not cover some aspect, like difference of the tissue in the real head and phantom, however their influence is small compare to other sources. The main limitations of presented results are assumption used in order to estimate unknown parameter distributions. Usage of uniform distribution for detector positions and true skull activity is maybe too conservative; however it is not reasonable to introduce more complex assumption on them without some reliable information.

Uncertainty estimates carried out from the current study are lower than ones given in the paper by Malátová (Malátová et al. 2000). The main difference, between these two study is: more realistic estimates of same uncertainty sources (e.g. inhomogeneous americium distribution in the skull), including of some new uncertainties sources (detector position), introduction of calibration function, based on simulation with voxel phantom. Usage of the efficiency curve from the simulations reduced possible error because some physical voxel phantoms used in Malátová et al. are imperfect and thus introduces larger uncertainty.

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Quality assurance applied for intake estimation in a case study of I-131 ingestion using in-vivo assessment data

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Abstract

Human Body Monitoring Laboratory (WBML) from IFIN-HH, Bucharest, Romania, performs internal contamination IN VIVO assessments, respecting its own Quality Assurance System, agreed from the year 2000 and updated in accordance with the requirements of the standard SR EN ISO/CEI 17025:2005 and other specific national and international standards, regulations and guidelines[1-6].

The method implemented for I-131 monitoring in thyroid was applied in a case of an internal contamination through ingestion. There were performed several thyroid assessments using a Thyroid Counter equipped with a lead shielded NaI(Tl) detector for the thyroid, with an efficiency of 0.9×10^{-2} counts/Bq*s and MDA of 34 Bq for I-131.

The results obtained after the selection of the measurement data showed a good compliance of them with the model considered for the retention of the iodine in the thyroid, specified in ICRP78.

Introduction

The WBML is notified to execute: (1) the monitoring of internal contamination for workers involved in nuclear activities from hospitals, industry, research, national security, with potential risk of internal exposure to ionizing radiation, and in case of nuclear accident, for persons belonging to the public; (2) the estimation of gamma emitting radionuclide intakes; (3) the estimation of committed effective doses

The radionuclides identified during the assessment of the workers were Na-22, Co-60, Tc99m, Zn-65, I-125, I-131, Cs-137 and Ir-192, specific to different types of nuclear activities, as production of radiopharmaceuticals, production of sealed sources, nuclear medicine, cyclotron maintenance and research activities.

During a routine monitoring of workers, one case of internal exposure revealed to be special due to the fact that it was not an occupational intake, but the result of the ingestion of a capsule of I-131 used in the diagnosis of a thyroid disease. The value of intake was known by the physician that prescribed the diagnosis procedure, but unknown to the Whole Body Monitoring Laboratory. Because it was possible, there were performed eight thyroid assessments, at different intervals after intake, to see the compliance with the expected decay for the I-131 in the thyroid.

Material and methods

The system used in WBML for the internal contamination assessment with I-131 was a Thyroid Counter, a gamma spectrometer, equipped with lead shielded NaI(Tl) scintillation detector of 40mm diameter, and 50mm thickness.

The efficiency calibration was performed with a plexiglass thyroid phantom, simulating the adult thyroid anatomical shape and volume, filled with certified radioactive solution of known activity of I-131. The values determined for the efficiency and for the Minimum Detectable Amount were of 0.9×10^{-2} counts/Bq*s and of 34 Bq for I-131 (364keV), respectively.

The associated electronics of detectors consists on state-of-art analog and digital ORTEC equipments and for spectra acquisition is used the software Gamma Vision 32 supplied, also, by ORTEC.

The estimation of the intake and of the effective dose was made following the steps suggested in the new document, FZKA 7243, General Guidelines for the Estimation of the Committed Effective Dose from Incorporation Monitoring Data, for acute ingestion. All these steps are parts of different stages of intake and effective dose evaluation implemented in a flowing chart that establishes, clearly, any action to be done for an optimum estimation of the intake and of the committed effective dose.

The stages followed were:

Stage1. Check for need of evaluation

Stage2. Check on significance of a new measurement and consistency with previous evaluation

Stage3-4. Identification of pathway of intake for special evaluation

Stage6. Special procedure for ingestion

The evaluation procedure of the intake and effective dose is not very simple because the monitoring values indicate an important internal exposure and there are necessary many steps of evaluation, just to be sure that it is applied the quality assurance of data for reliable results.

Results

The thyroid retention predicted values (Bq per Bq intake) following an acute ingestion of I-131 were considered those from the ICRP 78.

Stage1. It was identified the first monitoring value and determined M_C = 3.8Bq, the critical monitoring quantity.

Stage2. (i) It was determined the order of magnitude of the intake and of E(50) from the first measurement data.

Intake = 3397143 Bq

E(50) = 73.75 mSv

(ii) There were considered realistic estimates of the overall uncertainty on each data point expressed as a total scattering factor, SF, taking into account the total uncertainties of type A and type B.

Stage3-4. It was decided that it was necessary to perform a special monitoring for a case of pure ingestion.

Stage6. There were analyzed the monitoring data to see the compliance with the iodine model for ingestion declared in ICRP78 and to look for outliers.

The monitoring data and the trend of data considering the thyroid retention predicted values (Bq per Bq intake) following an acute ingestion of I-131 from the ICRP 78 were represented in Fig.1.

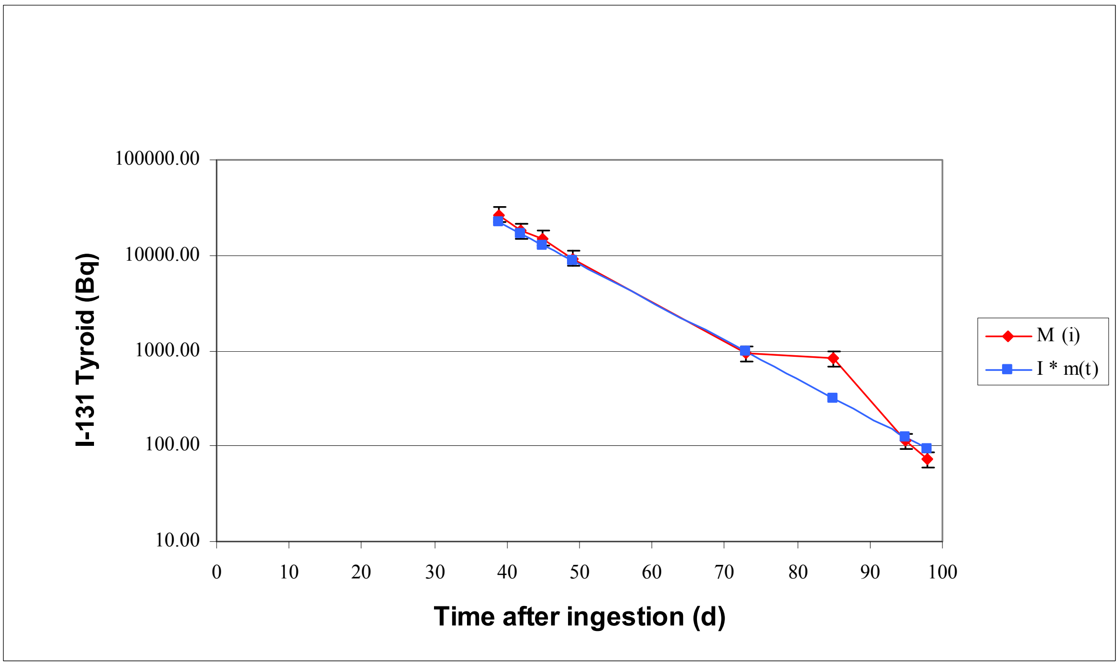

Fig. 1. I-131 Thyroid measurements.

One measurement data was found to be outlier and it was eliminated from the subsequent evaluation.

Considering the remained 7 monitoring data, they were processed for intake and effective dose estimation in 2 steps:

– the first 3 data, considering that they were obtained all in an interval of 7 days
– all the 7 monitoring data , considering that the high effective doses require more data for the evaluation. The results obtained are presented in Table 1.

Table 1. Estimated Intake and committed effective doses, E950) for 2 selection of monitoring data.

Nr.of processed data	Estimated Intake(Bq)	Real Intake (Bq)	E(50) mSv	Error (%) (absolute value)
First 3	3234628	2960000	71.16	9.3
7 (without outlier)	2859570	2960000	62.9	3.4

To qualify the fit of the measurement data, it was performed the evaluation of the observed chi-square. The values of 1.16 for $\chi_0^2(I)$ and of 0.88 for P-value certified us to consider that, there is a good fit of data.

Conclusions

The results obtained showed a good compliance of the measurements data with the model for iodine from ICRP78, fact that emphasizes the good quality of processed monitoring data. Moreover, a greater number of good monitoring data reduces the error of intake estimation. This gives us the certitude that the implemented method for the intake and dose estimation is reliable and valid.

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Internal dose formation in rural society after the Chernobyl accident

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Abstract

Dose is formed in person due to the personal psychological, physical and social characteristics and economic status during his practical activities interacting with polluted environment. These actively promotes consumption rate of the contaminated food stuffs, thus the dose is the property of the person.

Every person is a member of a family. A family is the same society, but concerning settlement it is a small one. Moreover, family is a social complex system, including certain individuals being in direct contact for a long time, united by connections and by the nature mutual relations. Every family as a social system determines actions of its individuals and in certain situations acts in relation to environment as a single whole.

The WBC-measurement Data Base for the period of 1989 – 2008 contains above 2 million records, which allow studying the internal dose peculiarities.

The provided analysis of internal dose distribution for inhabitants of rural settlements for 10-year period confirmed the hypothesis that every individual and also every family have their own certain place on a dose distribution curve and it is constant in time. In other words the same percentile value of dose distribution corresponds to certain person and certain family. These inhabitants have not only different doses but also different personal, social and economic characteristics.

Finally, the results of the research will allow predicting doses with enough high degree of accuracy in certain persons for any year using their own relative doses for one or several years or by known relative doses in members of their families.

This law can be used as a basis for methodological approach to reconstruction of individual doses for the subjects for practically any time period of the accident, and that is important for determining of individualized doses for people included into the Belarus State Registry of the Chernobyl affected persons and estimation of individual doses in epidemiological studies.

Introduction

The internal dose of the person is formed due to consumption of foodstuff contaminated by radionuclides and the consumption rate is defined by its personal characteristics (psychological, physical) and the social and economic status (professional occupation and a social status). The person during his practical activities interacting with contaminated environment actively promotes dose formation which is his property (Skryabin Vlasova 1995). Based on this fact, the distribution of inhabitants in a settlement by doses is not random. It reflects the attitude of certain individuals to radiation danger factor. This attitude also forms "food" behavior of people. Food habits of certain persons are formed depending on their perception of radiation danger factor. Therefore the attitude of certain persons to radiation danger is connected with their personal characteristics, such as gender, an educational level, psycho-emotional status (Skryabin 1997).

Under the same external conditions one refuse consumption of the 'contaminated' foodstuff, others act on the contrary. Internal dose distribution in the settlement is defined by personal characteristics of each of the inhabitants. Therefore obviously each person takes certain place, constant in time on a dose distribution curve (Vlasova 2007).

Besides each individual is a member of a family. Family is a social system which is compound organized, the ordered whole, including individuals united by connections and mutual relations, specifically social by the nature. Each family as the social system to some extent determines actions of individuals included into it and in the certain situations acts towards an environment as a single whole (Vlasova Stavrov 2005).

Within the framework of a family the direct consumption of foodstuffs is provided. This is preceded with formation of corresponding psycho-emotional perception of radiation danger factor. And apparently the direct factor of internal dose formation as a consumption rate of the contaminated food stuffs such as, for example, milk, forest mushrooms, game, is defined by a number of the indirect factors connected with social- demography-economic characteristics of a family. In this way, every family should be characterized by the dose. As the family forms a dose so the dose in it is a property of a concrete family (Vlasova Stavrov 2005). Thus, dose distribution in a settlement is defined also by characteristics or features of each family in a settlement. Therefore, obviously, each family, similarly to each individual, on dose distribution curve takes the certain place which is constant in time.

Really, analyzing internal dose distributions of rural settlements in dynamics for the 10-year period it was noticed that at certain persons and members of the families having numerous measurements during one and some years, relative doses are identical, in other words, the same value of a certain percentile of dose distribution in a settlement corresponds to the individual.

Material and methods

For conducting the study there was selected the settlement Kirov in the most contaminated Narovlya district of Gomel region located closely to a forest. As our studies showed the "forest" factor plays an important role in formation of internal dose caused by consumption of forest berries, mushrooms and game (Skryabin et al. 1995).

There were used data on internal doses calculated by WB-measurements on cesium content in the body of inhabitants of a settlement Kirov for 10-year period of 1990-1999, contained in the database of WB-measurements of the State Dosimetry Registry.

The methods of applied statistics as analyses of variance, multiple-factor statistical analysis were used. Statistical processing of materials was performed with the help of a package of statistical programs "STATISTICA 6.0".

Results

There were analyzed the data representative samples estimated by results of WB-measurements, internal doses for 10 years on the selected settlement Kirov. There were revealed the inhabitants having relative doses identical in time; the dose of each of them corresponds to the certain value of percentile dose distribution per every year. The descriptions of some of them are shown in the table 1.

It is necessary to note that at people distributed by this way through dose distribution` percentiles have not only doses which differ, but also different individual personal, social and economic characteristics.

Table 1. Stability of Relative Ingestion Dose at Individuals (the percentiles of Ingestion Dose Distribution), Kirov.

Name	Gender	BY	Occupat.	Years									
				1990	1991	1992	1993	1994	1995	1996	1997	1998	1999
Yu.T.	F	1976	pupil		16%		5%		30%			22%	
D. L.	F	1957	medicine				40%	25%	25%	19%		21%	38%
R. L.	M	1939	worker	90%	92%	98%	89%		93%	82%	96%		78%
P. I.	M	1909	pensioner	93%			98%				99%		

By method of the one-way analysis of variance there were allocated 2 seasons: spring - summer: March, April, May, June, July; autumn - winter: August, September, October, November, December, January, February (rural inhabitants use dry mushrooms in the winter prepared in the autumn) for which average internal doses significantly differ by average value (Vlasova 2007). Results are shown in table 2 and in figures 1-2.

Table 2. Comparison of Ingestion Dose Distribution Parameters for the Seasons of a Year, Kirov

Year	Season	N	16% Percent.	25% Percent.	Medi-an	Aver-age	Stand. Error	75% Percent.	84% Percent.	SGD
1990	Aut - Wint	47	0,20	0,24	0,59	1,04	0,29	1,10	2,18	3,70
1991	Aut - Wint	36	0,07	0,11	0,29	0,98	0,16	1,06	2,28	7,79
1992	Spr – Sum	47	0,06	0,10	0,19	0,36	0,09	0,39	0,52	2,75
	Aut - Wint	138	0,46	0,35	0,88	1,66	0,19	1,90	2,43	2,75
1993	Spr – Sum	234	0,44	0,62	1,10	1,78	0,14	1,97	3,38	3,08
	Aut - Wint	121	0,40	0,63	1,84	3,05	0,35	3,30	6,24	3,38
1994	Spr – Sum	92	0,04	0,18	0,40	0,84	0,12	0,87	1,51	3,73
	Aut - Wint	78	0,21	0,29	0,51	1,19	0,28	1,27	1,64	3,24
1995	Spr – Sum	243	0,12	0,19	0,37	0,72	0,05	0,90	1,37	3,72
	Aut - Wint	75	0,15	0,22	0,48	1,03	0,20	1,10	1,75	3,60
1996	Spr – Sum	176	0,17	0,26	0,50	0,88	0,09	1,16	1,39	2,76
	Aut - Wint	316	0,28	0,43	0,86	1,53	0,11	1,82	2,66	3,07
1997	Spr – Sum	162	0,11	0,15	0,32	0,65	0,08	0,67	0,97	3,06
	Aut - Wint	172	0,38	0,55	1,49	2,16	0,16	3,19	4,14	2,77
1998	Spr – Sum	211	0,12	0,17	0,56	1,12	0,11	1,46	1,97	3,49
	Aut - Wint	265	0,20	0,36	0,91	1,66	0,14	2,10	2,94	3,23
1999	Spr – Sum	212	0,21	0,27	0,53	0,85	0,07	1,00	1,37	2,90
	Aut - Wint	90	0,23	0,31	0,58	1,05	0,14	1,17	1,68	2,57

* SGD – standard geometrical deviation of the distribution: ratio of 84%- percentile to 50%-percentile or median.

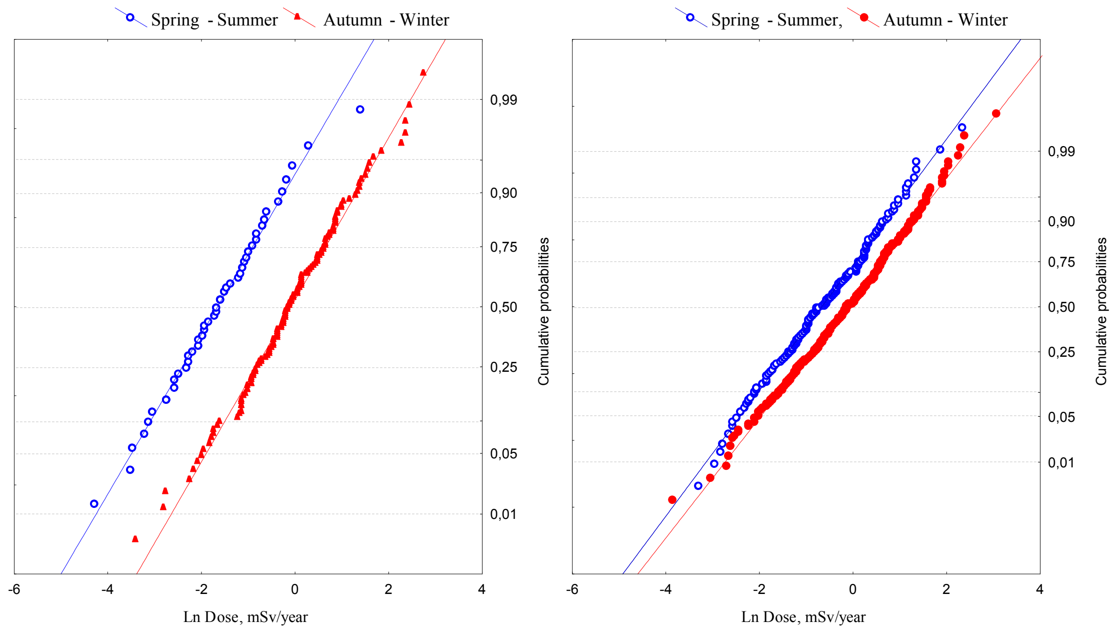

Fig. 1. Integral Internal Dose Distribution of Inhabitants of a Settlement Kirov, 1992, 1996 2 seasons of a year

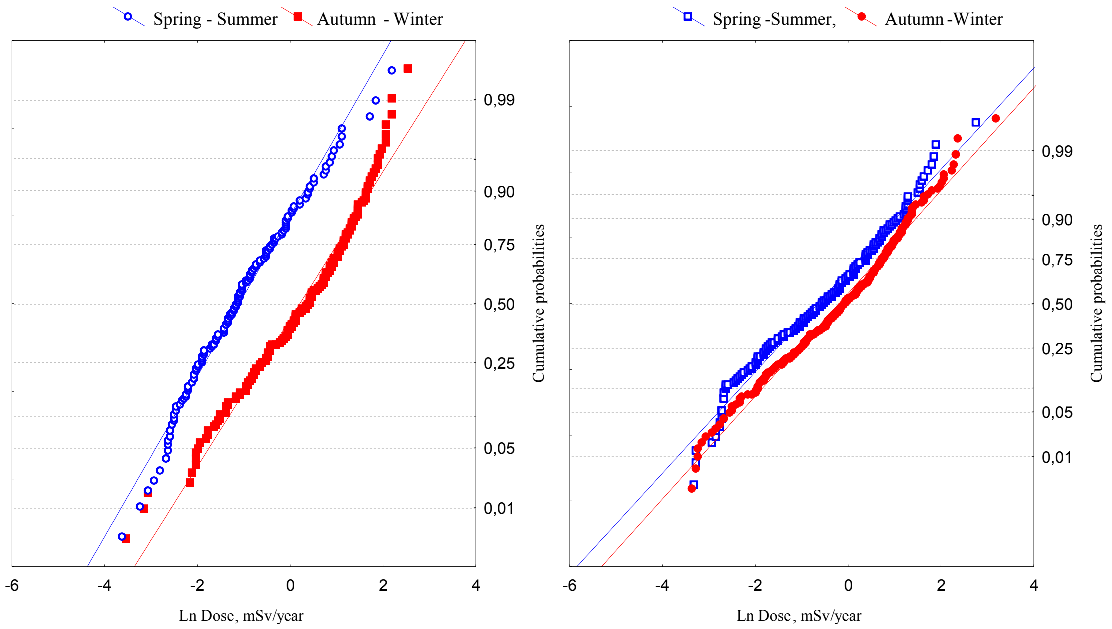

Fig. 2. Integral Internal Dose Distribution of Inhabitants of a Settlement Kirov, 1997, 1998 2 seasons.

Thus, in view of internal dose distributions on two samples of inhabitants of a settlement Kirov, corresponding to two seasons, by each year, for people and families (and members of families), having corresponding dose distributions percentiles (relative internal doses) on settlement per year as a whole (see tables 1, 2), there were identified the values of dose distribution percentiles of corresponding seasons of each of the years. The description is shown at table 3.

Table 3. Stability of Relative Ingestion Dose in Individuals (the percentiles of Ingestion Dose Distribution) Kirov, Autumn-Winter & Spring-Summer

Fam-ily	Gen der	BY	Occup.	Years Autumn-Winter & *Spring-Summer*									
				1990	1991	1992	1993	1994	1995	1996	1997	1998	1999
Yu.T.	F	1976	pupil		16%		21%		*31%*			*46%*	
D. L.	F	1957	medicin				*39%*	21%	*27%*	*14%*		*41%*	48%
R. L.	M	1939	worker	90%	92%	83%	*93%*		93%	*77%*	96%		*80%*
P. I.	M	1909	pension	93%			*87%*				99%		

Studying the individual characteristics and parameters of internal dose distribution in classes of the same behavior of inhabitants in Kirov settlement it was revealed that in each class there are the persons forming families (Skryabin Vlasova 1999). There was found to be a considerable share of such families in each class (see table 4).

Table 4. Family structure in classes of homogeneity.

Class №	Share of Inhabitants forming families, %	Average value				Head of a family		
		Internal Dose	Number of family members	Relative index of OAG*of a family	Family educational relative level	Share of workers, %	Share of female, %	Average educational relative level
1	72	0.34	2.52	1.36	1.07	30	30	1.10
2	62	0.74	2.60	1.37	1.00	42	24	0.95
3	57	1.15	2.65	1.39	0.88	47	27	0.87
4	51	1.75	2.70	1.46	0.93	59	7	0.96
5	69	5.48	2.78	1.51	0.80	70	4	0.80

* Using the analysis of varience there were distinguished two occupational-age groups (OAG) for which the average internal doses significantly differ. Structure of OAG1 included employees, housewives, female pensioners, handicapped persons and children; structure of in OAG2 included machine-operators, drivers, workers, foresters and male pensioners (Skryabin 1995).
Relative index of OAG of a family is the average value of the corresponding values of factor.

As evident from the data of Table 4, the average number of family and relative indicator of family OAG increases with growth of the dose from class to class; the average relative educational level of family decreases, the share of working heads of the family significantly increases, a share of women among them decreases, the relative educational level tends to decrease.

The family analysis of internal doses – classification of families of Kirov settlement, had been conducted. The methodology of family analysis of internal dose formation is similar to the corresponding one of the individual analysis (Vlasova Stavrov 2005): the multiple-factor statistical analysis, i.e. classification of objects by set of informative signs.

As a result of multidimensional classification of families in Kirov it was received 10 non-crossed classes giving full enough conception about a variety of types of families. In Table 5, the signs and statistical parameters of distributions of average-family and pseudo-collective doses for the whole sample and on the obtained classes are presented.

Average values and medians of dose distributions as avarage-family ones, as well as the pseudo-collective ones, essentially differ in classes. SGD values of dose distributions in each class are low enough in comparison with the whole sample that is obvious from data of Table 5 and Figure 3. It testifies to homogeniety of classes by dose and so to adequacy of the classification.

Table 5. Description and Statistical Parameters of Pseudo-Collective Family Doses and Average Family Doses Distributions by Family Classes in a Settlement Kirov.

Family number in class	Family' Index						Family Leader				Average Family Dose			Pseudo-Collective Family Dose			
	Number of members	Avg age	Education	OAG	Number of children	Contact with the forest	Age	Education	Occupation	Share of men, %	Average	Median	SGD	Average	Median	SGD	Relat. Range %
											mSv/year			mSv/year* man			
20	4.00	30.4	4.1	1.33	1.65	0.45	44.6	4.1	1.7	45	0.58	0.55	1.45	2.34	2.08	1.55	18
31	2.00	53.9	4.8	1.42	0.07	0.55	55.7	4.8	2.1	74	0.70	0.75	1.28	1.41	1.52	1.29	22
18	3.56	30.4	4.4	1.60	1.11	1.56	43.9	4.6	2.6	89	1.53	1.61	1.17	5.41	5.21	1.25	30
31	2.00	57.2	5.3	1.55	0.03	1.61	59.9	5.3	2.4	74	2.54	2.42	1.47	5.08	4.84	1.53	26
16	4.06	34.0	4.5	1.80	1.19	1.56	39.4	4.5	2.8	87	2.80	2.68	1.36	11.2	9.42	1.56	19
10	1.90	62.2	5.8	1.60	-	2.80	59.9	5.7	2.2	90	6.73	6.71	1.15	12.8	13.0	1.15	18
6	3.83	27.7	4.2	1.92	1.50	2.50	44.0	4.5	3.0	83	8.96	9.24	1.25	35.5	30.6	1.72	22

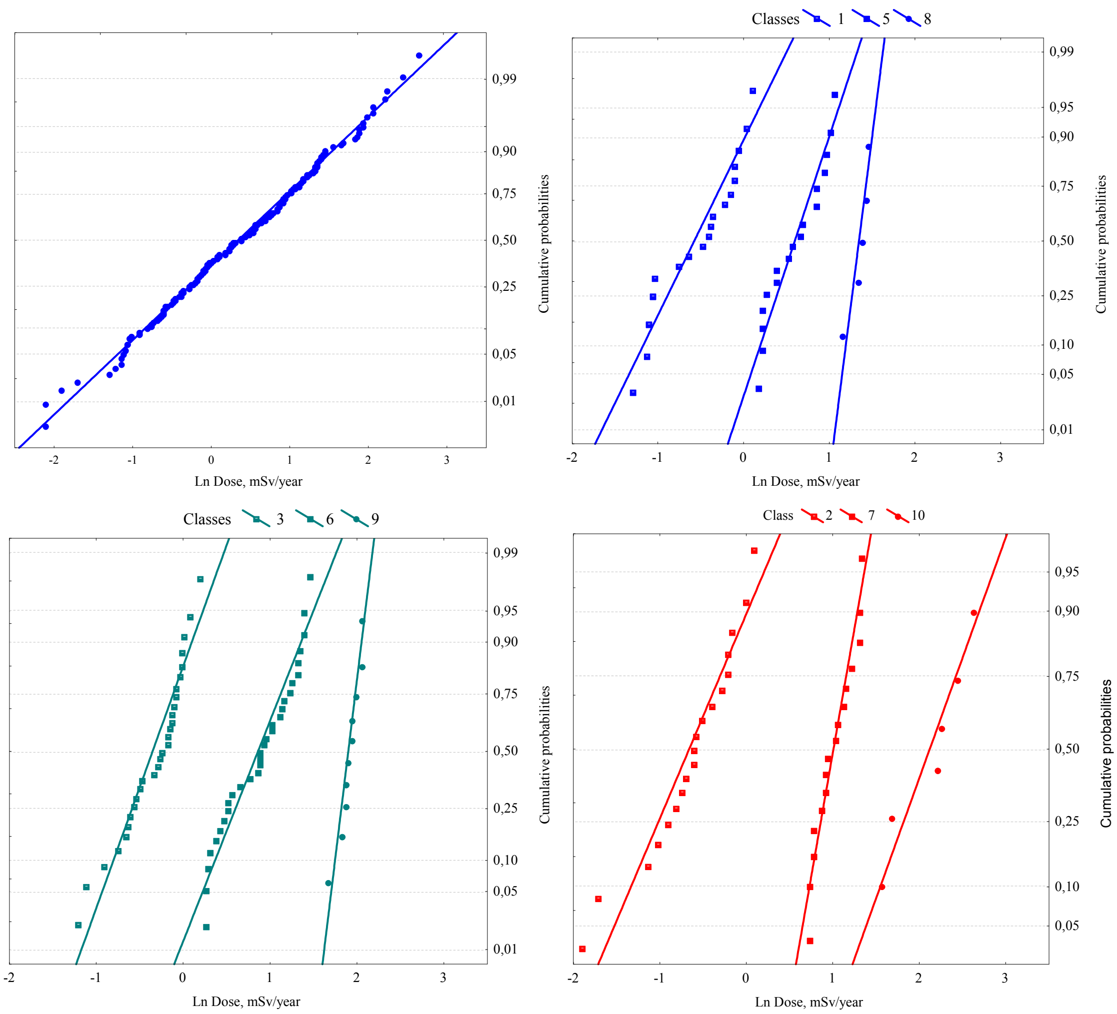

Fig. 3. Integral distribution of average-family dose of population of Kirov settlement in whole and by family classes.

The average relative scope of doses in the classes of families of Kirov settlement makes up 22 % on the average as in classes of small as well as of big number of families. That testifies to uniformity of food behaviour not only within the limits of a family, but also a class of families, that once again confirms the thesis about the defining role of a family as social system in dose formation.

Obviously, family’ classes are more significant not only by professional and forest signs, as it was for individual’ classification, but also by number of the family members, age, education, i.e. and by its social and demographic characteristics. The family’ analysis gives more clear conception of the mechanism of internal dose formating among inhabitants of rural society, at the same time it supplements the individual one. Based on the results of the family’ analysis one can adequately estimate or predict dose distribution not only in separate groups but also in settlement as a whole. The family analysis together with the individual one can form a reliable basis for revealing of the most exposed so-called “critical” groups of rural society.

ICRP recommends the following concerning critical group: “It is often convenient to class together individuals who form a homogeneous group with respect to their exposures to a single source. When such a group is typical of those most highly exposed by that source it is known as a critical group” (ICRP 1990). Following definition by ICRP, group’s criterion is homogeneity by dose.

It was noticed that dose distribution in a settlement is a mixture of logarithmetic normal distributions, each of which corresponds to a group of persons conducting the same way of life. If critical group is homogeneous by dose than a dose range is small enough, i.e. SGD of dose distribution in is low.

The analysis of dose distributions in settlements has shown that in the majority of them the homogeneous group “distant” from basic empirical distribution singled out in the “tai” of the distribution as it is shown in Figure 4. It is a critical group. Splitting of distribution into two groups, basic and critical, was performed by the corresponding value of percentile of the distribution (90 percentile in Figure 4). SGD of dose distribution in the critical group in comparison to the basic group is low (see Figure 4), which testifies to high degree of homogeneity.

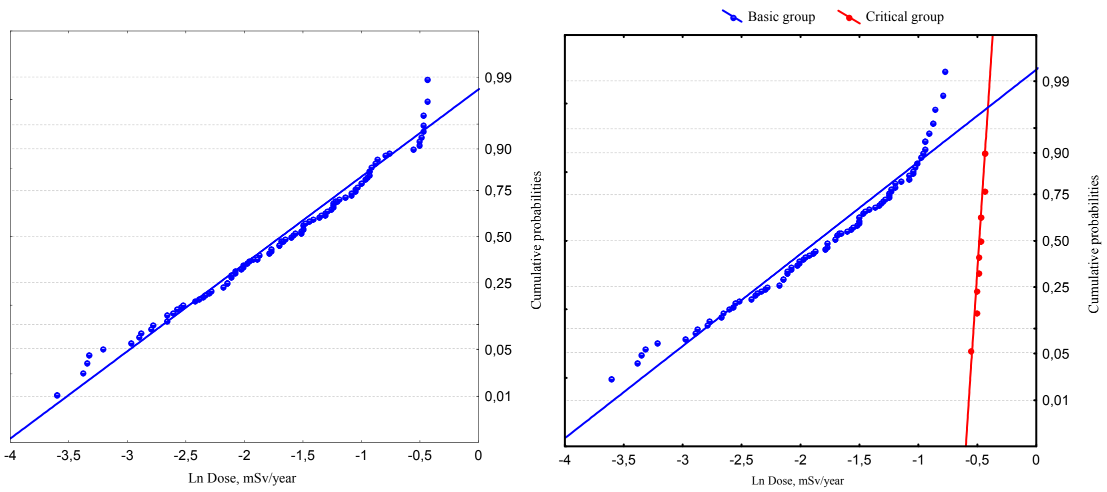

Fig. 4. Internal dose distribution of population of Neglubka settlement of Vetka district. On the left: whole settlement sampling, on the right: basic group and critical group 10% in size.

Probably, the critical group is the group of families with the same "food" habits causing the highest internal doses.

There were revealed families which members have identical in time relative doses i.e. the dose of each of them corresponds to the certain value of dose distribution percentile per every year, also and by the seasons of a year. The descriptions of some of them are shown in the Tables 6 and 7.

Table 6. Stability of Relative Ingestion Dose in Families (the percentiles of Ingestion Dose Distribution), Kirov.

Family	Gen-der	BY	Occup.	Years								
				1990	1992	1993	1994	1995	1996	1997	1998	1999
Bir-s	M	1949	driver			92%	92%		71%	68%		90%
	F	1955	teacher			97%	92%	70%			92%	98%
	F	1977	employer				84%					90%
	M	1981	pupil			98%		55%		96%	91%	99%
	M	1983	pupil		87%	99%		66%		99%		
Alex-os	F	1973	householder					75%	74%	88%		
	M	1962	driver		71%	69%		95%	87%	99%	90%	
	M	1969	worker		72%				78%			
	M	1992	child					79%	43%	33%		
	M	1991	child					73%	60%	54%		
An-os	M	1943	driver	83%	94%			89%		94%		
	M	1936	driver	89%		91%		90%			87%	
	M	1969	driver		96%	92%				98%		
K-s	M	1930	forest work	99%						99%		
	F	1938	pensioner	85%	89%	93%					90%	
Kov-s	F	1985	pupil	28%		23%		21%				
	M	1986	pupil	15%		28%		18%				
Atr-os	M	1949	driver			43%		56%	9%	46%		
	F	1965	worker			43%		26%	23%		20%	
	M	1981	pupil			21%	6%		9%	27%	9%	
	F	1983	pupil			25%	5%	2%	1%			

Table 7. Stability of Relative Ingestion Dose in Families (the percentiles of Ingestion Dose Distribution), Kirov, Autumn-Winter & Spring-Summer

Fami-ly	Gen-der	BY	Occup.	Years ***Autumn-Winter*** & ***Spring-Summer***								
				1990	1992	1993	1994	1995	1996	1997	1998	1999
Bir-vs	M	1949	driver			**84%**	**94%**		***64%***	***89%***	***68%***	**92%**
	F	1955	teacher			**97%**	***91%***	**72%**			***94%***	**98%**
	F	1977	employer				**78%**					**92%**
	M	1981	pupil			**95%**		**56%**		***96%***	**90%**	**99%**
	M	1983	pupil		**79%**	**100%**		**68%**		***99%***		
Alex-s	F	1973	householder					**77%**	**68%** ***68%***	**78%**		
	M	1962	driver		**52%**	**64%**		***96%***	***87%***	***99%***	**88%**	
	M	1969	worker		**53%**				**85%** ***73%***			
	M	1992	child					**76%**	**43%**	***33%***		
	M	1991	child					**75%**	**61%**	***56%***		
An-s	M	1943	driver	**83%**	**91%**			***91%***		***95%***		
	M	1936	driver	**89%**		***95%***		***93%***			**84%**	
	M	1969	driver		**86%**	**72%**				***98%***		

Table 7. Continuation.

Fami-ly	Gen-der	BY	Occup.	Years *Autumn-Winter* & *Spring-Summer*								
				1990	1992	1993	1994	1995	1996	1997	1998	1999
K-s	M	1930	forest worker	99%						93%		
	F	1938	pensioner	85%	74%	88%					92%	
K-s	F	1985	pupil	28%		20%		*20%*				
	M	1986	pupil	15%		25%		*21%*				
Atr-s	M	1949	driver			*48%*		58%	*7%*	26%		
	F	1965	worker			*48%*			*19%*		28%	
	M	1981	pupil			*20%*	*9%*	27%	13% *6%*	*43%*	10%	
	F	1983	pupil			*26%*	*5%*	2%	*1%*			

Obviously, it is not casual coincidence and it is a law, and thus, the last can have practical significance.

Discussion

The results of the present research will allow predicting doses with enough high degree of accuracy at persons for any calendar year (the time period) on their relative doses for one or any several years, or by known relative doses at members of their families.

It can be used for reconstruction of individual doses at a lack of dosimetric information.

It is necessary to have the information on parameters of dose distribution in a settlement for predicting or dose reconstruction at separate subjects: for example, average arithmetic or median.

Conclusions

There was confirmed the hypothesis about the stability of a relative internal dose of individuals and the members of families of rural society.

This approach can be a methodological basis for reconstruction of individual internal doses of concrete people of any time period of the accident including initial one what extremely is important at present: namely, for filling the Belarus State Registry of the Chernobyl affected people by individualized doses, and a retrospective estimation of individual doses for radiation-epidemiological studies.

Also this approach can be used for taking into account a contribution of cesium ingestion dose to individual thyroid ingestion dose.

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Determination of the main pathway of the ^{131}I intake to the urban residents of Belarus following the Chernobyl accident

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Abstract

Relationship between ingestion intake with milk and inhalation intake of ^{131}I for the residents of large cities after the Chernobyl accident depended on the ^{131}I contamination level of fresh milk distributed in the city trade network. This milk was delivered from different regions with various level of ^{131}I fallout. The leading contributor to the human intake might be ingestion intake (if milk was delivered from areas with higher or similar level of the ^{131}I fallout compared to that in the city) or inhalation intake (if milk was delivered from areas with less level of the ^{131}I fallout compared to that in the city). Selection of a wrong leading route of iodine intake might result in systematic error in individual thyroid dose calculation, which can be as high as a factor of four to five.

A method has been developed allowing for objectively determination of the main pathway of the radioiodine intake (inhalation or ingestion with milk) for the residents from one settlement. The idea of the method is as follows. The dependence of the variation with measurement date of the average estimate of the ^{131}I thyroidal content for adults and for children from the same settlement in case of inhalation radioiodine intake is significantly different from that in case of ingestion intake. Comparison of such dependency derived from actual results of the thyroidal iodine measurements with two theoretical curves allows for making a decision which pathway of two possible radioiodine intakes was the main contributor to the thyroidal exposure for the residents from a given settlement.

This method was applied to the results of in vivo thyroid measurements of the residents of two large cities in Belarus (Minsk and Gomel) following the Chernobyl accident. The analysis of those results allowed for derivation of an important conclusion that the intake of ^{131}I with milk was the main contributor to the thyroidal exposure for the residents of the cities of Minsk and Gomel.

Introduction

The accident at the Chernobyl nuclear power plant on April 26, 1986 resulted in internal exposure to thyroid from radioiodines for many people over the world. The highest exposure to radioiodines was delivered to the populations of Belarus, Ukraine, and Russia. Intensive efforts on assessment of exposure to thyroid from ^{131}I, the main contributor to the thyroid dose, for populations in these countries were done. An important part of those efforts was large-scale measurements of gamma radiation using detectors placed against the neck. Within a few weeks after the accident those measurements (called "in-vivo measurements" or "direct thyroid measurements") were performed for more than 300 thousand people in Belarus (Gavrilin et al. 1999), Ukraine (Likhtarev et al. 1996), and Russia (Zvonova and Balonov 1993, Stepanenko et al. 1996).

To assess individual dose to thyroid from ^{131}I for those people it is necessary to determine the time-dependent intake function of ^{131}I. This intake function depends upon the routes of radioiodine intake. Three pathways of intake of ^{131}I are usually considered for the population after the Chernobyl accident: (1) fresh milk, (2) contaminated air, and (3) leafy vegetables. According to the interviewing conducted in the most contaminated areas in Gomel and Mogilev Oblasts in 1988 the majority of the Belarusian residents began to consume leafy vegetables in middle and late May 1986 (Gavrilin et al. 1999). Thus, in general, the consumption of leafy vegetables did not provide any substantial delivery of ^{131}I to those residents. So, the leading route of the ^{131}I intake should be identified from the two possible pathways due to (1) inhalation of contaminated air and (2) ingestion of contaminated milk.

It is worth noting that there is a general agreement that for rural populations permanently living in contaminated settlements the main route of the radioiodine intake was consumption of cows' milk locally produced rather than inhalation of contaminated air. For urban populations consuming milk from shops the situation with determination of the main route of the radioiodine intake is more complex and depends upon local conditions, for example, areas from which milk was delivered to the city trade network, when and at what size countermeasures aimed at decreasing or cancellation of radioiodine intake with foodstuffs were introduced, etc. The relative importance of two pathways for the urban residents might be either in favor of milk intake (if milk was delivered from rural areas with similar or higher level of contamination of ^{131}I compared to that in the city) or inhalation intake (if milk was delivered from rural areas which were substantially less contaminated with ^{131}I compared to that in the city). Selection of wrong pathway considered to have been the leading one might have resulted to a systematic bias in the assessment of individual thyroid dose up to a factor of four to five. So, it is an important task in the problem of retrospective dosimetry to identify correctly what pathway of ^{131}I intake was the leading one.

The purpose of this paper is (1) to describe the method of analysis of direct thyroid measurements of the residents from one settlement to objectively determine which pathway (inhalation or ingestion) intake of ^{131}I is dominant and (2) to apply this method to available direct thyroid measurements for urban residents of two large cities in Belarus (Minsk and Gomel) following the Chernobyl accident.

Material and methods

Minsk city

An extensive monitoring of ^{131}I activity in human thyroids was conducted in Minsk city starting from May 1 through June 1986. People were invited to two urban policlinics #5 and #28 where the special measuring points were organized. The SRP-68-01 instruments were used at these measuring points to measure the level of gamma-radiation near the thyroid gland for the purpose of determination of the thyroid content, mainly ^{131}I, *in vivo*. The SRP-68-01 instrument has NaI (Tl) scintillation detector and analog output in terms of exposure rate (μR h^{-1}) and count rate (counts per minute).

With respect to conditions of radioiodine intake (and exposure to thyroid from ^{131}I) the population of Minsk city is not homogeneous because part of the residents temporarily lived outside the city for at least several days during the time period from April 26 to May 31, 1986 in the areas where conditions of thyroid exposure were different from that in Minsk. It is worth noting that the above-mentioned time period is the critical one, during which more than 99% of possible ^{131}I intakes had been realized for the overwhelming majority of the Belarusian inhabitants, accounting for a short half-life of ^{131}I (8.04 days) and the main pathways of ^{131}I intakes due to consuming fresh milk and breathing contaminated air. Thus, the measured Minsk residents were divided into two groups depending upon whether the residents left the city for substantial time during April 26-May 31, 1986 or not. Two criteria were applied in order to differ those people: (1) there is archival information that a given resident was outside the city in April-May 1986 and (2) no information is available regarding the place where the resident considered was in April-May 1986, but this person has enormously high result of measurement of the exposure rate near the thyroid, that was likely to be due to spending some time in contaminated areas.

To select the residents according to the second criterion the following inequality was used:

$$P(i,t_m) > 10 \times P_{gm}(i,t_m) \tag{1}$$

where $P(i,t_m)$ is the exposure rate near the thyroid of a person of age (i) having subtracted the background level of the person considered on the day of measurement t_m, μR h^{-1}; and

$P_{gm}(i,t_m)$ is the geometric mean of the exposure rate near the thyroid having subtracted the background level for all persons of age (i) measured in the urban policlinics on the day of measurement t_m, for whom no information related to their living outside the city in April-May 1986 is available, μR h^{-1}.

Eventually, the group of the measured Minsk residents (17,842 people) who thought to have been lived in the city during a few weeks following the accident was selected. Just this group has been analyzed below in order to determine what the main route of the radioiodine intake to the urban residents was.

Gomel city

Measurements of the ^{131}I content in thyroids of Gomel residents were primarily performed with the SRP-68-01 and the DP-5 instruments. The DP-5 instrument has a Geiger-Muller counter as a detector and analogue output in terms of exposure rate (mR h^{-1}). Similar to the Minsk residents two groups of the measured Gomel residents were formed. The group considered to be a representative one of those Gomel residents whose ^{131}I intake was mainly due to their staying in Gomel city consists of two subgroups. The first one includes 1,723 residents measured in Policlinics #1 of Gomel city. The second subgroup includes 2,499 Gomel residents measured at Policlinics #28 of Minsk city.

With respect to people related to the second subgroup it is important to stress that according to available information those Gomel residents left their city for Minsk city because of their concern on the radiation exposure due to the Chernobyl accident. Those people visited the measuring point at Minsk Policlinics #28, as a rule, on the day of arrival or within one-two days upon their arrival in Minsk city. Taking into account that Minsk city was much less contaminated compared to Gomel city (deposition density of ^{131}I in Minsk city was about two orders of magnitude less than that in Gomel city), it is reasonable to assume that their measured ^{131}I thyroidal content reflected intakes of radioiodine during their residing in Gomel city.

Method to identify the main route of the radioiodine intake to the residents with direct thyroid measurements

Unfortunately, no spectrometric measurements of ^{131}I concentration in milk distributed in the trade network of Minsk and Gomel cities are available. The results of spectrometric measurements of ^{131}I in milk for rural settlements show that milk was highly contaminated (up to 400 kBq L^{-1} on May 5, 1986 in a village of Khoiniki raion Gomel Oblast). In addition, it is known that during the very first days following the accident there was no official restriction on delivery of fresh cows' milk produced in the contaminated areas of Belarus to the other areas, including trade network of cities Minsk, Gomel and other settlements. Only on May 6, 1986 a special order of the Ministry of Health of the former USSR was issued to provide a radiation monitoring of milk and milk products and to prohibit the distribution in the trade network of fresh milk with contamination level exceeding 3.7 kBq L^{-1} of ^{131}I (Anonymous 1993). Thus, contaminated milk was definitely consumed by the residents living in contaminated areas where it was produced, and might have been delivered to the residents of other areas, including Minsk and Gomel cities.

It is impossible on the basis of a single result of direct thyroid measurement to determine what pathway of ^{131}I intake to the resident considered was dominant (inhalation or ingestion). However, considering rather large set of direct thyroid measurements of the residents of various ages (children plus adults) from one settlement it is possible to objectively answer to the question what pathway of ^{131}I intake to the residents considered was dominant. The idea of this method is to analyze the time variation of the ratios of the ^{131}I thyroidal content for adults to that for children of age (i) in one settlement. This time variation of the ratios is different in case of dominant inhalation or ingestion intake.

It is well known that the total inhalation intake of radioiodine by a person is primarily provided by inhalation occurred during the passage of a radioactive cloud through a settlement. The resuspension of radioiodine plays a minor role in an overall inhalation intake and can be neglected compared to that during the passage of a radioactive cloud. Taking into account that deposition of ^{131}I was prolonged in time (Makhon'ko et al. 1996), a prolonged inhalation intake was assumed for the rural and urban residents. It is also reasonable to assume for those residents a prolonged ingestion intake of radioiodine due to consumption of contaminated milk in rural areas where it was produced and in the cities where it was delivered from the contaminated areas of Belarus.

In order to reveal a relative importance of inhalation and ingestion intake for the residents it is necessary to analyze the time dependency of average ^{131}I thyroidal content for various age-groups of residents, which is different in case of only inhalation and only ingestion intakes.

Let's consider an inhalation intake. A general equation to estimate the ^{131}I thyroidal content for a person of age (i) at time t, G(i,t), due to prolonged inhalation intake has the following view:

$$G(i,t) = \int_0^t C_{air}(\tau) \times \nu_i \times f_l \times f_{th} \times e^{-\lambda_{th,i}(t-\tau)} d\tau \qquad (2)$$

where G(i,t) is the ^{131}I thyroidal content for a person of age (i) at time t, Bq;
$C_{air}(\tau)$ is the concentration of ^{131}I in the ground-level air in the settlement at time τ, Bq m^{-3};
ν_i is an average breathing rate of a person of age (i), m^3 d^{-1}; and
f_l is fraction of radioiodine entered into blood from lung, dimensionless;
f_{th} is fraction of ingested radioiodine absorbed by the thyroid, dimensionless;
$\lambda_{th,i}$ is an effective rate of elimination of ^{131}I from thyroid for a person of age (i), d^{-1}.

Assuming the deposition velocity of ^{131}I on ground to be constant, eqn (2) can be rewritten as follows:

$$G(i,t) = (\sigma_0/V_g) \times \nu_i \times f_l \times f_{th} \times \sum_{k=1}^{t} p_k \times e^{-\lambda_{th,i}(t-k)} \qquad (3)$$

where σ_0 is the total ground deposition density of ^{131}I, Bq m^{-2};
V_g is the deposition velocity of ^{131}I on ground in the settlement, m d^{-1}; and
p_k is the percentage of the total deposition of ^{131}I on ground occurred on day (k), dimensionless.

It is reasonable to assume that the parameters of the variation of the ^{131}I concentration $C_{air}(\tau)$ in the ground-level air in the settlement were the same for each age-group of the residents. The values of parameters f_l and f_{th} are assumed to be age-independent. Then, in case of inhalation intake of ^{131}I the average ratio of the ^{131}I thyroidal content for an adult (ad) and a child of age (i) at time t, $R_{inh}(ad,i,t)$, is estimated to be:

$$R_{inh}(ad,i,t) = (\nu_{ad}/\nu_i) \times \frac{\sum_{k=1}^{t} p_k \times e^{-\lambda_{th,ad}(t-k)}}{\sum_{k=1}^{t} p_k \times e^{-\lambda_{th,i}(t-k)}} \qquad (4)$$

Thus, if the inhalation intake for the residents was the main pathway, then the variation with time of the ratio between the average ^{131}I thyroidal content for adults and that for children of age (i) derived from direct thyroid measurements should be consistent with the time dependent function $R_{inh}(ad,i,t)$ calculated according to eqn. (4).

Let's consider now ingestion intake with contaminated milk. A general equation to estimate the ^{131}I thyroidal content for a person of age (i) at time t, G(i,t), due to prolonged ingestion intake has the following view:

$$G(i,t) = \int_0^t C_{milk}(\tau) \times V_i \times f_{th} \times e^{-\lambda_{th,i}(t-\tau)} \times d\tau \qquad (5)$$

where $C_{milk}(\tau)$ is the concentration of ^{131}I in milk at time τ, Bq L^{-1}; and

V_i is an average milk consumption rate of a person of age (i), L d^{-1}.

If the considered person of age (i) ceased milk consumption at time t_1, then the ^{131}I thyroidal content, G(i,t), at time $t \geq t_1$ can be assessed according to the same eqn (5) with a replacement of the upper bound of integration (t is replaced with t_1).

It is reasonable to assume that the parameters of the variation of the ^{131}I concentration in milk $C_{milk}(\tau)$ were the same for each age-group of the residents. Then, in case of ingestion intake of ^{131}I the average ratio of the ^{131}I thyroidal content for an adult (ad) and a child of age (i) at time t, $R_{ing}(ad,i,t)$, is estimated to be:

$$R_{ing}(ad,i,t) = (V_{ad}/V_i) \times \int_0^{t_1} C_{milk}(\tau) \times e^{-\lambda_{th,ad}(t-\tau)} \times d\tau \; / \; \int_0^{t_2} C_{milk}(\tau) \times e^{-\lambda_{th,i}(t-\tau)} \times d\tau \qquad (6)$$

where t_1 is the time when an adult ceased milk consumption $t_1 \leq t$, d and

t_2 is the time when a child of age (i) ceased milk consumption $t_2 \leq t$, d.

An analytical solution of eqn (6) can be found for any particular time-dependent function $C_{milk}(\tau)$. For example, if $C_{milk}(\tau)$ was close to the function describing concentration of ^{131}I in milk for cows put on contaminated pasture, then eqn (6) can be written as follows:

$$\begin{aligned} R_{ing}(ad,i,t) = {} & V_{ad}/V_i \times \exp((\lambda_{th,i}-\lambda_{th,ad}) \times t) \times \{[1-\exp(-(\lambda_g-\lambda_{th,ad}) \times t_1)]/(\lambda_g-\lambda_{th,ad}) \\ & -[1-\exp(-(\lambda_c-\lambda_{th,ad}) \times t_1)]/(\lambda_c-\lambda_{th,ad})\} \; / \; \{[1-\exp(-(\lambda_g-\lambda_{th,i}) \times t_2)]/(\lambda_g-\lambda_{th,i}) \\ & - [1-\exp(-(\lambda_c-\lambda_{th,i}) \times t_2)]/(\lambda_c-\lambda_{th,i})\} \end{aligned} \qquad (7)$$

where λ_g is effective clearance rate of ^{131}I from pasture grass, d^{-1} and

λ_c is effective clearance rate of ^{131}I from cow to milk, d^{-1}.

Thus, if the ingestion intake for the residents was the main pathway, then the variation with time of the ratio between the average ^{131}I thyroidal content for adults and that for children of age (i) derived from direct thyroid measurements should be consistent with the time dependent function $R_{ing}(ad,i,t)$ calculated according to eqn (6). For a particular considered case this ratio is calculated according to eqn (7). For the urban residents an exact view of function $C_{milk}(\tau)$ in unknown, but its general dependency should not differ significantly from the particular considered case. Thus, if the ingestion intake with milk for the residents was the main pathway, then the variation with time of the ratio between the average ^{131}I thyroidal content for adults and that for children of age (i) derived from direct thyroid measurements should be consistent with the time dependent function $R_{ing}(ad,i,t)$ calculated according to eqn (7).

The values of age-dependent parameters used in eqns (4), (6) and (7) for children with an increment of 1 y and for adults are given in Table 1. The values of breathing rate, ν_i, for newborn, children of 1, 5, 10 15 years old and for adults were taken from ICRP Publication 66 (ICRP 1993). The values of ν_i between those ages were obtained by a linear interpolation. The values of effective rate of elimination of ^{131}I from thyroid were derived from the data in ICRP Publication 56 (ICRP 1989). The values of the effective clearance rate of ^{131}I from pasture grass and from cow to milk were assumed to be 0.15 d^{-1} and 0.627 d^{-1}, respectively, according to Arefieva et al. (1988).

Results and discussion

The method described above was applied to the results of the direct thyroid measurements conducted for the Belarusian urban populations of the cities of Minsk and Gomel. The data for two groups of people were considered and compared: (1) adults (years of birth <1968) and (2) children from 1 to 6 y (years of birth 1979-1984). In principle, the less age of children the higher difference in the time variation of the ratios of the ^{131}I thyroidal content for adults to that for children, R(ad,ch,t), in case of dominant inhalation or ingestion intake of ^{131}I. However, infants and small children up to 1 y (years of birth 1985-1986) were excluded from consideration because they might have been breast milk consumers and this could have been an interfering factor in interpretation of the results of application of the method proposed. For assessing the "measured" ratios of R(ad,ch,t) the geometric mean values of the ^{131}I thyroidal content for adults (not less than 10 persons) and children (1-6)y (not less than 10 persons) measured on the same day for the residents of the same settlement were used.

Table 1. Values of age-dependent parameters used in equations (4), (6) and (7).

Age, i, y	Breathing rate, v, m^3 d^{-1}	Effective rate of elimination of ^{131}I from the thyroid, λ_{th}, d^{-1}	Milk consumption rate, V, L d^{-1}
0-1	2.9	0.132	0.30
1-2	5.6	0.119	0.22
2-3	6.5	0.117	0.20
3-4	7.4	0.114	0.20
4-5	8.3	0.111	0.20
5-6	9.3	0.108	0.20
6-7	10.4	0.105	0.20
7-8	11.5	0.103	0.25
8-9	12.6	0.100	0.25
9-10	13.7	0.097	0.25
10-11	14.8	0.096	0.25
11-12	16.0	0.096	0.25
12-13	17.2	0.096	0.25
13-14	18.3	0.095	0.25
14-15	19.5	0.095	0.25
15-16	20.3	0.095	0.25
16-17	20.7	0.095	0.25
17-18	21.2	0.094	0.25
>18 (adult)	22	0.094	0.20

Time variation of the ratios of R(ad,ch,t) derived from the direct thyroid measurements and calculated according to eqn (4) assuming only inhalation intake of ^{131}I and according to eqn (6) and (7) assuming only intake of ^{131}I with milk is given in Figures 1 and 2.

Analysis of the time variation of the measured ratios for the urban residents of Gomel and Minsk cities allows for revealing an obvious tendency that the "measured" ratios are scattered along the curve reflecting the intake of ^{131}I with milk as a dominant radioiodine pathway for the urban residents. Scattering points between the curves relating inhalation and milk intake in Figures 1 and 2 can be explained by at least two reasons: (1) insufficient statistics and (2) objective factor that part of people did not consume milk and inhalation intake was the only pathway of ^{131}I for them. So the geometric mean value of ^{131}I thyroidal content for such set of data should be definitely between the upper and lower curves. At the same time for the residents who consumed milk (the percentage of milk-consumers among urban residents is estimated to be about 80%) the measured "ratios" are far below "inhalation" curve locating close to "milk" curve.

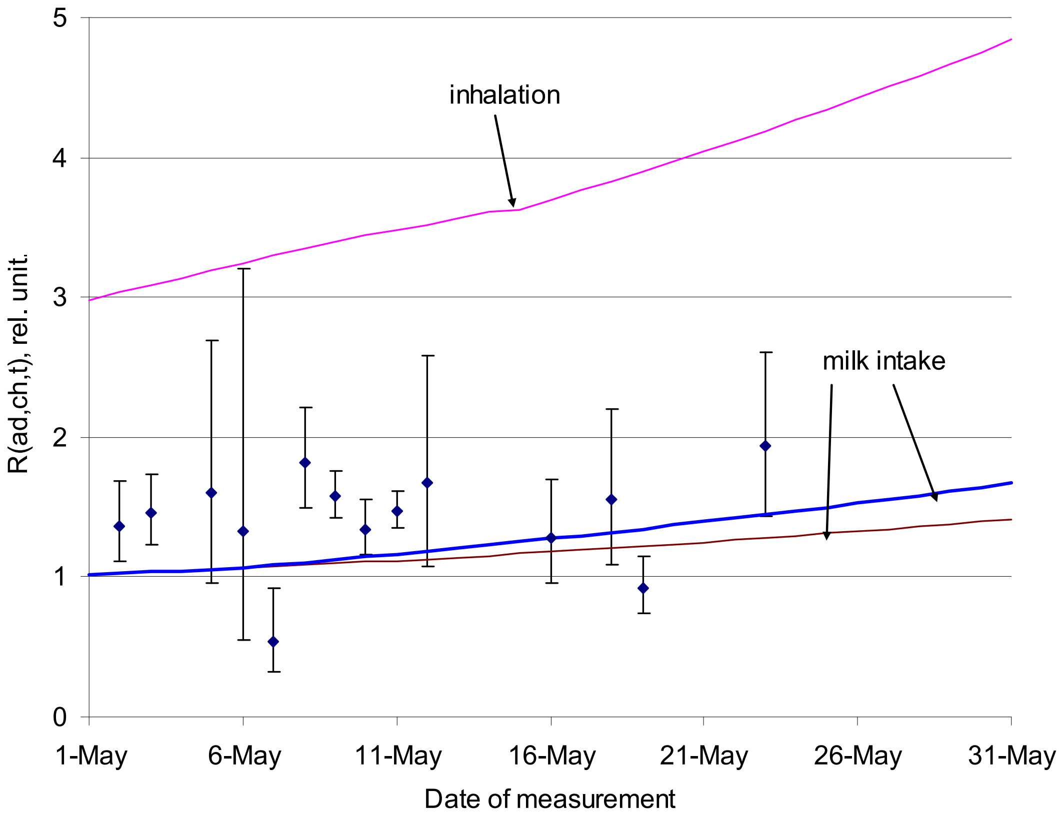

Fig. 1. Time variation of the ratios between the average ^{131}I thyroidal content for adults and that for children, R(ad,ch,t), derived from (1) direct thyroid measurements and calculated assuming (2) only inhalation intake of ^{131}I and (3) only intake of ^{131}I with milk for the adult residents and children (1-6) y of Minsk city. Diamonds denote measured ratios. Uncertainties related to 95% confidence intervals are indicated for the measured ratios.

Thus, a definite conclusion can be derived from the dependencies in Figures 1 and 2 that for the Belarusian urban residents in Minsk and Gomel cities the consumption of fresh milk was the dominant route of ^{131}I intake. It is worth noting that the dependency of R(ad,ch,t) versus time for the two variants of milk intake: (1) milk consumption was ceased on May 6, 1986 (upper curve) and (2) milk was consumed without any interruption (lower curve) shows insignificant difference, which does not play any important role in the analysis.

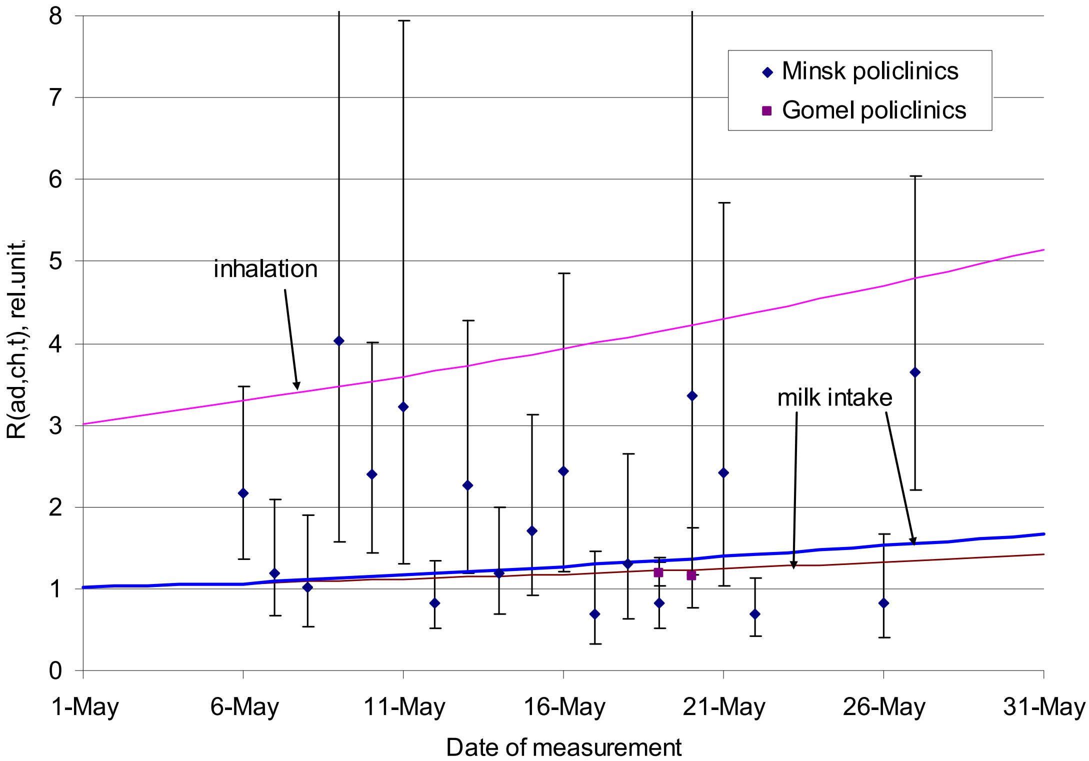

Fig. 2. Time variation of the ratios between the average ^{131}I thyroidal content for adults and that for children, R(ad,ch,t), derived from (1) direct thyroid measurements and calculated assuming (2) only inhalation intake of ^{131}I and (3) only intake of ^{131}I with milk for the adult residents and children (1-6) y of Gomel city. Diamonds denote measured ratios. Uncertainties related to 95% confidence intervals are indicated for the measured ratios.

Conclusions

A method has been developed to objectively determine the dominant pathway (inhalation or ingestion with milk) of the radioiodine intake for the populations with in-vivo measurements of ^{131}I in thyroids.

Application of the method described in the paper to the direct thyroid measurements of the Belarusian urban populations of the cities of Minsk and Gomel following the Chernobyl accident allowed for making a conclusion that the consumption of fresh milk was the dominant route of ^{131}I intake for those people.

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Accident dosimetry using chipcards

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Abstract

Accident dosimetry aims to determine the absorbed dose of exposed individuals involved in a nuclear accident. In order to achieve this, materials are needed that possess a certain radiation sensitivity and are worn close to the body. Chip cards are the perfect example and have the great advantage that nowadays everyone possesses at least one, in the form of bankcards, SIM-cards, identity cards, etc. Therefore the possibilities of chip cards as an accident dosemeter are further investigated aiming at establishing a uniform Belgian accident dosimetry system..

Using blue LED optical stimulation, several properties of chip cards that could be traced back to a single manufacturer were investigated. The shape of the OSL-curve and the fading properties were considered. Also the uncertainty on the results and the lowest detectable dose as a function of the integration area were taken into account. The results of this part of the research were used to determine the most promising part of the OSL-curve for dose determination.

Using these results, further research on the dose-response and possible sensitivity changes was made in order to propose a suitable protocol for the estimation of the absorbed dose.

Introduction

Accident dosimetry aims to determine the absorbed dose in case of a nuclear or radiological accident. The dosimetric results are useful to ensure the proper medical treatment of exposed individuals, to provide correct information to the public and are useful in epidemiological studies [4].

For the purpose of accident dosimetry objects can be used that are found within the accident area or that are worn close to the body of exposed individuals and posses a certain radiation sensitivity. Objects that meet such requirements, are e.g. ceramic materials and semiconductor devices. Previous studies have shown sensitivity for optical stimulation (OSL) and thermal stimulation (TL) of quartz in bricks [5]. Also personal objects, such as cell phone components and USB flash drive components, were investigated recently using TL and OSL [2, 9]. These devices can be regarded as personal dosemeters, since they are worn close to the body.

This study will be focusing on the pertinent OSL properties of various chip cards, in order to expand the list of personal objects useful in accident dosimetry. The great advantage of chip cards is that nowadays everyone possesses at least one, in the form of bankcards, SIM cards, etc. The exact properties of chip cards as an accident dosemeter are investigated using OSL as a sequel to earlier research [1, 7, 8, 10] in order to determine whether the results of that research can be adopted for 'Belgian' cards.

After a rough selection of various kinds of chip cards, based on the overall radiation sensitivity, a detailed investigation of the relevant OSL properties was done on one specific card type, i.e. the SIM card. The first part of the project was intended to determine the most promising part of the OSL curve for dose determination (integration window). The shape of the OSL curve was considered, in function of the sample number, the dose and inherent OSL fading. The lowest detectable dose in function of the integration window and the uncertainty on the result as a function of the integration window were taken into account as well.

In the second part of the project, properties such as fading, dose response and sensitivity changes were investigated using the proposed integration window. Finally, the dose recovery potential using a single aliquot regenerative dose method (SAR protocol) was investigated, with promising results.

Materials and methods

Several kinds of chip cards, SIM cards, SIS cards and bank cards - were investigated enabling identification of a specific wide spread type with good OSL properties. Sample preparation was done by extracting the chips from the cards using an industrial punch prior to irradiation. No chemical treatments were used. To avoid loss of information, the samples were irradiated in the absence of daylight.

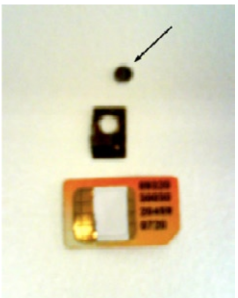

Fig. 1. Sample preparation was done by extracting the chip module from the card. Samples were taken using an industrial punch.

Fig. 2. The chip is surrounded by an epoxy encapsulant, responsible for the OSL signal [8].

OSL measurements of chip cards were executed with the Risø TL/OSL Luminescence Reader (model TL/OSL-DA-20). Continuous wave optical stimulation was performed using a cluster of blue LED's, emitting light with a wavelength of 470 nm. The LED's are equipped with a green long pass filter (GG-420) to prevent blue light from reaching the PMT [5]. A power of 45 mW/cm^2 was delivered to the samples. The stimulation time was 120 s and no thermal preheating was used. The luminescence emission was recorded using a Hoya U-340 filter in front of the photomultiplier tube (bialkali PMT, type EMI 9235Q), enabling detection in the UV-range [5]. The OSL curve was plotted using 400 datapoints. Stimulation with blue light was used, since the epoxy encapsulant, surrounding the chip (figure 2) and responsible for the OSL signal, contains SiO_2 which should respond well to blue LED optical stimulation.

The samples were irradiated using a ^{90}Sr/^{90}Y beta-source with an activity of 1,48 GBq, mounted in the Risø TL/OSL Luminescence Reader, at a dose rate of about 136 mGy/s. The gamma-equivalent dose rate was determined from Co-60 irradiated chip cards measured immediately after administration of the gamma dose. Gamma irradiations were performed with a Co-60 source delivering a dose rate of about 2 Gy/h (air kerma).

The obtained OSL curves were analyzed using the Risø Analyst software and Origin 8.0.

Results

Integration window

From the investigated chip cards, SIM cards, which are found in cell phones, turned out to be the most wide spread card with excellent radiation sensitivity. An analysis of the obtained OSL curves, following irradiation with the built in ^{90}Sr/^{90}Y-betasource (1360 mGy), revealed that three OSL components are responsible for the emitted luminescence after irradiation and optical stimulation. Normalised OSL curves were fitted with a 3rd order exponential decay:

$$y = y_0 + A_1 e^{\frac{-x}{\tau_1}} + A_2 e^{\frac{-x}{\tau_2}} + A_3 e^{\frac{-x}{\tau_3}} \qquad \text{(Eq. 1)}$$

The results in figure 3 indicate that every OSL curve can be decomposed into three OSL components with distinct fitting parameters A and τ, as can be seen from the three clusters. The clusters confirm the identical shape of all obtained OSL curves. This could mean a single protocol can be used to make an estimation of the absorbed dose, if other properties, such as fading properties and dose response properties, are identical as well for all samples. These properties are discussed in detail later in this paper.

Figure 3 shows that every OSL curve contains a so-called ‘fast’, ‘medium’ and ‘slow’ component. About 60% of the OSL signal is delivered by the fast component.

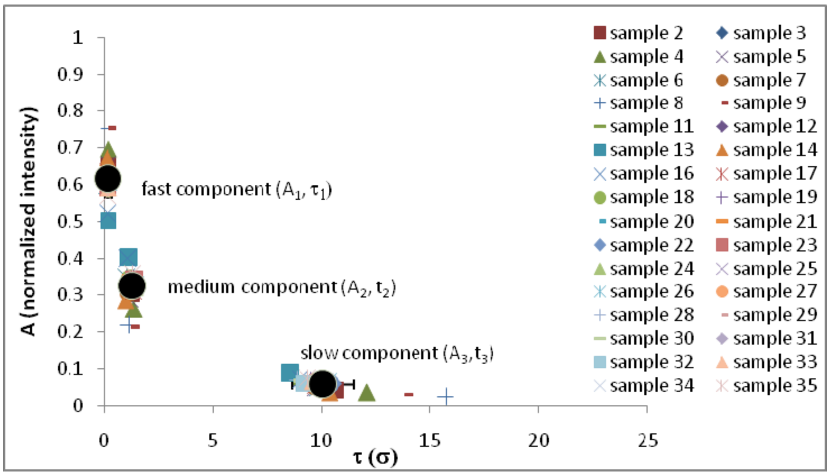

Fig. 3. OSL curves from several SIM cards were fitted using a 3rd order exponential decay. The parameter pairs (A_1, τ_1), (A_2, τ_2) and (A_3, τ_3) are consistent for all investigated cards, allowing to identify three components in the OSL signal.

The properties of the slow component, with a mean value of (10,08 ± 1,46)s for τ_3 indicate that a stimulation time of 120s would be sufficient to obtain the whole OSL signal. Exposure to different doses did not change the properties of the OSL curves. A stimulation time of 120s can thus be used in every situation, independent of the absorbed dose.

To obtain information on the stability of the components, 10 samples were subjected to a fading test. The optical stimulation was performed after varying delay times following irradiation (0, 10, 30, 100, 300, 1000, 3000, 10000 and 30000 minutes). Analyses of the obtained OSL curves showed that all components fade simultaneously, as pointed out in figure 4. This might suggest that all observed OSL components equally suffer from thermal and athermal instability (anomalous fading).

In order to make a proper choice for the integration window, the relative uncertainty as well as the lowest detectable dose (LDD) as a function of the integration window was determined for a set of 6 samples. A good integration window means that both the uncertainty and the LDD should be as low as possible.

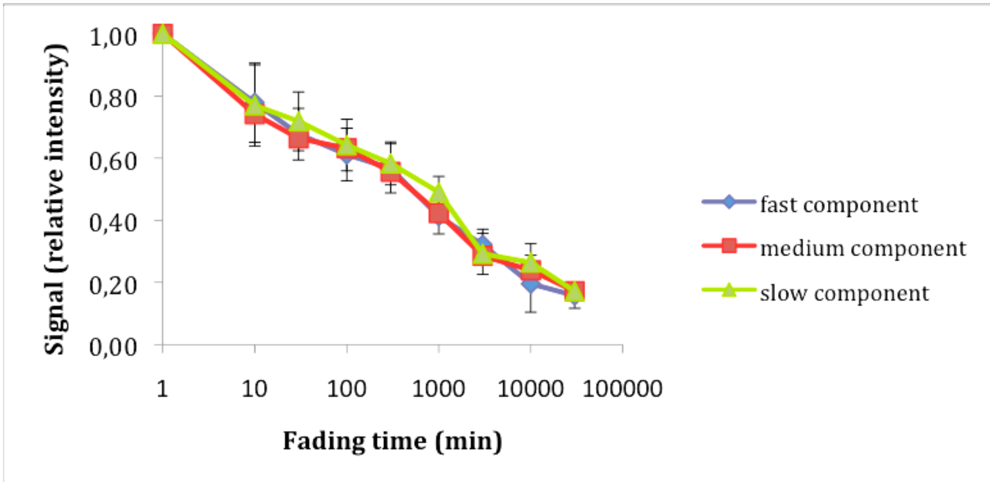

Fig. 4. Averaged fading results for the fast, medium and slow component. It can be seen that the three components have similar fading rates.

The uncertainty on the results is determined using the following formula [3]:

$$\hat{\sigma}_S = \sqrt{Y_0 - B + \left(1 + \frac{1}{k}\right)\hat{\sigma}_B^2} \qquad \text{(Eq. 2)}$$

$\hat{\sigma}_S$ being the absolute uncertainty on the calculated signal S, Y_0 the value of the entire signal including background signal and B the value of the background signal. k is equal to the ratio of the amount of channels used to determine the background signal and the amount of channels used to determine Y_0. $\hat{\sigma}_B$ is equal to the standard deviation of the background signal.

The lowest detectable dose (LDD) was estimated using the dose response curve of several samples. As can be seen in figure 5, the dose response appears to be linear over the investigated dose range from approximately 400 mGy to approximatley 10 Gy. The LDD was determined as the intersection point between the extrapolated dose response curve and the average background signal increased with three times the standard deviation of this background signal. The average background signal and the standard deviation on this background signal were determined by 10 zero dose measurements performed on every sample.

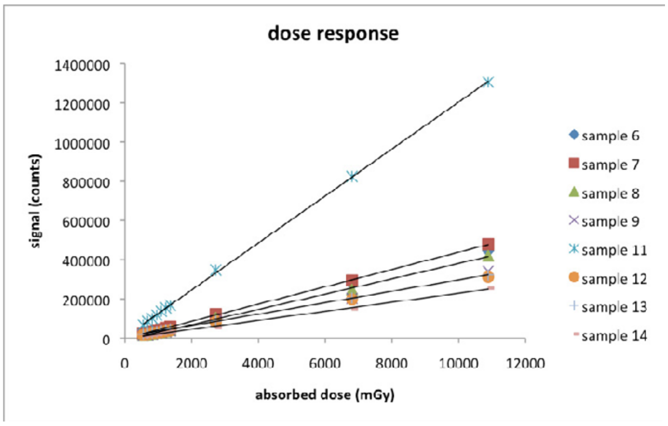

Fig. 5. the dose response of several samples appears to be linear in the range 0,4 - 10 Gy.

The multiplication of the LDD and the relative uncertainty as a function of the integration window was used as a proxy to determine the optimal integration window. The results show that multiplication minima are located within the first 2,10 s of the OSL curve. This means the fast component, responsible for about 60% of the OSL signal, is the most interesting part of the OSL curve.

The integration windows are therefore set at 0s to 2,1s to estimate the value of Y_0, and 105s to 120s to estimate the value of B. The LDD calculated with these integration windows is approximately equal to 2 mGy, depending on the sample.

Protocol for the estimation of the absorbed dose

Using the integration window as determined in the previous section, the fading rate, the dose response and sensitivity changes caused by repeated irradiation-OSL readout cycles were investigated.

Fading occurs simultaneously in all investigated samples as illustrated by figure 4. This offers the opportunity of using a universal fading factor. The latter was determined by fitting the composite fading curve of figure 6 using a 3rd order exponential decay.

$$f = 0{,}198 + 0{,}27e^{\frac{-t}{11{,}7}} + 0{,}319e^{\frac{-t}{0{,}110}} + 0{,}206e^{\frac{t}{104}} \qquad \text{(Eq. 3)}$$

with f being the percentage of the signal still available at time t (hours) following irradiation.

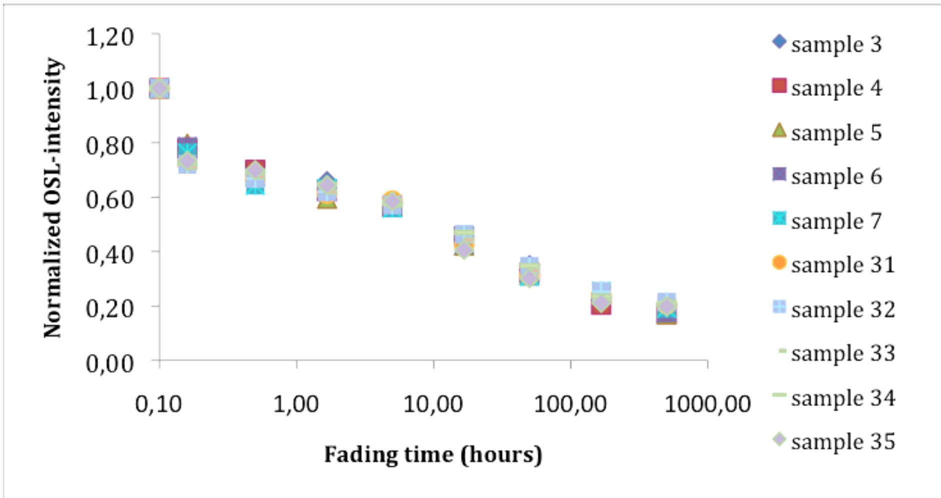

Fig. 6. Fading occurs simultaneously in every sample.

In principle, sensitivity changes can be detected using the SAR protocol (single aliquot regeneration protocol), by comparing the initial OSL output to the OSL output obtained after successive irradiations. The first OSL output gives a value for L_n, thus a value for the unknown dose D. Next, a known test dose is given, which delivers a value for T_n after optical stimulation. A known dose D_r is then delivered to the sample, giving a value for L_r after irradiation. Again the known test dose is delivered to the sample, so a value for T_r is obtained. The unknown dose can now be calculated taking sensitivity changes into account:

$$D = \frac{L_n}{L_r}\frac{T_r}{T_n} D_r \qquad \text{(Eq. 4)}$$

If no sensitivity changes occur, the values of T_n and T_r are equal, and the ratio is equal to 1, making correction for sensitivity changes unnecessary.

Ten samples were irradiated with the betasource during 10s and a SAR protocol was applied. The test dose response was investigated to check for sensitivity changes. The dose recovery test revealed that the given betadose of 10s could be recovered within ~10% uncertainty. The results of this test are shown in figure 5, with the horizontal line being 10s betadose delivered to the samples. Moreover no sensitivity changes were found as illustrated in figure 8, which means no sensitivity changes occur and no correction should be made. This, together with the linearity of the dose response and the uniform fading behaviour, opens the possibility to perform fast dose measurements using only one calibration point.

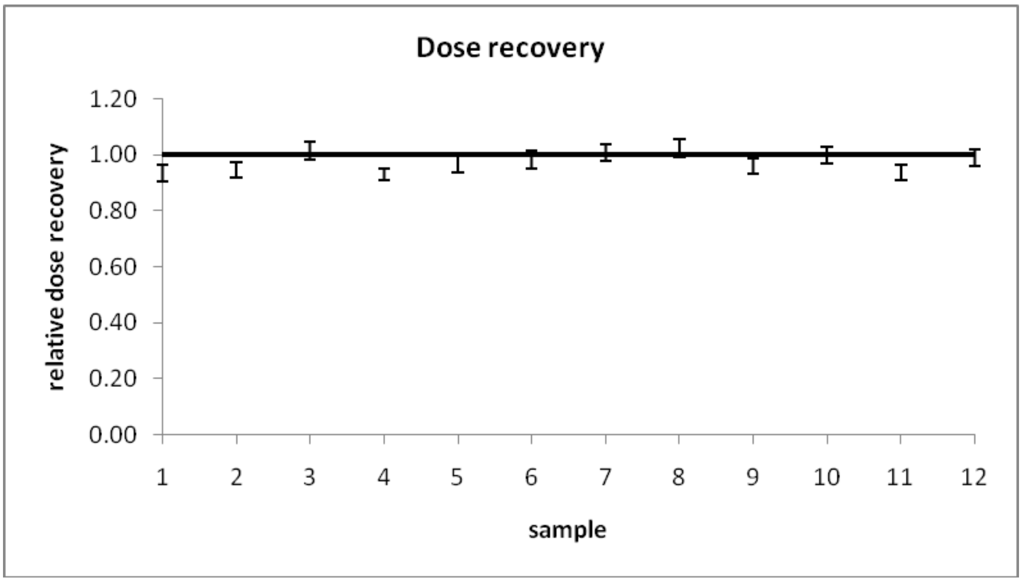

Fig. 7. Dose recovery potential of the SAR protocol using the chip module from SIM cards. The given beta dose can be recovered within ~ 10% uncertainty.

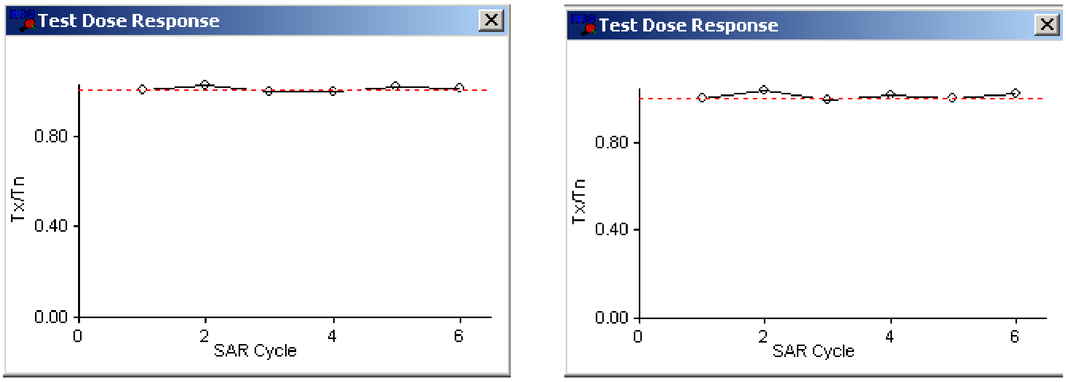

Fig. 8. Normalized test dose response of sample 13 and 16 monitored during several SAR cycles.

If the estimation of the dose is made several hours or days after the irradiation event, which is mostly the case in accident dosimetry, a correction for fading should be made. The results obtained after running the SAR protocol and processing with Risø Analyst software need modification in the following way:

– The result is given in the form $y = a + bx$, with $y = \frac{L_n}{T_n}$ and x the irradiation time in s. L_n is equal to $Y_{0,x} - B_x = S_x$, the signal caused by the accident dose. T_n is equal to $Y_{0n} - B_n = S_n$, the signal caused by the first test dose applied according to the SAR protocol. The meaning of the symbols are explained with equation 2.

– L_n should be corrected for fading, using the formula in Eq. 3 in order to obtain L_{n_corr}. L_{n_corr} is equal to $\frac{L_n}{f}$.

– By filling in this value in the equation $y = a + bx$ an irradiation time in s is obtained. By multiplying this value with the dose rate of the built in beta source the accident dose is obtained.

– The uncertainty on this value should be obtained by taking the uncertainty on $\frac{L_n}{T_n}$ into account based on the considerations in [6] and [3] as well as the uncertainty on *f*, *a*, *b* and the dose rate. The overall uncertainty is approximately equal to 8 %.

Discussion

Although the amplitude of the signal obtained from chip cards after irradiation and optical stimulation is different for every chip cards, a mathematical analyses of the obtained OSL-curves revealed that all of the investigated SIM-cards show a similar response to radiation and optical stimulation. The relative amplitudes and decay characteristics of the fast, medium and slow components show only a minor spread. This would imply that a general dose determination method can be used on this type of cards.

An important parameter to take into account is fading. Fading is clearly present in the signal obtained from the investigated chip cards. The main advantage is that background dose is essentially zero. A drawback is that large uncertainties are introduced when the original signal has to be recalculated in order to estimate the absorbed dose. Due to the fading properties, the time interval between irradiation and optical stimulation is limited, which means, in case of a severe nuclear accident, measurements must be performed within the first weeks after the accident.

The dose response is one of the most important parameters to investigate. Knowledge of the dose response relationship, allows correct estimations of the absorbed dose. For all investigated chip cards, the dose response is perfectly linear within a dose range from 400 mGy up to 10 Gy. Nevertheless a dose recovery test is performed using the SAR-protocol to check for possible sensitivity changes.

The results of the dose recovery test using the SAR-protocol are very promising, and it is also proven in this test that no sensitivity changes occur. Together with the linear dose response relationship, this means dose recovery should be possible using only one calibration point. This would offer a very fast method for determination of the absorbed dose in case of a large scale nuclear emergency situation.

Of course a protocol using only one calibration point needs testing in order to check the accuracy and the uncertainty of the obtained results.

Conclusions

The results of this study seem to confirm the potential of chip cards in accident dosimetry. Especially SIM cards seem to be the most promising given their good sensitivity to irradiation and optical stimulation using blue LED's. The lowest detectable dose is extremely low in comparison to other techniques used in accident dosimetry, but is dependent on the chip card.

Individual chip cards show differences in radiation sensitivity necessitating individual calibration for dose assessment. Fading is ubiquitous but the rate seems to be similar for all investigated samples, enabling the use of a universal fading correction factor. In order to make an estimation of the absorbed dose, the SAR protocol seems to be a very useful tool. Corrections for fading can easily be performed on the obtained results.

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A numerical statistical method application experience for the interpretation of Co-60 special monitoring data for nuclear power-plant workers

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Abstract

The procedure for assessing doses on the basis of routine monitoring results includes use of standard model parameter values for the dose calculation and at least simple estimation of the associated uncertainty due to uncertainty of the worker exposure conditions, the uncertainty in the time of intake and the uncertainty of the measured bioassay quantity method. If the upper value on assessed dose exceeds the reference limit a special monitoring procedure is required that consist in performing additional bioassay measurements to increase the reliability of the dose assessment by obtaining better information on the model parameter values and by using statistical methods for the calculation of the associated uncertainty. An example of a numerical method application for the interpretation of in vivo and in vitro bioassay measurements of Co-60 performed during special monitoring of four nuclear power-plant workers is presented to illustrate a possibility of improving the dose assessment. It was shown that from the two compounds used in the ICRP model for cobalt the type S was the best option and the range of AMAD values could be reduced from (0,1 до 20) to (3 – 8) microns. This resulted in the reduction the uncertainty of the dose assessment from (144-155) % to (17-28) %. It was also shown that absorption type S compound was not slow enough to describe the retention of the radionuclide Co-60 in lungs in the case under study and a slower type like super S was needed.

Introduction

The annual intake of radionuclides (I) and the corresponding committed (and committed effective) dose equivalents (CDE and CEDE) are the main safety limits for occupational internal exposure. They are defined by monitoring an individual occupational intake of radioactive material and assessing the resulting dose. A routine monitoring programme usually consists in a set of measurements of the radiation emitted from the body or the radioactive materials contained in biological samples such as urine or feces during the calendar year. The required frequency of measurements in a routine monitoring programme depends upon the retention and excretion of the radionuclide, the sensitivity of the measurement techniques available, and the acceptable uncertainty in the estimate of intake and committed effective dose (ICRP 1997). The procedure for assessing doses on the basis of routine monitoring results includes use of standard model parameter values for the dose calculation and at least simple estimation of the associated uncertainty due to uncertainty of the worker exposure conditions, the uncertainty in the time of intake and the uncertainty of the measured bioassay quantity method. If the upper value on assessed dose exceeds the reference limit (annual dose limit for workers or part of it) a special monitoring procedure is required.

The special monitoring procedure consist in performing additional in vivo and in vitro bioassay measurements to increase the reliability of the dose assessment by obtaining better information on the model parameter values and by using statistical methods for the calculation of the associated uncertainty. Inhalation is the most often route of intake for workers and the exposure conditions in this case are characterized by the particle size distribution of the aerosol, as described by the Activity Median Aerodynamic Diameter (AMAD) and the absorption characteristics of the material, as described by the absorption classification (Type M or S for cobalt compounds). Numerical statistical methods find an increasing application for the interpretation of internal exposure monitoring data for atomic industry workers. This is due to increasing potentialities of modern computers allowing a direct modeling of biophysical processes of the radionuclide intake, retention, excretion and measurement in a human using Monte Carlo simulating technique.

The aim of the paper is to present the results of a numerical method application (Molokanov et al. 2010) for the interpretation of in vivo and in vitro bioassay measurements of Co-60 performed during special monitoring of four nuclear power-plant workers.

Material and methods

The assessment of dose from intakes of radionuclides is a stepped process of measurement and computation that is in essence a virtual biophysical experiment where computation is the process of connecting and ordering of known data by means of relations based on theory or established models in order to create new data and to reveal new insight (Siebert 2006). Our virtual biophysical experiment is aimed to find a relation between a measured quantity and a desired quantity on basis of ICRP biokinetic model. Any physical quantity used in the biokinetic model can be considered as a desired quantity such as organ absorbed dose versus time, committed and committed effective dose equivalents, organ activity and any parameter of the model including the aerosol absorption type and AMAD. For such a biophysical experiment the model can be formulated as a cause-effect relation where a single intake at time τ is a cause and quantities M and X are effects:

$$\frac{M(t_M)}{X(t_X)} = \frac{I \cdot m(t_M - \tau, \xi(\tau))}{I \cdot x(t_X - \tau, \xi(\tau))} = m_{M/X}(t_M, t_X, \tau, \xi_i(\tau)) \tag{1}$$

where I is a value of a single intake of a radionuclide occurred at time τ; $m(t-\tau, \xi)$ and $x(t-\tau, \xi)$ are functions of time t derived from ICRP model; t_M is the measurement (or sampling) time; t_X is the desired quantity fixed time; $\xi_i(\tau)$ is a set of the model parameter values at time of the intake.

Two tasks can be considered using the model described by equation (1): evaluation of quantity X and evaluation of the model specific parameter $\xi_i(\tau)$. In the first case we analyze quantity M/X in the second case we analyze quantity M_1 / M_2 where M_1 and M_2 are two independent measurement results that may define the parameter $\xi_i(\tau)$. In the case of a single inhalation intake of Co-60 the following equations sequent to equation (1) will be used.

For the dose and intake calculation:

$$\sum_{n=1}^{N} M_L(t_n) = \sum_{n=1}^{N} X \cdot m_{M_L/X}(t_n - \tau, w, AMAD) = \Phi(X) \quad (2)$$

where $M_L(t_n)$ are result of Co-60 measurement in the worker lungs at time t_n; X –the intake or the dose value; w is a parameter, characterizing absorption type (fraction of type S compound in a mixture of M and S), N is the number of measurements during the observation period.

For absorption type investigation:

$$\sum_{j=1}^{N-1}\sum_{n=j+1}^{N} M_L(t_j)/M_L(t_n) = \sum_{j=1}^{N-1}\sum_{n=j+1}^{N} m_L(t_j - \tau, w, AMAD)/m_L(t_n - \tau, w, AMAD) = \Phi_L(w) \quad (3)$$

For AMAD investigation:

$$\sum_{n=1}^{N}\sum_{k=1}^{K} M_L(t_n)/M_U(t_k) = \sum_{n=1}^{N}\sum_{k=1}^{K} m_L(t_n - \tau, w, AMAD)/m_U(t_k - \tau, w, AMAD) = \Phi_{L/U}(AMAD) \quad (4)$$

and

$$\sum_{z=1}^{Z}\sum_{k=1}^{K} M_F(t_z)/M_U(t_k) = \sum_{z=1}^{Z}\sum_{k=1}^{K} m_F(t_z - \tau, w, AMAD)/m_U(t_k - \tau, w, AMAD) = \Phi_{F/U}(AMAD) \quad (5)$$

where $M_U(t_k)$ and $M_F(t_z)$ are results of Co-60 measurement in urine and feces of a worker at time t_k and t_z correspondingly; $m_L(t, w, AMAD)$, $m_U(t, w, AMAD)$ and $m_F(t, w, AMAD)$ are functions, derived from ICRP model showing the activity value in the lungs, daily urine and feces samples correspondingly versus time t after a single inhalation intake of the radionuclide.

Equations (2) - (5) give experimental value of the measured quantities or their combination – left side, and a theoretical value of the measured quantities or their combination according to ICRP model – right side of the equation.

For determination of the desired quantity value (intake, dose, *w* and AMAD) the result of measurement (or the combination of the measurement results) is compared with the theoretical values of the measured quantity or a combination of the measured quantities calculated using equations (2) - (5) for the set of the desired quantity values and fixed values of the other parameters of the model. Namely, $\Phi(X)$ in equation (2) is calculated for the set of X_1, X_2, ..., X_j values, $\Phi_L(w)$ in equation (3) is calculated for the set of w_1, w_2, ..., w_j values and $\Phi_{L/U}(AMAD)$ in equation (4) and $\Phi_{F/U}(AMAD)$ in equation (5) are calculated for the set of $AMAD_1$, $AMAD_2$, ..., $AMAD_j$ values. The value of a parameter at which the calculated theoretical value of the measured quantity agrees with the measured experimental value represents the estimate of the parameter according to the accepted ICRP model.

In fact, the measured quantity values are variables and the model parameter values are not known exactly. In this case a likelihood function is used to describe the probable value of the quantity that was measured and the probability density function (PDF) is used to describe probable values of the model parameters. Likelihood function, $L(M|X,\xi_i)$, describes the probability of observing a value M provided the values X and ξ_i are known. Values of the likelihood function can be calculated numerically using Monte Carlo integration technique. For that at the first step the model parameter values, ξ_i, are sampled from their PDFs and the value of the measured quantity M or a combination of the measured quantities $\Phi(M_n)$ is calculated using the likelihood function(s), $L(M_n|X,\xi_i)$ and a fixed value of the desired quantity X. Repeating this procedure many times a set of the measured quantity values $\{M_k\}$ or $\{\Phi_k(M_n)\}$ is generated. A frequency distribution of these quantities is found By sorting $\{M_k\}$ or $\{\Phi_k(M_n)\}$ that represents a random sample of the likelihood function $L(M|X)$ or $L(\Phi(M_n)|X)$ for the measured quantity M or a combination $\Phi(M_n)$.

Thus, the theoretical values of the measured quantity or a combination of the measured quantities are represented as the likelihood function $L(M|X)$ or $L(\Phi(M_n)|X)$, where M or M_n are arguments and the desired quantity X is a parameter. In this case the experimental value M_0 or a combination of the experimental values $(\Phi(M_{n0})$ are compared with a set of the likelihood functions $L(M|X)$ or $L(\Phi(M_n)|X)$ for a set of desired quantity values $X_1, X_2, \ldots, X_j$. The value X_α at which fulfills a condition:

$$\int_0^{M_0} L(M\,|\,X_\alpha)\cdot dM = 1-\alpha \quad \text{or} \quad \int_0^{\Phi(M_{n0})} L(\Phi(M_n)\,|\,X_\alpha)\cdot d\Phi(M_n) = 1-\alpha \qquad (6)$$

represents the estimate of the parameter with a confidence level α (Neyman 1937, Yao W.-M. et al. 2006) according to the accepted ICRP model, the model parameter assumptions and the assumed likelihood function for the measured quantities.

Results

Special monitoring measurements of four nuclear power-plant workers consisted of in vivo measurements of Co-60 in lungs during 200 days after an acute intake and several in vitro measurements of urine and feces samples performed 200 days after an acute intake that occurred 05.02.2009. The results of the measurements are presented in tables 1 and 2.

Table 1. Results of in vivo measurements of Co-60 in lungs for four nuclear power-plant workers during a year 2009, Bq/Lungs.

Wor-ker	Date of measurement										
	26.02	11.03	16.03	23.03	24.03	30.03	03.04	06.04	29.07	10.08	24.08
W1	80053		78889	76450	81908			75016	57284		40664
W2	32747		31965	32180			32564	32200			28107
W3		25275			27633	24918		18748		18299	19828
W4					40530		42000		35319		26996

Table 2. Results of in vitro measurements of Co-60 in urine (U) and feces (F) samples for four nuclear power-plant workers taken in august 2009, Bq/day (in brackets are values normalized to average mass of the fecal daily sample equal 152 g).

Worker			Date of sampling					
			20.08	21.08	24.08	25.08	26.08	27.08
W1	U	V., ml/day	2.5	1.5		1.4	1.4	1.4
		A, Bq/day	20	10.5		14.5	18.7	25.7
	F	Mass, g/day			274.8			329.1
		A, Bq/day			160 (88)			165 (76)
W2	U	V., ml/day	1.0	1.5		1.5	1.5	1.5
		A, Bq/day	2.3	2.0		6.0	9.0	11.0
	F	Mass, g/day		120.2			93	70.2
		A, Bq/day		17 (21)			22 (36)	17 (37)
W3	U	V., ml/day	3.0	2.0		2.5	1.8	1.5
		A, Bq/day	9.0	4.0		7.5	10.2	6.0
	F	Mass, g/day		243.8			200.7	166.5
		A, Bq/day		65 (40)			50 (39)	31 (28)
W4	U	V., ml/day	2.0	1.5		1.5	1.5	1.4
		A, Bq/day	4.7	5.0		6.5	11.5	9.3
	F	Mass, g/day		170.7			118.9	96.7
		A, Bq/day		30 (27)			20 (26)	8.6 (14)

Uncertainty of in vivo measurements of Co-60 in lungs presented in table 1 has two components: type A and Type B. Type A uncertainty arises from the statistics of counting measurements and described using the Poisson distribution. Type B uncertainty includes variability of the measurement geometry and uncertainty of the calibration. Type A uncertainty for the observed activity levels did not exceed 1 % and Type B uncertainty was described by lognormal PDF with geometric standard deviation equal 1.2 (Doerfel et al. 2006).

Uncertainty of in vitro measurements of urine and feces samples presented in table 2 also has two components: type A and Type B. Type A uncertainty for the observed sample activity levels was about 20 %. Type B uncertainty due to variation of excretion rate is usually described by lognormal PDF with geometric standard deviation equal 1.3 – 1.8 for urine and 1.5 – 5 for feces /6/ . Using table 2 data we estimated variation of urine excretion by the value of 1.4 and feces excretion by the value of 1.5.

Results of absorption type investigation using equation (3) and the results of measurements of four nuclear power-plant workers presented in table 1 are shown in figure 1.

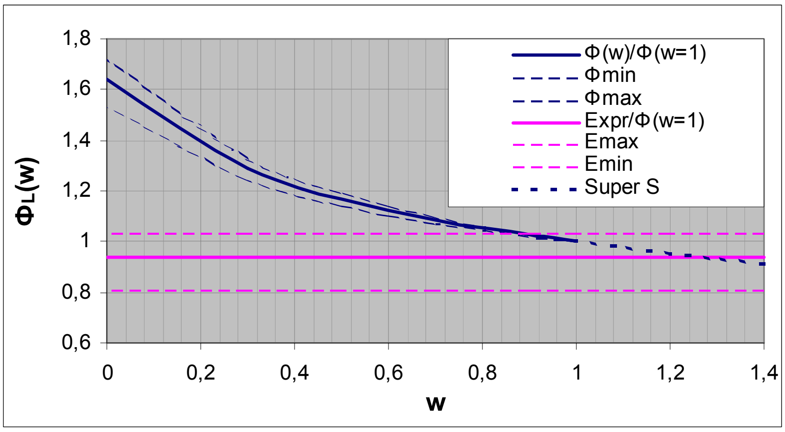

Fig. 1. Theoretical value of function $\Phi_L(w)/\Phi_L(w=1)$, calculated using equation (3) versus parameter *w* in comparison with values $\sum_{j=1}^{N-1}\sum_{n=j+1}^{N} M_L(t_j)/M_L(t_n)$ obtained in experiment.

Absorption characteristics of the material in case of cobalt compounds classified by two types: M and S. We introduced a parameter w to characterize a fraction of type S compound in a mixture of M and S compounds. In figure 1 a theoretical value of the ratio $\Phi_L(w)/\Phi_L(w=1)$ is presented as a function of w. Horizontal lines are the values of the combination of measured results $\sum_{j=1}^{N-1}\sum_{n=j+1}^{N} M_L(t_j)/M_L(t_n)$ normalized to $\Phi_L(w=1)$. Average, maximum and minimum values of the above quantitities for four workers and different values of AMAD in the range 0.1 – 10 microns are presented in figure 1. It follows from figure 1 that the most probable value of the parameter w is w >1. It means that Type S compound is not slow enough to describe the retention of the radionuclide Co-60 in lungs in the case under study and a slower type like super S is needed. But from the two compounds used in the ICRP model type S is the best option.

Results of the Activity Median Aerodynamic Diameter (AMAD) investigation using equations (4) and (5) and the results of measurements of four nuclear power-plant workers presented in table 1 and 2 are shown in figures 2 and 3.

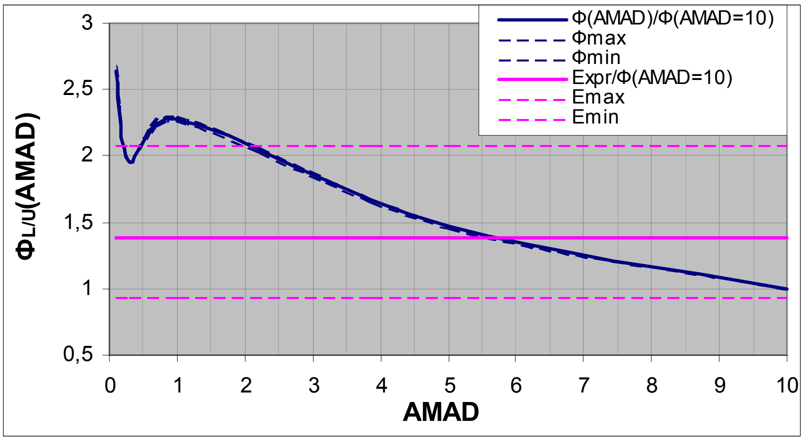

Fig. 2. Theoretical value of function $\Phi_{L/U}$(AMAD)/$\Phi_{L/U}$(AMAD=10), calculated using equation (4) versus parameter AMAD in comparison with values $\sum_{n=1}^{N}\sum_{k=1}^{K} M_L(t_n)/M_U(t_k)$ obtained in experiment.

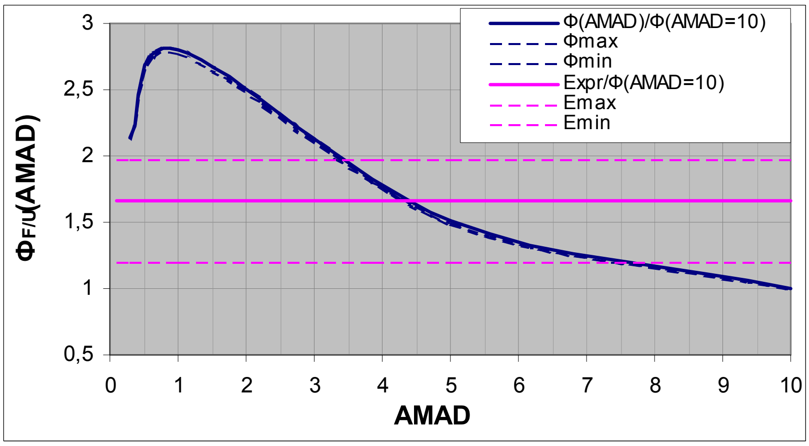

Fig. 3. Theoretical value of function $\Phi_{F/U}$(AMAD)/$\Phi_{F/U}$(AMAD=10), calculated using equation (5) versus parameter AMAD in comparison with values $\sum_{z=1}^{Z}\sum_{k=1}^{K} M_F(t_z)/M_U(t_k)$ obtained in experiment.

Parameter AMAD in working area vary in the range from 0.1 to 20 microns. Aerosols of desintegration usually have AMAD more than 1 micron and condensation aerosols - less than 1 micron. In figure 2 and 3 a theoretical value of the ratios $\Phi_{L/U}$(AMAD)/$\Phi_{L/U}$(AMAD=10) and $\Phi_{F/U}$(AMAD)/$\Phi_{F/U}$(AMAD=10) are presented as a function of AMAD. Horizontal lines are the values of the combination of measured results $\sum_{n=1}^{N}\sum_{k=1}^{K} M_L(t_n)/M_U(t_k)$ and $\sum_{z=1}^{Z}\sum_{k=1}^{K} M_F(t_z)/M_U(t_k)$ normalized to $\Phi_{L/U}$(AMAD=10) or $\Phi_{F/U}$(AMAD=10). Average, maximum and minimum values of the above quantitities for four workers are presented in figures 2 and 3. In all cases type S compound was used. It follows from the figures 2 and 3 that the avarage value of AMAD is in the range (4,5 - 5,5) microns and figure 3 data show a smaller range of (3-8) microns.

Results of the committed effective dose calculation for the four nuclear power-plant workers using equations (2) and (6) are presented in table 3.

Table 3. Results of the committed effective dose calculation for the four nuclear power-plant workers, mSv.

Worker	Dose estimate E_{50} (α=0,50)	Confidence interval		$(E_{max}-E_{50})/E_{50}$, %
		E_{min} (α=0,025)	E_{max} (α=0,975)	
W1	30.8 (20)	25.7 (5.2)	35.9 (51)	17 (155)
W2	13.6 (8)	11.1 (2)	16.8 (20)	24 (150)
W3	10.4 (7.1)	8.15 (1.8)	13.0 (17.3)	25 (144)
W4	17.9 (12.5)	13.9 (3.3)	23.0 (31)	28 (150)

For the dose calculation only in vivo measurements of Co-60 in lungs were used and the following assumptions were taken: Type A uncertainty for the in vivo measurements of Co-60 in lungs equals 1%, Type B uncertainty is described by lognormal PDF with geometric standard deviation equal 1.2, absorption type is characterized by the type S compound, AMAD vary in the range 3-8 microns with uniform probability. Normal PDF is taken as a likelihood function. In table 3 the estimate with a confidence level α=0.5 is taken as the dose estimate and the estimates with confidence levels α=0.025 and α=0.975 are taken as lower and upper limits of the confidence interval. In brackets the same estimates are shown but calculated using the first measurement only and assuming that absorption type is unknown (M or S) and AMAD varies in the range 0.1 – 10 microns.

Conclusions

The presented example of a numerical method application for the interpretation of bioassay measurements of Co-60 performed during special monitoring of four nuclear power-plant workers showed a possibility of improving the dose assessment by obtaining better information on the model parameter values and by using statistical methods for the calculation of the associated uncertainty. From the two compounds used in the ICRP model for cobalt the type S was taken as the best option and the range of AMAD values was reduced from (0,1 до 20) to (3 – 8) microns. This result in the reduction the uncertainty of the dose assessment from (144-155) % to (17-28) %.

The presented committed effective doses for the nuclear plant workers are the estimates with a confidence level α calculated in accordance with the ICRP model and with the assumptions of the model parameter values and the assumed likelihood function for the measured quantities.

On the other hand the parameter model investigation showed that absorption type S compound is not slow enough to describe the retention of the radionuclide Co-60 in lungs in the case under study and a slower type like super S is needed.

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Comparison of the three most recent biokinetic models for plutonium

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Abstract

Biokinetic models for the assessment of individual internal doses represent the physiological processes in the human body which affect the internal distribution of the radionuclide of interest. The International Commission on Radiological Protection (ICRP) recommends compartment models which emerge from a development and refinement of originally crude approximations. The three most recent systemic models for plutonium were studied: the ICRP67 model, the model proposed by Luciani (Luciani and Polig) and the Leggett model (Leggett et al.), which will find its way in future ICRP recommendations.

The three models represent the physiological processes of a standardized human. The flow from one compartment into another is specified by transfer rates, which may diverge from person to person. The sensitivity of a model regarding changes in one transfer rate can be analysed with Monte Carlo simulations. Simulations show that only few transfer rates have an impact on the measurable excretion rates. The influence of a changed transfer rate on the excretion rates is time dependent and can cause a large variation once, while it is without big effects at other times.

For dose assessment with unknown incorporation scenario the variation of the amount of incorporated plutonium was estimated with Bayes methods using excretion rates for the three models and considering individual variations. Incorrect presumptions about particle diameters, f1-values, solubility, wound category … were assumed for inhalation, ingestion and invulneration. They result in considerable different incorporations and consequently different doses. Thus the effort to determine correct incorporation conditions should not be neglected.

Introduction

For the assessment of individual internal doses of occupationally exposed workers and for members of the public the International Commission on Radiological Protection (ICRP) proposes biokinetic models. The most important component of internal exposure for workers is often provided by plutonium and other actinides. Beside the recommended model for plutonium (ICRP 67, published 1993) there are two newer models: the Luciani model and the Leggett model (Fig.1). Luciani modified the ICRP 67 model eliminating unphysiological assumptions and changing the bone model to the more realistic model from Polig. In the year 2005 Leggett improved the ICRP67 model to reflect later developed data. His model will find its way in future ICRP recommendations.

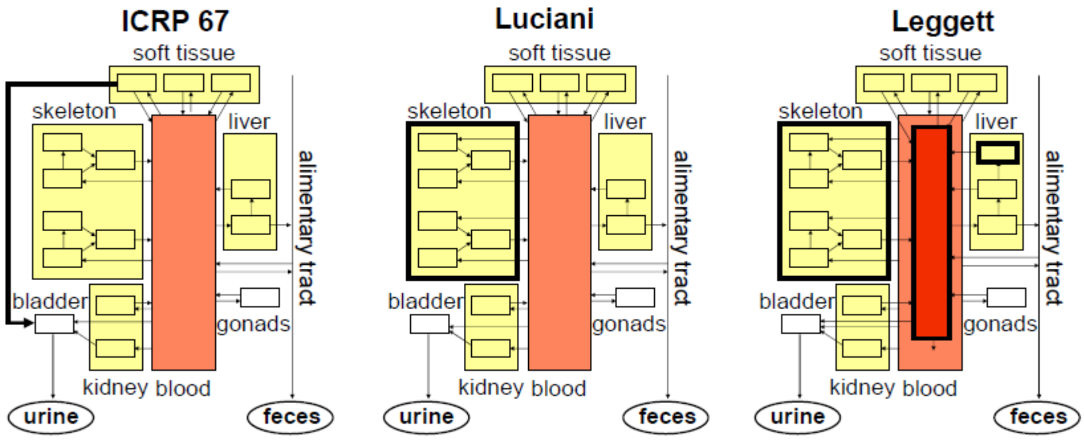

Fig. 1. Model structure of Pu-models from ICRP 67, Luciani and Leggett.

Air sampling and in vivo measurements can provide information on plutonium intakes, but in most cases the primary source of information is bioassay data obtained from urine and faecal analysis. In order to deduce intakes from excretion data, it is necessary to compare the measured amount of plutonium found in urine and feces with the model predictions (Fig. 2). Biokinetic models represent the physiological processes of a standardized human. In general the metabolism of an individual differs from the standardized one. Differences can be modelled with changes in model parameter values. Studies of the effect of these changes on the data of interest is called sensitivity analysis and can give some hints on critical model parameters and best date for excretion monitoring for dose assessment. If the incorporation scenario is unknown assumptions about the intake modalities have the main impact on dose assessment. Incorrect presumptions about particle diameters, f1-values, solubility, wound category … for inhalation, ingestion and invulneration lead to major errors in the estimation of occurred intake.

Material and methods

For sensitivity analysis transfer rates were generated to quantify the impact of changes in one transfer rate on the plutonium excretion within 10,000 days with Monte Carlo methods. As most of the measurable body parameters (e.g. body size, …) are lognormally distributed over the whole population a lognormal distribution was assumed also for transfer parameters. Previous studies indicated that independent variations of all transfer rates with a coefficient of variation of about 60% are able to represent the inter-individual variation for the biokinetic plutonium model best. Thus for the simulations of the sensitivity analysis each transfer rate fluctuates separately according to a lognormal distributions with the mean from the biokinetic model and a coefficient of variation of 60%. In this case Monte Carlo simulations lead to distributions for the predicted excretion rates with time variant statistical spread.

In real plutonium cases most intakes occur accidentally. Then in addition to the individual transfer rates the modalities of incorporation are uncertain as well. The effects of incorrect assumptions about particle diameters, f1-values, solubility, wound category … can also be examined with Monte Carlo methods. They can exceed effects from inter-individual differences by far. A 10,000 days monitoring period after incorporation was chosen. Virtual monitoring data was created exponential distributed during this period with at least 5 datasets. Incorporation modalities were chosen randomly (e.g. AMAD 1, AMAD 2, AMAD 5, AMAD 10 respectively solubility S, M, F) while the intake was calculated with standard assumptions (inhalation: AMAD 5, solubility M) and lognormally distributed transfer rates with coefficient of variations of 60%.

Results

As the decay of Plutonium produces mainly alpha particle with low penetration depth incorporated Plutonium commonly cannot be detected outside the human body. Thus dose assessment bases on bioassay data from urinary and fecal excretions and most human data derives from there. Therefore a correct reproduction of the observed excretion data is one of the most important aims of the biokinetic models. As a consequence the different models show only small differences in their charts for excretion rates of the standardizes (medium) subject while the observed data differs with a coefficient of variation of about 60-80% due to inter-individual variation (Fig.2).

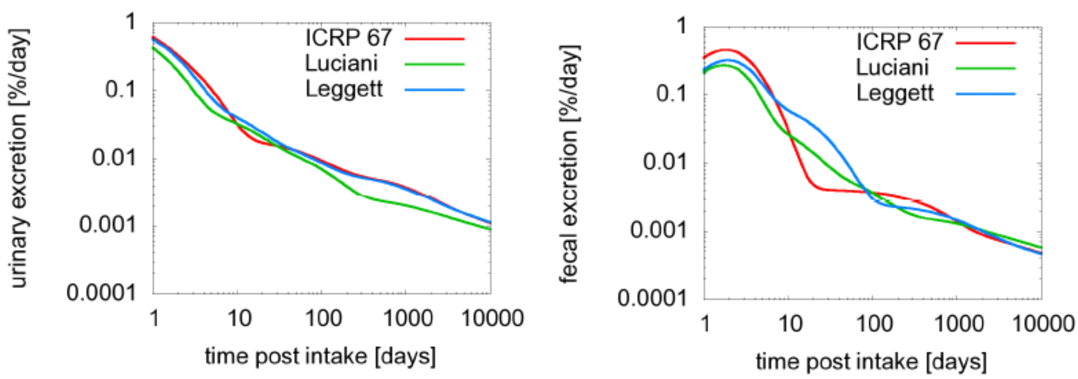

Fig. 2. Differences in model predictions for urinary and fecal excretions.

These individual differences can be modeled with changes in model parameter values. Parameter distributions and variations are unknown for the different models. As physiological parameters are distributed lognormally in the majority of cases, it can be assumed that the transfer rates of biokinetic models fluctuate according to lognormal distributions as well. Previous studies indicated that the observed inter-individual variation can be attained with independent variation of all transfer rates with a coefficient of variation of 60%. As it is unlikely that all parameters fluctuate similarly the influence of single parameters on the excretion rates should be studied in so-called sensitivity analysis. Therefore simulations for each transfer rate fluctuating separately according to a lognormal distributions with the mean from the biokinetic model and a coefficient of variation of 60% were compared with the standardized model. The study showed that only few model parameters determine the variation of the complete model while the contribution of other parameters can be neglected (Fig.3). The influence of the parameters is highly time variant. While one transfer rate determines the total variation of the excretion rate for one time it can be subordinate at other times.

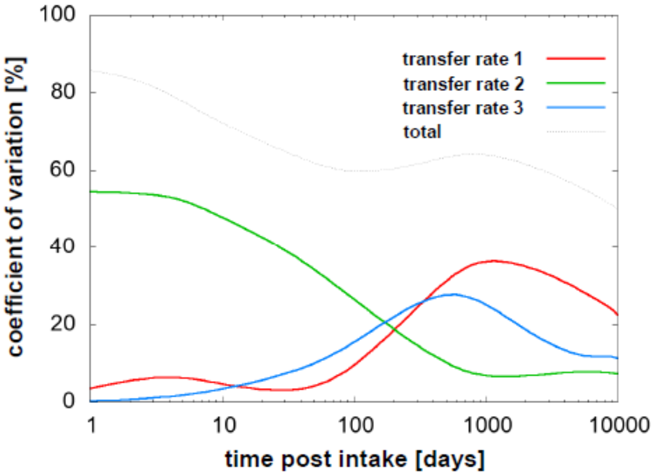

Fig. 1. Sensitivity analysis of the ICRP 67 model for urinary excretion.

In addition to the individual transfer rates the modalities of incorporation are often uncertain as well. The effects of incorrect incorporation assumptions exceed normally the effects of individual differences on excretion rates and thus on dose assessment. They can also be examined with Monte Carlo methods. Therefore intake calculation based on standard assumptions while excretion datasets were generated with different assumptions taking inter-individual variation in account.

The resorption in the alimentary tract is an example for a well studied parameter, which depends on the chemical form and on individual metabolism. Under experimental conditions (identical chemical form) values between $2 \cdot 10^{-3}$ and $12 \cdot 10^{-3}$ were observed for plutonium. The biokinetic models assume a medium of $5 \cdot 10^{-3}$. Dose assessment with Bayes methods shows a large under-/overestimation depending on the biokinetic model (Tab. 1).

Table 1. Influence of resorption fraction f_1 differing from standard value ($5 \cdot 10^{-3}$) on dose estimation.

model \ f_1	$2 \cdot 10^{-3}$	$12 \cdot 10^{-3}$
ICRP 67	-50%	+170%
Luciani	-60%	+150%
Leggett	-60%	+170%

For inhalation cases the particle diameters of the aerosol is an import parameter for the deposition of activity and the particle transport in the respiratory tract. It is specified as activity median aerodynamic diameter (AMAD). For reference workers the recommended value is AMAD = 5 µm (AMAD 5) if no further information about the particle size is available. Obviously the deposition and particle transport has an influence on the absorption into the body and consequently on the excretion rates and dose assessment. Nevertheless the effect on dose assessment is relatively small (Tab. 2). Monte Carlo simulations show a maximum mean difference of 21% for wrong particle size assumptions.

Table 2. Influence of aerosol particle size differing from standard value (AMAD 5) on dose estimation.

Particle size / model	AMAD 1	AMAD2	AMAD 10
ICRP 67	+3%	+10%	-18%
Luciani	+3%	+11%	-16%
Leggett	+13%	+16%	-21%

In contrast to the small influence of the particle size the correct assumption about solubility is crucial for dose estimation (Tab. 3). Different chemical forms are subdivided in only three solubility classes. The difference between estimated dose and true value lies in a range of -95% to +510% with wrong assumptions.

Table 3. Influence of solubility class differing from standard solubility class M on dose estimation.

Solubility / model	F	S
ICRP 67	+500%	-95%
Luciani	+510%	-95%
Leggett	+430%	-95%

Conclusions

Uncertainties in dose estimation have been calculated for the three latest biokinetic models for plutonium. In a first step the most sensitive transfer rates for urinary and faecal excretions were identified. The effort to determine individual transfer rates should concentrate mainly on these few parameters. In a second step the influence of individual variations and incorrect presumptions about particle diameters, f1-values and solubility were studied. The different sources of uncertainties lead to small under- respectively overestimates for the particle size (-21% to +13%) and bigger ones for the resorption in the alimentary tract (-60% to +170%) and the solubility class (-95% to 510%). Correct assignment of model parameters linked to the incorporation scenario like solubility, resorption and particle size can be much more important for dose estimation than the exact determination of the individual transfer rates of the systemic biokinetic model.

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Biokinetic models
Analysis of a numerical technique

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Abstract

Occupational exposure leading to intakes of internally incorporated radionuclides can occur as a result of various activities. This includes work associated with the use of radioactive sources in medicine, scientific research, agriculture and industry, and occupations which involve exposure to enhanced levels of naturally occurring radionuclides. The application of biokinetic models are a fundamental issue in estimation of doses and risks associated with incorporation of radionuclides within the human body. In biokinetic models of radionuclides, there are several techniques available to solve the systems of first-order differential equations. Here we present a straightforward implementation of the difference equations using an available computational spreadsheet like the EXCEL from Microsoft. It was chosen a well known model as a reference, allowing us to check the rigor of the technique. The main scope of this work is to make a detailed computational analysis of the role of all the parameters involved in this methodology. We obtained an accurate insight of the implemented technique, which validates its application in more complex biokinetic models where reference solutions are non-existing or difficult to obtain. In this technique of difference equations there are specific parameters such as time step values and compartment values for each time step that need to be well defined and well understood. Using appropriate algorithms, we developed techniques to optimise those parameters in order to obtain accurate solutions of complex compartment models. Moreover, in the scope of the Internal Dosimetry group activities at the ITN this tool will be of extreme value in order to estimate dose of measured personnel. Finally, another aim of this work is to provide a flexible tool that allows the study of biokinetic models with fast and easy addition or removal of compartments in any given model under analysis.

A methodology for studying cerium biokinetics using stable tracers and Thermal Ionisation Mass Spectroscopy

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Abstract

The double tracer technique as introduced in the 60s is a potent method to investigate the biokinetics of elements in humans. The technique was modified and applied over the last years to various elements of interest for radiation protection in order to validate and improve the available biokinetic models of ICRP. One project currently running at Helmholtz Zentrum München aims at studying the biokinetics of cerium in humans using stable, i.e non-radioactive tracers. For this purpose a methodology needs to be developed which enables to measure simultaneously different stable cerium isotopes in human body fluids such as blood and urine Thermal Ionisation Mass Spectroscopy (TIMS) using the instrument "Triton" from Thermo Scientific GmbH, Bremen, Germany. The measurement protocol was developed using cerium standard solutions with different concentrations and different combinations of filament materials. The best results were achieved adopting the double filament configuration and using the tantalum filaments. The optimal current for the ionisation filament is about 4700 mA and for the evaporation filament between 1250 mA, the temperature is around 1700°C. Under these experimental conditions the measured standard ratios agree within 1% with the IUPAC values. For measuring biological samples with TIMS a chemical treatment is necessary because it is not possible to coat the sample holder directly with the sample. A chemical method for eliminating all elements except cerium was therefore established, consisting in two steps: first blood plasma and urine samples are subjected to microwave-assisted acidic pressure digestion, and then cerium is extracted by column chromatography to obtain pure cerium salts, which can be optimally ionised in the TIMS ion source without interfering elements. The methodology was applied to biological samples collected during tracerkinetic studies in humans and preliminary results of urine excretion and plasma clearance of cerium will be presented.

First international intercomparison for Serbian Institute of Occupational Health EURADOS IC2008

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Abstract
Laboratory for personal dosimetry (LPD) in the Serbian Institute of Occupational Health, Belgrade (SIOH), has taken part for the first time in international intercomparison EURADOS IC2008. At the same time it was the first participation for any Serbian dosimetry service outside Serbia. LPD had participated in national intercomparisons organized in Former Yugoslavia during last twenty years.

The aim of this paper is to open the discussion on applying method, which usually stays hidden behind results.

Introduction
Preliminary results Intercomparison 2008 as presented at participants meeting, Braunschweig, 27 January 2009 (Figure 1) was free for access at EURADOS web site.

SIOH LPD is 38th participant.

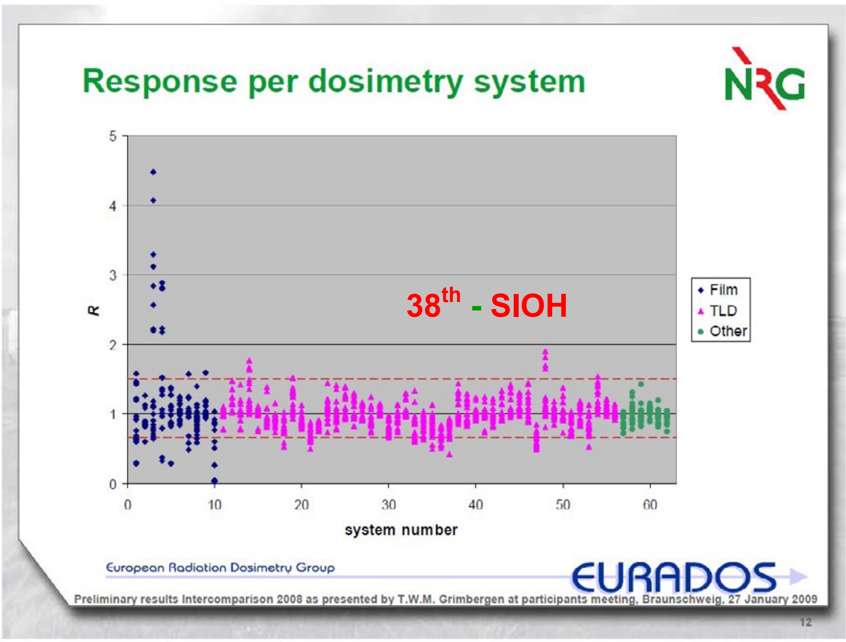

Figure 1. EURADOS IC2008 preliminary results for all participants.

Results

SIOH LPD reported results are given in Table 1, columns S7 and S8 together with important intermediary results.

Table 1. The main steps in the LPD procedure

0	S1	S2	S3	S4	S5	S6	S7	S8
tld	(ii) mikroSv	(iii) mikroSv	(ii) nC	(iii) nC	(ii)/(iii)	quality	Hp(10)mSv	Hp(0.07)mSv
12671	2968	3211	75.09	91.19	0.82	100 kV	3.13	2.86
12596	151000	159400	3820.3	4527	0.84	100 kV	159.18	145.63
12510	3115	3199	78.81	90.85	0.87	Cs-137	3.16	
12613	500.1	534	12.653	15.17	0.83	100 kV	0.53	0.48
12660	507.4	548.3	12.837	15.57	0.82	100 kV	0.53	0.49
12542	180.9	178.9	4.5768	5.081	0.90	Cs-137	0.18	
12682	4585	5153	116	146.3	0.79	60 kV	4.83	4.54
12551	3192	3535	80.758	100.4	0.80	60 kV	3.36	3.16
12642	175.2	166.3	4.4326	4.723	0.94	Cs-137	0.17	
12529	163100	167300	4126.4	4751	0.87	Cs-137	165.20	
12530	186	164.8	4.7058	4.68	1.01	Cs-137	0.18	
12704	3417	3757	86.45	106.7	0.81	60 kV	3.60	3.38
12608	10570	11110	267.42	315.5	0.85	100 kV	11.14	10.19
12639	2875	3059	72.738	86.88	0.84	100 kV	3.03	2.77
12569	3035	3132	76.786	88.95	0.86	Cs-137	3.08	
12527	3537	3867	89.486	109.8	0.81	60 kV	3.73	3.50
12571	10150	11220	256.8	318.6	0.81	60 kV	10.70	10.05
12706	4318	4810	109.25	136.6	0.80	60 kV	4.55	4.28
12590	2381	2507	60.239	71.2	0.85	100kV	2.51	2.30
12659	3389	3642	85.742	103.4	0.83	100 kV	3.57	3.27
12582	2438	2677	61.681	76.03	0.81	60 kV	2.57	2.41
12710	165.6	174.7	4.1897	4.961	0.84	100 kV	0.17	0.16
12700	175	177.5	4.4275	5.041	0.88	Cs-137	0.18	
12694	4589	4690	116.1	133.2	0.87	Cs-137	4.64	
12673	158.1	150.6	3.9999	4.277	0.94	Cs-137	0.15	
12707	4543	5081	114.94	144.3	0.80	60 kV	4.79	4.50

TLD has two crystals. The first step is to read-out dosemeters using calibration factors for S-Cs (columns S1 and S2). In the aim to estimate applied quality, inverse calculation gives signal intensity in *nC* (columns S3 and S4). Present calibration for quality appraisal is based upon results given in column S5. Column S6 represent estimation particularly important for Hp(0.07) calculation.

The biggest error of quality estimation was missing cobalt. Why? It is not yet answered.

Examples of good estimation are dosemeters ID 12694 and 12707. Nevertheless that was reported the same quality (S-Cs+N60-0^o) of irradiation for those two dosemeters the reference value ratio Hp(0.07)/Hp(10) 0.98 and 0.95 (Table 2.) indicates different qualities. It is possible that S-Cs and N60 quotum in these two irradiations were different. This is very close to estimation in Table 1: irradiation quality for ID 12694 Cs (Hp(0.07)/Hp(10)=1); for ID 12707 N60 (Hp(0.07)/Hp(10)=0.94).

Table 2. Comparison of LPD results and reference values

		Hp(10)			Hp(0.07)		
TLD ID	Quality	Your value (mSv)	Reference value (mSv)	Ratio	Your value (mSv)	Reference value (mSv)	Ratio
12671	N150-45°	3.13	3.10	1.01	2.86	3.04	0.94
12596	S-Co	159.18	185.00	0.86	145.63	185.00	0.79
12510	S-Cs	3.16	3.20	0.99	-	3.20	
12613	S-Cs	0.53	0.40	1.33	0.48	0.40	1.20
12660	S-Cs	0.53	0.40	1.33	0.49	0.40	1.23
12542		0.18	NIR		-	NIR	
12682	S-Cs + N60-0°	4.83	4.20	1.15	4.54	4.00	1.14
12551	N60-45°	3.36	2.60	1.29	3.16	2.64	1.20
12642		0.17	BGR		-	BGR	
12529	S-Co	165.2	185.00	0.89	-	185.00	
12530		0.18	NIR		-	NIR	
12704	N60-0°	3.60	2.90	1.24	3.38	2.72	1.24
12608	S-Cs	11.14	12.00	0.93	10.19	12.00	0.85
12639	N150-45°	3.03	3.10	0.98	2.77	3.04	0.91
12569	S-Cs	3.08	3.20	0.96	-	3.20	
12527	N60-45°	3.73	2.60	1.43	3.50	2.64	1.33
12571	S-Cs	10.70	12.00	0.89	10.05	12.00	0.84
12706	S-Cs + N60-0°	4.55	4.60	0.99	4.28	4.52	0.95
12590	S-Cs	2.51	2.70	0.93	2.30	2.70	0.85
12659	N60-0°	3.57	2.90	1.23	3.27	2.72	1.20
12582	S-Cs	2.57	2.70	0.95	2.41	2.70	0.89
12710		0.17	NIR		0.16	NIR	
12700		0.18	NIR		0.16	NIR	
12694	S-Cs + N60-0°	4.64	4.60	1.01	-	4.52	
12673		0.15	BGR		-	BGR	
12707	S-Cs + N60-0°	4.79	4.20	1.14	4.50	4.00	1.13

Six dosemeters (grey rows, Table 1) had average dose 0.17 *mSv*. According routine practice background has not been subtracted.

It was possible to recalculate Hp(10) after quality estimation using adequate calibration factors (column S7) as well as Hp(0.07) (column S8) given in Table 1.

Discussion

Here is present only an example of many open issues for discussion and improvement.

International intercomparison involving a lot of dosimetry services should be great opportunity for experience exchanging.

Conclusions

Besides the prime aim to reach results inside trumpet curve and being in accordance with standards international intercomparison gives chance to find solutions for many open dilemma in everyday practice.

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Research and applications in the Laboratory for Environmental and Personnel dosimetry

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Abstract

This paper presents the activity of the Laboratory for Environmental and Personnel Dosimetry. This laboratory is a part of the Department of Life and Environmental Science from IFIN-HH.

The laboratory has 4 units; each is dealing with a specific activity:

– the dosimetric surveying unit which uses thermoluminiscent dosemeters for personnel (USD-TL). In USD-TL unit, the personnel monitoring is performed with thermoluminescent dosimetric system SD-TL which uses TL-detectors of LiF: Mg, Cu, and P with the measuring range 10µSv – 100mSv;
– the unit for gammaspectrometrical analyses;
– the unit for measuring the environmental radioactivity where the environmental samples (soil, surface and underground water, vegetal, aerosols and atmospheric sediments) are analyzed from the point of view of the gross alpha, beta and gamma activity. In the same unit, the environmental monitoring is performed also by TL system SDTM type with LiF: Mg, Cu, P detectors;
– the unit for photodosimetrical survey.

All these units are working according to the ISO/IEC: 17025:2005 requirements. They are recognized by the Romanian Nuclear Authority (The National Commission for the Control of the Nuclear Activity – CNCAN) and accredited by the RENAR (The National Accreditation Body).

The R&D activity of the laboratory is focused on the personnel and environment dosimetry:

– the development of survey methods for personnel using in TL dosimetry for the extremities;
– the behavior of the TL detectors in ultra-low radiation background in the salt mines in Romania and intercomparisons of the TL-measuring systems.

Introduction

Horia Hulubei National Institute of R&D for Physics and Nuclear Engineering, IFIN-HH, is deeply involved in nuclear physics research and technical nuclear activities. Important nuclear installations, such as: the UVVRS Reactor- under decommissionning, the U-120 Cyclotrone, the Van der Graaf Accelerator-Tandem, the Radioisotope

Production Centre, the Radioactive Waste Treatment Station, and the Multipurpose Irradiator-IRASM are situated on its site. A significant number of professionals are implied in the nuclear activities as workers. Consequently, the radiological survey of the personnel and environment of IFIN-HH and its neighbouring zone of influence, are a priority. A dedicated laboratory for this activity, as well as for developing new systems and methods was set up many years ago. At the present time the laboratory, known as the Laboratory for Dosimetry of Personal and Environment (LDPM), is a part of the Department of Life Physics and Environment (DFVM).

The basic and continuous LDPM activities are: (i) the personnel dosimetrical monitoring, both by the use of the thermoluminescent dosimeters (TLD) and photographic films; (ii) monitoring of the environmental radioactivity: directly, by the measurement of the radioactive content of air, water, fallout, soil and vegetation samples, and indirectly, by continuous dosimetrical monitoring by using high sensitivity TLD. The high quality of equipment and the refined methods of prelevement and measurement allowed for developing new special applications, such as: measurement of ultralow level background in underground laboratories, or in special radiological areas of interest.

The laboratory succeeded in accomplishment of requirements of the quality standard EN ISO/IEC 17025:2005, "General requirements for the competence of testing and calibration laboratories", being accredited by the national accreditation body RENAR, and designed as a notified testing laboratory by the nuclear authority, CNCAN; this implies the periodical verification of the quality of testing methods by the participation in intercomparisons or proficiency testing. This paper describes the activities developed in LDPM as a whole.

Material and methods

LDPM description

The LDPM is officially organized in 4 distinct units:

- The Unit for the dosimetrical survey of the personnel by using the Thermoluminescent Dosimeters (TLD);
- The Unit for Photodosimetric survey;
- The Unit for the measurement of the activity of the environmental and food samples by the gamma ray spectrometry;
- The unit for the measurement of the gross alpha, beta and gamma activity of environmental samples and gross gamma activity of air samples. A parallel continuous dosimetric monitoring is performed with a high sensitivity TLD system with LiF: Mg, Cu, P detectors, Model SDTM.

Dosimetrical survey of personnel with thermoluminescent dosimeters

Our system is used for the measurement of the dose equivalent within the interval: 10μSv 100mSv. It consists from:

- A TL reader type READER 770A
- The dosimetrical coded case, containing the TL detectors type LiF: Mg, Cu, P.The dosimeters are worn by the persons along a one month period on the breast zone. The integral dose equivalent is registered [1]. Then they are brought in the

laboratory, and the information is obtained from the TL READER. It is delivered both as a Hard Copy Dosimetrical Bulletin, delivered to the user, and as a Protected Stock File in the Personal Computer and as CD. It contains the following data: the person's name, the TL case code, the survey period, the value of the dose equivalent with its uncertainty for a k = 2 interval, [2] equivalent to the former 95% confidence level. Two times per year the results are reported to the CNCAN. The system is implemented for the whole staff of IFIN-HH.

The Unit for Photodosimetry

This monitoring is accomplished by the use of dosimeters with AGFA films [3, 4].

The measurement interval of dose equivalent with this system is: 0.2 mSv-1000 mSv for gamma-rays with energy within the interval 30-3000 keV, and 0.1mSv-1000 mSv with energies of 30-200 keV. In IFIN-HH, the system is used as a redundant means of survey, doubling the TLD basic system, but it is still used as the sole survey mean for the workers belonging to about 30 external Romanian units, outside our Institute.

Gamma ray spectrometry Unit

The gamma ray spectrometry system type provided with a HPGe detector and a software Model ORTEC DigiDART are used. In this laboratory, gamma spectrometry of the environmental samples, foods and underground areas measurements are done. His work and results are beyond the scope of this paper.

The Unit for the measurement of gross alpha, beta, gamma activity

The monitoring of the environmental radioactivity in the IFIN-HH and in its influence area consists from the prelevement, processing and measurement of environmental samples of: soil, surface and subterranean waters, spontaneous and cultivated vegetation, aerosols, rain and atmospheric fallout. The specified prelevement points and their frequency are established in agreement with the Monitoring Program of the Environmental Radioactivity approved by the CNCAN. The points are representative for the radiological characterization of the zone of interest. They are placed inside IFIN-HH area and outside it, on a radius of 5 km, preponderantly distributed on the direction NE towards SW, where the population representing the critical group lives. The prelevement is made by a qualified personnel, according to the technical procedures elaborated in the laboratory. All the samples are weighted by using the analytical balance Model WAX 220, 0.1-220 mg weighing interval. The processing of samples generally consists from the concentration of the specific sample activity by: evaporation, drying, roasting, depending of its physical shape.

After the processing, the samples are measured in terms of the gross alpha, beta, gamma activity. The equipments used for these measurements are:

– Installation for the measurement of gross alpha and beta activity in ultralow level background. Model 9300 PC-6 FL Soft VISTA 2000.
– Gross alpha/beta/gamma with low background and automatic sample changer Model S5 XLB- 6e, Soft ECLIPSE.

The systems are calibrated in terms of detection efficiency with standard sources, as follows. For the gross alpha activity measurement a ^{241}Am and respectively for beta activity a ^{90}(Sr+Y) standard sources are used. They are provided by the Radionuclide Metrology Laboratory from IFIN-HH, accredited and notified calibration laboratory, according to the EN ISO/IEC 17025:2005 standard, traceable to the International System of Units (SI).

Our laboratory participated also at two international proficiency testing exercises, organized by the National Physical Laboratory (NPL), UK, organized in 2008 and 2009, regarding the measurements of gamma-ray spectrometry, gross alpha,beta and gamma activity in solid and liquid samples. An extensive paper describing the monitoring of the radioactivity concentration in air was published in [3].

The environmental radioactivity survey by the sensitive TLD system

The measurement of the integral ambient dose equivalent [mSv] for a given monitoring period and the calculation of the ambient dose equivalent rate [nSv h^{-1}], due to the gamma radiations of the radionuclides present in the environment inside IFIN-HH and its influence area is done by using the Environmental TLD system. It was qualified and authorized by the CNCAN to be used as an official monitoring mean. It is similar to the personal survey system, but it is more sensitive. It is composed from the following parts:

– A TL reader type Analyzer RA'94
– The dosimeter for the environment monitoring. It is in shape of a plastic cylindrical container with external dimensions: Φ=50 mm and h = 9 mm and contains a three TL detectors based on LiF:Mg, Cu, P.

A detailed description of its dosimetrical characteristics is given in the paper [6].

The system was extensively used for monitoring of IFIN-HH area and its influence zone. The results are reported to the CNCAN as a component of the official Environment Survey Report. The results of a one year monitoring of IFIN-HH is also presented in paper [6].

This system was also used for special measurements, such as: (i) The monitoring of special places of interest in IFIN-HH [7] ; (ii) The measurement of the very low level background values in underground laboratories: the mine salt Slanic Prahova-Romania [8,9] and Praid Mures –Romania mine [10].

Results

The paper presents two types of results: (i) Results obtained by using the equipment for gross alpha, beta, gamma activity and (ii) Results in the validation of the Environmental TLD system used for environmental radioactivity survey.

(i) The results obtained for one year follow of the samples preleved, processed and reported in the Unit for the measurement of gross alpha, beta, gamma activity

The results, obtained according to the use of the equipment and procedures of the Unit, are presented in Table 1, as mean values of gross activity concentrations, for various samples.

Table 1. Mean values of gross activity concentrations, during the year 2008

Type of sample	Gross alpha	Gross beta	Gross gamma
Evacuation channel of water	-	(0.81±0.22 Bq/L)	-
Points in amonte and in aval of the evacuation channel to the river	-	(0.75±0.08) Bq/L	-
Drinking water	(0.030±0.008) Bq/L	(0.25±0.05) Bq/L	(0.10±0.02) Bq/L
Underground waters	-	(0.55±0.15) Bq/L	(0.21±0.08) Bq/L
Sediment	-	(710±90) Bq/kg	(280±36) Bq/kg
Soil	-	(810±135) Bq/kg	(250±50) Bq/kg
Spontaneous vegetation	-	(100±15) Bq/kg	(15±4) Bq/kg
Cultivated vegetation	-	(85±10) Bq/kg	(10±13) Bq/kg

(ii) Results in the validation of the Environmental TLD system, by participation in the intercomparison with the NPP Cernavada TLD system

The inercomparison consisted in the placement of a pair of dosimeters, one Model SDTM, IFIN-HH and another Model Panasonic- belonging to the Laboratory for Environmental Survey of the NPP Cernavoda. Ten points of interest, situated outside the NPP area, but inside of its influence area, were chosen. Both dosimeters were placed at a height of 1 m from the ground level and the integral ambient dose equivalent was registered for a period of 1 month. At the end of the exposition period thy were read separately in the two laboratories with the respective TL-READERS and the results, expressed in terms of ambient dose equivalent, H*(10) [mSv], were compared.

The mean values obtained for monitorized ten points have the following values:

– for the laboratory from IFIN-HH: 0.105±0.013 mSv;
– for the laboratory from NPP: 0.096±0.020mSv.

The combined standard uncertainty of the IFIN-HH measured values was 12% (k=1) and was calculated from the standard deviation of the 3 detectors' readings and the system calibration uncertainty, 7.5%. The uncertainty of the NPP values is expected to be the same. In order to establish the quality of comparison, the u-statistic Test, defined in [11] was applied.

$$u - statistic = \frac{|result\,1 - result\,2|}{\sqrt{uncerta\,\mathrm{int}\,y1^2 + uncerta\,\mathrm{int}\,y2^2}} = \frac{|result\,1 - result\,2|\%}{21.5\%} \qquad (1)$$

For these kinds of low level measurements for environment, the u-test is applied in the judgment of the agreement of results states that two result are in agreement if:

$$u - statistic \leq 3.29 \qquad (2)$$

In this comparison the maximum allowed difference is: (result 1 - result 2) =55.7%. That means that the reported values are in good agreement.

Discussion

The results presented above referred only to two types of measurements: direct measurement of the gross environmental radioactivity and indirect measurement by the high sensitivity TLD system, as other results were extensively presented in the following references .

(i) The results obtained for one year follow of the radioactivity concentration.

The final result of the measurements is the judgement on the accomplishment of the Legal limits of the Maximum Admitted Concentrations (MAC) . From the reported results, such limits are established for: (a) Surface waters: gross beta activity, less than 1.85 Bq/L; (b) drinking water: gross alpha: less than 0.1 Bq/L and gross beta: 1.00 Bq/L.

Such as it can be seen from Table 1, the reported values are under these legal limits. Regarding the other reported values: for gross beta and gamma, they are mainly due to the natural ^{40}K and uranium series, as well as to the remaining ^{137}Cs from the Chernobyl accident.

(ii) Validation of the Environmental TLD system in an intercomparison

The statistical analysis of the two independent results demonstrated that the reported values are in good agreement.

On the other hand, if one takes into account a 30 days =720 hours of exposition, from above mean values one calculates the mean values of the ambient dose equivalent rate for all locations , dH*(10)/dt, [nSv/h]: IFIN-HH value =145.83 nSv/h and NPP value =133.33 nSv/h. These two mean values are in very good agreement, difference 9.4%.

It is very interesting to compare these values with other reported literature data. They are a little higher that the world and Romania mean reported values for outdoor rates, 59 nSv/h [10,11] and comparable with previousely values reported for IFIN-HH area, respectively: (127±14) nSv/h in 2006, (107±11) nSv/h in 2007, (113±12) nSv/h in 2008 and (110±10) nSv/h in 2009. The explanation is that in Bucharest area, the contribution of radon is superior to Cernavoda, such as reported for the Bucharest area in general.

Conclusions

– A general presentation of the Laboratory for Dosimetry of Personal and Environment (LDPM) is done. It includes also the main fields of monitoring of the personnel and environment, the main equipment and the work procedures.
– The measurement results are in agreement with the legal imposed limits for the environmental radioactivity
– The quality of work is demonstrated by the accreditation and notification of the laboratory obtained and by the good results obtained in intercomparison with similar system in the case of environmental TLD measurements

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Mean energy required to form an ion pair (W value) for various ionizing particles in air

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Abstract

The mean energy required to form an ion pair (W value) is defined as the mean energy spent by the incident particle of energy E for the formation of an electron – ion pairs after complete dissipation of the initial energy, W = E/N, where N is the total number of produced electron-ion pairs. For high-energy particles in thin media, when only a fraction of the particle energy (dE) is deposited in a medium, it is necessary to consider the differential w value, w = dE/dN. For sufficiently high incident energy, W value is approximately constant, and w = W is a good approximation. Available data on W or w are often fragmented, dispersed, and missing systematic. Several compilations and reviews, such as ICRU Report 31 and IAEA TECDOC 799, provide a basis for assessing the present knowledge of W values for different charged particles in various gases. Most available data exist for electrons. The W value for high-energy electrons (>10 keV) in dry air is well defined, 33.97 ± 0.05 eV, and also a dependence on air humidity has been studied. W and w values for high-energy protons (>20 MeV) have been also carefully analyzed. The value of 34.23 eV, with ± 0.4%, has been recommended. The existing W or w data for heavy ions used in radiotherapy (He, C, Ne, Si, Ar) are fragmentary and most of them are measured for relatively low energies (< 1 MeV/amu) where major variations in energy dependence are observed. There is a lack of experimental data at energies > 1 MeV/amu of interest for radiotherapy. For carbon ions a few sets of experimental data on W exist at low energies and have been recently extended to higher energies. The average W value for C ions at energies > 10 MeV/amu is 34.7 ± 0.9 eV, which is slightly higher than the recommended value (34.5 ± 0.5 eV) for all ions at energies > 1 MeV/amu. There is a need for new W value measurements for heavy-ions in air at energies > 10 keV/amu, and especially for C ions of several hundreds of MeV/amu that are of interest in carbon-ion therapy.

Comparison of local ion density determined by recombination chambers with biophysical models

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Abstract

The basic parameter characterizing initial recombination of ions in ionization chambers is local ion density μ, defined here as a density of ions averaged over a short segment of an ionising particle track. It can be determined with a recombination chamber, which is. tissue equivalent, high-pressure ionization chamber operating under conditions of initial recombination of ions. It was deduced from theoretical models, that the length of the segment mentioned above or the size of cluster of ions in recombination chambers is equivalent to about 70 nm of unit density tissue. For practical reasons, the values of μ in the ion recombination models were often approximated by restricted LET, in order to correlate ion collection efficiency in the chamber with radiation quality factor.

In 2004 we published the values of μ, measured for low-LET radiations of energies from about 30 keV up to ^{60}Co gamma rays. More recently, D. Harder and co-authors considered the role of electron track-ends for dicentrics formation in human blood lymphocytes and published the curve showing the Monte-Carlo calculated number of correlated track-end groups per unit dose, relative to ^{60}Co gamma rays. There is striking similarity between this curve and μ(E) curve, especially when results of our new measurements, extending the range of E to lower energies are taken into account. The similarity can be easy explained because both phenomena- in the chromosoms and in the gas – depends on the distances between the points of ion formation. In our opinion, expressed also in earlier papers, μ is a physical, measurable parameter, associated with track segments of the length similar to important distances within the cell. The paper will present the reconsidered model of μ(E) dependence, for low and high-LET radiations, with track-ends taken into account, the set of μ values determined for different kinds of radiation and short comparison with biophysical models of dicentrics formation.

Energy and angular dependence of radiation monitors in standard X radiation beams

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Abstract
In Brazil, most of the monitoring instruments are calibrated in terms of exposure rate and dose rate in X, gamma and beta radiations, using reference systems calibrated at primary and secondary standard laboratories. In this work, the ionization chambers were calibrated in standard X radiation beams with the objective to study their response in relation to the energy and angular dependence in the operational quantity ambient dose equivalent H*(10), as recommended by the International Organization for Standardization (ISO 4037). For the energy dependence study, the ionization chambers were tested in standard beams, at the distance of 2.5 m from the tube. The X-rays system utilized was a Pantak/Seifert equipment, model MXR-160/22, with effective energies of 48, 65, 83 and 118keV, respectively for the N-60, N-180, N-100 and N-150 radiation qualities. For the angular response of the detectors, a special holder was made of PMMA with a goniometer for the monitor rotation. The results obtained were satisfactory, according to the international standards ISO 4037-1(ISO, 1996) and ISO 4037-3 (ISO,1999) and to the Brazilian standard ABNT-NBR 10011 (ABNT, 1987).

Introduction
Since the discovery of X-rays, the use of ionizing radiation has been intensified and expanded in several areas of applications, as in medicine and in industry. Many investments have been applied in research and technology. All this innovation requires the use of irradiators, X-ray equipment, particle accelerators, and radioactive sources. To regulate the use of ionization radiation, the International Commission on Radiation Units and Measurements (ICRU) and the International Commission on Radiological Protection (ICRP) established standards with the objective of the risk perception and optimization of radiation protection, using the ALARA principle ("as low as reasonably achievable"). One form of ionizing radiation control is through the use of measurement instruments such as dosimeters in the clinical area, and portable detectors, which quantify and qualify radiations. There are some types of clinical dosimeters made of various materials as well as different types of radiation detectors such as Geiger-Müller detectors, scintillation detectors, proportional detectors, semiconductor detectors, ionization chambers. Working with portable area detectors requires a quality control that is guaranteed in part by the calibration, which certifies the good functionality of the

equipment and offers reliability in their measurements. The instrument calibration can ensure if it is working properly and determines the indication of an instrument as a function of the measurement (the quantity intended to be measured).

In this work, the radiation monitors were calibrated at the Calibration Laboratory of IPEN which offers calibration services of gamma, X, beta and alpha radiation detectors. The objective was to study the energy dependence of radiation monitors for low and medium energy X-rays, using the ambient dose equivalent quantity, $H^*(d)$ and the angular dependence. Theses tests can show if the instrument is working correctly.

Materials and methods

The system of X radiation, in the Calibration Laboratory of IPEN, consists of a Pantak / Seifert, ISOVOLT HS 160, model 160, equipament with mean energies of 48 keV to 118 keV. The characteristics of the X radiation beams, used in this work, are described in Table 1. The caractrization of the X radiation beams, radioprotection level, was performed using the ionization chamber PTW, model w32002 – 35, calibrated at the German Laboratory Deutscher Kalibrierdienst (DKD) for the radiation qualities N-60, N-80, N-100, N-150, N-200 and N-250 (DKD, 1995). This ionization chamber has an expanded uncertainty of 3.0%, and it was used with the electrometer PTW UNIDOS, model 10001, with an uncertainty estimated as 0.5%, composing a measurement system with a total uncertainty of 3.5% for the value corresponding to the calibration factor. The uncertainties were determined according to the recommendations of the ISO 15572 (ISO 1998) and Brazilian standards (ISO 2003) for a coverage factor k = 2 and confidence level of 95.45%.

Table 1. Characteristics of X radiation beams (series of narrow spectrum), radiation protection levels established in the Pantak/Seifert system.

Radiation quality	Mean energy (keV)	Voltage (kV)	Half-value layer (HVL) (mmCu)	Additional filtration (mm)
N-60	48	60	0.25	0.6(Cu)
N-80	65	80	0.61	2.0(Cu)
N-100	83	100	1.14	5.0(Cu)
N-150	118	150	2.40	2.5(Sn)

Twenty one ionization chambers of six manufactures and eleven different models were tested in X radiation beams. Of the twenty one ionization chambers, nineteen were studied in relation to their energy dependence and four of them were tested in relation to their angular dependence.

Initially, the measurements of the exposure rates and air kerma rates were converted to ambient dose equivalent rates. The conversion coefficients from air kerma to ambient dose equivalent are presented in Table 2.

Table 2. Conversion coefficients from air kerma to ambient dose equivalent for the radiation qualities of ISO 4037-3 (ISO, 1999), at the reference distance of 2m.

Radiation quality	Conversion coefficients (Sv/Gy)
N-60	1.59
N-80	1.73
N-100	1.71
N-150	1.58

Then, the calibrations factors, C_f, were determined. The calibration factor is the ratio between the conventional true value (reference), C_{TV}, and the measured (indicated value), M, by the radiation detector, for the N-60, N-80, N-100 and N-150 radiation qualities (narrow beams), according to Equation 1.

$$C_f = \frac{C_{TV}}{M} \qquad \text{Equation 1}$$

For the angular response of the detectors, a special holder, Figure 1, was made of PMMA with a goniometer for the monitor rotation. The response of the ionization chambers in relation to the angular dependence was taken for the angles 0^0, $\pm\,15^0$, $\pm\,30^0$, $\pm\,45^0$, $\pm\,60^0$ and $\pm\,90^0$.

The characteristics of the ionization chambers tested in this work are presented in Table 3.

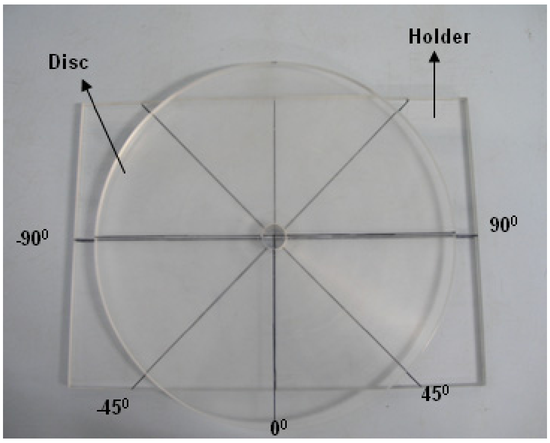

Fig.1. Goniometer for the angular dependence study of the ionization chambers.

Table 3. Characteristics of the ionization chambers.

Ionization chamber	Model	Chamber code	Radiation detected	Operating ranges
Babyline	81	A1	β, gamma, X	0 a 100 R/h
Fluke Biomedical	451 B-ryr	B1 e B2	β, gamma, X	0 a 50 R/h ou 0 a 500 mSv/h
Inovision	451P	C1 e C2	gamma, X	0 a 5 R/h ou 0 a 50 mSv/h
	451B	D1	α, β, gamma, X	0 a 50 R/h ou 0 a 500 mSv/h
Radcal	9010/10X5-1800	E1...E4	X	0,01 mR/h a 65 R/h
	9015/10X5-1800	F1	X	0,01 mR/h a 65 R/h
	2026C/20X6-180	G1	X	1 mR/h a 1 kR/h
Step	RGD 27091	H1	gamma, X	20 mSv/h a 2 Sv/h
Victoreen	450P	I1	gamma, X	0 a 5 R/h
	451B	J1...J4	α, β, gamma, X	0 a 50 R/h
	451P	K1...K3	gamma, X	0 a 5 R/h ou 0 a 50 mSv/h

Results and discussion

The calibration factors of the ionization chambers in standard X-ray beams were determined, and the results are presented in Table 4.

Table 4. Calibration factors of the ionization chambers for X radiation.

Chamber code	C_f: Calibration factor			
	(N – 60)	*(N – 80)*	*(N – 100)*	*(N – 150)*
A1	1.01	1.03	1.00	0.90
B1	0.88	0.81	0.78	0.81
B2	0.99	0.99	0.97	1.05
C1	1.09	1.05	1.05	0.91
C2	1.30	1.26	1.20	1.01
D1	0.96	0.93	0.95	0.99
E1	1.00	1.03	1.09	1.05
E2	0.98	1.01	1.08	1.06
E3	1.01	1.04	1.11	1.06
E4	0.98	1.01	1.09	1.06
F1	0.97	1.00	1.07	1.06
G1	1.01	1.04	1.09	1.04
H1	1.03	1.13	1.10	0.98
I1	1.20	1.14	1.15	0.91
J1	1.05	0.98	1.02	0.93
J2	1.11	1.00	0.96	0.82
K1	1.15	1.08	1.07	0.94
K2	1.12	1.07	1.07	0.95
K3	0.96	0.93	0.95	0.99

The calibration factor shall not be higher than 1,2 (IRD, 2004). In this case, the ionization chamber B2 would not be able to measure radiation intensity in the radiation quality N-100 and the ionization chamber C2 would not be able to measure the radiation intensity in the qualities N-60 and N-80.

The data of Table 4 can be visualized in Figures 2, 3 and 4. In the case of the ionization chambers D1 and H1, the response was presented for the radiation energies of the ^{137}Cs and ^{60}Co sources too.

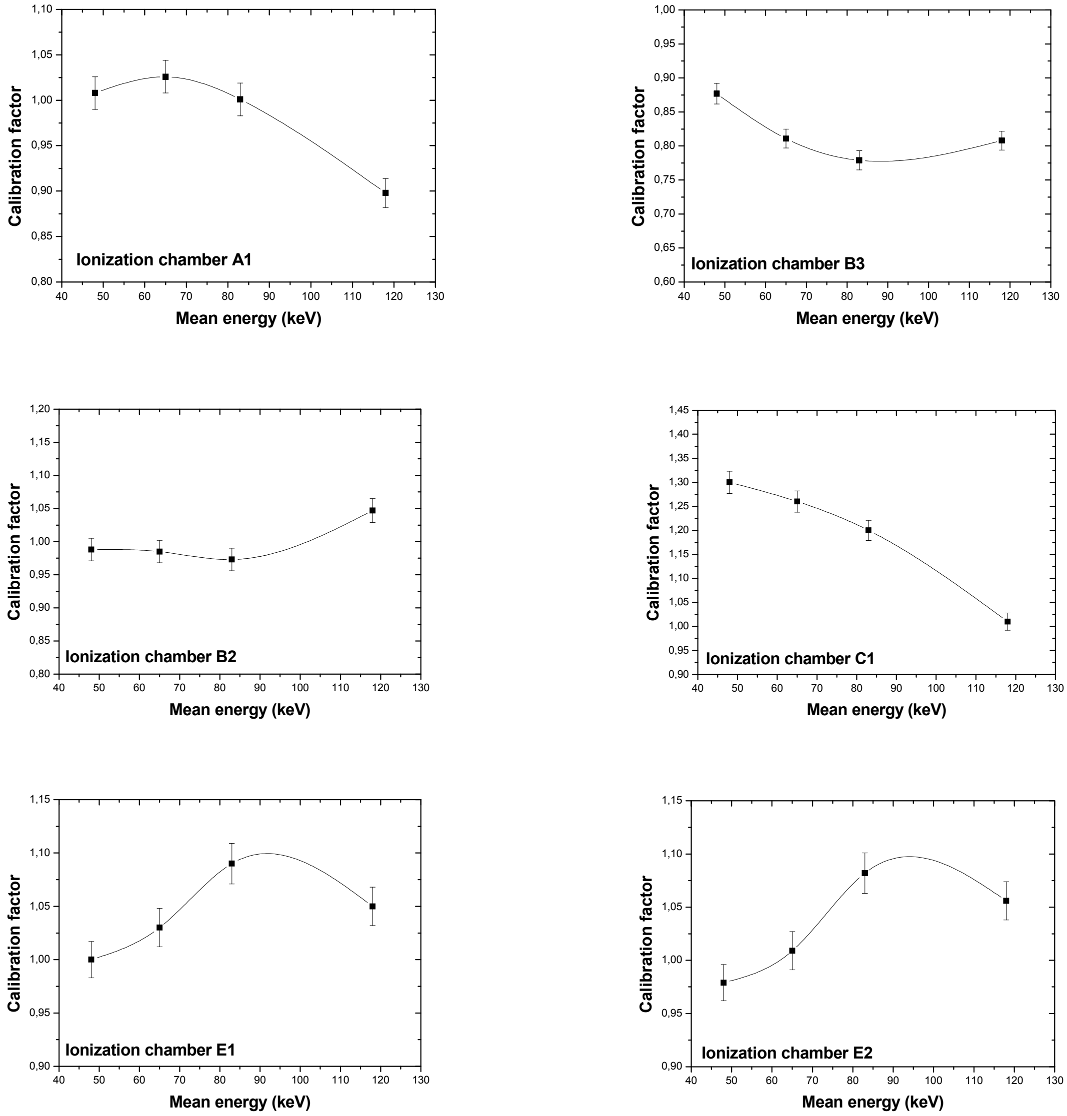

Fig.2. Calibration factors of the ionization chambers: A1, B1, B2, C1, E1 and E2 for different X radiation qualities.

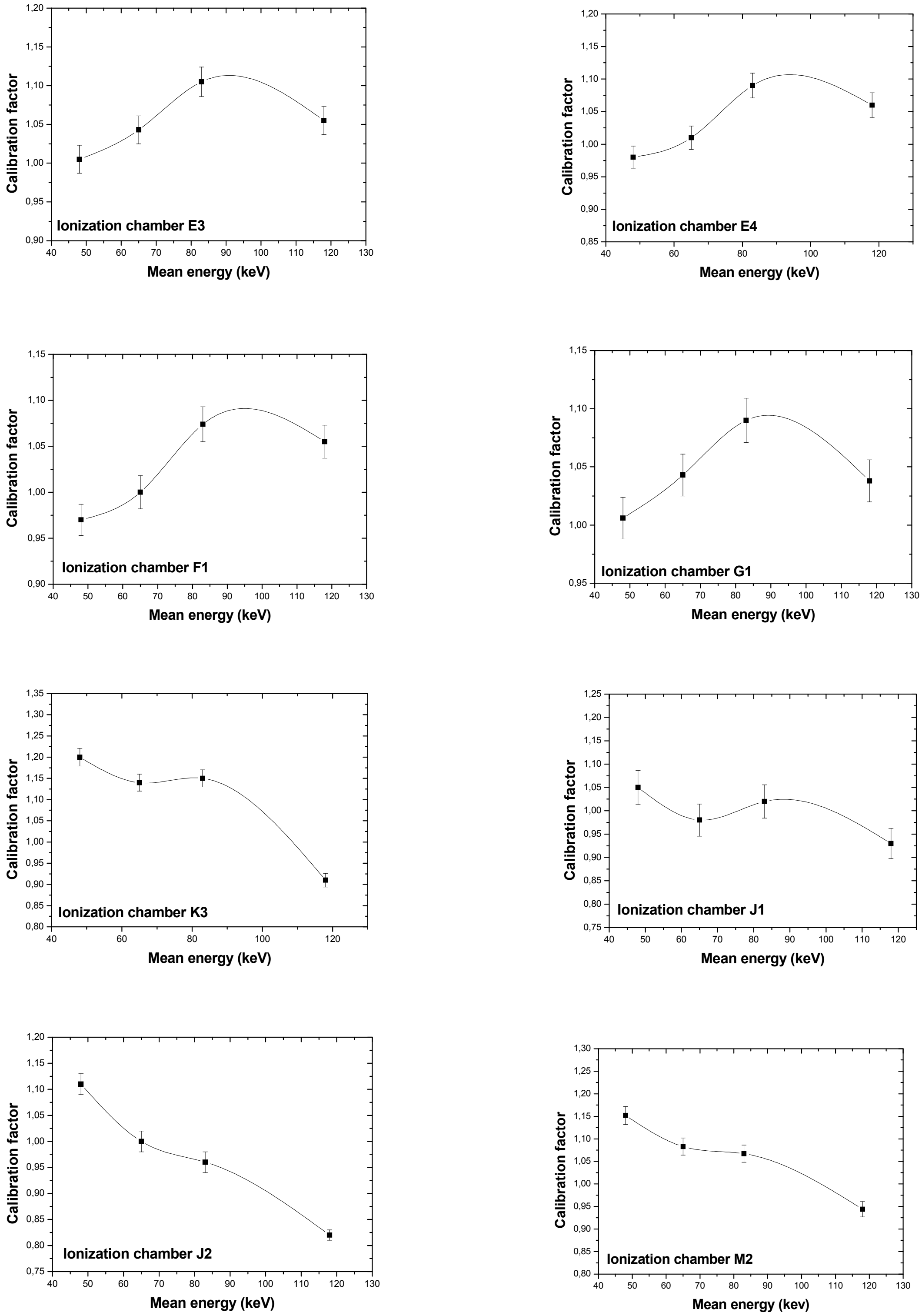

Fig.3. Calibration factors of the ionization chambers: E3, E4, F1, G1, I1, J1, J2 and K1 for different X radiation qualities.

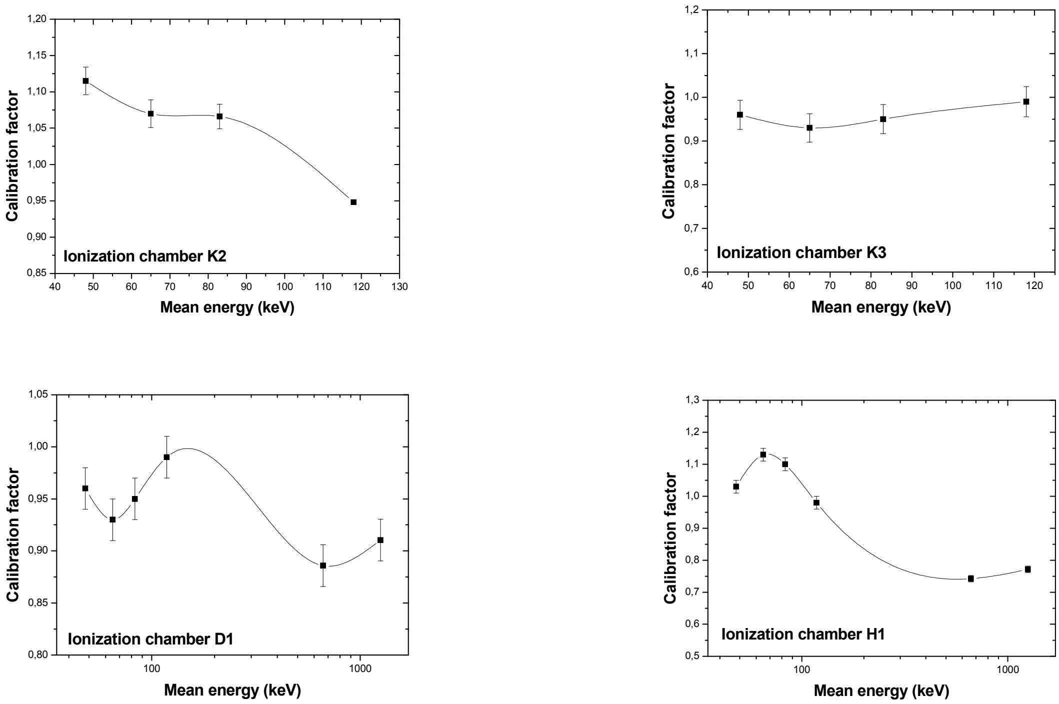

Fig.4. Calibration factors of the ionization chambers: K2 and K3 for different X radiation qualities; and D1 and H1 for gamma and X radiation qualities.

Different behaviors are observed in the response of the ionization chambers of different models. When comparing the curves, the behavior of the response of ionization chambers of the same model is quite similar, such as in the case of E1, E2, E3 and E4 ionization chambers. The curves obtained for the ionization chambers D1 and H1 in Figure 4 show that the calibration factors for X radiation is higher than the calibration factors for gamma radiation.

The energy dependence of these monitors is presented in Table 5, using the calibration factor data presented in Table 4.

Table 5. Energy dependence of the ionization chambers, radiation protection level, for X-rays (48keV-118 kev).

Chamber code	Energy dependence	Expanded uncertainty (U)
A1	1.14	0.07
B1	1.13	0.06
B2	1.08	0.06
C1	1.20	0.07
C2	1.29	0.06
D1	1.06	0.06
E1	1.09	0.06
E2	1.10	0.06
E3	1.10	0.06
E4	1.11	0.06
F1	1.10	0.06
G1	1.08	0.06
H1	1.15	0.06
I1	1.32	0.07
J1	1.13	0.06
J2	1.35	0.08
K1	1.22	0.07
K2	1.18	0.07
K3	1.06	0.06

The energy dependence of the ionization chambers varied between 6% and 35% (Table 5), according to the ionization chambers K3 and J2, which respectively showed energy dependence of (1.06 ± 0.06) and (1.35 ± 0.08). Monitors utilized at radioprotection level are divided into different classes, according to ABNT 10011 (ABNT, 1987). In this standard, the ionization chambers studied in this work are within the definition of Class II, where the maximum variation may be ± 25%. Only the ionization chamber J2 showed no agreement with the standard.

The angular dependence was obtained for the ionization chambers A1, C2, J3 and J4 without changing the measurement quantity to dose equivalent ambient. The two last models were not tested in relation to the energy dependence. The angle variation was from 0^0 to $\pm 90^o$ and only the radiation quality N-150 was utilized. The curves are presented in Figure 5.

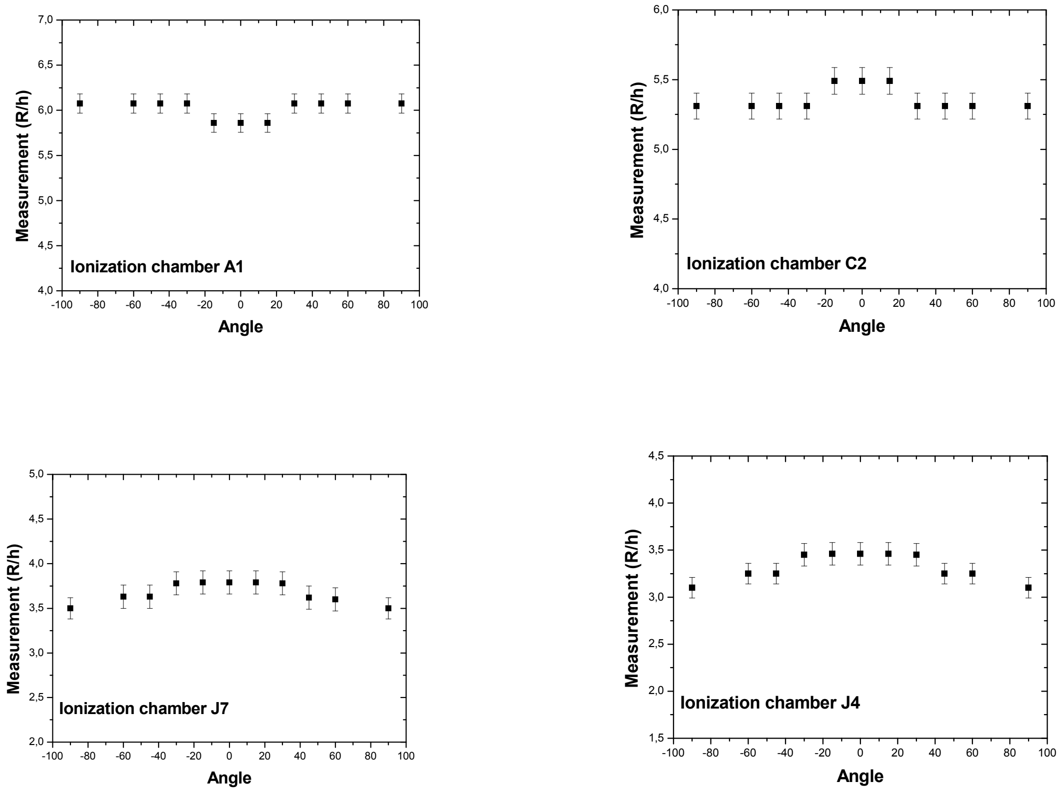

Fig.5. Angular dependence of the ionization chambers: A1, C2, J3 and J4; radiation quality: N-150.

The ionization chambers (Figure 5) presented low or almost no angular dependence for certain angle intervals. Their measurements are stable when varying the positioning of the ionization chamber in relation to the beam incidence. The ionization chambers J3 and J4 present higher angular dependence than models A1 and C2, because of their radiation entrance windows. The ionization chamber A1 has just an external wall (protection cap for build-up), but during the calibration with X-rays it is not utilized. The ionization chamber C2 is pressurized, and does not have an entrance window. All ionization chambers with x-rays agree with the standard EN 60846 (EN, 2004) that determines that the angular dependence may vary up to 40%.

Conclusions

The results obtained for the calibration factors and energy dependence for the ionization chambers were satisfactory, according to the ISO-4037-1 (ISO,1996); ISO-4037-3 (ISO,1999) and IAEA SRS16 (IAEA, 2000) recommendations. Only one of all nineteen ionization chambers presented an energy dependence higher than 25%. The energy dependence curves presented differents behaivors, depending on the ionization chamber model. The angular dependence of the monitors showed agreement with the standard EN 60846 (EN, 2004).

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A simplified OSL-algorithm for low-dose retrospective dosimetry using household salt

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Abstract
In retrospective dosimetry and for the purpose of geological and archaeological dating by optically stimulated luminescence (OSL) in materials such as quartz and feldspar, the single-aliquot regenerative-dose (SAR) protocol is widely used. This protocol is designed to account for the non-linear dose response between OSL-response and absorbed dose in e.g. quarts. The SAR-protocol normally involves five regeneration doses with subsequent read-out to be used for a signal-to-dose plot, enabling a calculation of the original absorbed dose in the sample. Currently the SAR-protocol is being optimised for NaCl and various brands of household salt, to be used as potential environmental and personal dosemeters. In this particular study we have investigated if only the initial part of the SAR protocol can be used for household salt, in an attempt to find a more rapid read-out protocol in case of nuclear and radiological emergencies. Stopping at only two regeneration doses appears to yield the same result within 2% as the full SAR-protocol for household salt, while at the same time saving up to 50% of read-out time per sample.

Introduction
Optically stimulated luminescence (OSL) has in the past few years been considered for retrospective dosimetry after nuclear emergencies [Bailiff 1997, Bøtter-Jensen and Jungner 1999, Thomsen et. al. 2002, Bernhardsson et. al. 2009]. The technique has for example been used for reconstruction of absorbed dose to human habitats after the Chernobyl accident [Bøtter-Jensen et. al. 2000]. In these events, burnt building materials such as bricks and tiles, containing quartz and feldspar crystals, have been used to retrospectively determine the absorbed dose. The sample preparation and measurements for OSL dosimetry of building materials are, however, altogether a laborious process. The minimum detectable absorbed dose (MDD) is estimated to be in the order of 30 mGy. It has been found that the MDD for OSL-determination of absorbed dose in household salt is much lower, down to 1 mGy [Thomsen et. al. 2002], and for some brands of salts even as low as 0.2 mGy [Bernhardsson et. al. 2009]. In combination with its relatively easy handling before OSL-read-out it has been suggested as an emergency dosemeter in case of nuclear and radiological accidents.

The OSL-response for various types of household salt and chemically pure NaCl has been investigated in Malmö using a TL/OSL reader (TL/OSL-DA-15; Risø National Laboratory, Roskilde, Denmark). In an on-going study, the SAR protocol originally used for quartz is now being modified to better fit the OSL-response and luminescence characteristics of household salt [Christiansson et al., 2010; Bernhardsson et al., 2009].

In the SAR protocol, a known additional dose, a so-called regeneration dose, is given to the sample after the optical destruction of the original OSL-signal in the sample, and the OSL-read-out is repeated. In all, five regeneration doses are given to the sample to determine the unknown dose. One problem with regeneration protocols using quartz and feldspar is the sensitivity changes in the sample when re-using the aliquot. This has been solved using a so-called test-dose after read-out of the regeneration dose. By normalizing the luminescence from each generation dose to that of the subsequent test dose, sensitivity changes during the process are compensated for. The SAR-protocol then makes use of the linear relationship between the detected luminescence and the given doses, and fits a linear expression to the dose-luminescence plot. From this linear expression the original luminescence of the sample is translated into an absorbed dose. A drawback of this method is the prolonged read-out times for the salt samples leading to a slow through-put of samples and delayed assessment of collected samples from e.g. an accident site. Our aim has been to test whether the initial part of the SAR-protocol can be used as a rapid substitute for the established SAR-protocol, provided the absorbed dose can be determined with sufficient accuracy (within 20% 1 SD).

Material and methods

The OSL measurements have been carried out using a TL/OSL reader (TL/OSL-DA-15; Risø National Laboratory, Roskilde, Denmark). The reader is equipped with an internal beta source ($^{90}Sr/^{90}Y$) of 20 MBq, which irradiates the sample at an absorbed dose rate of 0.9 mGy s^{-1} at the sample position. The stimulation light source was the 28 blue LEDs which are divided into four clusters, with a peak emission at 470 nm and a maximum power of 50 mW cm^{-2}. To detect the light from the stimulated sample, the reader contains a light detection system, which includes a PM-tube and light filters. The PM-tube is an EMI 9235QA PMT that has maximum detection efficiency around 400 nm. The light filters consist of a GG-420 long-pass filter that reduces contributions from the low end of the emission peak, and a 7.5 mm Hoya U-340 that reduces scattered light from the LEDs that reach the PMT. The reader is also equipped with a heater that can increase the sample temperature to any point between room temperature and 700°C.

In this part of the study we have specifically tested a salt brand that previously has shown the highest luminescence yield per unit given absorbed dose using a read-out algorithm based on the SAR-protocol. The salt is a mine salt (Falksalt fint bergssalt) aimed for household use. Before the measurements, the salt was bleached by exposing the salt to sunlight and fluorescent lamp-light for at least 24 hours, in order to reduce the background luminescence originating from e.g. prior exposure to background radiation. The salt was then sieved using “test sieves” (Retsch GmbH, Haan, Germany) to receive grains in the size range 100–400 µm. Upon the irradiation experiments the salt samples were divided into three subgroups which were subjected to an absorbed dose corresponding to 20, 50 and 100 s of irradiation by the internal ^{90}Sr-source, respectively.

The test dose read-outs were preceded with pre-heat temperatures ranging from 140°C to 200°C for 10 s, while the OSL-read-outs of the regeneration doses were preceded with a 220°C pre-heat for 10 s. All the OSL measurements were performed with a constant stimulation power (continuous wave) at 100°C for 40 s at 40% of the maximal stimulation power.

Results

The ratio between the dose measured and calculated with the SAR-protocol and the absorbed dose given to the samples (which are the regeneration dose of a given order 1st, 3rd and 5th) is plotted in Figure 1 for various pre-heat temperatures of the test dose. The pre-heat temperature at 140°C will not give the accurate dose value for one, three or five regeneration doses.

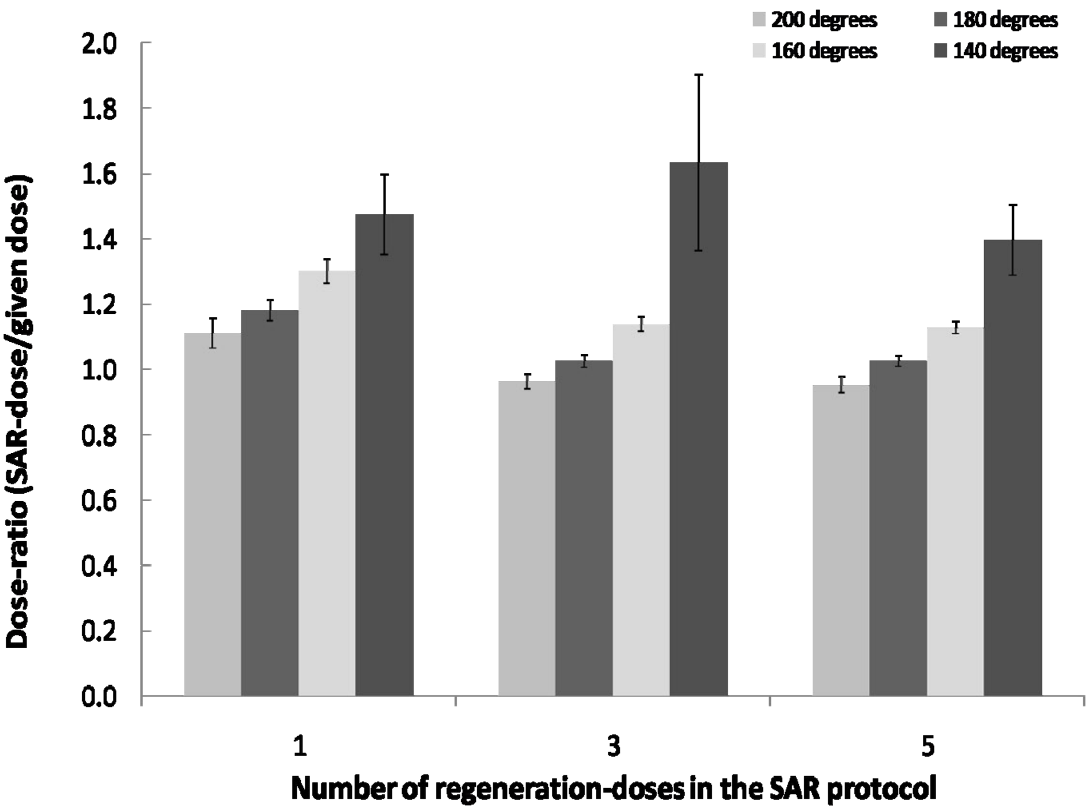

Fig. 1. The ratio between the dose measured with the SAR-protocol and the absorbed dose given to the samples of the investigated brand of household salt which are the regeneration dose of a given order (1st, 3rd and 5th). Different pre-heat temperatures ranging from 140 to 200°C have been used.

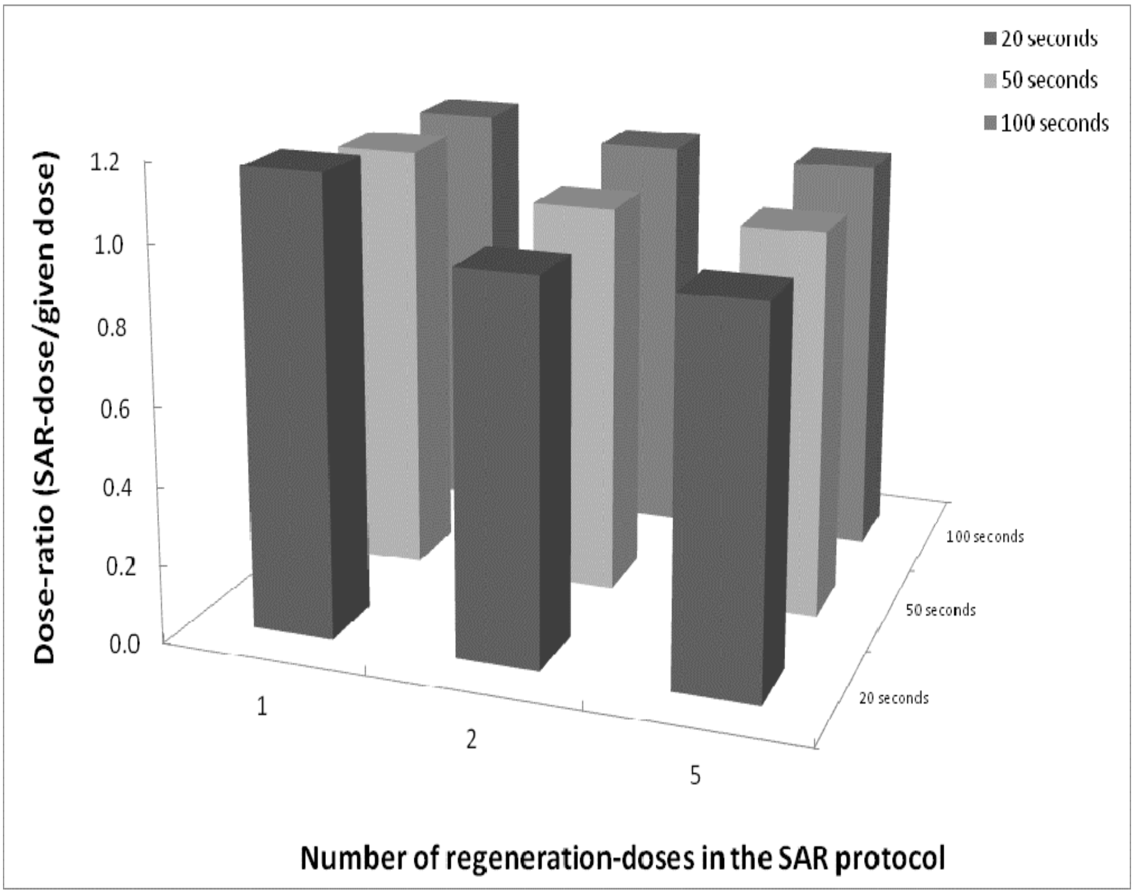

Fig. 2. The ratio between the dose measured with the SAR-protocol and the absorbed dose given to the samples is plotted for the pre-heat temperature 200°C. The given dose is received during 20, 50 or 100 seconds.

In figure 2 the ratio between the dose measured and calculated with the SAR-protocol and the absorbed dose given to the samples (which are the regeneration dose of a given order 1st to 5th) is plotted for the pre-heat temperature 200°C. In the latter figure it can be seen that at the second read-out, the ratio between the measured SAR-dose and the given dose appears to be sufficiently close to unity, and further regeneration will not improve this ratio drastically. Numerically the second regeneration dose protocol gives values that coincide within 2% of the ones from the full SAR-protocol.

Conclusions

With an optimised value of 200°C for the pre-heat temperature at the OSL-read-out protocol of household salt, two regeneration doses will give an adequate estimate of the given absorbed dose within 2% compared with the complete SAR-protocol of 5 regeneration doses. Employing the two-regeneration-dose protocol saves up to 50% of the reading time. Thus, this simplified SAR-protocol is recommended for use in case of a large nuclear or radiological accident involving many individual cases of exposures that need to be reconstructed.

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Uncertainty analysis of internal dosimetry using expert judgment and probabilistic inversion

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Abstract

The uncertainty of internal dose coefficient was analysed using expert judgment implemented by joint activity of U.S. Nuclear Regulatory Commission and European Community. In this study, the transfer coefficients in biokinetics model were taken to sources of uncertainty, and their probability distributions were estimated by iteration PARFUM of probabilistic inversion method with expert judgment. Their transfer coefficients sampled from their distribution were substituted into internal dosimerty model, and the uncertainty of dose coefficient for ingestion of ^{137}Cs by adult was analysed. The uncertainty factor (95%/ 5%) was approximately 5.1.

Introduction

The direct measurements of internal dose are difficult in most of cases. Then, the internal dose is estimated using both indirect measurements (e.g., bioassay and environmental concentration) and mathematical models which simulate metabolism and dose of radionuclides in body. The estimated internal dose includes errors of measurements and uncertainties of input parameters substituted to model which were estimated by human and laboratory animal data. In this study, the uncertainty of internal dose coefficient is analysed through propagation of uncertainty in input parameters.

The probabilistic distribution of input parameters is needed for implementing the uncertainty analysis of internal dose. Since their distribution cannot be obtained in most of cases, the method of expert judgment is applied to set probabilistic distribution of input parameters (Cooke et al., 1999). This expert judgment is subjective probability and is consisted of 5th, 50th and 95th percentiles which the quantity is well known by experts. The expert judgments do not always correspond to input parameters. Therefore probabilistic distribution of input parameters is estimated by probabilistic inversion with expert judgment. This study used the probabilistic inversion developed by Du et al. (2006).

The goal of the study is an estimation of uncertainty in internal dose coefficient for ingestion of ^{137}Cs by adult recommended by International Commission on Radiological Protection (ICRP). To implement this, the probabilistic distributions of transfer coefficients in biokinetics model of cesium are estimated by probabilistic inversion with expert judgment given Jones et al. (2001).

In this paper, it is described that the method of uncertainty for internal dose coefficient with expert judgment and the result of uncertainty of internal dose coefficient for ingestion of ^{137}Cs by adult.

Material and methods

In this study, the probabilistic distributions of the transfer coefficients in biokinetics model of cesium are estimated by probabilistic inversion with expert judgment. This section describes the biokinetics model, expert judgment, probabilistic inversion and analytical procedure.

Biokinetics model of cesium

The biokinetics model of cesium refers to model recommended by current ICRP. As illustrated in figure 1, this model is consisted of two compartments which are named Body Tissue A and Body Tissue B. Each compartments represent short and long retention.

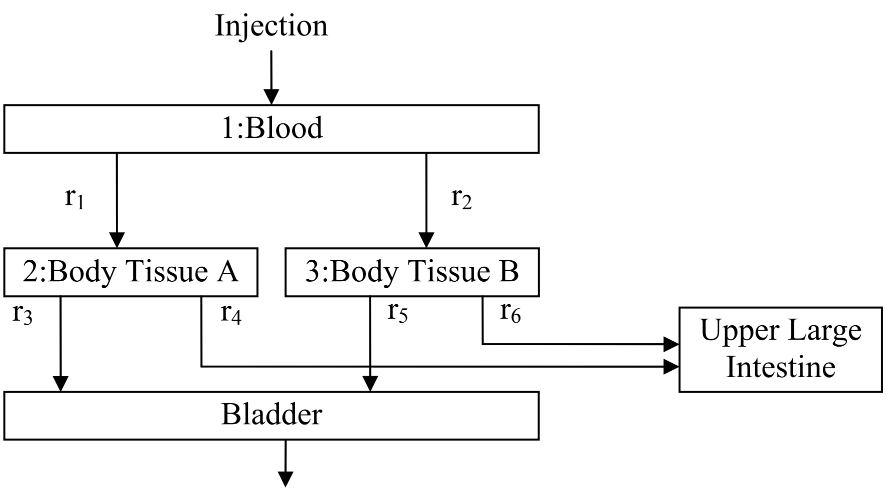


Fig. 1. Structure of ICRP's biokinetics model for cesium (r_i: transfer coefficients).

The retention of cesium in this model is described by first order kinetics. Suppose q_i is retention and r_i is transfer coefficient, the first kinetics is given by equation (1). When the initial conditions are $q_1(0)=1$ and $q_2(0)=q_3(0)=0$, their solutions are given by equation (2).

$$\begin{aligned}\frac{dq_1}{dt} &= -(r_1 + r_2)q_1 \\ \frac{dq_2}{dt} &= r_1 q_1 - (r_3 + r_4)q_2 \\ \frac{dq_3}{dt} &= r_2 q_1 - (r_5 + r_6)q_3\end{aligned} \tag{1}$$

$$\begin{aligned}q_1 &= e^{-(r_1+r_2)t} \\ q_2 &= \frac{r_1}{(r_3 + r_4) - (r_1 + r_2)}\left\{e^{-(r_2+r_2)t} - e^{-(r_3+r_4)t}\right\} \\ q_3 &= \frac{r_2}{(r_5 + r_6) - (r_1 + r_2)}\left\{e^{-(r_1+r_2)t} - e^{-(r_5+r_6)t}\right\}\end{aligned} \tag{2}$$

Expert judgment

The expert judgment of biokinetics model for cesium is illustrated in table 1. This expert judgment was given in Jones et al. (2001) and was represented by 5th, 50th and 95th percentiles which were retention fraction of body tissues by injection of cesium.

Table 1. Expert judgment of biokinetics model for cesium.

Time after Intake	Retention fraction of body tissues by injection of Cs		
	5%	50%	95%
1 day	8.70E-01	9.62E-01	9.92E-01
1 week	7.45E-01	8.59E-01	9.43E-01
1 month	5.45E-01	7.24E-01	8.93E-01
1 year	2.38E-03	6.48E-02	2.64E-01
5 years	1.21E-10	1.08E-05	6.30E-03

Probabilistic inversion

The probabilistic distributions of input parameters are needed to analyse uncertainty of dose coefficient through input parameters in biokinetics model of cesium. Their distributions cannot be obtained from data of measurements in most of cases. In such cases, the expert judgment of subjective probability is applied to uncertainty analysis. Since this expert judgment constructed quantity that experts well know, it does not always agree to input parameters. Therefore, the distributions of input parameters are needed to produce by probabilistic inversion from expert judgment. Though there were a few methods of probabilistic inversion, this study applied the method developed by Du et al. (2006). The uncertainty parameter of target refers to target variable and the variable asked to experts refers to elicitation variable.

In this study, the target variables are six transfer coefficients (r_1,...,r_6 in figure 1) and the elicitation variables are five retention fractions of body tissues at 1 day, 1 week, 1 month, 1year and 5 years. The 6-dimension space for target variables and 5-dimension space for elicitation variables are formed. The probabilistic inversion estimates the marginal distributions of space in target variables from the marginal distributions of space in elicitation variables. Since the target and elicitation variables are related to model equations and they are one-to-one mapping, the joint distribution in space of target variables is automatically evaluated if the joint distribution in space of elicitation variables is estimated. As a result, the marginal distributions of targets are evaluated from the joint distribution.

Since it is difficult to show figure of more than 3 dimensions, the probabilistic inversion for 2 dimensions about each of target and elicitation variables is explained as illustrated figure 2. The procedure of probabilistic inversion is as follows.

Step1 The analyst sets an appropriate sampling range of target variables and assumes uniform distribution as initial probability. The N sets are sampled from this region.

Step2 The sampled target variables are substituted for model equation and the obtained model values are mapped in space of elicitation variables. The N images are made in space of elicitation variables.

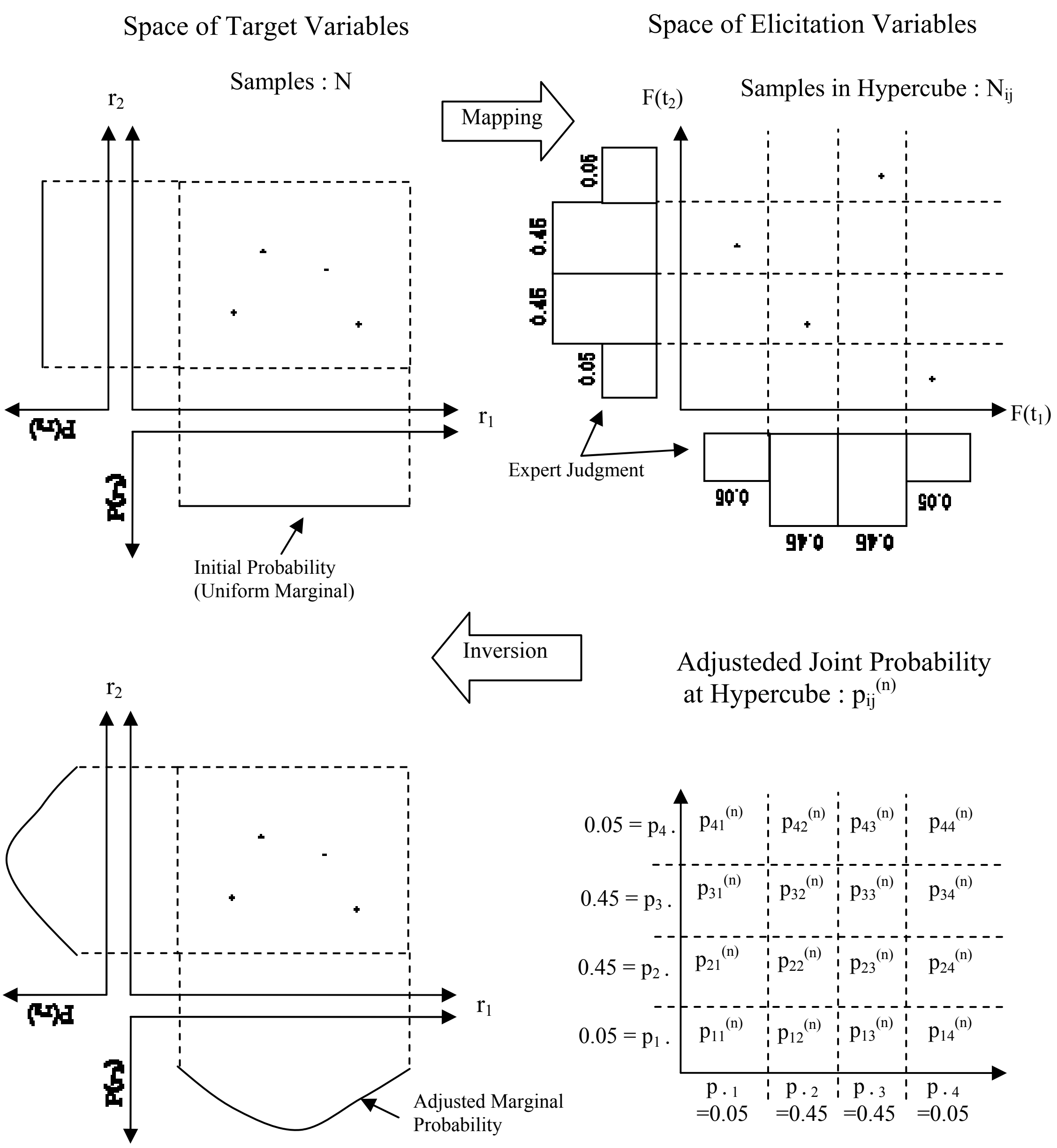
Step3 Counting number of points (N_{ij}, i,j=1,2,3,4) in hypercube partitioned 5th, 50th and 95th percentiles in space of elicitation variables.

Step4 The initial joint probability of hypercube are set to $p_{ij}^{(0)} = N_{ij} / N$. The iteration PARFUM is projected in n cycles and the joint probability of hypercube is adjusted as fitting to marginal distribution ($p_{i\cdot}$, $p_{\cdot j}$) of expert judgment.

Iteration PARFUM $$p_{ij}^{(n)} = \frac{1}{2}\left(p_{ij}^{(n-1)} \times \frac{p_{i\cdot}}{p_{i\cdot}^{(n-1)}} + p_{ij}^{(n-1)} \times \frac{p_{\cdot j}}{p_{\cdot j}^{(n-1)}} \right)$$

Step5 If the joint probability $p_{ij}^{(n)}$ could be evaluated, the probability of each points in hypercube is evaluated as $p_{ij}^{(n)} / N_{ij}$. Their probabilities are inversed to space of target variables and the joint probability of target variables is set up.

Step6 Consequently, marginal probability of target variables can be calculated from joint probability.



$$p_{ij}^{(n)} = \frac{1}{2}\left(p_{ij}^{(n-1)} \times \frac{p_{i\cdot}}{p_{i\cdot}^{(n-1)}} + p_{ij}^{(n-1)} \times \frac{p_{\cdot j}}{p_{\cdot j}^{(n-1)}} \right)$$

$$p_{ij}^{(0)} = \frac{N_{ij}}{N}$$

Fig. 2. Procedure of probabilistic inversion.

Analytical procedure for uncertainty of internal dose coefficient

The analytical procedure for uncertainty of model output caused by uncertainty of model input is shown to figure 3. The uncertainty analysis consists of two parts in this study.

The first part is that probabilistic distributions of model inputs are estimated by probabilistic inversion with expert judgment. The target variables are sampled by initial probability assumed uniform distribution. The PREP code (Homma et al., 1991) was used for sampling of target variables in this study. When Kullback-Leibler divergence with respect to initial probability of adjusted probability was evaluated after first probabilistic inversion, it is more than zero for inconsistent initial and adjusted distributions. The initial probability is replaced by adjusted probability and the probabilistic inversion is repeated to come closer zero. When the PREP code is used to sampling, their distributions approximate piecewise uniform distribution as illustrated in figure 3. When Kullback-Leibler divergence comes closer to zero, the distributions are set to final distribution.

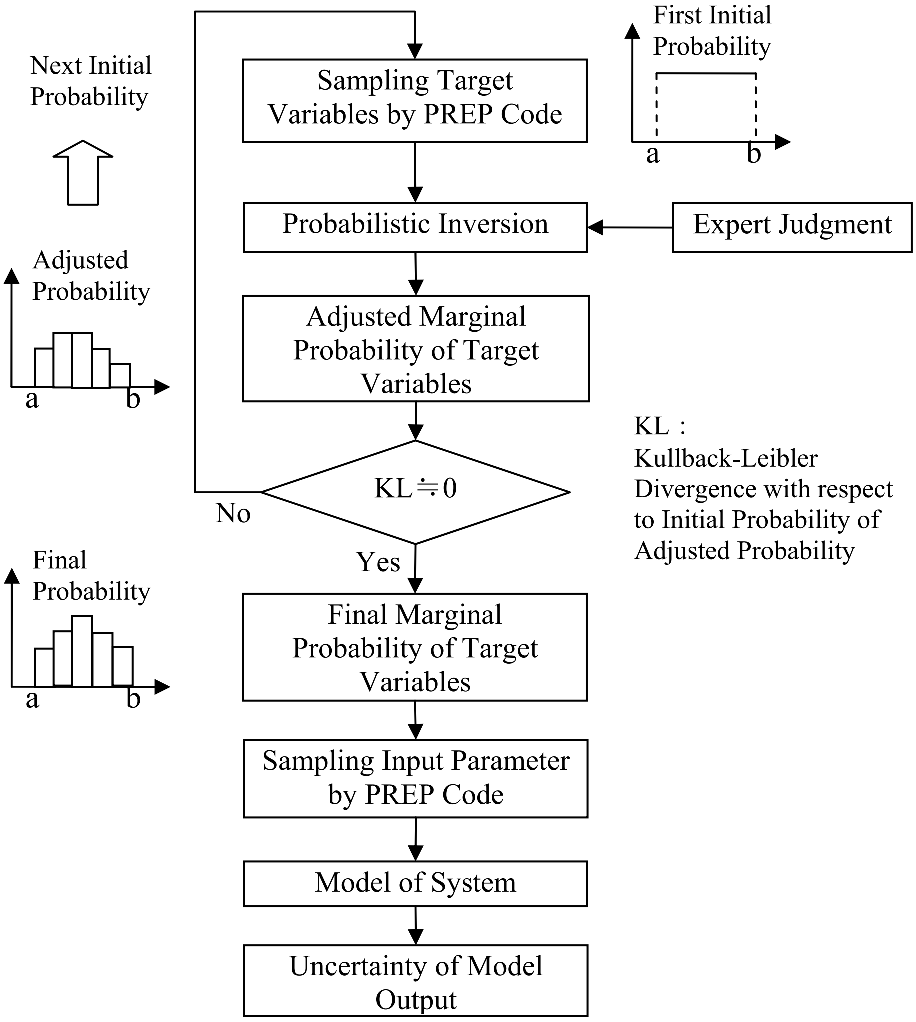


Fig. 3. Analytical procedure for uncertainty of model output caused by uncertainty of model input.

The second part is to analyse the uncertainty of model output. The input parameters are sampled with PREP code by estimated marginal distributions and outputs of model are evaluated by substituting their parameters to model equation. The uncertainty of dose coefficients for ingestion of cesium as illustrated in figure 4 is estimated by uncertainty of transfer coefficients in biokinetics model of cesium.

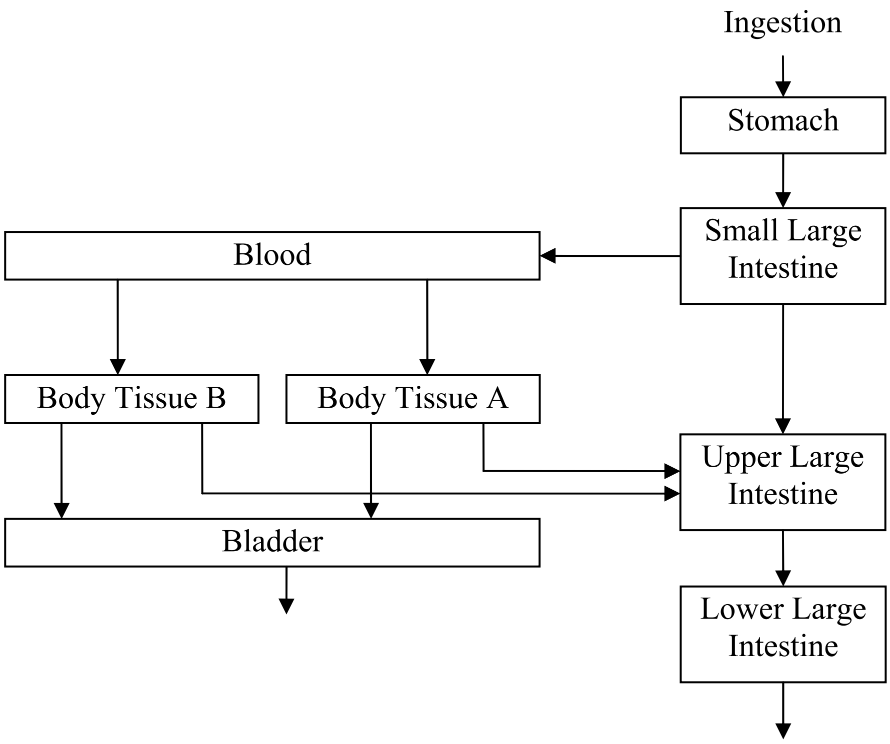


Fig. 4. Internal dosimetry model for ingestion of cesium.

Results

The probabilistic distributions for transfer coefficients in biokinetcs model of cesium estimated by probabilistic inversion and the uncertainty of dose coefficient for ingestion of ^{137}Cs by adult will be described in this section.

Probabilistic inversion

The six transfer coefficients were estimated by probabilistic inversion with expert judgment as illustrated in table 1 if the samples were 50000, the iterations of the iteration PARFUM were 200 and the iterations of probabilistic inversion were 10. In the results, the Kullback-Leibler divergence with respect to initial probability of adjusted probability was 6.04 and 0.38 in first and tenth probabilistic inversion, respectively. The probabilistic distributions of six transfer coefficients were shown in figure 5 as cumulative probabilistic distribution. This figure shows the first assumed uniform distribution, tenth adjusted probability and transfer coefficient recommended by ICRP. The 5th, 50th and 95th percentiles for retention fraction of body tissues by injection of Cs were illustrated in table 2 among expert judgment, 1th and 10th initial probability.

The first initial probability was very different from expert judgment, but the tenth adjusted probability resembled to expert judgment.

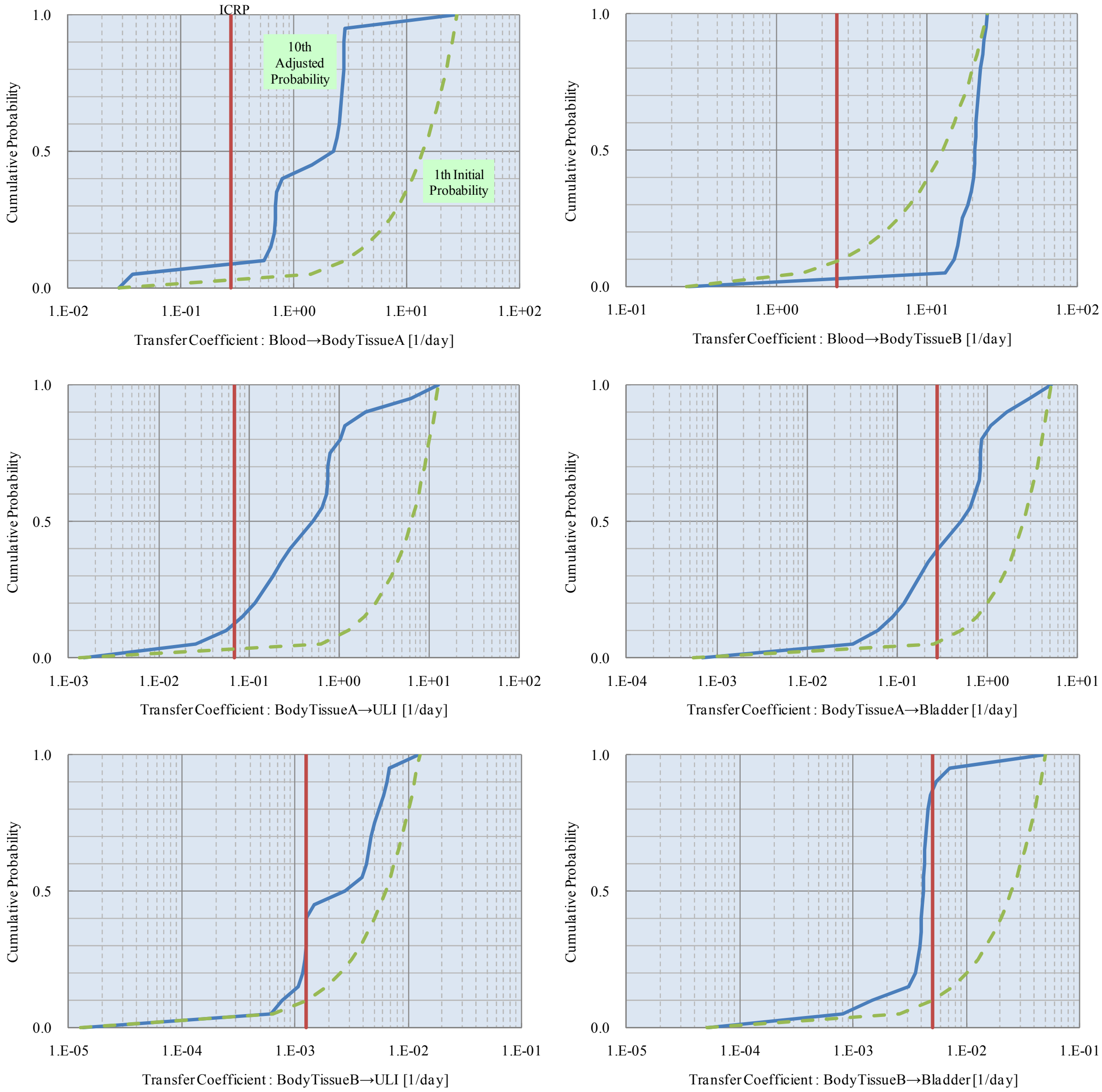


Fig. 5. Cumulative probabilistic distributions of six transfer coefficients in biokinetics model of cesium estimated by probabilistic inversion with expert judgment.

Table 2. 5th, 50th and 95th percentiles for retention fraction of body tissues by injection of Cs, for expert judgment, first initial probability and tenth adjusted probability.

Retention fraction of body tissues by injection of Cs		1day	1week	1month	1year	5years
5%	Expert Judgment	8.70E-01	7.45E-01	5.45E-01	2.38E-03	1.21E-10
	1th Initial Prob.	1.02E-01	7.66E-02	3.37E-02	7.60E-10	1.31E-44
	10th Adjusted Prob.	8.70E-01	7.46E-01	5.43E-01	5.55E-03	9.62E-12
50%	Expert Judgment	9.62E-01	8.59E-01	7.24E-01	6.48E-02	1.08E-05
	1th Initial Prob.	4.71E-01	3.80E-01	1.74E-01	4.18E-06	4.38E-26
	10th Adjusted Prob.	9.62E-01	8.59E-01	7.24E-01	7.07E-02	5.86E-06
95%	Expert Judgment	9.92E-01	9.43E-01	8.93E-01	2.64E-01	6.30E-03
	1th Initial Prob.	8.75E-01	7.33E-01	4.89E-01	2.15E-02	1.22E-07
	10th Adjusted Prob.	9.91E-01	9.52E-01	8.84E-01	2.93E-01	4.43E-03

Uncertainty of dose coefficient

The uncertainty distribution of effective dose coefficient for ingestion of ^{137}Cs by adult was shown in the figure 6. This uncertainty distribution was produced by estimated probabilistic distribution of transfer coefficients in biokinetics model of cesium. This figure also shows recommendation of ICRP, 5th and 95th percentiles. The statistics of uncertainty were shown in table 3. The recommendation of ICRP was equivalent to median and less than mean. The uncertainty factor of ratio of 95% to 5% was 5.1.

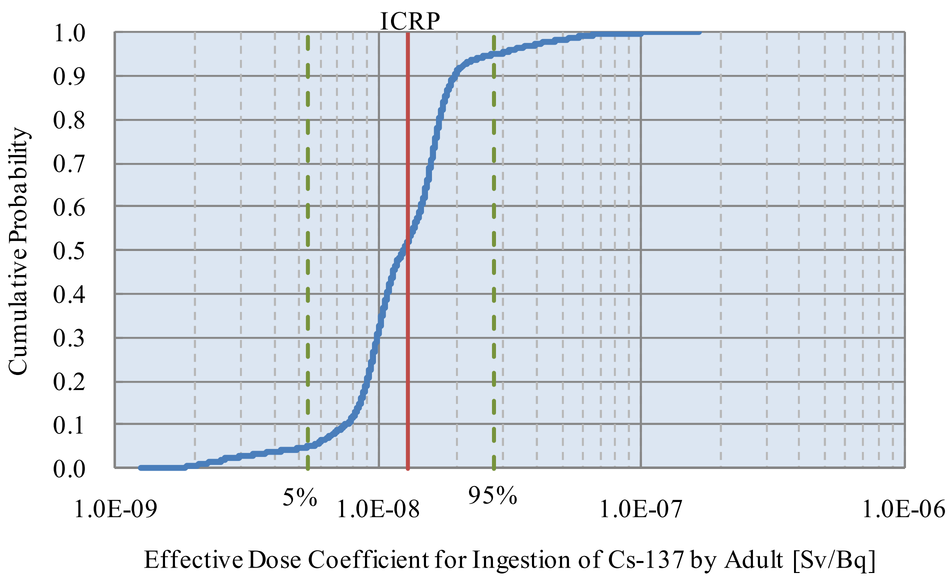


Fig. 6. Uncertainty distribution of effective dose coefficient for ingestion of ^{137}Cs by adult.

Table 3. Statistics of uncertainty of effective dose coefficient for ingestion of ^{137}Cs by adult.

Effectiv Dose Coefficient for Ingestion of ^{137}Cs by Adult	Value
Recommendation by ICRP	1.3×10^{-8} [Sv/Bq]
Mean	1.4×10^{-8} [Sv/Bq]
Median	1.3×10^{-8} [Sv/Bq]
5th percentile	5.4×10^{-9} [Sv/Bq]
95th percentile	2.7×10^{-8} [Sv/Bq]
Uncertainty Factor (95%/5%)	5.1

Conclusions

The uncertainty of internal dose coefficient was analysed with expert judgment. In this study, the expert judgment of Probabilistic Accident Consequence Uncertainty Assessment (Cooke et al., 1999) was applied. The probabilistic distributions of transfer coefficients in biokinetics model of cesium were estimated by probabilistic inversion. The uncertainty of dose coefficient for ingestion of ^{137}Cs by adult was analysed with estimated distributions of transfer coefficients.

In the result, the recommendation of ICRP was equivalent to median and less than mean. The uncertainty factor of ratio of 95% to 5% was 5.1, and its value was thought relatively small. It is thought a reason that the transfer coefficients in biokinetics model are taken as sources of uncertainty. The uncertainties of gastrointestinal model are also need to take as sources of uncertainty in dose coefficient for ingestion.

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Internal and external dosimetry organization in the Joint Research Centre of Ispra

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Abstract

The Joint Research Centre of Ispra, one of the research Sites belonging to the European Commission, Directorate General JRC, was created in the late '50s, in order to steer European research on nuclear industry. It hosts numerous nuclear facilities, some of which are maintained in operation, while others were shutdown in past years, namely: two research nuclear reactors, hot cells facilities, radiochemical laboratories, one Cyclotron (still in operation), facilities for studies on fissile material (in operation), and some facilities for the treatment and storage of liquid and solid waste (in operation).

The JRC accounts for 21 nuclear licences, 14 Controlled Zones and 12 main Surveilled Zones, on its Ispra Site.

The Radiation Protection Sector employs the services of some internal laboratories for the assessment of external and internal doses: the "Dosimetry" Laboratory (for personal and ambient TLDs), the Whole Body Count Laboratory, the Radiotoxicological Laboratory (for analyses on excreta), the Radiation Protection Sector itself (for ambient dose and contamination reporting, and electronic personal dosimeters readings).

Some of these Services are open also to the external market, and JRC-Ispra is among the few being able to provide, in Italy, either for internal and external dosimetric services to Customers.

The paper will discuss the organization and the structure of the Dosimetry and the Whole Body Count Laboratories, and their functions in the management of daily Radiation Protection tasks at the JRC.

Moreover, in order to follow-up and control Personnel radiation doses, the Radiation Protection Sector has developed and put in place, in 2007, the "**Unified Dosimetry System**", a wide and flexible *database* centralizing all dosimetric data and making them on-line available to JRC Radiation Protection experts and to the JRC Qualified Expert.

The "Dosimetry" Laboratory

Today, the mission of the JRC Ispra Dosimetry Service, belonging to the Radiation Protection Sector, is:

- preparation and distribution of personal and environmental TLD dosemeters;
- dosemeter reading and data processing;
- communication of dosimetric data to the Qualified Expert.

The number of dosemeters processed in the year 2009, with their monitoring frequencies, are shown in Table 1.

Table 1. Capability of JRC-Ispra Dosimetry Service.

Type of dosemeter	Number of dosemeters processed in the year 2009	Monitoring frequency
Whole body	4500	Monthly
Extremity - Ring	500	Monthly
Extremity - Wrist	1000	Monthly
Environmental	1000	Quarterly

Dosemeters' description: structure and composition

The Whole body dosemeter is sensitive to beta, neutronic, X and γ radiation. It consists of a transparent plastic outer casing (of dimensions 50x50x14 mm) with two Panasonic TLD badges; the first is of UD-802A series and is located at the top; the second is of UD813-A6 series and is contained within a capsule of B4C ("albedo capsule"), inserted at the bottom. Each badge contains four thermoluminescent detectors coated with different protective material and/or shielding. The choice of thickness and front and rear construction of the dosimeter serves the purpose of providing a more selective response to radiation of different nature and energy. Table 2. shows the composition of each thermoluminescent detector and the overall front and rear thickness of the different materials that shield each of them.

Table 2. Whole body dosemeter: structure and composition.

Badge series	Detector	Composition	Overall front filtration (mg/cm^2)	Overall rear filtration (mg/cm^2)
UD-802A	1	$^{n}Li_2B_4O_7$:Cu	Plastic: 14	Plastic: 28
	2	$^{n}Li_2B_4O_7$:Cu	Plastic: 303	Plastic: 303
	3	$CaSO_4$:Tm	Plastic: 303	Plastic: 303
	4	$CaSO_4$:Tm	Plastic: 143 – Lead: 874	Plastic: 143 – Lead: 874
UD-813-A6	5	$^{6}Li_2{}^{10}B_4O_7$:Cu	Plastic: 80	Plastic: 165 – B_4C: 450
	6	$^{7}Li_2{}^{11}B_4O_7$:Cu	Plastic: 80	Plastic: 165 – B_4C: 450
	7	$^{7}Li_2{}^{11}B_4O_7$:Cu	Plastic: 165 – B_4C: 450	Plastic: 165
	8	$^{6}Li_2{}^{10}B_4O_7$:Cu	Plastic: 165 – B_4C: 450	Plastic: 165

The ring dosemeter is made of plastic with flexible brackets to fit any finger size. The thermoluminescent detector is $^{n}Li_2B_4O_7$:Cu (Panasonic UD-807 series). Front filtration: 11 mg/cm^2 – Rear filtration: 28 mg/cm^2. The dosimeter does not allow selectivity in radiation of different nature and energy.

The wrist dosemeter is the Panasonic UD-802A badge inserted in a strip of clear plastic of adjustable length. The front and rear filtrations have characteristics of the standard Panasonic badge. The dosemeter is characterized by a selective response to radiation of different nature and energy.

The Panasonic UD-802A badge is also an environmental dosemeter, but inserted in a double plastic tube; the first tube (diameter 2 cm – height 5 cm) guarantees weather protection of the dosemeter in all weather conditions; the second allows the dosemeter correct positioning (diameter 4 cm – height 14 cm).

Starting in 2010, ambient Radon dosimetry service will officially be offered in the JRC. CR-39 detectors will be employed, and specific measurement campaigns will be started both in JRC nuclear facilities and in all underground working places.

Fig. 1. Personal dosemeters at the JRC: wrist, whole body and ring.

Dosemeters' calibration

Dosemeters are irradiated at the JRC Ispra Calibration Centre which is an accredited SIT ("*Servizio di taratura in Italia*": SIT is one of the signatories to the Mutual Recognition Agreement EA-MLA and ILAC-MRA for calibration certificates), using radiation sources (X and gamma) specified in ISO 4037-1 and in accordance with ISO 4037-3 (irradiations performed on ISO phantoms for personal dosemeter and in air for environmental dosemeter).

Calibration is performed for each new batch of detectors and, afterwords, with annual periodicity. Each thermoluminescent detector is characterized by a sensitivity factor every year.

For each monitoring period and for all types of dosimeters a calibration curve is read together with the associated dosimeter.

The dosemeters are read via a Panasonic UD-716 AGL automatic reader.

Performance of JRC personal dosemeters

Since 1986 the JRC-Ispra Dosimetry Service has successfully participated in the technical *audits* promoted by the national working group ENEA-EDP, for the whole body TL dosimetric system. From 2000 on, the national working group ENEA-EDP is no longer active, and in Italy new legislation to regulate the matter is pending.

According to the main standards in this field, the JRC-Ispra Dosimetry Service tested its dosemeters. Irradiations were performed at the Calibration Centre of the JRC Ispra. Table 3. and Fig. 2. show some of the test results.

Table 3. The standard requires that the detection threshold shall not exceed 0,1 mSv.

Type of dosemeters	Detection threshold (mSv)
Whole body	0.05
Extremity - Ring	0.07
Extremity - Wrist	0.03

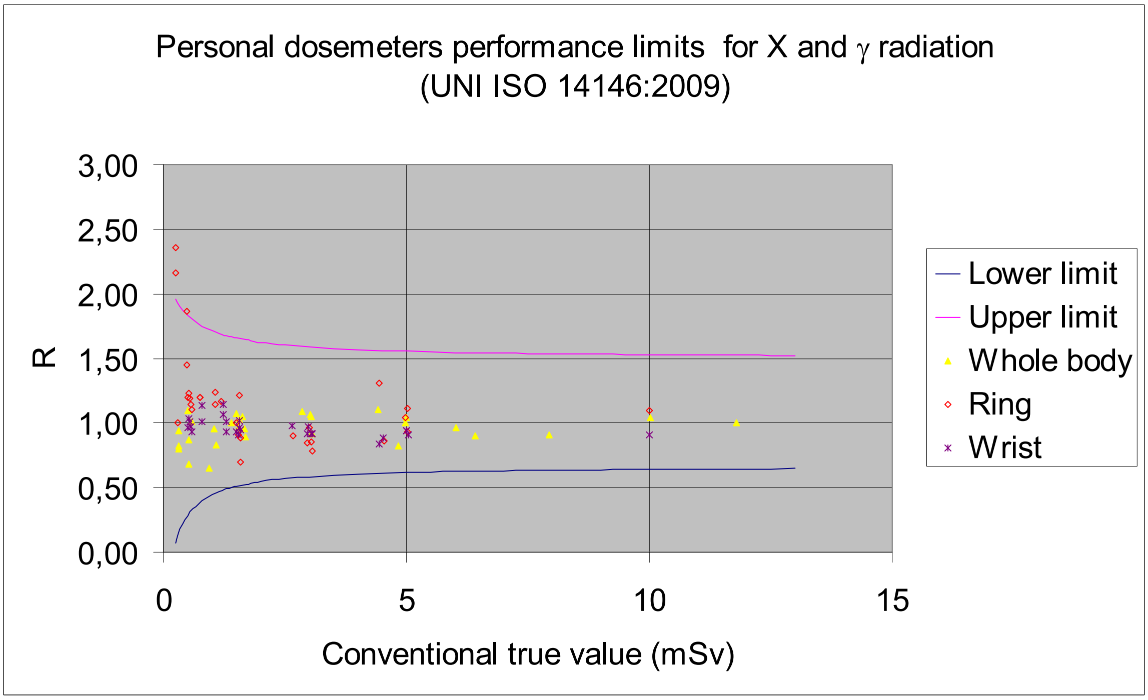

Fig. 2. Performance limits – R : ratio between measured dose and conventional true value.

The “Whole Body Count” Laboratory

The mission and the scope of Whole Body Counting (WBC) service of the JRC Ispra Radiation Protection Sector is:

- to conduct measurements of internal radioactivity of the human body on routine basis;
- to conduct measurements of internal radioactivity of the human body in special cases, like work missions outside JRC, in case of incidents etc.;
- identification of the radionuclide, and if possible its distribution within the body;
- communication of acquired data to JRC Qualified Expert.

The statistics of WBC examination performed in the year 2009 is presented in Table 4.

Table 4. Statistics of WBC measurements in the year 2009.

	Number of examinations performed in the year 2009
WBC total	599
WBC - JRC	361
WBC - external	238
no artificial radiation found	563
^{60}Co	6
^{137}Cs	26
^{125}I	1 (localized in thyroid)
^{18}F	2
^{99m}Tc	1 (undeclared medical examination)

WBC laboratory description

The internal contamination of the human body is measured by a "direct method": gamma rays emitted by radioactive nuclides from inside the body are detected using an appropriate detector, allowing determination of the quantity of incorporated activity and, in certain cases, the localisation of a radionuclide (e.g. lungs or thyroid).

For this reason the very good shielding of the measurement room against background radiation is a prerequisite. To achieve this, both, the special materials and a specific construction of rooms and equipment are employed:

- the laboratory is located in the building's basement, in order to reduce cosmic radiation,
- the 40 cm thick walls are made of low activity concrete providing additional shielding of the room,
- the floor of the room is made of marble poor in potassium (presence of 40K) and other naturally occurring radionuclides,
- the measurement room is shielded with a multilayer absorber consisting of:
  - bricks of purified lead, 10 cm thick, providing the outer shielding,
  - 1 mm thick cadmium sheets,
  - 2 mm of copper cover inside, to absorb fluorescence radiation from lead.
- A "maze" shape in the entrance of the measurement room
- the room is well ventilated and supply of fresh air through air filters is realized to reduce the radiation from radon and products of its decay present in ambient air,
- careful choice of other components of room equipment and spectrometer parts, containing low residual activity,
- low noise electronics.

With use the all above listed measures, an environmental radioactivity background as low as 10 nSv/h was obtained in the WBC measurement room.

WBC spectrometer description

The spectrometer of the WBC laboratory is equipped with three gamma detectors, one NaI(Tl) scintillation detector, commonly used for whole body counting and two HPGe detectors dedicated for lungs counting. Each detector is associated with separate spectrometric electronics - Multi Channel Analyser (MCA) and all three can work simultaneously. The spectra collected are processed and stored on the hard disk of dedicated PC, specifically intended for data analysis and archiving. Each spectrometric line dedicated for whole body counting or lung counting, except dedicated detector consists of:

- NIM module – Ortec 556 high voltage power supply,
- NIM module – Ortec 672 spectroscopy amplifier,
- NIM module – Canberra Multiport II multichannel analyser,
- PC (common for all three lines)on which the software is installed,
- software – Gennie 2000,
- software – ABACOS.

The scintillation detector is based on single crystal of NaI(Tl) with the diameter of 20 cm and 10 cm height, with an aluminium window and equipped with four independent photomultipliers. The energy resolution of this detector is 50.5 keV at 662 keV (^{137}Cs line) and 76.6 keV at 1461 keV (^{40}K line). Due to the high density of the crystal (3.67 $g \cdot cm^{-3}$) and its relatively high effective atomic number, the detector may detect electromagnetic radiation in the energy range from 50 keV up to around 2500 keV. The electronics associated is of high quality, in order to minimize noise and therefore ensure higher sensibility, accuracy, precision and overall performance of the system.

The two HPGe detectors are based on crystals of 5.1 cm in diameter and 2 cm height, with beryllium windows allowing the detection of low energy photons. The energy resolution of the detectors is in the range of 0.55 keV for 60 keV radiation and of 0.88 keV for the energy of 350 keV. Due to the high energy resolution and efficiency characteristics is more suitable for detection of the low energy photons, especially in the range 10 – 300 keV. The HPGe detectors are also connected to high quality electronics.

WBC calibration

Prior to WBC contamination measurement, system calibrations with specially designed, certified sources (phantoms) are needed. The WBC Laboratory has employed a Bottle Manikin Absorption (BOMAB) phantom for efficiency calibration of NaI(Tl) detector (commonly used for whole body radioactivity measurements). The BOMAB phantoms consists of 10 sealed bottles simulating shape of the human body and containing selected radionuclides homogeneously distributed inside the matrix (absorbing radiation in a similar way to a human body). The two HPGe detectors (used for lungs counting) are calibrated with a Livermore Phantom. This kind of phantom consists of plastic elements simulating the body and its internal organs. Such construction allows calibrating the system in order to measure the radioactivity localised in single organs (e.g. lungs). The calibration of HPGE detectors for lungs counting is shown in Fig. 3.

Since the WBC examination involves spectrometric measurements, also the energy calibration is performed on regular basis in order to allow nuclide identification. This process employs the measurements of radiation emitted by selected sources in order to assign radiation energy to channel number of MCA and is generally identical for all detectors.

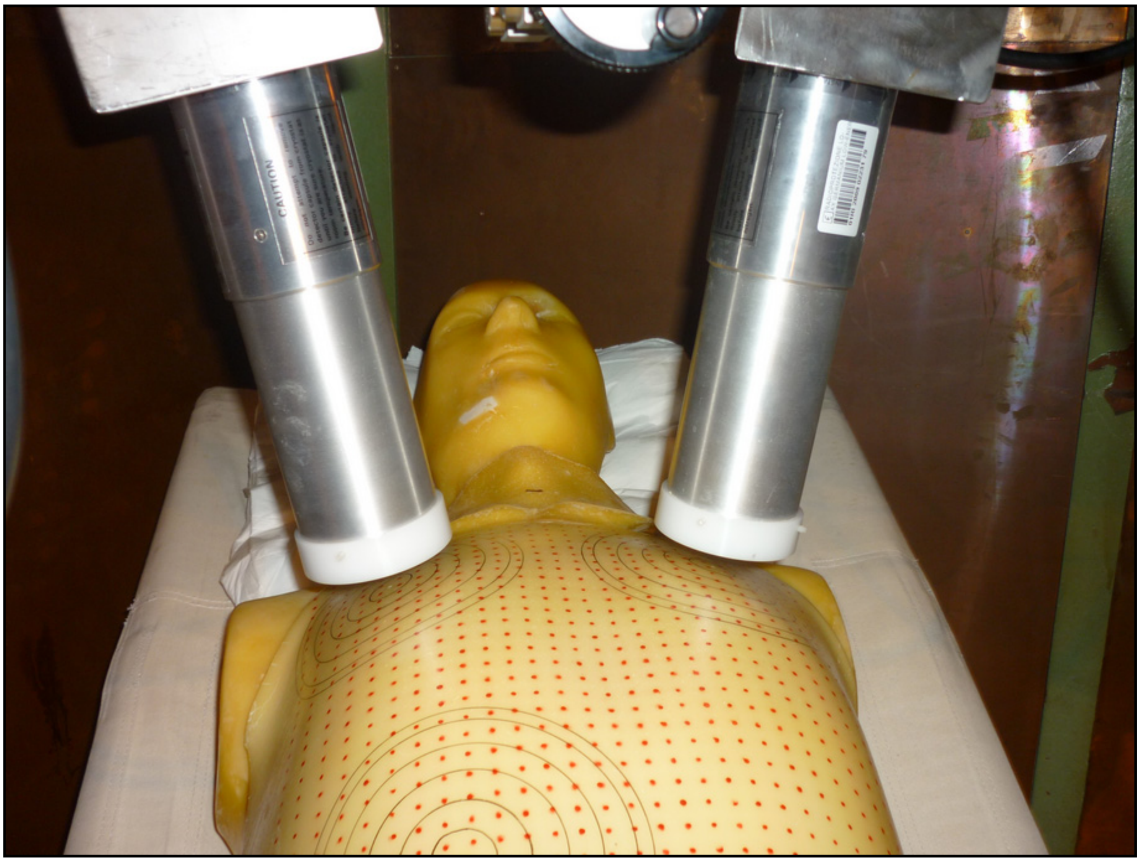

Fig. 3. The efficiency calibration of HPGe detectors with Livermore Phantom equipped with contaminated lungs.

WBC measurement

The WBC measurements are executed on a routine basis on JRC Exposed Workers, and also at start (or closure) of job contract with the JRC or related external company. The measurements are also performed on *ad hoc* basis in urgent cases (like suspicion of contamination of workers) or prior and after work missions. The duration of the measurement is between 10 minutes for standard check, up to 45 minutes in special cases. The collected spectra are evaluated ant the results are communicated to the JRC Qualified Expert.

The "UNIFIED DOSIMETRY SYSTEM"

The Radiation Protection Sector (RPS) introduced the "**Unified Dosimetry System**" (UDS) in 2007. This wide database is collecting all exposed Personnel's dosimetric data. It is also the current way of managing the TLD and WBC data, either for storage and for examination/evaluation by the JRC Qualified Expert.

The JRC accounts, on its Ispra Site, in 2010, for 21 nuclear licences, 14 Controlled Zones and 12 main Surveilled Zones.

On JRC classified zones, during the last 12 months, a considerable number of JRC exposed workers have been employed (for a total of 180 JRC exposed workers), as well as 190 external companies' exposed workers, belonging to around 30 different Companies. In the last 12 months, more than 50.000 passages of Personnel have been registered by the Electronics Dosimeters' System, inwards JRC Controlled Zones, among which almost 4.500 passages with non-zero dose (i.e., associated with a registered "electronic" dose higher than zero).

The RPS has developed and first put in operation the UDS aiming the follow-up and control of radiation doses received either by JRC Personnel intervening in nuclear facilities' operation (and during future decommissioning activities).

The Italian Legislation, implementing relevant European Directives, requires that personal doses be evaluated, recorded and registered. For some aspects, though, Italian Radiation Protection Legislation is stricter than the Directives from which it comes from, reflecting the peculiar attention granted to nuclear industry in Italy.

The main role in the management of doses and of exposed workers risks is a responsibility assigned, by Law, to the Radiation Protection Qualified Expert (QE).

It is the QE's responsibility, among many others, to inform the Employer about radiation risks, to perform radiation measurements and samplings, and to suggest the prescription of PPEs, personal dosimeters and additional controls and verifications that may result necessary.

Moreover, the QE is the sole responsible for evaluating workers' effective doses, unlike what happens in other EU countries, in which an approved dosimetric service may assign and communicate doses to the Employer (and not only dosimeters' readings).

In the case of workers of Category B, the Italian legislation requests that the evaluation of external exposure may be based on ambient dosimeters, but, for workers of Category A, personal dosimeters must always be assigned to workers[1].

Each worker automatically turns on and off his/her EPD at the entrance/exit of the specific Controlled Zone, making dose collection automatic and quicker. Moreover, "task codes" have been identified for some specific work activities, which allow to separate and track doses pertaining to long-lasting projects or to planned repetitive operation and decommissioning activities on JRC nuclear facilities.

[1] For workers of category A, proper *in vivo* and *in vitro* methods must be employed for the evaluation of internal exposure.

At present, active task codes are:
1. Radiation protection (for RP assistance activities all throughout the JRC)
2. Research activities at the JRC Cyclotron (IHCP Institute)
3. Radiopharmaceutical commercial production at the JRC Cyclotron (IHCP Institute)
4. Research activities concerning fissile material (IPSC Institute)
5. Decommissioning activities
6. Licensing and operation of JRC facilities
7. Maintenance activities on JRC facilities
8. Management of nuclear material
9. Visitors

To overcome the need for the Qualified Expert to manually treat all data relating to internal and external dosimetry (some of which are normally made available with some weeks of delay) and other radiation protection archives data (medical aptitude, previous dose records, training, passports, missions, etc.), the need for a very more complex database has been felt. The UDS database, in fact, needs to be able to store a huge number of data, namely:

- Personal workers' data (name, birth date, address, company, category, last medical visit, medical aptitude, previous doses, etc.)
- Access data (identification of controlled zone reader, date and hour of entry, date and hour of exit, errors in communication with the reader, etc.)
- Data related to doses (integrated dose (either gamma and neutron) between entry and exit, average dose (either gamma and neutron), dose or dose rate threshold exceeded, dose and dose rate alarm, integrated dose alarm (on a time period basis), etc.)
- Data related to "task codes" (task code typed in during entrance)
- Informatics data

This informatics tool has been specifically developed in order to comprise also:

- Workers' personal data archive, recording personal ID, arrival date to the JRC, category of exposure, medical visits and aptitude, doses received in previous activities (if present), working location, appointed QE, training information, emission of certificates and radiation passport, etc.
- Workers' calendar, presenting milestones in the worker's personal history (general RP training date, specific RP training date, medical aptitude date, in vivo and in vitro examinations dates, incidents and contamination dates and references, etc.)
- The dosimetric archive, either for TLD personal dosimeters, for EPDs and for TLD ambient dosimeters, importing and regrouping raw dose data coming from the Dosimetric Service and the RP Sector
- The radio-toxicological in vitro archive, importing and regrouping data from Medical Service examinations
- The WBC in vivo archive, importing and regrouping data from Whole Body Count examinations
- The Qualified Expert evaluations archive, regrouping evaluations of internal and external doses and other QE's official acts

- The dose communications' archive, regrouping the official dose communications from the Employer to workers and the communications of dosimeters' readings to Outside Companies

The functions of the UDS include the possibility to issue personal radiation sheets, signalling doses and intakes from the last months/years, and indicating additional information on medical aptitude and RP training. Finally, as dose evaluations must regularly be communicated to the Employer and the Workers, as requested by Law: the data for communications are easily obtained via the UDS, regrouping raw data and Qualified Expert's evaluations in a single view.

At present, the UDS incorporates the possibility to issue the dose evaluation document itself, inserting individual evaluations in a standard format letter, which is used for JRC dose communications, workers' dose evaluations and other data.

The "Unified Dosimetry System", is a wide and flexible tool, which allows a thorough follow-up of regulatory personal dosimetric data, as well as statistical analyses over dosimetric data. It is based on the integration of dosimetric data of different origin: either the use of both regulatory TLD passive dosimeters (either personal and area monitoring TLDs) and operational active electronic dosimeters (EPDs); and also data of internal dosimetric evaluations, both direct (WBC) and indirect (RTX).

The UDS allows the printout of dose communications, with the aim, in the future, to replace the paper ones, according to the requests of Italian Legislation.

The UDS is also a system of workers' dose control, in an ALARA perspective, and will be of paramount importance during future decommissioning works, for which a much higher collective dosimetry is expected.

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High dose values due to contaminated badges

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Abstract

In some cases extraordinary high monthly radiation doses have been determined by the PTPA for radiation workers of the Department of Nuclear Medicine of the General Hospital Vienna. These values only could be explained to be caused by contaminations of the cover of the dosimeters (badges). These incidents were the motive for the presented study, with the aim to find out the amount of radioactivity sufficient to produce a monthly dose of 1.67 mSv, derived from the maximum permissible annual dose (20 mSv).

For this reason 50 badges (UD-802A) have been contaminated systematically with calibrated radioactive solutions, using a 10µL-pipette. The drops have been placed on the resin cover above one of the four TL phosphors respectively in order to consider different spots of possible contaminations. The badges have been shielded by lead to avoid irradiating each other. They were stored for one month at most and evaluated afterwards. The procedure was performed with the nuclides F-18, P-32, Tc-99m, In-111, I-123, I-125 and I-131. In order to consider different times of contamination the results of the measurements were converted for 2, 14 and 30 days respectively before the evaluations.

As a result just low amounts of radioactivity are causing relative high doses. Due to the radionuclide and – for nuclides of longer half lives in addition due to the moment when a contamination occurred – the values differ significantly. Our study shows, that possible contaminations of the badges can be considered to be the reason for conspicuous high evaluated doses.

Introduction

In some cases extraordinary high monthly radiation doses have been determined by the PTPA for radiation workers of the Department of Nuclear Medicine of the General Hospital of Vienna. These values only could be explained to be caused by contaminations of the cover of the dosimeters (badges). The incidents were the motive for the presented study, with the aim to find out the amount of radioactivity sufficient to produce a monthly dose of 1.67 mSv, derived from the maximum permissible annual dose (20 mSv) [1].

Material and method

Thermoluminescence dosimeters (TL-badges; Panasonic UD-802A) were contaminated systematically with calibrated radioactive solutions, using a 10µL-pipette. The drops were placed on the resin cover above one of the four TL phosphors respectively, in order to consider different spots of possible contaminations. The procedure was performed altogether more than 300 times, with the nuclides F-18, P-32, Tc-99m, In-111, I-123, I-125 and I-131.

The contaminated dosimeters were covered with 2mm lead to avoid exposure towards each other. In the case of P-32 a cylindrical shielding of 8mm acryl with an additional cover of 1mm lead was used. The dosimeters thereafter were stored for a month at most, at a place with low radiation background before they were evaluated by the PTPA.

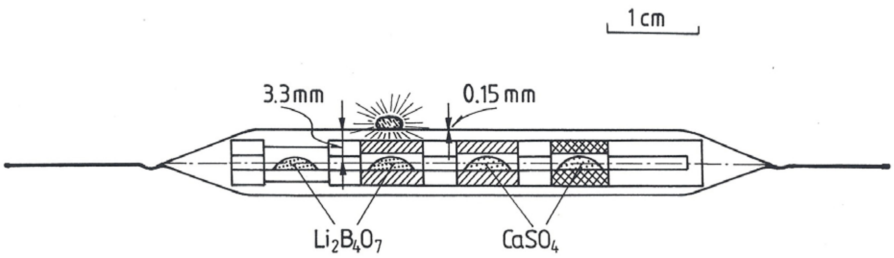

Fig. 1. Cross section through a TL-dosimeter, with resin jacket.

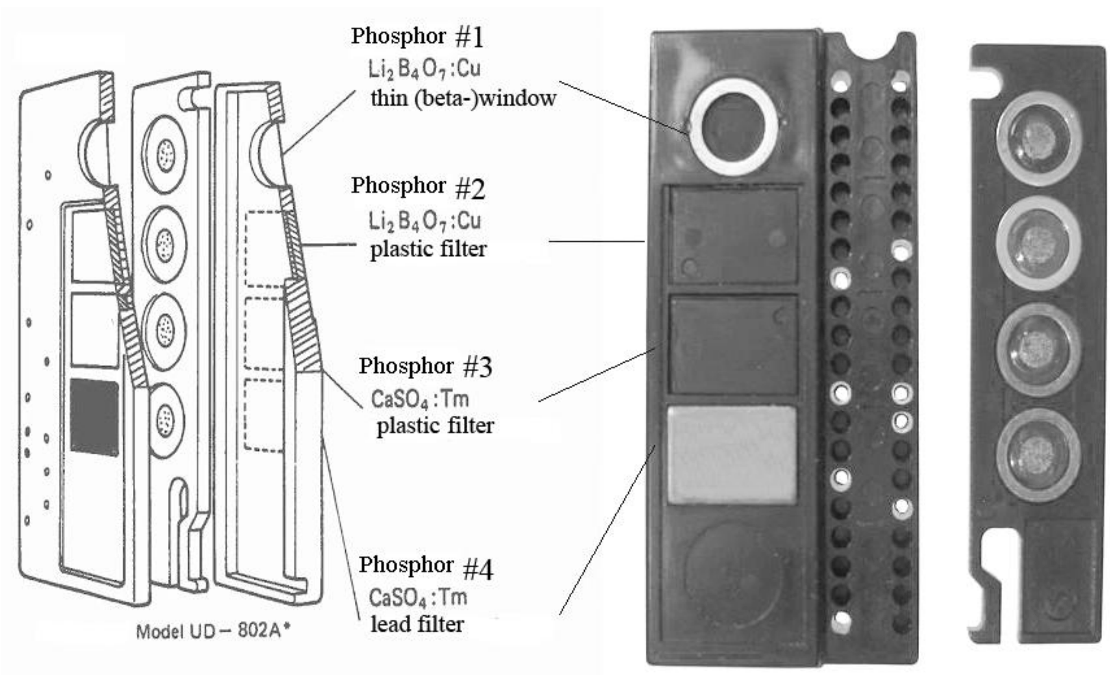

Fig. 2. TL-dosimeter Panasonic UD-802A [2].

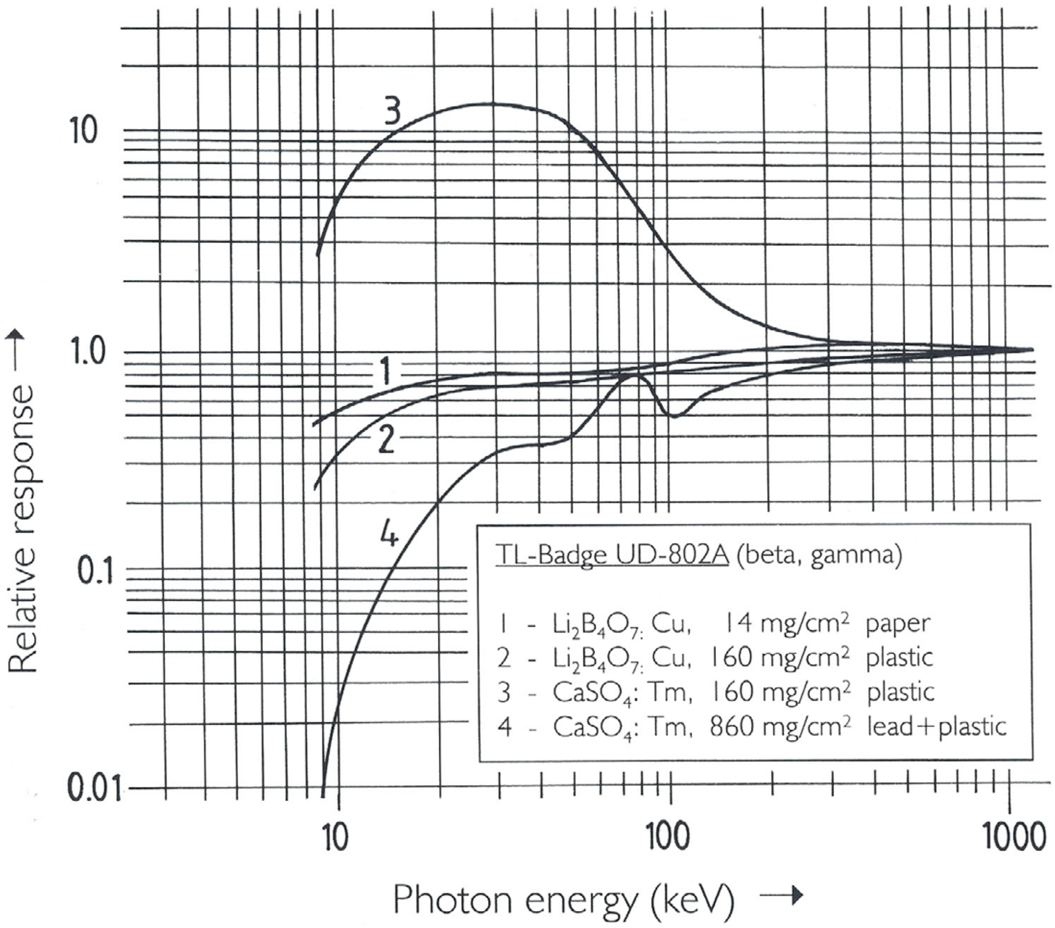

Fig. 3: Efficiancy curves of the four TL phosphors between the diverse filters [2].

Results

The results are summarized in the tables 1 to 7. They are corrected for radiation background and are normalized to an activity of 1 MBq (column 2). In order to consider different times of contamination before the time of evaluation the results of the measurements have been converted for 2, 14 and 30 days. The activities, sufficient to cause 1.67 mSv ($A_{1.67}$) are listed in column 3 to 5 of the tables. The respective irradiation times are indicated in parentheses.

Meaning of the terms in the first column of the tables:
"phosphor #1": contamination over the first Phosphor; mean of 7 values
(just as for "Phosphor #2" to "Phosphor #4")
"Ph. 1-4, mean": mean of the values "Phosphor #1" to "Phosphor #4"
"Ph. 1-4, uniform: uniform distributed contamination over the four Phosphors (singular values)

Table 1. Contaminations with fluorine-18.

F-18	mSv/MBq (3d)	$A_{1.67}$ (2d)	$A_{1.67}$ (14d)	$A_{1.67}$ (30d)
Phosphor #1	763.3	2.10	2.10	2.10
Phosphor #2	20.1	79.50	79.50	79.50
Phosphor #3	96.9	16.51	16.51	16.51
Phosphor #4	10.3	155.11	155.11	155.11
Ph. 1-4, mean	**222.7**	**7.19**	**7.19**	**7.19**
Ph. 1-4, uniform	56.1	28.50	28.50	28.50

Table 2. Contaminations with phosphorus-32.

P-32	mSv/MBq (22d)	$A_{1.67}$ (2d)	$A_{1.67}$ (14d)	$A_{1.67}$ (30d)
Phosphor #1	20372.8	0.56	0.10	0.07
Phosphor #2	39510.7	0.29	0.05	0.03
Phosphor #3	1307.2	8.68	1.63	1.05
Phosphor #4	30.7	369.75	69.42	44.60
Ph. 1-4, mean	**15305.3**	**0.74**	**0.14**	**0.09**
Ph. 1-4, uniform	10786.6	1.05	0.20	0.13

Table 3. Contaminations with technetium-99m.

Tc-99m	mSv/MBq (14d)	$A_{1.67}$ (2d)	$A_{1.67}$ (14d)	$A_{1.67}$ (30d)
Phosphor #1	30.8	52.2	52.0	52.0
Phosphor #2	23.9	67.3	67.0	67.0
Phosphor #3	5.1	313.5	312.2	312.2
Phosphor #4	1.8	890.4	886.8	886.8
Ph. 1-4, mean	**15.4**	**104.3**	**103.9**	**103.9**
Ph. 1-4, uniform	10.0	160.3	159.7	159.7

Table 4. Contaminations with indium-111.

In-111	mSv/MBq (28d)	$A_{1.67}$ (2d)	$A_{1.67}$ (14d)	$A_{1.67}$ (30d)
Phosphor #1	173.5	23.64	9.50	9.22
Phosphor #2	425.8	9.63	3.87	3.76
Phosphor #3	1477.0	2.78	1.12	1.08
Phosphor #4	313.5	13.08	5.26	5.10
Ph. 1-4, mean	**597.5**	**6.87**	**2.76**	**2.68**
Ph. 1-4, uniform	769.8	5.33	2.14	2.08

Table 5. Contaminations with iodine-123.

I-123	mSv/MBq (11d)	$A_{1.67}$ (2d)	$A_{1.67}$ (14d)	$A_{1.67}$ (30d)
Phosphor #1	13.5	129.09	118.76	118.76
Phosphor #2	50.6	34.39	31.64	31.64
Phosphor #3	296.9	5.86	5.39	5.39
Phosphor #4	9.4	185.04	170.23	170.23
Ph. 1-4, mean	**92.6**	**18.79**	**17.28**	**17.28**
Ph. 1-4, uniform	79.2	21.95	20.20	20.20

Table 6. Contaminations with iodine-125.

I-125	mSv/MBq (45d)	$A_{1.67}$ (2d)	$A_{1.67}$ (14d)	$A_{1.67}$ (30d)
Phosphor #1	1087.5	26.13	4.00	2.04
Phosphor #2	3573.6	7.95	1.22	0.62
Phosphor #3	12962.5	2.19	0.34	0.17
Phosphor #4	2740.3	10.37	1.59	0.81
Ph. 1-4, mean	**5091.0**	**5.58**	**0.85**	**0.44**
Ph. 1-4, uniform	6740.3	4.22	0.64	0.33

Table 7. Contaminations with iodine-131.

I-131	mSv/MBq (11.7d)	$A_{1.67}$ (2d)	$A_{1.67}$ (14d)	$A_{1.67}$ (30d)
Phosphor #1	5652.7	1.14	0.26	0.19
Phosphor #2	356.0	18.04	4.07	3.09
Phosphor #3	527.7	12.17	2.75	2.08
Phosphor #4	186.0	34.52	7.80	5.91
Ph. 1-4, mean	**1680.6**	**3.82**	**0.86**	**0.65**
Ph. 1-4, uniform	3652.7	1.76	0.40	0.30

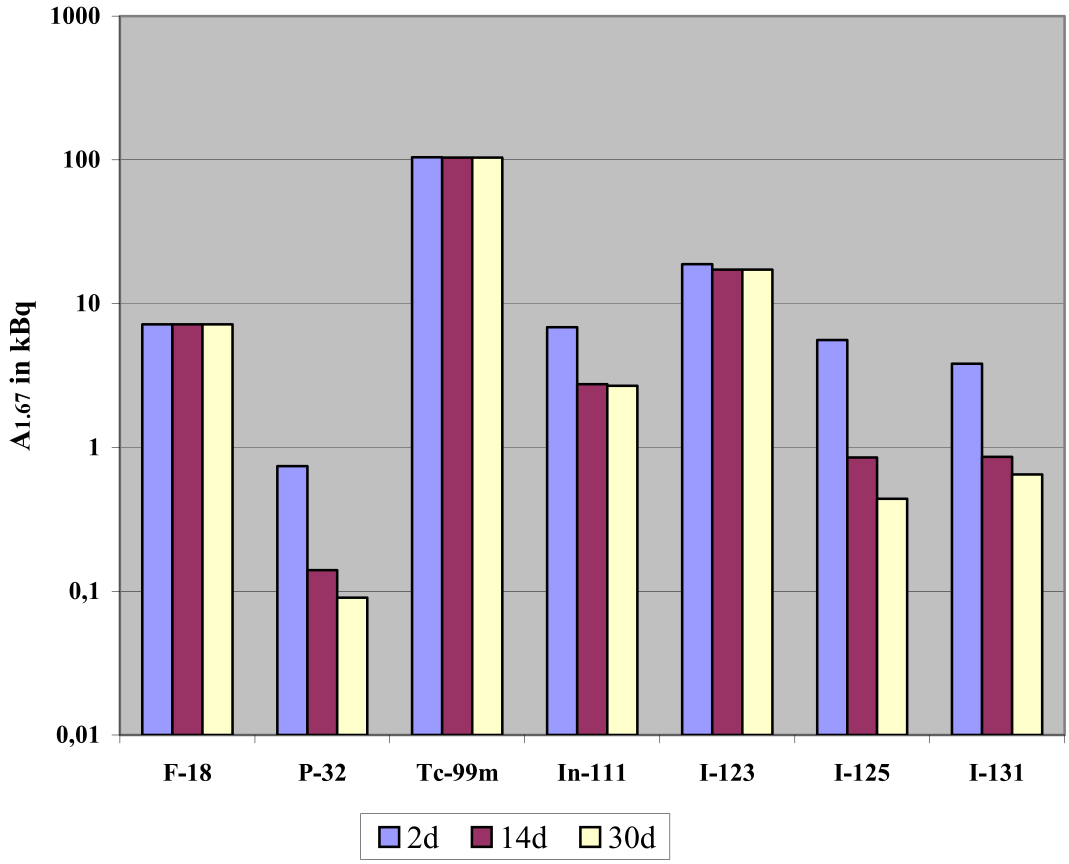

**Fig. 4: $A_{1.67}$ = "Contamination-activity", sufficient to cause 1.67 mSv
the days represent the time between contamination and evaluation**

Discussion

Just low amounts of radioactivity are causing relative high doses. Contaminations with less than 1 kBq of the nuclides P-32, I-125 or I-131 are sufficient to exceed the maximum permissible monthly dose. Compared to these nuclides, about 100 kBq of Tc-99m are necessary to reach this dose.

For nuclides with longer half-lives (P-32, I-125, I-131) the values differ significantly, depending on the moment of contamination. As expected this does not apply to nuclides with short half-lives (F-18, Tc-99m, I-123).

The evaluated doses depend on the radiation energy, on the half-life of the radionuclide and consequently on the cumulated activity $\tilde{A}$ (number of decays during the irradiation time). For example $\tilde{A}$ for an irradiation time of 14 days for P-32 is about hundred times higher than that for F-18. Due to the position of the contamination on the dosimeter the doses differ from 1:8.5 (In-111) to 1:1.300 (P-32). The mean values "Ph. 1-4, mean", written in bold letters, are the most evident results. In most cases they are similar to the values "Ph. uniform".

The doses for P-32 seem to be surprisingly high, but are explainable with the relative long half-life and with the fact, that the Phosphors #1 und #2 are within the range of the beta radiation and that adjacent phosphors can be reached by the slanting rays.

Furthermore, the results are due to the evaluation algorithm for the TL-dosimeters, generally intended for uniform external irradiation rather than for contaminations.

Conclusion

Our study shows, that contaminations of personal dosimeters with low amounts of radioactivity should be considered to be the reason for conspicuous high evaluated doses.

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[2] Technical information, Matsushita Electric Trading Co., Ltd., Osaka, Japan

Tooth dosimetry for residents of Techa riverside territories

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Abstract

The method of EPR dosimetry with teeth is practically the only method acceptable for retrospective individual dose measurements for non-occupational exposure. Application of the method for external dose reconstruction in the Techa River region (Southern Urals, Russia) is complicated by combined character of tooth exposure: internal (mostly due to $^{90}Sr/^{90}Y$ incorporated in the tooth tissues) and external radiation. The aim of the study was evaluation of the external doses for Techa riverside residents. The external dose can be assessed by subtraction of the internal dose component from a total EPR-measured dose in the tooth enamel. The internal dose in the enamel was estimated based on TL contact beta detection of the radionuclides incorporated in the tooth tissues. About 300 teeth obtained from 170 Techa riverside residents permanently lived in different settlements along the river stream were investigated by EPR. For many of them parallel TL-measurements of ^{90}Sr concentration were performed. The paper present results of tooth dosimetry accumulated during 10 years of extensive measurements.

Introduction

The extensive increase in plutonium production in the Mayak Production Association (Southern Urals, Russia) during 1948–1955, as well as the absence of reliable waste-management technology, resulted in significant releases of liquid radioactive effluent into the rather small Techa River. This resulted in chronic external and internal exposure of about 30,000 residents of riverside communities (Vorobiova et al. 1999). Since 1992, EPR tooth dosimetry is used for the Urals region for external dose estimation. Eleven EPR methods applied in 7 different laboratories were used under this 18 years study. Different methods had unequal quality that should be taken into account under combined analysis. Additional problem of EPR data interpretation is the presence of ^{90}Sr in the tooth tissues. ^{90}Sr and his daughter ^{90}Y are bone seeking beta emitters responsible for additional internal exposure of enamel. Moreover some of donors were treated by x-ray medical exposure influenced on the total enamel dose. Therefore, the following steps are required to evaluate the external doses from EPR measurements performed by different methods during long-term study:

- unification and harmonization of available EPR data obtained by different methods, including:
  * adjustment for systematic bias by estimation and subtraction of background doses, that consists from both the dose due to environmental exposure and possible systematic error;
  * decision on data processing below the detection limit;
  * algorithm for averaging of repeated measurements performed for the same tooth and/or different teeth of the same donor by different methods;
- estimation of internal dose due to 90Sr incorporated into the tooth tissues
- estimation of medical doses that could have contributed to the dose in the tooth enamel
- evaluation of the individual external dose according Fig1.

As it can be seen from Fig. 1 the individual external dose is evaluated from weighted EPR measurement (in case of several measurements for the same donor) by subtraction of average background, internal dose fraction and individual medical dose.

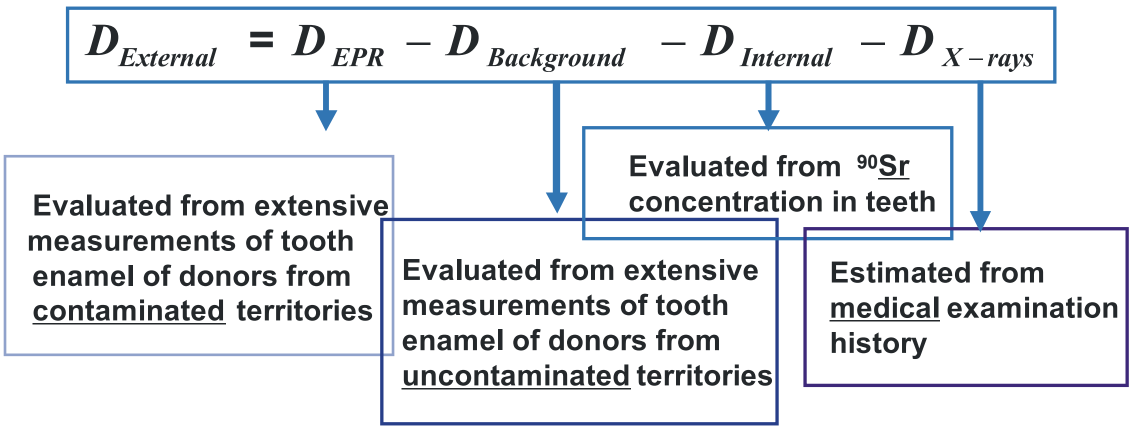

Figure 1. Illustration of assessment of the external dose for Techa river residents from EPR measurements.

Material and methods

The investigated teeth were all extracted because of medical indications at local dental clinics. Tooth donors were identified to allow matching with the roster of exposed persons (Vyushkova et al. 1996). During long-term study teeth were measured at 7 different laboratories using 11 different methods (with different accuracy and precision). The following laboratories participated this long term study: Institute of Metal Physics (IMP, Russia); Helmholtz Centrum Munich (HMGU, Germany); Institute of Chemical Physics (ICP, Russia); Institute of Biophysics (IBP, Russia); National Institute of Standards and Technology (NIST, USA); Medical Radiobiological Research Center (MRRC, Russia); Istituto Superiore di Sanità (ISS, Italy).

Table 1 represents description of the EPR data accumulated in the long-term study and available for analysis. Currently 973 results of EPR measurements for exposed and unexposed population is available in URCRM database. Moreover about 10% of teeth from exposed population were measured twice by different methods and 30% donors donated more than 1 tooth. Teeth from unexposed population were collected in the Urals rural territories from donors with the life style and diet similar to exposed population. Enamel dose obtained as a result of X-ray examinations of tooth donors at the URCRM clinic were estimated from personal medical histories. In total 152 persons with EPR measurements were identified as medically exposed. Calculations of medical doses were done using a method that will be published soon (Wieser et al, to be published). Internal doses from incorporated ^{90}Sr in the teeth were calculated with the method described in Shishkina et. al. (2010).

Table 1. Data on EPR measurements conducted for exposed and unexposed populations obtained by the 11 methods in 1952-2009.

EPR method	Years	Number of teeth (persons) Exposed population		Number of teeth (persons) Unexposed population Arial Bold 9 pt	
		Anterior	Posterior	Anterior	Posterior
1	1992, 1996–2007	87 (62)	206 (125)	23 (19)	84 (80)
2	1993–2002, 2007–2009	36 (30)	83 (60)	17 (17)	70 (55)
3	2000–2002	4 (4)	34 (24)	–	44 (44)
4	2004–2005, 2008	4 (4)	6 (6)	5 (5)	10 (10)
5	2004–2008	23 (13)	4 (2)	24 (24)	26 (26)
6	2001	–	5 (5)	–	2 (1)
7	2001	–	3 (2)	–	–
8	2000–2001	4 (3)	6 (4)	–	–
9	1994	–	14 (14)	–	2 (2)
10	1992–1993	10 (7)	18 (18)	–	–
11	2001	4 (4)	14 (8)	–	–

Results

Unification and harmonization of available EPR data

Pooled analysis of exposed samples has required adjustment of EPR results for proper background levels. The results of EPR measurements for unexposed population have demonstrated that the background level depends on EPR measurement protocol (due to possible bias) and differs for anterior and posterior teeth. In turn, lateral and lingual parts of incisors' enamel have different background doses. Thus, average background dose was calculated for each combination of the methods of sample preparation and EPR measurement separately for anterior and posterior teeth. For some EPR methods tmeasurements of background doses were not conducted (see Table 1) and the decision about background was made from the results of interlaboratory comparisons (Wieser et al., 2000a; Wieser et al., 2000b). Background doses for incisors were calculated separately for lingual and lateral fractions. Some incisors where measured in IMP without separation of lateral and lingual fraction. However it was found no significant difference between whole enamel and lateral fraction. This is reasonable because most fraction of incisor enamel is lateral. Therefore we combined measurements of lateral and whole enamel together for estimation of background doses.

Only one method (method 3 in Table 2) demonstrated the reliable age dependence in background level. For all the EPR methods, most of background doses were below the method detection limit. Therefore average background doses do not reflect real environmental exposure and corresponds to systematic bias. Therefore the average background dose can be assumed as adjustment for systematic bias (parameter for data harmonization).

All measurements of doses in the tooth enamel of exposed individuals which are below the detection limit should be replaced with a value of mathematical expectation of radiation dose. For each method, such value was calculated using the comparison of measurement results that were below the detection limit (DL) for exposed and unexposed donors. Figure 2 illustrates the approach for estimation of dose ascribed to EPR results below the DL. The obtained results are shown in the Tab. 2. These values are also harmonization parameters.

Table 2. Parameters of data harmonization.

EPR method	Years of measurements	Dose ascribed to EPR results below the DL (Δ), mGy	Adjustment for systematic bias ±st.dev, mGy		
			Posterior teeth	Anterior teeth	
				Whole enamel/lateral enamel	Lingual enamel
1	1996–2007	4	110±90	200±130	110±90
2	1993–2002 2007–2009	70 50	50±40 120±60	230±110	200±150
3	2000–2002	70	67±140	–	–
4	2004–2005	20	20±100	–	50±160
5	2005–2008	20	0±70	–	150±130
6, 7	2001	70	60	–	–
8, 9, 10	1992–1994, 2000–2001	4	110±90	200±130	110±90
11	2001	4	120±90	230±110	200±150

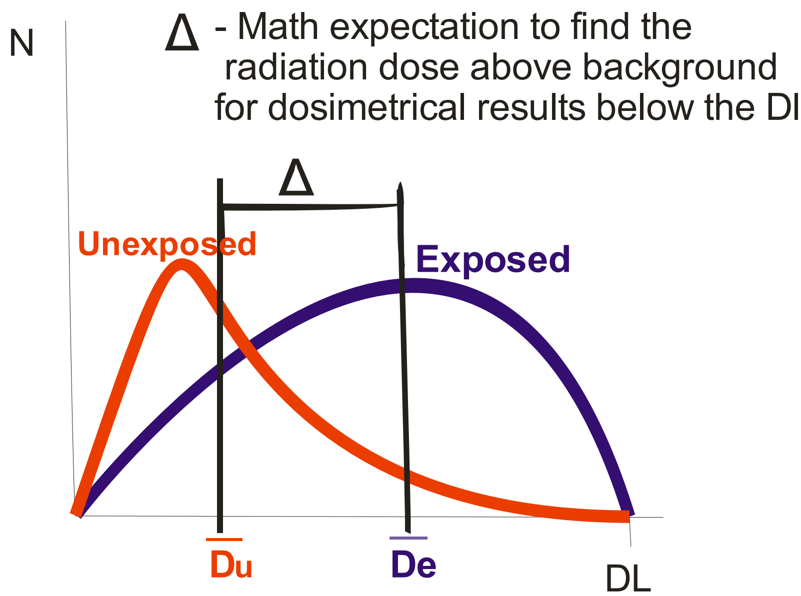

Figure 2. Illustration of the algorithm for estimation the mathematical expectation for dose of incidental exposure which attributes to results of EPR dosimetry below the DL. Du is the average dose in the sample group from unexposed population. De is the average dose in the sample group from exposed population.

The algorithm for pooling data obtained by different methods together for combined analysis was done according to scheme presented in Figure 3, where D is the background dose; Δ is mathematical expectation of incidental dose for measurements below the DL. Doses for teeth which have been measured several times by different methods were averaged accounting for different uncertainties of the methods (δ) and repeatability of the measurement (σ). Doses estimated for different teeth of the same donor were also averaged. As a result, a set of doses for individuals was available for analysis. Conversion of EPR dose in a tooth into external dose ($D_{tooth} \rightarrow D_{tooth\ external}$) was done by subtraction of individual medical exposure and internal dose component.

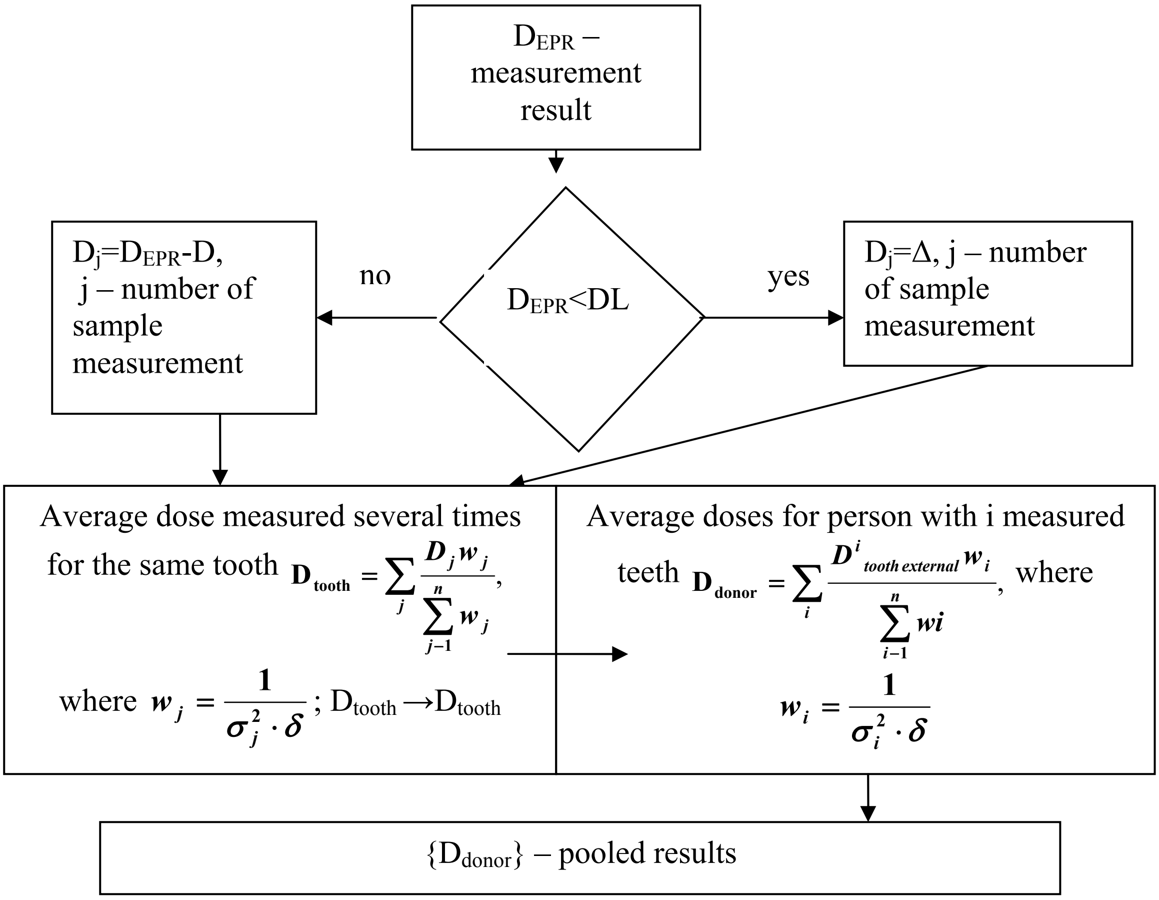

Figure 3. Scheme of data unification for combined analysis. Averaging are performed with weighting factors *w*.

Internal enamel doses due to ^{90}Sr incorporated into the tooth tissues

In Shishkina et al. (2010) it was shown that current levels of ^{90}Sr concentration in different tooth tissues of Techa riverside donors are age-dependent. It was shown that for external dosimetry only teeth aged more than T ≥ 6 years in 1951 can be used for the external dosimetry. The average current concentration of ^{90}Sr in enamel is 0.14±0.12 Bq/g. Additionally some influence on enamel dose of adjasted dentine existes. Based on the model of radionuclide contamination in different tooth tissues and dose conversion factors described in Shishkina et al. (2010) the expected average levels of internal doses were calculated for different tooth age (Table 3). For teeth with known individual contamination of ^{90}Sr internal dose component was calculated individually.

Table 3. Expected age-dependent average levels of ^{90}Sr concentration in the tooth tissues and correspondent cumulative internal doses

Tooth age, years	Concentration of ^{90}Sr, Bq/g			Internal dose in the enamel, mGy
	Enamel	Crown dentin	Root	
6	0.14±0.12	7±5	21±24	120±80
10	0.14±0.12	3±2	15±17	80±50
15	0.14±0.12	1.02±0.40	7±8	42±30
25	0.14±0.12	1.02±0.40	0.4±0.4	40±27

Doses of medical exposure

Medical histories of patients of URCRM clinic were investigated. Table 4 shows the information on X-ray examinations that resulted in additional exposure of tooth enamel, including statistics for donors who have EPR measurements on teeth. The last two columns include the enamel doses expected to be accumulated in the anterior and posterior teeth due to medical observations.

Table 4. X-ray observations and typical doses cumulated in the enamel

Tested organ	Total number of persons	Number of persons with EPR measurements	Average dose in enamel, mGy	
			Posterior teeth	Anterior teeth
Teeth/jaw	496	7	12	64
Skull	1071	34	10	9
Nosal cavities	1365	49	2	1
Neck-bone	1417	61	12	23

The results of individual dose calculation indicated that for 96% of cases of X-ray examinations of donors the additional medical doses in tooth enamel were below 50 mGy. Contribution of medical exposure was subtracted from the EPR dose for all persons who had X-ray examinations to adjust enamel dose estimates for this source of confounding exposure.

External doses for Techa riverside population depend on residence

External enamel doses were estimated from EPR measurements for 170 permanent residents of Techa riverside whose teeth were totally completed at the beginning of exposure. Each donor lived permanently at the same settlement during the period of maximum radioactive release: from 1950 through 1952. Individual EPR-based doses have been grouped according to a person residence site. Figure 4 illustrates group-average doses for residents lived in settlements located at different distances from the release site.

As can be seen, average external dose in tooth enamel was about 200-520 mGy at distances up to 50 km from the release site (individual doses reached 2.3 Gy). Two distinct peaks are visible in this length corresponding to residents of Metlino (located at

7 km frome release site) and Nadyrovo (located at 50 km). Also, large variations of individual external doses for residents of the upper Techa River require continuation of EPR measurements for more reliable conclusions.

At the distances of about 60-90 km from the site of releases the external dose decreased to 120-130 mGy, and was about 50 mGy for those who lived at distancesmore than 100 km downstream from the release site.

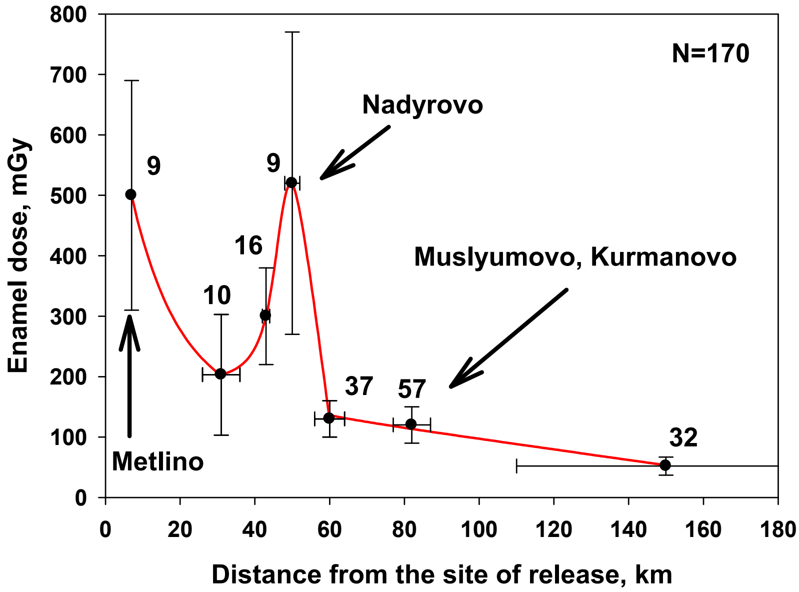

Figure 4. Average EPR-based external dose for permanent residents of the Techa River settlements dependent on the distance from the site of release. Bares indicate standard errors.

Discussion

The EPR-based estimates of individual external dose are used for validation of the external dose calculation made using the Techa River Dosimetry System (TRDS, Degteva et. al. 1998)). The Techa River cohort contains about 30,000 residents and some of them died. Therefore physical retrospective dosimetry with EPR is not sufficient to evaluate the doses for the whole cohort, and mathematical modeling is the only way for estimation of individual external doses. However the models used in TRDS include some assumptions which cand be validated with EPR.

In this connection the peak in external doses observed in Nadyrovo (Fig. 4) is not predicted with the current version of TRDS. This peak could be explained by exposure conditions specific to this part of the Techa River and should be considered more carefully in the further study. The obtained results of EPR external dosimetry will be used for improvement in external dose calculation in TRDS.

Conclusions

1. Algorithm for harmonization of available EPR data obtained by different methods during long-term study was elaborated. Parameters for harmonization were evaluated.
2. Internal doses were estimated. Average internal doses in the enamel for teeth with completed crown are in the range from 40 to 120 mGy

3. The influence of medical examinations on the total enamel dose was evaluated. 96% of doses due to medical exposure was found below 50 mGy.
4. Individual external doses in the Techa River region evaluated based on EPR tooth dosimetry are following:
   - average external dose in tooth enamel was about 200-520 mGy at distances up to 50 km from the release site (individual doses reached 2.3 Gy);
   - at the distances of about 60-90 km from the site of releases the external dose decreased to 120-130 mGy;
   - the doses are about 50 mGy for those who lived at distances more than 100 km downstream from the release site;
   - highest individual doses (up to about 2 Gy) were found in two sites: Metlino (7 km) and Nadyrovo (50 km)

The obtained results of EPR external dosimetry will be used for improvement in external dose calculation in TRDS.

Acknowledgments

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Finland’s approach to licensing and regulatory control of geological repository for spent nuclear fuel

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Abstract
Finland’s program for the disposal of spent nuclear fuel is among the most advanced in the world. The first licensing step, the Decision-in-Principle, which sealed public acceptance on local and national level, was taken in 2000-2001. Site confirmation studies, the construction of an underground rock characterization facility and the development of the safety case for the facility are ongoing and the program is approaching the next step, the submittal of construction license application, expected in 2012. All this was preceded by more than 30 years of work with a long-term goal. In this review, the approaches to licensing and regulatory control of the facility are presented.

Introduction
There are four nuclear reactor units in operation in Finland, two VVER-440/213-type pressurized water reactors (488MW each) in Loviisa and two ASEA-Atom boiling water reactors (860MWe each) in Olkiluoto. The reactors were commissioned between 1977 and 1980. One reactor unit, a 1600MW EPR, is under construction at Olkiluoto. There are applications pending for new reactors by three companies, on four candidate sites. Currently, about one third of the electricity consumption in Finland is produced by nuclear reactors.

From an early stage in the Finnish reactor program, it was considered that the whole nuclear fuel cycle needs to be taken into account when addressing the safety of nuclear energy production. Over 30 years of systematic research and development has been carried out to select a site, to develop engineered barriers and the safety case and, in parallel, a stepwise regulatory approach for the disposal of spent nuclear fuel produced in Finland.

These activities have been done following Government’s long term strategies since 1983. As the first regulatory step, public acceptance at local, Governmental and Parliament levels was gained in the Decision-in-Principle (DiP) in 2000 to locate the repository at Olkiluoto.

The operators of nuclear facilities/producers of nuclear waste, now Fortum Oyj and Teollisuuden Voima Oyj, have full responsibility for nuclear safety, including the safety of disposal of spent nuclear fuel. For the implementation of the disposal, the NPP operators have formed a jointly owned company, Posiva Oy. The Ministry of Employment and the Economy is responsible for the supreme command and control of nuclear matters, while the Radiation and Nuclear Safety Authority (STUK) is responsible for the supervision of safe use of nuclear energy. STUK is an independent regulatory body with broad authorities to ensure that nuclear power is produced in a safe manner, and to give necessary orders for this purpose.

An application for a Construction License of the disposal facility is expected to be submitted at the end of 2012 and operation of the facility to begin in 2020.

Stepwise licensing

In Finland, licensing a nuclear facility proceeds in three steps:

Decision-in-principle

Construction of a nuclear facility of considerable general significance requires a Government Decision-in-Principle on that the construction project is "in line with the overall good of society". Nuclear facilities deemed to be of considerable general significance include nuclear waste disposal facilities.

The Decision-in-Principle is the major political decision on the societal acceptability of the project. An Environmental Impact Assessment process with public consultation and stakeholder involvement must be performed. As a prerequisite for the decision, the Government must obtain the consent of the host municipality and a preliminary safety evaluation by STUK indicating that "no facts indicating a lack of sufficient prerequisites for constructing a nuclear facility have arisen".

The Decision-in Principle is made by the Government and endorsed or reversed by the Parliament.

Construction license

The license to construct a nuclear facility is granted by the Government. It requires an application which must include the justification of safety for the facility, the Safety Case of the designed facility. Based on the application, STUK makes a safety evaluation, which must be positive for the construction license to be granted.

Operating license

The license to operate a nuclear facility is also granted by the Government. The Safety Case of the constructed facility is reviewed by STUK and a positive safety evaluation is required for the Government to grant the licence.

Regulatory control

From a regulatory viewpoint, the Olkiluoto spent fuel disposal project can be divided into the following main phases:

1. Research phase from the late 1970's to the Decision-in-Principle licensing phase (DiP),
2. Research, development and design phase including construction of an underground rock characterization facility (from DiP to Construction Licence),
3. Construction phase (from Construction Licence to Operating Licence),

4. Operating phase
5. Decommissioning and closure phase.

Before the DiP, the progress in siting and research and developed were followed by annual reporting with STUK's statements to the ministry.

In the year 2000, the Government made the Decision-in-Principle. The decision also authorized the construction of an underground rock characterization facility, now called ONKALO, in Olkiluoto. The facility may be later used as the access tunnel to the disposal facility and is therefore constructed under full regulatory control.

In addition, research, development and design work by Posiva continued with the aim to arrive at a facility design and a Safety Case for the next licensing stage by 2012. STUK has reviewed Posiva's plans and evolving drafts of Safety Case documentation, commenting on their adequacy compared to regulator's expectations.

Construction of ONKALO

For long term safety, the main safety functions of the Olkiluoto disposal facility are ensuring the integrity of the engineered containment of the disposed waste and maintaining sub-criticality. The secondary safety functions are the limitation and retardation of the release of radioactive nuclides and isolation of the engineered barrier system from external impacts.

Therefore, it is important that such chemical and mechanical conditions are maintained in the bedrock that the safety functions are not jeopardized over a long period of time in a variety of normal and abnormal circumstances.

Construction of ONKALO to the planned disposal depth (-427m) disturbs the geo-environment and conditions in a variety of ways. The purpose of STUK's regulatory control of ONKALO construction is primarily to ensure that the design, location, orientation and construction are carried out in such a manner that the geo-environment retains its favourable characteristics and conditions needed for the safety functions.

In particular, this implies the minimization of

- host rock responses to excavation, the excavation disturbed zone,
- groundwater leakages to the tunnels and shafts, and
- introduction of foreign, potentially harmful, substances to ONKALO during excavation and operation (cement and other grouting materials, reinforcement materials, explosives etc.).

STUK's regulatory activities (approvals, review and assessment, inspection) of ONKALO are implemented in a graded approach. All structures, systems, components are classified based on their assessed significance to safety.

Posiva's management system is also subject to STUK's regulatory control. STUK primarily inspects safety and quality management of Posiva's organization and reviews Posiva's self assessment of safety culture.

As a basis for review and assessment, STUK requested from Posiva a plan of the documents to be submitted for information or approval. The plan was an adaptation of that defined in the regulatory guide for nuclear facilities. The documents that must be submitted include description of the constructing organization, staff competences, regulations, codes and standards to be used in the construction, quality system documentation, design data, drawings, construction documentation, etc.

In addition, Posiva was required to submit to STUK a plan describing how the company intends to communicate to STUK the progress of the construction work. The purpose of this document is to facilitate well planned and timely regulatory activities.

In STUK's review and assessment of ONKALO construction, requirements in current regulatory guides (for nuclear facilities) are used, as applicable, but in many cases they can only be used as a model and new requirements and criteria suitable for this part of the underground disposal facility have to be created.

ONKALO inspection activities cover all areas of STUK's responsibilities. Inspections are carried out in order to ensure that Posiva is in compliance with regulations, conditions and approvals of STUK. Inspection activities can be divided into:

- Construction Inspection Program,
- Inspections concerning the readiness to begin excavation and work phases, and
- Inspection on construction works on site.

From the Construction Inspection Program, five to ten inspections are performed annually. The inspection program has a three-level hierarchy:

- Level A, Management system: safety management, organization, safety culture, quality assurance, competence of staff, communication with STUK,
- Level B, Main Operations: construction project management and resources, safety issues, quality assurance for construction work, monitoring program, related R&D,
- Level C, Functions and Activities: Posiva's inspections and QC, excavation and excavation disturbed zone, drillings, mapping of features, construction impacts (to geochemistry, rock mechanics, hydrogeology, groundwater leakages to tunnels and shafts, introduction of foreign materials, grouting, enforcement works and materials), physical protection and emergency preparedness.

ONKALO construction is divided into different phases. The purpose of the inspections concerning the readiness to begin excavation and work phases is to ensure that all the arrangements and conditions at the construction site are in order for the next construction phase to start (previous phase is properly completed).

Examples of this type of inspections are inspecting the preparedness to begin shotcreting a specified tunnel section, and inspecting the preparedness to start a new excavation piece-work.

Inspections on construction works on site, carried out at least once in two weeks, focus on work processes, methods and practices, and their quality and compliance with approvals.

Posiva's research, development and design activities

During this period between the DiP and the submittal of Construction License application, STUK's regulatory control of Posiva's RD&D activities focuses primarily on the gradually evolving facility design and the drafts of Safety Case documentation.

Posiva has a program of producing the documentation for the long-term Safety Case, 10 main reports, in a several advancing versions, before final versions submitted with the Construction License application. STUK has reviewed these reports and commented on them, compared with regulatory expectations, and identified safety questions that need to be resolved before the next licensing phase.

As a required maturity test, Posiva submitted recently the current drafts of their Construction License application material to STUK and Ministry of Employment and the Economy. This material includes the draft Preliminary Safety Analysis Report and drafts of most reports included in the long-term Safety Case. The regulators review if the pre-license application demonstrates sufficient readiness from scientific, technological and safety viewpoint to justify the actual construction licensing process to start in 2012.

Regulations, guides and regulatory decisions are developed and taken in parallel with the proceeding RD&D work.

In support of its regulatory staff, STUK has organized the three international experts groups for

- Olkiluoto site safety investigations,
- Engineered Barrier System and Technology, and
- Safety Assessment.

Nuclear Safeguards

As ONKALO is foreseen to become a part of the disposal facility for spent nuclear fuel, STUK decided in 2003 to start to require the application of nuclear safeguards measures to ONKALO. Posiva was obliged to implement safeguards procedures from the beginning of the excavation to the closure of the disposal facility site. As required by STUK, Posiva has prepared and documented all necessary safeguards procedures and measures in their "Safeguards Handbook" and submitted it regularly to STUK for review and approval.

Safeguards activities for spent fuel disposal in Finland have four main objectives:

- to ensure that all safeguards relevant information about the final disposal facility will be available in due time;
- to be able to confirm that there are no undeclared activities relevant to safeguards at or near the final disposal site;
- to enable the IAEA to perform integrated safeguards in Finland;
- to enable the IAEA and the European Commission to plan for their future safeguards activities.

STUK's current safeguards activities consist of auditing Posiva's safeguards implementation, reviewing Posiva's safeguards relevant reports and confirming by on-site inspections that ONKALO is in compliance with Posiva's as-built documentation.

STUK's audit of Posiva's safeguards implementation includes review of the documented results and the observations made throughout the year in connection with report reviews and on-site inspections. Result of STUK audits are fed back to STUK's regulatory process and to Posiva.

Conclusions

Finland's program for the disposal of spent nuclear fuel is approaching the construction license phase. In parallel with the progress of the disposal project, an approach to regulatory control has been developed, as described in this paper.

Preliminary results on Cernavoda NPP operation impact on terrestrial and aquatic biota

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Abstract

Recently, the awareness of the vulnerability of the environment has increased and the need to protect it against industrial pollutants has been recognized. The concept of sustainable development, requires new and developing international policies for environmental protection. (Protection of the environment from the effects of ionizing radiation. IAEA-TECDOC-1091. International Atomic Energy Agency, Vienna.). As it is recommended in "Cernavoda Unit #2 NPP Environmental Impact Assessment it is Cernavoda NPP responsibility to conduct an Ecological Risk Assessment study, mainly to assess the impact of Nuclear power plant operation on terrestrial and aquatic biota. Long records from normal operation of Cernavoda Unit 1, wind pattern, meteorological conditions, and upgread source terms data were used to evaluate areas of interest for environmental impact, conducting to a circle of 20 km radius around mentioned nuclear objective. The screening campaign established tritium level (because Cernavoda NPP is a CANDU type reactor, and tritium is the most important radioisotope evacuated in the environment) in air, water, soil and vegetation, focusing the interest area on particular ecosystem. Using these primary data it was evaluated which are the monitored ecological receptors and which are the measurement endpoints. This paper presents the Ecological Risk Assessment at Cernavoda NPP technical requirements, and the preliminary results of evaluating criteria for representative ecosystem components at Cernavoda NPP.

Optimization of management of radioactive waste generated in research and education centres

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Abstract

The radioactive waste generated in biological research has low specific activity and it is candidate for clearance from regulatory control. Some issues of the management of this waste could be technically improved. With the aim, it has been initiated a research project with a grant of ENRESA.

The objectives are: 1) Characterization of radioactive waste generated; 2) To propose criteria for the management of mixed waste (scintillation vials full of scintillation solution); 3) To suggest clearance values for liquid waste; 4) To propose criteria for the management of uranyle and thorium salts wastes. 1) We suggests to measure the activity of liquid and mixed wastes and to estimate the amount of solid waste. The radiological characterization of selected radiosotopic techniques applying this new protocol has begun. Also, we are complying information about chemical characterization of the waste. 2)-3) It has been considered the incineration as the final way for the mixed waste. This waste can be characterized attending their specific activity and it can be applied the clearance levels for solid waste. Therefore, it is necessary to determine the released activity airborne and the activity burnt to ashes. These calculations are being carried out at this moment. Regarding the organic liquid waste, the final way is also the incineration if the radioactive concentration is below the clearance values.

However, for the aqueous liquid waste, the final way is the discharge via the sewer system. To determine the maximum discharge concentration, the calculations are being carried out taking into account the water flow of the sewer systems and the Annual Limit on intake for ingestion applying the committed effective dose per unit-intake for the ingested radionuclide. The results obtained allow confirming that it will be possible to obtain a final document to serve as a guide to simplify and to standardize the management procedure of radioactive waste generated in biological research.

Introduction

The radioactive installations of Research and Education Centres generate residual materials with radioactive content, they are very heterogeneous and, generally with low activity. Therefore they are mainly clearance. The International Agency of Energy Atomic (IAEA) names this waste as low and medium activity waste. It is necessary that these radioactive installations have an adequate management program for this waste, with technical and administrative actions to control these materials, to optimize their management and to minimize its radiological impact of human health and the environment. Some technical documents, of international reference, have been recently published, to improve the management of this waste in the installations of Research and Education (IAEA-TECDOC-1000 1998, Krieger et al. 2002, IAEA WS-G-2.7. 2005, IAEA-TECDOC-1528 2006).

Two technical guidelines were published in Spain in order to optimize the management of this waste. The first document (Castell et al. 1996) defines the basic components of general management system for this waste. The second document (Macías et al. 2002) developed the theoretical and practical aspects of the first document, in order to propose a Radiological Characterization Protocol presenting the results of radiological characterization of different techniques. The radiological characterization of this waste is a fundamental action to establish the most adequate disposal ways: conventional evacuation or transfer to authorized company. However, the acquired experience during the application of this second document (Macias et al. 2002) highlighted some difficulties due to the fact that the Radiological Characterization Protocol proposed is too complex. On the other hand, some issues related to liquid and mixed waste and other materials have not been resolved and others could be technically improved.

With the aim to try resolving these issues four important research and education centres started a research project with a Grant of ENRESA (National Company of radioactive waste in Spain). It began in July 2008 and it will finalize in December of this year. The centres are Universidad de Alcalá de Henares, Instituto de Investigaciones Biomédicas "Alberto Sols" (Consejo Superior de Investigaciones Científicas -CSIC- and Universidad Autónoma de Madrid -UAM), Centro de Biología Molecular "Severo Ochoa" (CSIC-UAM) and Centro Nacional de Biotecnología (CSIC). These centres have approximately 600 radioexposure workers. The main object of this project is to show significant improvement in the management procedures of radioactive waste just as to achieve homogeneous waste management programs in the radioactive installations of Research and Education. In order to develop this project and to take the technical guideline of 2002 (Macías et al. 2002), as a starting point, the following objectives have been proposed: 1) Radiological and chemical characterization of radioactive waste generated in different radioisotopic techniques. 2) To propose criteria for a homogenous and adequate management of mixed waste (composed of scintillation vials full of scintillation liquid). 3)To suggest reference Levels for liquid waste in accordance with the values propose in different national and international technical documents. 4) To propose a correct management of wastes generated in electronic microscopic techniques.

Material and methods

Gathering and classification of radioisotopic techniques

In order to obtain the data of the techniques performed in the indicated centres, a written form has been applied for, it contains the following information: technical name, frequency of realization and radioactive compound data used.

This process has been carried out in the radiological areas of 101 laboratories of the indicated centres. The obtained data have allowed to identify and to classify the radioisotopic techniques performed actually, taking into account the initial classification in technical groups and subtechniques used by Macías et al. 2002. Ten groups of generic techniques have been established: nº 1. Enzymatic Assays nº 2. Cellular culture Assays; nº 3. Nucleic Acid Hybridisation; Nº 4. Nucleic Acid Characterization; nº 5. Radioinmunoassay; nº 6. Binding assay; nº 7. Radioiodination; nº 8. Radiolabelling of animals; nº 9. *in vitro* transcription ; nº 10. *in vitro* translation. Each group has different subtechniques performed with several radionuclides.

Radiological Characterization Protocol of residual materials with radioactive content

Taking into account the experience acquired with technical Guideline of 2002 (Macías et al. 2002) as a starting point, a new Radiological Characterization Protocol has been developed. It is easier and more functional and it allows a fast characterization of the generated waste in a specific technical. The methodology applied is as follows:

- Acquisition of previous information of the radioisotopic technique to characterize using a data format.
- Weighing all material that will be generated as solid and mixed radioactive waste in each technique.
- Measurement of the radioactive concentration of liquid and mixed waste, and estimation of specific activity of solid waste taking into account:
  - Total activity used in each technique or initial activity.
  - Activity measured in liquid waste and calculations of the radioactive concentration taking into account the whole volume of liquid waste.
  - Activity measured in mixed waste and calculations of the specific activity considering the whole weight generated.
  - Estimated total activity of solid waste applying the expression 1

    A sólid = A initial – (A líquid + A mix) Expressión 1

    A solid: Estimated total activity of solid

    A liquid: total activity measured in liquid waste

    A mixed: total activity measured in mixed waste

  Likewise, the activity percentages of solid, liquid and mixed waste have been calculated with regard to the initial activity used in the assay.

  All these data are collected in a specific form of collection of results.
- Number of characterization for each technique: this paper proposes to carry out twice the radiological characterization at least.
- Measurement procedures: they are indicated in Macías et al. 2002.

Bibliographical revisions

The bibliographical revisions have been carried out using search engines in the web as the ISI Web of Knowledge, Scirus, Google Scholar o Live Search Aca-demic. At the same time, different documents and web pages of international institutions of Radiation Protection, with special attention to management radio-active waste have been consulted, as the IAEA, NEA and UNSCEAR. Further-more, web pages of the Regulatory Bodies as CSN and NRPB.

Calculations of Reference Levels

The necessary calculations have been carried out to propose the Reference Levels applicable to the incineration of residual materials with radioactive content (scintillation vials) and those corresponding to the activity concentration levels in the initial point of discharge to the normal sewerage system

Results

Radiological and chemicals characterization of radioactive waste generated in different radioisotopic techniques

The selection of techniques to characterize have been performed according to the following criteria: high frequency of use, generation of mixed waste and subtechniques belonging to a technique group which did not was characterized in the Guideline of Macias et al. 2002. Table I shows the results of characterization of some characterized subtechiques according to the indicated criteria, applying the described protocol.

Table I. Percentage distribution of activity of generated waste in the characterized techniques (number in brackets indicates the corresponding technique group)

Subthecnique	% in Solid	% in líquid	% in mixed
^{3}H- Adenilil cyclase activity (1)	84.88 ± 0.97	-	15.14 ± 0.98
^{32}P- Ceramides determination (2)	12.37 ± 2.91	87.18 ± 2.94	0.16 ± 0.03
^{45}Ca- Intracellular calcium release (2)	24.49 ± 1.26	75.33 ± 1.47	0.34 ± 0.05
^{125}I- GMPc radioinmunoassay (5)	17.89 ± 4.19	82.07 ± 4.29	-
^{125}I - Melatonine radioinmunoassay (5)	81.19 ± 1.75	18.84 ± 1.77	-
^{125}I- Somatostatin binding (6)	35.86 ± 1.05	64.14 ± 1.05	-
^{125}I- Intestinal vasoactive peptide binding (6)	3.67 ± 0.25	96.33 ± 0.25	-
^{125}I- Somatostatine Radioiodination (7)	47.73 ± 6.01	40.57 ± 4.68	-
^{14}C- Leucine Decarboxilation (1)	99.98 ± 0.87	0.006 ± 0.02	0.005±0.03
^{14}C- Pyruvate carboxylase activity (1)	99.33 ± 1.03	0.670 ± 0.05	-
^{32}P- Northern blot (3)	93.41 ± 2.30	6.59 ± 0.03	-
^{14}C- Etanolamine kinase activity (1)	19.50 ± 1.25	77.72 ± 2.76	2.78 ± 0.12

To confirm the validity of new protocol, some techniques characterized and described in the Guideline Macias et al. 2002 have been characterized with the new document. The results (they are not presented in this paper) demonstrate that the significant differences in the assignation of activities do not exist. At the present, the characterization of remaining techniques is finalising.

In relation with the chemical characterization of liquid and mixed waste, the gathering of chemical compounds of this waste is being carried out. The final results are not available at the moment.

Management of mixed waste with radioactive content

The bibliographical revision performed regarding the management of mixed waste (vials with scintillation solution) has allowed to obtain the following results: 1) The mixed waste is not considered as a defined typology of waste (they are not specifically classified); 2) Very often, the term mixed waste refers to the dangerous or biological waste but not to the vials full of scintillation solutions. (IAEA-TECDOC-1183 2000); 3) Some technical documents suggest the incineration as disposal way (IAEA-WS-G-2.7. 2005). Other minority documents, propose the emptying out of scintillation vials and to dispose the scintillation solution in the specific sink (Northumbria University, 2007).

Quantification of the weights and volume of mixed waste.

The percentage distribution of solid and liquid material in the mixed waste has been determined, weighing the empty and full scintillation vials and applying the scintillation solution density used. The results show 62% of solid content and 38% of liquid content in this waste.

Reference levels applicable for incineration of scintillation solution vials

To propose the reference levels for the possible incineration of scintillation vials two different situations have been contemplated: a) all radioactive content escape as gaseous effluents after incineration; b) some part of the radioactive content remain in the ashes after the incineration. The results shown in this paper derive from the calculations performed by the authors and the study carried out in the Unit of Environmental Radioactivity (CIEMAT), spanish reference laboratory in this field (Mora y Robles 2010).

a) Assessment of the activity that will produce an effective dose of 10 µSv per year in an individual after the incineration of mixed waste

Twenty four hours daily have been considered during which the individual may be exposed to the disposal, considering this during 365 days of the year. The breathing rate applied for the public is shown in table II (ICRP-66, 1994).

Table II. Breathing rate for age groups.

<1 year	1-2 years	2-7 years	7-12 years	12-17 years	>17 years
1043.9	1883.4	3182.8	5584.5	7336.5	8103

The presented calculations have been carried with normal atmospheric conditions and a procedure, previously, established (Mora y Robles 2010).

- For ^{3}H the expression 2 is used:

$$(C_A)_{X1}^{\max} = \frac{(X)_{X1}^{\max}}{(H)_{X1}} \quad \text{expression 2}$$

$(C_A)_{X1}^{\max}$ is the tritium concentration in stationary phase as water vapour, at a distance x1 from the point of gaseous discharge, $(X)_{X1}^{\max}$ is the tritium concentration obtained in air previously and $(H\ )_{X1}$ is the absolute humidity in the atmosphere. It is recommended to use of 6 x 10^{-3} l m^{-3} (IAEA- Safety Reports Series 19; 2001) for the last value.

To obtain the effective dose of 10 μSv/year it is necessary a continual emission of 3.9 x 10^{11} Bq/year of Tritium.

- For the ^{14}C the expression 3 is used

$$(A)_X^{\max} = \frac{(X)_X^{\max}}{(C)_X} \quad \text{expression 3}$$

$(X)_X^{\max}$ is the calculated concentration initially due to the dispersion of Gaussian plume, $(A)_X^{\max}$ is the specific activity of ^{14}C in relation to the total carbon and $(C)_X$ is the Carbon concentration in the air (IAEA-Safety Reports Series 19, 2001).

To obtain the effective dose of 10 μSv/year it is necessary a continual emission of 5.4 x 10^9 Bq/year of ^{14}C.

- When a mixture of both radionuclides is incinerated, the expression 4 must fulfill

$$A(^3H) + 72\ A(^{14}C) \leq 3.9 \times 10^{11} \quad \text{expression 4}$$

Where A(^{3}H) and A(^{14}C) are the eliminated activities (Bq/year) by the chimney.

b) Assesment of the activity that will produce a effective dose of 10 µSv per year in an individual due to the ashes

Considering that the 4.74 % is the percentage in weight of ashes generated in the incineration of scintillation vials (Cañadas et al. 1991) and taking into account that neither the 10^6 Bq/g for Tritium nor the 10^4 Bq/g for the ^{14}C must not exceed if both radionuclides are in the ashes (Spanish Order ECO/1449/2003), the expression 5 must be accomplished.

$$A(^3H) + 100\ A(^{14}C) \leq 10^6 \quad \text{expressión 5}$$

In this case A(^{3}H) and A(^{14}C) are expressed in terms of specific activities (Bq/g) contained in the ashes.

Management of liquid waste with radioactive content

The bibliographical revision carried out allows identifying the necessity of, initially, knowing the chemical composition of the liquid waste to establish the suitable management.

Management of organic liquid waste:

This waste, considering its chemical composition, will always be transferred to an authorized company for radioactive o dangerous waste. If the activity of this waste is lower than the indicated results for scintillation vials (they contain organic liquid) incineration, this paper proposes its transference to the dangerous waste company and subsequent incineration. Moreover, the organic liquid waste with activity more than the indicated values (point 2.2) and its half-life is greater of 100 days; it must be transferred to the radioactive waste company (in Spain ENRESA)

Management of aqueous liquid waste:

The aqueous liquid waste could be directly discharged to the normal sewerage system whenever they can be clearance._Therefore, their radioactive concentration must be lower than the activity concentration levels in the initial point of discharge to the normal sewerage system.

Activity concentration levels in the initial point of discharge to the normal sewerage system. To propose these levels it is necessary to calculate:

- The Annual Limits on Intake (ALI_{ing})
- Maximum limit of concentration of activity in the final point of the the normal sewerage system (CV_{max})
- Maximum limit of concentration of activity of the waste to discharge in the initial point of the normal sewerage system (Cv_{PI})..

a) Calculation of Annual Limits on Intake (ALI_{ing}):
Applying the committed effective dose equivalent of 1mSv by one year for members of the public, 1 mSv (RPSCRI, Real Decreto 783/2001; IAEA-TECDOC-1183. 2000) and the dose coefficient per unit intake for ingestion $e(g)_j$ Sv/Bq for members of the public (age group older than 17 years). (RPSCRI, Real Decreto 783/2001; IAEA-TECDOC-1183; 2000).

b) Calculation of the maximum limit of concentration of activity in the final point of the normal sewerage system CV_{max}, applying the expression 6

$$CV_{\max} = \frac{LIA_{ing}}{600}\frac{Bq}{l} \quad \text{expression 6}$$

The annual ingestion rate of water for adult individual, 600 l (ICRP-23;1975) has been used

c) Calculation of the maximum limit of concentration of activity of the waste to discharge in the initial point of the normal sewerage system (CvPI).
Cv_{PI} : is the maximum limit of concentration of activity of radionuclide in the container (initial point of discharge)

$$Cv_{PI} = \frac{CV_{\max} \times (V_c + V_{ec})}{V_c} \quad \text{expression 7}$$

V_{ec} is the water volume released in the centre daily

Table III shows the results of ^3H y ^{14}C for a centre that releases 45,000 l daily.

Table III. Values of concentration of activity for ^3H y ^{14}C in the initial point normal sewerage system.

Radioinuclide	Compound	ALI (Bq)	*CVmx* (kBq/l)	Cv_{PI} (kBq/l)
^3H	H_2O tritiated	$5.56 \cdot 10^7$	92.593	138,981.48
	OBT	$2.38 \cdot 10^7$	39.683	59,563.49
^{14}C		$1.72 \cdot 10^6$	2.874	4,313.22

When it is necessary to discharge a mixture of radionuclides the expression 8 must be applied.

$$\sum_{i=1}^{n} \frac{Cv_i}{Cv_{PI\,i}} \leq 1 \quad \text{expression 8}$$

These values may be considered as Reference Levels to clearance of aqueous liquid waste, with the approval of the Regulatory Body.

The Consejo de Seguridad Nuclear (CSN), Regulatory Body in Spain, limit the annual amount of discharge to the normal sewerage system to a maximum of 10 GBq of ^{3}H, 1 GBq of ^{14}C and 1 GBq as the sum of the rest of radionuclides, ensuring that in the final point of discharge the concentration of activity does not exceed the concentration limits obtained between the ALI_{ing} (age group older than 17 years) and the annual ingestion rate of water for the adult individual, 600 l (ICRP-23; 1975). This criterion is reflected in the Operation Authorization of the radioactive installations.

Management of waste with Uranyl Acetate salt

In relation to the management of waste with uranyl acetate salt, the gathering to characterize the waste generated in electronic microscope techniques have been carried out. The values of specific activity of Uranyl Acetate salt used are well known, the same as the technique protocol applied. The characterization of this waste has begun. They are, basically, solid and liquid waste and the rest of Uranyl Acetate salt have not been used, in the commercial containers.

The preliminary results are available. They indicate that the total activity of some solid waste and liquid waste is lower than exemption levels (IS-05 – CSN; 2003). Therefore, this waste could be a candidate to clearance and transfer to Hazardous Waste Company.

Table IV. Exemption Levels for U235 / U238

Radionúclide	Activity (Bq)	Concentration (KBq/Kg)
U^{235} / U^{238}	10^{4}	10

However, in some cases the specific activity of solid waste exceed the exemption levels shown in table IV, in this case it will be necessary its transfer to the radioactive waste company.

Discussion and conclusions

We can consider, by the data shown, several comments. Firstly, the application of new Characterization Protocol derived from the indicated project has allowed the easy and fast radiological characterization of the waste. The comparative analysis of obtained results to characterize the same technique applying both protocols supports the suitability of this document. Nevertheless, to determine the waste activity in each step of the technique, it is recommended to apply the characterization protocol described in Macias et al. 2002.

On the basis of the gathering carried out to classify the techniques it is possible to confirm that the amount of mixed waste is low in relation with the solid and liquid waste generated.

The percentage distribution of activity shown in table I indicates that the activity of mixed waste exceeds 10% of initial activity only in one technique, the rest of techniques have very low levels of radioactivity (it is not a significant activity).

The mixed waste is not always considered as a determined type of residual material with radioactive content, in accordance to the bibliographical revision. However, to establish its management it is necessary to classify them correctly. Taking into account its solid content in relation with its liquid content, the mixed waste is considered as solid waste in this project. Therefore, to propose a suitable disposal way for this waste, the clearance levels for solid waste (Order ECO/1449/2003) may be applied. Considering the percentage of activity of this waste shown in table I, it is possible suggest its clearance. However, taking into account its chemical composition (scintillation solution) its disposal as conventional solid waste is not possible, making necessary its transfer to hazardous waste company, the final treatment of which will be incineration. To propose the incineration as management way for this waste, according to IAEA WS-G-2.7, the necessary calculations have been performed to determine the reference levels applicable to the incineration of scintillation vials. Two different scenarios, very conservative, have been considered. In both situations, so the activity that will produce an effective dose of 10 **µSv** per year in an individual after the incineration (390 GBq /año de H-3 and 5,4 GBq/año de C-14) as in the amount of radioactivity that will produce an effective dose of 10 **µSv** per year in an individual due to the ashes are much higher than the used activities in the radioactive installations of research and education.

In relation with the organic liquid waste, regarding the results shown, the more suitable management may be the incineration.

It is possible to confirm that all mixed and organic liquid waste generated in radioactive installations of research and education are clearly candidates to clearance, using as final disposal way the incineration. With regard to the activities, weights and volume of generated waste susceptible for the incineration, the theoretical model used shown that the public dose will be two o three magnitude orders lower than the public dose of 10 µSv to year.

For the aqueous liquid waste, the discharge to the normal sewerage system whenever their radioactive concentration is lower than the activity concentration levels of discharge authorized, it is suggest. Different approximations have been carried out to define these values; finally the calculation of the maximum concentration of activity levels in the initial point of discharge was selected. This approximation has been chosen in relation with others since it is as conservative as the others, but the calculations to identify the clearance waste in each discharge are made easier. The maximum concentration of activity levels in the initial point of discharge, proposed in this paper, must be authorized by Regulatory Body, before its application. Likewise, the discharge limit to the normal sewerage system proposed by the CSN, is adequate for the management of aqueous liquid waste.

Waste generated in electronic microscope techniques, as it has been mentioned, is the most part clearance. However, as happens with the mixed and organic liquid waste, attending to its chemical composition, they might be transferred to the hazardous waste company.

Finally, considering the incineration as the most suitable disposal way for the mixed, organic liquid and Uranyl Acetate salt waste, it is necessary to provide rules and/or technical documents to the hazardous waste companies to facilitate the removal of this waste of the installations.

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Radiation Protection organization in radioactive waste management at the Joint Research Centre of Ispra

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Abstract
The Joint Research Centre of Ispra, one of the research Sites belonging to the European Commission, Directorate General JRC, was created in the late '50s, in order to steer European research on nuclear industry. It hosts numerous nuclear facilities, some of which are maintained in operation, while others were shutdown in past years, namely: two research nuclear reactors, hot cells facilities, radiochemical laboratories, one Cyclotron (still in operation), facilities for studies on fissile material (in operation), and some facilities for the treatment and storage of liquid and solid waste (in operation).

The JRC accounts for 21 nuclear licences, 14 Controlled Zones and 12 main Surveilled Zones, on its Ispra Site.

This paper will discuss the organization which has been developed, during the years, and put in place to guarantee highest safety levels in all radiation work activities at the JRC-Ispra, with an emphasis on radioactive waste management.

The present strategy for waste management at the JRC will be briefly discussed, and its major facilities will be introduced in detail.

Processes discussed will include:

1. Decontamination of surfaces via Abrasive Blasting Unit (ABU)
2. Measurements of materials in the Material Clearance Facility (MCF)
3. Radiological characterisation of waste (active and passive, XDRS and WCF)
4. Radioactive transports and accountancy
5. Interim storage at the JRC

Introduction

Historical background
The European Commission is responsible for the management of its nuclear installations present on the Ispra site of its Joint Research Centre throughout their life by mandate of the Euratom Treaty and the Agreement with Italy.

JRC-Ispra is thus required to safely conserve and subsequently decommission its shutdown installations and to manage the associated radioactive waste. These post-operational activities, together with the management of existing radioactive waste, are the Commission's so-called "Historical Liabilities" at JRC-Ispra. The Nuclear Decommissioning Unit of the Ispra Site Management is responsible for managing these liabilities.

AREA 40, a small nuclear island in the north-eastern corner of the Ispra site, was born in the early 1960s as the nucleus of radioactive waste management services for the site. AREA 40 was initially conceived to provide routine services including temporary and long-term storage of a variety of radioactive wastes originating from the nuclear research and development activities underway at the site. At that point in time, no provisions had yet been made for the support of decommissioning activities, and AREA 40's development proceeded accordingly. By the mid 1990s, the capacity and potential of the waste management infrastructure remained limited for the scope of a full scale decommissioning activity.

In the late 1990s the overall decommissioning and waste management policy underwent a transition from safe shut-down and delayed decommissioning to an accelerated reduction of JRC-Ispra's historical liabilities. This policy change resulted in a sudden need for significant improvements in AREA 40's infrastructure and its management practices: many development and planning projects were launched, soon followed by a first wave of investments and construction activities. This phase is now nearing completion as many important facilities are becoming operational.

Material and methods

JRC-Ispra’s waste management policy

At the highest level, the prioritization of waste management activities is guided by the needs of the decommissioning program and the applicable legislation. The latter can also be considered a prioritization of the safety of both the population and the environment since the evolving legislation is mainly driven by continuous demands to reduce and manage such risks.

At the waste management level, prioritisation follows the natural logic of implementation of a decommissioning program, according to which facilities are first realised and then operated. JRC-Ispra's waste management strategy choices are largely based on the foundations set by many internal and external studies and their critical assessment by JRC-Ispra's hierarchy. Feasibility studies have been conducted that covered the decommissioning and waste management of critical facilities, including JRC-Ispra's two main reactors (Ispra 1 and Essor) and the waste management infrastructure, including a super-compactor, the upgrade to an old cementation facility and a new storage facility. The prioritization of the implementation of these projects is determined by a critical path analysis. In practice, the Interim Storage Facility (ISF) and the definition of a Final Waste Package (FWP) turned out to be the highest priority projects. Other projects are being undertaken as material and human resources become available.

In some cases the range of strategy options has been pre-constrained by external stakeholder's opinions, such as Italian Control Authorities and Italian Regional and

Provincial Government. In this respect, it should be noted that Italian regulations sometimes set stricter criteria than those of related European Directives or their associated technical documents, an example being interregional radioactive waste transport. In particular, JRC-Ispra's general waste management policy is based on the criteria listed in the national guide for waste management, Guida Tecnica 26. These criteria are:

- radiological and environmental protection, as stated by the ALARA principle, i.e. including social and economic factors as well as the impact of the activities on future generations and the environment;
- waste volume reduction at the origin and by implementation of specific treatment and by optimization of waste management processes; and
- classification of waste into three different categories based on the present radioisotopes and their quantity, leading to different confinement times and waste management strategies.

The volume reduction criterion is translated into three further principles specific to the Ispra site:

- minimisation of the amount of un-irradiated nuclear materials and other obsolete materials by recycling them within the nuclear industry;
- maximisation of the quantity of material, either simply suspect or following its decontamination, that is removed from regulatory control; and released from the Site;
- reduction, where practical, of the volume of remaining radioactive waste for temporary storage on the Ispra site.

AREA 40 is intended only to provide interim - not long-term storage - of radioactive waste, since JRC-Ispra's radioactive waste and nuclear material are ultimately destined for a “final national repository”. Accordingly, it is the policy of JRC-Ispra’s to also harmonise, whenever practical, its waste management activities with current and evolving national guidelines. The general waste management strategy adopted by the Nuclear Decommissioning Unit of the JRC-Ispra is based on the application of these principles within the constraints posed by safety and economic considerations.

AREA 40's mission of providing waste management support for JRC-Ispra's decommissioning and waste management program translates into three strategic objectives:

- acquire appropriate waste management capability (internal or external, and on-site and/or off-site, as appropriate);
- manage existing wastes (including, where necessary, their re-conditioning); and
- manage future decommissioning wastes and materials.

Due to the necessity of significant upgrading of the existing waste management infrastructure the waste management activity is initially concentrated in the construction and refurbishment of the needed infrastructure. As the facilities are commissioned and become available for operation, this activity will give way to routine waste processing activities. Although the variety of waste is comparable even with large scale research centres, the actual quantities of waste are relatively small. Hence, what might have been obvious strategic investments for larger establishments was not necessarily the optimal approach for JRC-Ispra, especially over the time horizon of the Decommissioning and waste management Programme established in the Communication to the European Council and Parliament. JRC-Ispra has considered a range of technical solutions as well as their possible provision according to the following the four criteria:

- incur minimal technical risk;
- ease of licensing;
- cost effectiveness; and
- public acceptance.

The waste delivered to AREA 40 for processing and storage must comply with well-defined Waste Acceptance Criteria (WAC) in order to ensure their compatibility with the services offered.

Since plans for a long-term national repository are not yet finalised in Italy, it is assumed that such a repository will be accepting waste starting in 2023 and that all of JRC-Ispra's waste could be transferred by 2028.

Waste management facilities

The waste management plant (*Stazione Gestione Rifiuti Radioattivi*, SGRR) comprises buildings and installations located in AREA 40 and a Post Operational Clear Out (POCO) waste buffer. The eastern part of AREA 40 is used to store radioactive waste; the western to process it. All buildings in the controlled zone of AREA 40 are inside a fenced zone to which both personnel and vehicle access is limited and controlled. The whole area is covered by a fire prevention certificate thoroughly revised after the first wave of infrastructure development. Those buildings that are served by specific Heating and Ventilation (H&V) Systems are limited to the controlled area of building 40 and, more recently, the pair of buildings 41/41c. These buildings are served by a dubious liquid effluent collection network that drains to suitable collection tanks located inside building 40b integrated with the liquid effluents treatment station (Stazione Trattamento Effluenti Liquidi, STEL). Improvements are underway to expand the network to service installations such as a wash cabin.

The routine services offered cover the whole spectrum of radioactive waste management with the exception of waste disposal and are listed and summarised below following the natural time sequence of waste management operation:

- <u>Transport</u>: operations associated with the movement of radioactive waste material on the Ispra site. The main means of transport are forklifts, tractor trailers, vans and a truck with appropriate ADR-certifications.
- <u>Characterization</u>: determination of the physical, chemical and radiological properties of waste to establish its future management. The main installations are the waste Characterization Facility (WCF), the Radioanalytical Analysis Process Laboratory (RAPL), the Waste Characterization Laboratory (WCL) and the X-ray Digital Radiographic System (XDRS).
- <u>Pre-treatment</u>: operations prior to waste treatment, such as collection, segregation, and decontamination. The main installation is the Abrasive

Blasting Unit (ABU). In the near future, other supporting pre-treatment areas like wash and cutting cabins will be refurbished in the hot wing of building 40.

- Treatment: operations such as volume reduction, removal of radionuclides from liquid waste (decontamination), and change of composition that intend to benefit safety, economy or both. The main existing plant on site is STEL; some other capacity for solid waste, such as a sorting cabin and a compactor, might be refurbished or added based on operational experience.
- Conditioning: operations that produce a waste package suitable for handling, transportation, storage and disposal. The existing installation (RWCP, Radioactive Waste Cementation Plant) is presently in standby until a definite strategy, a qualified FWP and validated grouting and solidification methods will be established. Once these aspects will have been defined and necessary authorizations obtained, refurbishment of the existing RWCP or realization of one or more new installations can be undertaken.
- Material clearance: removal of radioactive materials or radioactive objects within authorised practices from any further radiological control by the regulatory body. The main installation is the MCF (Material Clearance Facility).
- Storage: safe and secure conservation of waste in a nuclear facility where confinement, environmental protection and human control are provided. The main installations will be the ISF, the Tank Farm Facility (TFF) and the POCO Buffer Store (PBS);
- Inventory and tracking: maintaining records of waste packages including e.g. data on their physical location, contents and classification. The main tools are the Electronic Transport Module (ETM) and Waste Information and Tracking System (WITS).

Waste infrastructure

This section presents a brief description of the infrastructure development

- Final Waste Packages (FWP): the process of optimisation of the ISF involves the standardisation of waste containers. JRC-Ispra adopted 440-l drums and 5.2 m^3 boxes as the (UNI CC-440 and CP-5.2) as the containers for final category I and II waste packages. The former will be used for sludge, the latter for grouted waste. Precise plans for category III waste packaging and storage are currently on hold while national guidelines are under redefinition.
- Waste Transport: the transport fleet has been designed on the basis of the FWP and will consist of fork lift trucks (3, 7 and 20 tons for the transport, respectively, of 4 × 200-l drums, 4 × 440-l drums, and CP-5.2), a flatbed truck and a tractor trailer equipped with two tanks for liquid effluents. Roads will be upgraded to allow the passage of heavy loads; the upgrade will minimise the use of asphalt in anticipation of the problems that might be involved in its disposal.

- Interim Storage Facility (ISF): Since the availability of a national repository is not expected before 2023, JRC-Ispra has made plans to develop its own interim storage facility to service its decommissioning and waste management activities. This is a lightweight structure to be built in AREA 40 by the end of 2011.
- Post-Operational Clear Out (POCO) Waste Buffer: low-level technological waste will be centralised in a suitably authorised building outside AREA 40 until the ISF will become available.
- Supercompaction: preliminary studies have shown the economic case for a mobile supercompactor, the specifications of which are under definition.
- Grouting: the immobilisation of waste in CC-440 and CP-5.2 FWPs will take place in a new grouting facility that will replace the existing facility, which was designed to treat waste packages no longer compatible with the evolving normative guidance.
- Solidification: due to insurmountable difficulties involved in the authorisation process for the off-site transport, cementation of the about 150 m^3 of treated effluents will take place in a new, on-site solidification installation.
- On-site radiochemical analysis laboratories: JRC-Ispra's current in-house radiochemical analysis capabilities are limited in scope (nuclide and matrix composition) and activity range (very low). A new laboratory for managing samples arising from decommissioning and waste management activities must be developed to extend these capabilities and allow the refocusing of the current laboratory on its original core activity of environmental sample analysis. Plans to develop two laboratories are underway: one for clearable material and low-activity samples, the other for samples needing glove box handling and separate working areas to avoid cross-contamination.

Waste streams

This section presents a brief description of the waste streams and their planned treatment and pre-conditioning.

- Intermediate Level Liquid Waste (ILLW): research and development activities on fuel and high level waste reprocessing resulted in a small quantity of liquid waste essentially comprised of a solution in nitric acid of irradiated fuel. If treated and conditioned alone, the waste would fall under the scope of category III as defined in Guida Tecnica 26 (highest concentration waste). A significant effort has been spent exploring processing options. Off-site options are significantly limited by authorisation questions; on-site options are limited by economic justification of a small-scale intervention. At the present time, the most feasible alternative seems to blend the liquid with the sludge present on site if this should result in category II waste (intermediate concentration waste).
- Sludge: two batches exist on site. The first, currently held in a tank, carries α contamination (210 Bq/g ^{241}Am); the second is stored in drums. Both batches will be centralised in the two new tanks of the TFF designed to store all sludge on the site. Drums will be emptied and washed in a glove

box equipped with a custom-built interface to connect safely to the drums. The empty, used drums will be either washed, cut and possibly blasted in ABU or supercompacted. The sludge will be solidified in 440 l FWP with a suitable cementation recipe. FWPs will be characterised at the WCF and transferred in pallets hosting six drums each to the ISF.

- Orphan sources: AREA 40 stores disused radioactive sources. High activity sources still have industrial interest and their liability is being transferred to third parties, effectively recycling both their active and shielding components within the nuclear industry. It is planned that smaller sources will be characterised (which might involve the on-site laboratory), grouted and packaged in 440 l FWP.
- Bituminised drums: more than 6000 drums of various bituminised wastes are present on site, buried in trenches. The drums will be retrieved, characterised in the WCF and, where necessary, invasively, prior to their transfer and grouting into FWP for storage in the ISF. Compaction of the drums before grouting is also being considered.
- Roman Pits: in the late 60s and early 70s, fifteen Roman Pits built from prefabricated concrete rings superimposed on one another were filled with waste, mostly activated metals, and sealed with concrete. Recovering the pits entails inserting a sleeve around each pit after excavating the surrounding ground and immobilising the sleeve with cement to allow lifting the pit from its burial hole. Once out of the ground, the pits will be laid down in an existing temporary facility. Following the extraction of the pits, the terrain will be radiologically mapped to confirm that no dispersion of radioactivity has occurred either during the thirty years of underground storage, nor during their extraction. Once extracted, a non destructive γ scanning campaign will be undertaken to characterise the pits. Samples will be taken where indicated if possible. The reference strategy is not to treat the pits prior to their conditioning, unless the results of the analyses should give indications to the contrary. In preparation for their shipment to the national repository, the pits will be inserted in special transport containers (essentially stainless steel liners) and grouted. Slicing the 7-m tall pits to allow transportation with conventional means is being evaluated in light of the justification and optimization principles.
- Specific wastes (e.g. soil, wood, asphalt): over the years about 100 m^3 of contaminated soil, wood and leaves were collected in drums and, where contamination was very low, in plastic boxes. Whereas a study has indicated that decontamination by chemical leaching would be feasible and potentially allow the release of 90% of the material, the relatively small volume involved does not justify the construction of a decontamination facility, the cost of which is projected to exceed supercompaction or even direct packaging in FWP. Materials will be characterised in the WCF.
- Concrete blocks: during the 1980s various waste conditioning activities resulted in the production of 570 m^3 in concrete blocks, for a total activity of about 62 TBq mostly from activation isotopes but with significant α content as well as exotic materials. It is assumed that non destructive

characterisation will provide satisfactory verification of existing inventories. The current treatment and pre-conditioning strategy entails packaging and grouting for direct shipment to the national repository.

- Decay pool aqueous effluents: the Essor decay pool have been measured to contain a low level of βγ nuclides (50 Bq/g) and no chemical contaminants. The Ispra 1 decay pool, on the other hand, may contain chemicals that inhibit the precipitation and flocculation process of STEL, thus creating the need for a pre-treatment step currently under investigation, after which the effluent will be sent to STEL. Evaporation may be used to bring the effluent closer to STEL's acceptance limit of (400 Bq/g), thus reducing the effluent volume. STEL's liquid effluent will follow the path via characterisation at the onsite lab to discharge in the environment; the sludge will be treated in the TFF (see Sludge above); and filters and other technological waste will be supercompacted and packaged appropriately for storage in the ISF.
- Reactor graphite: Ispra 1 contains about 22.4 m^3 of graphite which was used as moderator and reflector. The presence of significant quantities of ^{14}C, ^{36}Cl and ^{166m}Ho can be expected and will be defined by a characterisation campaign. Treatment and conditioning will be defined according to specific guidance of the control authority.
- Metallic waste: metallic radioactive waste resulting from both historical liabilities and decommissioning activities is contaminated mostly with ^{60}Co and ^{137}Cs. In some cases ^{3}H cannot be excluded. Most waste is steel and its alloys but lead and aluminium are also found. Pre-treatment will include segregation and cutting; parts that can be potentially decontaminated by blasting will be sent to the wash cabin and ABU. Non-releasable parts will be packaged, grouted and sent to ISF. On-site supercompaction has the potential to offer volume savings and is also being considered as a possible treatment option that, however, could be limited to a part of the waste by the presence of α contamination.
- Demolition waste: the biological shields of both Ispra 1 and Essor and the shavings of all contaminated areas make up most demolition waste. Expected activation products are ^{133}Ba, ^{152}Eu ^{154}Eu and ^{3}H, ^{14}C and ^{41}Ca must also be considered along with ^{60}Co and ^{137}Cs and α contamination. The biological shields will be cut to fit into CP-5.2 FWP. Characterisation will be performed by in situ γ spectrometry and sample analysis. FWPs will be transferred to AREA 40 for grouting. If significant α contamination should be found, the waste will be packaged into Category III FWP, characterised in the WCF and grouted following the guidance of the control authority.
- Combustible waste: this stream includes technological and process waste as well as some specific waste such as wood, leaves, carcasses and oil for a total of up to 110 m^3 from historical liabilities and an additional 500 tons from decommissioning. Due to authorisation difficulties, on-site incineration has been ruled out, and the volume of most combustible waste will be reduced by supercompaction. Other streams, such as oil, solvents and biodegradable waste, will be sent to an external provider after the

necessary characterisation. Return ashes will be packaged for storage in the ISF.

- Compactable waste: this stream includes mainly technological and process waste and potentially metallic parts, building rubble, contaminated soil and ashes. Until a supercompaction service will be available on site, volumes can be managed in the interim with a simple 3-ton press that will be used to compact waste in 220-l drums. These drums may be supercompacted later.
- Liquid effluents: decontamination and cutting operations are expected to produce about 100 m^3 of liquid aqueous effluents that will be treated in STEL, along with reactor pool effluents.

Potentially clearable waste

JRC-Ispra's waste management strategy calls for the maximisation of the quantity that can be cleared as conventional waste. The essential elements for clearance are:

- maximisation of the use of historical information;
- optimising in-situ pre-clearance measurements;
- minimisation of samples and scope of the analysis;
- maximisation of waste material handling and flow;
- optimisation of clearance measurements; and
- efficient application of quality assurance and control procedures.

If, e.g. on the basis of historical information, contamination of waste material is suspected, or if it has been decontaminated, 100% radiometric controls will be implemented to assess the homogeneity of the activity distribution where practical. Samples will be taken to assess the presence of emitters not detectable with field detectors. If the stream will result homogeneous, further analysis will produce the nuclide vector of a sample. The waste will then be routed to clearance measurements. 0.7-m^3 containers were chosen as a compromise between handling and measurement needs. The maximum averaging volume and mass allowed by normative guidelines were also important factors in the choice. The Material Clearance System includes two total gamma and a gamma spectrometry measurement chain. These measurements are complemented with sample-based data. A characterisation plan, addressing the qualification of the verification measurements will be submitted for evaluation, comment and, eventually, approval by the regulatory authority.

Radiation Protection organisation

All operational activities in Waste Management area are covered by permanent assistance by the JRC-ISPRA Radiation Protection Sector (RPS). This Sector, providing services also to operating facilities and during pre-decommissioning works, is presently composed of 15 Permanent Agents (Civil Servants of the European Commission) and is supported by an external Radiation Protection Assistance Contract.

The RPS is involved in the evaluation and discussion of all projects related to waste management, and is providing operation assistance on a daily basis.

The RPS is structured upon some main pillars:

1. an **Operational Radiation Protection group**, subdivided in three Teams, providing operational assistance on a daily basis, dose and contamination measurements on controlled areas, pre-clearance spot checks on materials, routine dose and contamination mapping, air contamination evaluations, workers' electronic dosimetry readings, etc., and proving guidance and support to workers;
2. the **Dosimetry Laboratory**, providing TLDs for whole body, finger and wrist dose evaluations, gamma, neutron and radon dosimetry;
3. the **Whole Body Count Laboratory**, providing direct measurements of internal contamination, either on a routine or incidental basis;
4. the **Electronics Laboratory**, dedicated to the management and maintenance of the over 1.000 fixed and portable radiation measuring equipments in the JRC-ISPRA;
5. the **Transport & Accountancy group**, intended to account for presence, movements and uses of radioactive material (sources, fissile material, waste, etc.) in the JRC-ISPRA, and managing the JRC "*Committee for Fissile and Radioactive Materials*";
6. the **SIT Calibration Laboratory**, a certified centre for the calibration of radiation measuring equipments in the JRC-ISPRA
7. the **nuclear emergency** preparedness and response group;
8. the **Radiation Protection Archives**, managing all workers' dosimetric data (WBC, RTX, TLDs, EPDs, medical reports and aptitudes, training, etc.)
9. a **Documental group**, managing the drafting and diffusion of technical procedures and documents, and managing Radiation Protection training actions in the JRC-ISPRA;
10. the **Radiation Protection Assistance contract**, providing assistance in operational and Laboratory activities (25 individuals in 2010)
11. the **Qualified Expert**, an internal Advisor, appointed by the Ispra Site Management Director, who is legally responsible for the correct management of Radiation Protection issues on Site.

The safe decommissioning of two plutonium contaminated facilities at Dounreay

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Abstract
The Dounreay site is situated on the north coast of Scotland, mainland United Kingdom, and is operated under contract to the UK's Nuclear Decommissioning Authority (NDA) by Dounreay Site Restoration Limited (DSRL), a wholly-owned subsidiary of UKAEA Limited. Dounreay was instrumental in fast breeder research and fuel reprocessing plant development. The site's business is now one of safe decommissioning and remediation.

This paper discusses the decommissioning of a plutonium criticality test facility and an experimental pulsed column technology solvent extraction facility. There is a detailed discussion of the radiation protection aspects of the decommissioning and the evaluation and implementation of novel, as well as "tried and tested" radiological protection controls within plutonium contaminated environments.

The paper concludes with a review of the radiation protection challenges and successes of the decommissioning techniques used, from a radiological protection perspective, and the lessons learned in successfully completing these challenging decommissioning projects.

Introduction
The Dounreay site was instrumental in fast breeder research and fuel reprocessing plant development. The site's business is now one of safe decommissioning and remediation. This paper discusses the decommissioning of a plutonium criticality test facility and an experimental pulsed column technology solvent extraction facility.

Decommissioning the Criticality Test Facility
The Criticality Test Facility was a laboratory (Fig. 1.) built in the late 1950s to provide data on solid and liquid plutonium criticalities in support of reprocessing plant design and development. The criticality cell (Fig. 2.) comprised a seven metre high by eight metre diameter domed cylindrical steel pressure vessel. This was surrounded by a 1.5 metre thick hexagonal concrete biological shield.

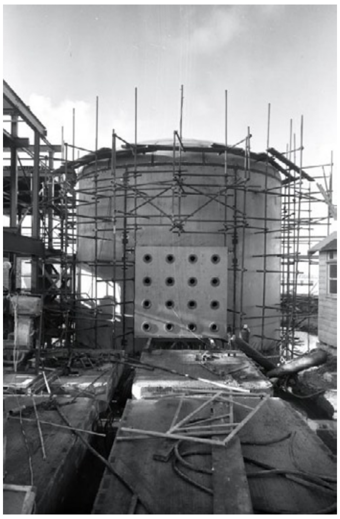

Fig. 1. The Criticality Cell during construction

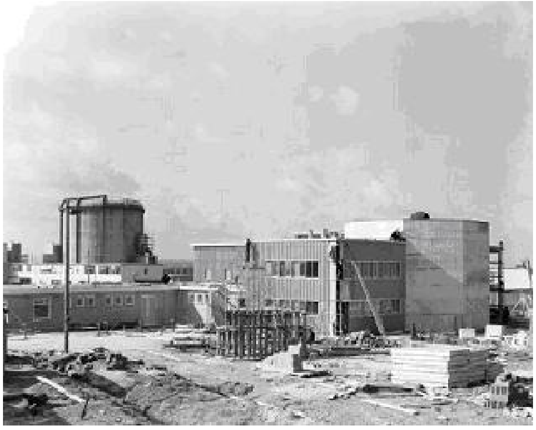

Fig. 2. The Criticality Test Facility

By 1994 most of the plutonium inventory process vessels had been decommissioned. The radiological protection aspects of this initial work have been documented in Nicol 1994. The work after this involved the removal of the highly contaminated criticality test cell and decontamination of the building structure before demolition to “brown field” status. This started in 1999 and involved carefully planned pressurised suit entries into the plutonium contaminated criticality cell. The radiological protection aspects of the early phase of this work were reported in Thompson 2002.

Unfortunately the cell contained a large amount of historical waste that had been left over from previous work. This waste was successfully size reduced, packaged and sentenced to the appropriate waste route. The cell’s internal structure (approximately $300m^2$) was decontaminated using the “sponge jet” technique. This successfully reduced the steel liner from Intermediate Level Waste (ILW) to Low Level Waste (LLW).

Installation of a new ventilation system (designed to be easily removed) was completed to allow the decommissioning of the now 50 year old ventilation system, whilst still providing the required ventilation capability for the remainder of the plants decommissioning. This also involved re-sealing the criticality cell to allow the ventilation capacity to be diverted to support decommissioning of other rooms within the facility. This work was carried out on a room by room basis. On some walls, particularly at locations where wet systems were located, the contamination had penetrated deep into the structure and required complete demolition, as decontamination via scabbling and other methods was not viable. To reduce manual handling, and mitigate radiological and industrial safety hazards much of the demolition was carried out using a Brokk robotic machine (Fig. 3. and Fig. 4.).

Fig. 3. Demolition of internal walls with a Brokk **Fig. 4. Completed internal wall removal**

On the remaining walls and support structures a thin sodium iodide detector was used to detect the low energy X-rays from the plutonium isotopes, to assess whether there was significant contamination below paint and within the structures. This allowed a decision to be made about whether further decontamination was required before demolition. This type of detector arrangement has been used successfully during a number of projects at Dounreay, to identify areas of plutonium contamination which could not be found using conventional surface contamination monitoring techniques.

Following a re-configuration of the ventilation system, attention then reverted back to the removal of the 13 mm thick steel pressure vessel of the criticality cell. Following an option study, work to remove the vessel was carried out using hand held oxy gas cutting equipment by operatives gaining access from an extensive scaffold structure (Fig. 5. and Fig. 6.). Essential preparatory work included the local monitoring and decontamination of all the cut lines, to ensure that contamination did not migrate with the highly mobile cutting fume. The monitoring included checking of the steel surface with the thin sodium iodide detector to provide confidence that contamination, undetectable with conventional instrumentation, was not present in the surface of the metal. This was backed up by laboratory analysis of samples of the metal. This process additionally prevented the inside surface of the bio shield from becoming contaminated.

Fig. 5. and Fig. 6. Hot work to remove the Criticality Cell liner and pit

The work culminated in a complete radiological survey of the building which included conventional health physics monitoring, plutonium X-ray monitoring and sampling of the building structure before the building was declassified and freed for demolition. Demolition was completed using conventional techniques (Fig. 7.).

The completion of this challenging project removed a significant liability from the site (Fig. 8.) and was a very significant success story for the Dounreay site.

Fig. 7. Demolition of the Criticality Test Facility

Fig. 8. The completed project

Decommissioning the Pulsed Column Laboratory

The Pulsed Column Laboratory (PCL) was built for full scale reprocessing flow sheet trials, using pulsed column technology (Fig. 9), which was a more efficient solvent extraction process and a replacement for "mixer settler" solvent extraction technology. The laboratory housed a large sectional glass pulsed column rig inside a glovebox with dimensions of 11.5 metres (high) by 6.5 metres (wide) and 1.2 metres (deep) which extended up four floor levels within the building (Fig. 10.).

Fig. 9. Pulsed Columns

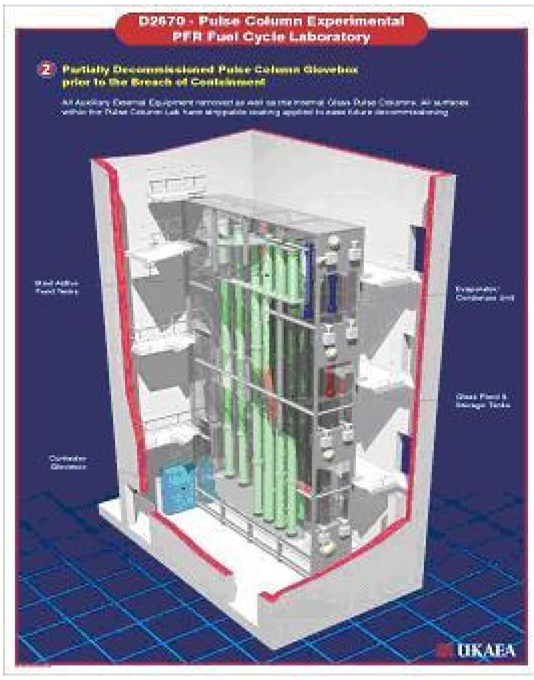

Fig. 10. The Pulsed Column Laboratory

During the early stages of the decommissioning work (Thompson 2001, Thompson 2002) in the PCL there were a number of low level intakes of radioactive material identified following the provision of nose blow samples. The contamination in the nose blow samples was not accompanied by significant surface contamination or airborne activity monitoring results.

A thorough investigation into the intakes highlighted that the contaminant was likely to be of very small particle size (sub micron) and that this was proving to be a significant challenge to the containment and ventilation system that had been installed for the decommissioning work.

The initial strategy involved breaching the glovebox to remove its internal glassware and services which were then size reduced in a purpose designed glass crusher before being disposed of as ILW. This resulted in a spread of contamination to the laboratory. Although tie down materials and decontamination campaigns were regularly undertaken, the laboratory effectively became the primary containment. It was clear that the contamination in the laboratory was providing a challenge to the ventilation system. To address this, a significant change in strategy was implemented with the fundamental principle being to regain and retain the glovebox as the primary containment.

The following text discusses the improvements that were made to the decommissioning strategy, containment, ventilation and access arrangements.

Ventilation and access/egress arrangements

The original ventilation design was set up to draw air through each of the four pressurised suit access corridors to the laboratory. The effectiveness of the ventilation system to provide adequate containment velocities at the containment barriers was sensitive to prevailing external weather conditions and internal movement (opening of doors, etc.) within the facility.

These access arrangements (Fig. 11.) were also convoluted causing problems with turbulent airflow at the containment barriers and curtains. This created a significant challenge to the dynamic containment of the facility, due to poor consistency of face velocities at barriers and turbulence at barriers and barrier curtains.

To improve this situation a number of significant modifications (Fig. 12.) to barrier arrangements and the supply ventilation flow distribution were implemented, including:

- Sealing of two access floors, which reduced the volumetric flow through these areas. This provided increased ventilation capacity to support dynamic containment at the two remaining access barriers and increased the volumetric airflow rate through these access areas. The increased volumetric flow rate also allowed the removal of the barrier curtains and barriers without any reduction in the required containment velocity of ~1m/s.
- Using prefabricated 'oval' doorways (without curtains) and profiling to streamline corners, etc. This reduced the amount of turbulence in the access arrangements and the fluctuating air flows noted during previous decommissioning entries as curtains are opened and closed (Fig. 12.). Furthermore, this modification eliminated the risk of contamination spread onto curtains.

- Modification of supply ductwork was necessary to implement improvements to airflow in the main change area, where original duct arrangements resulted in air being blown across the face of the barrier rather than evenly over it. A new supply grill and baffle plate was installed to deliver supply air to the ground floor sub change area and to promote an even flow pattern in the sub change areas.
- New supply ducting and ventilation grills were installed to the space ahead of the main change barrier to increase supply volume to the main change area and to promote an even flow pattern over that barrier. To feed these new supply points supply air was transferred from the closed off floors to the ground floor.

Fig. 11. The original PCL access corridor

Fig. 12. The improved PCL access corridors

Following implementation of the above, an extensive study of the airflow pattern involving smoke tests, recording and graphing of airflow velocities was carried out. Whilst the barrier flows improved, there was still a need for further improvement. To increase the total extract the power cabling and components for the duty and standby fans were uprated, this included the replacement of the two extract fan motors (duty and standby). To safely take advantage of the increased fan motor power, additional extract capacity was utilised by permanently using all four extract filter banks, as the original plant set up was to use three duty and one standby.

Further inspections and measurements were carried out after the fans were commissioned. Barrier flows were visually inspected using smoke tests generated by an industrial generator. This allowed visual confirmation that the access modifications had resulted in the smoothing of airflow profiles and subsequent much reduced turbulent eddies in the flow characteristics. The smoke tests were replicated with an operator standing at the oval door way which demonstrated that the air flowed past the operator without any significant backflow being induced. This exercise also proved to be a useful demonstration to the operators of what had been achieved.

Containment

The completion of the access arrangements and ventilation improvements allowed re-entry to the laboratory to be made under better conditions. Work to decontaminate the laboratory and re-instate the glovebox containment could now begin. This was achieved by installing a purpose made cover and bag posting arrangements (Fig. 13.) to the open glovebox. This and further covers were installed progressively as the glovebox was emptied and decommissioned (Fig. 14.). Material posted out of the glovebox was via installed bagging ports (Fig. 15.) to be either sentenced as ILW or transferred to the Dounreay decontamination facility for cleaning and disposal as LLW.

Fig. 13. Installation of the PCL Top Hat

Fig. 14. Installed posting ports for the removal of glovebox internals

Fig. 15. Posting operations within the PCL

Development of a new pressurised suit

As part of the thorough review of the arrangements at the PCL it was acknowledged that there is a period in the airline suit undressing process, when respiratory protection has to be removed and personnel are not fully protected. To alleviate this issue, a new single use suit with an air fed respirator integrated within the suit was developed (Thompson 2008). This new suit (Fig. 16.) offers the following features:

- An air fed respirator integrated within the suit, offering continuous protection both within the suit and following suit removal.
- Disposable, offering single use and minimising cross contamination potential.
- Improved wearer comfort, when worn with new moisture wicking undergarments.
- User involvement (and importantly ‘buy in’) in suit development.
- Less folds, creases and therefore contamination traps than the two piece suit that had been used previously.
- Easy to remove, minimising disturbance of the air and potentially contamination on removal.

The suits were used for the rest of the decommissioning work and proved to be very popular with the operators. They are very straightforward to put on and remove. It was novel in the United Kingdom for operators to wear respirators inside a pressurised suit, but feedback from the operators has been very favourable as the air fed respirator provides a good flow of air to the head area and has not caused problems with perspiration.

Fig. 16. The new pressurised suit with integrated air fed respirator.

The work to remove the PCL Glovebox and decontaminate the laboratory was completed in March 2010 (Fig. 17.), removing a significant hazard by employing innovative techniques to enable the safe decommissioning of the largest glovebox on the Dounreay Site.

Fig. 17. The PCL Glovebox is decommissioned

Conclusions

Our experience with the decommissioning of these two highly plutonium contaminated facilities, has provided a number of lessons, including:

- New facilities should be designed with decommissioning in mind.
- Containment of contamination at source is paramount in the safe decommissioning of plutonium contaminated facilities. Restricting the spread of contamination to the “work area” reduces the challenge to the ventilation system and the pressurised suit access areas.

- Glove boxes should be Post Operative Cleaned Out (POCO) immediately after cessation of work.

- Facilities should not be left in care and maintenance for long periods before decommissioning as facility specific knowledge and experience can be lost.

- Ventilation system requirements for the decommissioning of plutonium facilities should take into account potential routine and accidental airborne contamination challenges to the working area. The challenge posed by sub-micron particulate to the containment must be considered. The ventilation system should also be constructed with contingency beyond the minimum calculated requirement to ensure that there is additional capacity available.

- To enable unrestricted movement of personnel in and out of the high-risk area, without a significant cross-contamination potential, the space available for ingress and egress should be adequately sized and not convoluted. The addition of barriers and convoluted access ways can also lead to turbulence and the instance of turbulent eddies and back-flow.

- A considerable amount of time and effort should be allowed for the final surveys and decontamination of facilities. The use of instrumentation that measures the low energy X-rays from the plutonium isotopes to assess whether there was significant contamination below paint and within the walls and support structures etc can be extremely useful in the declassification process.

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Radiation protection issues related to the decommissioning of the DR3 research reactor

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Abstract
The research reactor DR3 was closed in 2000 after 40 years of operation. It was a Pluto type reactor operated at 10 MW (thermal power). It was heavy water cooled and heavy water moderated and it had a graphite reflector. The reactor was used for neutron physics experiments and isotope production. At present the reactor fuel has been removed and most of the peripheral systems have been dismantled. The main construction parts will be decommissioned from 2012 to 2016. The planning of the decommissioning is made by a group of former reactor engineers, mechanical engineers, reactor physicists, technicians and health physicists. Through discussions in the group possible schemes have evolved on how to decommissioning the reactor. Some of the main constructions parts have high contents of activity e.g. many GBq of ^{60}Co and they are posing a major challenge regarding external exposure during dismantling. Graphite dust containing ^{14}C and primary tritiated heavy water leaked into the graphite and concrete are potential sources of internal exposure. A characterization study of the reactor components together with constructional and operational information is the basis for dose rate calculations related to the future dismantling operations. Examples of calculated dose rates anticipated in some of the critical decommissioning operations are presented together with other important radiation protection issues.

1. Introduction
The DR3 research reactor was a Pluto type reactor supplied by the English company Head Wrightson Processes in 1960. It was in almost continuous operation at 10 MW until 2000 where it was permanently shut down. The reactor was moderated and cooled by heavy water and had a graphite reflector. The reactor was primarily used for physics research (neutron scattering), radionuclide production and silicon doping. 18 experimental tubes gave access to positions close to the core position. Neutrons (thermal and cold neutrons) were guided out of four openings in two horizontal tubes and dispersed to several spectrometers. Figure 1 shows a schematic cross section of the reactor construction.

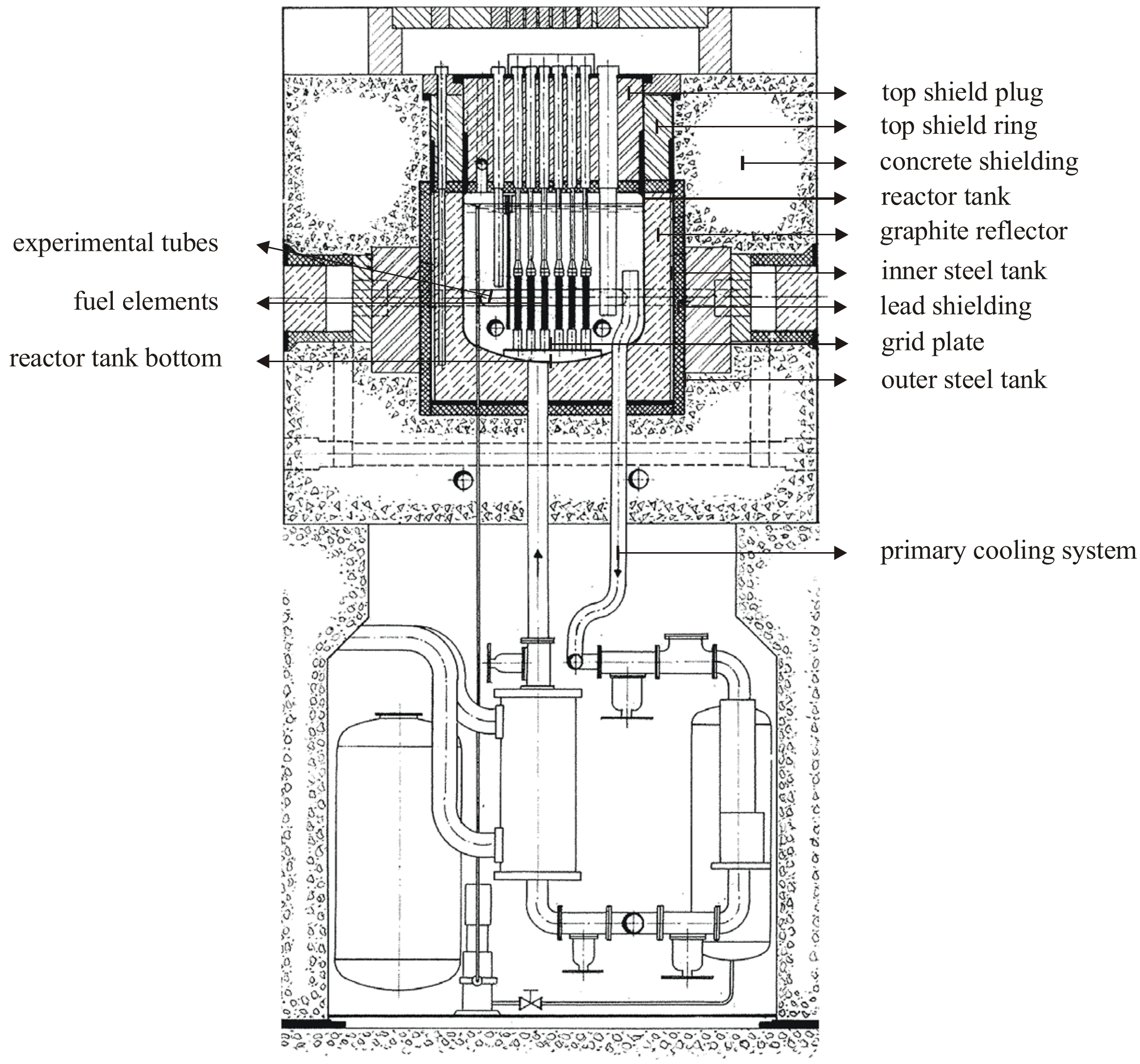

Fig. 1. Schematic cross section of the DR 3 reactor construction. The reactor aluminium tank contains the core and the heavy water moderator. Outside the tank the graphite reflector is situated in the inner steel tank. Outside the steel tank a lead and Boral shielding are situated in another, outer steel tank. Outermost is the concrete biological shielding. The reactor aluminium tank is closed at the top by a top shield plug (TSP) and likewise the steel tank with the graphite reflector is closed by a top shield ring (TSR). Below the reactor construction is the heavy water room containing the pumps for the primary cooling circuit.

The reactor block is placed in a steel containment which has an entrance for persons and an entrance for trucks. An auxiliary building used for active handling and storage of radioactive items is connected to the containment building at the truck entrance.

2. Preparing for decommissioning

Shortly after the final shut down the fuel was removed and the heavy water was drained from the primary and secondary systems. In 2003 the Danish parliament decided that DR 3 (along with the other nuclear facilities at the Risø site) should be decommissioned, preferable to a 'green field' status and a state company Danish Decommissioning (DD) was established with a core of personnel from the former nuclear safety department at the Risø National Laboratory.

In 2004 to 2005 a characterization project was carried out [Ølgaard 2006] with the aim of determining the amount of activity in the primary reactor systems and in the concrete in the reactor block.

From 2005 to 2006 another characterization project [Ølgaard et al. 2007] characterized the activity contents in components situated in the DR 3 storage facilities.

From 2006 and to the end of 2010 the peripheral reactor systems i.e. systems outside the reactor block are or will be dismantled and removed, leaving the bare reactor block and equipment necessary for the decommissioning e.g. the crane in the containment.

The decommissioning planning for the DR 3 started in 2008 and will continue until the actual decommissioning takes place from 2012 to 2016. Experience gained from the decommissioning of the DR 1 and DR 2 reactor will be used (Ingemann Larsen et al. 2008).

3. Radiation protection issues

Many radiation protection issues will arise during the decommissioning of the DR 3 reactor. Below some of the most important issues are addressed.

3.1 Radiation protection personnel

The Section of Radiation and Nuclear Safety at DD includes four health physicists (HP) and 11 health physics technicians (HPT). The personnel have a comprehensive education and training in radiation protection (Lauridsen 2009). Several have experience from former reactor operation and most have experience from the decommissioning of the DR 1 and the DR 2 reactors that were on the site.

The section serves all radiation protection works at DD including a 24 hour on site presence of a HPT. During the preparatory and planning phase 1.5 HP and 1 HPT has been allocated directly to the DR 3 decommissioning project. Additional manpower has been used for clearance measurements on equipment from the peripheral system parts. When the DR 3 decommissioning starts it is foreseen that 2 HP and 2-3 HPT will be allocated.

3.2 Sources of radiation and contamination

From the characterization project and some additional measurements the activity concentration and total activity content in the different reactor components have been assessed. Table 1 gives the values of ^{60}Co (which is the dominating source of external exposure) in some of the components primo 2012. The uncertainties on the values are estimated to be 20-200%, as they are obtained by sampling, which need not to be fully representative.

Table 1. Assessed average activity concentrations and total activity of ^{60}Co in selected reactor component primo 2012.

Component	Average activity concentration [MBq/kg]	Total activity [GBq]
Reactor tank (Al), lower part	80	30
Reactor tank (Al), bottom part	55	10
Experimental tubes	55	15
Grid plate	80	10
Graphite reflector	5	65
Inner steel tank, lower part	23	190
Outer steel tank, lower part	2.3	6.6
Top shield plug, bottom iron plate	7300	2200
Top shield ring, bottom iron plate	1300	300

3.3 Planning

A group of 10 people including one HP is responsible for the decommissioning planning. This group has expertise within reactor operation, reactor physics, mechanical engineering, radiation protection and decommissioning. All the group members have a basic understanding of radiation protection. The group has held one meeting every month from the end of 2007 and two meetings every month from the end of 2008.

Based on knowledge of the reactor construction and the activity contents in the construction parts, critical radiation protection elements in the project have been identified. Some of these have required additional sampling and measurements.

The group has produced preliminary plans for major steps in the decommissioning of the reactor some of which are alternative ways of dismantling. When the preliminary plans have been evaluated, the group will decide for a final decommissioning scheme and a detailed plan will then be made. The planning group will also participate in the actual decommissioning.

Plans are discussed and presented within the Dido Group which has members from other nuclear facilities with a Dido or a Pluto type reactor (Harwell, Dounray, Jülich and Ansto). Also an international expert panel has commented on some of the preliminary plans and will look thoroughly on the final overall plan as will the Danish radiation protection authorities.

3.4 External exposure

To assist the planning and evaluation of plans in terms of dose and shielding costs, radiation transport calculations have been made using the Monte Carlo method (MCNP) and the point kernel method (summation of point source responses). In cases where both methods have been used good agreements have been obtained. Figure 1 and Figure 2 show MCNP-calculated dose rates primo 2012 at different heights above the bottom of an open reactor pit, with only the top shield plug removed. Dose rates are shown as a function of distance to the central reactor axis. In the Figure 1 calculations air was in the reactor tank while in the Figure 2 calculations water filled up most of the reactor tank. It is seen from the two figures that filling up the reactor tank reduce dose rates by roughly an order of magnitude.

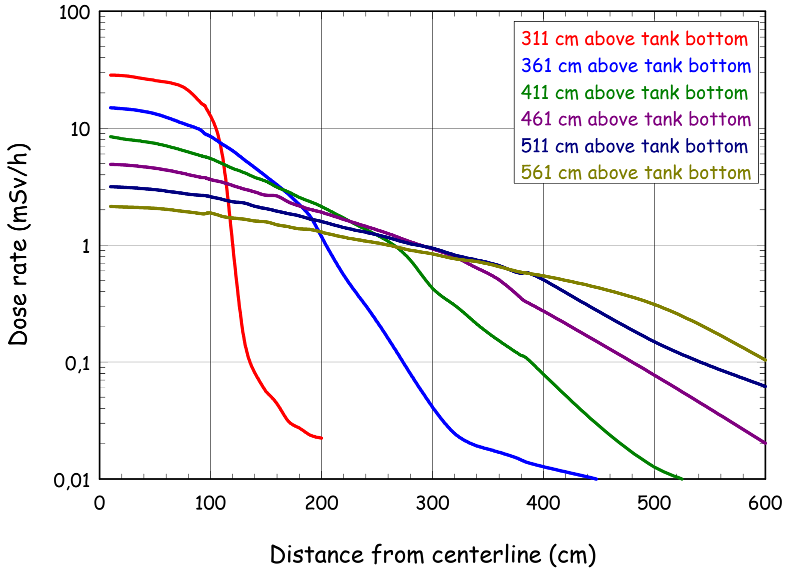

Fig. 1. Calculated dose rates primo 2012 at different heights above the bottom of an open reactor pit with only the top shield plug removed. Dose rates are shown as a function of distance to the central reactor axis. Top of the reactor is 311 cm above bottom.

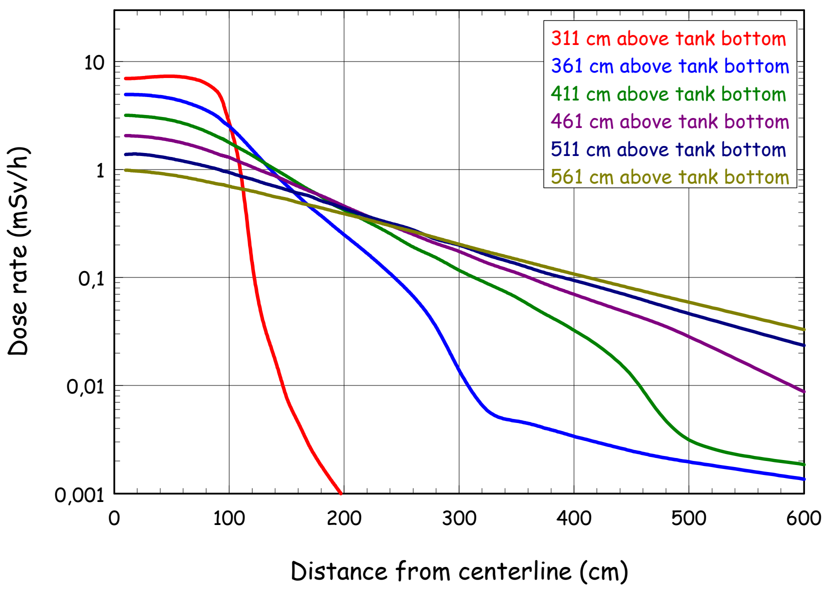

Fig. 2. Calculated dose rates primo 2012 at different heights above the bottom of an open reactor pit with only the top shield plug removed and with water in the tank. Dose rates are shown as a function of distance to the central reactor axis. Top of the reactor is 311 cm above bottom.

3.5 Internal exposure

The induced activity contents in the reactor components constitute sources of internal exposure as the dismantling will require the use of cutting tools. In addition a leak in the primary cooling system has supposedly coursed a widespread contamination with tritiated heavy water in especially the graphite and the concrete shielding. Also it is expected that graphite dust is already present some places in the graphite. From the estimated contents of radionuclides in the different reactor components the potential for internal exposure has been calculated. Table 2 gives the mass concentration in air equivalent to an effective dose rate of 1 µSv/h. Conservative activity to effective dose conversion factors have been used and tritium contents due to leakage has not been taking into account. For materials containing more than one radionuclide, the equivalent masse concentration is calculated as:

$$\left(\sum_i \frac{C_i}{dc_i}\right)^{-1}$$

where C_i is the concentration of nuclide i in the material (Bq/kg) and dc_i is the activity concentration in air (Bq/m^3) of nuclide i which is equivalent to an effective dose rate of 1 µSv/h.

Table 2. Mass concentrations in air of different reactor materials equivalent to a dose rate of 1µSv/h.

Reactor material	Equivalent mass concentration in air (µkg/m^3)
Graphite reflector	1.4
Al reactor tank	0.4
Steel inner tank	1.4
Concrete shielding	30
Primary heavy water	0.2

3.6 Dose constrains and acceptable doses

Dose constrains will be set for individual doses for each of the years the decommissioning takes place. The dose constrain for effective dose will be in the interval 5-10 mSv pr. year depending on the operations planned for the particular year. For all years the equivalent dose to the eyes will be constrained to 15 mSv/year and the equivalent dose to skin and to extremities will be constrained to 150 mSv/year.

Due to the inherent associated large uncertainties in determining internal doses, planning should strive to entirely avoiding internal exposure with tritium as an exception.

Radiation doses below the dose constrains should be as low as reasonably achievable. To assist the more detailed planning in controlling exposure for individual operations a guideline on “acceptable doses” has been made. Table 3 gives values of acceptable doses for operations with different benefit and frequency.

Table 3. Guideline for "acceptable doses" from working operations.

Benefit from operation and frequency of operation (pr. year)	"Acceptable" effective dose	"Acceptable" equivalent dose to eyes	"Acceptable" equivalent dose to skin and extremities
Small benefit, high frequency	up to 10-20 µSv	up to 30-60 µSv	up to 300-600 µSv
Medium benefit, high frequency	up to 50 µSv	up to 150 µSv	Up to 1500 µSv
Large benefit, low frequency	up to 100-200 µSv	up to 300-600 µSv	Up to 3000-6000 µSv
Very large benefit, A few operations	Up to 500 µSv	up to 1500 µSv	up to 15000 µSv
Very large benefit, one operation	Up to 2000 µSv	up to 6000 µSv	up to 60000 µSv

3.7 Clearance

An infrastructure for clearance called the Clearance Function has been established within the Section of Radiation and Nuclear Safety. This Function comprises laboratory facilities, measuring equipment, measuring procedures, waste handling software, software for clearance related calculations and trained personnel (Søgaard-Hansen et al 2008). The Clearance Function was granted an accreditation in 2007 from the accreditation body, DANAK, according to the standard ISO/IEC 17025 (ISO 2005). The accreditation process was completed over a period of approximately one year. Many of the items from the peripheral systems have been classified as candidates for clearance measurements and have subsequently been cleared; in contrast all materials within the reactor block are to be considered as radioactive waste.

4. Conclusions

The Section of Radiation and Nuclear Safety at DD which has been in existence since the days of reactor commissioning will be responsible for the radiation protection guidance during the decommissioning project. Procedures used during reactor operation such as participation of health physics personnel in planning and execution of operations which involve high dose rates and/or risk of contamination and health physics classification of areas will be continued during the decommissioning phase.

From the assessed activity contents the radiation exposure during the various operations will be calculated and used in the planning of the operations, with adequate considerations taken to allow for large uncertainties on the activity estimates. Calculations will, when possible, be checked with measurements.

A guideline for acceptable doses has been communicated to the planning group to help the detailed planning in ensuring that the dose constrains are respected.

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Alara aspects in dismantling of the irradiated fuel reprocessing pilot Plant MTR Type (M1 Plant)

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Abstract
The irradiated fuel reprocessing plant MTR type was designed to treat fuel elements plate type from reactor JEN-1 (Ciemat). This facility (M-1 Plant) also served to develop fuel reprocessing processes.

Dismantling Works, which have been conducted by ENRESA, were carried out by MONCASA-LAINSA (MONLAIN) , the most experienced Spanish joint venture in dismantling nuclear facilities.

The dismantling of M-1 plant was performed from 2007 until the beginning of 2008.

This paper presents the performed dismantling works of the three M-1 Plant main systems:

♦ Hot Cell M-1.
♦ Glove Boxes Assembly L-1.
♦ Radioactive Liquid Waste Storage Cell (F-1).

The dismantling of these systems required methodical tasks planning, development of specific execution processes, execution of Alara studies and detailed safety plans.

Development of specific tools for dismantling tasks, containment auxiliary systems design, namely ventilated tents with local extraction systems and HEPA filters, operation assembly for equipment and components' extraction represent ones of the examples presented in this work.

A multidisciplinary team formed by the two companies joint venture, working together on an equipment designed by ENRESA, has carried out a unique work in Spain, executed with efficiency, optimizing resources and wastes, and minimizing the dose in accordance with the Alara criteria.

Introduction
The pilot plant for reprocessing of irradiated fuel type MTR (Plant M-1) is located in the eastern part of CIEMAT, in Building 18. It was designed to treat plaque-type fuel elements of the JEN-1 reactor and for the development of fuel reprocessing processes.

After two campaigns of treatment of irradiated fuels the Plant was shut down.

The dismantling of the facilities of the Plant M-1 has been performed in the frame of the Integrated Plan for the Improvement of the Facilities of the Ciemat (PIMIC

The dismantling of these systems has needed careful planning of tasks, development of specific execution procedures, and performance of Alara studies and detailed Safety Plans.

Description of the plant

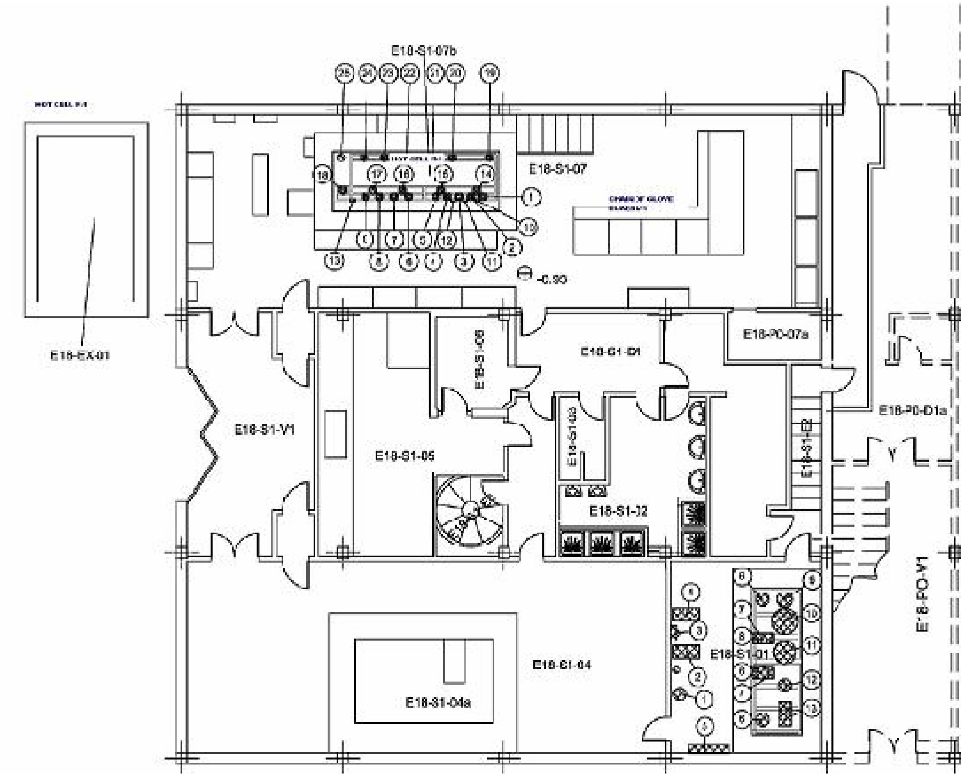

Fig. 1. Layout of the M1 Plant.

The pilot plant for reprocessing of irradiated fuel type MTR (Plant M-1) is a building of 352 m^2 and 6 m high, divided into three modules, with two floors in the central module.

The areas of interest for this paper are on the ground floor and northern area (Figure 1): Glove box chain L-1 and M1 Hot cell and outside the building 18, F-1 cell, Liquid Radioactive Waste Storage.

Glove Box Chain L-1

It contains the medium and low activity process equipment. The glove box chain L-1 has an "L" shape with seven separate modules. The second and the third are joined in a unique module.

The modules are divided in two parts. The low part of the second, third and sixth modules are not a part of the glove box and were used for the installation of auxiliary material. Each module is made of a metal profile structure to which panels of methyl-methacrylate (Plexiglas) of 8 mm of thickness are mate and constitute the side faces, as well as the separation between modules. The funds, roofs and separations between areas of modules are sheets of steel of 5 mm thick.

Cell M-1

Inside the cell the equipment for the first cycle of uranium and plutonium separation is found. Having dimensions of 1.60 x 4.80 x 2.40 m. The ceiling and walls of the cell M-1 are composed of reinforced concrete with a thickness of 65 cm. Part of the front wall (1.2 x 4 m) is shielded with three layers of lead bricks 50 mm thick. In the right side wall of the cell there is an opening that gives access to the interior with a free surface of 1.70 x 0.70 m, which is closed by a staggered shape door made of 60 cm thick barite concrete.

The M-1 cell contains diverse equipments of the first purifying fuel cycle, including tanks, mixers, settlers, valves and interconnecting tubing, mainly.

On the roof of the cell M-1 are found: the shielded glove box (sampling system of the cell deposits) and shielded cubicle C-1 (filtration system of the main process) and C-2 (volumetric pumps remote heads feeding radioactive solutions to process equipment).

Cell F-1

The F-1 cell for the reception and temporary storage of radioactive liquid waste is an underground cell of 2.6 x 4.5 x 4 m with reinforced 40 cm thick concrete walls. The roof is 20 cm above the street level and consists of 6 concrete slabs of 40 cm thick, which allows access to the interior.

The cell is divided internally into two areas, by means of barite concrete blocks (density 3.3 g / cm ^{3}) with a thickness of 50 cm, in order to separate the reception and storage of high and intermediate liquid effluents activity to the reception and storage of those of low activity.

The storage area of high and intermediate liquid effluents activity contains the tank A-1 (high activity), A-2 and A-3 (intermediate activity). The area of low activity includes tanks B-1 and B-2.

Dismantling Works

Planning for decommissioning has been carried out with reference to data provided ENRESA in the technical specification. The work was divided into units of intervention and for each of them a dismantling card was prepared to collect data related to the radiation risks, radioactive inventory, list of materials depending on their typology, estimate of waste, radiation protection and safety, equipment, components and systems, etc.

The first stage consisted of preparing the auxiliary facilities, in particular containment enclosures, called SAS (Figure 2). Due to the risk of alpha contamination the need to use a double barrier of containment was considered. This SAS consisted of a double enclosure, internal and external SAS, with an exhaust system that ensured an air flow towards the interior and that was complemented by a HEPA filtering system.

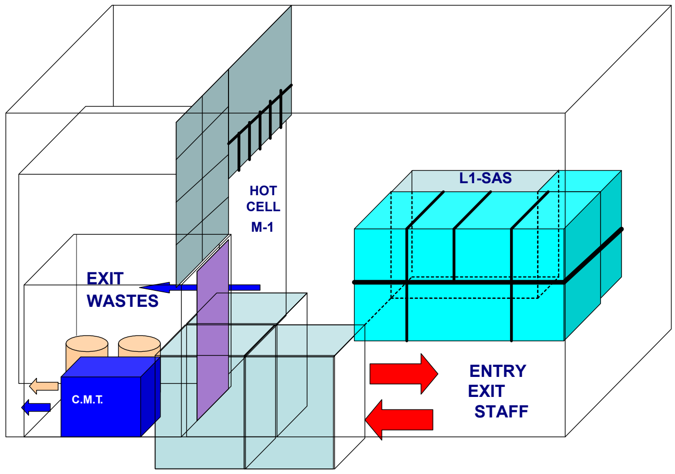

Fig. 2. SAS of Glove Boxes L-1.

The internal SAS had rectangular form and covered all the glove boxes L1 for both sides. It consisted of a metallic structure covered by methacrylate in the lower and not protected areas.

The external SAS covered the half of the room where there were the glove boxes L1 and the entry of the M1 cell. It was placed, partly, on the cell M1 and the rest in the corridor between M1 cell and the south wall. It had a metallic structure and it was covered in most cases by methacrylate. It is necessary to clarify that in the low part a door of methacrylate was enabled to allow the access to the waste management area.

That area had two methacrylate doors: one of them, in the East side, allowed the entrance of the wastes and the other was in the West side where the waste left. A rotary alarm with a door blocking system was placed to avoid the two doors were opened at the same time.

The system of assisted breathable air for the pressurized suits of the staff was located at the entrance area from inside the room, in the SAS of staff.

Dismantling of glove boxes L-1.

First, ancillary equipment (mixer-settler) located at the bottom of the modules C2, C3 and C6 was removed. The retrieval of these elements to the metallic containers was done by double bagging.

Next, sampling systems, glass columns (filled with silica gel), elements of ventilation and separate items mounted inside the box in the upper level of the modules C2, C3, C4, C5 and C6 were dismantled.

The method used to extract these systems from the inside of the glove boxes was the bagging.

Then dismantling of ducts, capping and closure of the holes, and connections with the lower level boxes C2, C3 and C6, was done.

Once emptied out the items with the previous method, and sealing the connections pipes, we proceeded to clean the inside (mopped) prior to the implementation of fixing paint inside the boxes.

Then we proceeded to the dismantling of the conduits located on the roof of the glove boxes L1, using cutting techniques such as confined cutting or sealed cutting with polyurethane foam.

Once the boxes were painted, the fastening pieces and methacrylate sheets were removed and cut to adapt them to the size of the metallic container.

Tanks on the lower level of each of the glove boxes were cleared.

Funds, ceilings, separating plates of 5 mm thick steel, and metal profiles were dismantled by cold cutting.

Hot Cell M-1

Once the Glove Box L-1 dismantled, it was necessary to restructure the SAS (see figure 3), to adapt it for the dismantling of the Cell M1.

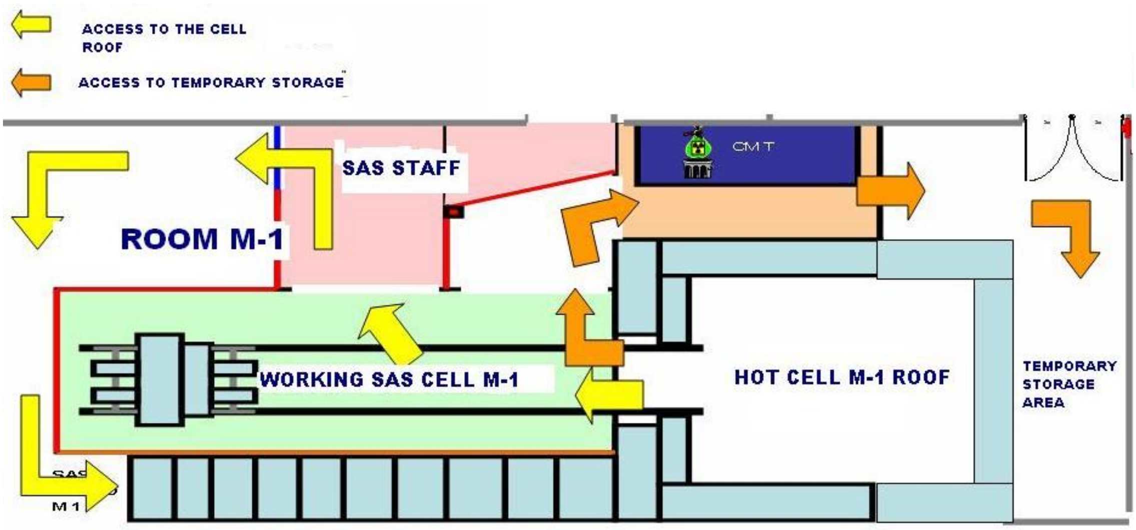

Fig. 3. SAS of Hot Cell M-1.

Given that the there was little room inside the Cell and the expected dose rates in certain systems, it was necessary to plan the works in detail attending on the ALARA principle.

An inventory of the equipments and systems of the interior of the Cell was performed, and a few cards reflecting the relevant information of every component (radiological characteristics, situational scheme inside the cell) were prepared.

A few auxiliary systems were prepared:

- Vacuum of particles.
- Decontamination.
- Collection of liquids
- Plasticized and bagging.
- Shielding.
- Mechanical manipulation of loads.
- Supply of breathable air.

The main methods of work used were:

- First vacuum. Adaptable canes of aspiration to the geometry of the environment of work. Periodic warnings in the filters of the system of collection of dust.
- Cutting and dismantling of pipes. Hydraulic shears that allows a rapid and clean cut. It cuts carbon steel as much as stainless steel.
- Dismounting of tanks, components and shieldings. Manual tools, nut-cutter, loosen bolts.
- Retreat of pipes. Double bagging and transport to a container (CMT).
- Retreat of tanks. Manual or with hoist according to the weight. Previous retreat of liquid from the inside of tanks.
- Cutting of work surfaces and supports. Hydraulic shears.

Cell F-1

The dismantling of the underground Cell F-1 had to be planned meticulously due to the risk of radiation and contamination, in addition to the little available room.

As in case of the Cell M1 it was necessary to build a specific ventilated SAS, but with an auxiliary installation that would allow removing the tanks (figure 4).

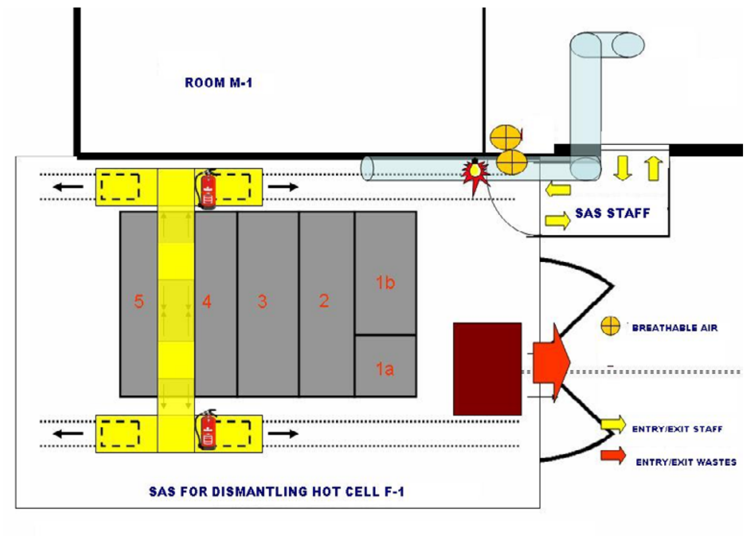

Fig. 4 . SAS for dismantling Hot Cell F-1.

This installation consisted of a bridge crane with a rail that allowed the movement across the SAS. The design of the SAS and of the bridge crane were thought to extract all the elements of the Cell, included the tank of discharged waste, which complexity was made clear along the works.

The sequence of the main activities was:

A. Opening and movement of concrete slabs.

B. **Dismantling of the Low Activity Tanks**

- Dismantling of valves and pumps.
- Cleaning of the outer tank walls.

- Cutting of pipes and supports of the tanks.
- Application of fixed paint on accessible surfaces of the tank.
- Hoisted of the tanks.
- Application of fixed paint in the rest of tank surfaces.
- Partial cutting of tanks inside the cell and the rest of cuttings outside. With vacuum hood system.
- Vacuum and cleaning inside the cell.

C. **Dismantling of Intermediate Activity Tanks**

- Extraction of side wall of concrete blocks adjacent to High Activity tanks.
- Dismantling of the roof blocks and cutting of feeding pipes of the tanks.
- Sealed-cutting of drainage pipes.
- Hoisted tank. Application of the first layer of fixation.
- Application of the second layer of fixation. Plasticized.
- Hoisted and laying in CMT of the tanks.

D. **Dismantling of the High activity tank**

- Extraction and dismantling of concrete blocks and cutting of feeding pipes.
- Opening a hollow and refilled with a layer of concrete the bottom of the tank.
- Fastening and hoisting the tank a little. Application of the first layer of fixation.
- Application of the second layer of fixation and covering with plastic.
- Hoisted of the tank and laying in shuttle.
- Introduction in spécial container CE2A.

Results

Next there the most important results, related to the work load, the dose and the management of materials are presented.

Work load

A whole of 4.427 hours x person has been used to achieve the three activities described previously. That indicates the difficulty of the above mentioned works. The percentage distribution of the hours is shown in figure 6. The percentages are very similar in the Cell M1 and in the F-1 and there do not move away too much of those of the glove boxes. Although the radiation risks glove boxes were very lower than those of other systems, certain time was used in the preparatory works, principally in the assembly of the SAS. In case of the Cell M1, the construction of the SAS did not take so much time, since the principal part, the separation in two halves of the Room M1 was already done in the previous phase.

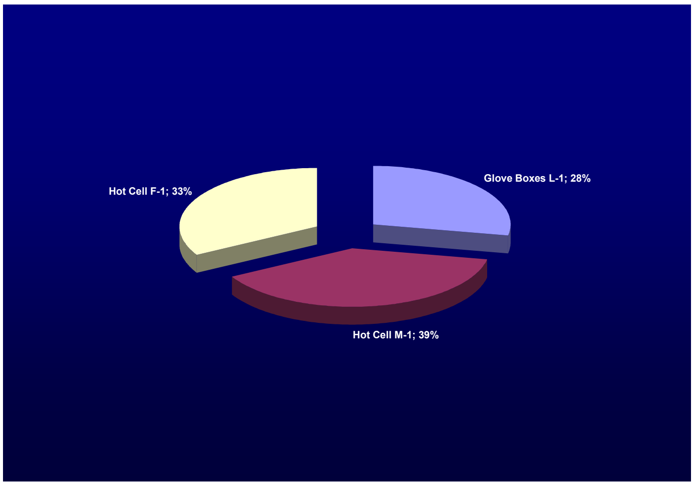

Fig. 5 . Work Load Breakdown.

Management of materials

Table 1. Balance of materials.

	CLEARENCE Materials	RADIAOCTIVE WASTES
Glove Boxes L-1	60 %	15 %
Hot Cell M-1	39%	40 %
Hot Cell F-1	1 %	45 %
Total	100 %	100 %

The whole of treated materials has been 62 Tm approximately. Table 1 shows the percentage distribution between the glove boxes and the cells for the potentially clearance materials and the radioactive wastes. 17 per cent of the materials have separated as clearance and therefore they have had to surrender to a process of measurement in the box counter, to perform a detailed radiological characterization and to decide his final destination. 83 % remaining has been considered to be radioactive residues, which is coherent with the type of elements on those who have been worked.

As it was expected the percentage of clearance materials for L-1 compared to those the Cells has been very higher. The biggest percentage of residues has been of the Cell F-1, due to the weight of the tanks and to that it has not been possible to segregate any more clearance components, due to the high dose rate inside the Cell.

Dosimetry

The activities with important collective dose have been the ones related to the Cell M-1 and the Cell F-1, since in case of the glove boxes L-1, the biggest risk was that of contamination. In this sense, there has been nor external contamination of the personnel neither internal contamination due to the applied confinement and protection measures. Photograph 4 shows the exit of personnel of the SAS of the Cell M1.

The biggest percentage of the collective dose has been allocated to the dismantling of the Cell F-1, as it was expected, since the levels of radiation were high.

Next the Cell M-1, since although the area dose rate was not too high, nevertheless it was certain equipments having points with high values.

The total collective dose in the Cell M1 has been 6 mSv x person approximately. This dose is 3 times lower than what had been foreseen in the Alara studies. We can find the cause of this deviation in the reduction of the time of execution, for what the exposition has diminished considerably. The detailed planning previous to the works has been a key point in this sense.

In case of the Cell F-1, the total collective dose has been 10 mSv x person. It represents approximately twice less than the expected one. In this case the attenuation of the term source, by means of shieldings and the meticulous planning of the works have been the main reasons.

Conclusions

The intervention of a multidisciplinary team of two constituent companies of the JV, collaborating with the team designated by ENRESA, has reached very good results in a work in Spain considered as unique. This work has been performed in an efficient and effective form, optimizing resources and residues and minimizing the doses in accordance with the Alara principle.

This work had not been possible without the fundamental support of the CIEMAT in the execution of the operations.

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Protocol for the clearance and release for metal materials from SLAC National Accelerator Laboratory – Application to BaBar Detector Dismantling

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Abstract

In 2000, the DOE announced a suspension on unrestricted releases of any metals that have been inside a radiological area. SLAC is an electron accelerator facility which has many valuable metal components for off-site reuse or recycling. BaBar is a particle physics detector currently being dismantling for that purpose. Based on the DOE relief process, a protocol for unrestrictive release of non-radioactive metals is developed for BaBar dismantling.

At electron accelerators, surface contamination is very unlikely and volume activation is the main interest. The BaBar protocol can be generalized for other applications at SLAC or other electron accelerators. The SLAC protocol consists of an evaluation to determine the potential radioactivity of metal components based on process knowledge, calculations and measurements, as well as a technical basis for unrestrictive release.

Facility operations (beam or non-beam related) that can potentially contaminate or activate property are evaluated as follows: 1) obtaining process knowledge such as BaBar historical operational models and beam losses, 2) radioactivity calculations using the FLUKA Monte Carlo code, which allows a zoning approach to classify the components as radioactive or not for the purposes of planning, radiological control, and measurements, and 3) gross beta-gamma field survey on all surfaces of every item. Gamma spectroscopy measurements are conducted on pre-determined critical items to compare with FLUKA results.

The technical basis for the protocol consists of determining the detection limits of the measurement methods. Survey procedures and release criteria are developed based on the DOE Order 5400.5, MARSSIM, and ANSI standard N13.12. The SLAC instrument's detection limit is 10 times lower than the N13.12 clearance level for volumetrically activated materials based on the dose of 1 mrem/yr. Only items that have non-detectable radioactivity from both calculations and measurements will be released.

Development of radiochemical analytical method for the determination of radionuclides difficult to measure for decommissioning of nuclear facilities

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Abstract

The decommissioning of a nuclear facility requires estimating the total inventory of radioactivity in various materials and its variation with time, which has to be carried out by the determination of the radioactivity of various radionuclides presented in the materials. The neutron activation products of components and impurity in the materials, such as ^{3}H, ^{14}C, ^{36}Cl, ^{41}Ca, ^{60}Co, ^{55}Fe, ^{63}Ni, ^{133}Ba, ^{152}Eu, ^{154}Eu, and some transuranics, are the main contributors to the total radioactivity. But some fission products, such as ^{90}Sr, ^{99}Tc, ^{129}I, and ^{137}Cs, may also exist in the materials due to the contamination of the leaked nuclear fuel. Of these radionuclides, the beta and alpha emitters including ^{3}H, ^{14}C, ^{36}Cl, ^{41}Ca, ^{55}Fe, ^{63}Ni, and some transuranics, have to be determined by radiochemical analysis including a completely separation of individual radionuclides from matrix and other radionuclides before measurement by beta counting, alpha spectrometry or mass spectrometry. This work presents various radiochemical analytical methods developed in the authors laboratory in the recent years for the determination of these radionuclides in various materials for the decommissioning of nuclear facilities, which includes:

1. rapid determination of tritium and ^{14}C in solid materials, such as graphite, concrete, steel, aluminium, paint, silica gel, soil, and dust;
2. determination of ^{14}C in high tritium samples, such as heavy water, waste water, and oil;
3. determination of ^{36}Cl and ^{129}I in graphite, steel, concrete, waste water, and dust;
4. determination of ^{41}Ca in concrete;
5. determination of ^{55}Fe and ^{63}Ni in graphite, concrete, steel, aluminium, sediment, sand, waste water, seawater, and lichens.

Meaning of site-specific data in dose assessment: case of concentration ratios in boreal forest

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Abstract

Posiva is implementing a deep repository for spent nuclear fuel in Olkiluoto, Finland. A site-specific safety case is being produced, and data has been acquired for the assessment models. However, neither site or literature data alone are sufficient for the safety case; a balanced combination is needed. The case of the concentration ratios from soil to plants in boreal forest, dominant ecosystem at the site, is discussed in this contribution.

Combining literature data with site data is not straightforward. Our approach is based on updating probability density functions (pdf). First, an *a priori* distribution is established from site-applicable literature data to contribute to the width of the distribution. Then, the pdf is updated with individual, site-relevant literature data taking into account the number of samples as a weight. Finally, the distribution is updated with the site data. In the weighting, also additional weighting factors of confidence or relevance for which quantitative measures have been established can be used. Repeating the procedure for different groupings of data, e.g. plant types, reveals whether the grouping is statistically reasonable.

The meaning of the site data is pronounced where the site data alone is not sufficient and where the literature data would have not been adequate: few site data do not necessarily reveal the entire width of the distribution, and the literature data may not span to the extremes in the site data.

Furthermore, the national forest soil inventory methodology applied required development of a concept to calculate concentration ratios from multi-layer soil data where the rooting depth is taken into account, as opposed to the conventional method of average concentration in soil core to a fixed depth. Examples of concentration ratios specific to soil layers are presented.

Introduction

Posiva is implementing a deep repository for spent nuclear fuel in Olkiluoto, Finland. A site-specific safety case is being produced to support a construction license application in 2012 (Posiva 2008), and data has been acquired for the assessment models (Hjerpe et al. 2010). In the assessment, a large amount of data is required of which only part can be produced in the site investigations with reasonable resources. Furthermore, there has been discussion on whether the site data alone would be sufficient anyway since it may not capture the entire variability relevant to the assessment (e.g. Sheppard 2005), the less the fewer the site data is. In this paper, methods to derive concentration ratios from site data and to combine site and literature data are outlined and demonstrated, and the need of site-specific and generic data is discussed based on some examples.

In Posiva's repository programme, ecosystem characterisation is an iterative process aiming to achieve an adequate site understanding, in order to evaluate the appropriateness of different models and of preliminary data to the site, and subsequently to provide data of sufficient scope and quality to underpin the safety case development (Haapanen et al. 2009). Ideally, a totally exhaustive characterisation of the properties and processes of the ecosystem could be taken as the aim. However, with any limited resources that can never be achieved, neither is it necessary to achieve to present a sufficient safety case; the extent of the ecosystem characterisation efforts needs to be in reasonable relation to the overall repository programme, the significance to the safety of the spent fuel disposal and the regulatory requirements. In practice, this means continuous improvement at a moderate level (i.e., reasonable in the context of overall repository programme) in ecosystem characterisation with a focus on issues that have significant safety relevance. Identification of key issues is not a straightforward task, but an iterative process, preferably done with feedback from the regulator and other stakeholders. During the initial overall characterisation little data has been acquired in some key nuclides and their chemical analogues – the programme is currently shifting focus from the general characterisation to these assessment-driven key issues. In the interim, information also from other programmes of similar sites or literature needs to be utilised.

Material and methods

Multi-layer approach to concentration ratios

In Posiva's biosphere assessment BSA-2009 (Hjerpe et al. 2010), the terrestrial ecosystems are represented by a compartment model shown in Figure 1. The model has been reformulated from earlier ecosystem-specific compartment models to correspond the features at the site as recognised in the biosphere description process (Haapanen et al. 2009). Following from its specific function and on the other hand its varying thickness (about 2-20 cm in the rather young sites at Olkiluoto Island, emerged from sea by post-glacial uplift only during few past millennia), the humus layer has been separated from the underlying mineral soil layers in forest objects. The compartment structure is also consistent with the available information from the various site studies, where the humus layer, the mineral soil layer of 0...30-cm depth and the subsoil are sampled separately to follow more general sampling schemes (e.g. the national forest soil inventory), and in the case of subsoil also in separate campaigns with lower spatial resolution due to the need of machinery and the disturbance to the site studied.

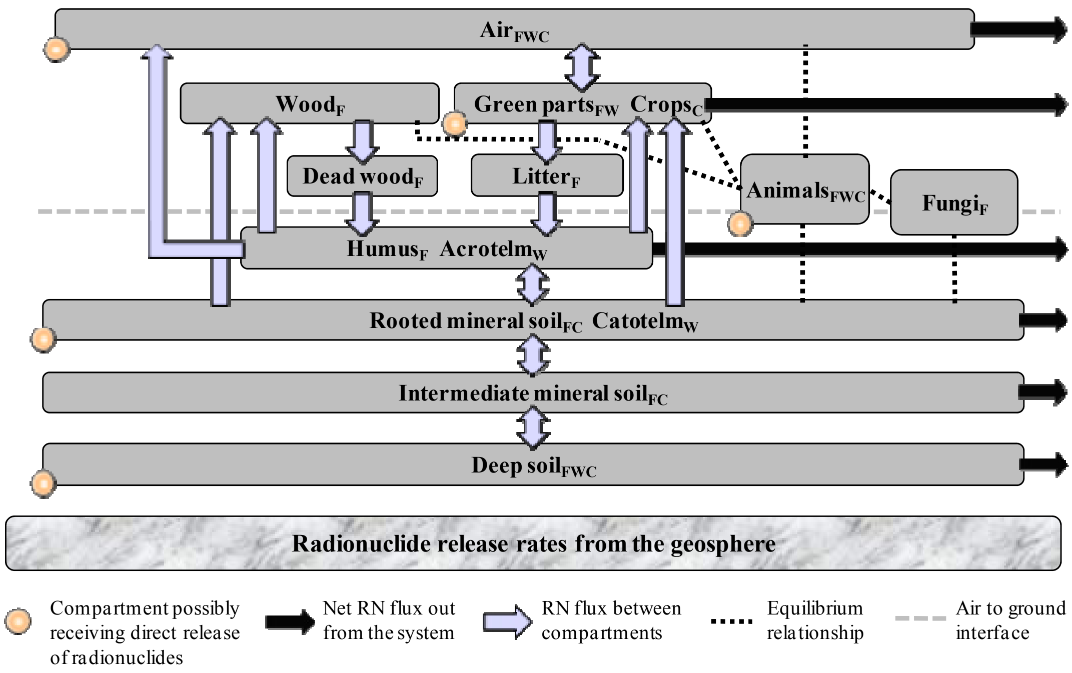

Figure 1. The conceptual radionuclide transport model for terrestrial ecosystems (Hjerpe et al. 2010). The indices in the compartment names define for which ecosystem(s) they are valid, where: F denotes forest, W wetland and C cropland.

Following the compartment division described above, concentration ratios are calculated for the transport from the humus layer and from the rooted mineral soil using the fine root biomass distribution (Table 1) as a weighting factor (Haapanen et al. 2009); it is assumed that the uptake from the compartments hosting the roots is proportional to the amount of roots in them:

$$f_i C_j = CR_{i,j} C_i$$

$$C_j = \Sigma_i\, CR_{i,j} C_i = CR_{eff} C_s$$

where f_i is the proportion of fine root biomass in soil layer i, C_j is the concentration in the recipient compartment (e.g. wood or foliage), $CR_{i,j}$ is the soil layer-specific concentration ratio from soil layer i to compartment j, and C_i is the radionuclide concentration in soil layer i. To calculate the concentration in the recipient compartment using the conventional concentration ratio (CR_{eff}), the corresponding average concentration in the soil layers relevant to the root uptake (C_s) shall be used:

$$CR_{eff} = C_j \,/\, [\, (\Sigma d_i \rho_i C_i)/(\Sigma d_i \rho_i)\,]$$

where d_i is thickness of a considered soil layer and ρ_i the respective bulk density of soil.

Certainly using the layer-specific concentration ratios is not compatible to the general definition of the concentration ratios, assuming a topsoil sample regardless of the soil layers, most often 30 cm. Thus for comparisons or combination with literature data, CR_{eff} needs to be estimated and used.

Table 1. Distribution of fine root biomass (mass-%) in humus layer and in mineral soil layer of 0-30 cm at Olkiluoto (Helmisaari et al. 2009).

Stand type	Plant group	Humus	Min. soil	Application in assessment
Pine	Trees	38	62	Pine trees
Pine	Shrubs	85	15	Shrubs in coniferous stands
Pine	Grasses	89	11	Grasses and herbs in pine stands
Spruce	Trees	59	41	Spruces and coniferous bushes
Spruce	Grasses	99	1	Grasses and herbs in spruce stands
Birch	Tree seedlings	71	29	Deciduous trees and bushes
Birch	Shrubs	63	37	Shrubs in deciduous stands
Birch	Grasses	76	24	Grasses and herbs in deciduous stands

Combination of various data sources using Bayesian updating

A Bayesian approach has been developed by Kristofer Stenberg (Facilia AB) and adopted by Posiva (Helin et al. 2010) to obtain combined probability density functions (pdf) from site measurements and literature data. The goal is to obtain pdf's that take into account the more generic data when the site data is judged insufficient and also the variability between the studies included.

As a ratio quantity, concentration ratios are expected (and thus assumed) to be lognormally distributed by the central limit theorem. Most data compilations that have studied the form of the distribution do not represent arguments against the assumption, although some other possible distributions have been derived from the data.

For small number of observations there can be large uncertainty in the estimates of mean and variance. Bayesian updating makes it possible to reduce this uncertainty by including further knowledge (prior knowledge) in the estimation: a prior distribution of the two unknown parameters is assumed and the posterior distribution, or the probability of the parameters conditioned on observed data, is given by Bayes' theorem. There are different methods of estimating the posterior distribution depending on the interpretation of the prior knowledge and thereby on the form of the prior specification. In this paper we demonstrate only combining the site and literature data with *direct updating* where they are considered equally appropriate to the site: the posterior is a combination of the observations *a priori* and the new observations, weighted with the sample sizes. This method utilises the concept of full conjugacy of the prior distribution (Gelman et al. 2003). This and the other methods, population and hierarchical updating, are described in more detail in (Helin et al. 2010).

Results

Soil-layer-specific concentration ratios from the Olkiluoto site

Table 2 lists results on concentration ratios of nickel and iodine to forest plants, derived from the data from three sampling sites at Olkiluoto in 2008: FIP4, FIP10 and

FEH914254 (Haapanen 2009). FIP (forest intensive monitoring) plots have been established to continuously follow changes taking place in the nutrient budgets and fluxes in the soil, tree stands and vegetation at both the stand and the catchment level. FIP4 is a Scots pine-dominated advanced growing stand and FIP10 is a mature Norway spruce-dominated stand. Both FIP plots are located in herb-rich heath forests. FEH plots are used for inventories of vegetation (composition, biomass, nutrient concentrations) and soil (soil profile description, physical and chemical properties) at 5–10 years intervals, in addition to regular tree stand measurements. FEH914254 is a black alder-dominated stand growing on herb-rich heath forest site.

As an example of a larger dataset, for comparison, respective concentration ratios of nickel are presented in Table 3 as derived from a forest inventory at Olkiluoto in 2005 (Tamminen et al. 2007). Further details of the calculations can be found in (Ikonen et al. 2010).

Combinations of site and literature data

To illustrate the combination of datasets using the Bayesian direct updating described above, datasets on concentration ratios of iodine to plants are used (the choice was limited by availability of literature data comparable to the site data). The site data on effective concentration ratios to understorey plants (excluding mosses) presented in Table 2 and Fig. 1 was combined with concentration ratios to generic plants derived by Sheppard et al. (2006). The latter data was converted to the dry-weight basis by assuming a dry matter content of 0.50 kg_{dw}/kg_{fw}. The combined data, still dominated by the literature data (N=80 vs. N=7 of the site data), exhibits a slightly higher geometric mean following from the update with the site data having a higher geometric mean than the literature data. The geometric standard deviation has also increased in the combination due to the difference of the datasets; in the combined one the variability of the both original datasets has been included.

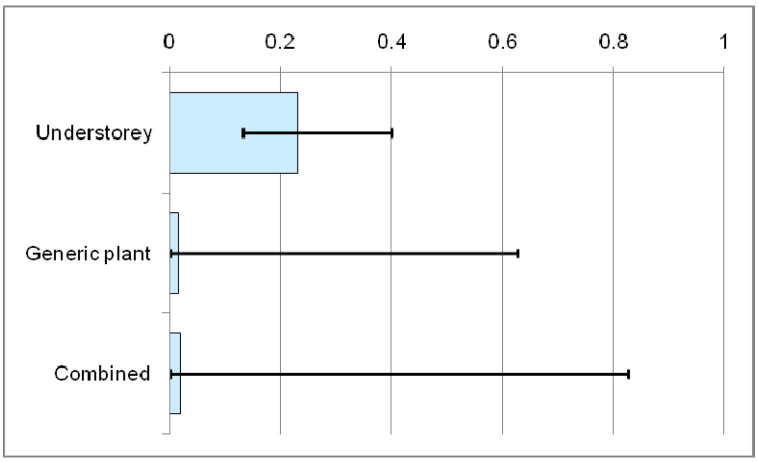

Fig. 1. Distributions of concentration ratios of iodine ([kg/kg$_{dw}$]/[kg/kg$_{dw}$]): site-specific data for understorey (CR_{eff}, Table 2), data for generic plants (Sheppard et al. 2006) and their combination derived by Bayesian direct updating. Geometric means are presented by end of the thick bars and 5th and 95th percentiles as the lines.

Table 2. Soil-layer-specific and conventional, or effective, concentration ratios ([kg/kg$_{dw}$]/[kg/kg$_{dw}$]) of nickel and iodine to forest plants at Olkiluoto in 2008, geometric mean (geometric standard deviation; number of sample pairs). The humus layer (*h*) has been taken as the actual at the sampling sites, and the mineral soil layer (*min*) includes the top 30 cm.

	CR$_h$, Ni	CR$_{min}$, Ni	CR$_{eff}$, Ni	CR$_h$, I	CR$_{min}$, I	CR$_{eff}$, I
Tree foliage	0.11 (2.9; 3)	0.15 (2.6; 3)	0.31 (3.1; 3)	0.05 (-; 1)	0.14 (-; 1)	0.29 (-; 1)
Grasses & herbs	0.17 (2.8; 5)	0.03 (6.7; 5)	0.35 (1.8; 5)	0.09 (1.7; 5)	0.02 (5.4; 5)	0.28 (1.2; 5)
Bilberry leaves*	0.14 (1.9; 2)	0.03 (1.5; 2)	0.19 (1.3; 2)	0.07 (-; 2)	0.03 (-; 2)	0.16 (-; 2)
Understorey **	0.15 (2.4; 8)	0.02 (4.5; 8)	0.24 (2.2; 8)	0.08 (1.7; 7)	0.03 (4.0; 7)	0.23 (1.4; 7)

* *Vaccinium myrtillus*
** All concentration ratios to understorey plants (except mosses) pooled

Table 3. Soil-layer-specific and conventional, or effective, concentration ratios ([kg/kg$_{dw}$]/[kg/kg$_{dw}$]) of nickel to forest plants at Olkiluoto in 2005, geometric mean (geometric standard deviation; number of sample pairs). The humus layer (*h*) has been taken as the actual at the sampling sites, and the mineral soil layer (*min*) includes the top 30 cm.

	CR$_h$, Ni	CR$_{min}$, Ni	CR$_{eff}$, Ni
Tree foliage	0.07 (1.7; 90)	15 (7.3; 80)	3.4 (4.6; 90)
Grasses & herbs	0.15 (1.7; 125)	3.6 (11; 122)	5.8 (4.3; 125)
Bilberry leaves*	0.07 (1.4; 58)	6.3 (4.9; 54)	4.2 (2.5; 58)
Understorey **	0.10 (2.0; 240)	3.8 (8.3; 223)	4.5 (3.8; 240)

* *Vaccinium myrtillus*
** All concentration ratios to understorey plants (except mosses) pooled

Table 4. Concentration ratios of iodine to understorey plants in Olkiluoto (Table 2), to generic plant type (Sheppard et al. 2006) and their combination by Bayesian direct updating ([kg/kg$_{dw}$]/[kg/kg$_{dw}$]). GM denotes the geometric mean, GSD the geometric standard deviation and N the number of data.

	GM	GSD	N
Understorey	0.23	1.4	7
Generic plant	0.016	9.3	80
Combined	0.02	9.6	87

Discussion

If we consider, for example, the transport parameters, like Sheppard (2005), it is indeed nearly impossible to know whether there is an unexpected bias when there are only few data (from the site or the literature), and also to exemplify whether a species in a group is different from another, for example, is bilberry bioconcentrating more effectively than lingonberry? In any case, the more data available, the more we can know about the underlying processes and relevant differences in conditions. Since the literature data on these parameters are notoriously scarce (with compendium values usually lacking description of the range of conditions the data are derived from), the contribution of even a few site data would be useful. Still, a few site data alone could be insufficient since the inherent variability of the transport parameters is so large that site data would be significantly different only for "fairly exceptional on-site properties" (Sheppard 2005). Demonstration of such cases turned out to be more difficult in case of forest plants than foreseen; the literature data accessible with moderate effort is rather plenty in caesium data but very scarce for other nuclides, specifically of those most of interest of Posiva's assessment (Cl-36 and I-129, dominating the dose together with C-14 in the dose assessment time window of 10 000 years (YVL 8.4, YVL E-5)).

By comparing the data in Tables 2 and 3 it is notable, that when all forest types in an island of 12 km² are included in the data, the geometric standard deviation (GSD) increases significantly for the concentration ratios from the mineral soil and for the effective concentration ratios – this is the opposite to the expected situation with the humus-specific concentration ratios where the GSD decreases with more data. This is likely due to the surrogate data used to the mineral soil layers, practically assuming a default value for each soil type (Ikonen et al. 2010). In the concentration ratios specific to the humus layer this does not affect, and it can be concluded that for foliage and herbs and grasses the GSD decreases to a half with samples increasing from 3-5 to 90-123. With bilberry leaves and understorey in general the change is smaller but exists anyway. In general, it appears that in the larger dataset (Table 3) the humus-specific concentration ratio the values are on the lower end of the 5th-95th percentile range represented by the fewer samples (Table 2) – species accumulating more nickel may have been collected into the latter by chance.

Conclusions

In the repository programme the ecosystem characterisation provides with data to and supports model development in the dose assessment, and this is an iterative process where focus should be on the parameters which have the highest impact on the doses as only part of the needed input data can ever be acquired by site studies. To ensure that large enough variability is propagated through the assessment, it is necessary to try to identify any bias with help of literature data. However, as the emphasis of radioecology has been almost solely in the consequences of the Chernobyl accident, very little data is available on the radionuclides of interest in the deep geological repository – with the naturally occurring nuclides as an exception, which are on the other hand out of the dose assessment time window in the Finnish regulatory context as they may reach the biosphere only after the next glaciation when release constraints apply instead of dose limits. Furthermore, the few compendia data tend to be too general in the description, and the old original research papers are seldom available – it is difficult to judge whether a piece of literature data is appropriate to a specific site. All this emphasises the importance of extensive site studies, which should be targeted to cover the variability not only at the site itself but within the full assessment context.

To optimise use of resources, Posiva's approach has been to implement the ecosystem characterisation at the site by adopting methodologies used more widely. By this, large independent datasets become comparable. However, as the sampling and analytical protocols have not been tailored to the needs of the radionuclide transport modelling and dose assessment, some gaps remain. For example, in an extensive forest soil inventory, shovel pits are used instead of volumetric samples due to the laboriousness of the latter, and for the integrated benefits humus layer and mineral soil are sampled separately. Calculating traditional concentration ratios from the top 30 cm of soil to the plant needed thus bulk density data for calculating the average concentrations, but few of them are available from the sampling plots. To overcome these difficulties, a concept of multi-layer concentration ratios has been outlined above. With it more of the site data can be utilised, but on the other hand comparison to the traditional concentration ratios in the literature data requires surrogate to the soil density data. This is not seen as a major problem, though, since the literature data is relatively scarce for the most radionuclides of interest of a deep geological repository.

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A graded approach to dose assessment in the Posiva safety case

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Abstract

Posiva Oy is responsible for implementing a final disposal programme in Finland for spent nuclear fuel. The next step of the nuclear licensing is by the end of 2012 submit a construction licence application for a KBS-3 type of repository at the Olkiluoto site. Currently, a safety case is produced to support this application, were a preliminary version was completed in 2009. A three-tiered graded approach was developed and applied in the dose assessment. Tier 1 and 2 are generic screening evaluations, and Tier 3 is based on site-specific state-of-the-art radionuclide transport modelling and dose assessment. The main aim of the screening evaluation is to reduce the number of radionuclides needed to be considered in Tier 3, which is especially valuable for long-term assessments that are associated with large uncertainties. This approach assures that the level of detail of the assessment is appropriate to the magnitude of the potential radiological consequences, and strengthens the confidence in the assessment and the demonstration of compliance with regulatory criteria. Tier 1 derives defensible and extremely cautious Risk Quotients, assuming that all radionuclides released from the geosphere expose one person, or a penalizing species for other biota, without any dispersion in the biosphere. Tier 2 derives defensible and cautious Risk Quotients, based on a generic model including three sub-models, a terrestrial, an aquatic, and a well. The screening evaluation was applied to the geosphere releases derived in the interim safety case. The result shows that 24 of the 35 radionuclides in the geosphere releases can be screened out. Especially noticeable is that no radionuclides in the naturally occurring decay chains need to be propagated to Tier 3. This paper presents the methodology of the graded approach, the screening models, and the results from the screening evaluation in the preliminary safety case.

Introduction

Posiva Oy (Posiva) was established in 1995 by the two Finnish nuclear power companies, Teollisuuden Voima Oyj (TVO) and Fortum Power and Heat Oy (Fortum), to implement the final disposal programme for spent nuclear fuel and to carry out the related research, technical design and development. The spent nuclear fuel is planned to be disposed of in a KBS-3 type of repository to be constructed at a depth of about 400 metres in the crystalline bedrock at the Olkiluoto site. Posiva is currently preparing for the next step of the nuclear licensing of the repository, which involves submitting the construction licence application for a spent fuel repository by the end of 2012. A safety case will be produced to support this licence application; the Posiva plan for conducting the safety case is documented in (Posiva 2008).

A key contributor to the safety case is the biosphere assessment. The overall aims of the biosphere assessment are to describe the future, present, and relevant past conditions at, and prevailing processes in, the surface environment of the Olkiluoto site; to model the transport and fate of radionuclides potentially released from the repository through the geosphere to the surface environment; and to assess possible radiological consequences to humans and other biota.

The regulatory requirements for the long-term safety are set out in detail in the Radiation and Nuclear Safety Authority's (STUK) Guide YVL E.5 (STUK 2009) on disposal of nuclear waste. STUK (2009) states that "In any assessment period, during which the radiation exposure of humans can be assessed with sufficient reliability, and which shall extend at a minimum over several millennia: 1) the annual dose to the most exposed people shall remain below the value of 0.1 mSv, and 2) the average annual doses to other people shall remain insignificantly low". Furthermore, "disposal shall not affect detrimentally to species of flora and fauna, this shall be demonstrated by assessing typical radiation exposures of terrestrial and aquatic populations in the disposal site environments".

The main approach in the biosphere assessment is to develop a fully dynamic model for the development of the surface environments, radionuclide transport and radiological consequences analysis. In the dose assessment part of the biosphere assessment, the fate of radionuclides potentially released to the biosphere is assessed with radionuclide transport models and radiological consequences are assessed by deriving radiation doses to humans and other biota. For the dose assessment, a graded approach based on three tiers has been applied. Tier 1 and 2 involve conducting generic evaluations to screen out radionuclides that have insignificant radiological consequences, using two levels of inherent pessimism, and Tier 3 is based on based on site-specific state-of-the-art radionuclide transport modelling and dose assessment. The graded approach to dose assessment has been developed for, and applied in, the recently produced interim safety case (Hjerpe et al. 2010, Posiva 2010). This paper presents the methodology used in the graded approach, the models underpinning the screening evaluation (Tier 1 and Tier 2) and results from the screening evaluation performed in the interim safety case.

Methodology

When conducting the biosphere assessment, two important goals are to ensure that the assessment is based on an appropriate level of understanding of the biosphere and its potential behaviour, and that the level of detail of the assessment is appropriate to the magnitude of the potential radiological consequences.

In the biosphere assessment performed for the interim safety case, a three-tiered graded approach is implemented to achieve the goals. The common denominator in tiered approaches is that the complexity and realism are greater for higher tiers compared with lower tiers. In the present approach, Tier 1 and Tier 2 are generic radionuclide screening evaluations, requiring a minimum of site-specific data. Tier 3, which uses the landscape model, is the most realistic. The approach is to use the results from Tiers 1 and 2 to identify radionuclides that are highly confidently expected to have insignificant radiological consequences, and do not need to be considered in Tier 3. This allows the computationally demanding landscape model used in Tier 3 to focus on key radionuclides. The approach thus avoids the need to obtain Tier 3 data, and evaluate uncertainties in these data, for radionuclides to which the overall assessment results are not sensitive. It also facilitates and strengthens confidence in the biosphere assessment and strengthens the demonstration of compliance with regulatory criteria, especially by increasing the transparency of the biosphere assessment. Furthermore, it provides an instrument for analysing model uncertainties, and providing guidance for the development of the landscape model, and the associated environmental monitoring programme. The landscape model is presented in Hjerpe et al. 2010 and the models used in the screening evaluation are described below.

The screening evaluation

This section describes the approach and the models used in the screening evaluation (Tier 1 and Tier 2). The quantity of interest in both tiers is the Risk Quotient (RQ), which is the calculated nuclide-specific dose rate divided by pre-selected Screening Dose Rates (SDR). The approach must be sufficiently cautious so that there is a high degree of confidence that the potential radiological consequences are below the relevant regulatory requirements when the RQ is below 1. The SDR itself must be assigned a value substantially below regulatory dose constraints; how much lower depends on the desired degree of conservatism in the screening evaluation. The approach here is to select the SDR low enough to allow Tier 1 and 2 to screen out individual radionuclides for which the calculated RQ is less than or equal to 1. Two SDRs are used throughout the evaluation: 10^{-5} mSv for humans, which is two orders of magnitudes below the lowest regulatory annual dose constraint, and 10 µGy/h for the other biota, which is the default generic screening absorbed dose rate in the ERICA Tier 1 (Beresford et al. 2007) and is also recommended by the PROTECT project (Andersson et al. 2008).

The procedure for the screening evaluation is similar to the one recommended in IAEA (2001) for use in assessing the impact of discharges of radioactive substances to the environment, and the models are in line with the recommendation by the ICRP (2000, 2007b) on how to conduct a dose assessment. In Tier 1, an extremely cautious approach is taken, in which it is assumed that a hypothetical individual receives the maximum exposure over one year to the whole integrated release from the geosphere. If the RQ calculated for a specific radionuclide in Tier 1 is greater than 1, then it is necessary to continue to Tier 2. In Tier 2, a screening model is applied that includes a higher degree of realism than the model used in Tier 1, but is still sufficiently cautious for screening purposes. The generic model includes two generic ecosystem-specific sub-models, one terrestrial and one aquatic, and a well sub-model. The screening model used in Tier 2 does not require site-specific parameters and therefore can be called "generic". If the RQ calculated for a specific radionuclide in Tier 2 is greater than 1, then it is necessary to consider that radionuclide in the site-specific landscape modelling (Tier 3).

Tier 1

The first tier is conceptually illustrated in Figure 1. It is designed to ensure extremely pessimistic RQs, by several orders of magnitude. Thus, radionuclides screened out at Tier 1 are indisputably insignificant for radiological consequences during the biosphere assessment time window. The evaluation regarding humans is carried out by assessing the doses due to each radionuclide in three exposure situations, where the whole integrated release from the geosphere is

- totally routed to one person for intake by ingestion,
- totally routed to one person for intake by inhalation, and
- transferred to the ground surface and exposes one person externally.

When assessing the dose due to external exposure, it is pessimistically assumed that the whole activity is transferred to one square meter of ground surface; then the external dose rate is, unrealistically, derived by multiplying the activity concentration in the contaminated surface by the dose coefficient for a source distributed over an infinite ground surface. Further, to derive the annual dose due to external exposure, it is assumed that the person is exposed to that dose rate over one year. The highest annual dose from the three exposure situations considered is then divided by the SDR, for humans, to obtain the RQ_{humans}. Thus, Tier 1 is an extension of the integrated radiotoxicity flux that may be used as an indicator of safety.

The evaluation, regarding other biota, is implemented in a somewhat different fashion. Instead of calculating absorbed dose rates to compare with the selected SDR for other biota to obtain the RQ_{biota}, the SDR is used to determine radionuclide-specific environmental media concentration limits (EMCL), based on the ERICA integrated approach (Beresford et al. 2007). The most penalising EMCL for different solid media (soil or sediment) and liquid media (freshwater and marine water) are denoted as $EMCL_{solid}$ and $EMCL_{liquid}$, respectively.

Furthermore, it is pessimistically assumed that the habitat for the worst case reference organism has an activity concentration numerically equal to the total integrated activity, for each radionuclide, in the geosphere release. For example, if the integrated activity is 100 Bq and the worst case reference organism habitat is lake water, an activity concentration of 100 Bq/L in the lake water is assumed. The RQ_{biota} is then determined as the highest ratio of the derived activity concentrations in solid media, or liquid media, and $EMCL_{solid}$, or $EMCL_{liquid}$.

In addition, to be even more certain that the RQs are not underestimated, the integrated geosphere release rates used are not corrected for any radioactive decay, but include build-up of progeny radionuclides. This is, of course, a non-physical approach. However, this is deliberate, so that there is no potential for underestimating the contribution either from the parent or from its progeny. The integration takes places over a time window of 15 000 years (from year 2 020 to year 17 020), which is 5 000 years longer than Posiva's interpretation of the biosphere assessment time window where regulatory dose constraints apply. Thus, this means that one person, or one individual of the limiting reference organism for other biota, is exposed to everything released from the geosphere up to the year 17 020.

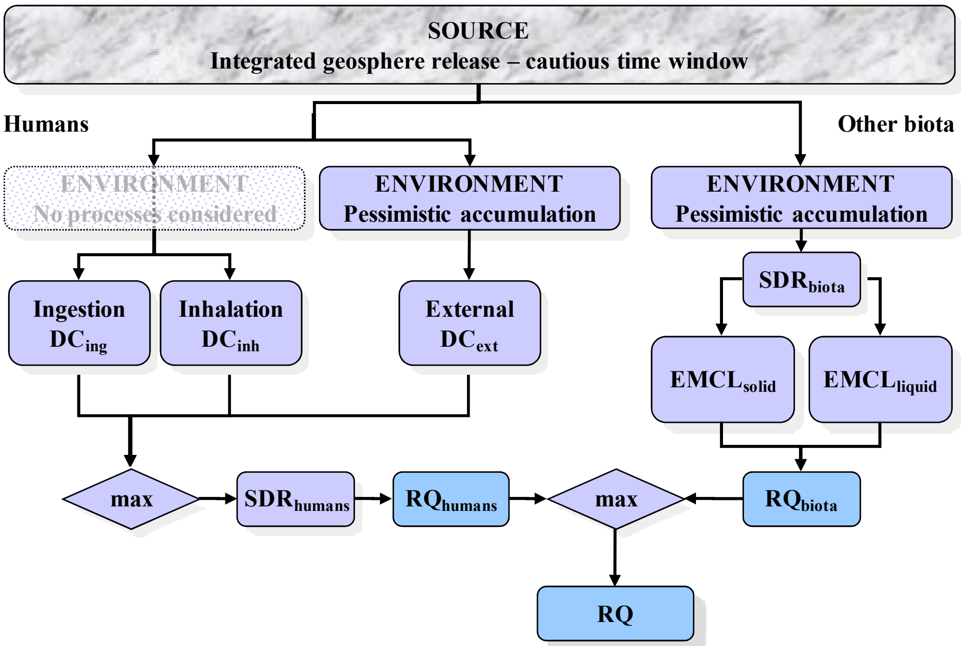

Fig. 1. Conceptual model of Tier 1. The environment box is shaded for the ingestion and inhalation pathways to emphasize that no environmental processes are considered – the source is directly inhaled/ingested (DC_{ing}, DC_{inh} and DC_{ext} are the dose coefficients for ingestion, inhalation and external exposure, respectively).

Tier 2

The second tier is illustrated in Figure 2, and is designed to ensure defensible and very cautious RQs. In contrast to Tier 1, Tier 2 takes dispersion and accumulation in the biosphere into account, and thus the degree of realism is increased. The generic model applied is still sufficiently cautious for the screening purpose, and includes generic biosphere sub-models: a cropland, a lake, and a well. These three sub-models are evaluated together, and the highest exposures by different pathways are used for deriving the RQs, which may differ for different radionuclides.

Radionuclide releases from the geosphere are directed to the well and small lake, and from the lake to a small terrestrial area (cropland) via runoff. The irrigation of vegetation with water from the lake and the well is also considered. Losses of radionuclides in the well and the lake are neglected and, therefore, the whole release reaches the terrestrial area, where radionuclides accumulate. The retention in the soil is maximised by using a high value of the distribution coefficient (Kd) in estimation of the sorption of radionuclides.

Each sub-model applies cautious assumptions regarding transport and retention. For example, the geometrical properties of the lake and cropland are selected to support exactly one person with food, and the mixing capacity of the well is cautiously selected. The parameter values are also, to as great extent as possible, cautiously selected (see Hjerpe & Broed 2010 for more details and justifications for the selected data). For example, the selected solid-liquid distribution coefficients (Kd) in soil and the concentration ratios from soil and water to biota are the 95th percentiles of internationally recommended distributions (IAEA 2009). In addition, geosphere release rate maxima are used. Tier 2 is applied over the same time window as Tier 1.

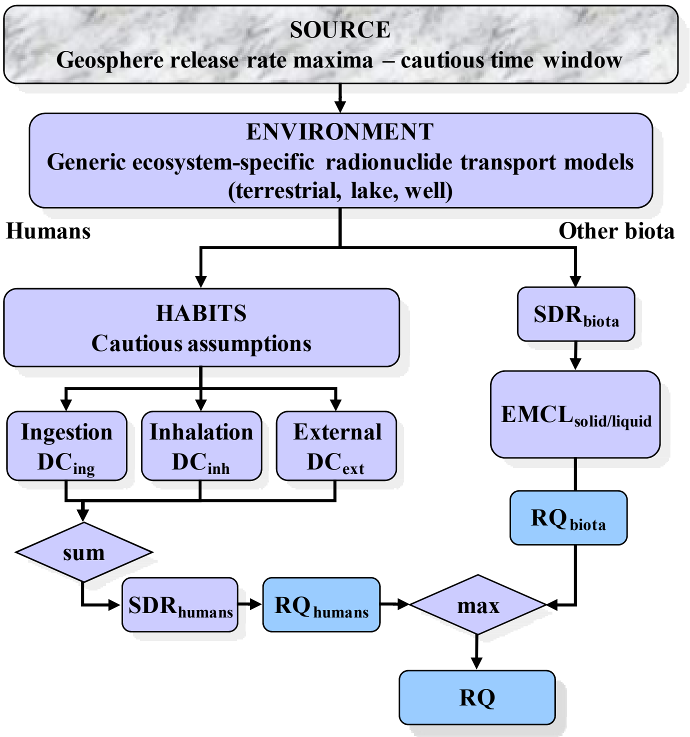

Fig. 2. Conceptual model of Tier 2 (DC_{ing}, DC_{inh} and DC_{ext} are the dose coefficients for ingestion, inhalation and external exposure, respectively).

Doses are calculated to a hypothetical individual that spends 100% of the time on the contaminated land and is exposed via inhalation and externally. It is assumed that the exposed individual obtains 100% of the ingested water and food from the contaminated environmental media. Moreover, it is assumed that all consumed food and water have the highest calculated activity concentration values.

Results

The screening models have been applied to the derived radionuclide release rates from the geosphere to the biosphere, results from *repository calculation cases*, to screen out the radionuclides considered to be insignificant for the biosphere assessment, from a radiological consequences point of view. The repository calculation cases considered in the screening evaluation mainly originating from the most recent assessment for a KBS-3V repository (Nykyri et al. 2008); but also a few cases are selected from the KBS-3H assessment (Smith et al. 2007) and the earlier assessment for a KBS-3V repository ("TILA-99" Vieno & Nordman 1999). All repository calculation cases addressed have their origin in the same type of repository scenario – assuming releases from a single canister with an initial penetrating defect at the time of emplacement. The full descriptions of the cases are presented in the above-mentioned references. The results from the screening evaluation are presented in Hjerpe et al. (2010) and summarised here, limited to the results from the repository base cases (cases Sh1, PD-BC and SH-sal50) and the cases with the highest derived annual effective doses in the landscape modelling (cases Sh4 Q and PD-EXPELL). This limited set of calculation cases also contains all individual RQ maxima for all cases considered in the dose assessment performed in the interim safety case, thus the outcome presented below is identical to the outcome that dose assessment.

The resulting RQs from the screening evaluation, for the five repository calculation cases considered here, with Tier 1 are presented in Table 1. At this tier, 11 radionuclides are screened out. The resulting RQs from the screening evaluation, for the five repository calculation cases considered here, with Tier 2 are presented in Table 2. At this tier, an additional number of 13 radionuclides are screened out. The reaming set of radionuclides to consider in Tier 3, and their RQ maxima from the Tier 2 evaluation, are summarised in Table 3.

Discussion

The radionuclide releases from the geosphere to the biosphere, in the repository calculation cases considered, contain 35 radionuclides, some also including progeny. At Tier 1, 11 of them are screened out and 13 more are screened out Tier 2, t. The set of remaining 11 radionuclides, and their progeny, is denoted the *key set of radionuclides* for the biosphere assessment performed for the interim safety case. It is only this set of radionuclides that are considered in Tier 3: the site-specific state-of-the-art radionuclide transport modelling and dose assessment.

It is notable that all actinides and all radionuclides in the naturally occurring decay chains are excluded from the key set of radionuclides (Table 3). However, the applied screening assessment is only valid for the time window where the dose constraints are assumed to apply and for the analysed scenario with a single defective canister. Hence, this does not mean that it is only these 11 radionuclides, and their progeny radionuclides, that are important for long-term safety. A similar screening evaluation carried out beyond the dose assessment time window, or for other scenarios, such as human intrusion, would most likely return another set of radionuclides; in this case, Ra-226 and Pa-231 would certainly be regarded as key radionuclides (see for example the rock shear cases in Nykyri et al. 2008).

Conclusions

This paper has presented the methodology for the three-tiered graded approach to dose assessment adapted in the Posiva safety case. Further, results from the successful implementation of the screening evaluation part (Tiers 1 and 2) of the graded approach in the interim safety case have been presented. The outcome shows that by applying a simple screening evaluation, a significant part of the radionuclides existing in the potential releases from the geosphere do not need to be assessed with more realistic, and more complex, biosphere models. Using simple screening models it can be shown with a high confidence that these radionuclides would have insignificant radiological consequences during the time window where the regulatory dose constraints are assumed to apply.

Table 1. Resulting RQs from the Tier 1 screening evaluation.

Radionuclide	Sh1	PD-BC	SHsal-50	Sh4 Q	PD-EXPELL
C-14	5E+07	5E+08	1E+08	8E+08	2E+10
Cl-36	4E+06	3E+08	4E+07	4E+07	9E+08
Ni-59	4E-13	3E-13	2E+03	7E+04	8E-04
Ni-63	6E-12			1E-07	
Se-79	3E+00	3E+02	3E+05	3E+01	4E+06
Sr-90+d	9E-09		5E+04	2E+04	
Zr-93+d	1E-16	1E-16		1E+00	2E-03
Mo-93+d	3E+00	2E+02		4E+01	3E+09
Nb-94	6E-03	4E-01	7E+02	3E+07	3E+04
Tc-99	7E-18	1E-16		1E+00	2E-03
Pd-107	2E-01	2E+02	1E+01	1E+02	3E+06
Sn-126+d	1E+00	3E+01	3E+07	9E+02	1E+07
I-129	1E+07	5E+08	2E+08	8E+07	1E+09
Cs-135	2E-10	9E-11	6E+03	1E+07	2E-04
Cs-137	3E-07			1E-06	
Sm-151	7E-26			3E-09	
Pb-210	6E-13	3E-09		4E+03	
Ra-226	1E-12	4E-09		7E+03	
Th-229	3E-18	3E-16		2E+01	
Th-230	2E-18	3E-17		3E+00	
Th-232	3E-23	1E-22		1E-07	
Pa-231	1E-12	5E-12		5E+05	
U-233	2E-19	1E-17		2E+00	
U-234	2E-18	2E-17		5E-01	
U-235	4E-19	4E-18		4E-03	
U-236	1E-17	2E-16		7E-02	
U-238	5E-17	5E-18		6E-02	
Np-237	2E-17	4E-18		2E-01	
Pu-239	2E-12	5E-11		1E+02	
Pu-240	1E-12	9E-11		3E+01	
Pu-242	2E-14	3E-13		2E+00	
Am-241	9E-14	5E-15		2E+01	
Am-243	9E-15	9E-16		2E+03	
Cm-245	9E-14	3E-15		2E+01	
Cm-246	2E-19	9E-21		9E-01	

Table 2. Resulting RQs from the Tier 2 screening evaluation.

Radionuclide	Sh1	PD-BC	SHsal-50	Sh4 Q	PD-EXPELL
C-14	1.1E+01	2.5E+02	3.2E+01	2.0E+02	1.6E+04
Cl-36	1.3E+02	1.8E+04	8.1E+02	8.7E+02	1.6E+05
Ni-59	5.0E-17	3.3E-17	3.0E-01	8.1E+00	1.1E-07
Se-79	2.0E-03	2.6E-01	1.7E+02	2.0E-02	2.6E+03
Sr-90 + d	2.9E-11		1.3E+02	4.5E+01	
Mo-93+d	5.5E-04	5.0E-02		1.1E-02	1.0E+06
Nb-94	5.8E-09	3.7E-07	7.3E-04	2.9E+01	3.4E-02
Tc-99	3.6E-23	1.8E-21		1.8E-06	5.2E-09
Pd-107	9.2E-06	1.2E-02	3.6E-04	4.3E-03	9.8E+01
Sn-126+d	9.6E-05	3.1E-03	2.0E+03	8.2E-02	7.7E+02
I-129	5.9E+02	7.7E+04	8.6E+03	3.7E+03	6.6E+05
Cs-135	1.7E-12	6.6E-13	4.6E+01	7.9E+04	1.5E-06
Pb-210	4.6E-19	2.1E-15		3.1E-03	
Ra-226	1.3E-16	6.0E-13		8.8E-01	
Th-229	2.0E-26	1.0E-24		1.6E-07	
Th-230	8.1E-27	1.0E-25		5.6E-08	
Pa-231	1.2E-19	6.3E-19		5.2E-02	
U-233	6.1E-27	1.8E-24		2.1E-07	
Pu-239	5.0E-21	2.7E-19		8.6E-07	
Pu-240	4.3E-21	5.7E-19		1.9E-07	
Pu-242	4.6E-23	1.7E-21		1.4E-08	
Am-241	8.4E-22	4.3E-22		1.1E-07	
Am-243	9.5E-23	2.3E-23		1.0E-05	
Cm-245	1.7E-21	1.5E-22		3.2E-07	

Table 3. Resulting set of radionuclides not screened out in the screening evaluation.

Radionuclide	RQ maximum In Tier 2	Radionuclide	RQ maximum In Tier 2
Mo-93+d	1.0E+06	Sn-126+d	2.0E+03
I-129	6.6E+05	Sr-90 + d	1.3E+02
Cl-36	1.6E+05	Pd-107	9.8E+01
Cs-135	7.9E+04	Nb-94	2.9E+01
C-14	1.6E+04	Ni-59	8.1E+00
Se-79	2.6E+03		

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On a simple method for proving clearance conditions for radioactive waste

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Abstract

Radioactive waste is one of the main problems of nuclear power. Any unusable material (waste) occuring in a part of a nuclear power plant is considered radioactive waste. An important part of these wastes have very low radioactivity, it may be excluded from the system of the radioactive waste regulation. Exclusion of a material from the regulation of radioactive waste is based primarily on a series of measurements to demonstrate the radioactive elements in the material are under the exclusion limits set by law. This paper presents a methodology including a series of relatively simple measurements which prevents time and human resources consuming radiochemical separation procedures. The basic idea is the fact that it is not necessary to know the exact amount of radioactivity, it is sufficient to demonstrate it is under a certain value (clearance limit). It is not necessary a complete characterization of the waste to demonstrate it meets the criteria for unconditional release. In a CANDU type radioactive waste may occur difficult to measure alpha active elements as $^{234/235/238}U$, $^{238/239/240}Pu$ and ^{241}Am; beta active elements as ^{14}C, ^{63}Ni, ^{99}Tc, ^{90}Sr and ^{90}Y and easy to measure gamma emitting nuclides as ^{54}Mn, ^{60}Co, $^{95}Zr/^{95}Nb$, ^{106}Ru, ^{110m}Ag, ^{134}Cs, ^{137}Cs, ^{144}Ce, ^{152}Eu and ^{3}H (^{3}H is considered easy to measure, the LSC technique is quite easy to apply). Determining the gross alpha and beta activity, high resolution gamma spectrometry and LSC counting can reach a well-founded conclusion the material meets all conditions for the release from the radioactive waste regulation system. The method consists of 12 stages, the theoretical basis for each stage will be presented separately. It will present the conditions under which these measurements are made and the basic properties of the equipment (background, efficiency, detection limits, stability, etc.). We will insist particularly on the need of the long-term stability of the installations used for the measurements.

A comparative overview of waste management concepts for large components

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Abstract
During the decommissioning of nuclear power plants large metallic components like steam generators or reactor pressure vessels are playing a relevant role. Depending of their radiological properties a disposal or a further use is possible. To fulfill the requirements of the intended products, different strategies are used. These strategies are following the basic conditions like the radiological or the site situation. A relevant aspect is the possible clearance of the waste material which can be achieved after decay storage where necessary. Based on these premises different strategies are resulting. Some of these strategies are already used in practice. For example, large components can be deconstructed on site with the objective of a final storage. Other strategies are pursuing the goal of the clearance of at least a part of the large components. This can typically be achieved by an after treatment of the fragmented or entire large components, eventually including decay storage. It is within the scope of this article to show up different strategies from a point of view of practical application of dismantling. The strategies will be compared concerning their relevant advantages and disadvantages like the legal situation or radiological protection.

Introduction
Large metal components play an important role in the deconstruction of nuclear power plants. Examples of large metal components include items like steam generators, reactor pressure vessels or pressurizes. Their treatment depends on several factors: the legal framework, the respective radiological situation, the installation situation and the necessity of their removal for the deconstruction planning. These factors are building a background for the decision which strategy for disposal will be used. There are several possible paths for the disposal of large metal components or their disassembled parts: e.g. disposal as radioactive waste or release. It is also possible to use combined strategies as we will show. Via the disposal path of release, metals can be returned into the circulation of secondary resources. On the basis of these framework conditions, there can be imagined several strategies for the disposal that have already been implemented. It is to be stated, however, that these strategies form approaches only. They must be checked as to their usability in each individual case. In part, these strategies can also be combined with one another in order to guarantee optimum disposal under the existing outline conditions. The TÜV NORD SysTec GmbH & Co.

KG was and continues to be involved in many ways in such activities and had gained substantial experience with the different strategies.

Strategies

Cut and Dispose

Under the "Cut and Dispose" disassembly strategy, the large component is disassembled in its installed position. Afterwards, the disassembled parts are subject to further treatment on site or externally according to their radiological situation. This results either in disposal as radiological waste or in release. By means of methods that are well proven in nuclear engineering, it is possible to produce packages suitable for interim or final storage.

Examples of this strategy are the disassembly of the steam generator of the nuclear power plant Gundremmingen A by means of ice-sawing or the currently running disassembly of the reactor pressure tank of the nuclear power plant of Stade.

It is the advantage of this strategy that the operating staff with its experience is available during the direct deconstruction process when the deconstruction is started shortly after operation. That way, the operating history of the components is well known. That experience can be used in the disassembly process. The infrastructure available at the plant, such as lifting gear, can also be used in the disassembly of the components.

The disadvantages include a possible higher collective dose because the facilities are narrow in many cases. An installation of additional shielding may be possible, but will require additional expenditure. Moreover, an increased effort for equipment is required for a remote-controlled dismantling when tailor-made solutions for the respective component are used.

Pack and Go

Under the "Pack and Go" disassembly strategy, the large component is transported to a disposal facility not situated at the location. The further disassembly and disposal of the large component will take place there.

Examples for this strategy are the steam generators of the nuclear power plants of Obrigheim and Stade. The steam generators from Obrigheim were transported to the interim store Zwischenlager Nord, the steam generators from the KKS were transported to the firm Studsvik Radwaste in Sweden (Fig. 1 and 2).

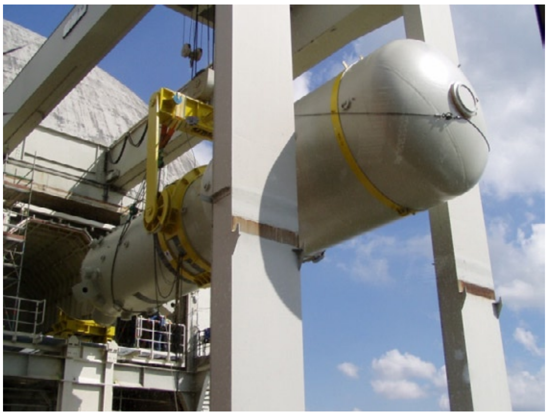

Fig. 1. Steam generator after passing material lock.

Fig. 2. Steam generator during loading on a ship.

It is the advantage of this disassembly strategy that the large components can be removed from the plant at an early stage. That creates space for further dismantling activities. Moreover, this strategy serves to accelerate the deconstruction on site.
It is the disadvantage that the large components need to be carried away in a non-disassembled condition. That requires the preparation of a transport path that allows for a smooth process free from disturbance. Moreover, the selection of the paths and means of transport depends on the possibilities of the plant to be deconstructed as well as of the receiving disposal facility.

For the internal handling of the non-disassembled large components, it must be guaranteed that the transport is possible. That refers for example to the spatial conditions or the lifting equipment available. In this concept it is beneficial that there exists worldwide substantial experience in the disassembly and transport of steam generators, as steam generators have been replaced in numerous operating nuclear power plants.

Pack and Wait

Under the "Pack and Wait" disassembly strategy, the large component is transported to an interim storage facility. There, it is disassembled after a sufficient decay period. The objective is the release of the residual materials to the largest possible degree.

Examples for this strategy are the reactor pressure tanks of the nuclear power plants of Rheinsberg and Greifswald as well as the steam generators from Greifswald (Fig. 3 and 4).

Fig. 3. Large components in the Zwischenlager Nord (ZLN).

Fig. 4. Removal of a steam generator in Greifswald (KGR).

It is the advantage of this strategy that the activity of the large components is reduced by their decay storage. That way, the collective dose for the disassembly staff is reduced. With the release, there can be achieved a higher degree of recycling. Moreover, transport on public traffic routes can be avoided when a local interim storage facility is available. The intermediately stored components are subject to the usual requirements on interim storage. The residual materials must be chemically inert and physically stable on a long-term basis.

It is the disadvantage that the interim storage facility with its infrastructure must be maintained during the decay storage for a period of several decays. That includes for example the lifting equipment, the fire protection and radiation protection installations. During the decay storage, work on the large components is not possible. In the course of the decay storage, there is to be feared a loss of knowledge with regard to the disassembly of large components from the sector of nuclear technology. Additional uncertainties result from legal aspects, e.g. in an unfavourable scenario, the release values might be changed during the decay storage, which might render the planned released more difficult.

The transports are subject to the same conditions that apply also to the "Pack and Wait" strategy. With an interim storage facility at the location, however, no difficulties of a transport on public traffic routes arise.

Pack and Dispose

Under this strategy, the components to be disposed of are directly stored at final disposal sites near the surface. For this method, a respective final disposal site must be available. This method is used in the USA for example.

The transports are subject to the same conditions that apply also to the "Pack and Go" strategy because the large component is transported in a non-disassembled condition.

Transport Paths

For the transport of non-disassembled large components, the internal transport paths as well as the public traffic routes must be taken into account.

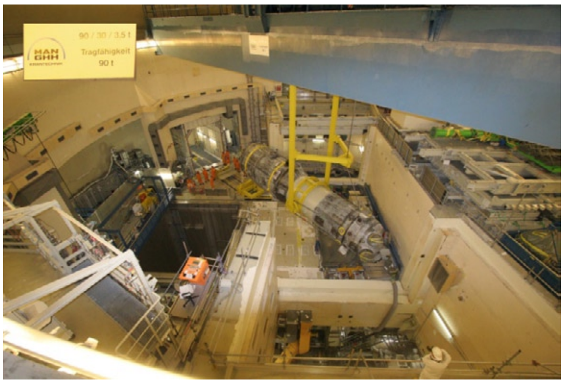

Fig. 5. Handling of a steam generator in the containment.

Fig. 6. Steam generator during loading.

The internal transports require a special consideration of the lifting equipment in many cases. As an example, the reactor-hall crane at a German nuclear power plant shall be mentioned. That crane is designed for operation according to the specifications of the German Nuclear Safety Standard KTA 3902 "Design of Lifting Equipment", Section 4.2 "Cranes, Jacks, Crane Trolleys and Load Lifting Devices with Additional Requirements". With that design, the crane is dimensioned for an operating load of 90 t. The steam generators, however, have a weight of about 180 t each. Nevertheless, it was possible to use the reactor-hall crane for these transports (Fig. 5). That was possible by examining the consequences of an assumed crash of the load. These considerations showed that the operation according to the design under KTA 3902, Section 3 "General Requirements" was possible with a mounting load of 180 t. Afterwards, the crane was recurrently checked and reclassified under KTA 3902 Section 4.2.

The transports on the state territory must meet the requirements of the legal regulations on the transport of radioactive goods. So the pipe connections of the steam generators of the nuclear power plant of Stade (KKS) were closed. The reactor pressure tanks of the nuclear power plants of Greifswald (KGR) and Rheinsberg (KRR) were enclosed in additional shielding for transport and storage.

Moreover, an optimal transport route must be selected. The requirements on the transport route result from the dimensions of the large component and from the possibilities on site. For example in Stade, it was possible to load the steam generators on a ship that took them to Sweden for further treatment (Fig. 2 and 6). The steam generators from Obrigheim (KWO) were transported by ship, too. The dimensions of the reactor pressure tank of the nuclear power plant of Rheinsberg (KRR) permitted a transport by rail. A transport by road would have been possible, too, in that case. But that possibility was rejected in favour of the technically more simple transportation by rail.

Capacities of TÜV NORD

TÜV NORD SysTec GmbH & Co. KG was and continues to be involved in many ways in a lot of such activities. As an example, we accompanied the disposal of the steam generators of the nuclear power plant of Stade from the preparation of the expert opinion up to the return transport of the radioactive waste from the disassembling facility and the release of the residual metallic materials.
Our activities comprise in particular the following fields:

- Radiological safety
- Decontamination / clearance measuring
- Evaluation of the suitability of the radioactive waste for interim storage
- Disposal of radioactive residues and waste
- Process strategy / disassembly engineering / remote-controlled dismantling
- Documentation audits

Conclusions

There are many approaches to the disposal of large components made of metal. Specific disposal strategies can be developed depending on the existing framework conditions. The framework conditions comprise for example the legal situation, the radiological situation, the spatial conditions and the planned ways of disposal. On the basis of these framework conditions, the strategies must be checked as to their applicability. All of the above mentioned strategies had been or are actually realised during the decommissioning of large metallic components according to technical feasibility or legal requirements.

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[4] Pictures: EWN, M. Gerngroß, KKS, S. Wiese.

Recovery from old intermediate level liquor spillage in redundant magnox waste Silo

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Abstract

A Silo at Sellafield has been used to store historic intermediate level waste from irradiated magnox fuel decanning operations undertaken from 1964 until 1991. This material is stored under water in 22 separate compartments.

In 1999, there was an event on one compartment that resulted in a liquor spill on the main silo operations floor with contact dose rates estimated at approximately 2 Sv/h in contact and general waist height dose rates of tens of mSv/h.

Retrieval of the contents from all of the Silo compartments in the building is one on the UK's most important nuclear risk reduction activities. In order to do this, it has been necessary to decontaminate and shield the affected area as far as reasonably practicable so that Silo emptying equipment can be deployed.

Detailed evaluation of the spillage was performed over the course of several years, until equipment was identified capable of removing an acceptable layer of the silo operations floor while maintaining its structural strength. Deployment of this equipment will be covered along with the ALARP processes used to identify an acceptable level of shielding for the floor area, taking account of expected future operations.

The presentation will describe the significant radiation and contamination challenges and the ALARP techniques undertaken during the course of the work. Photographs and radiological images will be used to illustrate how the project progressed through the phases of work, include a summary of the immediate follow up to the event.

A summary of the lessons learnt through this project will be presented. This will review the capability of dose prediction, suitability of the contamination control arrangements and utilisation of "radscan" imagery in an operational environment.

Efficient and environmentally sound management of radioactive waste streams from maintenance, upgrade and decommissioning

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Abstract

Power generation of today struggle with many challenges and as the effects of global warming become obvious many countries seek alternatives to fossil fuel. The increasing interest in nuclear technology is based on its many advantages but the technology also offers challenges in how to cope with prolonged lifetime support and sustainable waste routes.

Studsvik have been processing Low Level Waste (LLW) at its licensed facility in Studsvik, Sweden since the mid-1970s. The facility has historically processed metallic and combustible waste and can today demonstrate a well defined, cost effective and robust waste volume reduction route for both the Swedish and the international nuclear industry.

Introduction

Significant volumes of Low Level Waste are today generated by the nuclear industry. Some waste has well defined routes for disposal using economically and technically efficient solutions. Other waste is temporarily stored on power plant sites waiting to be conditioned before final storage at dedicated repositories. The reasons for intermediate storage of waste may be several, such as radiological or physical complexity, and therefore on site conditioning for final repository may be technically challenging or time consuming and as such expensive.

Various technical options are available in the industry where one suitable route for waste includes external treatment by specialists. This helps the industry to minimize the cost for volume reduction and conditioning of waste before final repository. One such route includes Studsvik Nuclear AB in Sweden. With more than 60 years of experience in the nuclear industry in Sweden, Studsvik has since the 1970's treated combustible LLW and since the 1980's treated metallic LLW for the national and the international market. Today Studsvik is a leading specialist on waste volume reduction of Low Level Waste.

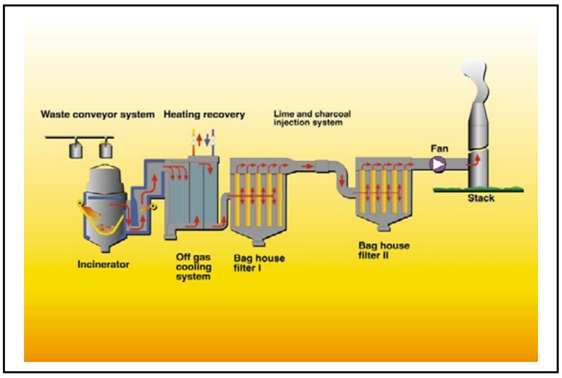
Waste treatment is also logistics and transportation. Studsvik has over the years developed a good understanding of the complexity in national and international transportation of radioactive waste. Such transports always require close cooperation with the customer as well as with regulatory authorities in both Sweden and the country of origin for non-Swedish waste.

Waste categories and methods

Studsvik focuses on treating two typical waste streams generated by the industry: low level combustible waste suitable for incineration and low level contaminated metallic scrap.

Combustibles are treated in a high temperature incinerator for maximum destruction, resulting in minimized volume and a very stable residue suitable for final repository.

Typical combustible waste is protective clothing, wood, paper, plastics, etc. According to Studsvik acceptance criteria waste is typically pre-packed by the owner into transparent plastic bags, < 25 kg to allow manual handling, sorting and inspection. This waste is typically shipped in 20-foot IP-2 containers and is well characterized for physical properties as well as radiological content (maximum dose rate and nuclide inventory).



Schematic overview of incinerator

Typical combustible waste

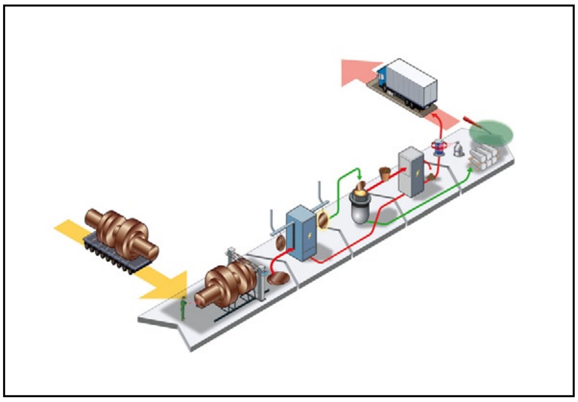
Treating combustible waste typically consists of:

- Arrival inspection and verification of transport documentation, including radiological data
- Safety inspection of all waste and an assessment of the scope of work required
- Loading waste on the conveyor system into the incinerator
- Collection of residues resulting from the treatment, ashes and fly-ash as secondary waste
- The secondary waste, including radioactive contamination, is analysed to verify that the activity content matches the original declaration by the customer to ensure all is returned back to its rightful owner
- All residual secondary waste is conditioned and shipped back with a detailed production report declaring the result of the processing

After processing the waste the remaining ash and the fly-ash accumulated in the bag house filters is collected in drums. This secondary waste is conditioned according to customer specifications and is returned to the owner together with a detailed production report.

Environmental aspects is an important part of Studsvik operations and off gas filtering is enhanced by using lime and activated char coal to minimize emissions. To further reduce the environmental impact, Studsvik is recovering energy (~1,5MW) by reusing it within the site to heat facilities.

Metal scrap is comprised of a wide variety of components such as pumps, pipes, duct work and other small waste fitting into a container. Studsvik is also capable of processing large components. This includes heat exchangers and turbine components such as housing and turbine axel, with a typical weight of 150 - 250 tonnes. Large PWR steam generators, weighing 300 tonnes or more, are successfully treated on a regular basis.



Schematic process flow for metal scrap

Typical metal scrap recycled by Studsvik consists of carbon steel, iron alloys, aluminium, lead, copper, brass, etc. Waste is delivered in transport packages such as 20-foot IP-2 containers and large components are transported under alternative arrangements according to international regulations. The acceptance criteria are well defined for physical properties as well as radiological content (maximum dose rate and nuclide inventory).

The treatment of metal waste is shown in the schematic illustration above and can be described as follows:

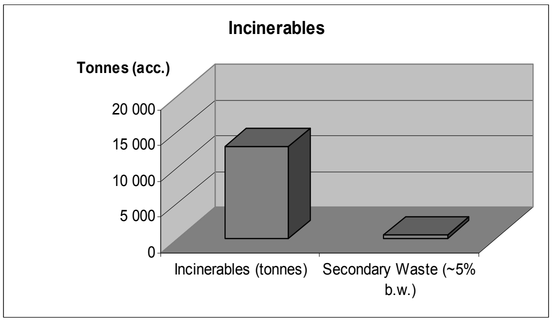
- Arrival inspection and verification of transport documentation including radiological data
- Safety inspection of all scrap and an assessment of the scope of work required
- Segmentation as needed to suitable size and weight
- Surface contamination removed by shot blasting and collected in customer isolated campaigns
- Decontaminated scrap metal is melted into ingots and analytically sampled to verify successful processing

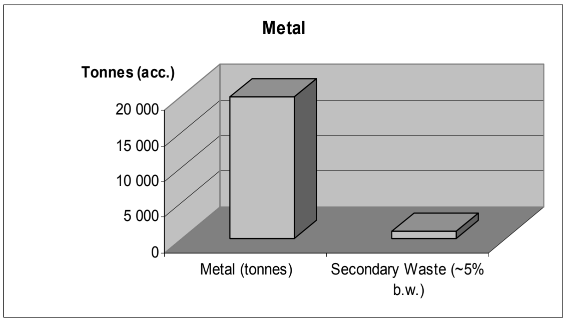
- To ensure all is returned back to its rightful owner the residual secondary process waste is analysed to verify that the activity content matches the original declaration by customer
- All residual secondary waste is conditioned and shipped back with a detailed production report declaring the result of the processing
- Metal ingots analytically verified to be in compliance with free release regulation is declared exempt by Studsvik, recycled as valuable raw material and routed back into the metal industry through internationally accepted procedures

Typical scrap metal processed at Studsvik, Sweden (Misc. scraps at left, Turbine at right)

Results

Studsvik has in total treated an excess of 10’000 tonnes of combustible material and 20’000 tonnes of metal at its site in Sweden. Today the annual processing rate is approximately 500 tonnes of combustibles and 3500 tonnes of metal with a very high rate of volume reduction, shown below in the two graphs as the average over the historical production. Specific details depend on waste character and may vary.





The resulting secondary waste is prepared according to customer specifications before it is returned. The conditioning typically includes packaging in a 200 litre drums and returning the drum back to the owner where the package undergoes final preparations before storage in a repository in the country of origin. Other forms of packaging may be required on the international market and can be arranged after individual agreement.

All final packages are analyzed and labelled, the results of which is reported in the final production report. After a completed campaign all secondary waste is returned to customer in country of origin to ensure volume reduced waste (containing the radioactive inventory) is brought back to its rightful owner for final storage.

Gamma-scanning

Exempt metal for recycling

Metal samples from each melting batch is analyzed to document data and to allow recycling as exempt metal. Metal ingots meeting exempt levels will be recycled after transfer of ownership from customer to Studsvik.

Transportation

Transport of radioactive waste is well regulated at the international level. In order to safely and compliantly transport waste for treatment a variety of regulations must be understood and applied. Studsvik has many years of experience in this field and has a detailed understanding of the regulations for land (ADR - Accord européen relatif au transport international des marchandises Dangereuses par Route), sea (IMO – International Maritime Organization) and air (Dangerous Goods Regulations) transport. Radioactive waste is, however, not transported by air freight due to several reasons, the main one being cost.

To perform a shipment of radioactive waste not only the relevant transport regulation must be complied to fulfil safety concerns but also a special transport authorization must be granted. This authorization must be given according to directive Euratom 2006/117 and approved by the competent authority in both countries. The approval operates on a system of prior notification and approval where-by the applicant (consignee) must apply to the competent authority in the country of export for consent to ship the waste internationally. They in turn seek approval from the competent authority in the country of import (and any countries transited across) prior to approving, or otherwise, the shipment. The Euratom 2006/117 system also enables the competent authorities to place conditions on the shipment such as a requirement to return the secondary waste (ash, dust, slag, etc.) to the country of origin, or to give advance notice of when the consignment will ship.

Studsvik regularly performs, in close cooperation with customers, their competent authorities, and Swedish authorities, many international transports from, and back to, countries within EU as well as other international countries. A mandatory condition in all cases is that any individual campaign must in normal cases be concluded within 24 months and all generated secondary waste (any and all waste not possible to classify as exempt and free releasable) must be returned to country of origin.

Conclusions

As nuclear power generation experiences a revival many questions on waste treatment must be answered. Prolonged lifetime and worldwide discussions of new power plant constructions highlights the obvious demand of modern, effective and safe waste treatment. Cost for storage at repositories is often volume driven and as such large savings can be obtained by volume reduction before disposal, which enables economically feasible management of waste. The process not only ensures that the volume of waste is significantly reduced but also produces a stable residue suitable for final disposal in a highly cost effective manner. This route offers clear benefits with reduced demand of required space in final repository with prolonged life time at reduced over all cost.

Effective waste treatment is also about a responsible engagement in environmental aspects. Recycling of valuable raw material and energy helps to further reduce the carbon footprint from the energy industry. By using the heat generated during incineration of combustible waste as energy (~1,5MW) to heat on site facilities the carbon foot print is dramatically reduced by replacing fossil fuel energy sources. Metal meeting exempt criteria will be circulated back into the metal industry as a valuable raw material. Reusing high quality metal saves on the environmental impact from our industry and uses limited natural sources efficiently.

Determination and quantification of NORM radionuclides

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Abstract
This paper discusses methodologies which can be used to identify naturally occurring radionuclides in substances using automatic spectrum analysis. The possibility to use automatic spectrum analysis in order to obtain an estimation of the activity concentration of the detected radionuclides is examined. The goal is to determine whether or not the activity concentration of a measured substance meets the regulation defined in the new upcoming European Basic Safety Standards with respect to naturally occurring radioactive materials (NORM). Existing techniques which are primarily used in lab settings are applied in industrial settings and tested. These techniques include artificial neural networks and automated spectrum analysis. The methodology is assessed and a comparison is made between NaI(Tl) and LaBr(Ce) based multi-channel analysers. The methodology is applied to several NORM materials with distinct activity concentrations: zirconium, ilmenite, bauxite, and fluorspar. In this way a methodology is constructed which can be applied in situ to allow determination and quantification of naturally occurring radionuclides.

Introduction
Radionuclides of the natural radioactive series of uranium and thorium are present everywhere in the earth crust. Concentration of these nuclides is depending on the composition of the soil. Besides uranium and thorium nuclides, an amount of potassium-40 is also present. All these nuclides produce radiation doses to all human beings. In addition, industrial processes can lead to an accumulation of naturally occurring radioactive materials (NORM) due to operations in end-, by- or waste-products. The concentration of nuclides can be accumulated in such a way that radiation protection is necessary.

A uniform approach towards NORM by all European member states is a primary goal. The European Commission is therefore currently recasting Council Directives with respect to natural radiation sources (European Commission 2010). One of the new elements is the construction of a positive list of industrial activities in the non-nuclear sector that may be subject to notification. As already described in UNSCEAR 2000 and ICRP Publication 75 the acceptable dose rate threshold can be aligned with the activity concentration of materials. As a result the activity concentration will be used to determine whether regulatory authorities need a prior authorisation.

This means it is important to have a technique to determine the activity concentration of NORM. Since the activity concentration has to be measured in all the products, by-products and residues in an industrial process, a fast and usable methodology is mandatory. Nowadays the activity concentration is generally being measured by sample analysis which is a cumbersome, time and money consuming job.

In this work the objective is to construct a tool that can automatically decide in an industrial setting whether a measured substance complies with the new European Directive's activity concentration limits. This has to occur in known geometries by using a relatively cheap NaI(Tl) probe-based multichannel analyzer (MCA) connected to a tablet-PC. Custom software on the tablet steers the MCA to capture a spectrum and to interpret the spectrum straightaway. Based on the spectrum, the weight of the substance and its density, an estimation of the activity concentration is made and presented to the user.

To assess the technique proposed we captured several spectra of distinct NORM materials in a known geometry of a NORM handling company. The company stores and handles sugars, fertilizers, chemicals, minerals, iron, steel and wood products. The handling comprises bagging, repacking, sieving, sifting, weighing, mixing and conditioned storage in contamination-free warehouses. The accuracy of the tool was assessed by comparing the results with the results of an accepted method based on sample analysis.

In this work we focus on tool support. The objective is to construct a tool that can automatically decide in an industrial setting whether a measured substance complies with the new European Directive's activity concentration limits.

Tool support for determination and quantification of NORM radionuclides in industrial settings

An important goal of our current research project is to make an inventory of the issues of NORM in a NORM handling company in collaboration with the Federal Agency of Nuclear Control.

The measurement tool was designed in such a way that at the end of the project the employees of the industry can use it independently. Therefore it has to be user-friendly and applicable in distinct industrial circumstances. Software is constructed that can be easily adapted according to the environmental context: several distinct geometries are present so that the end user can select the geometry that is applicable to the current context-of-use. For example: when the substance is contained in a shipping container, this geometry can be selected and subsequently parameters with respect to a shipping container (dimensions, probe location, container wall properties etc.) can be filled out by the user. On the other hand when the substance is contained in a bigbag (common used bag in industry containing dry bulk substances) this can be selected by the user and geometry properties can be filled out by the user accordingly (diameter, circumference, etc.).

As big bags were the most common geometry in this company, the tool was at first adjusted for this model (Fig. 1). Big bags contain dry bulk substances with a mass typically between 1 and 2 ton. Activity concentrations of nuclides in big bags were determined on the side but also on the top of the bag. The measurements on the top correspond the most to the sample analysis.

In practice, the bags were placed outside the warehouse to avoid interference with radiation caused by other bags. Geometry of the bag was determined (height, net mass, perimeter, volume). Two probes (NaI(Tl) and LaBr(Ce)) were set on the top and in contact with the bags and were connected with the spectrometer and software. Measurements were repeated 10 times for 15 minutes in order to control the reproducibility of measurements. A spectrum was automatically generated and analysed.

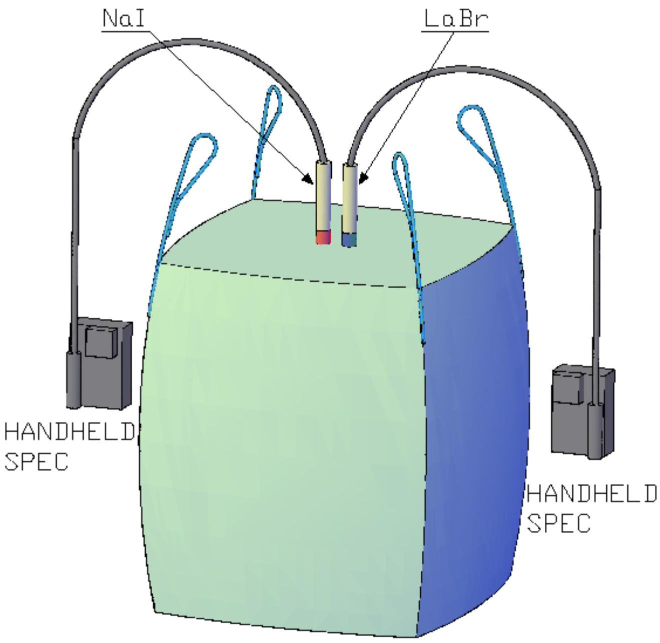

Fig. 1. A bigbag with two multi-channel analysers (NaI(Tl) and LaBr(Ce)) recording a spectrum of a known substance.

In the following sections two techniques are discussed which can be applied to identify NORM radionuclides by using Artificial Neural Networks and to quantify the activity concentration of NORM radionuclides by applying energy calibration by modelling the measured container. Both techniques make use of automated spectrum analysis of the spectra recorded in the industrial setting of the NORM handling company. The accuracy of the tool was checked by comparison of the analyses with sample analysis.

Determination of nuclides with Artificial Neural Networks (ANN)

This section elaborates on a technique to classify NORM radionuclides using artificial neural networks. The following section will discuss a technique to estimate the activity concentration.

Artificial Neural Networks for gamma-ray spectrum analysis

Artificial Neural Networks ANN simulate biological neural networks in a computational model that can be implemented in a software system. ANN already have a long history of applications in physics and chemistry in general and in nuclear science in particular with proven successful results on spectrum analysis (Long 1997) (Keller et al. 1994) (Yoshida et al. 2002) (Saritha et al. 2009). A key characteristic of ANN is the excellent performance on coping with data containing noise. Milford identified the generalization ability of ANN as the reason for this performance when increasing amounts of Gaussian noise were added to spectra (Milford 2002). Saritha et al. And Keller et al. compared a neural network to an Optimal Linear Associative Memory (OLAM) for the classification of linearly separable data (Saritha et al. 2009) and (Keller et al. 1994). While OLAM was superior to a neural network when Monte Carlo generated spectra were used, the ANN was again found to be the best solution for noisy data.

Practical Approach

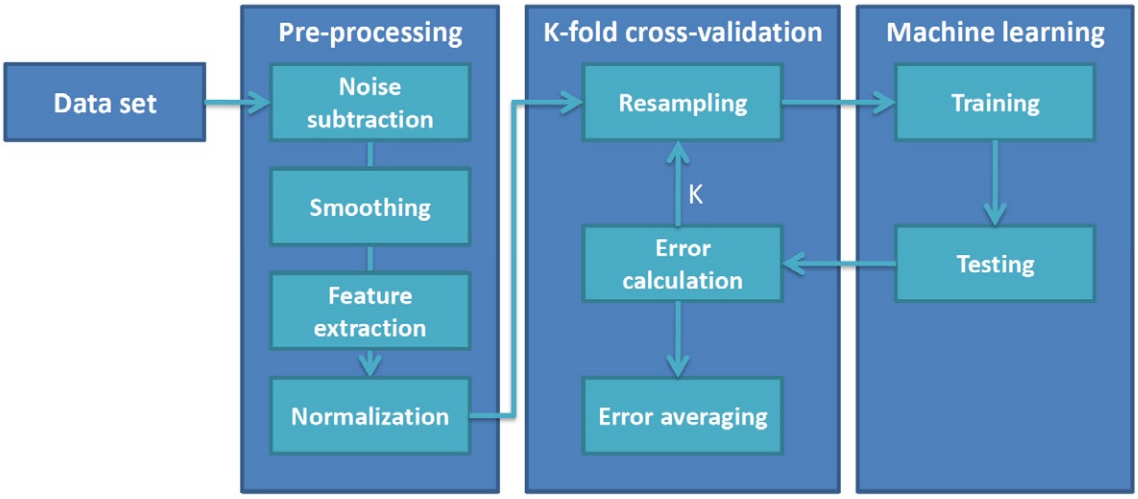

Fig. 2. Schematic overview of the machine learning sequence.

Figure 2 presents a schematic overview of the machine learning sequence of the ANN that has been implemented. The input data set is a collection of raw spectra obtained by the measurements in the industrial case study. The machine learning sequence consists of three steps: pre-processing, cross-validation, and machine learning.

Before the spectral data can be fed to the ANN a pre-processing step is applied to extract the most relevant data from the spectrum. Although this is strictly not necessary since the ANN can learn which data are irrelevant. Feeding 512 to 4096 channels used by an MCA to the ANN would be inefficient and decrease the training speed. Using trial and error the following pre-processing techniques generated the best results on the spectra recorded in the industrial case study: noise subtraction, running-average smoothing, peak-extraction based on Milford, and normalization (Milford 2002). The peak extraction algorithm investigates a specific number of channels surrounding a point to determine whether the point is a local maximum and thus a peak. This reduces the complexity of the input layer.

Figure 3 presents the implemented ANN. The ANN is a fully interconnected feed-forward network with one hidden layer. Back-propagation learning is used to train the ANN. Because of the pre-processing step, the number of input neurons is reduced to twelve, i.e. the respective position and height of six key peaks that can identify the presence of a radionuclide in the spectrum. The number of neurons in the hidden layer has to be chosen carefully in order to find a balance between the model's complexity and the generalization capacity of the ANN.

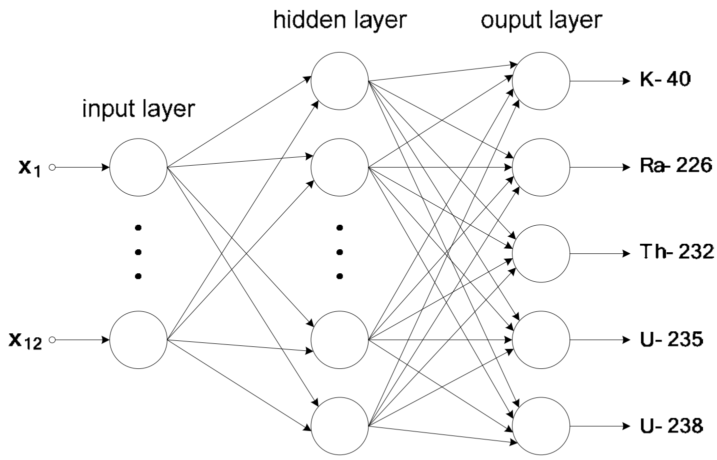

Fig. 3. Model of the implemented ANN.

To validate the ANN k-fold cross-validation was applied with k=10. This means the input data set was divided in ten equal collections of spectra from which nine are used to train the ANN and the remaining one is used for validation. Afterwards the error is calculated and the input data set is resampled for another run. This is repeated until the cross-validation error < 1%.

Results

The gamma spectra are recorded using a NaI(Tl) and LaBr(Ce) detector in the industrial environment. A training set of eighty spectral samples was used that represented all detectable isotopes (K-40, Ra-226, Th-232, U-235, and U-238) in equal numbers. The k-fold cross-validation was applied to exploit the reasonably small data set of eighty samples. A series of ten repeated 10-fold cross-validations were used to verify the performance of the ANN.

In order to optimize the performance of the network, the following parameters were isolated and compared by their accuracy, mean-squared error (MSE) and mean-absolute error (MAE):

- Momentum: scaling factor to specify to what degree the weights of the neurons change from the previous step of the learning algorithm. The momentum appeared to be acceptable with respect to the accuracy and the MSE between 0.1 and 0.3.
- Hidden-layer size: the MSE and MAE both reach a minimum when the hidden layer consists of four neurons.

Furthermore the learning rate and the number of epochs (iterations of an entire training set) were taken into account.

The results showed that the ANN classified the spectra with an accuracy of 85% on average which corresponds to an error rate of 15%. This large error was due to the problems that the ANN had with the classification of U-238. The best performance of the ANN on U-238 was 87%. The other isotopes (K-40, Ra-226, Th-232, and U-235) were classified with an accuracy of 100%.

Quantification of NORM radionuclides

In earlier work we discussed how the activity concentration of NORM radionuclides was quantified using a NaI(Tl)-based multi-channel analyser (MCA) using spectrum analyses on spectra captured in an industrial setting (Pellens et al. 2010). In this section we will compare results of spectrum analysis captured by an MCA equipped with a NaI(Tl) probe with spectra from the same substances captured in the same environmental circumstances but with an MCA equipped with a LaBr(Ce) probe. Furthermore the technique is assessed by comparing these results to the results of sample analysis performed by a certified institution.

Practical approach

The approach consists of automated spectrum analyses performed by customized software. Four substances known as NORM were measured in bigbags containing between 1000 kg and 2000 kg. The software was adapted to cope with spectra recorded by either NaI(Tl) and LaBr(Ce) probes. The final output of the software was an estimation of the activity concentration of NORM-radionuclides present in the measured substance. Measurements taken with NaI(Tl) and LaBr(Ce) are compared to results from sample analyses taken from the respective bigbags. The analyses on 250 g samples were performed by a certified lab. In the next section this comparison will be discussed.

The automated spectrum analysis was applied to ten spectra recorded on each substance. The same automated analysis algorithm was applied to the distinct substances.

Up to now, four distinct substances were considered: zirconium, ilmenite, bauxite, and fluorspar. The substances were chosen in a way that there was a variety in the respective activity concentration of the radionuclides present.

Results

The results are presented in three tables. Each table presents the results of one NORM radionuclide by comparing the automated spectrum analysis of spectra recorded with the two types of probes (Na(Tl) and LaBr(Ce)), the sample analysis by the certified lab and the recommendation as described in the reviewed basic safety standards by the European Commission. The results are presented for each of the four substances considered in this research.

Table 1. Comparison of spectral analysis results of 900s measurements using a NaI(Tl) and LaBr(Ce) probe with results of a certified lab for the Th-232 radionuclide.

Substance	NaI(Tl)		LaBr(Ce)		Certified lab SA	EC BSS
	Act Conc ($Bq·g^{-1}$)	Rel. Error (%)	Act Conc ($Bq·g^{-1}$)	Rel. Error (%)	Act Conc ($Bq·g^{-1}$)	
Zirconium	0.53	6	0.52	5	0.50	
Ilmenite	0.28	-16	0.36	6	0.34	
Bauxite	0.29	-2	0.33	12	0.30	
Fluorspar	<MDA = 0.03		0.08		<MDA = 0.02	

Table 1 presents the results of the Th-232 radionuclide. The sample analysis shows that the performance of the automated sample analysis of the zirconium spectra are nearly the same for the two different probes. There is an overestimation of 5% to 6% in comparison to the certified sample analysis. The activity concentration of Th-232 in ilmenite is performed way better with the LaBr(Ce) probe (6% overestimation) in comparison to the NaI(Tl) (underestimation of 16%). The automated analysis of bauxite presents a different picture: the analysis of the NaI(Tl) probe delivers a better result. Finally the activity concentration of fluorspar does not exceed the minimum detectable activity (MDA) for the NaI(Tl) analysis and the sample analysis. We can conclude that the automated spectrum analysis for the LaBr(Ce) probe delivers an overestimation of the activity concentration of Th-232 in comparison to the NaI(Tl) and sample analysis results. This introduces a more safe result than the NaI(Tl) which may deliver an underestimation. The last column of the table compares the results with the revised basic safety standards. None of the substances exceeds the limit of 1 $Bq·g^{-1}$ for the Th-232 series.

Table 2. Comparison of spectral analysis results of 900s measurements using a NaI(Tl) and LaBr(Ce) probe with results of a certified lab for the U-238 radionuclide.

Substance	NaI(Tl)		LaBr(Ce)		Certified lab SA	EC BSS
	Act Conc ($Bq·g^{-1}$)	Rel. Error (%)	Act Conc ($Bq·g^{-1}$)	Rel. Error (%)	Act Conc ($Bq·g^{-1}$)	
Zirconium	1.79	-8	2.13	9	1.94	
Ilmenite	0.09		<MDA = 0.053		<MDA = 0.31	
Bauxite	<MDA = 0.05		0.25		<MDA = 0.58	
Fluorspar	0.08		0.19		<MDA = 0.40	

Table 2 summarizes the results of the same analyses for the U-238 radionuclide. According to the sample analysis results of the certified lab, only zirconium exceeds the minimum detectable activity. The activity concentration delivered by the automated spectrum analysis of the NaI(Tl) and LaBr(Ce) spectra are comparable: an underestimation of 8% is produced by the NaI(Tl) and an overestimation of 9% by the LaBr(Ce). The activity concentration exceeds the limit of 1 $Bq·g^{-1}$ for the U-238 series.

Table 3. Comparison of spectral analysis results of 900s measurements using a NaI(Tl) and LaBr(Ce) probe with results of a certified lab for the Ra-226 radionuclide.

Substance	NaI(Tl)		LaBr(Ce)		Certified lab SA	EC BSS
	Act Conc ($Bq \cdot g^{-1}$)	Rel. Error (%)	Act Conc ($Bq \cdot g^{-1}$)	Rel. Error (%)	Act Conc ($Bq \cdot g^{-1}$)	
Zirconium	1.8	-15	2.1	-2	2.14	
Ilmenite	<MDA = 0.04		<MDA = 0.053		0.04	
Bauxite	0.27	-8	0.27	-9	0.30	
Fluorspar	0.11	-37	0.19	7	0.18	

Finally table 3 presents the results of the analysis of the Ra-226 radionuclide. For the substances zirconium and fluorspar the LaBr(Ce) outperforms the NaI(Tl): a respective underestimation of 2% and overestimation of 7% versus an underestimation of 15% and 37% for the Ra-226 activity concentration. The results of the spectrum analysis for bauxite are comparable. Again, the only exceeding activity concentration in comparison to the basic safety standards requirements is detected for zirconium.

Discussion

Looking at the results presented in the previous section, a conclusion can be made that the LaBr(Ce) performs better than the NaI(Tl) when automated spectrum analysis is applied en builds assurance because the error is mostly an overestimation. The results are significantly better when Ra-226 has to be quantified. The comparison of the performance on U-238 is about the same, but only zirconium exceeded the minimum detectable activity. The results of the Th-232 analyses show distinct performances in comparison to the certified sample analysis.

An important issue in the automated spectral analysis seemed to be the detection and quantification of U-238. The certified method using sample analysis could not deliver a justifiable quantification of the activity concentration for three of the four substances. The automated spectral analysis method delivered an estimation of the activity concentration below the minimum detectable activity of the certified lab. Using the automated spectrum analysis the proposed method is able to quantify the activity concentration accurately enough to determine whether the activity concentration is below or above the threshold used in the new European basic safety standards. Further research is required to narrow down the limitations of this methodology to determine when a more accurate analysis is necessary, i.e. when the activity concentration approximates the basic safety standards' limits.

Conclusions

Automated spectrum analysis was applied on spectra recorded in an industrial setting. The main objective was to study whether automated spectrum analysis can be used in providing a fast tool to check the applicability of the new European basic safety standards with respect to NORM material. A methodology has been constructed consisting of a tablet-PC running a software that steers a measurement and interprets the spectrum accordingly to approximate the activity concentration of the measured substance.

Both machine learning techniques and traditional automated spectrum analysis were studied in order to determine and quantify NORM radionuclides. Spectra were recorded with two different probes (NaI(Tl) and LaBr(Ce)). Results showed a better performance of the analyses with a LaBr(Ce) probe. In this paper, we showed one can use an ANN to determine the presence of K-40, Ra-226, Th-232 and U-235 with an accuracy of 100% while the determination of the presence of U-238 is 87%. It is in our intention to use these figures to optimize the quantification algorithms used in the automatic spectral analysis.

Results of analyses of four investigated NORM materials were compared with the Council Directives (whether the measured values were in the safe zone, around the threshold, or above the threshold). The four substances which were the subject of this study were accurately classified according to the basic safety standards taking the relative error into account.

Further optimization of the tool will be an important objective of this project. The tool will be able to tell in a very short period if a company needs notification according to the new European basic safety standards.

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The European Commission is currently in the process of recasting five Council Directives: Basic Safety Standards Directive (96/29/Euratom), Medical Directive (97/43/Euratom), Directive on High Activity Sealed Sources (2003/122/Euratom), Directive on Outside Workers (90/641/Euratom), Directive on Public Information (89/618/Euratom). 2010.

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Exposure of workers in France due to naturally occurring radioactive materials (NORM)

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Abstract

According to the council Directive 96/29 Euratom, handling or storage NORM or TENORM has to be considered from the radiological protection point of view. This directive has been implemented in French regulations by, in particular, the Ministerial order of May 25, 2005 related to activities operating NORM not used because of their radioactive properties. It imposes radiological characterization of any materials and assessments of the effective dose received by workers to be done. Since the publication of this text, ASN and IRSN have already received ninety studies which provide information about activities of materials and occupational exposure of nine different types of industrial facilities. These data show that activity concentrations strongly vary according to the type of material and industrial activity. Waste generally contain the highest activity concentration of natural radionuclides. Activities in raw materials and products sometimes exceed the exemption level recommended by IAEA for the use of NORM. Activity of ^{226}Ra and its daughters in waste or activity of ^{210}Pb and its daughters in ashes and dust related to heating processes are sometimes greater than the activities of the other radionuclides of ^{238}U series. Concerning occupational exposure, doses reported by operators range from less than 1 μSv.yr^{-1} to 82 mSv.yr^{-1}. Based on the French feedback, the following conclusions can be drawn. Assessments are still expected for some industries. About 10% of calculated doses are greater than the effective dose limit of 1 mSv.yr^{-1} for the public and need further examination. The highest doses correspond to the production of compounds with thorium. External and internal exposures are often of the same order of magnitude. Some types of industrial facilities currently not concerned by the French ministerial order, e.g. paper mills, are concerned by NORM and TENORM issue.

Introduction

Some industrial activities such as ceramic production or coal combustion in thermal power stations involve the use of materials, usually regarded as non-radioactive, but containing naturally occurring radionuclides (NORM). Handling or storage of these materials can lead to a significant increase of occupational exposure. According to the council Directive 96/29 Euratom (Council Directive, 1996), this matter has to be considered from the radiation protection point of view. French authorities implemented this directive to the French health and labour codes detailed in the Ministerial order of May 25, 2005 (Ministerial order, 2005). It sets out a list of ten types of industrial activities concerned by NORM and operators have to assess the effective doses received by workers on the bases of a radiological characterization of raw materials, by-products and waste involved in or produced by the technological processes. ASN asked IRSN to evaluate the methods adopted by some operators for assessing these doses and to draw the first conclusions in terms of radiological protection. Based on data presented by operators in the ninety studies received so far, a summary of the results of these studies in terms of activity concentrations and occupational exposure are presented hereafter.

Material and methods

Since the publication of the Ministerial order of May 25, 2005 (Ministerial order, 2005), ASN and IRSN have already received ninety studies which provide activities of materials and doses calculated for nine types of industrial activities (cf. table I). The majority of the studies deals with occupational exposure and about 91% of them report on effective doses received by workers.

Table 1. Breakdown of studies by different types of industrial activities.

Industrial activities sets by the French ministerial order of May 25, 2005	Part (%)
Production of refractory ceramics and smelting, metallurgy and glass industry using them	49%
Coal combustion in power plants	16%
Production of zircon and baddeleyite, and smelting or metallurgy plants using it	15%
Treatment of tin, aluminium, copper, titanium, niobium, bismuth and thorium ores	7%
Production of phosphated fertilizers and phosphoric acid	6%
Production or use of compounds with thorium	3%
Treatment of lanthanides series and production of pigments containing them	2%
Spas	1%
Underground water treatment by filtration	1%
Treatment of titanium dioxide	0%

Among the ninety studies, 43 present activity concentration measurements mainly carried out by gamma spectrometry for about 500 different samples. It represents more than 4200 results of measurements. Radionuclides of ^{238}U, ^{232}Th and ^{235}U series and ^{40}K represent respectively 47%, 29%, 15% and 9% of measurements and 69% of the measurements for ^{238}U series or ^{232}Th series are above the detection limit (LD), 32% for ^{235}U series and 59% for ^{40}K.

Uranium and thorium series activity concentrations

Measurements taken into account

Activity concentrations mainly measured by gamma spectrometry were analysed by IRSN. Uranium and thorium series have been divided into groups of radionuclides as recommended by the European Commission (Chen *et al*, 2003). For each chain segment, an activity was determined by the method whose principle is described hereafter (Maigret *et al*, 2008). For a given sample, if only one radionuclide activity of a group is measured, its activity stands for the group. If several radionuclide activities are measured and their measurements are consistent, the average of more reliable activities stands for the group. For example, regarding the chain segment of $^{226}Ra^{+}$[1], deconvolution of the ^{226}Ra gamma ray (186.2 keV) and the ^{235}U gamma ray (185.7 keV) cannot be done accurately by the classic HP-Ge detectors (detectors made of High Purity Germanium) mainly used in studies, whereas the activity of ^{214}Pb or ^{214}Bi is easy to measure with those detectors. So the average of activities of ^{214}Pb and ^{214}Bi stands for the group of $^{226}Ra^{+}$.

Activity levels and radioactive disequilibrium in NORM used in French industries

As a first step, all samples presented by operators were classified in four categories: raw materials, products, waste and environment. The category “Environment” represents a fifth of the samples and includes all the samples collected in the environment, e.g. soil, sediments, vegetables… These data have been excluded from our work. The category “Products” includes final products and by-products and the category “Waste” includes solid waste, effluents, sludge and dust. 26% of materials considered by operators are raw materials, 34% are products and 39% are waste.

In the second step, twelve industrial activities were identified: coal combustion, glass industry, production and use of zircon and baddeleyite, production of phosphated fertilizers, production of refractory ceramics, production or use of compounds with thorium, spas, treatment of aluminium ores, treatment of kaolin ores, treatment of lanthanides series, treatment of titanium ores and underground water treatment by filtration. All the samples were classified according to these categories.

Eventually, for each category of materials and for each chain segment, the minimum, the maximum and the median values, as well as the first and the third quartile values have been determined. Figure 1 presents activities of the most often measured chain segments: $^{226}Ra^{+}$ and $^{228}Ra^{+}$ groups.

[1] Symbol ‘+’ after a nuclide denotes a segment chain headed by that nuclide, e.g. $^{226}Ra^{+}$ corresponds to this segment chain: ^{226}Ra, ^{222}Rn, ^{218}Po, ^{218}At, ^{214}Pb, ^{214}Bi and ^{214}Po.

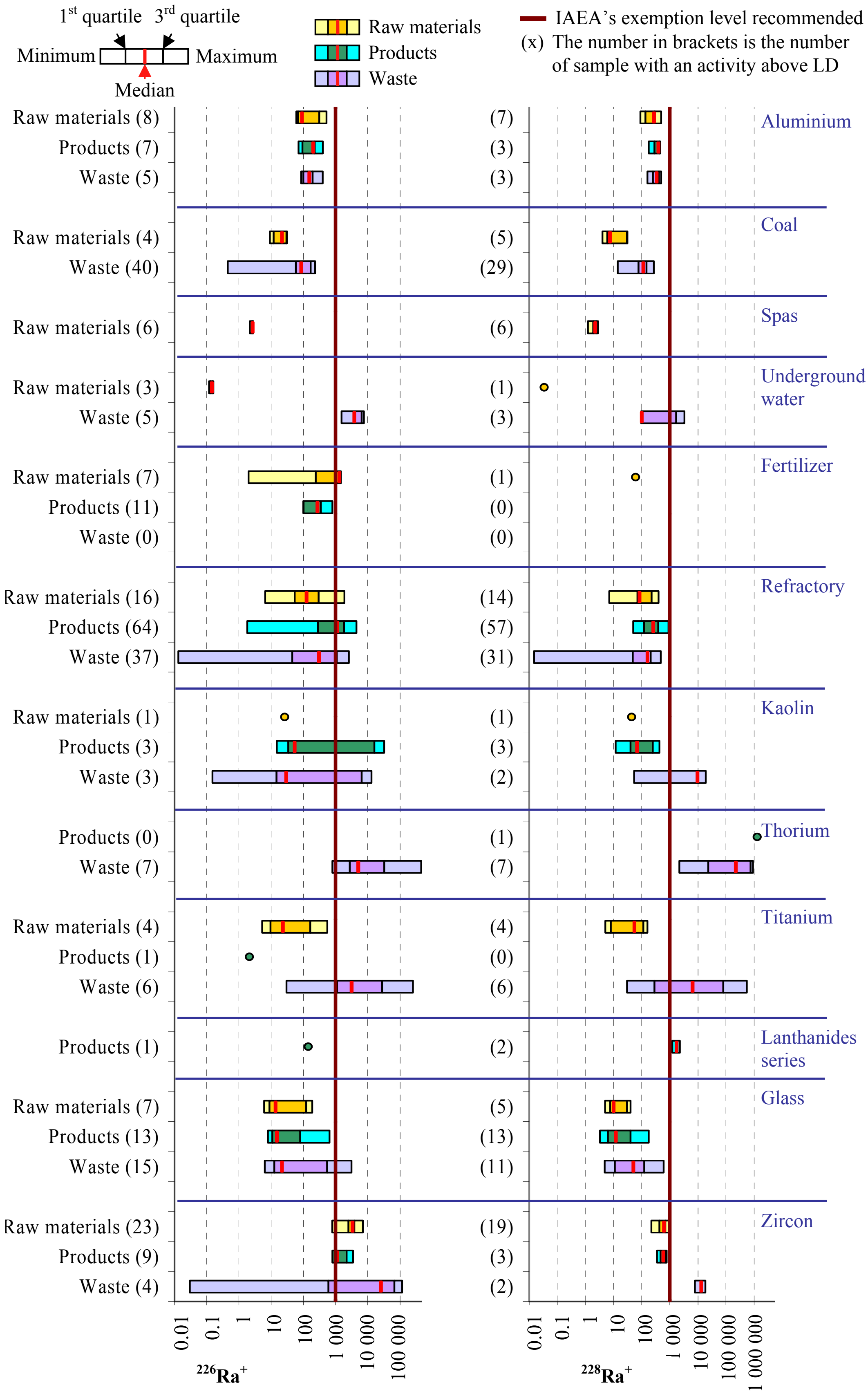

Fig. 1. Activities of $^{226}Ra^{+}$and $^{228}Ra^{+}$ (Bq.kg^{-1}).

Conclusions relative to activity levels

Based on French feedback, the following conclusions can be drawn:

- activity concentrations strongly vary according to the type of material and industrial activity;
- waste generally contain the highest activity concentration of natural radionuclides;
- 55% of materials with an activity higher than 1000 Bq.kg^{-1}, the IAEA's exemption level recommended for the use of NORM, are waste, 29% are products and only 16% are raw materials.
- the number of radionuclides measured and the accuracy of their activity determination are barely enough to assess the secular equilibrium or radioactive disequilibrium that occurs in NORM.
- imbalances have been identified: activity of ^{226}Ra^{+} group in waste (e.g. the category "underground water") or activity of ^{210}Pb^{+} group in ashes and dust related to heating processes is greater than the activities of the other groups of ^{238}U series. For example, figures 2 and 3 show two illustrations of excess of ^{210}Pb in dust from the refractory industries.

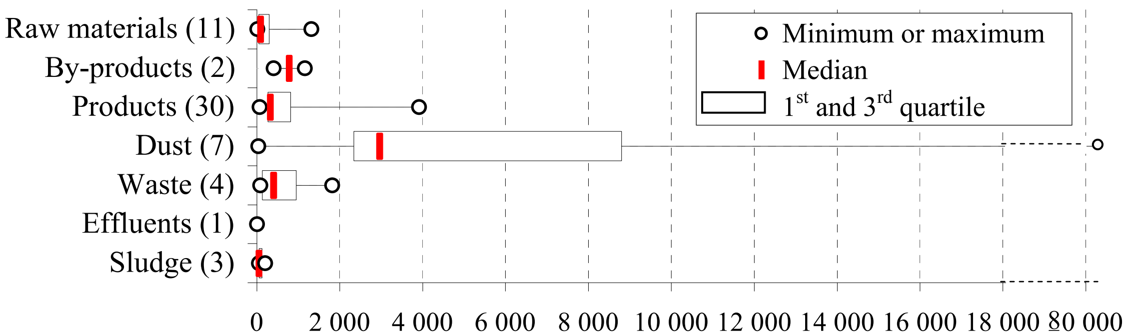

Fig. 2. Activities of ^{210}Pb^{+} in the refractory industries (Bq.kg^{-1}).

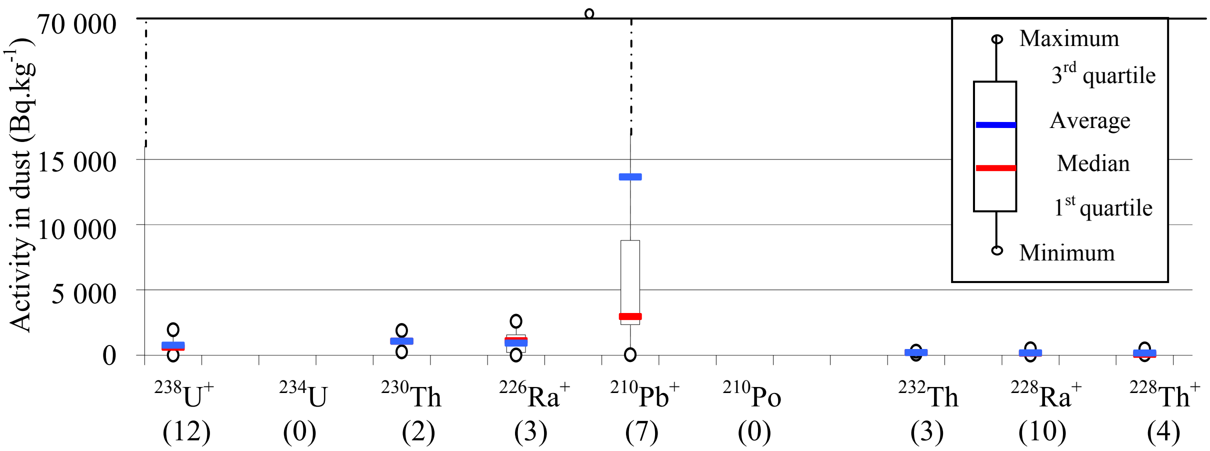

Fig. 3. Activities of ^{210}Pb^{+} in the refractory industries (Bq.kg^{-1}).

Occupational exposures

Data collected

The Ministerial order of May 25, 2005 imposes on operators to assess the occupational effective doses and to take into account external exposure, as well as internal exposure by dust inhalation and inhalation of radon progeny. Since the publication of this text, operators have assessed occupational exposure for more than 400 workplaces. A third of doses are less than 0.1 $mSv.yr^{-1}$, a half are less than 0.25 $mSv.yr^{-1}$ and 15% of doses are still greater than the effective dose limit of 1 mSv per year for the public in France (French Public Health Code, 2006) and so need further examination.

Firstly, some doses calculated by operators do not take into account external exposure or internal exposure by dust inhalation even if this route could be a significant pathway of exposure. Secondly, only some doses also consider exposure due to inhalation of radon progeny. Thirdly, only some operators take into account the exposure due to natural background in their assessment. Finally, the doses received by workers vary strongly according to the type of industrial facilities. Due to the different approach retained by operators, it is not possible to compare directly the doses reported. So, on the one hand, when data are available, exposure due to natural background was subtracted from the effective doses reported by operators. On the other hand, in order to compare the effective dose for each type of industrial activity and to identify the significant route of exposure, IRSN has analyzed the doses calculated for workers due to external and internal exposure for each type of industrial activity. Data relative to exposure to radon have been considered specifically.

Effective doses above the natural background

Production of refractory ceramics and smelting, metallurgy and glass industry using them

The production of refractory ceramics and smelting, metallurgy and glass industry using them is the most often assessed industrial activity: more than 100 workplaces have been assessed. Figure 4 shows the added effective dose. For two workplaces, doses are greater than 1 $mSv.yr^{-1}$ with a maximum of 1.5 $mSv.yr^{-1}$ but these doses are still consistent with literature (NORM V, 2008).

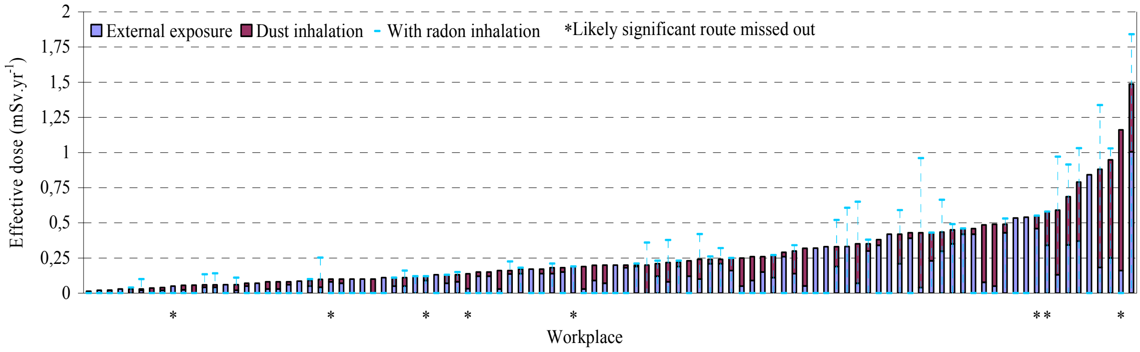

Fig. 4. Production of refractory ceramics - Added effective doses.

Production of zircon and baddeleyite, and smelting or metallurgy plants using it

The production of zircon and baddeleyite, and smelting or metallurgy plants using it is the second most often assessed industrial activity: more than 60 workplaces have been assessed. Figure 5 shows the added effective dose reported by operators. For eight workplaces, dose is greater than 1 $mSv.yr^{-1}$ with a maximum dose of 2.3 $mSv.yr^{-1}$. This dose is still consistent with results published in literature for zircone production by a thermic process (NORM V, 2008). Moreover, it is worth mentioning that the highest doses correspond to two studies in which a hypothetic and maximum time of exposure of 1600 $h.yr^{-1}$ have been considered.

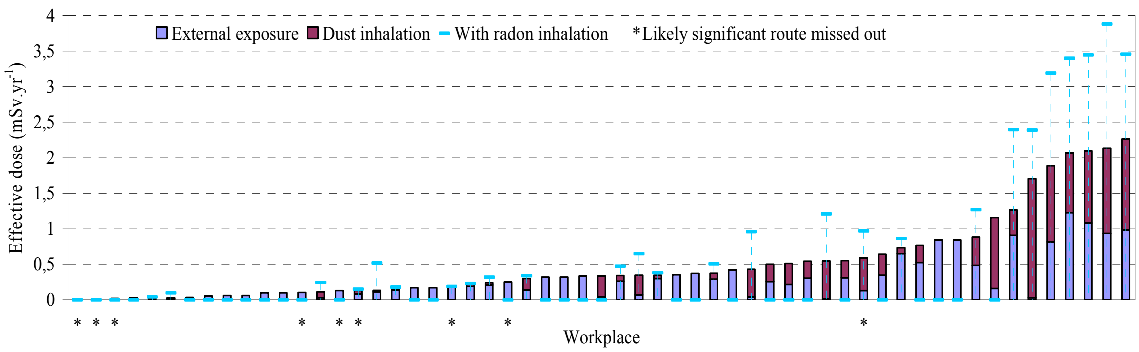

Fig. 5. Production of zircon and smelting or metallurgy plants using it - Added effective doses.

Treatment of tin, aluminium, titanium and niobium ores

The treatment of tin, aluminium, titanium and niobium ores is the third most often assessed industrial activity: more than 40 workplaces have been assessed. Figure 6 shows the added effective dose. For 13 workplaces, doses are greater than 1 $mSv.yr^{-1}$ with a maximum of 6 $mSv.yr^{-1}$ and are consistent with literature (NORM V, 2008), (UNSCEAR, 2000).

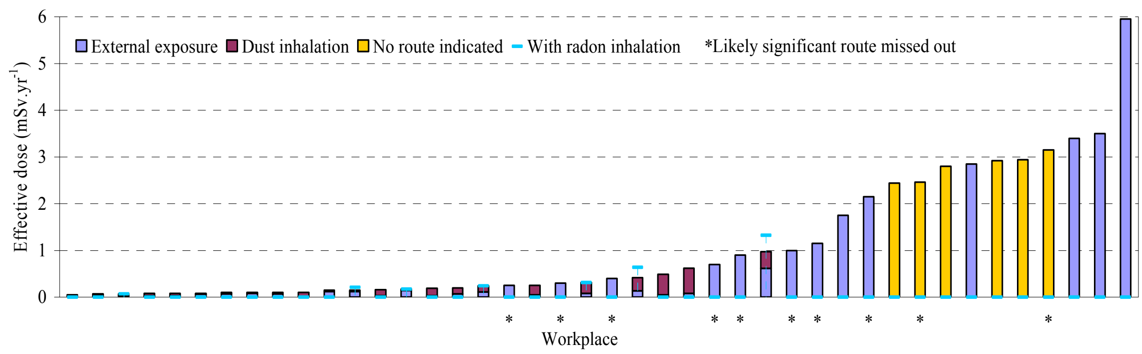

Fig. 6. Treatment of ores (Sn, Al, Ti, Nb) - Added effective doses.

It is worth mentionning that the highest doses correspond to only one study. New measurements in 2008 show an important reduction of dose rate: based on these new measurements, the maximum dose decreases to 3.2 $mSv.yr^{-1}$.

Coal combustion in power plants

30 workplaces for coal combustion in thermal power plants have been assessed. Figure 7 shows the added effective dose reported by operators. For any workplace, dose is less than 1 mSv.yr^{-1} with a maximum dose of 0.4 mSv.yr^{-1} which is consistent with literature (Smith *et al*, 2001).

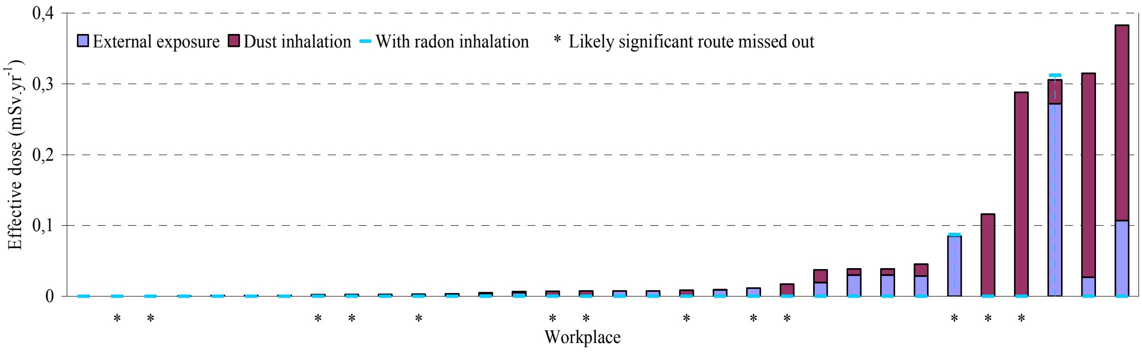

Fig. 7. Coal combustion - Added effective doses.

Other industrial activities

For the production or use of compounds with thorium, six workplaces have been assessed. For two workplaces, the doses are greater than 1 mSv.yr^{-1} with a maximum dose of 82 mSv.yr^{-1}. This dose is mainly due to dust inhalation. In order to reduce occupational exposure, the operator intended to equip workers with personal protective equipment and he should periodically clean the working planes and install a system of air filtration in his installations. These actions should significantly reduce the doses but IRSN and French authorities have not yet received the new study.

For the production of phosphated fertilizers, six workplaces have been assessed. Every doses are less than 1 mSv.yr^{-1} with a maximum dose of 0.5 mSv.yr^{-1} which is consistent with literature (NORM V, 2008), (AIEA, 2006).

For the treatment of lanthanides series, three workplaces have been assessed. Every doses are less than 1 mSv.yr^{-1} with a maximum dose of 0.3 mSv.yr^{-1} which is consistent with literature (AIEA, 2006).

Route of exposure

For only four types of industrial activities, workplaces have been sufficiently studied to allow to identify the significant route of exposure. Figure 8 shows the contributions of external and internal exposure for these four types of industrial activities.

– For coal combustion, data show clearly that external exposure is the significant route of exposure.
– For the production of refractory ceramics and smelting, metallurgy and glass industry using them and the production of zircon and baddeleyite, and smelting or metallurgy plants using it, data collected do not clearly show any significant route of exposure, though external exposure seems to be the most important pathway.
– For treatment of ores, no significant route of exposure could be identified.

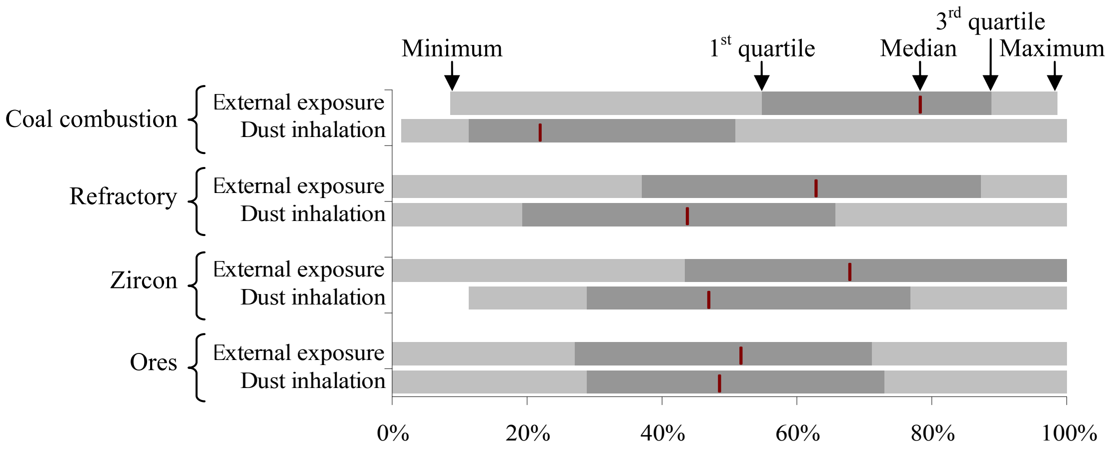

Fig. 8. Contributions of external exposure and dust inhalation.

Conclusions

Based on the feedback gained by IRSN and French authorities from the studies conducted by operators, the following conclusions can be drawn additionally to those presented in (Maigret *et al*, 2008):

– assessments are still expected for some industrial activities referred to in the Ministerial order of May 25, 2005 (Ministerial order, 2005). For example, in the next future, studies dealing with occupational exposure due to underground water treatment by filtration should be carried out;
– about 10% of added effective doses are greater than 1 mSv per year and need further examination;
– the highest doses were found in facilities which produce materials involving thorium (82 mSv.yr^{-1}) and which treat tin, aluminium, titanium and niobium ores (6 mSv.yr^{-1}). For the other types of industrial activities, the maximum dose remained below a few mSv per year;
– external and internal exposure are often of the same order of magnitude;
– some types of industrial facilities currently not referred to in the French ministerial order, e.g. paper mills, are concerned by NORM and TENORM issue and could be added to the list of industrial facilities set by the Ministerial order of May 25, 2005 (Ministerial order, 2005).

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Radiological baseline study on the planned Sokli phosphate mine in Finnish Lapland

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Abstract

Results of the radiological baseline study on the planned Sokli phosphate mine in Northern Finland are presented. From radiation protection point of view, the Sokli carbonatite massif contains considerable amounts of natural radioactive substances, i.e. thorium and uranium and their decay products. These substances are especially rich in the niobium ore. In connection with the Environmental Impact Assessment, the Radiation and Nuclear Safety Authority (STUK) carried out the radiological baseline study at the site of the planned Sokli phosphate mine and its surrounding environment. In the baseline study the activity concentrations were studied in ecosystems at the Sokli site, as well as in the immediate vicinity where the mine may have some radiological impact. The objective of the study was to obtain a detailed understanding of the radiation levels at and in the vicinity of the Sokli mining site before the commencement of mining and treatment of phosphate ore. The baseline study can be used in the future to assess possible impacts on the environment of the natural radionuclides released during the mining and milling process.

Introduction

The Sokli carbonatite massif is part the Kola alkaline province which covers an area of approximately 100,000 km^2, and is one of a few regions in the world where the paleozoic ultramafic-alkaline magmatism is well developed. A characteristic feature of the Kola carbonatites in many complexes is their close relation with phoscorites, rocks consisting essentially of apatite, magnetite and forsterite or diopside (Mi Jung Lee, et al. 2006).

Sokli is located in Savukoski municipality in the Finnish Lapland, close to the border with Russia (Fig. 1). The mineral deposit is associated with a carbonatite massif some 360 million years old, with a land surface diameter of 5–6 km. In addition to the phosphorus minerals, the carbonatite massif contains other ores, such as iron, niobium and vermiculite minerals. The Sokli carbonatite massif has been explored since 1967 and the pilot mining and milling of phosphorus minerals was performed in late 1970’s. However, in those days the commencement of phosphate ore concentrate and phosphate products in an industrial scale was not profitable. Today the plans for the Sokli mine project include opencast ore extraction and treatment, crushing and grinding of the ore, beneficiation, dewatering and storage of the products, tailings storage, water treatment and auxiliary activities. The planned production is 2.0 million tonnes of phosphorous concentrate per year. The amount of ore to be mined will be 4-10 million tonnes per year depending on the mining and milling strategy in the area. There are also plans to extract iron and other valuable minerals later. According the planned capacity, it has been estimated that the richest phosphorus ores will last for around 20 years of production. If the poorer parts of the deposit are exploited, it may be possible to extend the life time of the mine by twenty to thirty years, or even more.

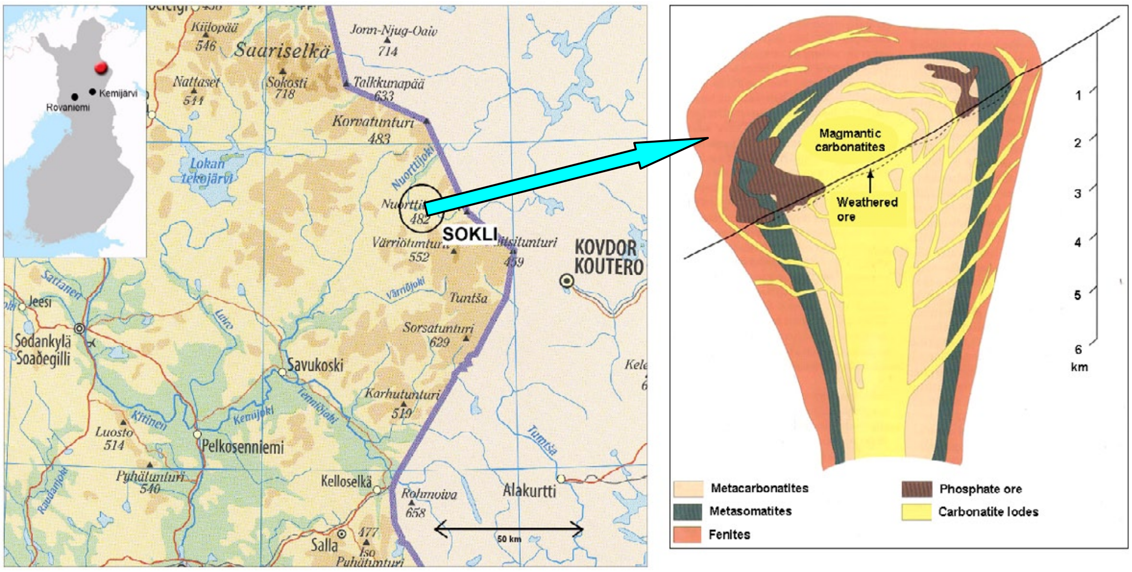

Fig. 1. Location and shape of the Sokli carbonatite massif. The magmatic carbonative ore continues tens of kilometers below the ground level.

Material and methods

The environmental samples were collected in 2008 and 2009. The samples were pre-treated and analysed at the STUK's Regional Laboratory in Northern Finland. In-situ gamma spectrometric measurements and radiochemical analyses of uranium in water samples were made by STUK's laboratories in Helsinki. Different environmental samples were collected: river water, river sediment, fishes, lichen, beard moss, water moss (*Fontinalis antipyretica*), mushrooms, berries, reindeer meat, ground water, soil, moose meat and pochard.

The environmental samples were analysed for gamma-emitting radionuclides with HPGe gamma spectrometers. The concentrations of ^{238}U, ^{235}U, ^{226}Ra, ^{228}Ra, ^{228}Th, ^{232}Th, ^{210}Pb and ^{40}K were analysed by using containers, which were inside the gas-tight aluminium bags in vacuum. The samples were kept in these bags at least three weeks before measurements so radium, radon and its short lived decay products were in equilibrium when measured. The ground water samples were also studied for ^{222}Rn.

In addition, radiochemical analyses were done to analyse activity concentrations for ^{210}Po and ^{210}Pb. All the solid samples were first dried at 105ºC and then homogenized. Suitable sample quantities were spiked with ^{209}Po tracer and digested by microwave. Polonium was deposited on silver plate and measured by alpha spectrometer (Vesterbacka and Ikäheimonen 2005). All samples were analysed before 6 months has gone from the sampling date. The solution remaining from the first deposition was stored about 6 months to allow the in-growth of ^{210}Po from ^{210}Pb. The second ^{210}Po deposition was then carried out by using ^{208}Po tracer and the ^{210}Po activity was measured by alpha spectrometer. The ^{210}Pb content of the samples was calculated and the first ^{210}Po determination was corrected for the radioactive decay between the time of sample collection and analysis and the in-growth of ^{210}Pb during sample storing before first ^{210}Po deposition was subtracted.

Results

The highest radioactivity concentrations were measured in the niobium ore and in the old mill tailings remaining at the site from the pilot enrichment in late 1970's. In the niobium ore the radioactivity concentrations were 8500 Bq/kg for ^{232}Th and 2200 Bq/kg for ^{238}U. In river sediments the activity concentrations of these radionuclides varied from few Bq/kg up to three hundred Bq/kg. In reindeer meat, game meat, berries, mushrooms and in river waters the activity concentrations remained in nearly all the samples below the detection limits of the used gamma spectrometric methods. More precise radiochemical analyses of uranium isotopes were performed for different water samples. The lowest concentrations were found in river water (0.001 - 0.024 Bq/L) and the highest concentration in spring water (0.01 - 0.02 Bq/L). The results are shown in Table 1.

^{210}Pb and ^{210}Po were found in all analysed samples. The lowest values were found in river waters and the highest values in water moss (*Fontinalis antipyretica*) which is a common aquatic plant in Lapland and known to be a good bio-indicator for heavy metals. High amounts of ^{210}Pb and ^{210}Po were found also in reindeer lichen (*Cladina rangiferina*) which is the main food plant of reindeers and in beard moss (*Bryoria*) growing on spruce and pine branches and in bolete. ^{210}Pb and ^{210}Po concentrations in different environmental media are presented in Table 2.

The activity concentrations of ^{222}Rn in ground water samples varied between 30-110 Bq/L. In river water samples radon concentrations were below detection limit 30 Bq/L.

Table 1. Concentrations of ^{234}U and ^{238}U in different water samples collected at the planned phosphate mine and its surroundings.

Water type	U-234, Bq/L		U-238, Bq/L	
	Year 2008	Year 2009	Year 2008	Year 2009
River water	0.0044 - 0.0241	0.0011 - 0.0099	0.0033 - 0.0199	0.0017 - 0.0088
Spring water	0.0189	0.0116	0.0203	0.0128
Ground water	0.0014 ja 0,0054	0.0011 ja 0.0045	0.0028 ja 0.0081	0.002 ja 0.0059

Table 2. Concentrations of ^{210}Po and ^{210}Pb in different environmental samples collected at the planned phosphate mine and its surroundings.

Sample	^{210}Po, Bq/kg	^{210}Pb, Bq/kg
Ground water	0.03 - 0.04	0.008 - 0.03
River water	0.003 - 0.01	0.0004 - 0.01
Bolete	129 - 832	1.9 - 3.4
Milk caps	15.6 - 42.9	9.9 - 21.7
Russule	6.4 - 18.3	4.1 - 6.7
Trout, grayling	2.0 - 19	1-3.6
Cloudberry	1 - 1.4	1.4 - 1.6
Lingonberry	4.6 - 4.9	2.5
Blueberry	4	2
River sediment	15 - 294	7 - 118
Water moss	182 - 1190	47 - 659
Reindeer lichen	167 - 186	89 - 96
Beard moss	302 - 472	166 - 251
Moose meat	4.3 - 4.7	0.7 - 1
Reindeer meat	8.0 - 83	0.8 - 2.5

Conclusions

The radiological baseline study in the area of the planned phosphate mine and concentration plant was performed in 2008-2009. It was considered important to investigate the natural radiological situation of the site and its environment before the commencement of mining and treatment of the phosphate ores to be able to assess possible radiological impact of these activities in the future. The Sokli carbonatite massif contains various ores and some of them contain elevated amounts of thorium and uranium. Radiological studies done before any industrial actions are the only way to verify the results of the Environmental Impact Assessment (EIA) because the studied radionuclides are occurring naturally in the environment. The local people have been worried about the radiological impact of the mining and enrichment activities to their health and the environment. It is important also to them to know the natural levels of radioactivity in different environmental media.

Preliminary results of the baseline study show that the amounts of natural radionuclides in mushrooms, berries, fish, lichens, beard moss, reindeer meat and moose meat samples were at the same level as in other environmental samples analysed from Finnish Lapland. The activity concentrations of ^{234}U, ^{238}U, ^{210}Pb and ^{210}Po in river water were close to the average concentration of Finnish tap water from waterworks 0.02. 0.015. 0.003 and 0.003 Bq/L, respectively. The highest ^{210}Pb and ^{210}Po concentrations were found in bolete, water moss, beard moss and lichen, which are good bio-indicators for radionuclides.

Radioecological investigations were performed at the Sokli area also in late 1970's when earlier plans to start phosphate production was current. The results obtained in 1970's were very similar compared with the results of this study. This indicates that the pilot production activities in 1970's had no significant radiological impact to the environment.

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Naturally occurring radionuclides in drinking water – An overview of the problem in Sweden

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Abstract

In Sweden, uranium rich soils and bedrocks can give rise to high concentrations of naturally occurring radionuclide (NORM) in groundwater. If the extracted water is then used as domestic water supply, an appreciable radiation dose can be received. Radionuclides that are of interest, originate principally from the decay series of uranium (^{238}U) and include radon (^{222}Rn), radium (^{226}Ra), polonium (^{210}Po), lead (^{210}Pb) and ^{238}U itself if the latter is present in high concentration (> 100 µg/l). A recent country wide mapping of about 700 drilled wells reveal that geological criterion cannot alone explain concentration variations of radionuclides in groundwater. 8% of the studied wells, located outside regions containing uranium rich bedrocks, had a radon concentration exceeding the recommended action level of 1000 Bq/l in water. A proper understanding of the factors guiding the mobilisation and transport of decay products of ^{238}U in the subsurface is a prerequisite to make accurate risk prediction. There are about 250 000 drilled wells that are used permanently in Sweden and between 5000 –10000 wells get constructed every year. The concentrations of radionuclides except for radon are seldom analysed. The National Board of Health and Welfare in Sweden recommends radon measurements in drinking water coming from private wells. Results of water analyses, if or when performed are not communicated to a centralised database that can be used by various national authorities but efforts are ongoing to build such a system. Knowledge regarding the extent to which private well owners check the quality of their water with regards to radionuclide and take appropriate remediation measures is poor. An ongoing survey regarding remediation measures is hoped to provide additional information on the issue.

NORM in the petroleum and geothermal industries: evolution of the French radioprotection legislation

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Abstract

Fluids produced from oil or gasfields and aquifers are naturally radioactive due to the presence of potassium-40 and isotopes from decay chains of uranium-238 and thorium-232 in these reservoirs. Thus, natural radionuclides can be carried away towards surface when petroleum or geothermal reservoirs are drilled and produced. These radionuclides can precipitate in deposits forming in process equipment, which can then be unusually radioactive, expose workers to hazardous materials and create waste disposal problems. Workers can also be exposed to high concentrations of radon. Due to their natural origin, these accumulations of radionuclides in equipment are called Naturally Occurring Radioactive Materials (NORM). They are also known as TENORM, because they are technologically enhanced.

In closed systems, exposition to gamma radiations penetrating equipment to the external surface can be reduced by restricting and controlling access to appropriate areas. But personnel may come into direct contact with NORM during maintenance and cleaning operations, when opening equipment and vessels.

In France, hygiene and safety rules concerning the protection of workers in the petroleum and geothermal industries are formulated in a specific legislation related to extraction and mining. Presently, both petroleum and geothermal industries are not included in the list of activities which have to follow radioprotection measures given by this legislation. French authorities are currently planning to update the list by including these two activities. There is a real need to realize a state-of-the art of the exposition level of French workers and to match the dose limits to those given by the Labor and Public Health Codes.

This paper will present the origin of NORM and radon contamination in the petroleum and geothermal industries and discuss the French legislation evolution for the radioprotection of personnel working in these two industries.

Ukrainian experience of monitoring of radiation exposure of population determined by building materials

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Abstract
This article contains results of monitoring of the specific radioactivity of the natural radionuclides in the minerals used as construction raw materials. In total, 22 quarries (opencast mines) of crushed stone (gravel), 10 quarries of sand and 4 of clay were studied and around 800 samples of mineral raw materials used in construction were analyzed. Effective doses of external radiation exposure of the population determined by building raw materials were defined.

It was found that the raw materials produced at 12% of the quarries do not comply with the relevant requirements of the national regulations, thus, such materials can not be used for the residential construction. Based on the assessment of the study, it was found that the average weighted effective exposure dose of the population resulting from radioactivity of the construction materials in the structure of the overall country housing is 0.23 mSv per year.

Introduction
External exposure of population to radiation in the houses is determined by naturally occurring radioactive materials contained in building materials. Studies of this source of radiation are currently done in various countries of the world, however, there is less data for radiation exposure levels inside houses than that for the outside, at the open space. This is related to the fact that there are no relevant regulations restricting exposure of population in the houses to radiation in the majority of the countries of the world. The most profound review of the results of research of such a source of radiation is quoted in the report of the United Nations Scientific Committee on the Effects of Atomic Radiation [1]. Thus, based on the Committee data, the average weighted radiation exposure dose of the population of the planet is in the range between 20 nGy to 200 nGy per hour, with the average level of 84 nGy per hour.

Furthermore, according to the Report [1], the lowest levels of radiation exposure were registered in New Zealand, Iceland and the United States, where it was lower the 40 nGy per hour level, whereas the highest (95 to 115 nGy per hour) were registered in

Hungary, China, Portugal, Austria, Spain and Iran. As a rule, the high radiation population exposure levels exist due to the use of stone and concrete with higher content of naturally occurring radionuclides in construction of housing. [2,3]. For example in Sweden lots of dwellings were constructed of light concrete before 1975. The raw material used for the production of light concrete contains large volumes of alum shale, rich in Uranium. The average dose to the dwellers of the light concrete houses is 1,1 mSv/a.

First regulations restricting the content of the naturally occurring radioactive materials (NORM) in building materials in the USSR have had come into being in 1976 [4]. This regulatory standard was developed by the specialists of the Leningrad Scientific and Research Institute for Radiation Hygiene then headed by Ye. Krisyuk. His monograph manuscript provided the scientific rationale for the methodological specifics related to substantiation of the numerical values of Radium-226, Thorioum-332 and Potassium-40 content in construction materials. These standards [norms] remain the same to date and are applied according to Norms of Radiation Safety of Ukraine, thus, it is worthwhile to look at those in greater detail.

Here is the main idea behind such standard. The value of the standard [norm] is no more than 370 Bq/kg of the specific effective radioactivity (C_{eff}) of the naturally occurring radionuclides (Radium-226, Thorioum-332 and Potassium-40) contained in construction materials. This was 4 times more the relevant average value at the territory of the USSR (93 Bq/kg). Expected excess of the standard throughout the whole territory of the country was around 1%. Effective external radiation exposure dose in the houses built with the use of construction materials with NORM value of $C_{eff} = 370$ Bq/kg is 1.75 mSv per year for the period of being inside a house of 7000 hours per year [6].

Construction materials with the higher content of NORM are permitted to be used in road construction (up to 740 Bq/kg within the limits of towns and other settlements and between 740 Bq/kg and 1850 Bq/kg beyoned such limits) [7].

Monitoring of the NORM content of construction materials and mineral raw materials used in construction is mandatory in Ukraine throughout the whole territory. The article includes the analysis of the study results for 95% of all quarries in Ukraine where construction materials and mineral raw materials used in construction are produced. Based on data, resulting from the direct studies, effective radiation exposure doses assessment was made for the population of different regions of the country.

Material and methods

Measurement method

Spectrometry method involving the use of ORTEK (US) high resolution gamma spectrometry system with NaI (T1) scintillation detector and well was applied in the framework of this article in order to do the measurements of mineral construction raw materials samples.

Measured samples were put inside Marinelli container, that fill the space of the well with the detector crystal completely. With this placement of samples the geometrical effectiveness of registration is substantially higher than when cylinder form crystals are used.

Large volume well, 150 mm diameter, 150 mm height, was used to provide for sufficient sensitivity. AMERSHAM (Germany) standard source was used for calibration of the gamma spectrometer.

For geometry 'Marinelli 1 litre' the value of the lowest detectable radioactivity was 2 Bq/kg for Thorium-232, 3 Bq/kg for Radium-226 and 10 Bq/kg for Potassium-40.

Effective specific radioactivity of a sample (A_{eff}) was calculated based on the values of specific radioactivity of the NORM, Thorium-232, Radium-226 and Potassium-40.

$$A_{eff} = A_{Ra} + 1{,}31 \cdot A_{Th} + 0{,}085 \cdot A_K,$$

Where A_{Ra}, A_{Th}, A_K – values of specific radioactivity of Radium-226, Thorium-232 and Potassium-40, Bq/kg.

Method dose calculation

Transition coefficients from air-absorbed dose to effective dose and parameters for time of humans being enclosed inside house, quoted in attachments to the United Nations Scientific Committee on the Effects of Atomic Radiation report, were used to assess the average annual effective dose values [1]. Transition coefficient from air-absorbed dose to effective dose was set at the value equal to 0,7 Sv/Gy whereas factor for length of time of being inside housing was 0.8 [1].

The following data was used in relation to the country overall housing structure to do the assessment of the average weighted population effective radiation exposure doses (obtained from Radon measurement certificates from IHME (Institute of Hygiene and Medical Ecology) databases: buildings currently in operation (28 000 buildings) and building in process to be commissioned (34 000 premises). The certificate contains information about materials, used to construct a building, year of construction, number of residents in a building as well as that on the number of floors, coverings of the walls, type of dividers between floor and space under floor, availability of basements and ventilation system in buildings. Such information makes that possible to do adjustments when calculating values of the effective doses of radiation exposure of population considering specifics of architectural and layout solutions pertinent to individual regions in Ukraine.

For instance, housing structure of the rural areas of the Kiev Oblast comprises 60% of the building made of bricks, 33% are wooden buildings, 2% - adobe brick buildings and around 1% was build with the use of shell limestone and slag stone.

Effective exposure doses were calculated separately for residents of the brick-made and concrete-plate buildings in order to assess average annual radiation exposure doses of the urban population, and then the resultant value was weighted against the housing structure.

Results

Results of measurements of the NORM content in samples of mineral construction raw materials from 22 crushed stone, 10 sand and 4 clay quarries were analyzed to assess relevant effective exposure doses.

It was found in the analysis that the average weighted specific radioactivity $\bar{A}_{eff}$ for selected quarries (580 samples) was 238 Bq/kg with standard deviation of 128 Bq/kg.

Relevant value for 10 sand quarries (80 samples) was 127 Bq/kg with the standard deviation 309 Bq/kg, and for 4 clay quarries (120 samples) it was 79 Bq/kg and 14 Bq/kg accordingly. Frequency distribution of the effective specific radioactivity values of the NORM was of the lognormal character.

Average value A_{eff} for granite quarries usually does not exceed 370 Bq/kg and corresponds to Class I for mineral construction raw materials set out in NRBU-97 (Norms of Radiation Safety of Ukraine) (para 8.5.1.). Findings of the study have shown that A_{eff} for the two crushed stone quarries (9% of the total number of all measured samples) exceeds that value, in other words, the mineral materials from such quarries can not be used for the residential construction. At present, such mineral materials are used for industrial and road construction.

According to the study, mineral raw materials produced at only 12% of all quarries of the country can not be used for the purposes of residential construction.

Tables 1-3 indicate data on NORM content in the mineral construction raw materials produced in separate regions of Ukraine.

Table 1. Average Values of NORM Specific Radioactivity Samples of Crushed Stone from Quarries in Ukraine.

Oblast (Region)	Specific Activity, Bq/kg		
	Radium-226	Thorium-232	Potassium-40
Vinnitsya	32	120	917
Dnipropetrovsk	143	420	1032
Zhytomir	45	72	1250
Kiev	42	131	1351
Kirovohrad	55	120	1217
Rivne	44	45	840
Sumy	99	69	1246
Khmelnitsky	44	52	976

Table 2. Average Values of NOR Specific Radioactivity Samples of Sand from Quarries in Ukraine.

Oblast (Region)	Specific Radioactivity, Bq/kg		
	Radium-226	Thorium-232	Potassium-40
Zhytomir	16	34	270
Kiev	6	7	88
Volyn	22	13	157
Rivne	28	33	401
Chernihiv	13	24	495

Table 3. Average Values of NORM Specific Radioactivity Samples of Clay from Quarries in Ukraine.

Oblast (Region)	Specific Radioactivity, Bq/kg		
	Radium-226	Thorium-232	Potassium-40
Dnipropetrovsk	33	43	447
Zhytomir	13	27	255
Kiev	21	33	481

Radiation exposure doses

External radiation exposure doses of Ukraine's population are calculated based on results of the NORM specific radioactivity measurements of mineral construction materials samples, depending on the housing structure of regions and type of construction material. Table 4 contains results of effective radiation exposure doses calculation for population of Kiev Oblast as an example.

Table 4. External Radiation Exposure Doses of Population Determined by NORM Contained in Building Materials.

Raw Material	Raw material use Coeff.	Absorbed Dose for Weight Fractions for Raw Materials in Buiding Materials wm, nGy/hour				ED with wm = 0,5, mSv/a
		wm = 1	wm = 0,75	wm = 0,5	wm = 0,25	
Crushed stone	2,09	168	126	84	42	0,41
Clay	0,67	54	40	27	13	0,13
Sand	0,14	12	9	6	3	0,03

Table 5 illustrates main results of the assessment of external radiation exposure doses to population determined by construction materials. It was found that average annual radiation exposure dose for the urban population of Ukraine (around 70% of the total population of the country) is in the range of 0,28 – 0,33 mSv/a and that such doses are practically no different for residents of both brick-made and concrete-plate-made buildings. Only the mineral construction materials of the Class 1 were used to construct such buildings.

Table 5. Effective Radiation Exposure Doses of Population of Selected Oblasts of Ukraine Determined by Radioactivity of Construction Materials, mSv/a.

Oblast	Urban Population	Rural Population
Vinnitsya	0,31	0,10
Volyn	0,29	0,09
Dnipropetrovsk	0,29	0,30
Zhytomir	0,27	0,30
Zaporizhzhia	0,27	0,11
Kiev	0,33	0,20
Odesa	0,31	0,12
Poltava	0,31	0,09
Rivne	0,22	0,07
Kherson	0,28	0,09
Cherkassy	0,31	0,21

Effective radiation exposure doses may vary by several times for rural population depending on a specific region. For example, most of the dwellers in Volyn and Rivne Oblast live in buildings made of wood, thus, effective dose for the residents of the above Oblasts is less than 1 mSv per year. In Dnipropetrovsk, Kiev or Zhytomir Oblast houses are, as a rule, made of bricks, therefore, effective radiation exposure dose for population there is 2-3 times higher.

Thus, average weighted effective external radiation exposure dose for the population of Ukraine in the context of the country housing structure is 0.23 mSv/a.

It is still worthwhile to note out that in the separate instances, and, in particular, with regard to residents of the old concrete-panel type buildings, constructed back in 1950-1960 (prior to implementation of restrictions on the use of construction materials) individual effective radiation exposure doses may be much greater. According to our assessment, in such instances the maximum external radiation exposure dose is 1-1.5 mSv per year.

Such doses may be even higher than that in some separate cases in the rural areas. This is related to the use of the high NORM content waste produced by the local industrial facilities as additional material utilized to underlay foundation of a building. So, individual privately owned houses in Dniprodzerzhinsk is an example, in construction of which process waste of uranium milling was used. Individual radiation exposure doses of residents of such houses may be several times higher than average throughout the whole country.

Some outstanding cases have no impact on the assessment of the average radiation exposure doses, however give rise to serious concerns on the part of the local public when such cases are identified.

In summary of the results it can be stated that the existing mandatory radiological monitoring of the NOR in mineral construction materials and raw materials has made impossible the use [of non-compliant materials].

Conclusions

1. Effective specific radioactivity $\bar{A}_{eff}$ of crushed stone is 238 Bq/kg for 22 analyzed quarries. Relevant value for 10 sand quarries was 127 Bq/kg and for 4 clay quarries was 79 Bq/kg.
2. Construction raw materials produced at 12% of the studied quarries can not be used for residential construction as effective specific radioactivity of the naturally occurring radionuclides in the relevant samples exceeds 370 Bq/kg.
3. Average weighted effective radiation exposure dose determined by radioactivity of construction materials for Ukraine’s population with regard to the country’s overall housing structure is 0,23 mSv/year.
4. Contribution of that component in the average annual radiation exposure dose of the country’s population is 2% approximately.
5. Restrictions with regard to content of natural radionuclides in the mineral construction raw materials and materials used in residential construction have practically excluded that component from the category of sources that impact the value of radiation exposure risks for the population of the country.

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The International Association of Oil & Gas Producers (OGP) naturally occurring radioactive material management (NORM) guidelines

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Abstract

It has been established that Naturally Occurring Radioactive Materials (NORM) may accumulate in the oil/gas production process. This can create a radiation hazard for workers, the general public and the environment if adequate controls are not established. Regulations and guidelines on radiation protection in general and NORM in particular have been issued by various National and International entities, however these are not specific to the Oil & Gas industry, and do not provide clear and concise guidance to enable NORM to be managed effectively. The International Association of Oil and Gas Producers (OGP) established a task force, headed by Saudi Aramco, to develop NORM Management Guidelines. This task was completed after research, review and thorough consideration of most available NORM regulations, and the industry's current best practices in NORM management. The OGP guidelines offer a simple and logical method of managing NORM impacted operations that are flexible enough to be implemented across the industry. This paper outlines major aspects of OGP NORM management guidelines and elaborates on key issues related to the management of NORM in the Oil & Gas Industry, in particular on NORM monitoring, control of NORM contaminated equipment, managing NORM waste handling and disposal, and worker protection, awareness and training.

Technologically enhanced NORM and heavy metals in iron and steel industry

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Abstract

Iron and steel industry was ranked as the largest industrial source of toxic environmental contamination in USA. About 2 – 4 tones of various solid wastes (slag, sludge, dusts and scales) are generated per ton of steel produced. These wastes contain a notable concentration of Naturally Occurring Radioactive Materials – NORM and heavy elements that could be a source of environmental contamination and occupational exposure. Composite samples of different iron and steel industry's wastes were collected from four iron and steel factories. Natural radionuclides (^{238}U, ^{232}Th, ^{40}K, ^{210}Pb, ^{210}Po) and trace elements (e.g. Cd, Cu, Pb and Zn) were measured using gamma-ray spectrometry, alpha particle spectrometry and ICP-MS analytical techniques. There is a wide range of variation in the concentration of natural radionuclides and other elements that depends on their concentration in the process input materials and thermal treatment process. Occupational dose due to dusts inhalation was calculated. According to the assumed scenario, the occupational exposure is much lower than the reference dose limit. The environmental impact due to wastes storage and/or usage should be considered generally and case by case.

NORM and trace elements fractionation in phosphate rock beneficiation processes: potential hazards and useful applications

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Abstract

Beneficiation processes (mainly crushing, washing, magnetic separation and wet screening) of phosphate ore rock aims at increasing the phosphate content percentage. It starts with the ore rock and end with wet rock, and different rejects or processes by-products. By-products have potential hazards due to their content of Naturally Occurring Radioactive Materials – NORM especially uranium-238 series and heavy metals, e.g, Cd, Cu, Pb and Zn. They have also potential industrial and agricultural useful applications due its physical and chemical properties such as their relative high content of clay and iron. Representative samples of ore and wet phosphate rocks and beneficiation processes by-products (clay and dolomite rocks, wet screening, magnetic separation and slim) were collected. Natural radionuclides (^{238}U, ^{232}Th, ^{40}K, ^{210}Pb, ^{210}Po) and trace elements (e.g. Cd, Cu, Pb and Zn) were measured using gamma-ray spectrometry, alpha particle spectrometry and ICP-MS analytical techniques. Potential hazards due to beneficiation processes and their by-products were discussed from both the radiological and chemical point of views. Radiologically, internal hazards index, external hazard index, representative level index, gamma absorbed dose rate and occupational dose equivalent due to inhalation were calculated. Chemically, input of toxic heavy metals (radioactive or stable) to the environment due to by-products dump and/or usage were estimated and their hazardous aspects were discussed. Some of the by-products could have potential useful applications such as clay rock and slim for agricultural soil reclamation. The aspects of these useful applications were discussed.

NORM in clay deposits

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Abstract

Clay minerals are among the most important minerals used by manufacturing and environmental industries. To reduce the occupational and environmental hazardous impacts of clays' mining, handling and applications, it is essential to investigate their radiological as well as physicochemical characteristics. Twenty one clay's deposit samples from three different regions (Al-Riyadh, Al-Kharge and Jeddah) were collected. Activity concentrations of naturally occurring radioactive materials (NORM), ^{226}Ra, ^{228}Ra and ^{40}K in Bq/kg dry weight were measured using a well-calibrated gamma-ray spectrometer. Radiation dose (e.g. absorbed dose rate- nGy/h and effective dose equivalent- µSv/y), radium equivalent (Raeq) value in Bq/kg and hazardous indices (e.g. external hazardous, internal hazardous and representative gamma level) due to natural radionuclides in clay samples were calculated. The variation of the average activity concentration of natural radionuclides (NOR) according to sampling region could be due to the origin of the geological formation and the geochemical behavior of the NOR. The average activity concentrations of NOR in Saudis' clay samples were much lower than their average concentrations in kaolin clays utilized in Egypt that imported from different countries. Based on calculated hazardous indices (external hazardous and internal hazardous), there are not hazardous effects of clay deposits utilization as a building material. This does not correct based on the average value of representative gamma level that was greater than unity. More detailed studies should be done to consider the occupational exposure due to clay mining and handling.

Introduction

Clay is a common name for a number of fine-grained, earthy materials. Chemically, clays are hydrous aluminium silicates, ordinarily containing impurities, e.g., potassium, sodium, calcium, magnesium, or iron, in small amounts. Clay deposits vary in the chemical and physical properties and mineral composition, especially in terms of the type of clay minerals dominant. Properties of clay minerals can be determined the field of the use in industry or agriculture. The most common minerals of clay are kaolinite, montmorillonite - smectite, illite, chlorite, attapulgite or palygroskite, feldspar and calcite. Clay deposits that contain high concentrations of smectite mineral are using for agriculture purposes, because they have desirable properties in agriculture such as a tendency to expand ,disperse, swelling - shrinking behavior as well as high cohesion, also they have a high cation exchange capacity (CEC). While clay deposits that contain kaolinite mineral are using for industrial purposes, because it has less plasticity, cohesion, CEC, and swelling than most other clay minerals [Sheta et al., 2006].

The application of natural clay deposits locally available in Saudi Arabia as soil amended material, soil conditioner, to improve the sandy soil physical condition such as relative swelling, cumulative infiltration, and water conservation were studied. It was concluded that using local abundance clay deposits could improve the predominantly sandy soils in the Middle Eastern countries [Sheta et al., 2006].

Radiation dose due to gamma emitter naturally occurring radionuclides represents the main external source of irradiation of the human body. More specifically, natural environmental radioactivity and the associated external exposure due to gamma radiation depend primarily on the geological and geographical conditions, and appear at different levels in each region in the world [UNSCEAR, 2000].

From the environmental point of view; it is essential to estimate the concentrations of the naturally occurring radioactive materials (NORM) in the clay deposits to ensure their radiological safety for their handling and applications. This study aims at measuring NOR activity concentration in clay samples from three different regions in Saudi Arabia, estimating the radiation doses and possible hazardous impacts of their applications.

Material and methods

Sampling and samples preparation:

Twenty one clay deposit samples were collected from, three Saudis' regions, Al-Riyadh, Al-Kharg and Jiddah. The selection of the sites was based on previous geological studies [Laurent, 1993]. Sampling was carried out by a scientific group from the soil sciences department, college of food sciences and agriculture, King Saud University. Samples were collected using the standard methods to get a composite sample that represents each site. Samples were dried in an open air of a dry place, mechanically crushed, and sieved through a 2 mm mesh sieve [Sheta et al., 2006].

Gamma-ray spectrometry:

The dried samples were transferred to polyethylene containers of 100 cm3 capacity and sealed at least for 4 weeks to reach secular equilibrium between radium and thorium, and their progenies. ^{226}Ra (^{238}U) series, ^{232}Th series, ^{40}K, 137Cs and 210Pb specific activities were measured using well-calibrated gamma spectrometry based on hyper-pure germanium (HpGe) detectors. The HpGe detector had a relative efficiency of ^{40}% and full width at half maximum (FWHM) of 1.95 keV for 60Co gamma energy line at 1332 keV. The gamma transmissions used for activity calculations are 352.9 (214Pb), 609.3, 1120.3 and 1764.5 keV (214Bi) for ^{226}Ra (^{238}U) series, 338.4, 911.1 and 968.9 keV (^{228}Ac) for ^{232}Th series, 1460.7 keV for ^{40}K, 661.6 keV for 137Cs and 46.5 keV for ^{210}Pb. The gamma spectrometers were calibrated using both ^{226}Ra point source and potassium chloride standard solutions in the same geometry as the samples (Khater, 2001).

Theoretical calculations:

Radium equivalent (Raeq) index in Bq/kg is a widely used radiological hazard index and a convenient index to compare the specific activities of samples containing different concentrations of ^{226}Ra, ^{232}Th (^{228}Ra) and ^{40}K. It is defined based on the assumption that 10 Bq/kg of ^{226}Ra, 7 Bq/kg of ^{232}Th and 130 Bq /kg of ^{40}K produce the same gamma dose rate. It is calculated using the following equation [Beretka and Mathew, 1985]:

$$Raeq = C_{Ra} + 1.43\ C_{Th} + 0.007\ C_K$$

Where; C_{Ra}, C_{Th} and C_k are the activity concentrations of ^{226}Ra, ^{232}Th and ^{40}K in Bq/kg, respectively.

The absorbed dose rates due to γ-ray the air at 1m above the ground surface for the uniform distribution of the naturally occurring radionuclides (^{226}Ra, ^{232}Th and ^{40}K) were calculated based on guidelines provided by UNSCEAR (1993, 2000). The conversion factors used to compute absorbed γ -dose rate (D) in air per unit activity concentration in Bq per kg (dry weight) corresponds to 0.462 nGy h−1 for ^{226}Ra (of U-series), 0.621 Gy h−1 for ^{232}Th and 0.0417 nGy h−1 for ^{40}K [UNSCEAR, 2000 & 1993].

$$D = 0.461\ C_{Ra} + 0.623\ C_{Th} + 0.0414\ C_K$$

To estimate the annual effective dose rates, the conversion coefficient from absorbed dose in the air to effective dose (0.7SvGy−1) and outdoor occupancy factor (0.2) proposed by UNSCEAR (2000) are used. Therefore, the effective dose rate in mSv.y−1 was calculated by the following formula [UNSCEAR, 2000]:

Effective dose rate ($\mu Sv.y^{-1}$) = Dose rate ($nGy.h^{-1}$) X 24 h X 365.25 d X 0.2 (occupancy factor) X 0.7 $Sv.Gy^{-1}$ (conversion coefficient) X 10^{-3}

According to ICRP (1977) the upper limit of radiation dose arising from building materials is 1.5 $mSv.y^{-1}$ [ICRP, 1977]. For limiting the radiation dose to this value, Krieger (1981) proposed the following conservative model based on infinitely thick walls without windows and doors to serve as a criterion for the calculation of external hazard index Hex - defined as [Krieger, 1981]:

$$H_{ex} = \frac{C_{Ra}}{740} + \frac{C_{Th}}{520} + \frac{C_K}{9620} \leq 1$$

Hewamanna et al. (2001) corrected this model after considering a finite thickness of walls and the existence of windows and doors. Taking these considerations into account, the equation used for the calculation of external hazard index becomes:

$$H_{ex} = \frac{C_{Ra}}{740} + \frac{C_{Th}}{520} + \frac{C_K}{9620} \leq 1$$

The value of this index must be less than unity for the radiation hazard to be negligible, i.e. the radiation exposure due to radioactivity in construction materials must be limited to 1.5 $mSv.y^{-1}$.

In addition to the external irradiation, radon and its short-lived products are also hazardous to the respiratory organs. The internal hazard index (Hin) is used to control the internal exposure to 222Rn and its radioactive progeny. The internal exposure to radon and its daughter products is quantified by the internal hazard index (Hin) which is given by the following equation [Krieger, 1981];

$$H_{in} = \frac{C_{Ra}}{185} + \frac{C_{Th}}{259} + \frac{C_K}{4810} \leq 1$$

For the safe use of a material in the construction of dwellings Hin should be less than unity.

Another radiation hazard index called the representative level index, Iγ, is defined from the following formula [NEA-OECD, 1979; Alam et al., 1999];

$$\mathrm{I}\gamma = \frac{C_{Ra}}{150} + \frac{C_{Th}}{100} + \frac{C_K}{1500} \leq 1$$

The safety value for this index is ≤ 1.

Results and discussion

The statistical summary of activity concentration of naturally occurring radionuclides (NOR), i.e, ^{226}Ra, ^{228}Ra and ^{40}K, and radium equivalent value (Raeq) in Bq/kg dry weight of clay deposit samples was given in table 1,and shown in figures 1 and 2.
There are a noticeable differences in the average activity concentration of ^{226}Ra, ^{228}Ra and ^{40}K, in clay samples from different regions (I:Al-Riyadh, II: Al-Kharg and III: Jeddah). The lowest average activity concentration of both ^{226}Ra and ^{40}K, and the highest average activity concentration of ^{228}Ra were found for Jeddah region samples. Average Raeq values for the three regions samples were comparable and less than 370 Bq/kg. That is equivalent to the maximum permissible limit for indwelling radiation dose due to NOR in building materials.

Table 1. statistical summary of activity concentration of ^{226}Ra, ^{228}Ra and ^{40}K, and radium equivalent value (Ra-Eq) in Bq/kg dry weight of clay deposit samples

Reg.	^{226}Ra	^{228}Ra	^{40}K	Ra -Eq
I	50 ± 3, 7 (45-59),6	28 ± 3 , 8 (16-35), 6	823 ± 143, 350 (298^{-1}181), 6	151 ± 17, ^{40} (92-200)
II	54 ± 11, 31 (11^{-1}08),8	39 ± 5.17, 15 (26-69),8	781 ± 143, ^{40}5 (262^{-1}387),8	169 ± 17, 48 (72-200)
III	29 ± 7, 14 (10-42),4	62 ± 13, 26 (34-85),4	583 ± 104, 209 (356- 768),4	162 ± 33, 66 (86-222)
ALL	47 ± 6, 23 (10^{-1}08),18	^{40} ± 5, 20 (16- 85), 18	751 ± 82, 347 (262- 1387),18	162 ± 11, 48 (72- 222)

*Average ± standard error, Standard deviation (range), No. of measured samples. I:Al-Riyadh, II: Al-Kharg and III: Jeddah

Average activity concentrations of ^{226}Ra, ^{228}Ra and ^{40}K, and Raeq value for clay samples were 47, ^{40} and 751, and 162 Bq/kg, respectively. These values were within the world activity concentration range of ^{226}Ra, ^{228}Ra and ^{40}K in soil; 10-50, 10-50, 100-700 Bq/kg, respectively [UNSCEAR 1988]. There is a trend for the NOR activity concentrations in the different regions where the maximum activity concentration of ^{228}Ra was found in region III while the minimum average activity concentration of both ^{226}Ra and ^{40}K were in region III. This trend is not clear for Raeq value because it is dependent on the activity concentration of the three radionuclides with weighting factor of 1, 1.43 and 0.07 for Ra, Ra and K, respectively.

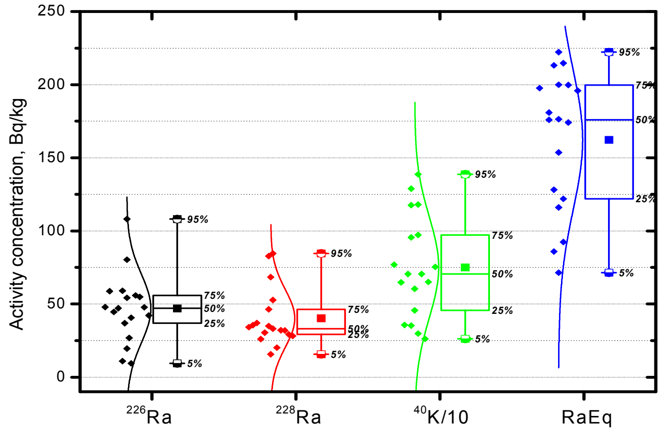

Fig. 1. Box chart of activity concentration of ^{226}Ra, ^{228}Ra and ^{40}K, and radium equivalent value (Ra-Eq) in Bq/kg dry weight of clay deposit samples.

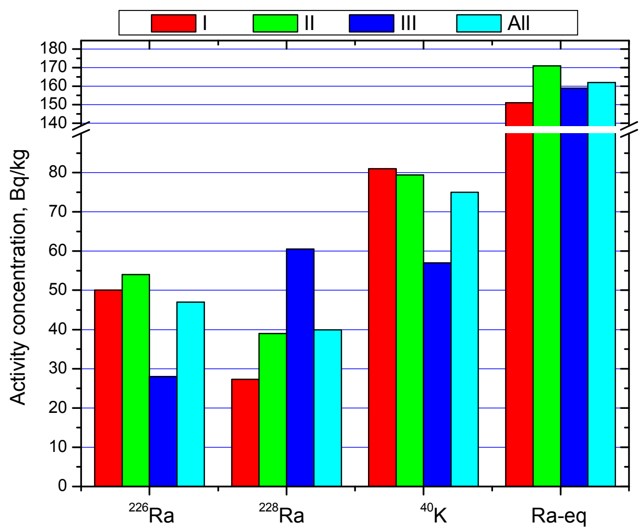

Fig. 2. activity concentration of ^{226}Ra, ^{228}Ra and ^{40}K, and radium equivalent value (Ra-Eq) in Bq/kg dry weight of clay deposit samples according to sampling region (I:Al-Riyadh, II: Al-Kharg and III: Jeddah).

Walley El-Dine et al., 2004, studied the activity concentration of NOR in local and imported kaolin (china clay) types used in Egypt. Kaolin is widely used in paper industry, ceramics, refractory bricks, white cement, textiles, rubber, medical industries, and special types of plastics. Their results show an obvious wide range of variation in ^{226}Ra, ^{228}Ra and ^{40}K activity concentrations, and radium equivalent values (Ra-Eq) that had mean values (ranges) of 965 (48-8633), 252 (96-1079), 59 (8-270) and 1329 (188-10185) Bq/kg, respectively [Walley El-Dine et al., 2004].

Clay is a widely distributed, abundant mineral resource of major industrial importance for an enormous variety of uses. Table 2 shows the average activity concentrations of ^{226}Ra, ^{228}Ra, ^{40}K and Raeq in clay bricks as building materials from

12 different countries [Hewamannaa et al., 2001]. The results of Saudi's clay samples are less than the average activity concentrations for ^{226}Ra, ^{228}Ra and Raeq, and slightly higher than the average activity concentration for ^{40}K.

According to IAEA (international Atomic Energy Agency) activity concentration values for exclusion, exemption and clearance (1000 Bq/kg for ^{40}K and 1000 Bq/kg for any other NOR), it is not recommended that the exposure dose be investigated for the studied clay deposits. Due to the limited number of studied samples and the possible variation in NOR activity concentration, it is recommended to measurement of NOR activity concentrations for more clay deposit samples from the same and other geological formation[IAEA, 2004].

Table 2. Comparison of activity concentrations and radium equivalents- Raeq (Bq.kg^{-1}) in clay bricks in different areas of the world [Hewamannaa et al.; 2001]

Country	No. of samples	^{226}Ra	^{232}Th (^{228}Ra)	^{40}K	Raeq
Australia	25	41	89	681	220
China	n.m.[a]	41	52	717	171
Egypt	1	20	14	204	56
Finland	33	78	62	962	241
Germany	109	59	67	673	207
Greece	6	49	24	670	135
India	1	48	52	381	152
Netherlands	14	39	41	560	141
Norway	6	104	62	1058	276
Sweden	n.m.	96	127	962	352
Sri Lanka	24	35	72	585	183
Saudi Arabia*	18	47	^{40}	751	162
Average± stand. Error (range)	-	55±7 (20^{-1}04)	59±9 (14^{-1}27)	684±70 (204^{-1}058)	191±22 (56-352)

[a] n.m. indicates that the number of samples was not mentioned in the published work. *Clay deposit samples
& Mean ± Stand. Error (range)

Clays and clay minerals occur under a fairly limited range of geologic conditions. The environments of formation include soil horizons, continental and marine sediments, geothermal fields, volcanic deposits, and weathering rock formations. Most clay minerals were formed where rocks are in contact with water, air, or steam. Examples of these situations include weathering boulders on a hillside, sediments on sea or lake bottoms, deeply buried sediments containing pore water, and rocks in contact with water heated by magma (molten rock). All of these environments may cause the formation of clay minerals from preexisting minerals. Extensive alteration of rocks to clay minerals can produce relatively pure clay deposits that are of economic interest (for example, bentonite, primarily montmorillonite, used for drilling mud and clays used in ceramics) [USGS, 1999]. The activity concentration and the environmental behavior of natural occurring radionuclides (NOR) in the geosphere depend on many parameters such as their geochemical properties. Radium (Ra) is an alkaline earth element, and can exist in nature only in the +2 oxidation state. In the pH range of 3 to 10, the uncomplexed ion Ra2+ is the dominant aqueous species for dissolved radium in natural waters. In sulfate- containing waters, precipitation and redissolution of calcium (Ca), strontium (Sr), and barium (Ba) sulfates, rather than adsorption/desorption, could control the concentrations of dissolved radium in the soil. Precipitation of radium is readily possible as the solid-solution solids (Ba, Ra)SO4 and (metal, Ra)CO3 in water where the concentration of dissolved sulfate and carbonate, respectively, are sufficient high. The adsorption behavior of radium will be very similar to that of strontium. Relative to other alkaline earth elements, radium is the most strongly sorbed by ion exchange on clay minerals. The adsorption of radium is strongly dependent on ionic strength and concentrations of other competing ions that adsorption of radium decrease with increasing ionic strength. Radium is also strongly adsorbed in mineral oxides present in soil, especially at near neutral and alkaline pH conditions. The results of some studies also suggest that radium may be strongly adsorbed by organic material in soils. [EPA, 2004].

The physical characteristics of clay deposits, as one type of sedimentary rock, are governed by many factors of which some include parent rock material, mode of formation of the minerals, the means and distance of transport, and the depositional environment. These characteristics together with uranium, thorium and radium content of the parent rock, and recent physical and chemical events (i.e. chemical leaching, transport with water, and precipitation/adsorption) can affect the final distribution of radium [Edsfeldt, 2001].

Neither radium itself, radium salts, radium carbonates, nor radium oxides are very soluble. However, radium solubility is enhanced by alpha recoil. During the decay of a radionuclide by alpha emission, alpha particles are rejected from the nucleus, carrying off most of the excess energy. The created progeny recoils in the opposite direction. Thus, uranium deposits can kick radium compounds into interstitial pore water due to alpha recoil process [Cothern and Rebers, 1991].

To evaluate the radiation hazardous due to the natural radionuclides in different clay deposits and their various applications, absorbed dose (D) in $nGy.h^{-1}$, effective dose rate (E) in $\mu Sv.y^{-1}$, radioactivity level index ($I\gamma$), external hazard index (H_{ex}) and internal hazard index (H_{in}) were calculated, Table 3. Calculations of both D and E were considered to estimate the radiation dose due to gamma ray emitter in clay samples. Calculated values are much lower than their value due to natural background (2.4 mSv/y) [UNSCEAR 1993].

Table 3. statistical* summary of calculated radium equivalent in Bq/kg dry weight of clay deposits sample, absorbed dose (D) in $nGy.h^{-1}$, radioactivity level index ($I\gamma$), external hazard index (H_{ex}), internal hazard index(H_{in}) and effective dose rate (E) in $\mu Sv.y^{-1}$

	D	I_γ	H_{ex}	H_{in}	E
I	75 ± , 21 (44-98)	1.16 ± 0.14, 0.33 (0.67^{-1}.53)	0.41 ± 0.05, 0.11 (0.25-0.54)	0.55 ± 0.05, 0.12 (0.38-0.70)	91 ± 10, 26 (54- 120)
II	81 ± 9, 24 (33.49^{-1}06)	1.26±0.13, 0.37 (0.53^{-1}.66)	0.46 ± 0.05, 0.13 (0.19-0.58)	0.610 ± 0.07, 0.20 (0.22-0.83)	100 ± 11, 30 (41^{-1}30)
III	76 ± 6, 31 (^{40}- 104)	1.20 ± 0.24, 0.48 (0.64^{-1}.63)	0.43 ± .08, 0.17 (0.23-0.60)	0.52 ± 0.10, 0.22 (0.24-0.71)	93 ± 19, 38 (49^{-1}28)
ALL	78 ± 6, 23 (33^{-1}06)	1.21 ± 0.09, 0.37 (0.53- 1.66)	0.44 ± 0.03, 0.13 (0.19-0.60)	0.56 ± 0.04, 0.17 (0.22-0.83)	95 ± 7, 29 (41^{-1}30)

Average ± standard error, Standard deviation (range

Other calculated indices considered when clay deposits use as a building material where their values should be less than unity. Both internal and external hazardous indices were less than unity that indicates their safe utilization as building material. While, the representative level index values in about 65% of the studied samples and the average value were higher than unity that indicates their unsafe utilization as building material.

Conclusions

- Average concentration of natural radionuclides (NOR) in clay deposit samples from different regions were varied. That could be due to the different of the geological formation origin and the geochemical behavior of the NOR. Variation of both Ra and K average concentrations were similar (I> II> III) while that of Ra was reversed (III> II> I). Radium equivalent values were comparable in the three regions.
- The activity concentrations of natural radionuclides (NOR) in the studies clay deposit samples are higher than the world average of soil.
- The average activity concentrations of NOR in clay samples were much lower than the NOR concentrations in kaolin clays that imported from different countries and industrially used in Egypt.
- The average activity concentrations of NOR in clay samples were fall in the low range of NOR concentration in clays blocks from different countries that were used as a building material.
- Based on calculated external and internal radiation hazardous indices, the studied clay samples could be utilized safely as a building material.

Acknowledgement

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Distribution pattern of NORM on Red Sea shore sediments in relation to non-nuclear industries

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Abstract

The Red Sea is a deep semi-enclosed and narrow basin that has an intensive non-industrial activities on and near its shore. Oil exploration, phosphate mining and trading, navigation activities and intensive touristic activities are consider as non-nuclear pollution sources that could impose a serious radiological and ecological impacts on the Red Sea marine environment. Both oil and phosphate related activities could increase the concentration of Naturally Occurring Radioactive Materials – NORM such as uranium-238 series, thorium-232 series and potassium-40. Forty representative shore sediment samples were collected from the Egyptian Red Sea shore, from Shuqeir to Marsa Alam City region. Activity concentration of ^{238}U, ^{232}Th, ^{40}K, ^{210}Pb and ^{210}Po were measured using gamma-ray spectrometry, alpha particle spectrometry and ICP-MS analytical techniques. Previous study showed the possible impact of industrial activities on the activity concentration of NORM in shore sediment. This study will investigate such relation and the distribution pattern of NORM in more details.

NORM and heavy metals partitioning during water treatment processes

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Abstract

Water samples were collected from underground water purification plant to study the variation in Naturally Occurring Radioactive Material (NORM) e.g., U, Th and K, and heavy metals e.g., Cd, Pb, Hg, Cu, and Zn concentration through the treatment processes and its relation to physical and chemical properties of water. Samples represent the different treatment processes (input, output, after filtration, sludge tank, reverse osmosis permit and reject, and waste water ponds). NORM and heavy metals concentration in the collected samples were measured using ICP-MS. Their concentrations in water samples show a wide range of variation that depend mainly the water treatment processes and the chemical properties of water samples. Water physical and chemical properties, i.e. pH, EC, major cations (Ca, Mg and K) and major anions (CO_3, HCO_3, Cl and SO_4) were determined. The effect of water treatment processes on NORM concentration; dose assessment due to water drinking (before and after treatment) and the radio-ecological risk assessment were discussed.

Soil-to-plant transfer factors of ^{210}Pb and ^{210}Po in boreal forests

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Abstract

The general goal in this project was to obtain overview on ^{210}Po and ^{210}Pb behaviour and mobility in the environment. Binding and mobility of ^{210}Po and ^{210}Pb in forest soil and transfer of polonium and lead from soil to plants were considered. The main study areas were located in Scots pine forests in southern Finland (62º9'N, 22º52'E) and in northern Finland (66º21'N, 26º44'E). The soil samples were collected and separated into different soil horizons (litter, organic and mineral soil layers). Vertical distribution and concentrations of ^{210}Pb and ^{210}Po in soils were determined. Wild berry samples and edible mushrooms were collected at the sampling sites. Activity concentrations of ^{210}Po and ^{210}Pb were analysed from roots and rhizomes of the berry samples as well as from berries (i.e. fruits), leaves and stems separately. Mushrooms were divided into caps and stipes and each part was analysed separately. The mean ^{210}Po/^{210}Pb ratio was 0.9 in the organic soil layer in the southern and the northern Finland sites. The activity ratios of ^{210}Po/^{210}Pb in the wild berry and mushroom samples were mainly higher than one, indicating elevated concentrations of polonium in the samples. In mushrooms the concentrations of ^{210}Pb and ^{210}Po were higher than in fruits of the wild berries. Soil-to-plant transfer factors for lead and polonium will be discussed. The research results gained in this project will enable an assessment of the mobility of ^{210}Po and ^{210}Pb in the environment and in the food chains and estimation of ensuing radiation doses to humans.

Radiological assay techniques associated with TENORM industry

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Abstract

In recent years there has been an increasing awareness of the radiological impact of non-nuclear industries that extract and/or process ores and minerals containing naturally occurring radioactive materials (NORM). Actions were undertaken to find and characterize the main industrial activities involving NORM and to assess the impact of these activities on the nearby public. The main concern was on past activities in phosphate fertilizers industry and the associated by-products disposal as well as the activities in oil and gas production and processing industry. Phosphogypsum is the main by-product generated during the phosphoric acid production process (wet process), and phosphate slag is the principal by-product generated from the production of elemental phosphorus (thermal process). During the wet process, a selective separation and concentration occurs for radionuclides which are naturally associated with the phosphate ores. In oil and gas production and processing equipment some elevated TENORM contamination might occur due to the physical processes to which the extraction fluids are exposed. Some radium and radium daughter compounds are slightly soluble in water and may become mobilized when this production water is brought to the surface. The paper presents the methodology used in some particular cases to deal with NORM characterization as a regulatory requirement for non-nuclear practitioners to improve their radiological safety and to achieve an adequate level of knowledge as concerns the risks associated with NORM industry.

NORM management in the oil & gas industry – Saudi Aramco experience

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Abstract

It has been established that Naturally Occurring Radioactive Materials (NORM) may accumulate at various locations along the oil/gas production process. Components such as wellheads, separation vessels, pumps, and other processing equipment can become NORM contaminated, and NORM can accumulate in the form of sludge, scale, scrapings and other waste media. This can create a potential radiation hazard to workers, general public and the environment if certain controls are not established. Saudi Aramco has developed NORM management guidelines and is implementing a comprehensive strategy to address all aspects of NORM management which aim towards enhancing: NORM monitoring, Control of NORM contaminated equipment, Control of NORM waste and disposal, Workers protection, awareness, and training The benefits of shared knowledge, best practice and, experience across the oil & gas industry are seen as key to the establishment of common guidance. This paper presentation outlines Saudi Aramco's experience in the development of a NORM management strategy and its goals of establishing common guidance throughout the oil and gas industry.

The needs and feasibility of land reclamation of areas affected by enhanced natural radioactivity

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Abstract

The major efforts devoted to disposal of radioactive materials are focused on those originating from nuclear industry. Far less attention has been paid to environmental burden of waste with natural radioactivity enhanced by non-nuclear industry. Such waste differs significantly from classical nuclear ones and the radiation risk is often associated with risk caused by other pollutants. Contrary to the nuclear waste that's strictly controlled, it has been a common practice to put TENORM *(Technologically Enhanced Naturally Occurring Radioactive Material)* into heaps, where they can reach thousands of cube meters or tonnes, without any protection. Exposure to meteorological conditions sets some chemical or physical processes in motion, leading to the selective transfer and accumulation of radionuclides and disequilibrium in decay series.

As a result of the inconclusive European law, that left the decision where risk caused by TENORM is significant under each member state's competences, non-nuclear industry is hardly ever aware of environmental problems caused by natural radioactivity or expect negative consequences in case of implementing radiation protection measures. This results in the substantial underestimation of the detrimental effects to the environment originating from TENORM. EU member states apart each other try to regulate this problem case by case when but the risk caused by TENORM is rarely taken into account when the treatment of such waste is planned.

The feasibility of different methods of land reclamation was discussed including hard land reclamation, bioremediation and phytotechnology. An approach based on dilution of TENORM with inert materials was considered. Contrary to the nuclear waste, where dilution is totally forbidden, natural radioactivity, which is present in small amounts elsewhere, is a case where such approach to decrease risk sounds rationally. In extraction industry where huge amount of gangue is present, such approach is well-founded also from economic point of view.

Introduction

Natural radioactivity is ubiquitous in human environment. According to the state-of-the-art radioprotection, radiation emitted by primordial radionuclides in their natural state that has not been altered due to human activity is not considered to be a source of harmful effects neither for human beings nor environment. There are many areas in the world having elevated so called "natural background" caused either by the geological and geochemical structure of the rocks, or by the radioactive content of water flowing from underground springs. Whether or not it can cause a negative or positive effect on human beings is a matter of opinion. But if concentrations of natural radionuclides have been changed as deliberate or accidental action carried out by human being it is quite another matter. The classical case where radiation risk caused by natural radioactivity is not negligible is uranium mining and milling. It is abundantly clear that such processes must be carried out at the region where reach uranium ore occurs. But such activity is considered to be an immanent part of nuclear industry to be enclosed in radiation protection domain at the very beginning. After the enhanced natural radioactivity had been thoroughly studied in other industries it became clear that such phenomena are very frequently present in the anthropogenic environment. Many processes beyond nuclear industry lead to a situation where the activity concentration of naturally occurring radionuclides is enhanced. Such alteration of natural state can result in increased radiation risk to the people as well as to environment and non-human biota. Hence, the monitoring and prevention of occupational radiation risks caused by enhanced natural radioactivity has become obligatory in many cases of industry of concern.

Enhanced natural radioactivity is usually associated with industrial processes where a significant mass reduction of raw materials occurs. As a matter of fact, these processes are not aimed at production of natural radionuclides or the deliberate use of radiation. Therefore, radioactive nuclides are often accumulated in useless industrial residues. From the point of view of the general principles of radiation protection, the activity concentration of natural radionuclides in such materials is sometimes high enough to rank them as radioactive waste. Their amount collected in one site frequently riches hundreds of thousands of cubic meters or tonnes. For instance, in coal mining industry radium activity deposited in single tailing ponds may reach as high as 300 GBq (Michalik et al. 2005). Probably, the biggest "producer" of waste with enhanced concentration of natural radionuclides are phosphate processing plants where radionuclides remain associated with the phosphogypsum particles, being subsequently stored in a disposal sites located in the vicinity of the factories with surface dose rate reaching 350 MBq/h (Bolivar et al. 2009). In spite of that TENORM-type waste (TTW) is often deposited directly into the environment, what is strictly forbidden in case of "real" radioactive waste. Contrary to the occupational risk, which is more or less controlled, by far, less attention has been paid to the environmental burden emerging due to TENORM-type waste (TTW). This results in sites where such waste have been dumped not to undergone adequate land reclamation and often even monitoring of radioactive pollution is not carried out. There are only few examples where TENORM type waste are treated in correct way (Welbergen and Wiegers 2008). There are significant number of cases where only non-radiological parameters are effectively taken into account during presumptive land reclamation, mainly due to the lack of proper regulation.

Environmental burden of TENORM residues

There is a lot of data dealing with the behaviour of natural radionuclides being in the natural state on the border of abiotic and biotic matter. Frequently, the processes of metabolism lead to concentration of some long-lived natural radionuclides in particular tissues of fungi, plants as well as animals (McDonald et al. 1996). Derived committed dose can be higher than doses resulted from artificial radionuclides accumulated simultaneously (Aarkrog et al. 1997). There are some examples of societies based on limited trophic chain where the related committed dose to individuals caused by biologically accumulated natural radionuclides such as polonium and lead ingestion is significant. J. Van Oostdam et al. (2005) indicated ^{210}Po derived doses as high as 10 mSv per year for some aboriginal northern communities consuming large amounts of caribou. Either, annual doses reaching 3 mSv were reported for population of fisherman just living on seafood (Alonso-Hernandez et al. 2002), (Camplin et all. 1996). If such processes are going on in unchanged environment, one can easily image what is going to happen in the vicinity waste dump where residues containing enhanced concentration of natural radionuclides had been collected.

Each particular occurrence of TTW presents a unique scenario of exposure – usually different from those caused by artificial radionuclides present in radioactive waste or spent nuclear fuel. As a result of the direct contact with environment, some transformation processes such as mobilisation of radionuclide species from solid phases or interactions of mobile and reactive radionuclide species with components in soils and sediments may be set in motion (Vandenhove and Van Hees 2007). Also considerable transfer of radionuclides to biota can be observed (Soudek et al. 2007). All these result in that the original distribution of radionuclides deposited in environment can change over time. Moreover, natural radionuclides are often associated with other pollutants as heavy metals or hydrocarbons that can escalate negative impact on environment if dumped out of plants. That's why, to plan the land reclamation of a site affected by TTW, information on radionuclide species deposited, interactions within affected site either local ecosystems or environment compartment and the varying in time distribution of radionuclide species influencing mobility and biological uptake is essential (Tamponnet et al. 2008).

TENORM-type waste characterization

TTW produced by non-nuclear industrial activities such as mineral production, mining activities or coal fired power plants contain a number of long-lived natural occurring radionuclides from the uranium and thorium decay series. The main, from radiation protection point of view, properties of TTW which make them significantly different from "classical" radioactive waste are (Michalik 2007);

- the occurrence in bulk quantities deposited directly in the environment,
- the wide variety of radionuclides speciation and different minerals content,
- the possible coexistence of other pollutants as heavy metals, sulphates, hydrocarbons

Taking into consideration the radionuclides that occur in TTW and either their activity concentration or total activity, some part of them should be classified as radioactive waste containing alpha emitters, (i.e. the limit of activity concentration for radium isotopes is 10 kBq/kg). Actually, the decision to rate TTW among the radioactive waste at all, not only as alpha emitters is rarely taken. It results in technical problems and economical consequences that would follow such decision (to remind, such waste must be sealed and deposited in a repository, in case of alpha emitters in deep underground repository). After they are classified as radioactive waste TTW would generate enormous cost- related implications and fill available repository very quickly. There is no possibility to treat them as such with regard to existing regulation, which had been prepared when thinking about nuclear waste. Finally, there are gaps, no special regulation and no correct way of deposition. Moreover, from environmental point of view, very interesting is, what one should do with waste that contain slightly lower than limit radioactivity content, in case of radium, for example, 9 kBq/kg. Expected effects on environment caused by waste with 10 and 9 kBq/kg of radium should not be significantly different. Actually, the proposed clearance level for natural radionuclides from uranium and thorium decay series have been set at the level of 1 kBq/kg. So, even if one wants to follow the existing regulation quickly will meet the big gap between 1 and 10 kBq/kg.

After deposition TTW in environment, enhanced concentration of natural radionuclides first of all, results in enhanced exposure of biota to external gamma radiation. So, accurate measure of the total radionuclides concentration in these waste materials is crucial to asses the potential radiological risk at a dump site. However, if one managed to gather data about physical presence of each particular radionuclide, this information would give only a part of the knowledge necessary to evaluate its harmful potential. The radiological hazards can be increased by migration process of mobile fraction of these radionuclides to the vicinity of a depository (Sheppard et al. 2005). Being released into an ecosystem, they can enhance the gamma radiation doses to biota. That is the reason why it is crucial to know the mobility of radionuclides (Martinez-Aguirre et al. 1995). The mobility and environmental behaviour of every element depend on their speciation in certain waste material (Zhongwen et al. 2002). The speciation of a radionuclide is generally related to its physical and chemical forms existing, that is, simple and complex ions in interstitial solution, exchangeable ions associated with waste material organic fractions occluded or co-precipitated with metal oxides, carbonates, sulphates and other secondary minerals.

But the exposure to external gamma radiation is only a tip of an iceberg. One should remember that in thorium and uranium decay series there are 7 and 12 alpha particles respectively (additional 12 are in actinium series). Even when one consider decay series starting from radium (^{226}Ra or ^{228}Ra) what is very common in TTW waste the number of alphas decreases only to 7 and 9, respectively. It is hardly ever taken into consideration but in environment, in case of plants, especially plant roots, the exposure to external alpha radiation is as important as exposure to alpha radiation emitted by incorporated emitters. Also one should remember about betas emitted by natural radionuclides. Actually the weighting factors for alpha and beta radiation established for human being can not be directly applied for plants but there are no rational reasons why they should be significantly lower. Direct measurements showed that absorbed dose resulted from alpha radiation can reach the same level as doses from gamma radiation (Michalik 2008).

Besides the problems caused by activity concentration, radionuclides speciation and migration, the evolution of the related risk must be taken into consideration. Namely, the most common radionuclides responsible for risk creation are radium isotopes. In case where source of TTW are formation waters (oil and gas industry, underground mining) radium isotopes, both from uranium and thorium series, ^{226}Ra and ^{228}Ra are dominant. It means that, significant disequilibrium in natural decay series exists. At the beginning, just after deposition, there are almost pure radium as the only contaminant (besides possible other, non radioactive ones, of course). Radium isotopes are usually weakly mobile and not bio-available in environment. As it was proven radium creates not soluble radium-barium sulphate or its atoms are strongly bonded with fine clay minerals. The bio-available part usually does not exceed 1 % of total activity (Leopold et al. 2007). So, the environmental risk is limited only to the exposure to external gamma radiation, or as it was shown above, alpha radiation and finally could be partially limited by a cover from an inert material. Moreover, the part of exposure derived from radium ^{228}Ra will relatively quickly decreases (the half live of ^{228}Ra is 5.7 years, so not supported by long-lived parent radionuclides ^{228}Ra will quickly disappear). The only difficult situation is that, in the decay series started with not-supported radium ^{228}Ra, the activity concentration of decay products i.e. ^{224}Ra can exceed the activity concentration of radium after few years. It can cause troubles during measurements and appropriate actual dose assessment.

Quite different situation exists in case of radium ^{226}Ra. It decays slowly. Not only from the point of view of a human being but also form point of view of the environment it lasts eternally. 1600 years is long enough to be eye witness of changes going on in an ecosystem.

As it was already mentioned, the bio-availability of radium is very weak, but such comfortable situation with immobilised radium does not last long. Taking into consideration the relationships inside the uranium decay chain one should expect not so quickly, but permanent growth of long lived radium daughters as lead ^{210}Pb and polonium ^{210}Po. Both these radionuclides are very well known as easy migrating nuclides in environment and available to biota. After decay, the nuclide of radium is released from the barium-sulphate or clay mineral cage. Moreover, between radium and lead and polonium, there is radon (or thoron respectively) occurring in gaseous form. So, after one hundred years one should expect in the site where TTW with radium had been dumped exactly the same activity of easy migrating and easy bio-available lead and polonium isotopes. Besides specific chemical properties, they are beta and alpha emitters respectively and both of them are chemiotoxic elements.

In summary, the proper land reclamation of sites affected by TTW is an interesting challenge.

Remediation trials

Among noticed approaches to the reclamation of land affected by TTW, the simplest one, compliant with radiation protection rules, is to treat such waste literally as radioactive ones and apply all required restrictions. But for many reasons, it is applicable in limited cases (Michalik 2008,2). Usually, only parts of dewatering systems from crude oil and natural gas industry (Varskog 2007) or gangue from mineral sand processing (Hutchinson and Toussaint 1998) are disposed to especially prepared repositories. Sediments had been created in surface tailings sometimes are treated in special way (Al-Masri and Suman 2003) but usually, they are left without any action. Some promising efforts to wash out radium from oil sludge were made in Egypt (Afifi et al. 2009). Also applications of biotechnology, in order to make radium more mobile from uranium milling waste were noticed (Muñoz et al. 1995). But any of them have been applied in technical scale. In uranium mining and milling industry, where implementation of strict rules of radiation protection has long history, usually a kind of hard remediation is applied. Land-filing and covering with an inert matter is sufficient to limit the exposure to external radiation as well as further radon exhalation (Krizman 1995), (Juhász et al. 2001). But this approach is not sufficient enough to stop radium and further polonium migration.

The possibilities of application of different methods of land reclamation have been considered towards the sites contaminated by radium rich sediments originating from underground coal mining (Chałupnik et al. 2001), (Michalik 2004). The first based on application of phytotechnology was tested on an abandoned mining settling pond. Six-year-lasting observation of contaminated area let one noticed that the process of natural plants transgression was so effective and, even without any support, good enough to stop the physical propagation of contamination. The plants overgrowing the pond created a tight cover able to stop water and air land erosion. It supposes that, in case of controlled and supported propagation of selected plant species, it would be actually an effective and cheap method for immediate land reclamation of contaminated with radium sites.

The possibility of phytoextraction at this area was evaluated for two plant species *Cirsium vulgare* and *Calamagrostis epigejos*. The balance of radium ^{226}Ra in plant and sediment at the tested areas showed that one can expect only less than 0.01% of total amount of radium will be extracted during one vegetation season (Michalik et al. 2009). So it is by far too small for effective application of this method. Moreover, the experiment was done based on sediments in which radium was mainly adsorbed at clay minerals. Other experiments, done on sediments with radium-barium sulphate, a hardly soluble mineral, let one to expect even lower results. So, such approach does not solve the problem at all.

The recommended and usually followed approach to utilisation of sediments that had been gathered in underground workings is to put them into old galleries considered never been used again. From the radiation protection point of view such approach is optimal for safe disposal of sediments from surface settling ponds too. However, from technical point of view a lot of obstacles exist as limited capacity of empty underground spaces or distance from settling pond to the nearest shaft in use.

Finally, an approach based on dilution of radium waste was taken into consideration. In general, dilution of radioactive waste is totally forbidden. But in case of natural radioactivity, which is present in small amounts elsewhere, such approach to decrease derived risk sounds rationally. Especially in case of hard coal mining, where technological process creates fovorable circumstances to do it in economic way. During coal exploitation process, at least, half of mine spoil is waste rock and gangue. The radium activity concentration in thse waste do not differ significantly from average value taken as background for earth crust. So that it would be a good "solvent" for radioactive sediment. Actually, mechanical mixing of huge amount of mineral waste is complicated and expensive but again, coal exploitation process provides an opportunity to do it as a "by the way". Namely, all excavated matter must pass through a coal treatment plant in order to clean coal from gangue and prepare expected fraction of coal by flotation. The total amount of flotation and coal cleaning waste is big enough to turn back radium activity concentration in sediments to background level after homogenous mixing.

The possibility to apply such approach was tested in a mine. The total amount of mine spoil: coal and all types of created waste were balanced against total amount of radioactive sediments gathered in water galleries at all mine levels. It have been done year-by-year since 2005 (table 1). Codes were given based on rules of European waste catalogue (Michalik 2009).

Table 1. The balance of the excavation process.

year	Mine productivity (t):					
	total	coal	Waste from coal cleaning process code: 010102	Waste from flotation code: 010481	Sediment from water galleries code: 190899	Sum of : 010202 and 010481
2005	6895262	3674000	3050585	170677	1740	3221262
2006	6760336	3703900	2894547	161889	1620	3056436
2007	7056571	3737600	3163933	155038	3668	3318971

The current and archive data concerning radium activity concentration in every kind of considered materials were used (table 2). Because there are no data about behaviour of such sediment during coal enrichment process, all possibilities were taken into account, it means: total amount of radium accumulated in particular kind of waste have been considered separately (table 3).

Table 2. Basic statistics of radioactivity in sediments and gangue.

	code: 190899		code: 010202 & 010481	
	Ra-226	Ra-228	Ra-226	Ra-228
	Bq/kg			
average	705,6	364,4	79,7	73,1
median	409,0	246,0	73,0	79,0
Minimum	21,2	19,0	24	10
Maximum	8272,0	2880,0	189	112
Number of samples	39		15	

Table 3. Radium activity concentration in mixed waste.

nuclide	year	190899 + 010202 + 010481		190899 + 010102		190899 + 010481	
		maximum	*average*	*maximum*	*average*	*maximum*	*average*
Ra-226 [Bq/kg]	2005	193,36	80,07	193,61	80,09	270,57	86,05
	2006	193,28	80,06	193,52	80,08	269,08	85,93
	2007	197,92	80,42	198,36	80,45	375,81	94,19
Ra-228 [Bq/kg]	2005	113,49	73,22	113,58	73,23	139,93	76,01
	2006	113,47	73,22	113,55	73,23	139,42	75,95
	2007	115,06	73,39	115,21	73,40	175,97	79,80

Obtained results seem promising. The average activity concentrations in all end-products of the process are slightly increased in comparison to their original value. Moreover, considering the worst case scenario, it means the total amount of radium remains in the flotation waste, the smallest contributor to the total mass of concern, the activity concentration does not differ significantly from other ones and there are no limitations in their disposal at surface mine spoil bank and further use as i.e. aggregate.

Conclusion

In the light of different approaches to remediation of areas affected by waste with enhanced concentration of natural radionuclides, the dilution method with inert material or waste originating from industry of concern seems to be well justified from technical and economical point of view. The example from coal mining industry shows, based on the balance of waste rock and gangue produced by every mine, that there are enough capabilities to use this technology for safe disposals of radium-rich sediments that had been gathered in surface settling pond due to either former or current mining activity. However, such approach needs to be approved by appropriate regulation.

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Uranium and heavy metals in narghile (shisha, hookah) moassel

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Abstract

Cigarette smoke is a source of some trace and heavy metals but also the smoking mixtures used in narghile. Some cigarette filters retain significant quantities of cadmium, lead, magnesium, iron, and other metals. However, data on narghile (hookah, shisha, water-pipe) are scarce. So, the objective of the present study was to investigate the contents of the main and widely used smoking product: moassel. Ten representative samples of 3 different moassel brands were collected from the local market of Cairo city-Egypt and Riyadh City- Saudi Arabia. Uranium and heavy metals (e.g., Cd, Cu, Pb and Zn) were assessed using ICP-MS. Among the 10 representative samples of the 3 different moassel brands, The results indicate the existence of a wide range of variations in uranium and other element concentrations that could be because of non standard manufacture processes. The concentration levels were compared to the results of other studies and that of cigarette tobacco. Our study shows that, as far as trace elements are concerned, harm can be reduced. Public health officials could include in the national prevention plans the use of smokeless tobacco, particularly when addressing heavy narghile smokers. However, it must be clear that there are important differences between smokeless products. On one hand, some of them, like the Sudanese "tumbak", contain high levels of carcinogenic nitrosamines and we fear that Arabian shamma might be of a similar nature. On the other hand, a moist snuff like the Swedish SNUS, is, in the view of prominent international experts, highly recommendable all the more that it is also very low in carcinogenic substances. We therefore encourage its use all the more that this harm reduction product is culturally adapted to the Arab world context and other similarly sanitary and socio-cultural ones.

External gamma radiation produced by materials: proposal of an evaluation model. Application case study NORM

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Abstract
External gamma exposure to natural radiations is due to a series of isotopes gamma emitting, contained in all building materials. Beginning from the set up of a calculation model to estimate the emission in a closed environment, it was possible to derive the effective dose in the environment. The case of materials generally rich of natural radioisotopes was investigated.

From an inverse process, starting from the examination of the radiation field it is possible to estimate the extension of an activated surface. A case of contamination and activation surface in a general context as the inner part of nuclear plant or a radioactive facilities was evaluated. This method, moreover, is also applicable to NORM.

Introduction
The emission of total gamma radiation of simple or variegated structural configurations does not represent a simple modeling phenomenon, since the factors associated with emission, absorption and reflection, principally due to radiation-material interaction, generate a series of secondary events, the precise estimation of which inevitably risks complicating the problem.

Emitted radiation comes into being, due to various processes, involved in a series of intermediary interaction, then to be definitively absorbed. A global balance can be created considering the equation.

$$\text{Incident Radiation Energy} \rightarrow \text{Converted} + \text{Absorbed Radiation Energy} \qquad (1)$$

At this stage, it becomes inevitable to refer to the “medium”, or rather to the space-material reference in which the radiation is to be found.
Introducing a medium in which radiation is propagated, has the substantial aim, not just of identifying parameters such as propagation speed or refraction index in the medium, but of verifying if such a medium is of influence in terms of radiation intensity reduction, and therefore of absorption - bremsstrahlung of the same.

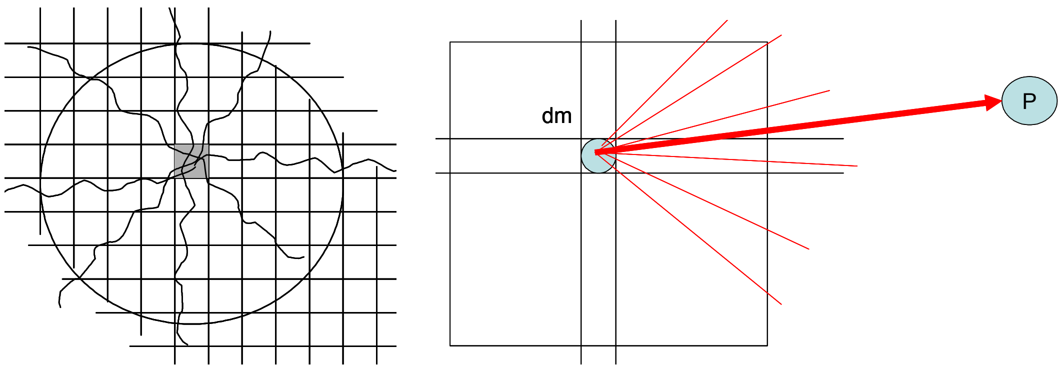

Fig. 1. Finite radiant element in the material matrix (left), and activity produced by a finite element on a P external point.

The study conductible on the emission of a finite source will start from a global approach, considering the attenuation properties of the finite elements and the data inherent to the absorption properties[1] of the interested medium being available. In the treatment of the subject, which will be carried out, the material presents prevalent gamma emissions (radionuclides greatly present), uniformly distributed and unequivocally determined by spectral lines. It is possible to hypothesise in order to reduce a sample in finite elements, considered as precise sources for simpler treatment, yet it is computationally suitable at the same time.

In the majority of cases, the level of activity of a source varies by varying the distance of the P observation point from the source, and from the properties of the medium crossed by ionised radiation.

$$A(P)=A(Ao, \underline{S}, \rho) \tag{2}$$

where:
Ao = activity in correspondence with the emission point;
P= observation point or radiation exposure target;
$\underline{S}$ = vectorial coordinate of point P compared to source origin;
ρ = specific density of material seen in relation to its mass attenuation;

Theory

The final objective of this study of materials is to determine the dosimetrical units associated with exposure to emitted radiation. The knowledge of the dose values descends directly from the values of the field typical parameters. In such a paragraph Laws are presented in order to determine the flow density for definite sources in space, in the hypothesis that the interplaced medium is air and that the point-source distance is such that it is able to disregard the effects of secondary absorption.

It is known that an isotropic point source is for the simplest case of emission. Photons are uniformly emitted in all the 4π solid angle, and in the absence of absorption, the φ flow density results as being thus defined (Pelliccioni, 1993):

$$\varphi = \frac{S}{4\pi r^2} \tag{3}$$

[1] Each medium considered has known properties such as density and degree of radiation absorption. Moreover the spectrometric analysis add information relative to the emissive properties of the material. The latter data can be evaluate according to an approach which takes into account each single radionuclide and of its own emission frequency, or rather, its total radiation contribution given by the finite material block.

At this point, also the attenuation produced by auto-absorption of the same materials comes into play, as well as the air. It is necessary to introduce a further complication (Salinas et al., 2006). Returning to the formula for photon attenuation, the pass-ratio results as being:

$$f = 1 - \frac{N}{N_0} = 1 - e^{-(\eta/\rho)\ \rho l} \tag{4}$$

For mono-energetic radiation the mass attenuation coefficient is known, as is the density of the material crossed. The term "l" represents the distance in the material with which the equi-balanced source sees the target[2]. The latter being dependant on the spatial coordinates of the integration dominion (Alitto, 2005), even the f function will depend on them.

$$\varphi = \iiint_V \frac{S_v \cdot f(x,y,z)}{4\pi r^2}\, dx\, dy\, dz \tag{5}$$

The study of an almost standard indoor environment with six walls, is conducted by the study of each individual wall. For exposure symmetry the maximum emission plane is situated along the axis of each wall. Radioactive emission of a wall presents a bell trend with the maximum situated in correspondence with the centre axis of the wall. Since gamma radiation covers a certain distance in the air ($1.2 \cdot 10^4$cm) it is a reasonable hypothesis to consider that the emissions of each wall are overlapped and culminate in the central area of the environment where people are more likely to be residing.

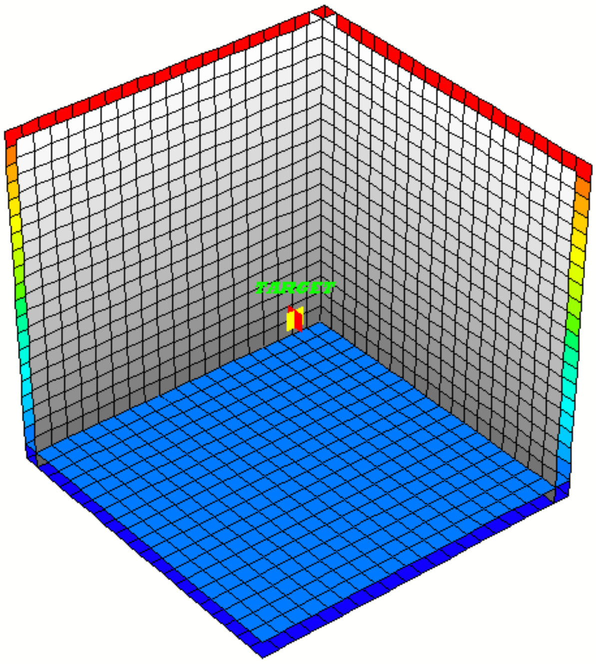

Fig. 2. Identification of the calculation target of a hemi – environment.

[2] Supposing that the wall is on an xy plane, that the target has P(x0,y0,z0) coordinates, that the distance of P from the wall is d, r the distance of a generic radiant point dV of the wall from P, then the distance $l = r(1 - d/|z - z_0|)$.

Furthermore, considering the absorption power of the air is translated into the introduction of a multiplicative factor similar to f(x,y,z), considered with the parameters of standard air composition.

$$\varphi = \iiint_V \frac{S_v \cdot f(x,y,z)\ f_a(x,y,z)}{4\pi r^2}\, dx\, dy\, dz \tag{6}$$

Such a further contribution leads to the natural extension of the formula to the case in which, in the virtual path of radiation, from the source to the target, a series of different mediums and materials, components such as mixed-layer thicknesses with different attenuation properties (Stranden, 1979), is in existence.

$$\varphi = \iiint_V \frac{S_v \prod_i f_i(x,y,z)}{4\pi r^2}\, dx\, dy\, dz \tag{7}$$

Nevertheless, the lack of "good geometry" conditions leads to the necessity of taking into account the build-up effect (Brar et al., 1999), on the basis of the components which characterize the materials. A corrective factor will be correlated to the formula (7) which is equal to

$$B(E,r) = 1 + \frac{(b-1)(K^x - 1)}{K - 1} \tag{8}$$

The formula to undergo simulation will be the result of the following mix (Alitto,2005):

$$\varphi_{sim} = \frac{S_v}{4\pi} \iiint_V B(E, r(x,y,x)) \frac{\prod_i f_i(r_{(x,y,z)})}{r^2}\, dx\, dy\, dz \tag{9}$$

Validation of the simulations was carried out using a comparison of the results with scenarios that have already been experimented and calibrated (R.P. 112). By means of the data provided by it, it was possible to carry out the direct identification of the external gamma dose, due to the presence of walls, floors, surfaces, thicknesses etc., by means of the conversion formula:

$$D^i_{\gamma\text{ - ext}} = \varphi \frac{c_i}{Sv} \tag{10}$$

Since the valuation of the external dose in mSv/year is through the knowledge of the fluence rate φ, obtained from the simulation, from the typical mass photon flow of the material and from c_i[3] which is a characteristic function of the i single radionuclide considered, in the case of mono-energetic radiation. The extension of poly-energetic radiation (Higgy et al.,2000), therefore a classical distribution of nuclides in the material matrix, is obtainable simply as a summary of the contributions of the single nuclide dose.

[3] The term Ci depends on the concentration of activity of the natural gamma-emitter element at a given frequency, and by a equal scale factor characteristic of the considered standard environment in R.P. 112 for 4x5x2,8 m³ , with wall thickness equal to 0,2 m and radiant material uniformity conditions.

Materials and methods

The field of radiation produced by a single finite wall, in which a certain concentration of natural radionuclides results as being uniformly distributed, can be inferred through the use of the equations seen previously, with the use of an appropriate calculator, through the use of volumetric numerical resolution, through libraries which are, in part, present in the software used (Matlab 7.0).

Such graphic representations of the emitting wall allow for the hypothesis that, in a preliminary and evidently superficial way, the previously formulated theory is correct, or rather that the field produced in a standard indoor environment (6 walls, of which two floors and four horizontals) is at its greatest in the central point.

The incorrectness of this assumption was verified proceeding with a volumetric analysis of the volume considered tridimensionally.

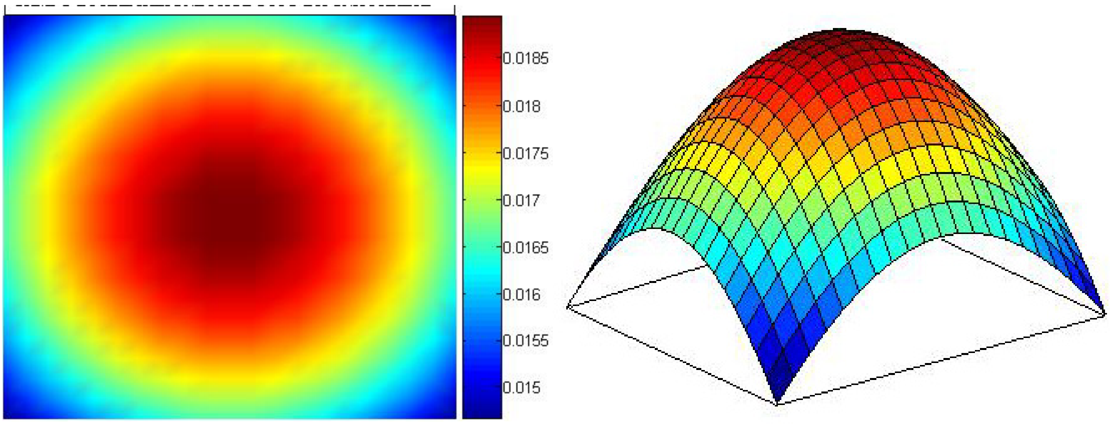

Fig. 3. Radiation field produced by a single wall by materials containing natural gamma emitters per unit of mass photon flow (left) and graphic visualization of the field produced by a wall as 3D surface (right).

In the first phase, the distribution of a reference mono-energetic source (K-40), uniform in all the material forming the 6 walls of a standard indoor environment, having cubic dimensions (4 x 4 x 4 m³), and equal thickness for each wall (0,2 m) was considered (Alitto, 2005).

A tridimensional calculation model was created which considered, for the evaluation of the field of radiation, volumetric elements of the room that were sufficiently small to be able to consider, the reasonably constant field of radiation within them.

The centres of each of these volume elements, aligned according to the three spatial directions, form the calculation plane of the field of radiation, or rather the planar portions of the environment space, to evaluate the variability of the field of radiation.

Results and discussion

The simulation was carried out, taking into consideration the aforementioned parameters, for the determination of the fluence rate of gamma radiation in each point-packet of the volume delimited by the walls according to the schematisation highlighted in the following image.

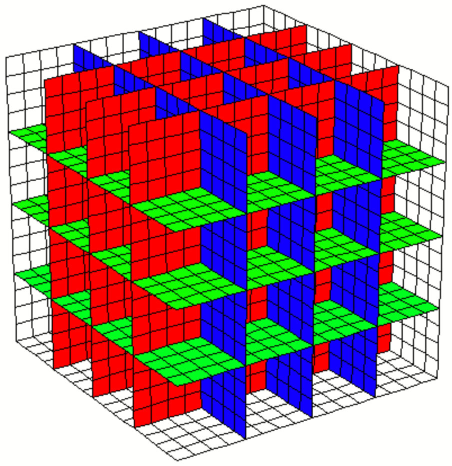

Fig. 4. The hemi-environment delimited by the walls (white grid), and the calculation planes (coloured) along the three directions. Some of them are shown as examples.

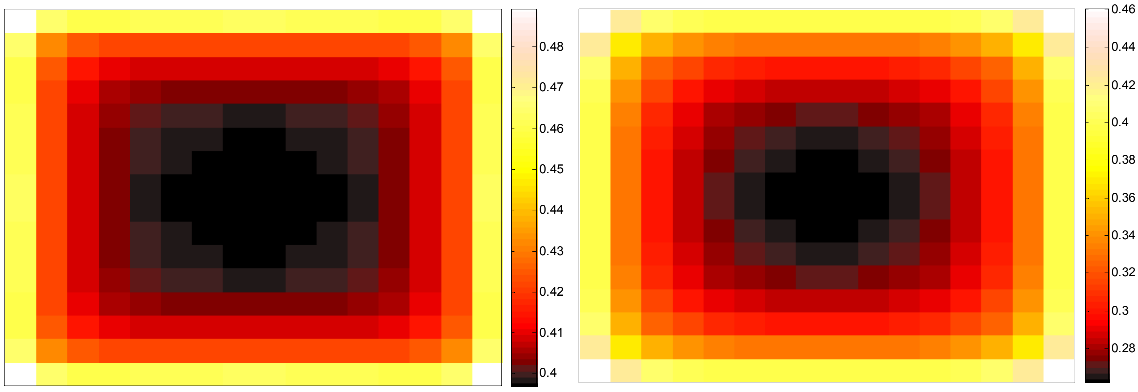

Fig. 5. Colorimetric evaluation of the field of gamma radiation induced by a source distributed per unit of photon volume flow, in proximity to one of the radiant walls (left) and for a central section (right).

The symmetry intentionally chosen for the case, which does not preclude the possibility of extending the model to scalene dimensional parameters, allows for the carrying out of an exhaustive evaluation of the calculation planes once for all components, taking one of them as a starting point.

From the compared images it is possible to note how, in reality, the hypothesis carried out previously is not correct. The most intense points of the field of radiation are represented by the areas contiguous to the walls. In substance, the greater effect of the field is in the vicinities of the corners. Instead, the centre of the environment must be considered as that having the lesser radioactive impact.

The example was executed in the maximum symmetry conditions: in this case, the reference environment was cubic, with uniformity also in the material considered. It is necessary to bear in mind that, in the reality of the cases, the environments are of scalene dimensions and that the materials used are composites and mixed-layers (Jalali et al., 2008).In order to render the example more realistic a simulation was carried out considering the case of an environment with unequal dimensions on three sides (4 x 5 x 3 m^3), with different horizontal wall compositions (masonry- full and perforated ricks) and lower and upper floors (granitic flooring, standard slab).

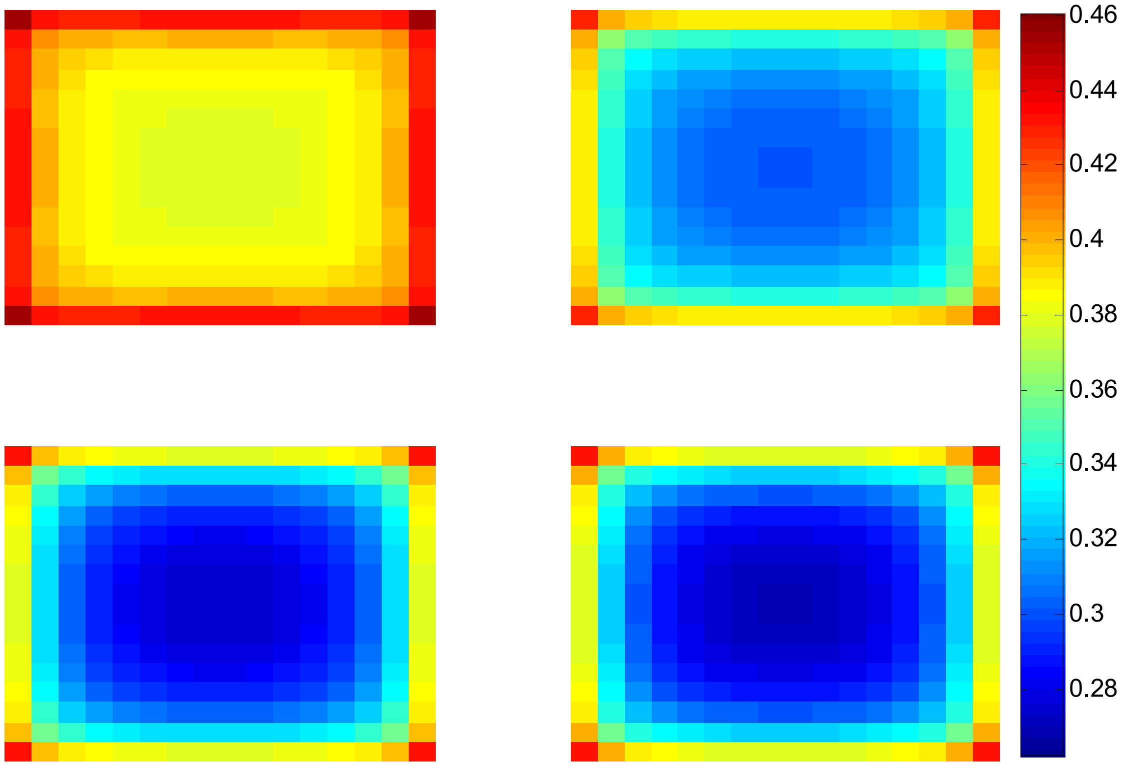

Fig. 6. Comparative evaluation of the field of radiation in the calculation planes which start from the area adjacent to the wall to the central one of the indoor environment.

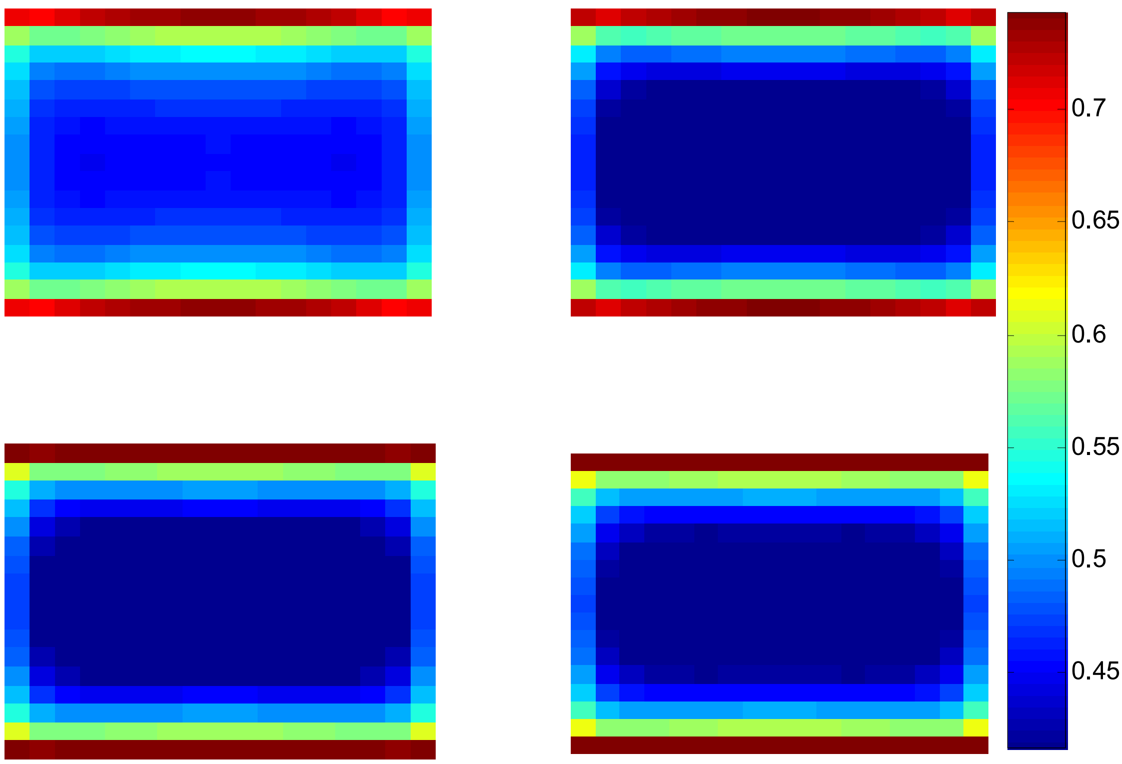

Fig. 7. Comparative evaluation of the field of radiation for an environment with scalene dimensions with a diversity of materials forming the walls (granite, masonry-bricks).

Such complications only introduce new weighting factors for the radiation field attenuation, modifying the geometry and symmetry of the field already seen in the previous case.

This new example cannot be exhaustive of the real conditions of use, and therefore of exposure that, obviously, depend on the typical occupation times but also furnishings, the presence of obstacles which more or less influence the trend of radiation.

The following values of concentration of activity referring to the single components of the simulation were inserted:

Table 2. Activity for materials used for the simulation.

Data in literature on commonly used materials (R.P.112)	Average activity (Bq/kg)			Conventional dose values (R.P.112	
	^{226}Ra	^{232}Th	^{40}K	I	E (mSv/a)
Cement	40	30	400	0,42	0,22
Pink Granite	150	120	1600	1,63	2,46
Perforated bricks	125	85	1220	1,25	1,50
Stucco – plaster	10	30	200	0,25	0,10
Wall paint	5	5	15	0,05	0,01

The value of the fluence rate per unit of mass photon flow, in the environment considered in figure 7, has an average value equal to about 0,5. The effective annual dose that results by applying the simulation, results as being equal to E[4] = 1.53 mSv. Such a value corresponds to the corresponding gamma radiation produced by the excess materials as well as the natural dose. This value can be significant of an uncontrolled situation since it concerns the values of indoor exposure to only gamma radiation. The limit introduced by National legislation, which regards exposed workers, imposed an annual limit value equal to 1 mSv. According to epidemiological studies (UNSCEAR 2006), this value corresponds to a probable mortality rate of 0.01%. This value risks further complication due to the coexistence of radon.

The evaluation was made taking in account conversion coefficient in UNSCEAR 1993, 2006 and conversion coefficient from the dose in the air to the effective dose received by an adult individual, 0.8 for the indoor occupation factor, and 0.2 for outdoor (Mulligan et al., 2004). From these values some dose evaluation samples follow, evidently starting from dosimetrical valuations:

Indoors: 84 nGy h^{-1} 8,760 h × 0.8 × 0.7 Sv Gy^{-1} = 0.41 mSv
Outdoors: 59 nGy h^{-1} × 8,760 h × 0.2 × 0.7 Sv Gy^{-1} = 0.07 mSv

The global average resultant for the effective dose is 0.48 mSv, with oscillating values from country to country ranging from 0.3 to 0.6 mSv. For children and infants, the values result as being from 10% to 30% higher, in direct proportion to the increase in the air dose conversion coefficient to the effective dose.

[4] In the dose value nuclides that are more significant to gamma emission, such as ^{40}K, ^{232}Th, ^{226}Ra are taken into consideration.

Application case study and conclusion

In this last part the possibility of identifying known radioactive sources in structures to be demolished in an indirect way will be shown, examining the model predisposed in the second paragraph.

In the case in which a nuclear plant is being dealt with, for which the principal radionuclides deriving from processes are known, the evaluation carried out previously is not applicable in a direct manner. In fact, considering a uniform distribution of the contaminant agent would not be correct. However, theoretically the considered parameter, Sv, should be substituted with a function that quantifies more or less realistically the contamination situation. In that case it could be possible to adopt a concentration distribution of nuclides equal to an expression such as:

$$Sv = Sv_0\, e^{-k\left((x-x_o)^2+(y-y_o)^2+(z-z_o)^2\right)} \qquad (11)$$

Where the mass photon flow is defined with Sv_0 per volume unit which then is configured in the central contamination position (x_0, y_0, z_0), k represents a scale factor.

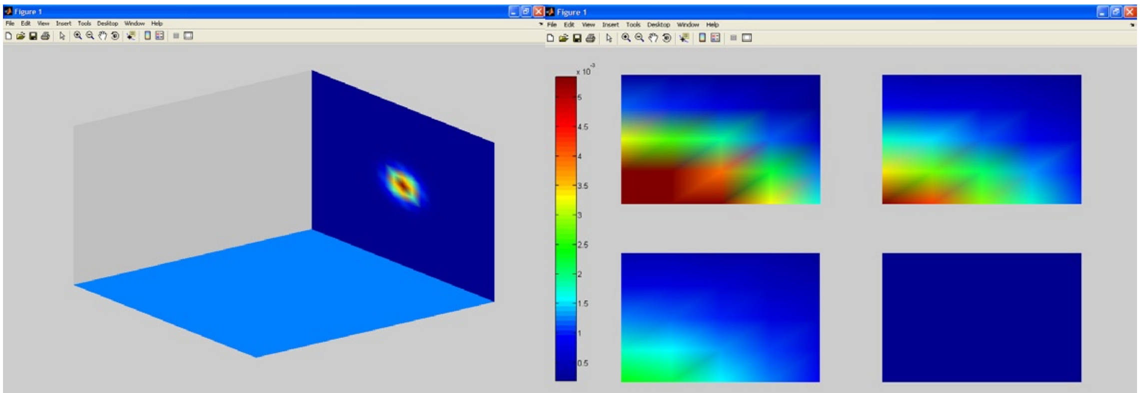

Fig. 8. Widespread Distribution hypothesis of a contaminated source or NORM (left) and distribution of the field of radiation of a source, concentrated in the corner of a standard room, starting from the wall adjacent to the radiation.

Within contaminated environments, there are automatic devices which capture the actual activity level (gamma-environment systems, dosimeters, instantaneous detectors, etc.). For practical reasons, such instruments are placed in correspondence of one of the delimiting walls of the environment.

Considering the case of a source that is concentrated in one point of the environment (e.g. in the corner) the distribution of the field of radiation, starting from the calculation plane corresponding to the wall adjacent to the contamination, to the one corresponding to the most distant one (hypothesising a regular environment), is described in figure 8.

Prior to resolving the problem it is first necessary to attempt to formulate it, the problem consists of the evaluation of the entity of a contaminating component, starting from the information provided by in-situ spectrometric data.

As already stated, there is in existence a diversified typology of portable or fixed instruments, even remotely connected, which acquire data providing the instantaneous value of parameters which can be the dose, the activity of a source, the fluence rate etc. Such instruments provide data which is more or less precise of the parameters considered.

Case histories require that the involved radionuclide are known, and with well known emission properties. It all rests upon the determination of the most suitable calculation positions for the evaluation of the actual contamination of the environments: an object of fundamental importance for the entire decontamination process.

The limit of decontamination at the desired clearance level consists, inevitably, of the computation and separation of the natural radiation component, which, as we have already seen, is not negligible and, often, complicates the foreseen removal process.

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The Safety Culture as a part of radiation protection in medical imaging

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Abstract
The safety culture means individual awareness of the importance of safety, competence, commitment, motivation, supervision and responsibility, concerning also attitudes of the staff at all levels. The enhancement of patient safety involves a wide range of actions in the recruitment, training and retention of health care professionals, performance improvement, environmental safety and risk management, equipment safety, safe clinical practice and safe environment of care. To find out some features of the radiation safety culture, several studies in different areas concerning radiation safety officers in diagnostic departments and patient doses were analyzed. The main factors affecting to the safety culture in medical use of radiation were regulatory and organizational environment, management styles, workers and their attitudes, patients' dose optimization as well as technological characteristics. Optimization of image quality and patient dose allows the patient dose to be decreased by more than 50 % in many patient groups. Digital imaging gives possibility to degree dose and still has image quality good enough. The attitudes and fear of change are the biggest barriers to reach the good safety culture. The next step is the establishment of safety culture, which takes into account the attitudes, behavior and other human factors, which have effect on safety.

Introduction
In industry the concept of "safety culture" has been known and in use for long time (Cooper 2000). To the area of radiation it has become from the OECD Nuclear Agency report (INSAG 1988) on the Chernobyl disaster. The safety culture means individual awareness of the importance of safety, competence, commitment, motivation, supervision and responsibility, concerning also attitudes of the staff at all levels. It describes the corporate atmosphere or culture in which safety in understood to be and is accepted as, the number one priority. (Cullen 1990.) International Atomic Energy Authority (IAEA 1991) defined safety culture as "assembly of characteristics and attitudes in organizations and individuals which establishes that, as an overriding priority, nuclear plant safety issues receive the attention warranted by their significance".

A British advisory committee on human factors in nuclear safety identified senior management commitment, management style, management visibility, communication, pressure for production, training, housekeeping, job satisfaction and workforce composition as key indicators of the safety culture. (Flin et al 2000.) Also the concept "safety climate" has been analyzed.

Safety culture does not operate in a vacuum by it self along and without any reflections. It affects to other organizations and is affected by other organizations. In the same hospital all users of medical radiation should have a common and together agreed goal how to apply the safety culture in clinical use and what it includes. The enhancement of patient safety involves a wide range of actions in the recruitment, training and retention of health care professionals, performance improvement, environmental safety and risk management, equipment safety, safe clinical practice and safe environment of care. (Flin 2000, Holopainen 2004, Niemi 2007)

The main trends in safety culture of radiation protection in medical radiology in Finland can be described as shown in Table 1. The Radiation and Nuclear Safety Authority in Finland (STUK) made in 1950's radiation safety measurements in hospitals, because hospitals were not able to do these measurements by themselves. The weight was in the radiation safety of the staff. Later the measurements developed from protection measurements to quality assurance, performance of the unit and patient dose measurements. The radiation safety of the patient was emphasized. Gradually the responsibility from measurements transferred to hospitals and the authority only control these measurements. In the 2000's new activities such as optimization of image quality and patient dose, clinical audit and continuous education and training in radiation protection were started. (Kettunen et al 2007, Holopainen 2004.)

Table 1. The rough schedule of activities in medical radiation protection in Finland. (Kettunen et al 2007)

(Action in Medical Radiation Protection	Year
Safety license and inspections	-1960
Occupational dose measurement	-1960
Radiation safety of the diagnostic units	-1965
Performance of the units	-1970
Quality assurance	-1980
Radiation dose to patients	-1990
Optimization of examination technique	-1990
Clinical audit	-2002
Education and training in radiation protection	-2003
Safety culture	-2004

Trend in radiation protection of medical radiology shows that the clinical audit is the new step to promote good practice because in audits the current clinical written practices are compared to so called 'good practice'. Clinical audit covers the whole imaging chain from writing the referral to imaging examination up to the report of imagines. The hospital is responsible for that action. (European Commission 2009.)

Follow up of the dose reference levels (DRL) can be also seen as a part of good practice (Säteilyturvakeskus 2008). The background of these guidelines and orders is very complex and several steps and organizations are needed (Figure 1).

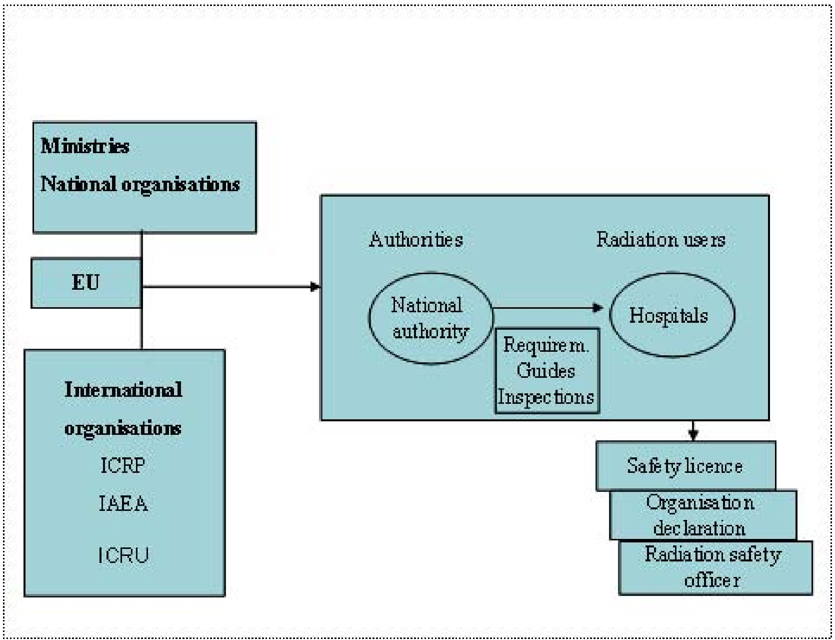

Figure 1. Factors affecting to the safety culture in medical use of radiation. (According to Holopainen 2004).

The latest, very fast development in the area of imagining modalities demands a lot of work in dose and image quality optimization and management. Deterministic harms have been reported both in interventional (e.g. Siromäki 2004) and diagnostic radiology (Yoshimasa 2005).

The next step could be the establishment of safety culture, which takes into account the attitudes, behavior etc human factors, of the staff. This has been already established in nuclear power plants where the consequences of the accident may be remarkable. The errors of the diagnoses reported in literature show high frequencies so the follow up and reporting system should be established. The safety culture may cover all those factors. One model for the assessment of safety culture has been made by IAEA 1998 (Figure 2).

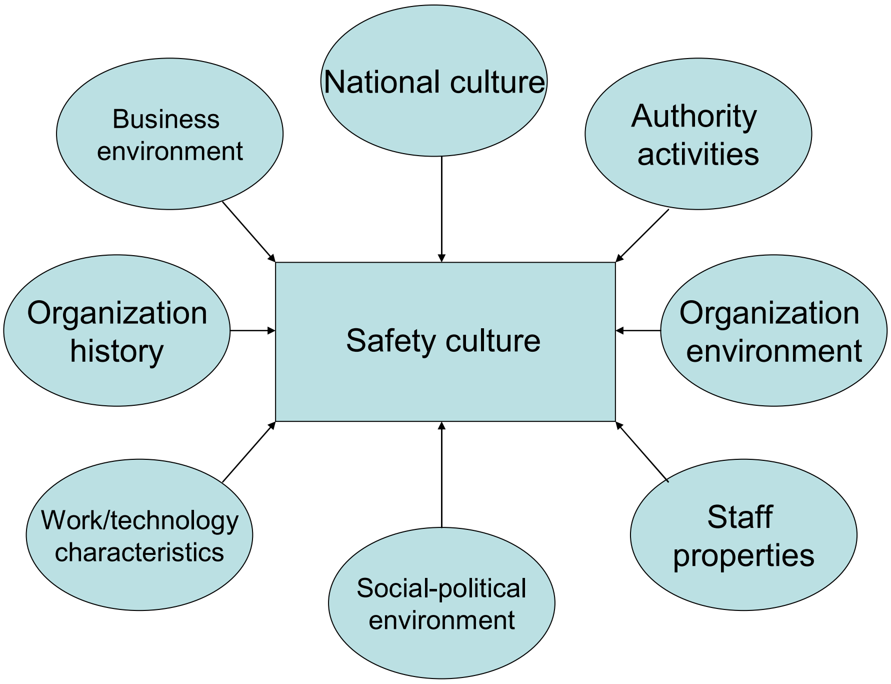

Fig. 2. A model for the assessment of safety culture. (IAEA 1998)

Material and methods

To find out some features of the radiation safety culture in Finland, several studies, reports and articles in different areas were analyzed concerning radiation safety officers in diagnostic departments and patient doses in medical imaging.

The volume of medical radiation activities of medical radiology in Finland has been quite the same level during last ten years. There are about 650 Safety licenses and 1500 x-ray units. Number of radiologists in dose monitoring in 2007 was 543 and number of radiographers in dose monitoring in 2007 was 2583.All personnel in health care in dose monitoring in 2007 was about 4900 including nurses, bioanalytics, dental personal, surgeries, orthopedics etc. (Rantanen 2009, Lehtinen 2008)

Radiation and Nuclear Safety Authority has given reference levels for adult patient in common projection examination, in computed tomography examinations and cardiac angiograms. There are also DRLs for nuclear medicine examinations as well DRLs for children in projection examinations. The adult's DRLs have been updated in 2009. (Säteilyturvakeskus 2008.) The Regulatory Guides on radiation safety (ST Guides) are updated approximately once in five years.

Two studies concerning radiographers' work and one concerning radiation safety officers work and attitudes where analysed as well a group of papers about the doses to patients or to patients groups. Examples of dose and image optimisation are given based on results of small optimisations in clinical practice made by the staff and radiographer students.

Results

The safety culture is divided here in two parts. First the group of people and their role in safety culture and second the dose management as a part of safety culture.

Regulatory and organizational environment, management styles, workers and attitudes as well as technology characteristics were the main factors affecting the safety culture. Niemi (2006) found four shared meanings in radiographer's safety culture in the medical use of radiation. The meanings were challenges of knowledge and skills structuring safety culture, dimensions of cooperation enabling safety culture, disorientation conditioning safety culture and multidimensional professionalism as the foundation of safety culture. The significance of radiation protection was emphasized but the radiographers were confused by the different directions and practices in implementing it.

Paalimäki-Paakki (2009) described ethical dilemmas in radiographer's work in diagnostics. Most important ethical dilemmas were found to consider the use of radiation, patient care, and radiographer's work community. In the use of radiation, implementation of justification and optimisation principles were found to be lacking. Dilemmas in work community consisted of problems among employees and insufficient practice. The radiographers noticed some times that the colleagues did not work as said in quality handbook (written good practice). Dose and image quality optimisations and the use of lead shields were not good enough. According to Paalimäki-Paakki is due the lack of safety culture.

Radiation safety officers are the key persons in the radiation safety culture of medical radiology. The study concerning radiation safety officers show that organizational aspects are the main factors in the safety culture of radiation protection in medical radiology. The most remarkable developing targets were stated the clarification of field of activity of radiation safety officer, possibility to affect decisions, increase of education and increase of cooperation with other radiation safety officers in various hospitals. New active partners in this field have appeared (quality assurance, education and training, occupational dose measurements, accreditation, clinical audit etc) secure the sufficient status and empowerment in the organization. Regulatory and organizational environment, management styles, worker as well as work/technology characteristics were the main factors affecting the safety culture of safety officers. (Holopainen 2004, Servomaa, Holopainen 2005.)

In the study of newborn premature babies 118 chest x-ray examinations to 43 newborns (gestational age from 26 to 42 weeks) were estimated. The effective dose from one chest radiograph varied from 7.5μSv to 54.8μSv. Retrospectively, the total number of radiation examinations to these newborns totalled 399 during the study; the mean was 9.3 (range 1-40). 98% of the examinations were produced during the first treatment period after birth. The total effective dose per child varied from 0.31 mGy to 3.7 mGy. (Kettunen 2004.) The smaller the baby is the more x-rays are usually needed for the follow-up during the first living months (Siironen 1994)

The study of pregnant woman' examinations showed that there was no common practice on how to exclude the possibility of pregnancy and the dose to a foetus was not estimated either before or after the pelvic x-ray examination. (Kettunen 2004.)

The dose study of pediatric reflux patients showed that about 6. 9 examinations / patient were made up to the age of 16 years. Average effective dose was 4.3 mSv and maximum dose about 11 mSv (Table 2). (Kettunen et al 2003.)

Table 2. Numer of examinations and effective dose, average and range to 12 patients under 16 years due to examinations of vesico-uretal reflux. (Kettunen et al 2003)

	MCU	MCU TC99m	Urography	Kidney with Tc99m	Total Number of examinations
Total Number	40	15	20	2	77
Average (range)/ one patient	3.8 (1-7)	1.3 (0-3)	1.7 (0-7)	0.2 (0-1)	6.9 (1-15)
Effective dose /patient annos(mSv)	1.7 (0.6-3.5)	0.25(0.15-0.6)	2.0(1.3-8.1)	0.3(1.9)	4.3(0.7-11)

The study of radiological examinations made for young accident patients showed that average number of examinations varied between 9-59 and effective dose varied between 5.3-33.9 mSv. (Huusko, Räsänen 2002, Servomaa & Kettunen 2005.)

These studies show some special groups of patients which need a lot of attention and especially tight indications for x-ray examinations. Next results will show how dose and image quality optimization can be applied with small steps in everyday clinical work.

The new full field mammography equipment had Dose- and Standard (STD) programs. The vendor recommended to use the STD program. The radiographers wanted to evaluate the difference in the dose and image quality between these programs. From ten females in mammography were exposed randomly either left or right breast with STD-program and other with Dose-program. Three experienced radiologists (used to read screening mammograms) evaluated them "blind" without knowing which are taken by which technique. All mammograms were good enough for diagnostic. There were no difference find between images taken by STD- or Dose-programs and now all mammograms are taken by Dose-program and lower dose. The dose reduction was about 30 %. The AEC (Automatic exposure Control) was tested in order to reach lower dose level. In Lumber spine lateral examination -16 % decrease and in hip joint -27 % dose decrease was achieved very easily image quality was still good enough. The PACS (Picture Archiving and Communication System) offered opportunity to analyse the quality of images afterwards. The use of air grid is not very common in Finland. In hip axio-lateral project the use of air gap (30 cm) instead of a grid decreased skin dose two thirds (Figure 2). (Henner et al 2009.)

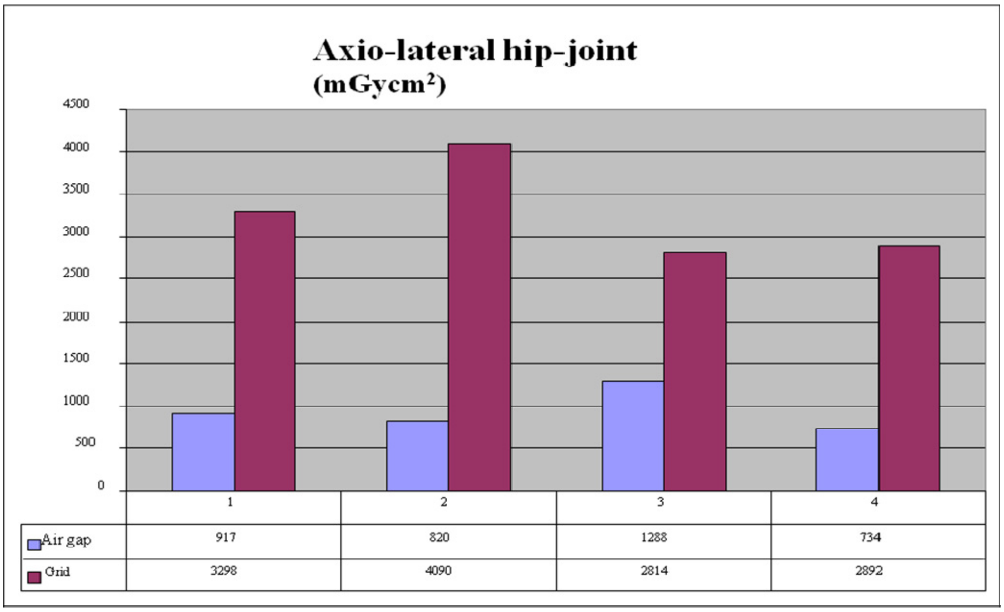

Figure 2. DAP in Axio-lateral hip joint with Grid and Air Gap. (Henner et al 2009)

Because in the lower extremity mechanical axis the required image quality for a diagnosis is not high, the results showed that the patient radiation dose could be lower when additional filtration and higher voltage were used. The results also showed that by using different AEC classes the patient dose could be affected and thus should be used. According to these results it is not necessary to use a grid while imaging a normal size patient, because the quality of the image does not need to be as high as for example when diagnosing a fracture. While imaging the hip predetermined exposure charts should also be considered. (Hilli, Hirvelä 2010.)

Discussion

The concept of safety culture and safety climate is not yet well-known and they are quite difficult to find out in every day clinical practice. Problems in healthcare all over the world are common: less money, more seriously ill patients, expensive equipments and medication, lack in education, more work and less staff. The role of Radiographer is expanding to new areas partly because of higher education and partly because lack in radiologists and physicists. Better equipment means better image quality with higher dose. Does this mean also more harms due to radiation? Who is responsible for the dose optimisation? As seen from the results of Niemi (2006) and Paalimäki-Paakki (2009) stress, hurry, new demands and more complicated situations are parts of radiographers work, every day. They are barriers to work as well as they want. They also purchase the further development of work conditions and dose and image optimization.

Radiation and Nuclear Safety Authority has given reference levels for adult patient in common projection examination, in computed tomography examinations and cardiac angiograms. There are also DRLs for nuclear medicine examinations as well DRLs for children in projection examinations. The doses have to be monitored at least every third year. (Säteilyturvakeskus 2008.) This part of safety culture has been applyid very well since 1996. The Regulatory Guides on radiation safety (ST Guides) are updated approximately once in five years and the guidelines are applied in all radiological departments. Second round of clinical audits is running (Faulkner et al 2009).

There are special patient groups which are exposed quite often (children with scoliosis or vesico-uretal reflux, premature babies). The new technical solutions offer a lot of possibilities for dose reduction, if we want. (e.g. Al Khalifah., Brindhaban 2004, Arreola, Rill 2004, Bush 2004, Hamer et al 2005 Uffman 2008, Geijer 2009, Lanca et al 2009).

One model for the assessment of safety culture has been made by IAEA 1998 is too large for this kind of work. It should be divided to smaller parts in research. Other vise there are a lot of very congregate and everyday ways to develop the safety culture. The management and legislation offer the basics but the work must be done among those who are working in radiological departments.

Conclusions

Optimization of image quality and patient dose allows the patient dose to be decreased by more than 50 % in many patient groups. Digital imaging gives possibility to degree dose and still has image quality good enough. The attitudes and fear of change are the biggest barriers to reach the good safety culture. The next step is the establishment of safety culture, which takes into account the attitudes, behaviour and other human factors, which have effect on safety.

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The role and responsibilities of the medical physicist as the radiation protection adviser in the healthcare environment

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Abstract

In the European Member States, the use of ionising radiation in the healthcare environment is regulated by the transposition into national legislation of Directive 97/43/Euratom of 30 June 1997 on health protection of individuals against the dangers of ionising radiation in relation to medical exposures.

EFOMP noted the definition of the Medical Physics Expert and the involvement of this expert in radiotherapeutic, diagnostic nuclear medicine and other radiological practices in this Directive. EFOMP has for many years sought to harmonise and promote the best practice of medical physics in Europe and has in this respect issued Policy Statement No. 9 in response to the above directive.

The EFOMP strategy is directed towards the recognition of the European Medical Physicist and for this purpose the Federation's approach has been to encourage registration schemes (on a voluntary basis) where no regulated scheme (as imposed by law) exists. EFOMP recognised that the most appropriate way to achieve harmonisation of standards across the whole of Europe was to express the duties and competencies

expected of the Medical Physics Expert (MPE), as set out in the 1997 Directive, in very practical terms.

A framework of five levels of competency that covers the whole career structure of the medical physicist was developed including the duties of the MPE. A system for recognised Continuing Professional Development is recommended by EFOMP and has been described in a Policy Statement.

This paper aims to present EFOMP's efforts to harmonise the education, training and competences of the Medical Physicist in Europe and thus to meet the role and responsibilities of the MPE as specified in the above directive.

Introduction

In the European Member States, the use of ionising radiation in the healthcare environment is regulated by the transposition into national legislation of Directive 97/43/Euratom of 30 June 1997 (EC, 1997) on health protection of individuals against the dangers of ionising radiation in relation to medical exposures. This directive supplements Directive 96/29/Euratom (EC, 1996) laying down basic safety standards for the protection of the health of workers and the general public against the dangers arising form ionising radiation.

In these directives, the definitions of the professionals involved are subject to various interpretations. For example under article 2 of directive 97/43/Euratom, the definition of the Medical Physics Expert is given as:

"Medical Physics Expert: an expert in radiation physics or radiation technology applied to exposure, within the scope of this Directive, whose training and competence to act is recognized by the competent authorities; and who, as appropriate, acts or gives advice on patient dosimetry, on the development and use of complex techniques and equipment, on optimization, on quality assurance, including quality control, and on other matters relating to radiation protection, concerning exposure within the scope of this Directive."

In the above directive extract, the recognition of the MPE is left to the national competent authorities and this has been recognised at different levels of competence in most of the European Union Member States, as evident from the latest EFOMP survey (Eudaldo, 2008). Furthermore Article 6 clause 3 of the same directive states:

"In radiotherapeutic practices, a medical physics expert shall be closely involved. In standardized therapeutical nuclear medicine practices and in diagnostic nuclear medicine practices, a medical physics expert shall be available. For other radiological practices, a medical physics expert shall be involved, as appropriate, for consultation on optimization including patient dosimetry and quality assurance including quality control, and also to give advice on matters relating to radiation protection concerning medical exposure, as required".

The phrases "closely involved", "shall be available" and "shall be involved, as appropriate" take a different interpretation in each and every setting. It is again left to the local competent authorities to interpret these phrases. The EFOMP survey (Eudaldo, 2008) has in fact shown that there is a large variation between the European Union Member States.

In a similar fashion Directive 96/29/Euratom defines Qualified Experts as:

"Qualified Experts: Persons having the knowledge and training needed to carry out physical, technical or radiochemical tests enabling doses to be assessed, and to give advice in order to ensure effective protection of individuals and the correct operation of protective equipment, whose capacity to act as a qualified expert is recognized by the competent authorities. A qualified expert may be assigned the technical responsibility for the tasks of radiation protection of workers and members of the public".

The term "Qualified Expert" is mentioned in a number of places in this directive implying a different professional such as Medical Physicist Expert, Radiation Protection Expert, Radiological Practitioner, etc. Again the competence of the Qualified Expert is recognised by the national competent authorities (which may in fact be different from those deciding the competencies of the MPE). Furthermore the role and responsibilities of the various Qualified Experts are not clearly stated in this Directive but are left to the National Competent Authorities and others to specify.

Therefore within the European Union, we have two directives relevant to radiation exposures in the Healthcare Environment and 27 national sets of legislations and regulations. This is due to the interpretation being understood differently, taking into account the culture, language and other political influences present in each national setting. This is hindering the harmonisation of the education, training, competences, roles, responsibilities and effectively the mobility of the professionals responsible for implementing the various functions covered by these directives.

With respect to the MPE, EFOMP noted its definition and the involvement of this expert in radiotherapeutic, diagnostic nuclear medicine and other radiological practices in this Directive. EFOMP has for many years sought to harmonise and promote the best practice of medical physics in Europe and has issued Policy Statement No. 9 (EFOMP, 1999) in response to the above directive.

In this Policy Statement, EFOMP lists in general terms the involvement of the MPE as follows:

"• *to carry out the physical measurements related to evaluation of the dose to the patient and to take responsibility for dosimetry;*
- *to improve any conditions that will lead to a reduction in unnecessary patient dose;*
- *to lay down tests in the field of quality assurance of the equipment;*
- *to assure the surveillance of the installations with regards to radiological protection;*
- *to choose equipment required to perform radiation protection measurements and to give advice on medical equipment;*
- *to take part in the training of medical practitioners and other staff in relevant aspects of radiation protection;*
- *to provide skills and responsibilities that complement those of medical practitioners."*

This Policy statement gives further guidance with respect to the education, training and continuous professional development of the MPE. These will be discussed in this paper.

Recently the Commission has undertaken a simplification of Community legislation in the area of radiation protection and has proposed the consolidation into a single text of the following Directives:

- Council Directive 96/29/ Euratom of 13 May 1996, laying down basic safety standards for the protection of the health of workers and the general public against the dangers arising from ionising radiation,
- Council Directive 97/43/Euratom of 30 June 1997 on health protection of individuals against the dangers of ionizing radiation in relation to medical exposure,
- Council Directive 89/618/Euratom of 27 November 1989 on informing the general public about health protection measures to be applied and steps to be taken in the event of a radiological emergency,
- Council Directive 90/641/Euratom of 4 December 1990 on the operational protection of outside workers exposed to the risk of ionizing radiation during their activities in controlled areas,
- Council Directive 2003/122/Euratom of 22 December 2003 on the control of high-activity sealed radioactive sources and orphan sources.

The latter four Directives cover different specific aspects complementary to the overall Basic Safety Standards (the first Directive).

The task of consolidation was given to the Euratom Treaty Article 31 Group of Experts. Their task was finalised in February 2010 and their "Draft Euratom Basic Safety Standards Directive" (This document is not yet publicly available) has been submitted to the European Commission for its approval and further submission to the European Council and Parliament for the necessary political scrutiny.

Definitions have evolved over time and have been adjusted to specific scopes. While many requirements may fit in the original context they could not be extended to act as a general application of the Standards, so the consolidation proceeded through a recast rather than a simple codification of the original texts.

From the latest publicly available summary reports of the Group of Experts (GoE) referred to in Article 31 of the Euratom Treaty meeting (EC, 2009a, 2009b), the new European Basic Safety Standards (BSS) Directive appears that will be defining the roles and responsibilities of services and experts who should be involved in ensuring that technical and practical aspects of radiation protection are managed with a high level of competence. It further appears that will be defining the role, responsibilities and competences of the Radiation Protection Expert (RPE) and the MPE and appears to be introducing the function of the Radiation Protection Officer (RPO).

It also appears to be strengthening the requirement for information, training and education addressing these in a specific title in order to highlight the importance of education and training in radiation protection. In the medical area, education and training should also raise the awareness of the medical profession, in particular with a view to risk communication with patients.

Changes may, however, be made before the Directive is approved and finally published in the Official Journal of the European Union.

Discussion

From what is known at the moment about the recast version of the European Basic Safety Standards Directive, it would seem that an overall improvement to the existing directives is being made. It is acknowledged that the role, responsibilities and competence of the RPE and RPO in facilities outside the healthcare environment are essential for the radiation protection of the workers and the public at large. There could, however, be considerable ambiguity with the roles, responsibilities and competence of the MPE within the healthcare environment. It is therefore necessary that clear guidance is produced to clarify the different roles and responsibilities of the RPE, RPO and the MPE, as well as the need for collaboration between them.

An attempt is made here to assist in the interpretation of these functions (MPE, RPE and RPO) from the EFOMP point of view.

Firstly ionising radiation is used in a large range of different healthcare facilities, from the dentist's office with a simple dental radiographic unit to the large university hospitals with highly sophisticated devices such as Multi-Detector Computed Tomography (MDCT) systems and Positron Emission Tomography (PET) and in their radiotherapy departments systems such as Linear Accelerators (LINACS) using highly complicated treatment procedures such as Image Modulated Radiation Treatments (IMRT) and Image Guided Radiation Treatments (IGRT), not to mention Tomotherapy and Cyberknife systems.

Secondly it is very important to accept that the definitions, roles, responsibilities and competence of the RPE, MPE and RPO are functions that are required to be fulfilled and are not necessarily separate professions or even individuals.

The unique feature of healthcare is that it is the only application of ionising radiation where there is a need for optimisation. Radiation Protection can never be seen in isolation for image quality (in diagnostic radiology) and tumour control (in radiotherapy). This is one of the reasons that radiation protection is part and parcel of the education process of medical physicist.

The continuous application of the radiation protection principles and the routine implementation of any daily quality assurance programme to assure the correct performance of the equipment would be the role of the RPO. The number of RPOs will depend on the number of modalities and sophistication of the healthcare facility. The function of the RPO may be undertaken by any individual with the appropriate training and competence (for example an appropriately trained technologist). The RPO must be a permanent staff of each facility. It is not acceptable to outsource the function of the RPO.

The function of the MPE is an inherent role of the Qualified Medical Physicist (QMP) and in more sophisticated facilities this function may need to be the role of a Specialist Medical Physicist (SMP) (Eudaldo and Olsen, 2010). It can be outsourced for small healthcare facilities, or it can be assigned to in house medical physics services, again taking into consideration the sophistication of the modalities and the workload of the healthcare facility.

Within the healthcare environment the function of the RPE can be undertaken by a QMP or in more sophisticated facilities by a SMP, since the role and responsibilities of the RPE within the healthcare environment are part of the education, training and competence of the QMP and SMP (Dendy, 1997), (EFOMP, 1997 and 1999). The function of the RPE can be outsourced for small healthcare facilities, or it can be assigned to in house medical physics services, again taking into consideration the sophistication of the modalities and the workload of the healthcare facility.

The EFOMP strategy is directed towards the recognition of the European Medical Physicist and for this purpose the Federation's approach has been to encourage registration schemes (on a voluntary basis) where no regulated scheme (as imposed by law) exists (EFOMP, 1995, 1998 and 2001). EFOMP recognised that the most appropriate way to achieve harmonisation of standards across the whole of Europe was to express the duties and competencies expected of the MPE, as set out in the 1997 Directive, in very practical terms.

As far back as 1997, a framework of five levels of competency, covering the whole career structure of the medical physicist was developed including the functions of the MPE (Dendy 1997):

"Level 1 - relevant first degree or equivalent
Level 2 – completion of specialist expertise
Level 3 – completion of practical experience
Level 4 – advance practical experience
Level 5 – mature overview and greater responsibility."

Over the years these were further refined with the latest version given in Policy Statement 12 (Eudaldo and Olsen, 2010). This framework now also includes an extensive requirement of Continuous Professional Development. Also other initiatives have taken place, which are briefly discussed here.

European School of Medical Physics

This is an annual event in collaboration with the European Scientific Institute - ESI and takes place in Archamps, France. The first school was held in 1998. Originally it consisted of five weeks of intensive training in Medical Physics (EFOMP, 1998b). One week is spent on covering each of Medical Imaging with Non-ionising Radiation, Medical Imaging with Ionising Radiation, Medical Computing, Physics of Modern Radiotherapy, Brachytherapy. As from 2008 a sixth week was added on Radiation Protection.

EFOMP Sponsoring Programmes

The sponsorship of meetings and congresses is instrumental in disseminating and encouraging the adoption of the policy statements. EFOMP organises, co-organises, supports and recognises meetings, congresses and courses together with its National Member Organisations (NMOs). Guidelines that explain the above terms and the requirements for interested NMOs to collaborate in such events can be found on EFOMP's website.

The purpose of these Guidelines is to help NMOs to obtain EFOMP sponsorship for their events by setting out the steps that they need to take and the conditions that must be fulfilled.

The biggest event is the biennial European Congress on Medical Physics. Note that there are detailed guidelines on the requirements for this event.

Furthermore, EFOMP collaborates with other European and International Organisations (ESR, EANM, ESTRO, ESMRMB, IOMP, IFMBE, IRPA) by having joint sessions at their conferences, with the objective of promoting its activities, mainly on Education, Training and Professional matters of the Medical Physics Profession (Caruana et al, 2009), (Christofides et al, 2008, 2009a, 2009b, 2009c).

Lifelong Learning

On the 23rd of April 2008, the European Parliament and the Council issued recommendations on the establishment of the European Qualifications Framework (EQF) for lifelong learning (EC, 2008). With these recommendations the European Union is encouraging its Member States to compare and align their education and training qualifications with the eight (8) levels of learning outcomes described in these recommendations.

The purpose of EFOMP's work in education and training is to position the various stages of the Education, Training and CPD of the Medical Physicist in Europe, as stated in the various EFOMP Policy Statements, within the eight levels of the EQF) for lifelong learning and identify any gaps that need to be addressed.

Figure 1 shows the preliminary results of the various stages of the education and training of the Medical Physicist in Europe according to EFOMP Policy Statement No.12 using the eight levels of the EQF for lifelong learning.

The level of graduation (Master's level) of the Medical Physicist corresponds to level 5 of the EQF for lifelong learning and the level of the QMP correspond to level 6. The level of the SMP corresponds to level 7.

Level 8 of the EQF for lifelong learning goes beyond the EFOMP Policy Statement No.12 and it corresponds to the Medical Physicist holding a Doctoral Degree (PhD) or higher, working in a research environment (University or Research institution).

EFOMP needs to develop a Policy Statement to bridge the gap between level 7 and level 8 of the EQF.

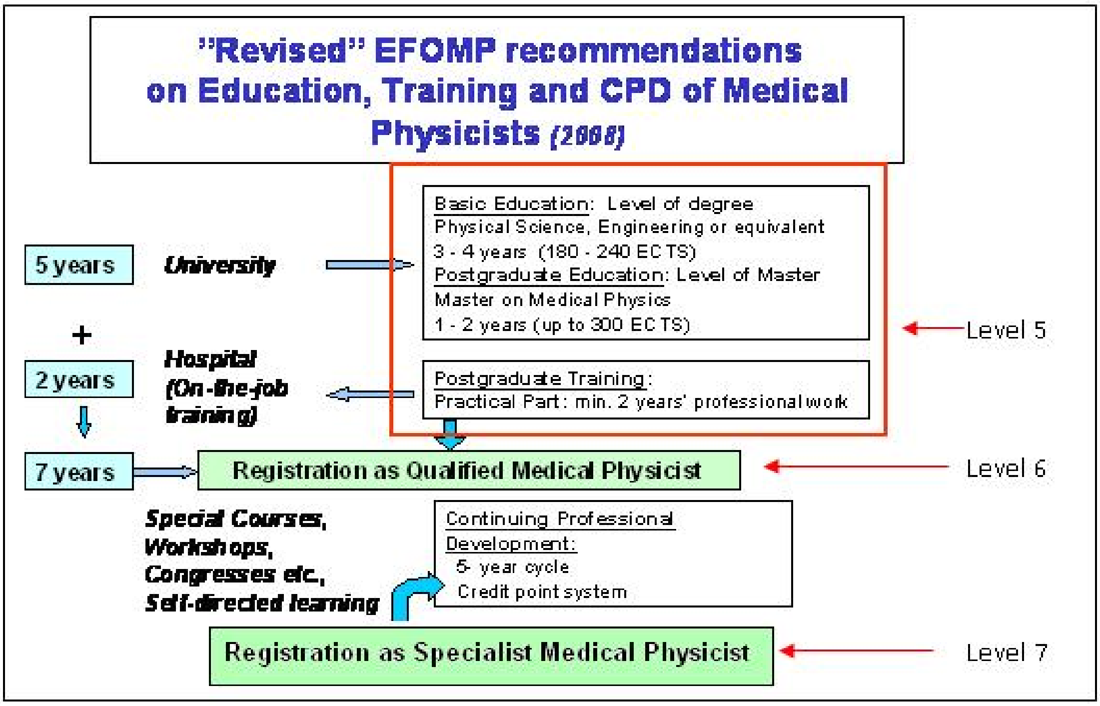

Fig. 1. The various stages of Education and Training of the Medical Physicist in Europe according to EFOMP Policy Statement No.12 using the 8 levels of the EQF for lifelong learning.

Medical Physics as a Profession

In due consideration of the following facts, EFOMP believes that Medical Physics should be recognised by the European Union as a regulated Health Profession in accordance with the EU Directive on the recognition of professional qualifications (EC, 2005):

a) Medical Physicists have a formal education and training in Anatomy, Physiology and Radiation Protection as applied to medical activities.
b) Medical Physicists have the necessary skills to manage the quality of the performance of the equipment used in hospitals.
c) Medical Physicists have a relatively long practical training in Hospitals.
d) Clinical professionals regard medical physicists as invaluable specialists who facilitate the safe use of radiation in hospitals.
e) Quality Assurance and Quality Control in Radiotherapy, Nuclear Medicine and X-Ray diagnosis is, normally, done by medical physicists. The results of these activities have clear implications on patient diagnosis, treatment and safety.
f) Medical Physicists have the experience to perform basic and applied physics research in medicine.
g) In a number of European Countries Medical Physics is a profession regulated by law.

There is more than enough evidence to prove all the above statements apart from two very important issues:

a) The education and training of the medical physicist is not harmonised across Europe according to the EFOMP Policy Statement No. 12.
b) The Medical Physics Profession is not regulated in all European Member States.

These two issues can only be resolved by the individual Member States by applying the EFOMP recommendations.

EFOMP participation in European Projects

The importance of medical physics in the area of radiation protection in medicine and the lack of uniformity in training and education has led the European Commission to commission research projects with aim the ultimate definition of the MPE with the required education, training and competencies. To this end, the Commission has tendered two research contracts in both of which EFOMP is a partner. These are:

- the European Medical ALARA Network –EMAN (Contract No: TREN/09/NUCL/SI2.542127), and
- the European Guidelines for the Medical Physics Expert (Contract No: TREN/09/NUCL/SI2.549828).

The contribution of EFOMP to these projects was considered essential by the European Commission and it was a condition that EFOMP should be represented in all of their work packages.

Conclusions

By accepting the fact that the role, responsibilities and competences of the MPE, RPE and RPO are functions to be undertaken by recognised professionals and that this recognition will be according to the complexity of the modalities of any given healthcare facility, then the new European Basic Safety Standards Directive can be implemented by the European Member States in a more harmonised way so facilitating the mobility of these professionals within the European Union. Additional guidance will be required to assist in the implementation and interpretation of the provisions of this directive.

Some of the activities of EFOMP in the area of Education, Training and Professional Development of the European Medical Physicist, with emphasis on Radiation Protection have been discussed.

These activities can only materialise through the collaboration of all the Medical Physicists of EFOMP's National Member Organisations (NMOs). The NMOs must actively adopt and implement the guidelines of the policy statements as well as participate in the various events organised by EFOMP in collaboration with its NMOs.

The contributions of all interested parties are more than welcome in order to further develop the harmonisation of the education, training and professional status of the Medical Physicist in Europe.

Acknowledgments

EFOMP acknowledges all those that have contributed to the development of its policy statements as well as all those involved in the organisation of educational and training events under its auspices.

EFOMP also would like to express its sincere thanks to all those that through the years have contributed actively in promoting the Profession of Medical Physics.

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Computer based radiation protection course for workers in the health care sector

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Abstract

Radiation protection aspects in the health care sector are a primary concern due to the very different kind of occupational activities and to the large number of people involved with ionising radiation (I.R.), for instance in a large hospital.

The Government of the Tuscany Region (Italy) has promoted the realisation of a computer based radiation protection training course for all I.R. workers of the National Health Service within the Tuscany region.

The challenging goal of this project is to provide the basic safety information in such a complex field, where people with very different education levels and duties do work together, with the aim of fulfilling the specific educational requirements of Directive 96/29/EURATOM as introduced in the Italian legislation.

The course is devoted to provide radiation protection education to all health workers, with special attention to those without high level of education in the I.R. field (i.e. physicians outside the radiology area, surgery room staff, nurses in nuclear medicine or radiotherapy and laboratory staff).

The course is composed of five sections dealing with the general aspects of radiation protection (basic I.R. physics, biological effects, regulatory system, dosimetry and radiation protection principles) while four other sections deal with sector specific radiation protection aspects: radiology, nuclear medicine, radiotherapy and laboratory. A further special chapter, summarising all aspects treated in the course, is devoted to workers with poor educational level or no-background in the field of physics, radiation protection and current legislation. Each section includes a multiple choice test with hint function, numerical examples, a glossary and bibliographical references.

First feedback outcomes and multiple choice tests results from hospital workers are also reported.

Introduction

According to EC regulation, all persons whose work may be associated with ionising radiation risk must be adequately educated to ensure that they are informed about the potential health risks which could result from radiation exposure, the basic principles of radiation protection and the relevant radiation protection regulations as well as safe working methods and techniques in radiation zones.

Radiation protection (RP) aspects in the health care sector are a primary concern due to the very different radiation sources, kind of occupational activities and large number of people involved with ionising radiation (I.R.), for instance in a large hospital.

The Government of the Tuscany Region in Italy has promoted the realisation of a computer based training RP course for all I.R. exposed workers of the National Health Service (N.H.S.) within the Tuscany region. The project was developed by the Health Physics Department of the Florence University Hospital (Azienda Ospedaliero-Universitaria Careggi) as leading institute, in collaboration with the other Health Physics Departments of the Tuscany N.H.S. The course is also open to contractors' personnel as complementary information in addition to the RP training they must receive from their own employers.

The main challenge of the project is to provide the basic safety information in such a complex field as the health care sector, where people with very different education levels and duties do work together (e.g. in a radiological interventional room).

In Fig. 1, the distribution of Tuscany region health care workers exposed to I.R. is shown. In the "Others" group are included, biological lab staff, logistics and cleaning staff and medical physicists A total amount of roughly 6000 I.R. exposed people work in the NHS of the Tuscany region, servicing a population of about 3.7 million inhabitants. In Fig. 2, the Tuscan NHS exposed workers distribution between health care activity sectors is reported.

The goal of the project is to fulfil the specific educational requirements of Directive 96/29/EURATOM as introduced in the Italian law.

In this paper we present a description of the course as well as an evaluation of the course effectiveness as resulting from the outcomes of the multiple choice tests.

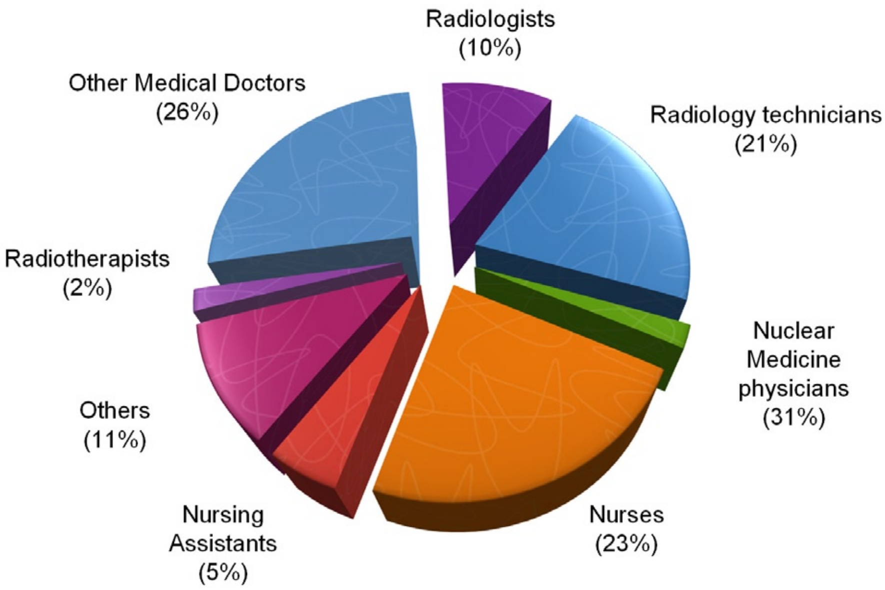

Fig. 1. Distribution of exposed workers, according to their professional role, in Tuscany region health care sector.

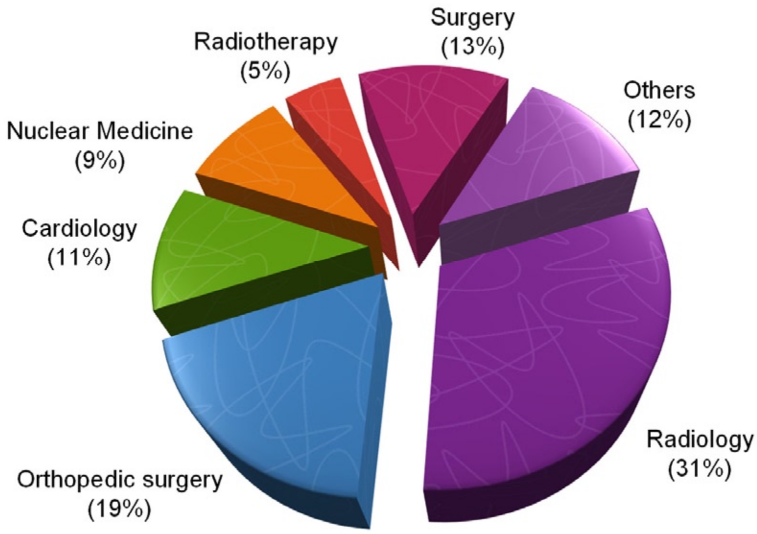

Fig. 2. Distribution of exposed workers according to medical specialties in the Tuscany region.

Material and methods

The course is addressed to all people working in the health care sector, with special attention to workers without high level education in the I.R. field (medical doctors outside the radiology area, surgery room staff, nurses in nuclear medicine or radiotherapy departments, laboratory staff, etc.)

Course description

The main course is composed of a few sections dealing with general aspects, covering the main topics of radiation physics, biological effects of I.R., national regulatory system, dosimetry and RP principles. The other sections deal with the specific aspects of RP in radiology, nuclear medicine, radiotherapy and laboratory. A special section, summarising all the fundamental notions of the course, is devoted to workers with poor educational level or no-background in the field of physics, radiation protection and related legislation, that is: hospital auxiliary staff, workers belonging to external service providers (i.e. cleaning services) and non-health (i.e. logistics) workers. In this section, each sub-section contains detailed information on how to act (things to do/ not to do) and a list of Frequently Asked Questions (FAQs) with related answers.

The main aspects of safety procedures, definitions and health hazards are stressed through a series of numerical examples, pictures and warning text boxes spread out in each chapter. Moreover, besides those data coming from international/national literature and reference publications (e.g. ICRP, IAEA Publications), lots of the data reported in tables and/or used in examples come from direct measurements carried out in several hospitals within Tuscany region in the typical operating conditions.

A summary of the course content, divided in chapters and sections, is reported in Table 1.

Table 1. Radiation protection course content.

Section	Sub-sections
1. Ionizing radiation (I.R.) principles	1.1. Atomic structure 1.2. Ionizing radiation 1.3. Sources of I.R. 1.4. Radioisotopes 1.5. Artificial radiation sources 1.6. Basic physical quantities and units
2. Biological effects of I.R. and epidemiological information	2.1. Radiation Interaction with cells and tissues (deterministic and stochastic effects) 2.2.Epidemiological information and radiation protection
3. Radiation dose and its measurement	3.1. Radiation dosimetry 3.2. Basic dosimetric quantities 3.3. Dose measurement 3.4. Personal dosimetry service
4. Introduction to radiation protection	4.1. The radiation protection principles 4.2. Types of radiation exposure, radiation hazard warning signs 4.3. Dose reduction principles 4.4. Protection devices
5. Radiation protection regulations	5.1. Introduction 5.2. The basis of radiation protection regulation 5.3. Italian national regulation 5.4. Classification of workplaces 5.5. Classification of workers 5.6. Limitation of doses 5.7. Employer's duties 5.8. Workers' duties 5.9. Special protection during pregnancy and breastfeeding
6. Radiation protection in diagnostic and interventional radiology	6.1. Radiation sources and risk sources 6.2. Hazard Assessment 6.3. Radiation safety measures 6.4. Local rules and operational procedures
7. Radiation protection in Nuclear Medicine	7.1. Radionuclides for diagnostic uses 7.2. Radionuclides for therapeutic uses 7.3. Hazard Assessment (External exposure, Contamination and internal exposure) 7.4. Radiation safety measures 7.5. Local rules and operational procedures 7.6. Decontamination procedures 7.7. Handling of radioactive wastes
8. Radiation protection in Radiotherapy	8.1. Radiation Sources (External beam radiotherapy, Brachitherapy) 8.2. Hazard Assessment (External Beam Radiotherapy, Brachitherapy) 8.3. Radiation safety measures 8.4. Local rules and operational procedures 8.5. Biological irradiators
9. Radiation Protection in clinical analysis and biomedical research laboratories	9.1. Radiation Sources 9.2. Hazard Assessment (External exposure, Contamination and internal exposure) 9.3. Radiation safety measures 9.4. Local rules and operational procedures 9.5. Decontamination procedures 9.6. Handling of radioactive waste
10. Radiation protection for non experts	10.1. Ionizing radiation 10.2. Sources of ionizing radiation 10.3. Biological effects of ionizing radiation 10.4. Dose measurement 10.5. Introduction to radiation protection (Classification of workers, Classification of workplaces, Radiation hazard warning signs, Dose reduction principles, How to prevent contamination) 10.6. Radiation protection regulations (Incl. Workers' duties) 10.7. Radiation protection in diagnostic and interventional radiology 10.8. Radiation protection in nuclear medicine 10.9. Radiation protection in Radiotherapy 10.10. Radiation protection in laboratories

Each section includes a multiple choice test, a glossary, a reference list and a "Focus on" sub-section for further information on specific topics.

The entire radiation protection course counts about 150 web pages, and about 6 hours of study are estimated being necessary in order to proficiently acquire the basic knowledge and to become able to correctly answer the test questions. A learning time of 3 hours is estimated for not experts workers, as they are supposed to read the section "Radiation protection for non experts" only.

A web based user interface was chosen to take advantage of the flexibility, in terms of information search and retrieving, hyper text features, document storage capability and eventual future upgrading. Other advantages are inexpensive distribution, reduced technical support and cross platform delivery.

The course provides additional learning tools, such as:

- interactive glossary: in order to make learning easier. When passing the pointer over a term defined in the glossary, a "mouse over" function interactively opens a box with that term definition
- hint function for the multiple choice tests: in case of wrong answer a pop up window linked to the web page containing the right information is opened
- PDF documentation of main national (i.e. Italian) regulations concerning exposure to ionising radiation
- links to external web sites of prominent international radiation protection committees and agencies
- bibliographic notes and links
- links to complementary material and curiosities related to radiation exposure.

In a first stage the course is distributed cost free as an interactive CD-ROM to all radiation workers in Tuscany.

In Fig.3 and Fig.4 the screenshots of two pages are shown as example.

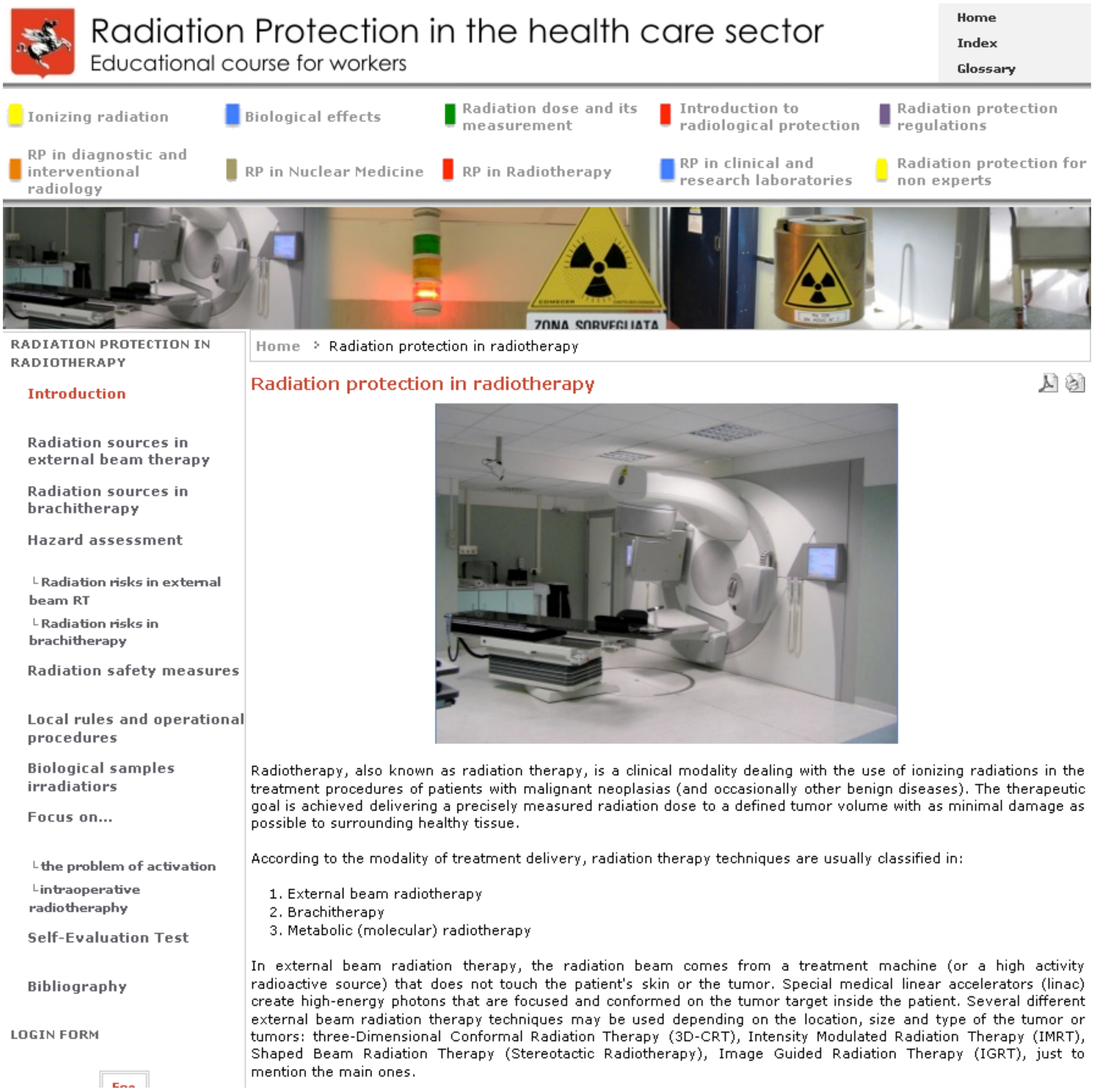

Fig. 3. Screenshot of a course page, showing the web user interface.

Course evalutation

A release candidate version of the course has been made available in February 2010 either on CD-ROM and online to a testing group of 49 workers for evaluation. The workers engaged in the testing group have been selected according to their competence profile. The testing group composition is the following:

- 12 radiological technicians
- 14 professional nurses in a radiological area
- 5 physician specialized in a radiological area
- 12 physician not specialized in a radiological area
- 6 non experts (2 cleaning staff and 4 auxiliary staff)

Each tester was asked to read only the group of chapters appropriate for its job. A test with multiple choice questions concerning RP topics was given before and after the reading of the content. The course has been submitted to the testers along with an assessment and satisfaction questionnaire concerning the relevance of the subject treated, the quality and usefulness of the information presented, the clearness of the explanations and in general the usability of the course and of the learning tools available (more details in the *Results* section).

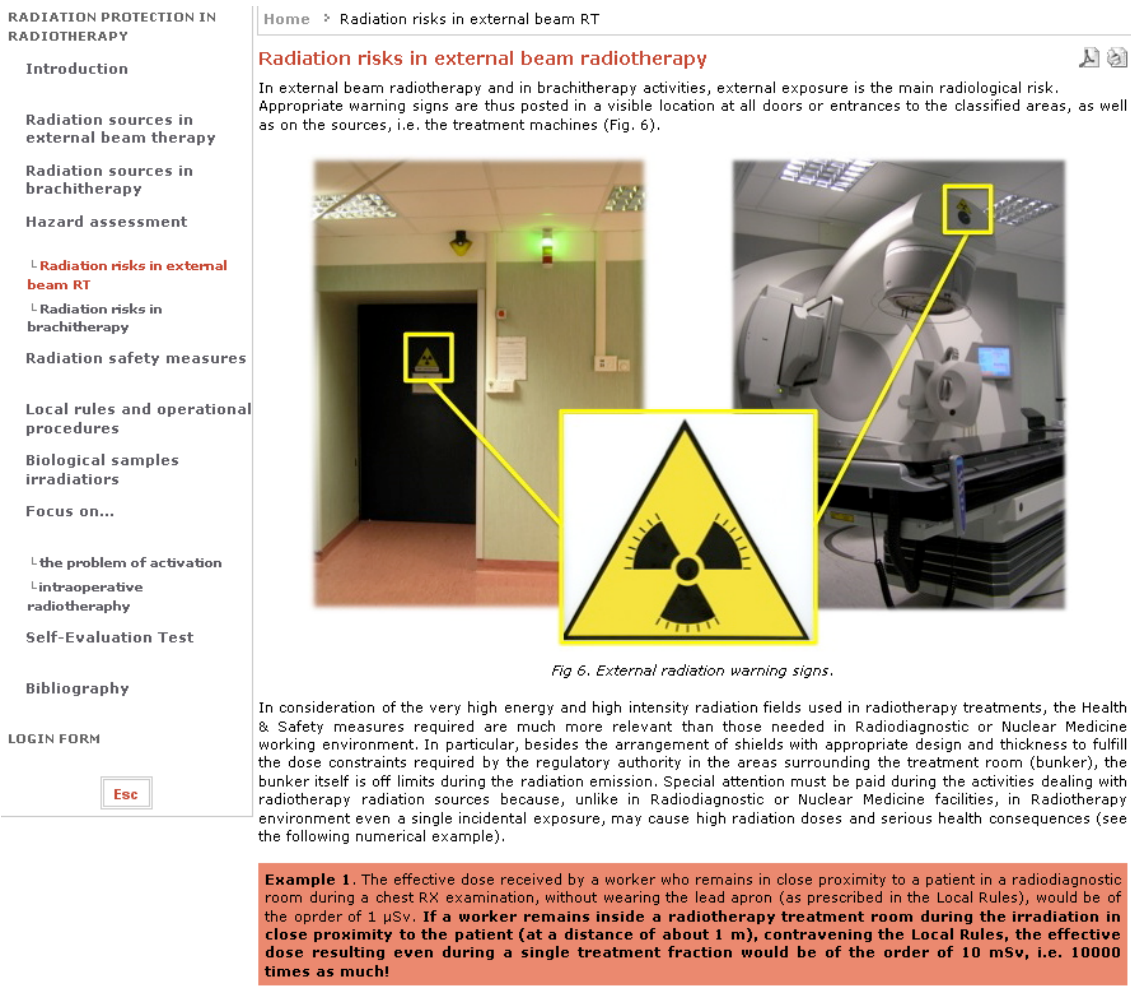

Fig. 4. Screenshot of a course page containing a numerical example.

The effectiveness of the course has been evaluated by comparing the results from the tests performed before and after the course delivering.

Results

The overall results of the satisfaction questionnaire is reported in Fig.5, together with the investigated aspects. Users were asked to express a four steps judgement (from “Insufficient” to “Very Good”) to rate the course educational features.

The multiple choice test results, expressed as wrong answer percentage, are shown in Fig. 6 for the tests performed before and after the course delivering. The educational effectiveness is reported for the different professionals groups and as average.

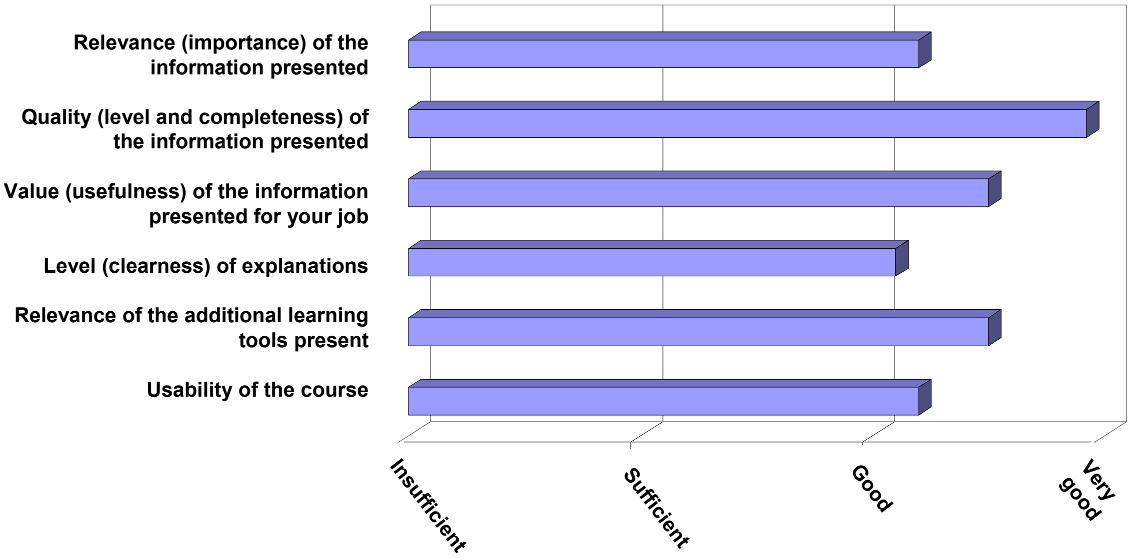

Fig. 5. Synthetic outcome of the assessment questionnaire returned by the workers of the testing group.

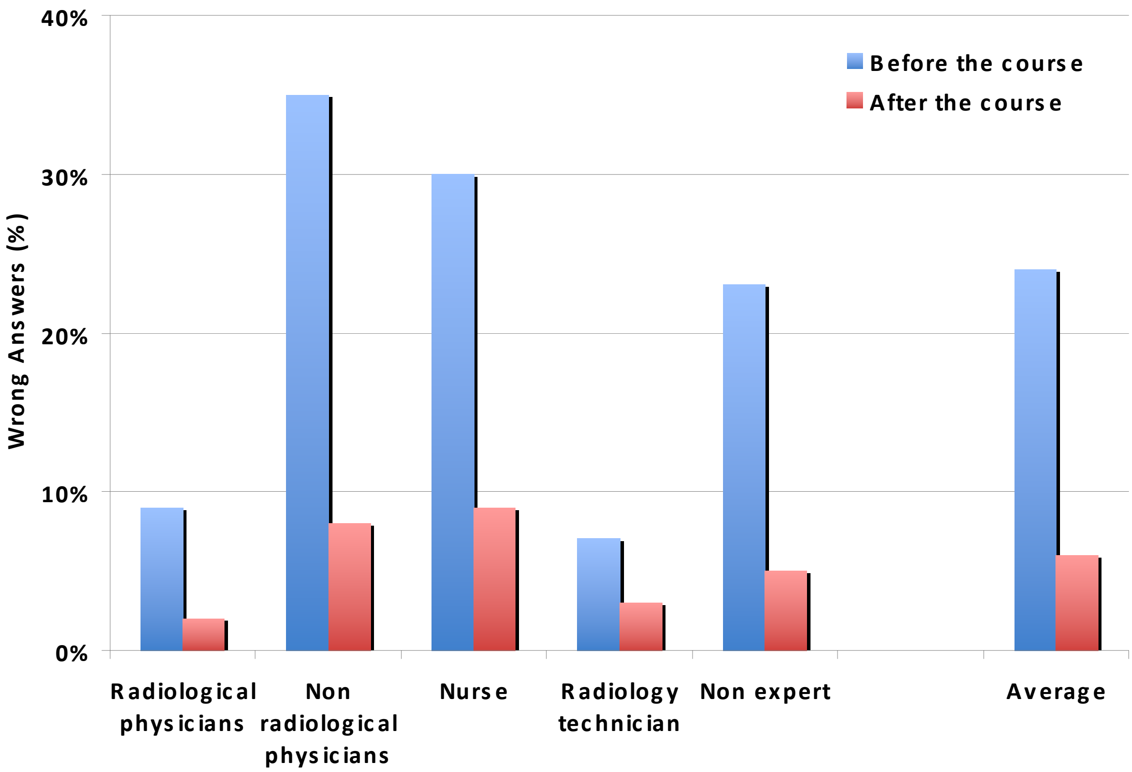

Fig. 6 Multiple choice tests results (expressed as wrong answers percentage) for the different professional groups.

Conclusions

A computer based radiation protection course for all radiation exposed workers of the National Health Service within the Tuscany region, Italy, has been developed. The main challenge of the project is to provide the basic safety information in such a complex field as health care sector, where people with very different education levels and duties work together. The course is addressed to all people working in the health care sector, with special attention to workers without high level education in the I.R. field (medical doctors outside the radiology area, surgery room staff, nurses in nuclear medicine or radiotherapy departments, laboratory technologists, etc.) The main course is composed of a few sections dealing with the general aspects, including basic radiation physics, biological effects of I.R., national regulatory system, dosimetry. Other sections deal with the specific aspects of RP in radiology, nuclear medicine, radiotherapy and laboratory. A special section, summarising all aspects treated in the course, is devoted to workers with poor educational level or no-background in the field of physics, radiation protection and current legislation concerning the exposure to ionising radiation.

First feedback outcomes and multiple choice test results from hospital workers are also reported. The response of first users resulted quite positive according to the questionnaire and the educational content proved to be well suited for the intended target user.

In a close future the course is going to be implemented as a Web Based Course to make it accessible to a larger number of workers, possibly outside the Tuscany region, on an e-learning platform. In the latter case the course could become part of the institutional CPD programme in any hospital.

e-Learning in DAP-measuring

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Abstract

According to studies lower limits for radiation dose setting can be established for computed radiography (CR) systems to produce diagnostic quality images than the standard values used for film/screen systems. This is especially important in examinations performed for the children. Flat panel Detectors (so called direct radiography, DR) gives even more possibilities for dose reduction and still having at least as good image quality as with CR. One of the most important quality assurance procedures is to evaluate the doses delivered to patients who undergo x-ray examination. The DAP-meter is a real-time patient dose monitoring system for auditing patient doses. Technical data from each exposure and for every examination type can be collected. Radiography students and lecturers of Metropolia University of Applied Sciences produced an e-based learning module about DAP-measuring in co-operation with Metropolia degree programme in multimedia technology and Finnish Radiation Protection Authority. In this presentation the contents and pedagogical solutions of this module are presented. Also the development process of this learning module is depicted. The educational module consist of articles to be read, different kinds of tasks to be solved e.g crossword about central concepts of dose-area product-meter (DAP). The group also made a film that shows how to measure patient doses with DAP when using digital radiography (DR) technology.

e-Learning philosophy and structure for a Nordic education project in evidence based radiography

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Abstract

Helsinki Metropolia University of Applied Sciences has initiated a Nordic project to increase competence of radiographers by evidence-based training in quality in radiographic imaging. The members of this project are radiography lecturers, principal lecturers and physicists from universities in Finland, Norway and Sweden. The goals of the project are to produce curricula and learning materials using internet solutions for ground-, master and PhD-level web-based education and finally to implement the education and evaluate the training program and the materials produced. The main contents will be evidence-based digital imaging, dose optimization and quality assurance of digital imaging systems. The main principle for the education project is that the way of training as well as the contents will be evidence-based. In according to this the members of the project group apply evidence-based way of working in planning the education and in the contents. To give the students understanding, knowledge and tools for working evidence-based, the training program will start with an introduction to evidence based knowledge and its use in radiography, which is elementary in the whole education. The training program applies various web-based methods depending on the subject and the expected learning outcomes. In some cases the method used will be group activities like lectures by videoconference, face to face meetings on the web and discussions in groups synchronously or asynchronously via Wiki-based IT-solutions. Some subjects will be suitable for individual studies by self-paced e-learning online e.g. when searching articles or offline e.g. in learning dose area product measuring or other procedures. Students from the participating universities are even involved in the project from the very beginning e.g. by writing thesis for the needs of the project.

Developing a radiation protection culture at school

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Abstract

Since September 2007, actions to develop a radiation protection culture have been undertaken in several schools in cooperation with experts from the Institute of Radiation Protection and Nuclear Safety (IRSN), the Nuclear Protection Evaluation Centre (CEPN) and the centre for scientific culture (Pavillon des Sciences – Franche Comte). The aims are to provide school-students with scientific and social bases on radiation protection in order to be better prepared for dealing with societal issues, to promote a scientific and technical culture, and to allow the students to better understand the "world" of radiation protection.

Each year, this approach includes two steps:

One part is dedicated to training during the school-year; each school works on different topics chosen according to local concerns (e.g. radon in dwelling, medical radiation protection, radioactivity in the environment, radiobiology…). In order to facilitate the development of this culture, practical aspects are favoured, notably with experiences and visits to laboratories, hospitals, … Special attention is given to multi-disciplinary approach including physics, chemistry, biology, philosophy, geography, economy… This task is lead by school teachers in cooperation with experts.

The other part consists in participating to a "student workshop" allowing them to present their work and to exchange with other students and experts. In 2008, the first workshop was held in Montbeliard (France), the second one, involving French as well as German, Ukrainian and Belarussian schools, was held in Poitiers (France) in 2009 and the third one was held in Paris in March 2010, involving more than 200 participants from French and foreign schools (Belgium, Belarus, Germany, Italy and Ukraine).

This paper details the actions developed during the last years and discusses the lessons drawn on the way to address radiation protection with young people combining scientific issues and social concerns.

Introduction

There is an increasing trend to involve stakeholders in radiation protection management. Some issues will be of particular importance in next years: radioactive waste management remains a sensitive issue, new development of nuclear energy is anticipated, medical exposure increases drastically, management of radon exposure needs to be improved… However, members of civil society need basic knowledge in radiation protection together with practical experimentation to be able to improve their level of protection regarding radiation exposures, to express their concerns and expectations on these issues and to play a role in the related decision making processes.

Therefore, Institute of Radiation Protection and Nuclear Safety (IRSN), Nuclear Protection Evaluation Centre (CEPN) and the "Pavillon des Sciences de Franche-Comte" (a centre for scientific and technical culture) decided in 2007 to initiate a pilot programme with some high schools (lycee) in France.

Material and methods

The programme is based on cooperation between school professors and radiation protection experts either from IRSN, CEPN, universities or environmental NGO.

In order to develop multidisciplinary approach, professors teaching biology, physics, philosophy, arts … are involved. School students, from 15 to about 19 years old, participate, on voluntary basis, in small group(s) up to 20 students in each high school.

Each school year, the programme includes two parts: the first one concerns practical experiments performed by school students with their teachers and the second one consists in the organisation of a workshop, involving all the schools, held in Spring.

The following topics are proposed: biological effects of radiation, cancer epidemiology in exposed population, radiation detection, radon exposure, radioactivity in the environment, medical use of radiation, ethics…

Each high school chooses topics, according to regional or personal interest.

For each topic, a radiation protection expert is identified on voluntary basis to accompany professor and students.

Experiments are designed for the students to cope with radiation protection bases through practice and to identify the issues at stake for the management of radiation protection in specific situations (mining residues areas, environmental surveillance around nuclear installations, radon management in dwelling, radiation protection at hospital,…). Protocols for experiment are defined in order to avoid risks for participants.

Figure 1 summarizes the approach adopted for developing a radiation protection culture at school.

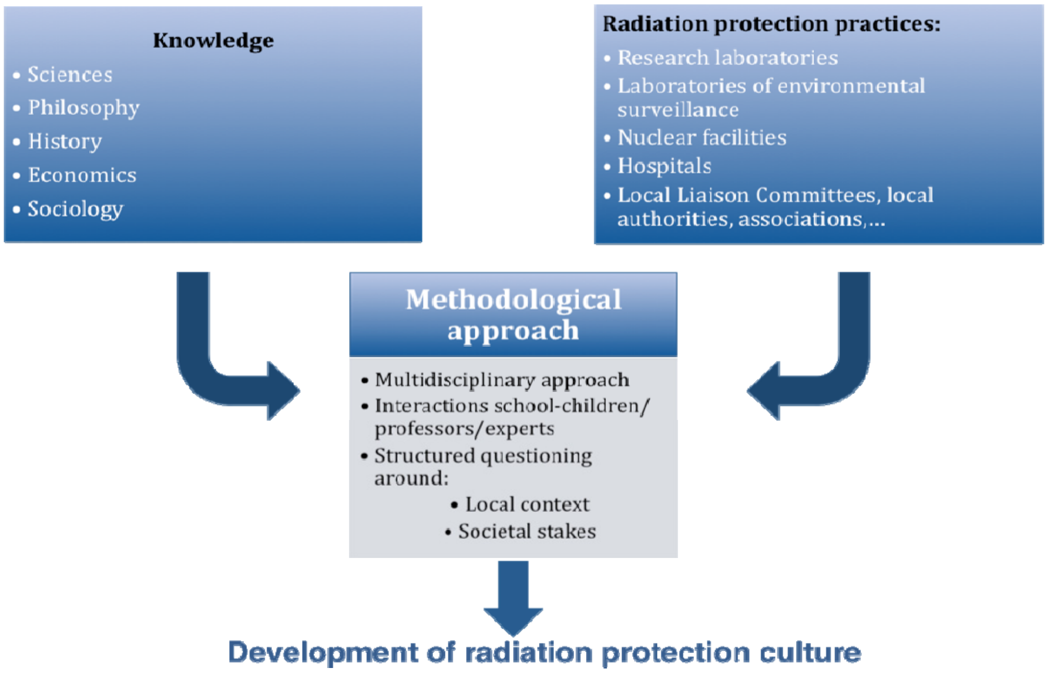

Fig. 1. Approach adopted with schools.

Besides, instruments needed must be either simple ones used in high school (colorimeter, computer, microscopes…) or tools which could be lent by experts (portable dosimeters…).

In most cases, first step is a meeting between professor(s) and expert to define experimental protocol. As second step, expert makes a short presentation of the topic to the students. Then, the experiment is performed by students with their professor, the expert remaining as support if necessary. Often, this step is followed by a technical visit of a unit specialized in the corresponding field; it allows students to discuss their results and work performed in the unit. Visits and discussions are also organised with practitioners form nuclear installations, surveillance laboratories, elected people, local liaison committee members around nuclear installations, …

The Workshop is an opportunity for students to present their work and exchange with the other students, but also with experts. It includes thematic sessions with oral presentations by students, as well as some lectures by experts on special topics. The Workshop programme also includes technical visits and entertainments.

Results

The programme has now been operated for three years. Except for special circumstances (retirement of school professor…), high schools remained involved through the years.

The programme was initiated with French high schools. From second year, schools form Belarus, Ukraine and Germany were involved, and during the third year schools from Italy and Belgium joined the programme, although the language used was French to facilitate exchanges.

For the year 2009-2010, 14 schools and about 230 persons participated to this action. Figure 2 presents the evolution of the participation to this action.

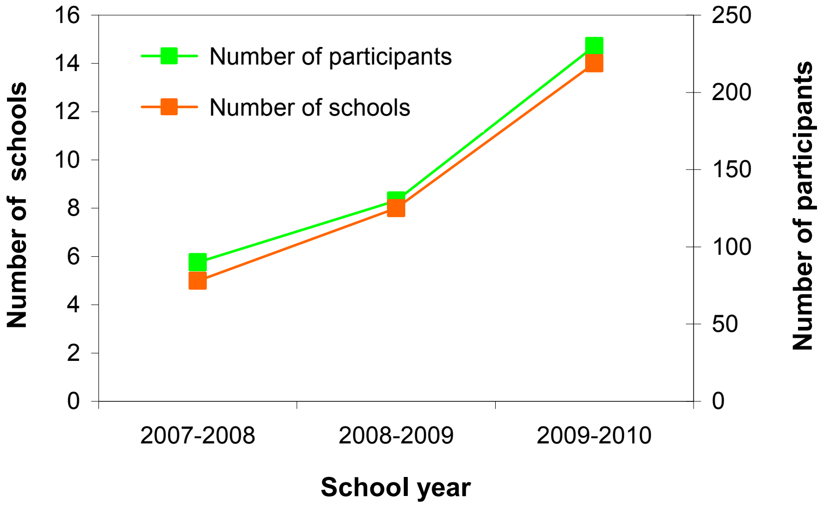

Fig. 2. Evolution with time of the numbers of schools and participants.

Each year, over 20 experts, who are technicians, engineers or researchers, participated to this action, with a large participation from IRSN.

From practical experiments, students learned about physical, biological, regulatory… bases in radiation protection. For instance, biological dosimetry experiment was organised to study effects of radiation. Students had to identify and count chromosome aberrations (dicentrics) on metaphase spreads from blood samples irradiated at various doses; after defining calibration curve, they had to evaluate dose on an unknown sample. From chromosomal aberrations, students learned about radiation induced DNA damages, cell death… Search of hidden sources with different dosemeters or analysis of tissue proportional counter spectra gave them an opportunity to study, for different radiations, interactions of radiation with matter, radiation detection, energy deposition at micrometer level... Students exposed radon dosimeters in various conditions: basement, living rooms, caves…; analysis of results gave them the opportunity to learn about radon exposure. To understand radionuclide release in the environment, students took samples in environment surrounding a nuclear facility, these samples were then analysed in laboratories.

In addition, the multidisciplinary approach together with the interaction with different experts and stakeholders give the opportunity to engage a reflection on the societal issues associated with different radiation protection practices. Inquiries have been performed by students around nuclear installations for example on the type of information available for the population. A measurement campaign has been organised with an environmental NGO in addition to the visit performed in a nuclear installation. Students living in territories contaminated by the Chernobyl accident in Belarus and Ukraine worked on the memory of the accident and the current consequences in the day-to-day life.

Fig. 3. Schools students searching for hidden sources.

The Workshop is an opportunity for the students to present their work, share the knowledge they had acquired, express their view on the societal issues at stake, exchange about their experience and discuss with experts.

In addition, few experts gave lectures on topics such as recent discoveries in radiobiology, SOCATRI accident in France, sources of exposure to ionizing radiations of French population, management of post-accident situations…

Fig. 4. Student lecture and presentation during workshop.

Discussion

Each year very different topics were selected, but interestingly radon, which is not generally considered as a sensitive issue by population, was over selected every year.

Since each group cannot work on all topics, part of the bases are not acquired directly, but though exchanges with other students. It would be interesting to check how far they are able to understanding the technical and scientific aspects of radiation protection issues as well as their capability to address societal issues at stake with different radiation protection practices.

So far, experts who participated to this programme are enthusiastic about sharing their knowledge and experience, but they are almost exclusively issued from institutions. It would be interesting to widen the spectra by involving more members of civil society.

One important condition is to work on voluntary basis to involve highly motivated professors, students and experts.

It is interesting to underline that this approach could contribute to favour the participation of citizen to the debate on radiation protection issues. Besides radiation protection purpose, this programme could also promote scientific careers at a time when interest for this field is sharply decreasing in France.

Conclusions

This pilot programme has been rather successful, in demonstrating that scientific and technical bases of radiation protection, which are known as complicated, can be rather easily acquired by school students. It has also demonstrated the interest of addressing the radiation protection issues with practical experiments and within local contexts. The challenges are now to adapt the programme to cope with anticipated increase of participants, to define a charter to precise conditions of participation, to involve a wider panel of experts and to further investigate the best approach for the diffusion of the radiation protection culture for school students.

How to share with children the basic knowledge about radioactivity and nuclear Risks?

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Abstract

This paper presents the strategy of IRSN to increase the basic Knowledge of young of children about radioactivity and nuclear risks in partnership with associations. 58 reactors are built on the French territory but a review performed by M Vroussos in 2004 highlighted that French people have a very poor knowledge about radioactivity and nuclear risk. This was confirmed by the Eurobarometer surveys. The consequences of such a low background knowledge should be, first, that the public cannot reduce the doses due to natural, medical exposition. Second, in case of nuclear event (example Tricastin 2008) or nuclear accident, the behaviours of concerned people and French consumers could create strong social and economical effects due to inadequate representations of the accident impacts. Third, in a more general way, citizens have few knowledge to discuss the nuclear renewal.

In order to diffuse a better knowledge IRSN and IFFO RME decided to have a partnership to develop tools for teachers and children. The Gafforisk fan was the first tool, with the objective to give fundamental knowledge about natural radioactivity, nuclear reaction, accident fallout, crisis management with a special emphasis on behaviour recommendation in case of nuclear crisis.

The tool was developed in a mobile exhibition which can be used by local information commission (CLI), for example during crisis exercise. The paper will present in detailed the strategy, the partnerships the different tools and their use.

Introduction

In 2004, after a large review to reinforce radioprotection in France, in his report asked by the French Nuclear Safety Authority (ASN), Pr Vroussos concluded that French people have a very poor knowledge about radioactivity and nuclear risk and he suggested main ways to improve Radioprotection. One of them is “to invent, or at least to experiment, new modes of information and dialogue with the population to share the knowledge and try to establish a dialogue between the citizen and the services in charge of the control, and to allow each one to form its opinion, in particular on the question of the risks connected to the use of the nuclear energy”.

To make it in a pragmatic way, he recommended "a strong action with the Ministry for Education so that the programs of the secondary education integrate the physical and biological bases of the effects of ionizing radiations, their diverse applications and the radioprotection within the citizen-centred approach to the environment topics and to the sustainable development".

In its strategic plan for 2010-2012, the ASN makes a commitment to associate more widely the public with the process of decision-making and to explain its decisions. For it, it would be necessary to increase the level of information of the public. According to the Eurobarometer about nuclear Waste (2005) [1], 22% of French people declared not to be informed on nuclear waste, and the knowledge on this topic was tested by 5 questions. As a result, France is eighth (60% of good answers), three first ones are Slovenia (72%), Finland (68%), Sweden (68%), countries where the number of nuclear installations is less than in France. This can be explained by the very long period (until 2006) during which many documents and recommendations on safety issues were not available to the public, thus without debates with the society.

The consequences of such a low background knowledge should be (1) in a more general way, citizens have few knowledge to discuss the nuclear renewal. (2) the public is not able to reduce the doses due to natural, medical radioactive risks to which they expose themselves and, (3), in case of nuclear event (example Tricastin 2008) or nuclear accident, the behaviours of concerned people and French consumers create strong social and economical effects due to inadequate representations of the accident impacts. Because, as said by M Covello in the conference in Bethesda in March 8, 2010 about risk perception, "when people are stressed and upset, the gap between perceptions and reality often becomes wider.", "perception equals reality", "that which is perceived as real is real in its consequences.".

For these reasons, the French Institut de Radioprotection et de Surete Nucleaire (IRSN) wondered about the best way for developing a knowledge on the nuclear power and the radioactivity in particular towards the school pupils, the citizens of tomorrow. Recognizing that the pedagogy was not its field of expertise, IRSN decided to have a partnership with the French Institute of Trainers at the Major Risks and Environment (IFFO RME) [2]. In this association, half of the 800 adherents are voluntary teachers to educate their pupils at the major risks (natural and industrial risks also too little handled in course of study), other are coming from French public services involved in risk management or risk specialists.

The most challenging aspects for education

The first step of the partnership was to well identify the most important challenges of the common work.

Nuclear accident seems to be the topic where the knowledge is the lowest and with the most important consequences on society if it occurs. IRSN risk barometer 2009 [3] shows that French people consider the accident in nuclear plant as the most important potential risk (fig 1). But they have little knowledge on it and on the risk management in such a crisis. They don't know who give information. In this area, authorities have a poor credibility [4] (fig 2), these verbatim are very frequent: «they lied to us. The cloud of Chernobyl stopped on the border", "they do not say to us everything ","If we do not die at once, we shall die from cancers ", "they speak about nuclear safety but not about risks". This shows that in the representations of the population on the nuclear risk, Chernobyl consequence and its management in France takes an important place and contributes nevertheless to a certain consciousness of the risk..

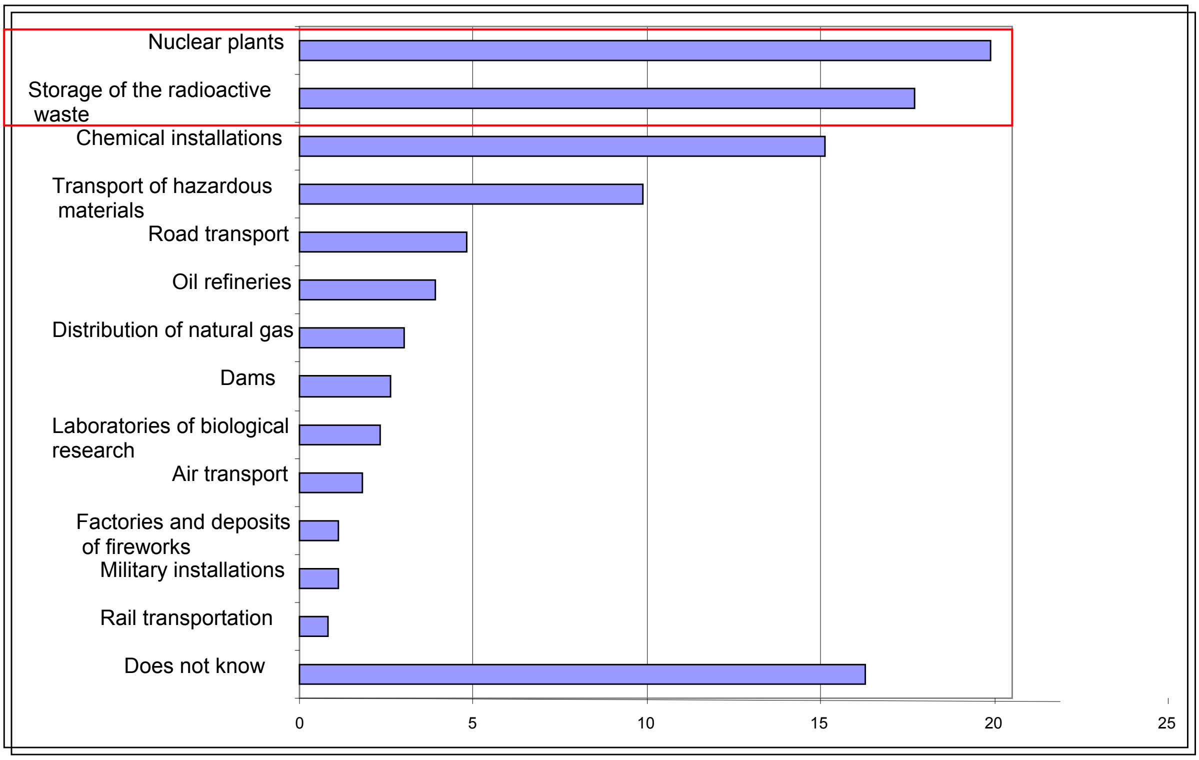


Fig. 1. IRSN risk barometer 2009 : Percentage of answers to this question " Among the diverse industrial or technological activities following ones, which are the ones which risk most to provoke a grave accident or a disaster in France? " (3 possible answers)

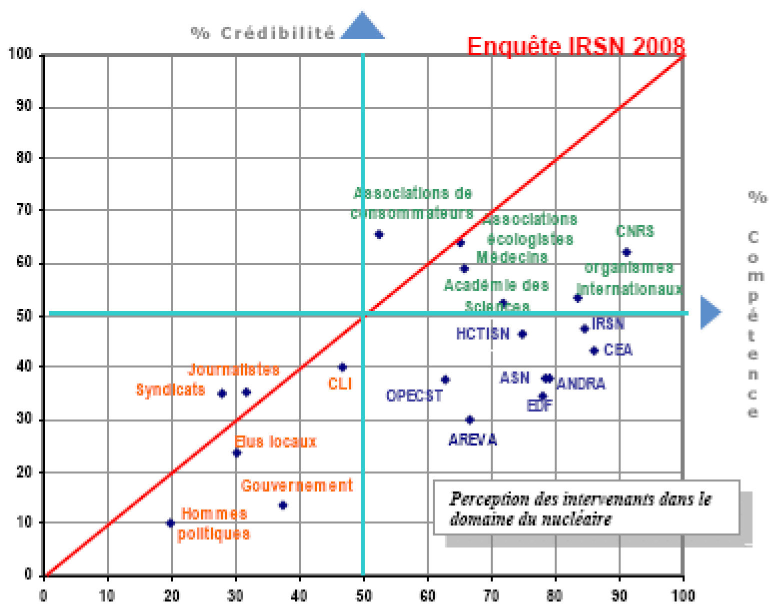


Fig. 2. " This picture give the rate of credibility and the rate for competence given to different actors. This shows that the credibility of ASN (nuclear safety authority) and IRSN is less than the average and the greater credibility is given to scientist and ecologist associations. "
From the IRSN barometer 2008

More, the population knows rather little the major risks which could occur as well as measures of prevention and protection they have to use in a crisis. Besides, the partners recognize that without a minimum of knowledge about a topic, it is impossible to ask questions and participate as a citizen in a nuclear debate. So, two major difficulties were identified to change the tendency: it is needed to popularize a scientific and technical domain (viewed as an elitist topic) and to take the heat out of the debate, without which we cannot build of information and educational.

So, for that, other partners join the working group such as Commission locale d'information (CLI), Association nationale des Commissions locales d'information (ANCLI), ACRO (Association pour le Contrôle de la Radioactivité de l'Ouest), Versailles Academy. The concerns of all the partners were to lock the cursor for the center and not to overturn into the debate "for" or "against". The approach was guided by Dr V. Covello principes in [5] on Risk Communication recommendations: the risk Communication is an interactive process of exchanges of information and opinion among individuals, groups, and institutions. It involves multiple messages about the nature of risk, not strictly about risk, but also legal and institutional arrangements for risk management.

The past experience of IFFO RME on risk education

Since 1990, the IFFO RME association worked about education on risk. But some recent laws reinforce their actions. Some events occured in France have drastically changed the policies on natural and industrial risk management and crisis organisation, to reinforce the regulations in prevention and in preventive information of the populations. The Ministry in charge of Environment prepared the Law N 2003-699 of 30/07/03 relative to the prevention of technological and natural risks and to the repair of the damages, the Interior Ministry, the Law n°2004-811 of August 13th, 2004 of modernization of the civil protection, and the government promulgated the Law N 2006-686 of June 13th, 2006 relative to the transparency and to the safety in nuclear domain.

This information will be perceived all the better than it will take support on an early educational approach instead of waiting the treatment of event only by the media. To train the future citizen in responsible behavior, the education in the prevention of major risks joins in an education the environment, the safety and the health, in other words in the sustainable development education. In this perspective, speak about nuclear risks need to consider the overall nuclear activities from the mine extraction to the energetic and medical uses. This education calls to the notions of country planning, scales of time (management with more or less long terms for wastes or spoilt soil contamination and radioprotection measures) and spaces (solidarity / responsibility international). This education to the complexity develops the critical thinking and the education in the choice. It puts the thought of accepting a certain level of risk towards our needs.

One of the last tool developped by the IFFO RME association was "the major risk and me", an exercise book. Its various exercises are proposed based on, first, the observation of local risks (it can be a nuclear plant or installation) in its immediate environment and, second, on personal research. The booklet also identifies the major risk players and offers thinking on collectively and individually safety. In this progressive educational approach, the knowledge is build step by step. Pupils can work on this book in class or independently. In this case, the tool takes part of education.

The experience shows that it is very important that teachers participate to the design of the tool and to give their feeling about the pedagogical approach. They have to be stakeholders in the educational development. They must find anchors in the syllabus of geography, of life sciences and earth, of history, of languages, of sports in link with civics education. The major risk education is fostered by interdisciplinary work In parallel to the work done to create the manual, web pages have been designed to support teacher. They specify the general approach of the support, the level of targeted instruction, the anchoring discipline, the knowledge and skills which have to be developed from the teaching notes and always the basic knowledge for the own use of teachers.

More, schools have to organize the pupil's safety inside their buildings, in case of major risk crisis, waiting for emergency assistance teams. In France the specific plan for safety, named PPMS, is recommended by the Education Ministry. To prepare such a plan, the entire school community (adults, students, parents) need to be mobilized and to trigger appropriate behaviour and solidarity. For being well understood and applied, this organisation have to be based on pedagogical approach: in order to know why this plan, what for, what to do, in which case, against which danger.

To conclude, two essential objectives are pursued by IFFO-RME. First, the cultural one: the construction of a risk culture which enlighten citizen and make him the actor of the public debate. Second, the operational aspects, which permit the comprehension of the procedures of crisis and its management. The stake is double and of equal importance because, if somebody does not understand source data, he cannot subscribe to the message of the authorities in a crisis.

An overall strategy

From this 15 years experience of IFFO-RME, 3 complementary levers can be activating to build a nuclear risk education or modestly hope to build a conscience of nuclear risk for young people (and indirectly for parents):

1. Partnership between scientists, teachers and administrations: The exercise was difficult, it is necessary to find the consensus between the rigor, the detail of the scientist and the popularization which does not infer false representations (ex: the cooling tower never produces radioactive fallout). A common understanding of partners and the conception of a common speech takes time. The partnership between IFFO-RME and IRSN permits to share knowledge without which the population could not understand and adopt the adequate behaviour in a nuclear crisis based on the knowledge of a nuclear institution but with the wording of the users.
2. Tools: In a society which is ceaselessly called, sought by the visual communication, the visual appearance's tools "funny", "attractive", "coloured" is a key to catch the attention and interest of young people on a difficult subject. It is not a question of using the competences of the communication "to sell yoghurts", but to put these in the service of the education. The tools are always developed with an additional document for the teachers use where more detailed information is given.
3. Adults trainings: special training are organized for teachers and trainers (with various background) to let them able to ask questions and to have a deeper knowledge.

The final decision of partners was to center the purpose of tools on the nuclear risk (and not nuclear energy), in a perspective of radio-protection aspects of accident, crisis and post-accident issues.

This overall strategy includes a diffusion plan for schools (duplication of tools, the manpower-ressources, downloading on internet) where key people coming from National Education are involved. Without them the project can be unsuccessful…
However, education is not the exclusivity of schools but it is built all the life. So, the planning of itinerant exhibition includes cities and public services.

The development of tools and events

The first tool was the fan "Gafforisk", developed for fourteen year old children even if nothing in their school programme is in link with the information. Ten pages (recto verso) the first ones gives information about major risks, then one page for general knowledge about atoms and material, radioactivity and radiations. An other one for the origins of the radiations received by the population in France and their percentage in the exposure doses.

One page about the use of fission reaction giving energy in nuclear plant, and the fission product problem in case of accident, another page summarized the Chernobyl catastrophe. The various radiation uses and the different layers for protection are illustrated in other pages,. The units to measure radioactivity and its impact on the body are explained. Differences between Irradiation and contamination are illustrated. Four pages are used to explain the crisis management: alert signal, seclusion and stake under cover. Some exercises are proposed to make active children.

To build this tool, partners discussed a lot to find the last word. The selected words were not the most scientific ones but the most understandable ones for a 14 years old children.

IRSN and the French Ministry in charge of environment and Sustainable Development bought more than 35 000 copies of this fan to be broadcasted in schools with the help of the IFFO RME net.

One itinerant exhibition was created : 14 panels pulled by the fan Gafforisk were published to be posted in the schools and places where an in deep discussion was wanted. A trainer or an IRSN specialist accompanied the exhibition. Taking care of the remarks of trainers, education experts and users, 2 panels were added on the specific topic of the Chernobyl consequences. Due to the French context, these panels are very important for allowing each one to get it of his chest on the management of this accident in France and to hear the information coming from the posterior studies on this subject At the national level, three copies of the exhibition can be borrowed for various events..

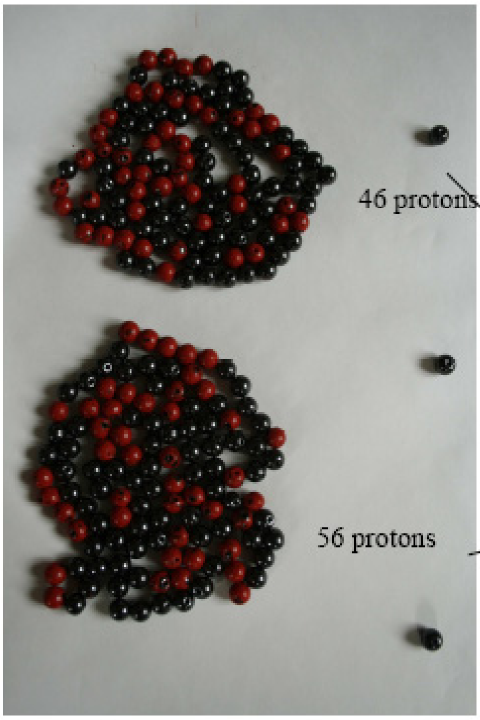
For example, it can be used when a crisis exercice is organised near a nuclear plant and when school participate to it, using the PPMS (the pupils confine themselves to the school), or when the mayor wishes to give an information to the city hall during the distribution of pastille of iodine to the inhabitants. Because in case of crisis, the key to risk management success is anticipation, preparation, and practice.

An other way to use it is during French education events such as the Week of sustainable development, or the days of Science Festival.

The lessons learnt is that that is exhibition is easy to used as well with the adults without any knowledge on radioactivity and nuclear that with the children (the youngest were 10 years old). The speaker has to adapt his language to the public.

Additional kits were developed to be used in front people to explain fission and fission products (235 +1 magnetic hematite pearls, 92 painted in red) in order to produce the feeling to be able to understand this elitist topic and to understand what is the iodine problem…They can feel what is a scientific approach and go very far in their understanding.

Fig 3 What are the fission products produced in this fission?



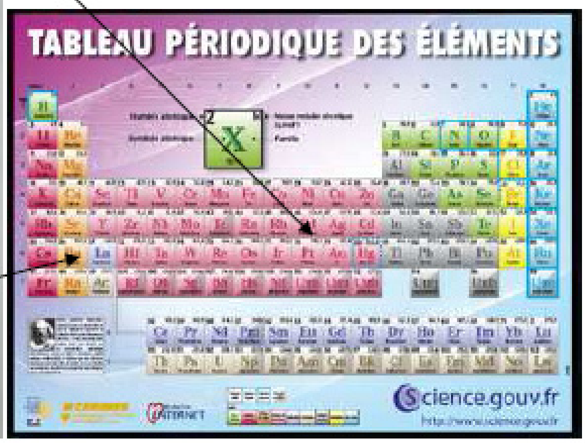


Fig 4 The answers found by people themselves by counting red pearls spread over the table and using Mendeleev table.

A local guide of exhibition

This document, composed with the partner "Local commission of information of the nuclear research center of Saclay" (CLI de Saclay) based on the fan Gafforisk. It was developed to be given to the public, alone in the exhibition when no speaker is present.

The booklet presents the exhibition in 5 parts and gives local informations. The five parts :

1. What's the radioactivity?
2. For what to do?
3. Minor incident or major accident?
4. What to do in case of crisis?
5. How to inform myself?

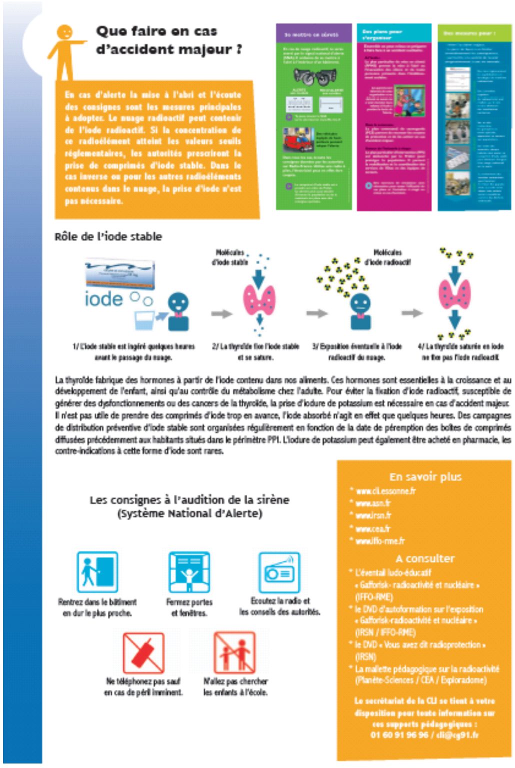


Fig 5 In this last part of the booklet, the behaviours recommended during crisis situation and protection against iodine radioactive contamination by distribution of iode tablets are explained.

Proposal of teachers training and trainer's tools

Because it is not easy to trust in ourself when we are not a specialist of the domain, trainings or tools for self-training must be developed for the attention of the voluntary persons. The academy of Versailles had organized in partnership with IRSN and IFFO-RME a session of 3 days about "radio-activity and nuclear". The training was opened to every voluntary professor to form on this subject. Professors of physics and natural sciences were present but also professors in industrial techniques as well as professors of geography, history, civic instruction, French. This unit of training will be renewed next year and suggested to others French academy.

DVD for trainers

Otherwise, IRSN and IFFO RME are preparing a DVD for auto-training based mostly on videos, free on internet, because new tools and media can be efficient vehicles for our purpose. Because, most of the concerns and questions of upset or concerned people can be predicted and prepared for in advance. It was considered as possible to anticipate them and to develop answers which will remind the aims developed in the panels of exhibition. The commentaries will be given by teachers and specialist of crisis management. There will be additional topics to about diseases, international responsibilities.

It will contain some practical experimentation about the atoms, the fission, the produce of fission, the dispersal of radioactive cloud…

Conclusion

In the context of sustainable development, the nuclear issues arise in terms of society choice: what are the risks/benefits of this choice.

To build an education on nuclear energy and especially on nuclear risk, we have to imagine various systems (paper, video, kits) with multiple sources of education messages. The messengers of education have to be diverse to multiply the meets between the citizen and rational representations of risks. So the citizen could access to information, understand it and take a better part to his security.

In the framework of the nuclear renewal, several questions arise.

– How different States can combine their efforts in the field of education in nuclear risk and radioprotection ?
– What international working group on teaching practices would allow the construction of white book of best practices and adapting existing tools in the countries concerned
– What evaluations of educational strategies in nuclear risk would measure its effectiveness in the long term.

These actions contribute to a better society resilience.

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European ALARA Network:
– Evolution, operation and key activities

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Abstract
The new ICRP recommendations (ICRP 103), and in particular the detailed treatment of optimisation in the ICRP Publication 101, define optimisation of protection as a source-related process aimed at keeping the likelihood of incurred exposures, the number of people exposed and the magnitude of their individual doses as low as reasonably achievable, also below constraints, taking into account economic and societal factors. The implementation of the optimisation of protection into practice is supported by the ALARA principle. The term ALARA is an acronym for "As Low As Reasonably Achievable". In essence, the ALARA principle requires that radiation exposure of man and environment be kept as low as reasonably achievable (also below constraints) when using ionising radiation. Practical implementation and further development of the ALARA principle has been achieved for many years now by the successful cooperation of experts from different European organisations; first as pioneers by establishing the European ALARA Network and then by enthusiastically supporting the activities of the network itself. This contribution presents the evolution, operation and key activities of the European ALARA Network (EAN) in the last years; the successful cooperation of experts from different professional backgrounds, advocating the ALARA principle in a range of radiation protection areas, and contributing to its further development by trading experience and networking. The interaction between the EAN and international organisations, which support the ALARA principle by including relevant activities in their work programmes, is described, as well as the cooperation between EAN and other networks to identify the role of ALARA in the process of improving radiation protection culture.

ENETRAP-II: development of European training schemes for RPE's and RPO's

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Abstract

ENETRAP II, aims at developing reference standards and good practices for education and training programmes for radiation protection experts and officers, reflecting the needs of these professionals in all sectors where ionising radiation is applied. The introduction of a radiation protection training passport as a mean to facilitate efficient and transparent European mutual recognition of these professionals is another ultimate deliverable of this project. It is envisaged that the outcome of ENETRAP II will be instrumental for the cooperation between regulators, training providers and customers (nuclear industry, research, non-nuclear industry, etc.) in reaching harmonisation of the requirements for, and the education and training of, radiation protection experts and officers within Europe, and will stimulate building competence and career development in radiation protection to meet the demands of the future.

Introduction

Radiation protection (RP) is a major challenge in the industrial applications of ionising radiation, both nuclear and non-nuclear, as well as in other areas such as the medical and research area. As is the case with all nuclear expertise, there is a trend of a decreasing number of experts in radiation protection due to various reasons. On the other hand, current activities in the nuclear domain are expanding: the nuclear industry faces a so-called "renaissance", high-tech medical examinations based on ionising radiation are increasingly used, and research and non-nuclear industry also make use of a vast number of applications of radioactivity.

Within this perspective, maintaining a high level of competency in RP is crucial to ensure future safe use of ionising radiation and the development of new technologies in a safe way. Moreover, the perceived growth in the different application fields requires a high-level of understanding of radiation protection in order to protect workers, the public and the environment of the potential risks. A sustainable education and training (E&T) infrastructure for RP is an essential component to combat the decline in expertise and to ensure the availability of a high level of radiation protection knowledge which can meet the future demands.

Today's challenge involves measures to make the work in radiation protection more attractive for young people and to provide attractive career opportunities, and the support of young students and professionals in their need to gain and maintain high level RP knowledge. This can be reached by the development and implementation of a high-quality European standard for initial education and continuous professional development for radiation protection experts (RPEs) and radiation protection officers (RPOs), and a methodology for mutual recognition of these professionals on the basis of available EU instruments, such as the European qualification framework (EQF) and/or the directive 2005/36/EC.

Within the framework of this project, the RPE and RPO should be interpreted as:
Radiation protection expert (RPE): *an individual having the knowledge, training and experience needed to give radiation protection advice in order to ensure effective protection of individuals, whose capacity to act is recognized by the competent authorities.*
Radiation protection officer (RPO): *an individual technically competent in radiation protection matters relevant for a given type of practice who is designated by the undertaking to oversee the implementation of the radiation protection arrangements of the undertaking*.

These are the definitions proposed by ENETRAP 6FP (www.sckcen.be/enetrap) and EUTERP (www.euterp.eu) to the EC DG TREN and Article 31 group, who is working on the revision of the Basic Safety Standards.

Project details

Determined to build further on the achievements of 6 FP ENETRAP, most ENETRAP partners participate in 7FP ENETRAP II (www.sckcen.be/enetrap2). The overall objective of this project is to develop and implement European high-quality "reference standards" and good practices for E&T in RP, specifically with respect to the RPE and the RPO. These "standards" will reflect the needs of the RPE and the RPO in all sectors where ionising radiation is applied (nuclear industry, medical sector, research, non-nuclear industry). The introduction of a radiation protection training passport as a mean to facilitate efficient and transparent European mutual recognition is another ultimate deliverable of this project.

With respect to the RPE the overall objective is to be achieved by addressing both education and training requirements.

In the field of education this project deals with high-level initial programmes, mainly followed by students and/or young professionals. It is foreseen to analyse the European Master in Radiation Protection course, which started in September 2008. Broadening of the consortium and quality analysis of the providers and the content of the modules can be performed according to, primarily, the standards and guidelines for quality assurance in the European higher education area (ENQA) and, secondly, to the ENEN standards.

In the field of RPE training the ultimate goal is the development of a European mutual recognition system for RPEs. Hereto, the ENETRAP training scheme initiated as part of the ENETRAP 6FP will be used as a basis for the development of a European radiation protection training scheme (ERPTS), which includes all the necessary requirements for a competent RPE. In addition, mechanisms will be established for the evaluation of training courses and training providers.

With respect to the RPO role the desired end-point is an agreed standard for radiation protection training that is recognised across Europe. Data and information obtained from the ENETRAP 6FP will be used to develop the reference standard for radiation protection training necessary to support the effective and competent undertaking of the role.

Furthermore, attention is given to encouragement of young, early-stage researchers. In order to meet future needs, it is necessary to attract more young people by awaking their interest in radiation applications and radiation protection already during their schooldays and later on during their out-of-school education (university or vocational education and training). Radiation protection experts and officers work more and more on a European level. It is therefore important bringing together all the national initiatives at a European level: tomorrow's leaders must have an international perspective and must know their colleagues in other countries.

It is envisaged that the outcome of ENETRAP II will be instrumental for the cooperation between regulators, training providers and customers (nuclear industry, medical sector, research and non-nuclear industry) in reaching harmonisation of the requirements for, and the education and training of RPEs and RPOs within Europe, and will stimulate building competence and career development in radiation protection to meet the demands of the future.

Specific objectives of the ENETRAP II project are to:

- develop the European radiation protection training scheme (ERPTS) for RPE training;
- develop a European reference standard for RPO training;
- develop and apply a mechanism for the evaluation of training material, courses and providers;
- establish a recognised and sustainable ERPTS "quality label" for training events;
- create a database of training events and training providers (including OJT) conforming to the agreed ERPTS;
- bring together national initiatives to attract early-stage radiation protection researchers on a European level;

- develop some course material examples, including modern tools such as e-learning;
- develop a system for monitoring the effectiveness of the ERPTS;
- organise pilot sessions of specific modules of the ERPTS and monitor the effectiveness according to the developed system;
- development of a European passport for CPD in RP.

The objectives of ENETRAP II 7FP will be reached by several activities dealing with
- the analysis of job requirements (RPE and RPO);
- the design and implementation of appropriate training standards and schemes to support these requirements;
- development and application of a quality assurance mechanism for the evaluation of the training events, used material and training providers;
- setting up a database of training events and providers conforming to the agreed standards;
- the development of training material (traditional texts, as well as the introduction of more modern tools such as e-learning modules) that can be used as example training material;
- monitoring the effectiveness of the proposed training schemes.

The final goal is the development of a European mutual recognition system for RPEs and the introduction of a training passport.

The different work packages (WP) defined in this project are:

WP1	Co-ordination of the project
WP2	Define requirements and methodology for recognition of RPEs
WP3	Define requirements for RPO competencies and establish guidance for appropriate RPO training
WP4	Establish the reference standard for RPE training
WP5	Development and apply mechanisms for the evaluation of training material, events and providers
WP6	Create a database of training events and training providers (including OJT) conforming to the agreed standard
WP7	Develop of some course material examples (text book, e-learning modules, …)
WP8	Organise pilot sessions, test proposed methodologies and monitor the training scheme effectiveness
WP9	Introduction of the training passport and mutual recognition system of RPEs
WP10	Collaboration for building new innovative generations of specialists in radiation protection

ENETRAP II 7FP is realised by 12 partners, each having relevant experience in policy support regarding E&T projects on radiation protection. It concerns SCK•CEN (Belgium), CEA-INSTN (France), Karlsruhe Institute of Technology, Centre for

Advanced Technological and Environmental Training KIT-FTU (Germany), Federal Office for Radiation Protection BfS (Germany), the Italian National Agency for New Technology, Energy and Environment ENEA (Italy), NRG (The Netherlands), CIEMAT (Spain), Health Protection Agency HPA (UK), the ENEN Association (France), the Nuclear and Technological Institute ITN (Portugal), the Budapest University of Technology and Economics BME (Hungary), and University Politehnica of Bucharest (Romania). Staff members of the different partners who play a key role in this project, have also proven to be highly involved with E&T matters, on national and international levels, and are member of several E&T networks. Most of them also have an advisory role towards the national regulatory authority.

ENETRAP II (grant agreement number 232620) is a coordination action that runs under the theme "Euratom Fission Training Schemes (EFTS) in all areas of Nuclear Fission and Radiation Protection" (Fission-2008-5.1.1). The project will run over 36 months.

Results

Although this project is still in its first phase, some work packages have already reached intermediate results. In the following paragraphs a summary is given of the work carried out in the different WPs, and the first achievements are highlighted.

Requirements and methodology for recognition of RPEs

WP2 deals with the requirements for recognition of RPEs and the development of a methodology for the recognition of RPEs. Although the execution of any recognition process is the responsibility of the national regulatory authority, ENETRAP II will put forward a harmonised methodology, in line with the national approaches. The existence of this European methodology will facilitate the ultimate goal: a European mutual recognition process for RPEs. Qualification, competence, and continuous professional development will be discussed and elements for these three requirements will be defined. An outline of the proposals for key elements of a national scheme for RPE recognition was put forward at the most recent ETRAP conference (Lisbon, November 2009 (http://www.euronuclear.org/events/etrap/index.htm). Here, also a questionnaire was presented which was sent out to all participants, EUTERP contact points and other relevant stakeholders. The results of the questionnaire are currently being analysed and will be used to provide guidance with respect to national schemes for recognition of RPEs. From there, a mechanism will be developed for the mutual recognition of RPEs between Member States.

Requirements for RPO competencies and establishment of guidance for appropriate RPO training

WP3 deals with requirements for RPO competencies and the establishment of guidance for appropriate RPO training. Employees, appointed to act as RPOs in hospitals, industrial companies or teaching and research institutions should have an adequate level of understanding of concepts related to radiation protection and understand the radiation protection issues pertinent to their radiation application. Therefore the level and format of training required by an RPO is dependant on the complexity of that application. It is therefore essential, on the EU level, (i) to define requirements for the competencies of

RPOs according to their area of work and specific radiation protection tasks, and (ii) to establish European reference standards for RPO training. The first intermediate report on this topic is submitted.

Establishment of the reference standard for RPE training

WP4 continues on the achievements of WP2. Here, it is the aim to develop appropriate European radiation protection training schemes (ERPTS), with objectives, target audience(s), audience prerequisites, required topics, suggested durations and evaluation methods for both initial and refresher training of RPEs, taking into account the nature and requirements of the RPE role. The starting point is the ENETRAP 6FP training scheme. Furthermore internationally recognised training material such as the material developed by the IAEA will be incorporated. The ERPTS should meet the requirements of the revised definitions of the RPE and should eventually replace Communication 98/C133/03, as a guide for the Member States to develop, or evaluate, their national strategies for RPE qualification and recognition.

Development and application of mechanisms for the evaluation of training material, events and providers

In the EU, a vast number of training events, material and providers exist. Given that formal recognition is required for RPEs, it would be prudent for training providers involved in the RPE training process to also be formally recognised. The aim of this WP5 is to develop a mechanism for the comparison, through a transparent and objective methodology, of training materials, courses and training providers, which can be used by regulatory authorities to evaluate their national radiation protection training programme for compliance with the ERPTS. First results are presented at this conference.

Database of training events and training providers

It is foreseen in WP6 to create a database of training event and providers conform to the agreed standards. The database will be made public through the ENETRAP II website and will thus be available for all interested parties. It is the aim that such a move would add credibility to the recognition process and would help to provide reassurance to RPE candidates and to employers that the training obtained satisfies an agreed European standard. It is foreseen that this database will also incorporate an overview of institutes hosting on-the-job-training possibilities. First announcements are foreseen by the end of 2010.

Development of course material

In order to provide examples of standardised training material, meeting the requirements of the ERPTS, WP7 foresees a European textbook for several modules of the ERPTS. It is suggested to launch a "cyber book", using the MOODLE platform that was introduced in ENETRAP 6FP. The modules that will be treated in this book are to be decided from the 66 entries WP7 received to their survey amongst the partners regarding the available course material. An additional suggestion is to create a sonorized (video + audio) PowerPoint presentation.

Pilot sessions

In the framework of ENETRAP II, pilot sessions of the European reference training scheme will be organised. The date put forward for the first pilot session, containing the basic RP modules and a specialized module on radiation protection issues in nuclear power plants, is spring 2011. This will be organised at KIT-FTU, Germany. Another specialized module on NORM issues will be organised at HPA, UK. At this conference the pilot sessions are presented in a poster presentation.

Introduction of a training passport and mutual recognition system

The ultimate goal of the ENETRAP II project is the introduction of an EU mutual recognition system for RPEs. In WP9 coordinating actions will be undertaken to establish such a system. Furthermore, the European training passport will be introduced as a tool for facilitating an efficient and transparent mutual recognition system.

Whenever possible, a collaboration will be established with the "training" working groups of the three EU "platforms" that were launched in 2007 (in particular, to discuss the added value of a "European training / skills passport" and the balance between theoretical and practical training that is desired to improve both the quality and the mobility of nuclear experts in public as well as private sector). The results of this WP are expected in the last phase of the project.

Building new innovative generations of specialists in radiation protection

Those people who developed concepts in radiation protection and held leadership positions at universities and research institutions to further develop radiation research and educate and train the next generation in the EU are retired or starting to retire. We are facing the same situation for numerous radiation protection experts and officers who devoted their knowledge and experience to build up a high level of radiation safety in all radiation applications in industry, medicine and research in the EU. In order to maintain this high level and to further develop a European safety culture, it is necessary to attract more young people by awaking their interest in radiation applications and radiation protection. More young people must be inspired to take an interest in radiation research and prepared to take leadership positions at universities and radiation applications in industry, medicine and research in the EU. Because high-level RP professionals often work in a European context, tomorrow's leaders will benefit from having an international perspective and knowing their colleagues in other countries.

Summary and conclusions

Based on the outcome of the ENETRAP 6FP, ENETRAP II 7 FP aims at contributing further to the EU harmonisation of E&T of radiation protection professionals. With the introduction of a modular European reference training scheme and European recognition methodologies, key issues will be delivered for the development and implementation of mutual recognition system of RPEs. In this way ENETRAP II meets the EC requirements to rely on the principles of modularity of courses and common qualification criteria, a common mutual recognition system, and the facilitation of teacher, student and worker mobility across the EU. ENETRAP II will structure research on radiation protection training capacity in all sectors where ionising radiation is applied. End users and specifically regulatory authorities are represented through foreseen participation in the advisory board which will advise about the best balance between supply and needs, thereby ensuring stable feed back mechanisms. The tasks defined in this project maximise the transfer of high-level radiation protection knowledge and technology, addressing young as well as experienced radiation protection workers. In this context, the proposed project will thus contribute to meeting the objectives of the EURATOM research and training work programme.

A Training Programme for Regulatory Inspectors under ISO 17020

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Abstract

The Radiological Protection Institute of Ireland (RPII) is the competent authority for all matters pertaining to ionising radiation in Ireland. In fulfilment of its statutory responsibilities it oversees a regulatory programme (licensing, inspection, enforcement and the provision of technical guidance and advice) for all users of sources of ionising radiation.

In December 2008 the RPII was awarded accreditation to ISO 17020: General Criteria for the Operation of Various Types of Bodies Performing Inspection by the Irish National Accreditation Board (INAB). In achieving this award the RPII became the first regulatory authority in Europe engaged in radiation protection inspections to be accredited for its inspection services under this standard. The RPII's quality manual and quality procedures describe in detail how its inspection programmes are planned, carried out and reviewed as well as setting out training requirements for inspectors.

This presentation will describe the formal training programme that was developed for a new inspector who joined the RPII's Regulatory Services Division in October 2007. The programme was developed in accordance with the requirements of the ISO standard and audited by technical assessors from INAB. The programme clearly identifies all aspects of training, both technical and non-technical, that a new member of staff has to undertake in order to be warranted as an inspector/authorised officer.

In addition to the development of a formal training programme for new inspectors the ISO standard also required the RPII to develop a formal programme for assessing the on-going competency of existing inspectors. This is achieved through a programme of inspection witnessing of individual inspectors by their technical managers. Inspectors are deemed competent provided their technical manager is satisfied with their knowledge and performance as witnessed during inspections.

Introduction

In 2005 the RPII commenced work on developing a quality management system for all of its work activities associated with the inspection of licensees holding sources of ionising radiation. The primary motivation for this work was the desire to ensure that the RPII performed all activities associated with inspections, such as planning and carrying out inspections, issuing of inspection reports, follow-up actions and inspector training, to the highest international standards. Through the development of an externally audited quality management system, containing elements such as documented work procedures, programmes of continual review and formalised inspector training programmes the RPII could ensure that all inspections are carried out in a professional and competent manner ensuring consistency of inspections and the approach taken by inspectors.

The RPII originally commenced work on developing a quality management system under ISO 9000 for all of its licensing and inspection activities. However, following meetings with the Irish National Accreditation Board (INAB), the national body with responsibility for accreditation in Ireland, a more appropriate standard for the RPII was identified, namely *ISO 17020: General Criteria for the Operation of Various Types of Bodies Performing Inspection* (ISO, 1998). This standard is typically sought by inspection bodies whose functions include the examination of materials, products, installations etc. and the determination of their conformity with requirements, including the subsequent reporting of the results of these activities. As the RPII has a statutory responsibility to license all holders/users of sources of ionising radiation, and to carry out inspections, it determined that to seek accreditation to the ISO 17020 standard would be the most appropriate standard for it.

Over the course of two years the RPII developed a quality manual which described in detail the systems and processes it had in place to address the requirements of the ISO 17020 standard. The quality manual was further supported by a comprehensive set of quality procedures that documented all the work activities that support its inspection programmes. A list of the quality procedures is given in Table 1.

Table 1. RPII Inspection Quality Procedures.

Procedure No	Title
QP01	Corrective/Preventative Action Procedure
QP02	Document Control Procedure
QP03	Management Review Procedure
QP04	Audit Procedure
QP05	Annual Inspection Programme
QP06	Pre-Inspection Procedure
QP07	Inspection Procedure
QP08	Post-Inspection Procedure
QP09	Inspectors Meetings
QP10	Training Programme Procedure
QP11	Equipment Procedure

In relation to the training of new inspectors the RPII's quality manual identifies a number of criteria which must be met:

- Staff responsible for inspections shall have appropriate qualifications, training, experience and a satisfactory knowledge of the requirements of the inspections to be carried out;
- The RPII shall establish a documented training system to ensure that the training of inspectors is kept up-to-date;

- The training shall depend upon the qualifications and experience of persons involved, and shall include an induction period, a supervised working period with experienced inspectors and continuation training to keep pace with developing technology;
- The RPII shall provide guidance on the expected conduct of inspectors.

The quality system went "live" in September 2007 and was operated for a full year before INAB carried out a pre-registration assessment visit over two days in September 2008. As well as reviewing the quality manual and all the documented procedures the INAB auditors witnessed RPII inspectors carrying out an inspection of a nuclear medicine department in a major teaching hospital as well as an inspection of a licensee carrying out non-destructive testing work. The INAB auditors were satisfied that the RPII had met the requirements of both the ISO standard and the IAF/ILAC guidance document (IAF/ILAC, 2004) and subsequently awarded it accreditation to ISO 17020 in December 2008.

Inspector Training pre accreditation

Prior to the introduction of a formal training programme under ISO 17020, the RPII had informal procedures for training new inspectors. Due to the limited usage of ionising radiation in Ireland, it is difficult to recruit inspectors with several years experience of working with sources of ionising radiation and, in practice, most new inspectors are employed straight out of university, the majority of them having studied physics.

Newly recruited inspectors would undergo an induction period during which they would be provided with on-the-job training in relation to the licensing and inspection processes and are provided with a full set of relevant legislation and guidance documents. The emphasis during this training focused on the hard skills such as knowledge of the applications of ionising radiation, inspection procedures, use of radiation monitors, report writing etc. rather than on the softer skills such as interviewing techniques and communication skills.

Over the course of their first couple of months a newly recruited inspector would accompany experienced inspectors to observe how inspections were conducted across all sectors. Prior to 2005, inspectors did not specialise in particular areas and were trained to be able to undertake inspections across all sectors such as medical, veterinary, industrial, education research etc. In 2005 it was decided that two distinct categories of inspectors would be created: those who would specialise in carrying out inspections in the medical, dental and veterinary sectors and those would conduct inspections in the industrial, educational and research sectors. This specialisation meant that inspectors could focus on relevant sectors and attendance at appropriate conferences, meetings and training courses could be arranged.

New inspectors would continue to observe inspections across a wide range of activities until such time as the experienced inspectors felt that the new inspector could lead an inspection with their assistance, and under their supervision. Once the inspector was deemed competent by their technical manager, the Board of the RPII would be asked to issue a warrant to the new inspector.

While this approach had worked very well over the years it was acknowledged within the RPII that its inspector training programme should be formalised to ensure consistency in how its inspectors were trained. The programme would include milestones indicating when new inspectors could be deemed competent to undertake inspections on their own and accordingly be issued a warrant. It was also acknowledged that the focus to date had been on teaching a new inspector how to plan and undertake inspections and that softer skills, such as interviewing techniques, assertiveness training and chairing meetings, all of which are necessary when carrying out inspections, were not given as high a priority.

In practice, during the period 2004 – 2007 there were only two new inspectors appointed. One inspector returned from working as a medical physicist in a large hospital to the RPII, having previously worked as an inspector in the 1990s, and required relatively little training to bring her up to date. When a member of staff transferred from a different department within the RPII into the Regulatory Services Division (RSD) it provided an opportunity to trial the new inspector training programme that was in the process of being developed.

Inspector Training under ISO 17020

Shortly after switching on its new quality management system in September 2007, the RPII recruited a new inspector to the Medical, Dental & Veterinary Section within the RSD. The new inspector joined the team responsible for licensing, inspection and enforcement activities in these sectors and was the first inspector to go through the formal inspector training programme developed under the quality management system.

The specification drawn up for the post required candidates to hold an honours degree in physics or in electrical, electronic, mechanical or clinical engineering. Additionally candidates were advised that experience working as a medical physicist was desirable.

The new inspector commenced work in October 2007 as a Scientific Officer. Holding a diploma in applied physics and a BSc in physics and physics technology she had previously worked as a research assistant in a third level college and as a field service engineer for a major semiconductor company. In the year prior to joining the RPII she completed an MSc in medical physics.

Training Programme

The RPII's quality management system allows for some flexibility when drawing up a training programme for a new inspector, as he or she will have different skills and experience. For the newly recruited inspector a formal training programme was drawn up and mutually agreed though discussions during her first week. As well as focusing on the skills required to carry out inspections the training also looked at wider training requirements, particular as this person was coming from the private sector into the public sector. The training programme was broken into 13 modules, which are summarised below:

- Introduction to the RPII: internal structures; relationship with/to government departments;
- Introduction to regulatory responsibilities: licensing; inspection; enforcement; provision of guidance and advice;
- Inspector behaviour: RPII staff handbook; code of business conduct; inspector's code of conduct;
- Health and Safety: RPII health & safety management system; manual handling; construction site safety (Safe Pass); fire safety;

- ISO 17020 Quality Management System: quality manual; quality procedures;
- Legislation: primary & secondary legislation; legal workshop; courtroom skills;
- Information technology: licensing & inspection database; IT usage policy;
- Equipment: personal protective equipment; types and use of radiation monitors; personal dosimetry;
- Inspection procedures: planning; reporting; follow-up;
- Inspections: observation phase; supervisory phase; competency assessment;
- Incident investigation: accidents; personal dosimetry investigations;
- RPII business planning: strategic planning; annual business planning, Performance Management Development System (PMDS);
- Training: on-the-job training; formal courses; technical meetings; conferences.

Based upon discussions between the new inspector and her technical manager it was agreed that she would accompany experienced inspectors on a total of 20 inspections as part of the observation phase of her training programme. These 20 inspections would cover diagnostic and therapeutic applications across the medical, dental and veterinary sectors. Over the course of these inspections the new inspector would gain experience of how inspections are planned and arranged, witness firsthand how an inspector interacts with the licensee during all stages of the inspection and how the findings of inspections are reported and followed up.

This would then be followed by a supervisory phase where the new inspector would lead five inspections supported by one of the experienced inspectors. She would be responsible for arranging and conducting the inspection as well as issuing the final report and dealing with follow-up actions. Finally, her technical manager would assesses her competence by witnessing her carrying out an inspection in each of the modalities of diagnostic radiology, nuclear medicine and radiotherapy in the medical, dental and veterinary sectors as appropriate. This graded approach to the competency assessment means that the inspector can be signed off for simple inspection types e.g. dental radiology, DXA etc. working up to more complex inspection types such as nuclear medicine and radiotherapy as she gains more experience.

As each phase of the training was completed both the trainee (new inspector) and trainer would formally sign off her training record.

On-the-job training

The RPII has developed inspection audit forms, which are part of the quality system, for all sectors which inspectors complete during inspections. Specific audit forms have been developed for modalities within individual sectors, such as diagnostic radiology, nuclear medicine and radiotherapy within the medical sector. As well as being an *aide-memoir* for inspectors the forms also ensure that different inspectors cover the same areas during inspections i.e., there is consistency among different inspectors on the issues examined during inspections.

At the outset of the training programme for the new inspector it was agreed that she would witness a total of 20 inspections by accompanying experienced inspectors. By the end of this phase of her training she had in fact witnessed 27 inspections. Table 2 summaries the inspection types observed during the period October 2007 – July 2008.

Table 2. Inspections observed during observation phase of training programme.

Inspection Type	Number of inspections observed
Diagnostic Radiology (Medical)	14
Nuclear Medicine (Medical)	1
Radiotherapy (Medical)	4
Diagnostic Radiology (Dental)	4
Diagnostic Radiology (Veterinary)	2
Equine Nuclear Medicine (Veterinary)	1
Distributor of Radioactive sources	1
Total	27

Following this period of observation the experienced inspectors deemed her competent to lead an inspection under their supervision. In practice this meant that she was the lead inspector but was assisted by one of the experienced inspectors. However, in order for her to be able to lead an inspection she had to be formally appointed an inspector by the Board of the RPII, in accordance with the provisions of the Radiological Protection Act 1991 (RPA, 1991). In July 2008, the Board of the RPII was advised that the inspector had reached a stage in her training where she had been deemed competent to lead an inspection, under the supervision and with the support of an experienced inspector. In order to complete her training she would be required to be appointed as a warranted inspector so that she could lead inspections. Accordingly, nine months after joining the RPII she was appointed as an inspector and issued with a warrant.

As part of the supervisory phase of her training programme she ended up leading six inspections, four in diagnostic radiology facilities within hospitals and two dental practices, assisted by an experienced inspector. She also participated on the inspection team for two radiotherapy and one diagnostic radiology inspections during this phase of her training.

In January 2009, for the final phase of her inspector training her technical manager assessed her competency as an inspector. It is important to note that as her technical manager had to assess her and sign off on her training he could not be involved directly in her training as this would present a conflict of interest i.e., he couldn’t both train her and be the person responsible for determining that she had been fully trained. Over the course of two days she was witnessed carrying out inspections of diagnostic radiology facilities in two hospitals, a dental practice and a veterinary practice. After each inspection her technical manager reviewed her performance during the inspection, posing a series of hypothetical scenarios in order to further assess her technical competence. Later that month, after all the inspection reports had been issued and follow-up issues closed out, the new inspector was formally deemed to have completed her initial inspector training and could undertake diagnostic radiology inspection on her own. To lead radiotherapy or nuclear medicine inspection teams she would have to be further assessed by her technical manager.

Formal Courses

While on-the-job training is very effective for developing hard skills, such as knowledge of legislation, using radiation monitors there are many soft skills that a new inspector needs to acquire. These soft skills include interviewing techniques, conflict management, negotiating skills, communication and presentation techniques, chairing meetings etc., all of which are put into practice each time an inspector carries out an inspection. As part of her training programme suitable training courses and opportunities were identified which cover both hard and soft skills. While it was possible to identify some of these at the outset of the programme, others only came to light as the training programme progressed. In addition to her initial training programme the new inspector continues to attend relevant training courses as part of her personal development. The courses attended by the new inspector since joining the RPII in October 2007 are listed below:

- "How to inspect" – a foundation course in non-technical aspects of inspection, Irish Medicines Board (Dublin) – October 2007;
- Quality customer services training, RPII (Dublin) – October 2007;
- A Course in Radiotherapy Physics, Royal Marsden Hospital & The Institute of Cancer Research (London) – March 2008;
- WHO Co-ordination Meeting - Radiation Safety in Healthcare Settings, WHO (Geneva) – June 2008;
- ICRP Seminar, ICRP (Dublin Castle) – September 2008;
- Presentation skills course, Irish Times Training (Dublin) – October 2008;
- "Radiation Protection in Nuclear Medicine", IPEM meeting, IPEM (London) – November 2008;
- Radiation Physics for Nuclear Medicine - Madiera Course, EC funded (Milan) – November 2008;
- Auditor training, RPII (Dublin) – May 2009;
- ADR basic training, (Dublin) – May 2009;
- ADR Class 7 specialisation course, One Photon Consultancy (Louth) – May 2009;
- Three week technical placement in Medical Physics Department, Royal Surrey County Hospital, Guildford, UK - June 2009;
- Radiological Protection Summer School 2009, IBC Global Academy (Cambridge) – July 2009;
- Court room skills: *The Expert Witness*, La Touch Legal Training (Dublin) – August 2009;
- Workshop on Incident Management, Health Protection Agency (Dublin) – October 2009;
- Legal training workshop, Harry Mooney & Co Solicitors (Dublin) – November 2009;
- Assertiveness Skills, Irish Times Training (Dublin) – February 2010;
- Safe Pass: Health and Safety Awareness Training Programme for the construction industry (Dublin) – April 2010.

Inspection witnessing

One of the requirements of the ISO 17020 standard is that the inspection body must have procedures in place to be able to assess the on-going competence of all its inspectors. To address this requirement the RPII has developed a programme of inspection witnessing whereby each inspector is witnessed by his or her technical manager carrying out a series of inspections.

In the medical sector inspections are divided into four broad categories: diagnostic radiology (including medical and dental), nuclear medicine, radiotherapy and the activities associated with the distribution of X-ray equipment and radioactive sources. Each inspector must be witnessed carrying out an inspection at least once each year and in each of the four categories at least once every three years. The technical manager for the Medical, Dental & Veterinary Section is in turn witnessed by the technical manager for the Industrial Section, and vice versa. A set of inspection witnessing audit forms have been developed which the technical manager completes as he witnesses the inspection. The competency assessment covers the entire range of activities associated with inspections, from the planning stages, though the actual inspection itself and finally the inspection reporting. In particular, the following issues are audited:

- Did the inspector plan and document all pre-inspection arrangements?
- Did the inspector bring photographic identification, his/her inspector's warrant, relevant legislation and appropriate personal protective equipment on the inspection?
- Did the inspector follow all leads in an objective and resolute manner?
- Did the inspector use an open/probing questioning style?
- Did the inspector use test or inspection equipment correctly and was it fit for purpose?
- Did the inspector investigate all areas pertaining to the scope of the inspection?
- Could the inspector answer all questions raised by the licensee?

Following each witnessed inspection the technical manager meets the inspector to discuss how the inspection went, posing a series of challenging scenarios to the inspector. Provided that the technical manager is satisfied with the inspector's performance the inspector is deemed competent to continue to inspect within that category.

This system of formal inspection witnessing ensures that all inspectors carry out inspections in a manner that is both consistent with their fellow inspectors and with the objectives of the RPII quality management system.

Conclusions

The RPII has successfully developed and implemented a formal training programme for new inspectors under its ISO quality management system. As well as providing a structured approach to training, it also provides both new staff and their technical managers a clearly defined set of training modules which are mutually agreed at the outset of the programme. The programme is flexible and can be tailored to suit the training needs of trainee inspectors, taking account of the expertise they bring to the organisation.

In addition to the development of a training programme for new inspectors, the RPII has also introduced measures to assess the on-going competency of all inspectors through its inspection witnessing programme. This programme provides a formal mechanism where the technical competence and ability of all inspectors are assessed on an annual basis. As well as providing reassurance to the RPII it also provides reassurance to licensees that the RPII has procedures to ensure that its inspectors are performing their duties to the highest standards.

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A European survey addressing needs for safety culture training

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Abstract

Nuclear Safety Culture is a topic of paramount importance for nuclear operators as well as those working with radiology and radiotherapy. Safety culture is a combination of individual and group beliefs, values, attitudes, perceptions, competencies and patterns of behaviour that determine the commitment and the proficiency of an organization's safety management. Implementation of safety culture requires continuous and multilateral efforts involving not only technical but also human and social aspects. Many principles of safety culture are generally applicable , and could be disseminated through seminars on best practices, case studies, feedback studies, pilot sessions, etc. Such dissemination would contribute to harmonisation according to high standards, and it would promote the mutual recognition of training throughout Europe.

A consortium of 19 partners working together in a 4 year project, TRASNUSAFE, will develop and test relevant training schemes on Nuclear Safety Culture, based on an evaluation of the training needs in a European context. Special attention will be given to the links between the ALARA principle currently used in the radiation protection community and the safety culture of the nuclear industry. Two user groups, involved throughout the entire development process, will contribute to the training schedule by providing their input and feedback and by participating in the training modules.

Training needs will be assessed throughout Europe using a questionnaire to be separately developed and validated prior to the assessment. The actual assessment of training needs will make use of web based surveys, e-mails and regular mails, telephone conversations with correspondents, as well as seminars and site visits.

This paper will focus on the training needs analysis. It will explain the methodology and will give a report of the first results.

Managing medical exposure through education

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Abstract
Nuclear physics is providing medicine investigation and treatment methods of a priceless value. During over one hundred years, radiological equipment become more and more performing from the point of view of easy handling and rapid answer, quality of radiation beams and, not the last, radiation protection of operators and patients.
Physicians' enthusiasm due to these possibilities and society concern for individual health risk to transform these achievements from friends to enemies of health due to their excessive utilization.

It is important to estimate supplementary cancers expected by un-relevant computer tomography examinations but it should be more adequate to develop measures and methods for preventing transformation of these potential figures into real ones.

In this context, education could be used as a mean to present both sides of these special applications involving individuals, on one hand by public information and on the other hand by improving physicians' education.

Public education could begin even from elementary school by graduate introduction of the necessary notions to understand the specific aspects of radiation biologic effects. It could continue through mass media communication and by physicians at their surgeries.

This paper proposes to represent guidance for patients' education in order to involve them in the medical act. Being aware of radiological examinations specific aspects, people would participate more active to the implementation of individual monitoring systems.

As a result of the international actions already initiated, it is presumed that prescribes would become more restrictive when recommending radiological examinations and radiologists and non-radiologists physicians would become more aware to comply with the regulatory requirements on justification of each examination.

Introduction
Taking into account that each science progress depends of its efforts to integrate the results of other sciences, it could be noticed that medicine took an impressive advantage in the 20^{th} century, as a result of cooperation with other sciences, such as physics, chemistry, biology, materials science, information technology.

This multidisciplinary approach determined the development of new branches, such as radiology, bio-chemistry, cell biology, etc., (attached to medicine) or medical physics, radio-biology, bio-materials, drugs production (attached to other sciences).

Medical radiation exposure

For Nuclear Physics, cooperation with Medical Sciences started in 1895, after X Rays discovery. X Rays applications for diagnosis rapidly extended (in one to three years) in all Europe and out of it and since then they have been continuously developed.
Three major directions could be identified in this expansion:

1. extension of applications in medicine by
   - using of radioactive sources (sealed and unsealed) beside Roentgen generators and electrons accelerators (in the last decades)
   - adding to radiological examination for diagnosis densitometry and interventional radiology
   - extending use of ionising radiations in radiotherapy and stereo tactic surgery.
2. technical development and improvement of equipment and data systems by developing more versatile equipment with fluorescent screen or film, image amplifier or image digitalization (with different handling and storage possibilities) and from 2D imagines to 3D (CT) or 4D. It is estimated that it was obtained a reduction of dose per examination by a factor of a few tens between the first and the last generations of radiological equipment.
3. development of regulations in compliance with the development of the extension of applications in medicine and technological development to reduce the biological risk for medical personnel and patients. Looking behind, it could be noticed that regulations implemented after the development of applications. Over 30 years of X Rays utilization in diagnosis were necessary to establish the International Commission on Radiological Protection (1928) – nongovernmental organization which suggests rules for ionising radiations utilization for protecting users against their adverse effects. Radiation protection concept appears as a definition of assembly of measures/rules for health protection and limitation of risks in radiation sources utilization after 1928. Radiation protection requirements addressed exclusively to operators of radiological facilities because they were the first persons who developed symptoms of over exposure and could be monitored during their activity.

Increase of radiological examinations procedures and extension of applications in other medical fields determined an important increase of population exposure. Medical exposure is the most important artificial source of exposure for population and it has an increasing tendency.

Consequently, the IAEA in cooperation with other international organizations introduced, in 1996, through the International Basic Safety Standards, the first requirements on patient protection that involved the importance of justifying and optimizing radiation doses.

Recent UNSCEAR estimations shows that medium dose per patient is about 200 times higher than that of medical personnel. Analyses of radiological practices in many states underlined an overuse of radiological examinations due, mainly, to medical system: lack of education in radiation protection of physicians (especially non radiologists), poor communication between different departments/hospitals and even technical mistakes.

These evaluations determined an international action plan for radiological protection of patient, involving a number of international organizations, such as UNSCEAR, ICRP, WHO, PAHO, European Commission, International Electro-technical Commission, International Organization for Standardization, medical physics (IOMP). All these actions dedicated to reducing medical public exposure take into account solutions such as:

– new requirements of training in radiation protection of medical personnel, including prescribes
– monitoring of each exposure and recording of all examinations per patient
– implementation of a national electronic health recording system which could also be accessed by EU countries

These initiatives should be easier performed if they are completed with educational and informing programmes for patients. Educated patients should cooperate with medical staff and other involved bodies to control their own exposures, being directly interested in results.

Programmes dedicated to education of patients should be developed in cooperation with the national education system and nongovernmental organizations for social assistance. If for physicians doctors it was considered worthy to issue a”guide on the utilization of radiological examinations and imagistic procedures”, an adapted guide for patients on such applications would be useful, too.

In some states these kinds of initiatives coming from medical side are already in place, consisting in dedicated websites. They could be completed by oral presentations in schools, universities and clinics. Oral presentations have the advantage of adjusting the information to auditors’ level of knowledge and interest for the subject and permit dialogue with participants. They could be completed by printed documents including images and graphics, attractive and easy to understand. For some countries the websites’ communications are quite restrictive and utilisation of direct communication is very suitable.

No matter how we address to the (potential) patients and their interest, we have always to answer to the same questions:

What are X-rays?
How do X-rays tests work?
What is the difference between X-rays tests?
What is the amount of radiation we receive from each of them?
How the organism responds to that amount of radiation?
What is the risk of developing cancer or other diseases from having X-rays tests?
What is the number of x-ray examinations that are advisable in one year?
What is the minimum period between two examinations?

To respond to these questions, accessible and friendly means and methods should be used, based on similitude with well known notions, such as: comparison of X rays with radio waves, light or ultraviolet rays, comparison of doses administrated in each examination with natural exposure expressed in days or months of exposure, graphics or images to compare different X ray tests.

Special attention should be given to the possibilities of minimizing risks of exposure in case of:

- repeating in short time some higher-dose X-ray tests (for example a CT scan or a barium meal or enema)
- being pregnant, or thinking you may be (your doctor and radiographer may look for other ways to make their diagnosis (If an X-ray is required in this situations, the test will either be delayed or supplementary preventive measures will be used to protect the unborn child).

Asking for a pregnancy test before having an X-ray procedure is, surely, the safest method to avoid an undesired exposure for young women.

One must know that if he/she needs to have an X-ray test soon after another one he/she would tell doctor about that. One of these examinations should be sufficient and another one would not be necessary. Patients should be educated to keep their own record on X-ray history to help medical decision.

To avoid a successively exposure due to patients movement during examination, they have to know before being tested that most X-rays and scans are entirely painless. Although a mammogram for example can be uncomfortable and for a number of scans an injection of dye into the arm is necessary, there is usually no other discomfort.

For children and their parents or accompanying persons it is necessary to be informed on special means used during the exposure if it is case, taking into account their possible reactions in unknown situations.

If a child is undergoing an X-ray test and their parents have been asked to hold him or her during examination, they have also to be protected with a lead apron.

Conclusions

- Radiology is an important method of investigation.
- Justification is necessary for each test.
- Medical stuff has to appreciate the importance of the test for establishing the diagnostic and decide when it is necessary and patients would make an option. Both medical staff and patients need information for a right decision.
- Basic education is important for both categories. If information may be required from the medical system, educational support should be provided by society.

Improving radiation protection culture: Social representations, attitudes towards risks and stakeholders involvement

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Abstract

The culture in RP is an integral part of the safety and prevention culture and it includes the complex of all those risk-based approaches chosen to set the standards and the philosophy governing those standards. The level of ambition against risk has evolved during the time on the basis of new information about the effects of radiation and considering changes of attitude towards risk. The concern on environmental protection is an emblematic example. Often it has been asked if really RP has been and continues to be a model with the capability to also influence the protection against risks of different nature than radiation, as regards to the proposed approaches in risk assessment and evaluation. In this paper the authors are proposing a multidimensional approach to the study of the RP culture by carrying out: 1) a comparative analysis of the various and different approaches which, during the past years, have been adopted in managing the technological risk both in the RP area and in other areas of technological and scientific knowledge; 2) the historical reconstruction of the evolution of the collective imagination about the possible risks connected with nuclear radiations. In the context of the modern societies, in fact, a wide range of factors of socio-cultural, ethic and political nature contribute to determine the social representations on which the attitudes to accept or to reject the technological innovation and its related risks are founded. We are convinced that by deepening the nature of the close connections of the co-evolution between the collective imagination about radiations and the different approaches to the governance of the technological risk, starting from the inclusion of different stakeholders, by also taking into consideration the lessons learned from experiences in other fields of activities, it will be possible to define useful cognitive tools aimed to strength the RP culture.

The EUTERP Platform: Towards a European approach for harmonisation in education and training for radiation protection professionals

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Abstract

In Europe, a common vision for maintaining competences in radiation protection is emerging, focussing on a common denominator for qualification of radiation protection experts (RPEs) and radiation protection officers (RPOs), and for mutual recognition and mobility of these professionals across the European Union and related countries. Started as an initiative of the European Commission the European Platform on Training and Education in Radiation Protection (EUTERP Platform) has been transformed into a legal entity under Dutch law. The Platform facilitates a permanent dialogue between all involved parties by the use of its website (*www.euterp.eu*), by issuing newsletters and by organising workshops. From the workshops several recommendations based on common agreement among the participants - were given to the EC, IAEA, IRPA and national authorities including proposals for definitions of the RPE and the RPO. Currently, the possible consequences for national legislation and E&T activities and guidance needed in relation to this proposed definitions of RPE and RPO that will be implemented in the Euratom BSS are the main focus of the EUTERP Platform. The role of EUTERP concentrates on the objectives to strengthen and harmonise education and training, and to facilitate the development of mechanisms of mutual recognition, based on a common approach. EUTERP will be provided with input from the ENETRAP II project. With these results it will: develop guidance on a methodology to compare the quality of training courses and training material; develop guidance on a standardized methodology of assessing the recognition of RP professionals as a basis for future mutual recognition, based on a description of roles and duties, education, training and work experience; develop guidance for a formal recognition process of the competences of RPEs and RPOs. EUTERP aims for being a European body on harmonisation of criteria and qualifications for and mutual recognition of RP professionals.

ENETRAP II: WP5 Develop and apply mechanisms for the evaluation of training material, events and providers

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Abstract

To maintain a high level of competency in Europe regarding radiation protection and to facilitate harmonisation and (mutual) recognition of Radiation Protection Experts (RPEs) and Officers (RPOs) quality assurance and quality control might play an important role. The ENETRAPII project (FP7-EURATOM) aims at developing European high-quality 'reference standards' and good practices for education and training in radiation protection. In Work Package 5 (WP5) the quality issue is addressed. Therefore, WP5 deals with the development and application of mechanisms for the evaluation of training material, training events and training providers by means of a transparent and objective methodology. The results can be used by regulatory authorities to benchmark their national radiation protection training programme and will be communicated to other networks, e.g. EUTERP. This paper addresses the first results of WP5: the comparison of training material. Training material is defined within WP5 as books and duplicated lecture notes.

The reference table that is developed for the comparison of training material is used to compare two Dutch text books for RPE/RPO training, and showed that the demanded content of this training is covered by both books on most of the subjects. For the remaining subjects additional material has to be used.

Introduction

The FP7 European Network for Education and Training in Radiation Protection II (ENETRAPII) project is a specific tool for EURATOM policy for E&T implementation in the radiation protection field and towards a mutual recognition of professional qualifications. The project will last three years.

Today's challenge in the field of radiation protection involves measures to make the work in radiation protection more attractive for young people and to provide attractive career opportunities, and the support of young students and professionals in their need to gain and maintain high level knowledge in radiation protection. These objectives can be reached by the development and implementation of a high-quality European standard for initial education and continuous professional development for Radiation Protection Experts (RPEs) and Radiation Protection Officers (RPOs).

For the purposes of this project the Radiation Protection Expert is defined as :

“An individual having the knowledge, training and experience needed to give radiation protection advice in order to ensure effective protection of individuals, whose capacity to act is recognized by the competent authorities.”

and the Radiation Protection Officer as:

“An individual technically competent in radiation protection matters relevant for a given type of practice who is designated by the registrant or licensee to oversee the application of the requirement of the Standards.”.

These are the definitions as proposed by the second EUTERP workshop in 2008 in Lithuania and submitted to the Euratom Article 31 Group of Experts.

To reach high-quality European standards for initial education and continuous professional development, there has to be agreement between the European countries concerning the duties and responsibilities of both RPEs and the RPOs. These standards are developed in Work Packages 3 and 4 (WP3 and WP4) of the ENETRAPII project.

When these standards are known, each country will be able to access and benchmark its own education and training against the European standards. It will also be possible for a country to benchmark an RPE or RPO, educated and trained in another country, to their own national standard. Shortcomings in education and training materials, events and providers become clear, when it is possible to compare education levels and national standards to the European standard. Therefore, one of the cornerstone work packages in ENETRAPII is Work Package 5 (WP5), entitled: *Develop and apply mechanisms for the evaluation of training material, events and providers*.

WP5 has started with an inventory of topics, items and subjects that need to be addressed in the education and training of the RPE and RPOs. These main subjects are subdivided in a reference table to come to a methodology of comparison. With this reference table each country can compare its own training and education methods with that of the European Standard (WP3 and WP4).

Material

The inventory has started with the subjects addressed in the syllabus EG133 (EC, 1998), the IAEA syllabus (IAEA, 2002), the European Master's degree in Radiation Protection syllabus – EMRP - (result of WP8 ENETRAP, www.sckcen.be/enetrap), the existing tables of subjects for education and training in radiation protection and similar tables used in different countries. This has to lead to a common reference table, which can be used to compare training material. Training material in WP5 is defined to be a text book or duplicated lecture notes.

IAEA Basic Syllabus PGEC

The IAEA basic syllabus (IAEA, 2002) can be used for post graduate students to become a qualified expert. The basic syllabus is split up in 11 modules, which cover the whole basic radiation protection. The duration of the course is 18 weeks.

Each module is divided in main subjects and these are subdivided in more detailed subjects. Only for the modules the number of hours spent is clear. For each main subject lecture notes and practical exercises are given.

EG Basic Syllabus 98/C133/03

In its communication 98/C133/03 (EC, 1998) the European Council guides the European countries in how to implement the basic safety standards 96/29/EURATOM in their own legislation. In this document the basic syllabus for the qualified expert in radiation protection is published as a list of subjects to be addressed in radiation protection training. Most of the subjects mentioned in the basic syllabus are not subdivided in detail. No information can be found about the hours spent on the different subjects, except for the statement: "the depth to which topics of the syllabus should be covered should depend on the level of advice/input required from the qualified expert". The listed subjects cover the basic radiation protection training and additional training in five different fields: nuclear installations, general industries, research and training, medical applications, and accelerators.

ENETRAP training scheme

The ENETRAP training scheme of the ENETRAP 6th FP project is based on the IAEA syllabus, the European basic syllabus, the EUTERP recommendations and other ENETRAP output. The scheme consists of different modules. The first three modules are the basic modules. Afterwards at least one additional module has to be followed, concerning the area of interest. This area can be Nuclear power plants or research reactors, waste management and decommissioning, non-nuclear research or oil and gas, medical, or NORM.

The ENETRAP training scheme modules are divided in main subjects, with the numbers of hours spent on all the main subjects. The main subjects are subdivided in more detail. It is not clear which level of education is required to enter the ENETRAP training scheme. The duration of the whole course is 42 days.

Tables of issues in radiation protection training

In the Netherlands a reference table – The first and the two last columns in Table 2 - is used since the 1980's for different levels of training in radiation protection. This table is divided in main subjects and subdivided in more detail. There are no numbers of spent hours in this table, but only a characterisation of the level of detail at which the detailed subjects are covered during the training, together with its training goal (Table 1). The advantage of using grades above hours spent on the different subjects is that the entrance level of students doesn't have to be set. Theoretical people with different levels can enter all courses.

Table 1. Grades at which subjects are covered in training material.

grade	Covered	goal
0	not covered	–
1	global, qualitative	aware of the subject
2	important subjects covered, quantitative	understanding of the subject
3	Detailed, quantitative	detailed understanding of the subject and able to work with it

Results

Since the main subjects of the different syllabi and reference tables are more or less the same, it does not matter which table to use. The reference table of the Netherlands is used, because it was ready to use.

As explained before, in this table no hours or pages spent on different subjects can be found, but only grades, corresponding to the grades in Table 1. The advantage is that the entrance level of students does not matter, just like the number of pages spent on a topic.

This reference table will be used for the comparison of training material and for the comparison of training courses. If the reference table is filled with the demands for the RPE, RPO and radiation worker (RW), it can be used to see which material can be used for which course and when other, additional material is needed.

The reference table is applied in a comparison of training material, i.e. two text books, to determine whether this table is suitable for this purpose. The first part of the comparison table can be seen in Table 2. In the first column the main subjects and more detailed subjects can be found. In the second and third column the grade of the different subject can be found for respectively book A and book B. Both books are written to educate student in radiation protection to the same end level. In the fourth and fifth column the demands for RPE are displayed for respectively small and large 'users' in the Netherlands.

The table shows that the books cover most of the items as asked for by the government. There are some minor deviations between the books, caused by the interest of the authors.

An exception is the mathematics which is not covered by either book. Therefore, an additional book has to be used during the training course.

In reality however, there is a difference between the levels of the books. This results in the usage of book A as study material for basic level RPE in small companies or institutions while book B is used for a more advanced level for small and basic level for large companies or institutions. This is not obvious in the reference table, since this is caused by the more detailed description of the topics covered in book B.

Table 2. The first part of the comparison table of training material. Book A and book B are compared to each other and to the demands in knowledge of the different RPE functions in the Netherlands.

grade	Subjects of basic radiation protection training covered	Goal
0	not covered	–
1	global, qualitative	aware of the subject
2	important subjects covered, quantitative	understanding of the subject
3	Detailed, quantitative	detailed understanding of the subject and able to work with it

	level of competence			
	book A	book B	3 (RPE of small institutes)	2 (RPE of large institutes)
Math				
- Differentiate, integrate, differential equations	0	0	2	2
- Exponential function	0	0	2	2
- Graphs (linear and logarithmic axis)	0	0	3	3
- Statistics (distribution, standard deviation)	2	2	2	2
General physical and chemical subjects				
- Composition of the matter	3	3	3	3
- Ionisation, excitation	3	3	3	3
- Nuclide Chart	3	2	3	3
Radioactivity				
- Proton - neutron ratio	3	3	2	2
- Radioactive decay, half-live	3	3	3	3
- Decay formula and –constant	3	3	3	3
- Mother - daughter relation	3	3	2	2
- Specific activity	3	3	3	3
- α-, β-, γ-decay, electron capture	3	3	2	2
- X-rays, Auger electrons	3	3	2	2
- Decay schemes	3	3	3	3
- Particle- and energy fluence and density	2	3	2	3
Activity from natural sources				
- U- and Th-decay diagram	3	2	2	2
- Cosmic radionuclides	2	2	2	2
- Other natural radionuclides	2	2	2	2
- Cosmic radiation	2	2	2	2
- Dose due to natural radioactivity	2	2	2	2
Artificial radioactivity				
- Nuclear fission, fission products	1	1	1	2
- Nuclear reactions, cross section	2	2	2	2
- Other sources	1	1	1	2
- Dose due to artificial radioactivity	2	2	1	2

Discussion and conclusions

The reference table can be used for the comparison of training material. All main subjects of the different syllabi can be put in this reference table and can be compared in detail with regard to these main subjects. Also, the different additional modules can be put in this table. For the competencies that need to be covered in the study material a similar table can be made.

When the table is filled out with the detailed grades of subject that have to be known by RPEs, RPOs and RW, the table can be used to show which book can be used by which part of the training modules for RPE, RPO and RW.

The reference table is probably too extensive to be used for a lot of comparisons. For this purpose the table can be made more condensed by taking some subjects together.

A further study will be carried out towards the usefulness of this reference table in the comparison of training modules and towards the usefulness of a competencies reference table in the comparison of training material and training courses.

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About implementation of EU requirements on education and training

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Abstract
The Council Directives require that each Member State shall take the necessary measures with regard to teaching, education and vocational in radiation protection. Moreover, it is foreseen that each Member State shall fulfill the requirements on establishment of an appropriate curricula and recognition system of competences in radiation protection and the undertaking is obliged to provide appropriate radiation protection training and information programmes for their personnel. In order to establish a harmonized framework and to avoid overlapping with national regulation, EU strategy on establishment of a common infrastructure for education and training in radiation protection throughout the European Union should be built. It should be defined the European agreement on the qualifications for training and education and requirements for mutual recognition of the competencies. Following the EU directions, each Member State should revise the national strategy in order to establish the national training needs, the system for credentialing radiation protection programs, national policy on the selection and training of trainers and to comply with the EU mutual recognition system. Participation in EUTERP and ENETRAP meetings on this subject represents a good opportunity for sustaining these proposals.

European Medical ALARA Network (EMAN): Supporting the ALARA principle in the medical field

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Abstract

The main objective of this project is to establish a sustainable European Medical ALARA Network (EMAN) where different stakeholders within the medical sector will have the opportunity to discuss and to exchange information on different topics relating to the implementation of the ALARA principle in the medical field. This network will also support the European Commission (EC) in its activities in this field. In addition, EMAN will aim to:

- Disseminate up-to-date information about literature, studies, research and good practices relating to the ALARA principle in the medical sector,
- Identify and communicate to the EC needs for development and update of European Union guidance,
- In particular cover the areas of education and training as well as continuous quality improvement as requested in the Directive 97/43 EURATOM
- Formulate proposals to the EC on harmonization issues,
- Propose to the EC solutions of identified issues at the European level,
- Establish co-operation with appropriate international organizations and associations.

To fulfil these objectives, EMAN will especially rely on:

- Three Working Groups, where three specific topics, all deserving special attention, will be widely discussed by professionals of the medical area:

on optimisation of patient and occupational exposures in CT procedures,
on optimization of patient and occupational exposure in interventional radiology,
on radiological safety for patients and personnel in activities using X-ray equipment outside the X-ray departments,

- A website to widely diffuse the information gathered and the work done by the network and to facilitate the exchange of information within the members of the network,
- A final workshop to present and discuss the work performed in the scope of the network and of the three working groups and to propose recommendations to the EC for improving the optimisation of radiation protection in the medical sector.

EMAN – the project

On the 23 October 2009 the European ALARA network project started. The project's main objective is to create a network supporting optimisation of medical and occupational exposures in the medical sector. Methodology of networking will be studied and efforts will be spent on setting up the network to create conditions favourable for its long term survival.

Three working groups will work with the optimisation process in different areas of medical applications, for computed tomography examinations, cardiac and non-cardiac interventional radiology procedures and x-ray examinations performed outside the radiology department, respectively. Much effort has been spent on recruiting highly experienced experts to the different working groups.

The project management consists of seven organisations with different perspectives and experiences of the field. The consortium consists:

- **Strålsäkerhetsmyndigheten (The Swedish Radiation Safety Authority), SSM, Sweden**
- **Bundesamt für Strahlenschutz (Federal Office of Radiation Protection), BfS, Germany**
- **Centre d'étude sur l'Evaluation de la Protection dans le domaine Nucléaire, (Nuclear Protection Evaluation Center), CEPN, France**
- **European Federation of Organisations for Medical Physics (EFOMP)**
- **European Federation of Radiographer Societies (EFRS)**
- **European Radiation Dosimetry Group (EURADOS)**
- **European Society of Radiology (ESR)**

In the consortium key professions are represented such as, radiographers, radiologists, medical physicists. It is planned during the project to broaden the medical competences involved, for example with prescribers, cardiologists and surgeons. Other stakeholders having an impact or interest in the optimisation process, such as manufacturers, hospital managers or the general public, will also be involved in the project. When the project ends in 2012 the objective is to have created an independent functioning European Medical ALARA network with a large potential to expand to the various fields involving medical exposures.

EMAN – the network

There exist many types of constellations called networks. Some networks are rather unstructured driven by common interests, others with a formal structure and management. Some are sponsored by organisations and some rely on voluntary work only. In the project we will investigate and make use of experiences from other networks in order to find the success factors and an optimised structure for EMAN.

Networking relies on communication and transparency, both internally and externally. Different communications tools, e.g. on information technology, has to be used efficiently. The opportunities for exchanging information are endless today. The project must develop a strategy for what tools to use and be flexible to use new tools.

Networks, both professional and personal, exist for different topics in the medical sector, and also in the radiation protection community. The future network will collaborate with other networks and organisations. Therefore it is necessary that the network, its mission and its vision is know by the other networks. Work has to be allocated to informing about the network.

EMAN – the working groups

During the project work of specified topics is going to be performed in 3 working groups.

- WG 1 on optimisation of patient and occupational exposures in CT procedures,
- WG 2 on optimization of patient and occupational exposure in interventional radiology,
- WG 3 on radiological safety for patients and personnel in activities using X-ray equipment outside the X-ray departments.

Each working group, in its special field of medical activity will:

- Collect up-to-date literature, on-going clinical studies and current reports concerning good practises on optimisation of radiation protection for patients and workers.
- Identify needs for development and update of guidance on optimisation of RP for patients and workers.
- Identifying needs and collect proposals to improve the European legislation on radiation protection in these medical fields.
- Contribute to the elaboration of the final workshop.

These working groups are also functioning to experience how collaboration and work can be performed in the future.

EMAN – the website

The EMAN website has been created at the beginning of the project with two main objectives:

- Facilitate the effective and efficient information exchange between the members of the network.
- Provide information to concerned stakeholders on the life of the network and on topics linked with ALARA in the medical field.

The web portal is organized into different sections. First of all, a specific section describes the objectives and the organisation of the network, and makes available a list of the network's members. Each of the Working Groups created in the scope of the network has a dedicated web page, where information on the work of the Working Groups will be found. Links to appropriate websites and to international organisations and associations cooperating with EMAN are also provided. After the final workshop of the project, the abstracts and slides of the oral presentations as well as the conclusions and recommendations will be made available.

The website will be regularly updated with news concerning the work undertaken by the network, publications, regulations, etc. related with ALARA in the medical field.

In order to collect information on the potential interested members of the future network, it is possible for any interested persons to register on the web site to the project newsletter which will be published regularly, as well as to register to a specific directory giving the names and contact of other interested persons.

EMAN – the workshop

In spring 2012 a European workshop is going to be arranged with the aim to widely diffuse the information gathered and the work done by the working groups and by the network in general and to facilitate the exchange of information within the members of the network. This workshop will also be an opportunity to draw and disseminate conclusions and recommendations from the network as well as to elaborate and discuss further actions to be made by the network.

EMAN – the future

The rapid change introducing new equipment and examinations is a huge challenge for all involved in the medical sector. A consequence of the increasing demand on productivity is that the optimisation process needs to be efficient. To have an efficient optimisation process will be very important but very demanding. More than ever it will be important to exchange experience and to share information in this field.

We believe that EMAN will contribute positively in this field; we have to work further on the methodology and gather experience from others and also listen to colleagues and interested parties in order to create an attractive network.

EMAN – contact information

All information can be found on the web site: **www.eman-network.eu**

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Thinking about stakeholders and safety culture in the ionizing radiation medical field

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Abstract
This works treats about:
1. The need to have a well built radiation protection frame not only from de scientific and technical point of view but from that of the organization and the stakeholder perception.
2. The importance of knowing how to relate and communicate to:
   Regulators
   Organization Directors
   Exposed and not exposed workers
   Patients
   People; media; judges and lawyers.
3. The need to improve and maintain the safety culture.

Conclusions:
- A deep built radiation protection organization system maintained and actualized is needed.
- The radiation protection experts must have a continuous training related to technical radiation protection aspects and how to communicate to stakeholders.
- The safety culture must be a part of the quality assurance program with a systematic incidents' review and the spread of the measures taken to minimize the appearance of errors.

Introduction
In relation to the majority of other risk agents presents in our actual society the ionizing radiations (IR) are one of the most studied and known ones. The justification, optimization and limits system is logic and a well built system.

Though many things should be susceptible of harmonization, the expert's language is understandable and common between them. Magnitudes, calculation units and so on, are well defined and there are plenty of reference organism publications and guides to support that. But, when we go into the social field the issue gets complicated and we go into other world with more and not so well defined variables. So we can say that if we are good experts but we do not control the play rules in every case, we are not doing our work well enough.

Nowadays, and after great efforts, the IR Medical Physics and the related RP , is included into the Health Sciences field specialities like that of medical doctors under the name of "Radiofísica Hospitalaria" (RFH) and his activity is developed in: Radiotherapy. Radiology, Nuclear Medicine and laboratories that use IR, including in, all these areas: The patient, workers and public RP.

The RFH title is obtained passing an exam similar to that of medical doctors, but referred to physics, and three years (we are asking for four actually) working at one hospital with a Health Ministry Accredited Teaching Unit.

The RI Spanish legislation that applies to heath field, is numerous and systematically updated [1, 2, 3, 4, 5]. The quality is contemplated in three laws, divers national quality protocols related to: Radiotherapy, Radiology and Nuclear Medicine and a dynamic Health Forum with participants of the: Two reference Spanish societies; Medical Physics (SEFM) and RP (SEPR) and so of the CSN (Regulator Organism). Looking at this we can say that the frame to support and develop the system is solid.

Material and methods

Our "Radiofísica y Radioproteccion" Service is integrated by seven RFH, five technicians and a secretary, but we have to work also at centres outside the hospital with Nuclear Medicine in two of them and plenty of RX units.

The actual philosophy of our hospital, concerning to RI field, is to get the ISO 9001/2008 certification. Our service and the Nuclear Medicine one have already gotten their certificate. Radiotherapy and Radiology Services are in process to achieve it this year. Besides that way, the hospital got, last year the 14001 Certification for environment.

Our certification is referred to:

- Patients, exposed workers and public RP.
- Physical and clinical dosimetry.
- Equipment quality control
- Radioactive installations management.
- RFH Teaching Unit.

Our "Radiofísica & Radioproteccion" Service is integrated by: Seven RFH , five technicians and a secretary, but we have to work also at centres outside the hospital with Nuclear Medicine in two of them y plenty of RX units.

The Service Head has, also, the RP Responsible title according to CSN requirement.

Results and discussion

Thinking about, a good ISO Organization, we must have efficient and practical procedures for:

- Making appropriate technical reports within the legal framework and to enable its compliance.
  Giving a rapid and scientific response in "special" situations, incidents and accidents.

Radiophysics and RP procedures should be indicated as a reference for medical services mentioned above; and a good management of the whole system must have an active role in the development and implementation of quality programs.

With respect to RP training programs for all workers exposed to ionizing radiation, there is a national program for physicians who are doing their specialty. This training is based on a 6-hours course that is organized by the RFH and RP services in each hospital. The course is mandatory for most medical specialties and it carried on during the early days after their incorporation to the hospital. In addition, for those who are more concerned with IR, there is 4-6 hours more of training. Those who are working in: Radiology, Nuclear Medicine and Radiotherapy have a 50 or more training hours which includes a course approved by the CSN. Technicians and some nurses of these services also have an accreditation or license granted by the CSN.

We believe that in order to have an effective system, day by day, there is a need for continuing education in open, practical, and enabling participation sessions and seminars without forgot interdisciplinary meetings in critical situations.

To organize and actively participate in all this is a very important job in the role of the RFH and RP Services.

The information of unexposed workers, patients and people who contribute to his comfort and the information associated with critical situations has to be translated from a technical language into something understandable and that built up confidence in the auditorium.

It is very important to disable the unfounded and unreasonable fear for radiation and prevent, in some cases, the use of this fear to obtain unjustified “collateral benefits”, but, at the same time, we must properly listen to the stakeholders in every circumstance and know their problems and perceptions. We must answer to them in an understandable language but without losing the technical foundation. It must seem truthful.

We must avoid the use of RP for other purposes. For instance, the porters, who transfer patients to RX services for radiological probes, said that they wanted personal dosimeters like the ones the technicians have. We organized briefing session for them and we realize, they felt something pejorative when we said they are classified as common public. We had to make clear all about the limits for the public and so on.

Information to patients and caregivers also has its own characteristics. It is important that the information shall be contained in clear and brief notes adapted to each set of circumstances but, at the same time, there should be an open communication line to resolve particular questions (this often occurs in the case of patients receiving radiopharmaceuticals, pregnant women etc).

Excessive exposed and unexposed workers IR fair may create situations of improper alarm or confusion. For instance, in a hospital, the Patient Care Service received a complaint letter from a father who accompanied his young son in a big common room in the Intensive Care environment area because he observed that when the RX technician went into the room to make a radiography to another baby he shouted ¡¡ RAYS!! And then all the staff leaved the place running. It is necessary that, in such situations, both, the technician that manage the RX equipment and the rest of the professionals act properly and inform the family that at a certain distance of that RX equipment, a person does not receive any measurable radiation.

It is obligatory to have visible written rules, but the process requires something more: To keep an open contact with the workers periodically; to collect impressions and feed back and to take into account the different perceptions, mainly for new workers.

Concerning Media issues related to external events, the SPR must have appropriate minimum information but the issue can be channeled to reference scientific societies experts well prepared for that.

With respect of internal issues, we have to act quickly but with the security that technical data provides.

There are two kinds of media: The scientific and documental and the "daily" one. In the first case we have time and possibility to use a logical and technical treatment of the problem. The real challenge is to manage the second one. It is frequent to listen something that "good news are not news" and " the bleeding news are more engaging for the auditorium". Then the RP experts should prepare themselves for that, because another kind of response and attitude is required.

In relation to lawsuits, the "linear not threshold hypothesis" can generate problems, if the judge finds that any microsievert can be carcinogenic, even if the RP system is correct and according to the norm.

If a negligible amount of radiation can have this consideration, the system ALARA would not support itself, and workers and those present in the radiation facility would have an excessive weighting factor and unjustifiable risk considerations. We must give an appropriate weight to radiation and not create unfairness to other agents.

We must add that the Radiation Protection Service should work together with the Occupational Risk Prevention and the hospital Health and Safety Commission in which there are trade union representatives who must be informed.

National and international organizations (ICRP, IAEA etc) have a great concern for the safety of radioactive sources. It is necessary to efficiently organize the safety of equipment and radiation sources in RT and MN. The safety of patients in RT is based on a good quality program that affects to equipment, facilities and treatments. All interdisciplinary team: Medical doctors, RFH and technicians must know their work very well, case by case, and they must interact appropriately between them in each patient.

Open communication, progressive incorporation of well informed new workers, regular sessions and good incidents follow up is essential.

This approach can be adapted to nuclear medicine and radiologic therapeutic interventional procedures.

Conclusions

- A deep built radiation protection organization system maintained and actualized is needed.
- Radiation protection experts must have a continuous training.
- The Clinical Medical Physicist; Qualified Expert in Radiological Protection, should include, into his skills, the relation with a broad stakeholders field.
- Each stakeholder group has to be listened and well considered but each special problem must be, always, managed under the suitable technical framework.
- The ionizing radiation workers need to be well informed and protected but shouldn't be treated better than other workers with similar level of risk

- The safety culture must be a part of the quality assurance program with a systematic incidents review and the spread of the measures taken to minimize the appearance of errors.
- A well informed media could be one of the best paths to spread the radiation protection
- Adequate Conduct Code (IRPA, IAEA, Montbeliard[6] … should be incorporated to the practice.

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4. Real Decreto 1085/2009 de 3 de julio por el que se aprueba el Reglamento Sobre Instalación y Utilización de Aparatos de Rayos X con Fines de Diagnóstico Médico.
5. Instrucción IS-18(2008) del CSN sobre notificación de sucesos e incidentes radiológicos.
6. Processes and tools for stakeholders engagement in radiological protection. Conduct Code Post Montbeliard (France) Meeting.

Better evidence-based quality in radiographic imaging by e-Learning?

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Abstract

The change from film-screen radiographic imaging systems to digital imaging systems has brought vast challenges in medical imaging services. Firstly, new general guidelines and working models optimising the radiation dose and image quality are needed.. Secondly, the challenge derived from the first one is to update the competence of the staff working with imaging units and to train the new health care personnel so that their competence could match the needs of the new technique.

The project's purpose is to increase the competence of staff working in imaging units by evidence based education in digital imaging, and dose optimisation according to the principles of the ICRP and DIMOND3 in the three Scandinavian countries: Finland, Sweden and Norway. The specific objectives of the project are to:

1) Plan and implement Scandinavian evidence-based course in digital imaging and dose optimisation, and the materials needed on the basis of national and international regulations about the subject in 1st, 2nd and 3rd cycle education degree and life long education,
2) Produce the materials for the education and
3) Evaluate the evidence-based course plan in quality of digital imaging and the materials produced.

The project management group consists of radiography lecturers, principal lecturers and physicists of College University of Gjøvik - Norway, Karolinska Institutet - Sweden, University of Oulu –Finland, Oulu University of Applied Sciences – Finland and Helsinki Metropolia University of Applied Sciences – Finland. Also students of these organizations are involved in the project from very beginning.

The project is realized between autumn 2008 and spring 2011. In the presentation the main results of the project tills summer 2010 are presented.

Introduction

The change from the film-screen (conventional) radiographic imaging systems to digital imaging has brought great challenges in medical imaging services. Firstly, general new guidelines and working models optimising the radiation dose and image quality of radiographs. Secondly, the obvious challenges derived from the first one are to update the competence of the staff working with imaging units and educate the new health care personnel who are in the middle of their studies to match the needs of the new technique.

According to the studies (e.g. Al Khalifah & Brindhaban 2004), lower levels of radiation dose setting can be established for computed radiography (CR) systems than the standard values used for film-screen systems to produce diagnostic images of equal quality. This is especially important in the examinations performed for children (Livingstone et al. 2008). Flat panel detectors (so called direct radiography, DR) gives even more possibilities for dose reduction and still gives at least as good image quality as with CR (e.g. Willis & Slovis 2004; McEntee M et al 2006; ICRP 93 2004) At the European level, European Councils Directives about radiation protection define at least the minimum criteriafor quality of imaging services (Directive 84/466/, 96/29/ and 97/43/Euratom). The Council Directive 97/43 Euratom defines Quality assurance as ‘’all those planned and systematic actions necessary to provide adequate confidence that a structure, system, component of procedure will perform satisfactory complying with agreed standards.’ The latter part of the Directive describes more specifically the general principles of radiation protection. At the moment Radiation Safety Authorities in all the Scandinavian countries are working on to make national guidelines for digital imaging taking account both the quality assurance and dose and image optimisation. They do this in co-operation with universities and Universities of Applied Sciences that are educating health care personnel who use radiation in their work.

Guidelines and working practices in radiography must be done systematically by using research findings and best available evidence as the basis. Evidence based medicine (EBM) has been defined as ‘the conscientious, judicious and explicit use of current best evidence’ in clinical decision making (Sackett et al. 1996). In studies in the radiography area, in addition to the concept *evidence-based radiography* (Keenan et al. 2001; Ebrahim 2005), also terms such as *evidence-based medical imaging* (e.g Smith 2008), *evidence-based practice* (e.g. Pitt 2004; Pickersgill 2007), *evidence-based clinical practice* (Bonnetti et al. 2006) and *evidence-based medical practice* (e.g. Omorphos & Kontos 2003; Banerjee & Van Dam 2006) are generally used.

In this project, our purpose is to increase the competence of staff working in imaging units by evidence-based education in digital imaging and dose optimisation according to the principles of the ICRP and DIMOND III in the three Scandinavian countries: Finland, Sweden and Norway. Targeted objectives are to: 1) plan and implement Scandinavian evidence-based course plan in digital imaging and dose optimisation and the materials needed on the basis of national and international regulations about the subject in 1st, 2nd and 3rd cycle education degree and life long education. 2) produce materials for the education and 3) evaluate the evidence-based course plan in quality of digital imaging and the materials produced.

The project unifies the Scandinavian knowhow in the radiography field in different levels of professional and adult education. It also fosters the professional status of the radiography education and radiography science. The competence of imaging units staff working in hospitals, health centres and clinics will be developed in a area of digital imaging and dose optimisation. As a result of this, quality of digital imaging will be developed and radiation doses of the patients will be lowered.

Material and methods

Organization of the project

The project group consists lecturers (radiography and physicists) and principal lecturers in higher education institutions. They give first cycle Bachelors level (juvenile and adult) and second cycle Masters level education in radiography and radiotherapy in three Scandinavian countries. From Finland the network includes two Universities of Applied Sciences, from Sweden one University and from Norway one University College.

The steering group of the project consists of the members of Radiation protection Authority in Finland (STUK), Oulu University, Finnish dentists association Apollonia and member of Finnish Research Society of Radiography.

The projects' working principles and ee-learning philosophy

The project group meets approximately once a month in a web conference via Adobe Connect Pro (ACP) videoconferencing system, and twice a year face to face. Mean while the project group members work in the Moodle-platform in the web. The project group uses evidence-based way of working: seeking consensus of peers, experts and member of radiation protection authority, making literature surveys, taking into account national and international guidelines about digital imaging quality assurance and radiographers competency and consulting the radiography departments' staff. Also radiography students take part the project. They prepare some contents and learning material for the e-based course, comment and test it. They work in a co-operation with the multimedia technology students preparing technical solutions for the course.

The learning philosophy of the course was chosen in the beginning of the project on the basis of the basic idea of the project, which is to learn evidence-based radiography. At first the core competencies of the course module were formed on the basis of literature survey (e.g. Medina & Blackmore 2006; Cronin 2009 a and b; Kelly 2009 a and b), national (STUK 2004) and international recommendations (e.g. DIMOND III 2003; ICRP 2004), critical views of project group members and experts. On the basis of core competencies, the core contents of the learning modules were derived. On the basis of core competencies and core contents, learning technological solutions were chosen. This means that the mode of learning is mainly chosen according to the substance to be learned.

Timetable of the project

The project started in autumn 2008, and it operates for two years, untill autumn 2011. During the autumn 2008 and spring 2009 the project organisation and common principles of action were formed. The Scandinavian project group had its first meeting in Helsinki in December 2008. It started planning the core competencies of the evidence-based course plan for the juvenile level. During autumn 2009 the core competencies and some of the learning material became ready. Also the first part of the course module was tested in Helsinki University of Applied Sciences. During spring 2010 the curriculum and learning materials for the Bachelor level was finished and tested. Experiences of those were collected and used for revision of the course. The group also started planning the core contents and learning materials for the Masters level course.

During autumn 2010, the project group will finish the Masters level educational package and test it in Oulu University and Metropolia University of Applied Sciences (Masters Programme in Clinical Expertise). During the spring 2011, the project aims to make PhD level course ready, and if possible, also to test it at least by some radiographers having their PhD studies in the Scandinavian Universities. Also the final evaluation of the learning materials and evidence-based course plan in quality of digital imaging will be performed. In the summative evaluation the aims of the project are compared to the achievements of the project. Also unintended results of the project are evaluated. The value of the educational program for the needs of the working life is asked (theme interviews for key persons at the imaging units) and evaluated. (see e.g. Frechtling et al. 2002)

Results

Evidence-based radiography is a learning philosophy as well as learning contents of the project. This is to say, evidence-based radiography is learned by evidence-based method. In evidence-based learning philosophy, the research evidence and other best knowledge (e.g. national and international guidelines and recommendations) combines with the technical and clinical experience of the student, his/her peers and experts. E-based learning fits to this kind of learning philosophy very well. That is why the project group started to call it ee-learning.

As a result of the project groups' evidence-based way of working described in Materials and Methods section of this article, ten core contents and respective themes of the learning module were formed for 1st cycle Bachelors level education. These were: 1) The basic concepts, methods and process of evidence-based radiography, 2) Decision making in radiography, 3) Technical Quality Features of Digital Images, 4) Visualizing the important information of the image in an optimal way, 5) Applying demands for positioning and collimation for digital radiography, 6) Managing digital radiographic equipment, 7) Applying diagnostic requirements in projection radiography, 8) Applying post processing techniques to improve diagnostic image quality, 9) Assuring and optimizing image quality, and 10) Motivating multiprofessional way of working and documentation. When defining core competencies and core contents of the learning module, it was taken into account that they apply European Qualifications Framework level six adopted by the European Council on 23 April 2008. The EQF is a common European reference framework, which links countries' qualifications systems together, acting as a translation device to make qualifications more readable and understandable across different countries and systems in Europe. (European Council 2008)

It was also defined core contents for each theme based learning module. As an example core contents for the theme 'The basic concepts, methods and process of evidence-based radiography' were defined as follows: The student knows concepts, principles and purpose of evidence-based health care, recognizes the phases of evidence-based radiography, is able to differentiate between sources of evidence, knows what are the most central health care databases, knows how to use scientific databases correctly, knows how evidence-based knowledge is applied to the decision making and development of radiographic practice, is able to make narrative reviews, is able to read and understand systematic reviews and recognizes the connection between radiography science and evidence-based radiography.

Core contents for learning module ' Technical Quality Features of Digital Images' were defined as: The student applies Fryback´s and Thornbury´s model of the efficacy of diagnostic imaging, understands the dilemma technical vs. diagnostic image quality, applies basic properties of a digital image: features relating to (technical) image quality (contrast, noise, geometric distortions, artefacts), and effect of exposure settings (kV, mAs, filtration, grid use….) and object positioning on image quality.

From the viewpoint of e-learning, the project applies mixed mode setting. Individualized self-paced e-learning is used online e.g. when the students search for evidence-based knowledge about the quality assurance (QA), and offline e.g. in learning dose area product measuring (DAP-meter). Also the modes of group-based e-learning synchronously (e.g. videoconference lectures and face-to-face meetings on the web) and asynchronously (e.g. in discussing about problem cases in QA via Wiki-based IT technical solution) are used.

Discussion

Working in a multinational and multiprofessional project group has been very interesting and challenging. It took for half a year for the project group to form a common view about what is quality assurance in digital imaging and how it should be defined in this project. This difficulty came up because radiographers responsibilities in quality assurance differ in Finland, Sweden and Norway. We had to find out what are the central competencies radiographer in all these countries should possess. A great help in this work was achieved through literature reviews and discussions with peers, experts and radiation protection authority. We also had to define the basic level presupposed when the students enter the course. It was defined what the students should know about evidence-based radiography and digital imaging quality assurance before starting this course. Although half a year in a two years project is quite a lot, as an experience we would say that it was a right thing to do. After having a clear view about what to do, the rest of the project is just to do it.

Project group works mostly at the web by using the Moodle platform and Adobe Connect Pro (ACP) videoconferencing system. At first, using the Moodle was a bit difficult at least for those who were not familiar with it. We had not clear user's instructions in English language, which caused that the learning method was consulting each other and trial and error. After some difficulties in the beginning, the Moodle proved to be quite a handy platform of working in this kind of project. The ACP videoconferencing system suited also very well for the work. Seeing each othe's faces and hearing the speech is quite different from just communicating via text. It gives a sense of belongingness to the project group and this can and was in our project been seen straight as a productivity of the project group.

Radiography students of the higher education institutes taking part of the project were involved in it at the early phase. The project was integrated to their studies. The students reported that they liked a lot studying in this kind of 'real' project, although it was sometimes very challenging. They learned many kind of things while doing the project: evidence-based way of working, English language, digital imaging quality assurance, problem solving skills, team work and project management. The students wrote articles about their working at the project and took part project groups'work in face to face and ACP meetings.

As the parts of the learning module were tested with the radiography students, it was reported that most of the students had difficulties in understanding the English vocabulary. However they felt that the content of the course was mostly easy. There were some difficult concepts, which needed clarification. A large part of the students hoped that the course would be in Finnish or at least have an English-Finnish dictionary for the difficult words. The course will continuously be tested by radiography students as well as by radiographers working at the hospitals. It will be revised according to feedback received between the testings.

The project group will translate the course to national languages after the project will be finished. The course has been built up to the Moodle in a way that it can be transferred to other web based platforms. This is necessary because higher education institutions use different platforms in their e-based teaching and there is no point bothering the students to learn the use of new platform for one course. Instead of it they can concentrate on learning the contents of the course.

So far the project has proceeded approximately in its timetable and reached its' objectives. The project produces common Scandinavian understanding about what is evidence-based radiography and digital imaging quality assurance. As unintended outcome of the project, R&D networking on the area of radiography among Scandinavian countries has got some impetus, which we hope, will continue.

Conclusions

As a result of the project, a learning module consisting of ten core contents and respective themes for evidence based digital imaging quality assurance was formed for 1st cycle Bachelors level education. The learning module also comprises learning material and learning tasks. In the multinational and multiprofessional quality assurance project one of the most challenging things is differing responsibilities of radiation using staff. This will become easier in the future as the national guidelines for digital imaging will be developed. According to the experience of this project evidence and e-based way of working is very useful to develop new applications in the field of radiation use quality assurance. Ee-learning has also proved to be a useful learning philosophy in radiography.

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Monte Carlo based PCXMC-program as a tool of learning dose optimisation in plain (project) radiography

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Abstract
Effective dose can be defined for X-rays as the sum of the weighted average absorbed doses in all the tissues and organs of the body. International Commission on Radiological Protection (ICRP) has produced a list of tissue weighting factors for a number of organs and tissues. The calculation of effective dose is based on the Monte Carlo technique or direct organ dose measurements. PCXMC is a widespread and well tested Monte Carlo program for calculating patients' organ doses and the effective dose in medical X-ray examinations. The anatomical data is based on the mathematical hermaphrodite phantom models of Cristy (1980). The program computes doses in freely adjustable x-ray projections and other examination conditions. The program describes patients of six different ages: newborn, 1, 5, 10, 15 -year-old and adult patients. Radiographer students calculated effective doses to patients of different age and changed technical parameters one by one. They analysed the results and compared their own result to those found in literature. With PCXMC it is also possible to evaluate the risk to the patient due the x-ray examination and compare the difference in organ doses calculated based on the data of the ICRP publications 60 and 103. The students have found out the benefits of PCXMC in dose optimisation. It is easy to point out, what kind of changes have to be done and how much the changes help to decrease the dose.

Introduction
It is well known that ionising radiation may course harms. From stochastic point of view, there is no evidence at present of a threshold to the radiation dose below which there is no risk. The probability - but not the severity - of stochastic harm increases along with increased dose. In addition, the latest studies have indicated that there may also be effects on the lower levels and consequently the optimisation of an individual dose is important. (Hall et al 2004, International Commission on Radiological Protection 103 2007.)

The effective dose was developed by ICRP to reflect the fact that some organs are more sensitive than other organs. They have a higher risk of producing a cancer or another deleterious effect. The effective dose also helps in evaluating the biological effects of radiation (International Commission on Radiological Protection 103 2007).

The effective dose is the sum of the effective doses for exposed organs. The calculation of these doses requires knowledge of the size and configuration of the individual, the geometrical projection of the beam, size and location of the primary and scatter beams, Entrance Surface Dose and x-ray beam energy spectrum. The radiation quality factor is needed because of the different biological effects when interacting with tissue. For x-rays, the quality factor is unity (Lampinen 2000), which according to International Commission on Radiological Protection 74 (1997) and Marttila (2002) means that the effective dose can be defined for x-rays as the sum of the weighted average absorbed doses in all the tissues and organs of the body. The unit is Sv (Sievert). International Commission on Radiological Protection has produced a list of tissue weighting factors for a number of organs and tissues and they are mean values for the whole population. The factors were published first time in Publication 16 (1970), re-evaluated in Publication 60 (1991) and once again in Publication 103 (2007).

. The calculation of effective dose is based on the Monte Carlo technique described in NRPB-R186 (Jones, Wall 1985) or direct organ dose measurements by TLD (McCollough, Schueler 2000). The anatomical data in PCXMC is based on the mathematical hermaphrodite phantom models of Cristy (1980), which describe patients of six different ages: newborn, 1, 5, 10, 15 -year-old and adult patients. PCXMC is a widespread and well tested Monte Carlo program for calculating a patient's organ doses and the effective dose in medical x-ray examinations (e.g. Tapiovaara et al. 2000, Geijer et al. 2001, Tort et al. 2001,Dimov, Vassilieva 2002, 2002, Hansen et al. 2003, Uffam et al 2005). The height and weight of the patient can be set from a real patient. All organ doses calculated by PCXMC relate to the patient entrance air kerma (free-in air, without backscatter) at the point where the central axis of the x-ray beam enters the patient. The datum can be obtained by combining data on the examination techniques and the radiation output of the x-ray source or by using the surface dose or Dose-Area Product measurements of actual patient examinations. (Tapiovaara et al. 1997, Tapiovaara et al 1999, Tapiovaara, Siiskonen 2008.) This enables the calculation of the effective dose related to the x-ray field size and provides a technique for monitoring the collimation of the x-ray field.

Material and methods

The concepts like absorbed dose, entrance surface dose, Dose Area Product, organ dose and equivalent and effective dose are first discussed and clarified. The PCXMC program is introduced to students by two hours lesson and demonstration. After that the students practice by them selves to calculate the doses according the given examples in small groups for two hours supervised by the lecturer. They calculate first dose to an adult person and then the perceptions are discussed with the whole group. The aims of this part are at first to make the program familiar with the students so that they understand the concepts used in PCXMC program and secondly that they can use the program independently in a correct way.

After that the students get the written tasks. First they calculate doses for five patients in some examination. This data has been collected earlier in clinical practice for the ESD (Entrance surface dose) calculation. Next they pick up one patient and calculate doses changing parameters one by one. The changed parameters are: Tube potential (kV), added filtration, height of exposed field, width of exposed field, projection (ap /pa), age and focus to skin distance and finally both kV and mAs (e.g. kV from 70 kV to 80 kV and from 20 mAs to 10 mAs to keep to dose on the same level). Students evaluate the effect of changes to the whole body effective dose and organ doses organ by organ. They also have to calculate effective doses to a new born, 1, 5, 10 and 15 years old child in one examination e.g. chest ap. The risk analyses are made based on some of those calculated effective doses. The PCXMC program calculates the whole body effective dose according to tissue weighting factors from ICRP 60 (1991) and ICRP 103 (2007). It is easy to find e.g. the differences in effective dose between female and male with the same dose as well as the correlation between the risk and age.

The students write also a report in pairs about what they have done and found. The results and observations must be compared to those from literature at least with ten scientific articles. They also have to evaluate and reflect their own experiences and learning outcomes as well as how to use the program later in clinical practice.

Some examples of these reports are shown and the students own learning outcomes has been analysed by content analysis method.

Results

The students understood the basic concepts of the PCXMC program during the two hours lesson and demonstration. The two hours training in small groups clarified the concepts and the students could after that session use the program correctly in order to calculate doses. In small groups the students had different examination (chest, pelvis and lumbar spine ap (anterior-posterior) and pa (posterior-anterior). the results were discussed and there was a lot of discussion about the differences and how to transfer them to clinical practice. The next tables will demonstrate one student's calculations in adult patient's chest examination (Table 1).

Table 1. Five patients in chest examination used in calculations.

	Length	Weight	kV	mAs	BMI	ESD (mGy)
Female	170	55	125	1,26	19.03	0.07
Female	178	80	125	1,73	25.24	0.11
Male	175	86	125	1,79	28.08	0.11
Male	166	60	125	1,08	21.77	0.07
Female	164	77,6	125	1,65	28.85	0.1

All patients are adult. BMI is body mass index calculated from the patient's height and weight. The ages varied from 23 to 55 years.

Table 2. Organ doses and effective doses according to ICRP 60 (1991) and ICRP103 (2007) for five patients introduced in table 1 calculated with PCXMC program in chest examination.

	Dose (mGy)				
Organ	Female 179/55	Female 178/80	Male 175/86	Male 166/60	Female 164/78
Bone marrow	0.009411	0.010732	0.010377	0.007748	0.010495
Breasts	0.008188	0.008722	0.008362	0.006013	0.007866
Colon	0.000241	0.000271	0.000354	0.000287	0.000506
Lungs	0.025360	0.031904	0.030964	0.021119	0.029790
Heart	0.012914	0.014500	0.013495	0.009985	0.013336
Thyroid	0.003530	0.004531	0.003427	0.003492	0.003874
Effective dose ICRP60 (mSv)	0.007070	0.008187	0.008047	0.006305	0.008884
Effective dose ICRP103 (mSv)	0.007849	0.009013	0.008939	0.006516	0.009112

The results in Table 2 show that effective dose is dependent of the patient's size and there is difference in doses calculated according to the ICRP 60 and ICRP103.

Table 3 shows differences in doses with same patient with different technical settings. In chest ap / pa it is easy to find the benefit in pa projection both in organ doses and in effective doses. Most important difference is in the dose of breasts and active bone marrow. Also the thyroid dose decreases in pa projection although it was not in the primary beam. The effective dose in chest ap is increasing more with the ICRP 103 weighting factors due the change in breasts factor. Added filtration decrease all doses and Cu more than aluminium. Higher tube potential and lower mAs decreases also all doses.

Change in the size of exposed area decrease patient dose also: two centimetres smaller area in height gives 18 % decrease in effective dose. Distance is also one parameter influencing to patient dose. With 30 cm longer distance from the x.ray tube focus to patient's skin, the dose to patient is lower and in digital imaging the mAs and kV can be on the same level, the image quality is still equal.

Table 3. Doses to one adult female patient in chest examination calculated with PCXMC program with different parameter settings (on the first row the basic dose with 125Kv, 3mmAl, FID 200cm, 125kV, 2,4 mAs, in others only documented parameter has been changes from the basic settings**)**

Parameters	Average dose in total body	Effective dose mSv (ICRP 60)	Effective dose mSv (ICRP103)	Breasts mSv	Lungs mSv	Active bone marrow	Thyroid	Oesophagus
Pa	**0.014488**	**0,016581**	**0,018372**	**0.017273**	**0.065154**	**0.021996**	**0,005304**	**0.037316**
Ap	0.014182	0.023271	0.029565	0.107345	0.055809	0.012905	0.008714	0.028139
Filtration +2mmAl	0.011650	0.013596	0.01506	0.014530	0.052790	0.017851	0.004491	0.031598
Filtration +1mmAl+0,1Cu	0.010305	0.012232	0.013608	0.013378	0.046896	0.015943	0.004156	0.029293
kV135, mAs1.2	0.008726	0.010116	0.011221	0.010731	0.039393	0.013369	0.003296	0.023265

The students also make risk assessment with the program. Figures 1 and 2 show the difference in risk with same dose and age between male and female. The risk is very small due one chest pa projection but, as seen, female has higher risk. Due to one chest x-ray a female has loss of life expectancy is 0,3 hours and and male 0,1 hours. In other risks the male has higher rate.

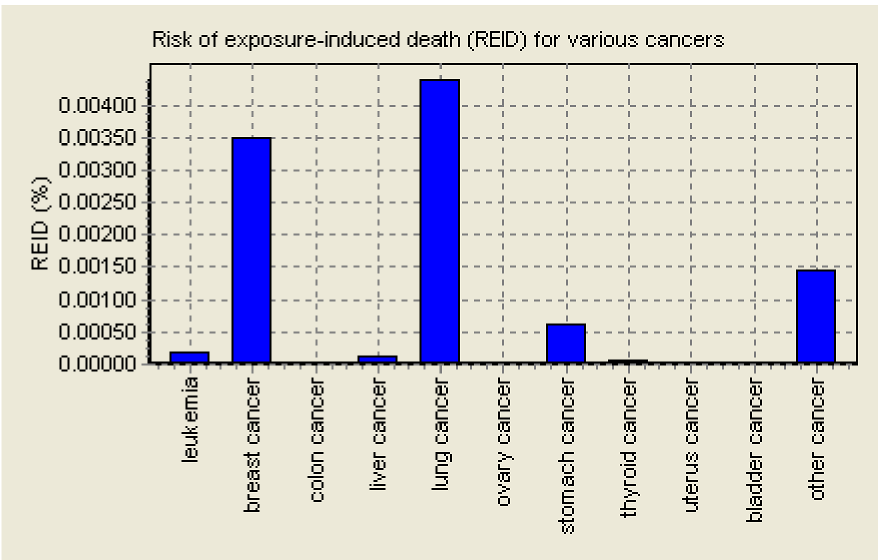

Figure 1. Radiation risk assessment: Stochastic radiation risks (Finnish mortality data). 20.0 year-old female, Expected length of remaining life 59.5 years, Risk of exposure-induced cancer death (REID): 0.000176 %, Cancer mortality for other causes; not related to this exposure 17.1 % , Loss of life expectancy (LLE): 0.3 hours.

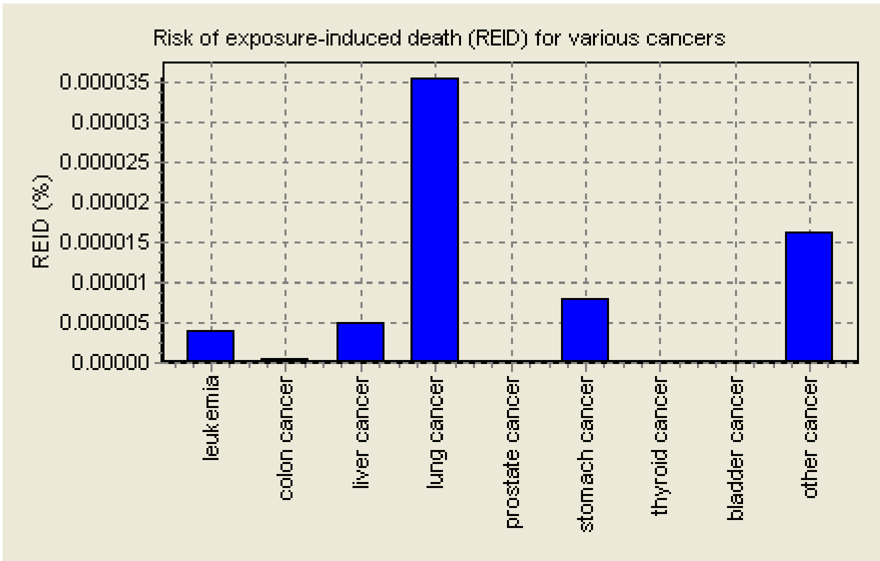

Figure 2. Radiation risk assessment: Stochastic radiation risks: Finnish mortality data, 20.0 year-old male, Expected length of remaining life 52.9 years, Risk of exposure-induced cancer death (REID): 6.95E-5 %, Cancer mortality for other causes; not related to this exposure: 20.1 %, Loss of life expectancy (LLE) 0.1 hours

The younger the child is the higher is the risk. It is also possible calculate cumulative risk due to many x-ray examinations.

Discussion

The PCXMC program can be used as a new tool to evidence the effects of the technical parameters and projection view to the patient dose. There are several examinations which can be taken either as an ap or pa projection. In dose optimisation it is easy to calculate both organ doses and effective dose and compare the difference between them and make the decision based on the evidence based results. It is also possible to calculate the dose to a foetus, if needed, especially during the first half of pregnancy (Perisinakis 1999, Kettunen 2004).

Tube potential (kV) affects to the contrast of the image. With higher kV the radiation beam is more penetrative and may for that reason cause loss in image quality especially in soft tissues and ribs (Uffman et al 2005). They used the PCXMC program to evaluate the effective dose and image quality with different kV levels (90, 121 and 150 kV) and as a conclusion they recommend lower kV at a constant effective patient dose.

Most x-ray equipment has possibility to use different added filtration in x-ray tubes. The effect of added filtration has been evaluated by Hamer et al (2005) and they pointed out about 31% reduction in ESD with 0,3 mmAl Cu-filtration. Decrease in effective dose can be found also in effective dose with PCXMC program.

PCXMC evaluates also the risk to the patient due the x-ray examination. As well known the children are more sensitive to radiation than adults (Paile 2002, ICRP 2000). There are also some special groups among small children and young people, which are frequently x-rayed, e.g. a child with scoliosis, heart problems, vesico-uretal reflux etc. Also premature babies might be exposed tens or even hundred times during the first living months. The risk assessment may give more evidence for the dose optimisation. (Kettunen et al 2003, Kettunen 2004., Servomaa, Kettunen 2005.) The program reports the risk as a loss of life expectancy or as a cancer mortality. The program shows also the cancer mortality due to other risks and that gives perspective to risk assessment.

The new detectors offer a good possibility for dose reduction. It is still quite difficult in clinical practice, because the more radiation is used; the better is the image quality. (Correa et al 2008,Geijer et al 2009.) The staff get more motivation to dose and image quality optimisation when they notice the effect of the change to the patient's dose. The students reported a lot of possibilities to use PCXMC program to demonstrate how to optimise dose and what is the effect of each technical parameter or direction of the projection. It is amazing to find out how small changes can effect so much to the patient's dose.

Conclusions

The students have found out the benefits of PCXMC program in dose optimisation. It is easy to point out, what kind of changes have to be done and how much the changes decrease the dose. Also the meaning of critical organs clarifies. The age –dependence in risks due to ionising radiation can be demonstrated as Students analyse their findings and compare them to the results of scientific articles published during the last years.

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Radiation protection of the staff in operating theatres

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Abstract

Purpose: The universal features of the safety culture are individual awareness of the importance of safety, knowledge of competence, commitment, motivation, supervision and responsibility. In Finland nurses and medical doctors are allowed to use the c-arm in operating theatres and emergency rooms. The purpose of this study is to point the key factors affecting to the radiation protection and safe use of mobile c-arm in operating theatres and emergency.

Methods and materials: About 40 courses (1,5 ects) concerning the safe use of mobile c-arms have been given to nursing staff all over Finland. The course consist of 5 areas given in EU legislation: Fundamentals of Radiation Physics and Radiation Biology, Radiation Protection Provisions, Radiation Safety Measures at the Workplace and Procedures Involving Exposure to radiation. In demonstration all features of that c-arm are shown step by step. In the end of the course there is a written exam and more than 1300 answers have been analysed for this research.

Results: About 80% of the participants passed the test in first exam and only 1% needed third exam. Mostly participants are nurses but also some medical doctors and other staff. According to this study the most difficult areas are the basic concepts: radiation and it's features, effects of radiation at the molecular, cellular and tissue levels, deterministic and stochastic effects of radiation and their identification, radiation user's organization, dose motoring and categories A and B and what they mean in everyday work, controlled and supervised areas and monitoring of radiation exposure of workers. The operational radiation protection is quite well known.

Conclusion: During the courses there has been discussion about factors influencing the interpretation of fluoroscopic images procedures exposing children and pregnant women to radiation. The nursing staff is very willing to involve to good safety culture. More hands on training is needed to the nursing staff.

Radiation protection training in the Joint Research Centre of Ispra

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Abstract

The Joint Research Centre of Ispra, one of the research Sites belonging to the European Commission, Directorate General JRC, was created in the late '50s, in order to steer European research on nuclear industry. It hosts numerous nuclear facilities, some of which are maintained in operation, while others were shutdown in past years, namely: two research nuclear reactors, hot cells facilities, radiochemical laboratories, one Cyclotron (still in operation), facilities for studies on fissile material (in operation), and some facilities for the treatment and storage of liquid and solid waste (in operation). The JRC accounts for 21 nuclear licences, 14 Controlled Zones and 12 main Surveilled Zones, on its Ispra Site. The Radiation Protection Sector has developed, during the last years, a new approach for training in Radiation Protection: this includes an improvement and extension of the traditional classroom-based training to Exposed Workers, and:

1. the generation of new training modules on radiation and radiation risks, specifically addressed to non-exposed Personnel;
2. on-the-job training to newcomers and Radiation Protection Officers;
3. development and operation of mock-up facilities for specialized training (glove boxes, high contamination areas, etc.).

Evaluation of training effectiveness has also been modified and enhanced. Moreover, the need for a more comprehensive set of "education and training" actions in the field of Radiation Protection is becoming a priority, in Italy, due to the reduced availability of competent Technicians in the Radiation Protection market. The JRC-Ispra is currently studying the possibility to re-evaluate its training offer to the market, in order to further improve, according to its mission, the dissemination of technical and scientific knowledge in this field.

Radiation protection related teaching in Estonia and using of web-tools

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Abstract
This paper will provide a time-line overview of the progress on teaching of radiation protection in Estonia following the 1991 declaration of independence. We will address several important factors and the means which promoted the teaching, and give the overview of different tools used for that. Also it is our intention to share the experience in order to facilitate the learning process.

Introduction
The Republic of Estonia (area of 45 227 km², population of 1.34 million) is situated in North-East of Europe, on the eastern coast of Baltic Sea. Estonia regained its independence in 1991, which created a need for changes in many areas, including radiation protection. During the Soviet period there was research in the field of nuclear physics and nuclear-related courses were taught at the universities. However there were no courses in Estonian universities about radiation protection or nuclear safety.

After regaining the independence a sudden need appeared to build up the radiation protection system in the country. Also taking into account the heritage sites – nuclear submarine training centre and uranium mining and milling facility, there was need for educated people. The first radiation protection courses were taught at the University of Tartu. These courses were addressed mostly to the students of environmental physics and were based almost entirely on the environmental radiation. Soon the courses concerning the use of ionizing radiation in the medicine followed. During the last decade the scope of the courses taught at several universities has been widening, so that currently the topics covered are wider and the courses are addressed for the several specialities.

From late nineties the radiation protection course is taught also in the Türi College of University of Tartu, where most of the environmental specialists of Estonia are educated. Türi College is a relatively small institution, which actually has only one curriculum – Environmental Regulation and Planning. Taking also into account that Türi is a small town (population of about 6000) situated in the middle of Estonia, without any research facilities, national environmental institutions or big universities, qualified lecturers usually will be available only during the lecturing course for three to four days. However the college has good technical facilities, including the special web-teaching rooms for students with good internet connection. Considering this, and also the lack of teaching materials available in Estonian language, the main radiation protection lecturers at the Türi College decided after almost 10 years of teaching to develop the web-based teaching tool.

Material and methods

Developed web-based teaching course is basic radiation protection course, which is part of the required curriculum for Türi College students. However the same course is available also for other students of Tartu University on optional basis, if they care to travel to Türi. There are no special requirements to students in order to enroll, however some knowledge about mathematics is useful. The course aims to provide the knowledge of basic principles of radiation protection, and to give the ability to assess some easier cases including activity estimation or simple shielding calculations for radioactive sources. At the end the student should also have understanding about the sources of ionizing radiation, biological effects caused by ionizing radiation and principles of the nuclear energy. The course covers also terminology and units used in radiation protection.

The Blackboard Learning System (Blackboard Inc, 2010) was used in preparation of the web-based course. The Blackboard Learning System is a web-based server software platform, which features include course management, a customizable open architecture, and a scalable design that allows for integration with student information systems and authentication protocols. Its main purposes are to add online elements to courses traditionally delivered face-to-face and to develop completely online courses with few or no face-to-face meetings. The options available in the system include for example announcements, discussions, mail, calendar, learning modules, assessments, assignments, media library, etc (Fig. 1). Regarding the quite specific terminology used in radiation protection, that students may not be familiar with, one of the most useful application in Blackboard system is the Glossary. Course developer can insert new terminology into the Glossary and particular word in the teaching material will become a hyperlink, clicking on which results a new window opening with the particular explanation from the Glossary (Fig. 2 and 3).

Fig 1. Student view of a learning material about III generation nuclear reactors. On the left side is visible the list of Blackboard capabilities used in this course.

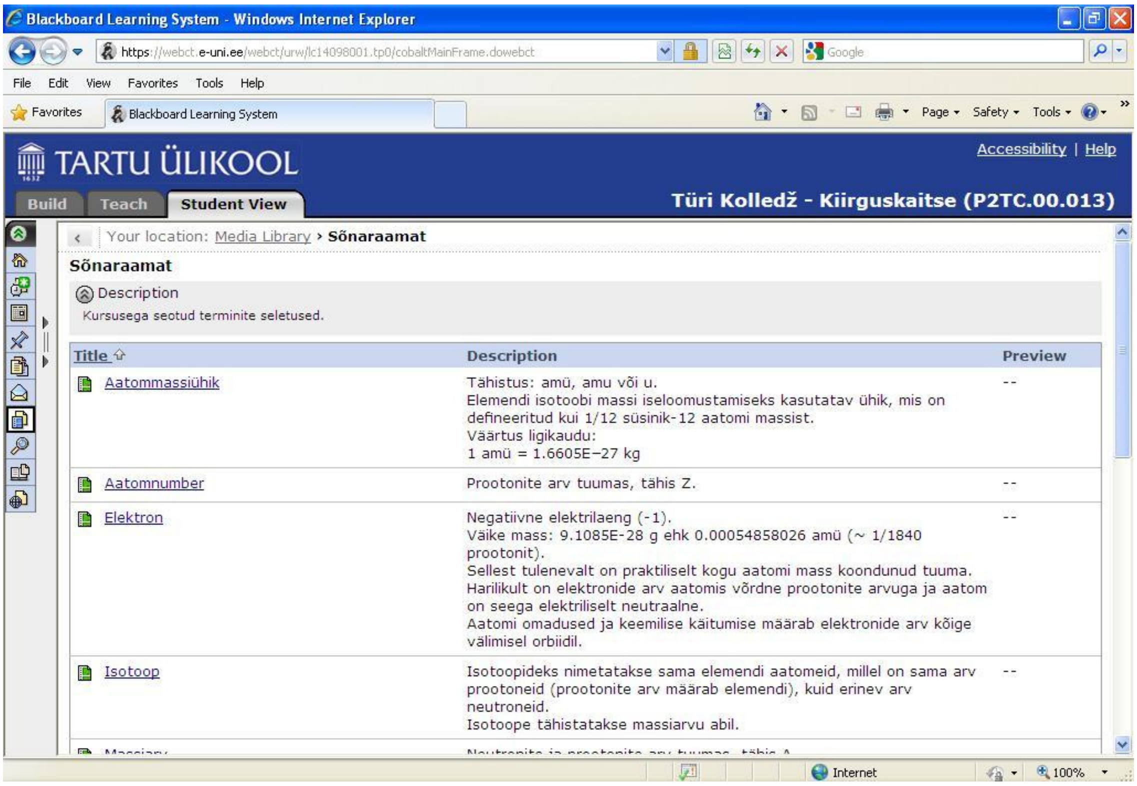

Fig 2. Example of a Glossary application.

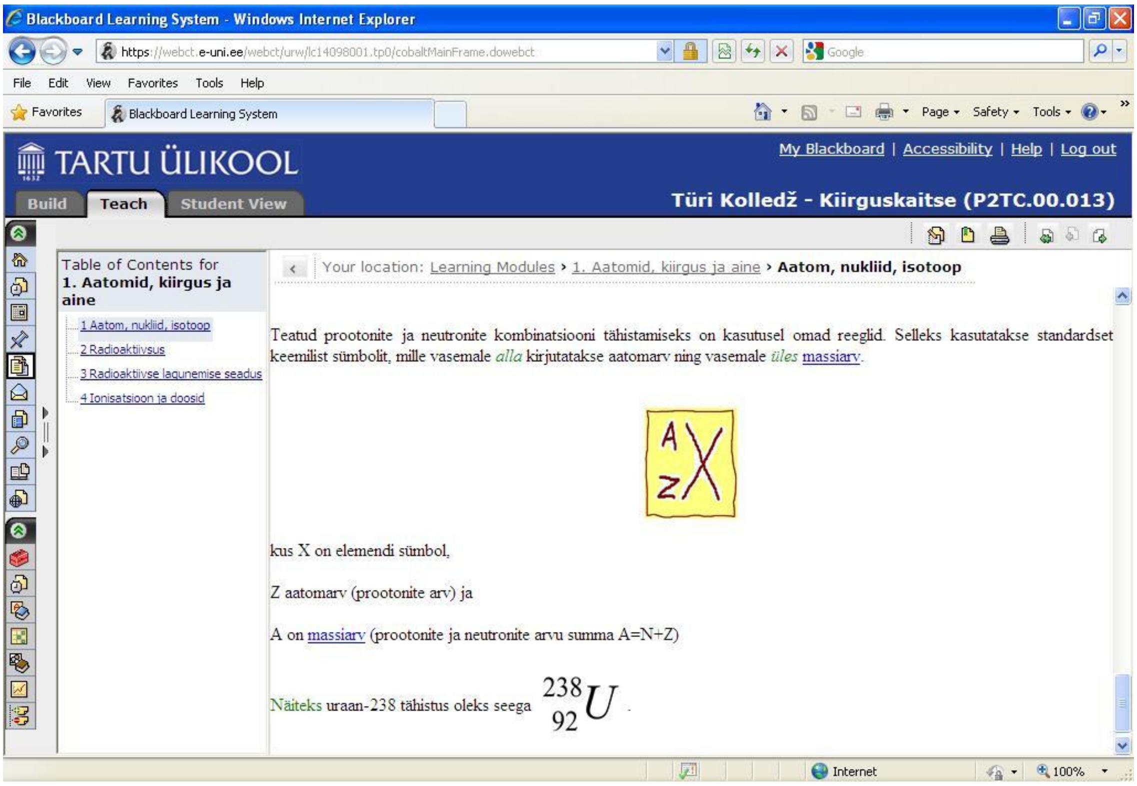

Fig 3. Example of teaching material. As you may see the word "*massiarv*" is automatically formated as a hyperlink, because it is defined in a Glossary seen on Fig. 2. On the left side of the screenshot is visible the Table of Contents for the first chapter of the course.

Results

The web-based teaching tool helps to facilitate better interaction between the lecturer and students while residing in different regions of Estonia, and also provide them learning environment with additional information. The preparation of current web-based course was started in spring 2009, it was finished and tested first time in autumn 2009. The prepared web-based basic radiation protection teaching course consisted of 6 learning modules:

1. Atoms, ionizing radiation and dosimetry

   Giving an overview of topics like: atom, nuclide, isotope, radioactivity, radioactive decay law, ionization and dosimetry.

2. Natural and man-made radiation sources

   Consisting of: environmental radioactivity, decay chains, radon, internal radiation, uses of radiation sources, medical radiation, industrial applications of ionizing radiation, results of nuclear weapons testing, accidents.

3. Biological effects caused by ionizing radiation

   Which includes: biological effects of ionizing radiation, four stages of the interaction between ionizing radiation and human cells, DNA and direct and indirect effects of ionizing radiation, deterministic and stochastic health effects, cancer development, hereditary effects, acute radiation syndrome.

4. Nuclear Energy

   Giving an overview of: nuclear energy and radioactive waste management, pressurised water reactors, boiling water reactors, III generation reactors, fuel cycle and waste, radioactive waste jurisdiction, releasing radioactive waste into the environment, liquid, solid and gaseous radioactive waste, used nuclear fuel, decommissioning.

5. System of Radiation Protection

   Covering the topics of: history of radiation protection, organizations, main principles of radiation protection, justification, optimization, limitation, national radiation protection system.

Different materials and experiences gathered were used in preparation of the course. One of the main supporting materials was booklet "Radiation, People and the Environment" (Lust *et al* 2006*,* Wrixon *et al* 2004). The book, which was produced by the IAEA in close co-operation with the UK National Radiological Protection Board, provides a broad overview on the subject of ionizing radiation, its effects and uses, as well as the measures in place to ensure it can be used safely. It also discusses the benefits and risks of practices that use such radiation in medicine, industry and energy production and considers some topical concerns about environmental pollution, waste management, emergencies and transportation safety. The book was translated and published in Estonia in 2006.

The course is usually held in the autumn semester. The course calendar will provide the exact dates for the assessments, assignments, lectures and exams. The teaching will be held as mix of the auditorial work and self-based work online, the course will start with introductory lecture. Each learning module provides explanation of the topic with a lot of illustrative material – photos and schemes. Considering that basic radiation protection course consists mostly of new information to read, different colours for text were used to make it easier for the student. Additionally each learning module has assessment and assignments. The assessments (Fig. 4 and Fig.5) were usually organized in self-testing form and consist of questions or calculation exercises. In order to pass the course the student has to do all the assessments. Most of the supporting materials were provided online. The final step will be passing the exam, which can be done online as well. This has been also quite an important feature in order to provide the security for the results. After collecting the information from the learning modules provided in the course, the result was booklet consisting of 111 pages.

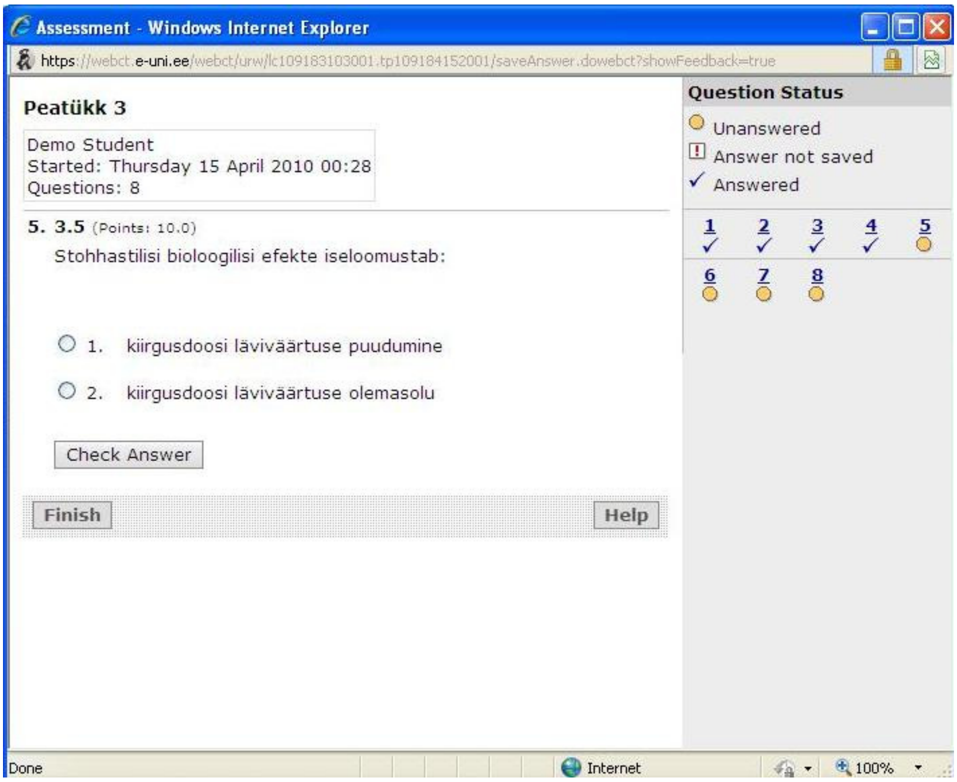

Fig 4. Example of a multiple-choice type self-assessment test. Questions are taken randomly from the question-bank filled by the teacher.

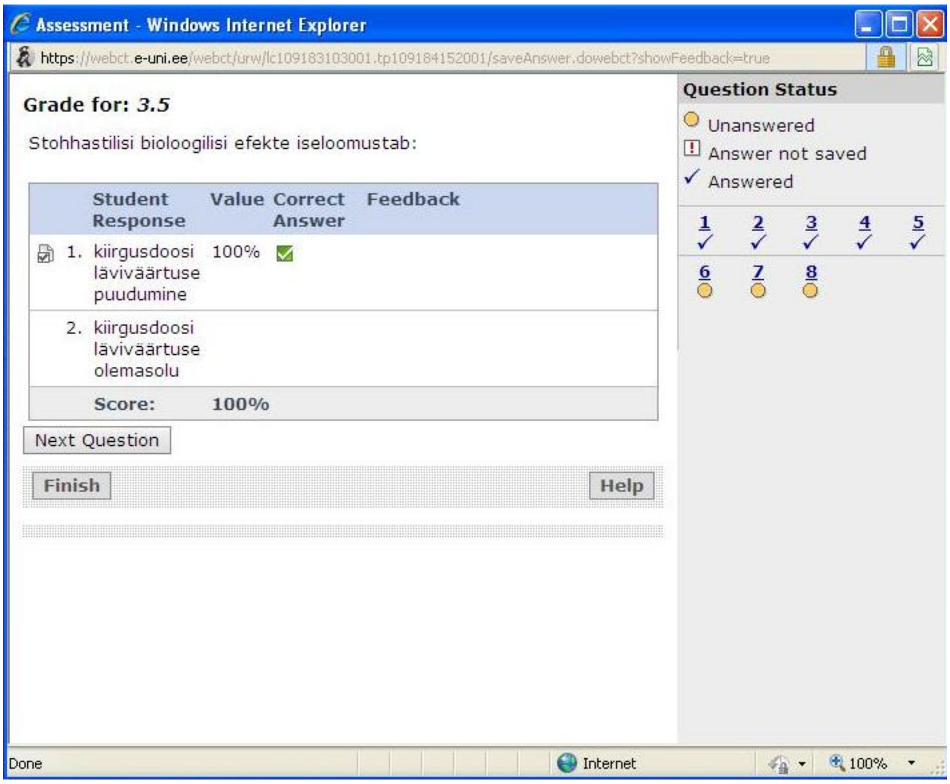

Fig 5. Instant feedback for the student that the answer to the 5th question on the self-assessment test was correct.

Discussion

Around year 2000 started the first courses for the users of radiation sources. Most of the first courses were organized through international co-operation projects, but during the years there has been teaching of the local trainers. This has lead to the situation, where basic radiation protection education for the operators can be provided by local experts. Estonian Radiation Protection Centre (since February 2010 known as Radiation Safety Department of Environmental Board) has facilitated translation and publication of several radiation protection related publications and has used its web-pages as a tool for providing the information. Now additionally the information gathered through preparation of the course can be used and published for wider public as well.

Other future applications and expansions of such a course may also relate more closely to nuclear energy. Taking into account the vision approved by the Parliament in June 2009, which lists the nuclear development as one of the possible future scenarios, there is a great need for education in nuclear safety. University of Tartu and Tallinn Technical University have started the preparation of nuclear safety and nuclear energy master course. The first students are supposed to be admitted in the autumn of 2011. The prepared course can be used in the curriculum of these master courses as well.

Conclusions

Estonia started building up the radiation protection infrastructure a bit more than 10 years ago. There was no clear governmental mandate and no accompanying support training system to start with. During the last decade the remarkable steps have been made in providing the information related to radiation protection to the public and also in preparation of the specialists at the universities. There is still much work to do, however using the technical tools available, the current human resources can be used much more wisely. The web-based course prepared last year passed the first test successfully and it also provides a good platform, which can be used in several occasions.

Acknowledgement

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Radiation protection education and training activities at the Belgian Nuclear Research Centre SCK•CEN

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Abstract
The scientific world of radiological protection (RP) is in constant motion, triggered by new research as well as by developments and events in the daily industrial and medical sectors. In addition, national and international regulatory policies try to streamline and guide daily practice along procedures that guarantee the protection of workers and public, and that at the same time also ensure optimization of all peaceful uses of applications of radioactivity. Harmonization and coordination are in this sense of the utmost importance, not only 'on the work floor', but also with regard to education and training (E&T). Within this spirit, the international school for Radiological Protection (isRP) of the Belgian Nuclear Research Centre SCK•CEN participates to several E&T activities: SCK•CEN experts lecture several courses within existing academic programs, and give guidance to Master and PhD students in the framework of their thesis. We also organize courses on a wide variety of RP topics for professionals of the nuclear and medical sector and - in parallel - we aim to play a role in national and international policy through active participation in several European networks.

Introduction
The Belgian Nuclear Research Centre SCK•CEN was created in 1952 in order to give the Belgian academic and industrial world access to the worldwide development of nuclear energy. It is a Foundation of Public Utility, with a legal status according to private law, under the tutorial of the Belgian Federal Minister in charge of energy. Since 1991, the statutory mission gives priority to research on issues of societal concern such as safety of nuclear installations, radiation protection, safe treatment and disposal of radioactive waste, fight against uncontrolled proliferation of fissile materials and fight against terrorism. The Centre also develops, gathers and disseminates the necessary knowledge through education and communication, and provides all services asked for in the nuclear domain (by the medical sector, the nuclear industry and the government). Today, about 630 employees advance the peaceful industrial and medical applications of nuclear energy, and realize a turn-over of about 85 M EURO.

Thanks to its thorough experience in the field of peaceful applications of nuclear science and technology, the Belgian Nuclear Research Centre SCK•CEN has garnered a reputation as an outstanding centre of not only research, but also education and training. SCK•CEN is an important partner for training projects in Belgium (to the nuclear sector, the medical and non-nuclear sector), as well as at the international level (IAEA, EC, …). The Centre's know-how and infrastructure are available for education and training purposes. One of the research topics that is strongly developed at SCK•CEN's Institute for Environment, Health and Safety, is radiation protection. From years of experience and recent knowledge that results from the latest innovative research, an extensive range of course modules has grown. Our courses are directed to the nuclear industry, the medical and the non-nuclear industry, national and international policy organizations, the academic world and the general public.

Academic collaborations

SCK•CEN experts are involved as lecturer in several academic programs, at Belgian and foreign universities and technical universities. They also deliver contributions to courses set up on specific occasions such as the courses given in the framework of ENEN, CHERNE, WNU, etc.

Radiation Protection Expert

Together with XIOS Hogeschool Limburg, ISIB and IRE, SCK•CEN organizes the Radiation Protection Expert course, given in Dutch and in French. This course broadens the scientific and technological basic knowledge of radiological and nuclear techniques with special attention for practical radiation protection. It is aimed at professionals working with ionizing radiation, in all sectors. The program is in line with the legal requirements of the Belgian Royal Decree of July 20 2001, mentioning the prerequisites for accreditation as Radiation Protection Expert by the Belgian Federal Agency for Nuclear Control. SCK•CEN collaborators contribute to the Dutch course, for 96 of the total 120 hours.

Guidance of PhD students

In a conscious desire to increase its pool of highly specialized young researchers and to tighten the links with the universities, SCK•CEN embarked in 1992 on a program to support PhD candidates and post-doctoral researchers. Since 1992, about 100 students started a PhD in collaboration with SCK•CEN and more than 50 post-docs joined our Centre. Today, about 35 young scientific researchers perform their work in research fields that reflect the priority programs and R&D topics of SCK•CEN, about half of them are working within the field of radiation protection.
Next to this, SCK•CEN experts also work with Master students, and even high school students (sixth year) have the opportunity to use our laboratories and other infrastructures in order to perform scientific experiments to support there thesis work.

Training courses for professionals

Training courses of the international school for Radiological Protection of SCK•CEN cover a very broad offer. Different modules are foreseen as a "standard", but in principle all our courses are tailored to the specific needs, field of operation, and level of the trainees.

Basic modules

The course series *Background and Basic Knowledge* collects general and more specific technical courses on radiological protection. The series consists of seven modules and provides the theoretical and practical knowledge required for implementing technical aspects of radiological protection in a medical or industrial working environment, both in the daily practice and in the management in the long term:

- Basic principles of nuclear physics
- Interaction of radiation with matter
- Radiation and dose measurements
- Biological effects of ionizing radiation
- Gamma spectrometry
- Standards and legislation
- ALARA and safety culture

Referring to questions such as *'what is radioactivity?'*, *'how can we use it?'* and *'how can we protect ourselves against it?'*, the series starts with an introduction to nuclear physics that is then linked to a practice-oriented part on radiation and dose measurements and spectrometry. The module on biological effects of ionizing radiation presents an understanding of the effects of high and low level doses of ionizing radiation on the human body. The series is completed with a state-of-the-art overview of standards and legislation and a rationale on ALARA and safety culture, including a demo session with virtual dose assessment software tools.

When composing a custom-made program, the course could start from this basic series, but some modules may be omitted and other more specialized modules can be added upon request.

Specialized modules

The *nuclear and radiological expertise modules* are follow-up modules that fit in directly with the basic course series, but provided sufficient foreknowledge, they may be taken separately. The series addresses technical practice-oriented issues with a link to radiological protection and relies fully on the nuclear expertise of SCK•CEN. The series include amongst others:

- Transport of radioactive materials
- Radon and increased natural radioactivity
- Ethical aspects of the radiological risk
- Management of radioactive waste
- Internal dosimetry assessment from bioassay measurements
- Quality assurance in nuclear safety
- Decommissioning and dismantling techniques
- Good safety practices in controlled areas (practical training)
- Organization of emergency planning
- Misuse of radioactive materials: prevention and response (safeguards)
- Radiochemistry

Practical exercises and visits

In the course programs, lectures and practical training sessions can be alternated with visits to relevant laboratories and installations of the SCK•CEN. These technical visits enable to enrich and illustrate the participants' acquired knowledge with the practice of 'real-life' situations, as well with regard to safety culture in controlled areas, as the techniques and know-how of the applications of radioactivity as such. SCK•CEN installations and laboratories that can be visited include:

- BR1, the 'natural uranium – graphite – air' type research reactor;
- BR2, the high neutron flux material test reactor;
- BR3, the first prototype Pressurized Water Reactor in Europe, and the first now in dismantling phase;
- The HADES underground laboratory for waste disposal research;
- The radioactive decontamination wing of the medical services;
- The emergency planning and follow-up room;
- The whole body counter laboratory (antropogammametry);
- The radiobiology and microbiology laboratories;
- The radioecology laboratories;
- The nuclear calibration services.

Practical organization

The international school for Radiological Protection is based on the site of the Belgian Nuclear Research Centre SCK•CEN in Mol, Belgium. The regions of the municipality of Mol and the adjacent municipality of Dessel have a historical concentration of nuclear activities of more than half a century, hosting research, nuclear fuel fabrication and waste treatment and storage.

All installations and labs that are taken up in the list of possible technical visits are located on the SCK•CEN site. On request, the course lectures can also be organized in the SCK•CEN offices in Brussels, or at the venue of the trainees, eventually completed with technical visits to laboratories and installations on the technical domain of SCK•CEN in Mol.

Lecturers

Among the isRP lecturers are technicians, physicists, biologists, medical doctors, engineers and social scientists who all bring insights and ideas from their specific background into the course programs. As SCK•CEN staff members, they have a solid knowledge and experience in their field, and can thus directly transfer their theoretical knowledge and practical experience into the various courses.

Transdisciplinary aspects

Understanding the benefits and risks of radioactivity requires technical insight and training, but also an insight in the context and a sense for the social and philosophical aspects of the situation. isRP concentrates on how to integrate this transdisciplinary approach in education and training programs.

In cooperation with the SCK•CEN PISA group (program of Integration of Social Aspects into Nuclear Research), isRP has build up experience with the theory and practice of participation and involvement in technology assessment and has set up a course module on ethical aspects in radiation protection. On various occasions, the two groups organize round tables, workshops and focus groups with schools and local communities, and this on topics such as medical applications of radioactivity, (nuclear) energy policy and radioactive waste management.

Policy support

The implementation of a coherent approach to education and training becomes crucial in a world of dynamic markets and increasing workers' mobility. Through networking and participation in international programs, SCK•CEN contributes to a better harmonization of education, training practice and skills recognition on a national and international level.

Covering electricity production, medicine and several activities within the non-nuclear sector the spectrum of applications of ionizing radiation is very wide. Although working with a variety of responsibilities and specific professional aims, practitioners have a triple common need:

- basic education and training providing the required level of understanding of artificial and natural radiation,
- a standard for the recognition of skills and experience,
- an opportunity to fine-tune and test acquired knowledge on a regular basis.

From an executive perspective, education and training are undoubtedly the two basic pillars of any policy regarding safety in the workplace. The radiological protection rationale that serves as the basis for this policy is the same all over the world, going beyond cultural differences and disciplinary applications. In this sense, the implementation of a coherent approach to education and training in radiological protection becomes crucial in a world of dynamic markets and increasing workers' mobility.

Through networking and participation in international programs, SCK•CEN aims to contribute to a better harmonization of training practice and skills recognition on a national and international level. In this frame, specific issues of interest to SCK•CEN in general and the isRP in particular are the standard requirements for course programs and educational material, the development of transdisciplinary training programs, e-learning and distance learning, the link between radiation safety and conventional safety, the organization of experience feedback, international exchange of knowledge and experience and the sharing of lecturers, training facilities and educational source material. These are the topics covered in European networks such as EUTERP (European Training and Education in Radiation Protection Platform – www.euterp.eu) and ENETRAP (European Network for Education and Training in Radiation Protection – www.sckcen.be/enetrap2), in which SCK•CEN is playing a prominent role.

Nuclear Training Centre experience in radiation protection culture

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Abstract

This paper aims to present Nuclear Training Centre (CPSDN) Romanian experience in education and training in order to achieve the required competences in radiation protection with a view to future recognition within the EU countries.

Carrying further education and training activity started with almost 50 years ago by the Institute of Atomic Physics and University of Bucharest-Faculty of Mathematics and Physics, CPSDN organized over 750 programs and trained more then 18500 participants.

For each of these programs, CPSDN take into account the following objectives:

- provide adequate training for workers, for radiation safety officers and specialists
- comply with the authorities regulatory requirements
- develop syllabus for general education, specialized training courses and upgrading programs in compliance with CE RP 116 and IAEA recommendations for the groups of professionals that have been identified
- contribute to the national radiation protection infrastructure
- establish the necessary mechanism and to provide appropriate administrative support, qualified and experienced trainers, practical demonstrations and exercises
- cooperate with national organizations developing activities in the field, for example the Romanian Radiation Protection Society, University of Physics, research institutes, main stakeholders.

Being concern by the continuously improvement of its services, CPSDN applied for TUV-CERT certification for the quality management system according to ISO 9001/2000, in 2007.

Introduction

Placed between the Faculty of Physics and the National Institutes of Research and Development for Physics in Magurele Platform, Nuclear Training Centre is developing post secondary school and post university training of the personnel involved in the nuclear field and/or in related areas, contributing, through its activity, to the human resources development and to the implementation of research results of "Horia Hulubei" National Institute of Physics and Nuclear Engineering and the other institutes from Magurele Platform.

Established through the Decision of Ministers Council No. 148/1970, Nuclear Training Centre (CPSDN) assumed the organization of the post university specialization programme entitled "Courses on the Utilization of Radioactive Isotopes" (CUIR), initiated since 1956 by the Institute for Atomic Physics (IFA) in cooperation with the Faculty of Mathematics and Physics of Bucharest University.

Organized as legal unit under the authority of the State Committee for Nuclear Energy (CSEN), CPSDN has yearly developed training methods suitable for the tackled applications, by categories of degrees of responsibilities in radiological safety assurance, such as: training courses, qualification and post qualification courses, post secondary school and post university specialization courses, endorsement and accreditation courses, managers dedicated courses. Training programmes curricula were permanently adjusted both to the technical upgrading of the envisaged fields and to the growing regulatory requirements.

As an illustrative example is the programme for the non destructive examination area entitled "Industrial Nondestructive Testing", which was organized during 1973 – 1989; it has been upgraded in methods programmes, starting with 1980.

In the medical field the initial "Radioisotopes and Radiation Applications in Medicine" programme has been upgraded and several programmes were developed, such as: "Radiation Protection in Radio Diagnostic Practice", "Protection of Patient and Operators from Nuclear Medicine", "Radiation Protection in Radio Therapy Practice", etc., dedicated to physicians and/or nurses.

Required themes approach envisaged not only participants' information but also their formation as specialists in the related practice by the development of a correct understanding of the physical basic aspects and an individual attitude for the field promotion.

CPSDN constantly benefited of the scientific potential of Physics Institutes from Magurele Platform cooperating with specialists within IFA as lecturers and using the laboratories for the practical applications.

Since 1996 CPSDN has been developed its activities as a Department of "Horia Hulubei" National Institute of Physics and Nuclear Engineering (IFIN – HH).

This kind of organization facilitated CPSDN's access to the IFIN – HH laboratories and specialists.

During 1970 – 2009, CPSDN contribution for training and specialization of personnel involved in practices with radiation sources and advanced physics techniques could be resumed as it follows:

~ 780 training programmes

~ 19.000 graduates.

Main training activities

Taking as reference the technical international bodies recommendations such as ICRP publications, IAEA Basic Safety Standards, European Directives, at the national level the Law No. 111/1996 on the safe deployment, regulation, authorization and control of nuclear activities, republished, was adopted. National Commission for Nuclear

Activities Control, the Romanian regulatory and control body in the nuclear field, issued the Fundamental Norms for Radiological Safety (NSR 01) and other norms requiring the specialization of the staff at the level of operators and mainly at the level of coordinators, with responsibilities in the assurance of radiological protection for population, environment, professionals, patients in medical applications. As the radiological protection aspects were the most significant requirements, CPSDN organized different kind of training envisaging the utilization radiation sources under radiological safety conditions in a specific practice. During the last period the following training programmes are constantly developed:

1. Radiation Protection on the Utilization of Measurement Systems with Radiation Sources
   – Participants are secondary school and university graduates involved in the management or operation of gauges with radiation sources or X fluorescent analyzers, electronic microscope, etc.
   – Schedule: 40 hours, Licensed by CNCAN for the level 1 (licence at operators level)
2. Radiation Protection on the Utilization of Radiological Facilities for Packages Control
   – Participants are secondary school and university graduates operating facilities for packages control (post office, customs, airports, army, etc.)
   – Schedule: 30 hours, Licensed by CNCAN for level 1
3. Radiological Safety in Uranium and Thorium Mining and Milling
   – Participants are university graduates developing activities in the field of uranium and thorium ores mining, processing, transport
   – Schedule: 90 hours. Licensed by CNCAN for level 2 (licence for persons with responsibilities in the controlled units).
4. Radiation Protection in Radio Diagnostic Practice
   – Schedule: 30 hours (Radiologist Physicians)/64 hours (Radiologists nurses)
   – Licensed by CNCAN for level 2
5. Radiation Protection of Personnel and Patients in Nuclear Medicine
   – Participants are university graduates – Physicians, Physicists, Chemists
   – Schedule: 80 hours, Licensed by CNCAN for level 2.
6. Radiological Safety on the Utilization of Radiation Unsealed Sources
   – Participants are university graduates
   – Schedule: 80 hours. Licensed by CNCAN for level 2.
7. Radiological Safety on the Utilization of Radiation Sealed Sources
   – Participants are secondary school and university graduates
   – Schedule: 70 hours. Licensed by CNCAN for level 2.
8. Applications of Radio Isotopes and Nuclear Radiation Sources
   – Post university specialization course for coordinators in laboratories with radiation sources from all fields of activity: mining, industry, medicine, education, research, army.
   – Schedule: 180 hours. Licensed by CNCAN for level 2, all domains
9. Radiological Safety on the Utilization of Sealed Sources (SI)/Unsealed Sources (SD)/Radiation Generators (GR). Knowledge Upgrading
   – Schedule: 30/40 hours. Licensed by CNCAN for level 2.

CPSDN is developing focused programmes for various domains of the nuclear field and, consequently, it organizes, on requirement, training programmes on a certain application, according to the related requirements and competence level.

In this respect, we could mention:

– training of personnel involved in the nuclear power programme, developing activities in the design, equipment and components supply units for Unit 1 of Cernavoda Nuclear Power Plant (NPP)
– radioecology specialization programmes organized for the Ministry of the Environmental Protection
– training programmes for the personnel involved in the activities related to VVR-S Research Reactor decommissioning
– training programmes for radioactive materials carriers
– training programmes on radiological safety in radioactive logging.

Results of the last years

Synthesis of activities developed during the last years (2004 – 2009) represents a relevant example of those above mentioned.

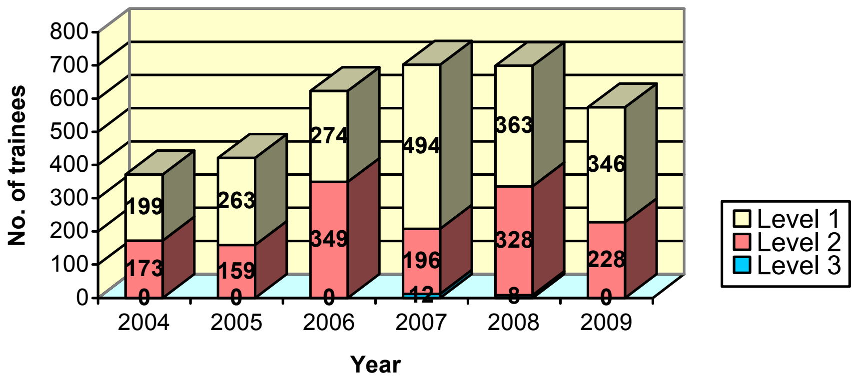

Fig. 1. Training activities during 2004 - 2009 and authorization levels.

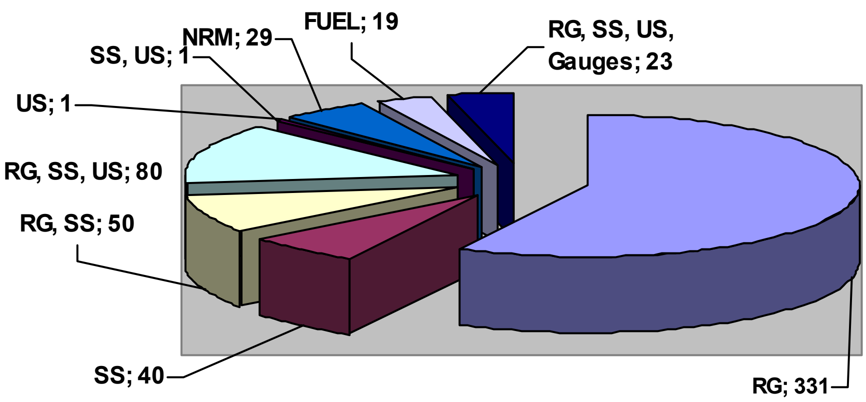

Fig. 2. Trainees and training fields in 2009.

SS – Seales Sources
RG – Radiation Generators
US – Unsealed Sources
NRM – Nuclear Raw Materials
Comparing the results, one may conclude the following:

(i) a spectacular increase in the number of trained persons in compliance with technologival development and new regulatory requirements
(ii) the number of programmes increased and the weight of various programmes reflects a RG and SS higher double specialization.

Future development of the Centre

CPSDN future objectives are in compliance with the CPSDN quality policy major objective, respectively:

Continuous improvement of training services quality by diversifying the training offers and improving services performances.

In this respect, CPSDN activity is focused on the following objectives:

- maintaining the Quality Management System certified by TÜV HESSEN
- diversification of training dedicated to radiation sources practices in the medical applications and other fields
- updating of web page for the dissemination of Centre's activities and improvement of communication with past and future beneficiaries (forum)
- development of web page for improving public information and customer relationship management
- e-learning implementation for knowledge upgrading of licensed workers
- new electronic/carbon copy training materials on radiation protection (manuals, booklets, etc.)
- development of training programmes dedicated to experts in radiation protection according to the national and international requirements.

Conclusions

- CPSDN represents a long history of training in the field of radiation protection.
- Training programmes cover a wide area according to the various requirements and applicants' needs.
- CPSDN resources benefits of continuous upgrading in order to comply with the new challenges in the field.
- CPSDN envisages being much more involved in the European programmes and is ready to develop national and European partnerships to respond to actual training needs.

The study of radon to understand the radioactivity and to know the environment

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Abstract

The work describes the collaboration among many Italian research groups that pursue since seven years the objectives to make students of secondary schools aware of the problematic of the radioactivity and of the ionising radiations, introducing they, when possible, in real research activities. For reach this goal, many approaches where adopted around the main path starting with the measurement of indoor radon concentrations and continuing considering most complex themes. The project obtained very good results from many points of view, and each year many schools that carried out their activity want to insert in the experience new students and news schools joined to the project. This implied a continuous increase of the students that approach themselves these problems and, through this contact, the world of the research and of the university;

Meetings and conferences have been regularly organized, in which the students communicate to a wide public their experience; and often students, teacher or researcher are invited in several contexts to describe the developed activity. Year by year the number of proven techniques increased and from the indoor measurements carried out with passive techniques, it has been passed to the dose measurements, to the analysis of materials, to the comparison of more techniques, to the use of active instrumentation which is often home made. Starting from 2008, the study of the artificial radioactivity in environmental samples, originated by radioactive fallout, began. These results were possible thanks to the 'commitment of research groups involved. A further increase of the schools involved would entail too great an effort to sustain. To avoid that, are being prepared various kinds of materials that can help schools continue to use this useful method of investigation of science topics, usually treated just in schools, but by shifting the focus of the operation on them.

Introduction

The project "ENVIRAD-SPLASH" collects the activities carried out in the within of the National Institute of Nuclear Physics (INFN) in the field of the spread in the Italian schools of the culture of the radioactivity and involves seven groups of six Italian regions working in the field of the Environmental Radioactivity [Esposito et al., 2005]. Each group has chosen their own strategy to reach this common objective in full autonomy, but privileging always the direct and practical approach of the students to the studied topics [Balzano et al., 2006]. This has been realized of time in time with: their direct participation in the development of simple instruments, in the running of experiments finalized to the measurement of physical and/or radiometric parameters, in the observation and measurement of environmental parameters, in the participation in first person to true research activities. The umbrella of the INFN has allowed to create a network that has optimized the effectiveness of these experiences, creating in this framework easy opportunities of comparison and exchange of experiences between the investigators, the teacher and the students, with positive feedback on the result that such activity can produce.

The inclusion of the students in these activities has produced a very good success in the scholastic community, as it is testified by the great participation and by the continuous demand of new entry. Beyond these good results in the educational field, a secondary but equally important objective reached has been the knowledge of the exposure of the students to radon and its decay products, which, for the region of the South Italy, has been estimated for the first time topics [Venoso et al., 2009].

Material and methods

The general topic proposed at the beginning to the students has been that one of the natural radioactivity, in particular the program consisted in the measurement of the concentration of radon in the scholastic buildings, using passive detector techniques. This continues as the main activity in many cases, also because it answers to a requirement of monitoring and control of sensitive atmospheres, control that in many regions comes completely disregarded. Later on, with the acquisition of experience by the students, the interest, and therefore the activity fields, included also other objectives, like the carrying out of the same measurements in buildings outside of the participating schools, the measurement of the activity concentration in minerals and soils, the measurement of the dose, the practice with more sophisticated instrumentation., the evaluation of the risk depending on the exposition to the ionizing radiations.

An other topic, previewed since the beginning, has been the monitoring of the radon concentrations in the ground, using a system set to point in the frame of the experiment [Roca et al., 2006]. This experience, which started in the schools, had important developments and allowed to begin a research path that will be carried out in collaboration within the Earth Science community.

In the framework of the project, they have been developed also some kits dedicated to specific applications, like the treatment in the schools of passive radon detectors, the realization of simple alpha and gamma-ray spectrometers, built by the students, the measurement of radiometric parameters, like the lifetime of a radionuclide, or the efficiency of a detector All these activities, developed following suitable educational pathways, contributed to approach the students to arguments that the ordinary scholastic programs generally neglect and that instead they touch very actual topics. Simple numerical simulations realized using the ordinary electronic sheets have supplied a useful support to the understanding of basic concepts (comparison between behavior of the single radionuclide and that one of the relative species, analogies between various phenomena (radioactive decay and interaction of the electromagnetic radiations with the matter), difference between the interaction of directly and not directly ionizing radiations,…) or the estimate of experimental parameters, like the energy resolution of a detector, as well.

Results

In the first place it goes emphasized the dimension of the impact that the experiment had on the world of the school. A row estimate allows to count just about 120 schools for a total of approximately 2500 students directly engaged in the activities. This number can be increased of at least a factor 3-4 if it is believed next to the number of not directly engaged students but that they have been "perceived" from the job of the companions and the teachers. Also in the familiar ambit the experience of the boys had an echo that has brought back terms, data and problematic linked with the radioactivity. Probably, therefore, it is not exaggerated to say that 10000 persons, thanks to this plan, have been encouraged to make reflections more or less complex, on various aspects of the radioactivity This is a argument a lot neglected as well as in the scholastic ambit, how much in that one of the information, where it only bounces in case of accidents and comes systematically and exclusively associated to disaster and risk contexts.

Since the treatment of the passive detectors has been made following well known and correctly applied techniques, the radon concentrations measured with one year long exposures can be considered absolutely correct (on the scientific side) results. This finding is particularly important because, as it as been yet said, for South Italy, these were the first available data relative to this kind of exposure.

An mainingfull aspect of a scientific activity is the communication of the results. To aid the students in this part of their work, each year a series of annual meetings has been carried everywhere, in which the students presented very effectively the job carried out in front of hundred of their colleagues, with the participation of academic authorities, personality of the INFN, experts of the national agencies that take care of the measurement and the effects of radiations (ENEA, APAT, ISS), managers of local agencies and the participation of representatives of the press.

Result obtained within this experience have been shown to national and international conferences. Many result have been produced also in other fields, and their have consisted in doctorate and bachelor thesis, participation to various initiatives, like the International Year of the Physics, the activation of courses in the curricula of the School of Specialization to the Instruction, contacts between different thematic areas, contacts with the local agencies, that often have been transformed in full collaborations and also have some times carried financial contributions in the project.

Discussion

A careful reflection on the developed activity has allowed to evidence the following points

The plan seems to be deep-rooted in the zones in which has been lead. The schools find its follow-up natural, year after year, and therefore they encourage the participation of new students in substitution of those which complete their cycle of studies

New schools that have indirectly known the project ask to participate to it, demonstrating to appreciate the contribution that the character naturally to multidisciplinary of the dealt arguments supplies to the continuous attempt to l ink the ordinary scholastic issues, than generally are considered from the students like not communicating sections. The degree of agreement and assimilation of the new topics by the students has been also studied proposing to they some questionnaires whose analysis has supplied information useful to correct the carried out action

In these years of collaboration with the schools, they have been tried and predisposed various study patterns, informative material, many experimental projects, laboratory kit directly entrusted to the students and their teachers, and a lot of other material that has been experimented on the field and that opportunely selected and organized it could continue to produce positive results in the future.

Also the university and in general the world of the research draws advantage from an activity that serves to establish with their basin of user a stable tie, deriving from a engagement of articulated common job on two years. For this community, moreover, also good scientific data come from the work with the schools.

All these considerations have convinced to bring at the project some modifications suggested from the experience, from the availability of schools and teachers who already have lived this experience, from this historical moment, that sees reopen the debate on the nuclear energy, and with the aim to place the bases of a stable activity to put on hand of the schools also to outside of the logic of a classic experiment, which usually is strictly limited in the time. In short, a new proposal will be soon made, based on few strategic and methodological points. On the strategic side, more attention will be dedicated to organize data produced from the students campaign, extending the look also to the search in the environment of radionuclides of artificial origin and proposing in the more favourable situation (motivated teaching expert and interested classes) more complex problematic, deriving more or less directly from the job already carried out;

On the methodological side, a critical analysis of the applied methods and of the available materials will be accomplished, in order to predispose a catalogue of arguments, projects, suggestions, instruments, didactic materials, verification test, useful to aid the work of the students groups interested to start or to continue the work in the project. These instruments will be all thought for two levels: the first useful to the activity of the schools that for the first time enter in the plan, oriented to a basic formation in the field of the radioactivity; the second, made of more specific arguments, destined to students already expert (as an example students of the fifth year with already a pair of years of experience) and followed from the more able and interested teachers.

A great attention will be turned at the formation of the teachers. In fact, indispensable circumstance so that the proposed piano runs and that the offered catalogue can be useful for the students, is that the teachers are capable to ménage the activity and the instruments that are included into it. To allow the developing of these points, the redaction of an adequate and detailed program is underway.

If this result will be reached, two objectives strictly linked will be reached together. Te set to point of a method to introduce students to the study of then radioactivity and to its practice, and the possibility to obtain in this framework results that, from a scientific point of view are fully valid.

In this way, the system constituted by the researcher community and the involved schools, will produce both didactical and scientific results. This method, which produced so useful results in the field of the evaluation of the radon daughters exposure, could be expanded to study of other parameters interesting for a better knowledge of the environmental. The topics which can be studied in this way are in fact many. The first one could be the search in environmental samples of artificial raduonuclides. The presence in the environment of small ^{137}Cs concentration is well known and easy to verify and quantify. Its systematic observation could be an useful "zero point" for the environmental monitoring in view of the possible re-starting, in Italy, of the activities linked to nuclear energy production.

Conclusions

The project Envirad-Splash pursues the objective of carry out in the secondary schools an experimental activity in the field of the radioactivity, in which the students have an direct and intense implication. Many good results have been obtained up to now, both in the educational and scientific field. This activity has been organized in the framework of the activities of the National Institute of Nuclear Physics and it has been designed as an normal experiment, with a project phase, a run phase and a conclusion. The good reached results have suggested to go over this schema and the experiment has enjoyed more than one extension. To consent to other schools to have such opportunity, a method which could allow to develop this activity also out of a normal experiment is under study.

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Rn-222 concentration measurements at Italian schools: a way to educate, train and disseminate radiation protection culture among young students

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Abstract

In Italy the "nuclear issue" was for a long time a taboo. A lack of information will lead to unwarranted fears, which will distort the risks we take in everyday life. In other words the subjective perception (sensation) of the risk doesn't correspond very often to the objective and real risk of human activity. In particular, our perception of radioactivity is often misleading because of the lack of accurate information. A way to approach this theme to make the public more trusting of nuclear issues is to discuss radioactivity and ionizing radiation starting from young students. An experimental activity that involves secondary school students has been developed. The approach is to have students engaged in activities that will allow them to understand how natural radioactivity is a part of our everyday environment.

On this basis started a project that gave students and teachers of the Italian secondary school system the opportunity to discuss and to experiment with nuclear related experiences. The students were provided basic but correct information and with the added benefit of being able to experiment. The core idea is that: a) to provide the students a furnished laboratory at their school so that they can measure the natural component of the radioactivity that surround us. In this exercise the measurement of the ^{222}Rn concentration is particularly well suited b) to show the different types of radiations including ionizing radiations and how they each relate to the other; c) to demonstrate how easily ionizing radiations can be measured; d) and to prove the fun a student can derive from discovery and detection of ionizing radiation in the environment.

One other interesting outcome has been that the measurements have been made in accordance to Italian radiation protection law. Therefore, the data collected could be used to determine the radon concentration mapping of the school buildings.

RadiaX – Radiac simulation for first responders

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Abstract

As a complement to the training of first responders in their preparedness for accidents and incidents involving radiation, a radiac simulation, called RadiaX, was developed.

RadiaX has a threefold purpose; to teach (i) the handling of specific instruments, (ii) the proper procedures in missions and (iii) basic principles in radiation physics and radiation protection. The simulation is developed as a modification of Half-Life 2, a famous computer game.

Introduction

Swedish preparedness for accidents and incidents involving radiation includes equipping and training first responders with radiation indication instruments. The intensimeter SRV-2000 (Rados Technology Oy, Finland) was introduced in Sweden in 2001 by the Radiation Protection Authority (SSM). 1200 instruments were distributed to about 300 fire brigades as well as 300 municipalities and other organisations involved in the national preparedness organisation. The Swedish Civil Contingencies Agency (MSB) provides annual training on the SRV2000 for all Swedish municipalities and county councils. The training includes a basic education about ionizing radiation and threats (MSB 2007). The missions are predicted to include evacuation, triage with subsequent personal decontamination, establishing boundaries of the safety and security perimeters and, in extreme situations, to secure and move sources (Runesson 2003, MSB 2008). Currently a new, complementary intensimeter with an optional gamma/beta probe is placed in service at fire brigades, emergency rooms and police departments. Training is important for building confidence in the ability to solve the task once an incident occurs, regardless of the likelihood of occurrence. The lack of training instruments and the difficulties and costs of safe exercises on dedicated training areas result in a need for a complementary training solution.

Computer games with an intention to convey to the gamer real life useful skills are categorized under many various labels with even more definitions. One popular label is “Serious Game”, somewhat simplified defined as a game with the goal to educate rather than entertain. A serious game has rules, objectives, and motivated consequences of the player’s actions. (Michael, Chen 2005). Yet another label is “Lightweight Simulation”, which refers to computer-based training systems designed to teach specific mission skills (Alexander et al 2005).

RadiaX has a threefold purpose; to teach (i) the handling of specific instruments, (ii) the proper procedures in missions and (iii) basic principles in radiation physics and radiation protection. Therefore it can fit both the labels mentioned above.

The gamer, the user of RadiaX, is henceforth denoted as the trainee and RadiaX is in the same way simply referred to as the game.

Material and methods

To provide an easily accessed and widely distributed training system the game is made as a modification based on an existing game, Half-Life 2 (HL2). Development cost and time is cut by starting from a released game and modifying and reusing modules from the original game, only adding the needed code for radiation simulation.

HL2 is a commercial computer game of the type FPS (First Person Shooter). HL2 is developed by Valve Corporation (Valve Corporation, USA) and was voted Game of the Year 2004 in several publications and engages over 10 million players. The game uses the 3D game engine Source and a modified version of the Havok physics engine. Utilizing the Source SDK user modifications (mods) are made possible and can be distributed to any holder of a license for HL2.

The levels are created in Valve Hammer Editor, included in Source SDK. It is the official Source mapping tool, and once mastered it is a powerful tool.

Creating levels requires only minor programming experience, and is intended to be attainable by the training instructors for ad-hoc or local debriefings and demonstrations. The existence of a mature content creation tool chain is a major factor in deciding to base a game on an existing engine instead of developing one from scratch.

The radiation model, including radiation sources, background radiation, photon attenuation and dose accumulation (for the trainee and instruments independently), is added to the engine using C++, compiled with Microsoft Visual C++ 2005 Express.

The ambient dose equivalent rate (ICRU 1998) at the trainee's position, $\dot{H}^*(10)_{trainee}$, is calculated momentarily as a function of gamma-ray constants applied to specific ideal point sources (Tschurlovits et al 1992), the distance (inverse-square law) and the attenuation in different materials in the pathway between the radiation source and the instrument (Beer-Lambert law). The calculations are formalized in equation 1 below. The gamma-ray constants, Γ, are given for 1 meter source distance.

$$\dot{H}^*(10)_{trainee} = \Gamma \cdot \frac{A \cdot e^{-\mu \cdot x}}{d^2} \quad [\mathrm{mSv/h}] \qquad (1)$$

where Γ = gamma-ray constant [mSv m^2/MBq h]

A = source activity [MBq]

μ = linear attenuation coefficient of attenuation material [cm^{-1}]

x = material thickness [cm]

d = distance between source and receptor [m]

The existence of attenuating material and its thickness is determined by tracing rays from the radiation source to the specified measurement point. Attenuation coefficients (Hubbell 1982) are chosen from an internal table depending on the material used (e.g. wood, concrete) for each piece of intersected level geometry (set by the level designer during level creation). This allows the game to use HL2's existing library of materials, and also allows levels to be designed without the level designer having any knowledge of radiation physics. Instead, attenuation accuracy simply becomes a function of mapping accuracy.

Dose accumulation is modelled through summation of discrete dose quanta. At fixed time intervals, for each radiation source the current dose rate is calculated at the trainee's position and the corresponding dose quantum is added to the trainee and to each simulated instrument the trainee carries and have switched on. This is done independently, the actual dose received and the instrument-measured dose may be different. Due to the Source engine design, the maximum attainable frequency at which the dose quanta can be calculated is 50 Hz. However, since the attenuation path calculation is computationally time demanding, the dose accumulation update frequency is intentionally kept lower, at 20 Hz, to keep the systems requirements down.

The radiation sources placed by the level designer are ideal point sources that are invisible and have no physical interaction with the world. They can however be coupled to other physical game objects, allowing movable, fully interactable radiation sources using Source's built-in rigid-body physics engine. When placing radiation sources, the level designer selects a nuclide name (e.g. Cs-137) from a list and specifies an activity (Bq) as seen in Figure 1. Appropriate gamma-ray constants and attenuation coefficients are retrieved from an internal list automatically by the game.

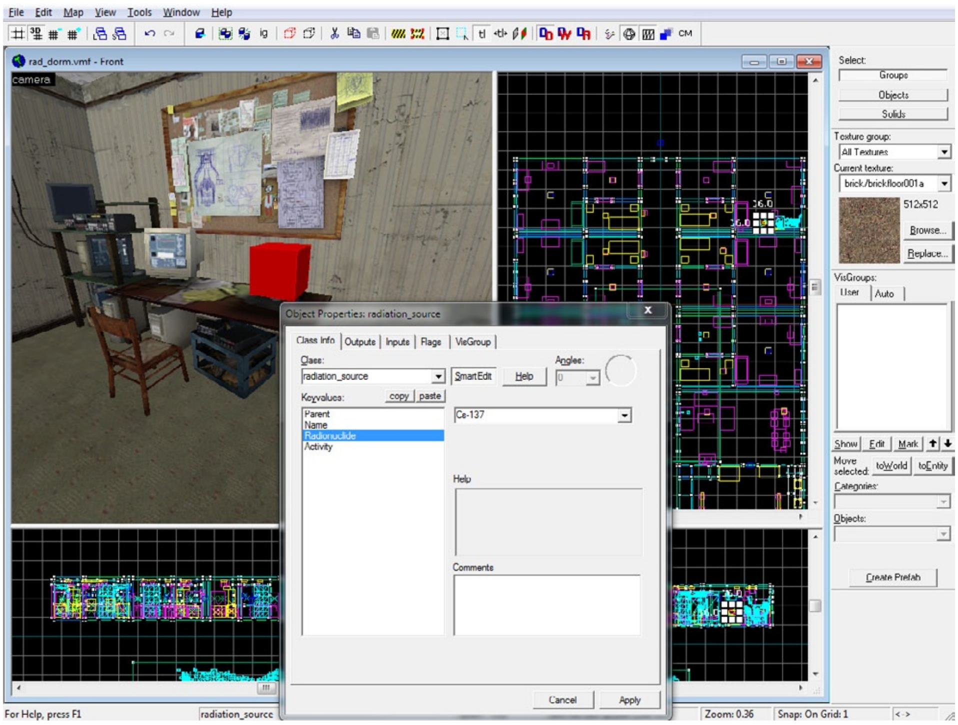

Fig. 1. A screenshot from placing and editing a radiation source using the mapping tool Hammer. The radionuclide (here Cs-137) and its activity is specified in a dialog window. The position of the source is shown in the floor plan (a white square in upper right corner) and in the camera view as a red box. During live training the source itself is invisible but coupled to any other object.

Background radiation is simulated uniformly across each level with stochastic fluctuation around a set mean value.

In accordance with the instrument specifications, inaccuracies are added to the calculated dose rate values, and time constants as well as delays are applied. Alert and alarm signals and adjustable level functionalities are implemented.

The dose rate error is modelled as having both a dose rate-invariant (absolute) component and a dose rate-relative component. This gives good error fidelity under both low dose rates, when the absolute error dominates, and high dose rates, when the relative error dominates. The errors are assumed to be normally distributed, with the deviations set on a per-instrument level.

Levels can be designed to present the trainee with a short in-game debriefing upon conclusion of the mission. This takes the form of a pop-up window showing, among other things, the trainee's accumulated dose during the mission, together with easy-to-understand information of the received dose. This information, e.g. “equivalent to x months' worth of background radiation”, is designed to help the trainee to gain an intuitive understanding of radiation doses and to gauge the real-life risk involved in the simulated tasks.

Supporting tools such as a distance measuring device and a position marker (paint spray can) is added to the arsenal.

The game is complemented by two separate user manuals. The first is intended for the trainee and includes installation manual, optional game settings, the game commands for movement and instrument action and a brief background. The second manual is intended for instructors, extended with more in-depth information (e.g. level design, radiation protection, instrument manuals and dose evaluations).

Results

Specified or observed features of the instrument SRV-2000 are implemented. Each setting and mode is accessed through standard keyboard and mouse by the corresponding buttons on the instrument as in real life. The SRV-2000 is always virtually held in the left hand (Figure 2) in front of the trainee, most of it visible at all times (buttons and display).

Fig. 1. A screenshot from RadiaX. The SRV-2000 is virtually held in the trainee's left hand and the buttons are selected using specified keys or mouse buttons. Each button is highlighted for a second when selected.

Table 1. The present set of game levels with a short description of their contents and purposes.

Game level	Brief description	Learning goals
Exercise: Basic motion	Obstacle course	Move around in the game environment using keyboard and mouse
Exercise: Characteristic features of SRV2000	Pinpoint hidden source among 17 boxes using different indicators	Start-up, buttons and the characteristic time constants of the SRV2000
Exercise: Basic radiation protection, part 1	Approaching a radiation source and observe the inverse-square law,	Fundamental principles of radiation protection: Distance
Exercise: Basic radiation protection, part 2	Different materials of different thickness between source and detector	Fundamental principles of radiation protection: Shielding
Exercise: Basic radiation protection, part 3	Accumulated dose during exercise and time spent.	Fundamental principles of radiation protection: Exposure time
Scenario: Dorm	Suspected hidden source for antagonistic objectives in student dorm	Search for radiation source
Scenario: Warehouse	Incident in warehouse with radiation sources in stock. In-house personnel remaining.	Search for radiation source, evacuate victims, establish boundaries of the safety and security perimeters using distance tool and marker.

The game levels are divided into exercises and scenarios. In the exercises the trainee is guided through the course and the radiation source is often visible. The exercises are either specific for SRV-2000 with the learning goals to teach characteristic features of the intensimeter, or general with the learning goals to teach fundamental principles of radiation protection.

In the scenarios the trainee is expected to solve tasks based on a specific scenario. Current scenarios (Table 1) are a demonstration of the capabilities of the game, while the exercises are general for the game or specific for SRV-2000.

Discussion

A simulation tool is a complement to live training by offering possibilities for debriefing, classroom as well as individual training, demonstration and increasing the repetition frequency as well as the numbers of trainees. Using existing standard PCs the training is cost effective and easy to distribute. This increases the availability of the training and might inspire to "off-hours" individual training.

The learning goal to teach radiation protection includes the awareness of accumulated dose levels in certain circumstances. The doses are maybe lower than the trainee first expects.

Extensive testing is required during development to ensure accurate reproduction of instrument behaviour, as simply reading the specifications might not be enough to cover all cases. If the real-life behaviour of a device differs significantly from the simulated behaviour, simulator training might have a detrimental rather than beneficial effect.

Further development

At present only one specific intensimeter is implemented. MSB and the National Board of Health and Welfare together with SSM have recently distributed a complementary instrument with a gamma/beta probe, the Intensimeter 28/T, a slightly modified AN/UDR-13 (Canberra Industries Inc, USA). The Swedish national preparedness organisation under SSM holds a wide and diverse range of instruments. The intensimeter 28/T particularly and many other instruments are justified to add to the arsenal available to the trainee in the game. This extends the target group to include customs staff, medical staff as well as radiation protection experts.

Further development (table 2) aim to include all scenarios listed in table 3. Exercises and tutorials supplementing new instruments and scenarios add to the table. Demonstration and reassess levels of occurred incidents or established training scenarios are of interest.

Giving feedback, perhaps the most important part, is a demanding and often neglected part of the preparations of a radiac exercise. Giving the mission of each game level scenario in-game and presenting the result, e.g. dose, time and effectiveness, to the trainee are important parts of all future game levels.

Table 2. A selection of intended future additional set of game levels with a short description of their contents and purposes.

Game level	Brief description	Learning goals
Exercise: Characteristic features of Intensimeter 28/T	With and without probe	Start-up, buttons and characteristics of the instrument
Tutorial: Basic body scan	Demonstration of body scan procedures	Scan strategies
Exercise: Basic body scan	Body scan with visible sources	Practice scan strategies
Scenario: Decontamination	Body scan and triage	Practice scan strategies for decision support
Scenario: Transport accident	Transport accident involving radioactive material and contaminated victims and bystanders	Search for radiation source, evacuate victims, establish boundaries of the safety and security perimeters. Triage.
Scenario: Urban RDD	RDD in an urban environment	Life saving prior to radiation risk

Table 3. Predicted scenarios (TMT handbook 2009, IAEA 2006).

Suggested scenarios
Radiological Exposure Device (RED)
Radiological Dispersal Device (RDD), “Dirty bomb”
Attack on, or incident with transport of, radioactive material
Contamination of food and water supplies
Attack on, or incident in, nuclear installation or installation containing radioactive material
Improvised Nuclear Device
Uncontrolled dangerous source

Conclusions

RadiaX is a novel complement in radiac preparedness training. It offers easily distributed and changeable training in a non-lethal environment.

Radiac training is well suited for simulation when the fidelity of the instrument is the primary focus and other simulations secondary, since the radiation does not need any tactile or visual simulation.

RadiaX has been demonstrated to training instructors and first responders and was well received.

Further development, with added instruments and scenarios, and evaluation of the training transfer is in progress.

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Radiological emergency exercises facing the collaboration issue of different response authorities

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Abstract
In Sweden, two large radiation emergency exercises with international participation were held in 2001 (Barents Rescue) and in 2006 (DEMOEX). Experiences from these exercises showed that conducting "close reality" emergency situations utilizing real radioactive sources was very valuable in the training of radiation experts and first responders. In the years 2007-2009 this concept of combined training was further developed in a set of smaller-scale exercises called Lärmät. The concept was to break down the large scale scenarios to pieces and learn its radiation protection difficulties in detail. This showed to be a successful and appreciated way of conducting exercises for Swedish specialists. The radiation specialist, however, will never have free hands to handle the situation in case of a malevolent act involving radioactive material. The situation would involve combined actions of several authorities. Therefore, Sweden's combined radiological emergency response has an urgent need for exercises where authorities work closely together across authority borders. This issue was specially addressed in the exercise Lärmät 09. Teams were put together from radiation protection specialists, rescue services, police, the National Laboratory of Forensic Science, The National Board of Health and Welfare and the Swedish Customs. The idea was to let people from each profession present their view (in each different scenario) of the situation in order to solve it safe and not destroying potential evidence. In Lärmät 09 six different scenarios were built with real sources and inspired from real radiation accidents. The radiation doses to the participants were kept low by radiation protection planning and technical arrangements. This paper presents the Lärmät 09 scenarios and lessons learned from this joint "close reality" exercise.

Introduction
Real radiation emergencies are extremely rare and experts and first responders only way to improve their practical and organisational skills are to expose them for unknown radiation scenarios. Training the Swedish specialists in radiation scenarios of different kinds has been one primary task for the Swedish Radiation Safety Authority (SSM) since 2006. This suite of exercises originate from the large and appreciated exercise Barents Rescue in 2001 and was followed by DEMOEX exercise in 2006, (Finck et al.

2008) which has been viewed as the starting point to the yearly training program held by SSM. The training is mostly as close to a real radiation emergency as practically possible without jeopardizing full safety and control.

Prerequisites for the exercises

The National Laboratory of Forensic Science (SKL), gave a brief lecture in crime scene behaviour. This was done to add the dimension of radiological difficulties at a crime scene investigation. The Swedish customs has participated in the radiological emergency exercises DEMOEX (2006), Lärmät 07 and Lärmät 08 with success and shown that also non-specialists benefit from this type of exercises. The focus of Lärmät 09 was to train small teams of radiation specialists, first responders and customs officers working together. The disparate background of the team members makes this a challenge. The participants were divided into teams with two radiation experts, two first responders and one customs officer in each team. All medical radiation physicists and persons working within the field of radiation protection were designated as radiation experts. In each team a radiation expert was selected as a team leader and as responsible for the radiological safety of the team members. It should be noted that the radiological issues were the main goal of the Lärmät 09 exercise. The drill of first responders and laymen came secondary.

The exercises

Activity	Container	Dirty Bomb	Strong Source	Bomb Workshop	Indoor malev. act	Contamina-ted person	Mobile Syst. Disp.
No. of Teams	12	6	6	12	12	12	12
Tot. No. Particip.	60	30	30	60	60	60	60

Fig. 1. Showing the number of teams exercised at each activity and number of exercised participants.

1. Container filled with contaminated scrap metal

1.1 Objective

In the last few years Sweden have had several incidents where companies who recycle scrap metal has caught radioactive material in their portal monitors. The objects has been of different kind, such as water filtration units with accumulated NORM and thorium treated air plane engine parts. The wide range of what might be found and the lack of experience how to safely and effectively sort out the radioactive objects from the rest of the scrap metal, called for a controlled exercise.

1.2 Exercise setup

The task for a team was to locate and identify the radioactive material inside the container using hand held instruments of a mobile emergency preparedness specialist team. The container was filled completely with approximately 20 tons of scrap metal. The inactive metal parts, the bulk, was so large in size and weight so moving them by hand was a very ineffective, if at all possible, way to proceed. An excavator, including operator was at hand with a scrap metal processing gripper. The container was prepared with 7 sources of different kind. A flight engine part containing thorium, a stainless pipe coupling contaminated with Co-60, smoke detectors of different kinds and instruments with illuminated paint containing Ra-226, were some examples of sources used. In order to complete the exercise in 3.5 hours the team had to plan their work efficiently. The team was informed that they were responsible for the radiation safety of the person operating the excavator.

Fig. 2. Left picture shows the container before activity start, right picture shows a team in action searching for sources or contamination.

1.3 Results

Several teams expressed their unfamiliarity with this type of scenario. In general the teams didn't move the active metal parts away far enough from the container and therefore had problems with finding radioactive objects remaining in the container. Participants also lacked the ability to explain to the operator what they wanted to be done. Some groups didn't have the correct instrumentation for the task. All sources were found for all 12 teams except one, which missed one NORM object. A few teams found the smoke detectors just by eye. The gloves used by the teams showed low level contamination of NORM activity, such as Thorium, after the completion of the exercise.

2. Dirty Bomb exercise

2.1 Objective

A dirty bomb is a device with the purpose of dispersing radioactive material in the blast. A dispersive device will most certainly have great impact on society infrastructure even if the radiological harm to people is negligible. As emergency teams have to handle a dirty bomb situation properly, training is needed.

2.2 Experimental setup

The scenario was built inside a military Operations Urban Terrain (MOUT) called SIB belonging to the P7 tank regiment outside Revinge in the south of Sweden. A blown up car was placed in a street crossing and the close surroundings of the car were fenced off to ensure radiation safety. Entering into the marked hot zone was strictly forbidden for all participants. A well collimated Ir-192 source of approx. 440 GBq (12 Ci) with its radiation field pointing upwards was placed on the floor in the car, producing only scattered photons and sky shine. The dose rate around the vehicle was under 0.2 mSv/h and inside the fenced off vehicle close to the source were about 1000 times higher.

On the ground, 35 hot particles in the form of small contaminated metal plates were placed to simulate debris.

Fig. 3. Left picture showing a team assessing the radiation field with the supervisor being the person i yellow to the left. Right picture showing the optimal geometry in search of ground contamination and hot particles.

Also two larger contaminated fragments of metal were placed on the ground close to the vehicle to indicate a blown source container. The activities on the small plates were 1-3 MBq and on the two larger fragments 300 MBq each. The radionuclide used in this exercise was Zr-89 with the half life of 78.4 hours. This radionuclide is both a positron and a gamma emitter and is produced at the cyclotron at Lund University Hospital. The purpose of the Ir-192 source was to create a radiation field that strikes out portable spectrometers by producing very high dead time. This forced the team to take samples for analysis, when the fragmented sources were to be characterized.

2.3 Results

One participant was slightly contaminated on the left shoe. One team had to be told that there were radioactive fragments on the ground. One team missed to equip the person closest to the car with a dose rate instrument at all times. The same team took it for granted that the radiation field levels would stay static in time. Problems with documentation during rain showed need for water proof writing pads and water resistant laptops. Few of the teams had the experience of taking smear samples in practice. The portable instrumentation was not well suited searching of fragments on the ground.

3. Strong source exercise

3.1 Objective

Small and very strong radioactive sources exist in society. Very strong sources impose special problems which are rarely dealt with in exercises. The objective of the strong source exercise was to familiarize the participants how to assess a strong source situation.

3.2 Experimental setup

Both geometry and source strength was disclosed to the teams before exercise start. A Co-60 radiographic source of 385 GBq (10.4 Ci) was used as the ”strong source” and was collimated to a cone beam exiting along the earth surface. The source was placed 200 meters in at one end of a large (400x1200m) grass field.

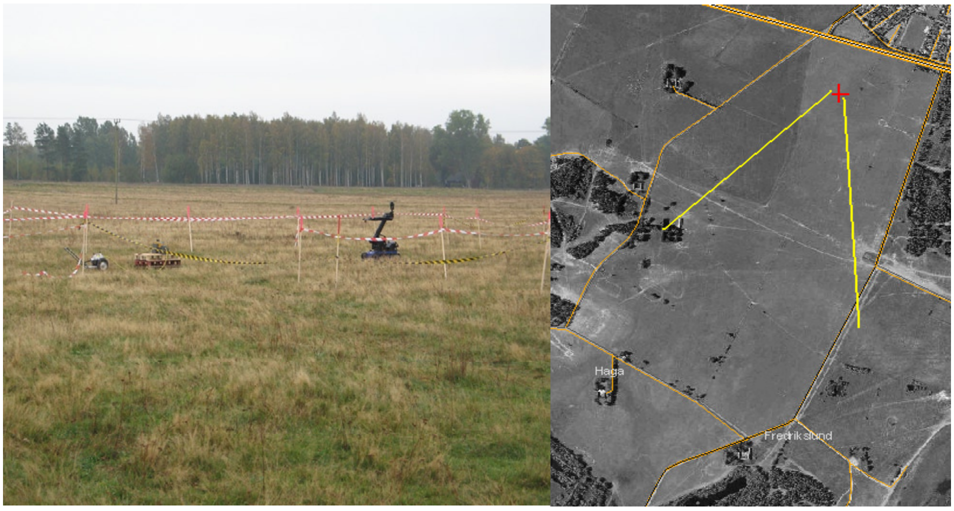

Fig. 4. Left picture showing the police bomb robot closing in on the cobalt source, right picture showing the extent of the radiation field.

Approximately 20 cm of lead and a 4 cm thick tungsten collimator was used as a radiation field delimiter to achieve the desired field geometry. A dose meter with alarm capability was placed close to the shield to indicate if the source got stuck on the way to the collimator or shield. To be able to practice measurements in a safe way, warm- and hot zone was marked by coloured fence posts, positioned at the dose rate levels of 100- and 500 microsievert an hour. The dose rate level was also written on the fence posts for clarification. The participants was allowed to work freely outside warm zone and allowed inside ”warm zone” delimiter only by making a valid request. No participant was ever allowed inside the fenced off ”hot zone” for safety reasons.

3.3 Results

One person chose not to participate in this exercise. The police bomb robot was granted access to the hot zone for practising measurements and also allowed to move freely inside the fenced off area. The Police bomb squad need more practice and guidance to be able to function in situations like this. Almost all teams requested to enter warm in order to test their instrumentation.

4. The Bomb Workshop exercise

4.1 Objective

09 included a exercise scenario illuminating how to preserve forensic evidence at a crime scene contaminated by radioactivity. Earlier exercises have included several

scenarios with gamma emitting nuclides as radioactive contaminant. This time a scenario including only a pure beta-emitter was chosen. The cause for this was mainly to show and train the participants in the complexity of estimating activity and extent of the contamination when only beta sensitive instruments were the proper choice.

4.2 Experimental setup

Two parallel rooms separated from each other where set up with covered windows, a workshop table carefully covered with construction plastic film, necessary tools to construct a explosive device and plastic film covered floors.

Fig. 5. A participant working on mapping the contamination in the Bomb Workshop activity.

Two teams worked on the scenario in parallel without communicating with each other. Contamination was of very low grade but carefully calculated and tested not to give any indication of radioactivity on gamma-only instruments. The nuclide used for this exercise was Y-90, a perfect simulant for the more common Sr-90. The 1.8 MBq/l water solution created a surface contamination of about 10 Bq/cm^2. Invisible, but radioactive, footprints were placed on the floors. On the tables radioactive fields the size of a hand was created to simulate grabbing marks. Not all tools were contaminated and surfaces not likely to have been used were clean.

4.3 Results

Two teams used only gamma detectors when entering the scenario the first time and one of these teams had contaminated members on exit. A couple of teams found it difficult to estimate activity from the beta measurements. Several participants found it hard to do quality measurements with protective suits and mask. Beta instruments with long time constants were hard to use in this scenario. All teams were not properly prepared to convert a cps-value to Bq/cm^2. No instruments were contaminated.

5. Indoor malevolent acts

5.1 Objective

This scenario was compiled out of two different activities introducing different problems. The ”School” scenario was based on a malevolent act playing with the thought of one person wanting to harm school children by placing gamma emitting sources in the school room. The ”Home” scenario presented a house in which a internally contaminated person had stayed. The objective in both scenarios was to assess the radiation fields and contamination levels. The teams should gather information through measurements and estimate radiation doses in both situations.

5.2 Experimental setup

The ”School” scenario was set up in a big room pretending to be a classroom with four tables and eight student chairs. In front of the classroom a teacher’s desk were set up to add realism. A fairly strong Co-60 in the cellar underneath the classroom enhanced the radiation level in the room to several hundred microsievert per hour. Additionally, four Co-60 point sources were located in the classroom underneath the chairs and tables. The ”Home” scenario was built around a home environment. A room with a kitchen, stairs, dinner table and a bed was set up to simulate a home situation. In order to make a traceable chain of contamination, the water was chosen to be the carrier of the activity. Therefore, all things that could have been in contact with water had to be carefully prepared to fit into the scenario. The nuclide used was I-131. Water I-131 concentration in both the coffee maker and sink had to match the activity concentration and the total activity in the earth of the potted plants as well. Also the bed and a reading chair were contaminated in a way to lead the teams to the conclusion of household water as the course for contamination.

Fig. 6. Left picture showing a team working in the ”Home” activity. Right picture is a view showing the farm house to the right where the ”School” activity was held.

A faked well was prepared outside the building in case the teams would search for a water supply in the close proximity. By placing a 30 MBq Iodine-131 point source in the well, it could be detected from a distance and simulate a real well full of contaminated water. A real sample of contaminated water (1.8 MBq/l in a 250 ml plastic can) was also placed in the faked well. The idea was that the team should use the activity concentration of the well water as a possible starting point for dose assessment calculations.

5.3 Results

In the school scenario several teams had difficulties locating the four point sources as they were obscured by the gamma field from the cellar source. A few teams reported wrongly a surface contamination in the classroom. Two out of six teams found all four point sources and completed all scenario tasks in a correct way.

In the home scenario all teams found the source of contamination and found the water sample in the faked well. All teams kept their members uncontaminated and several samples were taken and analysed. A few teams used their mobile spectrometer to estimate the I-131 water concentration. All teams collected enough information and did correct measurements to be able to correctly estimate the internal radiation dose.

One team showed only a calculation error which presented an accumulated dose 1000 times to low.

6. Mobile gamma system display exercise

6.1 Objective

The development and new commitment made by the SSM to enforce the radiation protection nationwide with new modern equipment, has accelerated the development of methods and new custom made software platforms. To show the first responders what instrumentation and equipment Swedish specialists use, a demonstration type of mobile gamma search was set up.

6.2 Experimental setup

Several systems from the Swedish customs, SSM and National emergency preparedness laboratories were set up aside a straight road to monitor passing cars. Several packages were prepared in advance with manmade radionuclides and some packages also included natural occurring radionuclides. Examples were potassium-40 and uranium ore. These packages were placed in one of four cars passing the measuring teams. The teams had to pinpoint any car containing radioactivity and disclose the radionuclide by means of gamma spectrometry. To enhance the test several cars passed the teams with blank packages. The speed of the car convoy was also varied in order to spot any differences in the ability to detect the content.

6.3 Results

The non-specialists found it hard to in some cases to decide with certainty if a weak source had passed the detector or not. It was not clear to all that mobile detection units (cars) often have different sensitivity for sources passing either on the left or right side of the vehicle. Some participants believed that this exercise was less relevant to them.

7. Contaminated person

7.1 Objective

In a radiological accident situation involving injured people the ability to do a quick, correct search for internal or external contamination is essential. In this exercise the goal was to educate the participants how to handle contaminated persons, in this case a contaminated dummy doll.

7.2 Experimental setup

The teams were given a pre-prepared doll with contamination in the hair, on the clothes, in the lungs and a flesh wound containing a hot particle. The activities used were 6, 60, 0.2 and 30 MBq in the previous given order. The radionuclide used was Tc-99m with 6.05 hour half life. The doll had to be undressed outside in a fenced off area and brought inside on a stretcher. The teams should estimate the extent of the contamination and make a report on their findings.

7.3 Results

The activity was much appreciated. The teams had problems with finding the lung contamination since the other contaminations dominated instrument response.

8. Follow up meeting

A vital and successful part of Lärmät 09 was the follow-up meeting two months later. The participants were asked to hand in all results and calculations obtained at each scenario prior to this one-day meeting for compilation. Time was also given for general discussion regarding lessons learned during Lärmät 09.

Discussion

The general response from the participants was very positive. All participants were asked to express their views on the different tasks by completing a questionnaire before leaving each scenario. In the Dirty Bomb and School exercises the standard gamma search instrumentation for Swedish teams was not less useful in the high "background" levels encountered. One team measured and mapped dose rate levels around the Dirty bomb vehicle and thereafter dose rate instruments were considered of no more use when working in close proximity to the vehicle. Acting as if a situation is static in radiation dose levels, thinking there is no need to carry a dose rate meter violates the plan of radiation protection optimization.

The Strong Source exercise showed the need for collaboration between specialists and police bomb squad units. The units in Sweden have different equipment and robots. Distance equipments and robots are indispensible dealing with very strong radioactive sources, but further training is necessary in this field. A part of the strong source exercise, the sky shine geometry, was postponed to a future exercise due to time constraints and the complexity of such a radiation field. The lesson learned from the Bomb Workshop scenario was the preconceived assumption that a gamma detector is enough, scanning a contaminated room, must be abandon.

Since several participants were contaminated more training in contaminated environments is needed. No instruments were contaminated. This showed that the participants handled their instrumentation carefully in this aspect, though, better instrument contamination protection was called for by the participants.

In the School exercise all teams made a survey outside the house before entering the building. Based on this survey most teams suspected a strong source in the inaccessible cellar compartment. Only a few teams asked for better search instruments, a Saphymo was at the teams disposal on place, and did not realized that their standard GR-110 and GR-135 instruments were less adequate in the high dose rate levels in the class room. In the Home scenario no instrumental shortcomings were recognized. The erratic dose calculated by one team was later corrected. All teams found the contamination chain and did a water I-131 concentration measurement which all gave correct results. A few teams who measured their samples in their mobile laboratories on site showed the same accuracy as others.

The Mobile Units Display exercise showed the need of trained specialists to discover sources giving weak signal in the gamma detection systems. The relevance of such a activity was discussed and was considered to be good general knowledge and adding dimensions for also non specialists, making them more all-round. The lack of sensitive neutron instrumentation in the National Radiation Emergency Preparedness Organisation was also highlighted.

The Contaminated person exercise introduced the participants on problems with measuring a contaminated person with different types of contamination. The simulated hot particle made it hard to find the much weaker lung contamination.

In general, in a real radiological event, the specialist upon arrival must gather information quickly to compile a first situation estimate and communicate this to the responsible authority. Lärmät 09 revealed that specialists in several cases didn't fulfil this role adequately. In order to function properly the specialists need more education and training. During the exercise the teams collected measurement data and samples in 3.5 hours, the predefined time slot chosen for all scenarios. In a real malevolent situation it can be expected that the demand from media, press and authorities, after such a time interval or less, are strong for a first verbal report from the radiation specialist. Details and calculated doses were later presented by each group during the follow up meeting. This meeting was much appreciated and gave participants and planners a sense of what to expect in present time and what field to focus on for improvements. The issue of effective surveying and reporting is recognized as very important. This will be addressed in Lärmät 10, which is to be held October 5-8, 2010.

Conclusions

The involvement of observers from SKL came out successfully and added new perspectives how to behave at a radiological crime scene. The set of Lärmät exercises has shown to be a good test and indicator of present status of the Swedish emergency preparedness organisation. It is believed that the Lärmät concept of small yearly exercises is a productive way to ensure ability and also control the status of a chosen radiological emergency organisation. The radiation protection planning and optimization kept the radiation doses below 100 microsievert for all participants during Lärmät 09.

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New educational technique using radiation sources fabricated from chemical fertilizers

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Abstract
Potassic chemical fertilizers contain a small amount of potassium-40, which is a naturally occurring radioisotope. The potassium-40 emits beta and gamma radiation. Therefore, potassic chemical fertilizers are often used for classroom demonstrations of natural radiation sources along with kelp, mushrooms, sinter, and others.

A previous study proposed a new educational approach to enhance understanding of radiation that is naturally emitted from potassium contained in fertilizer. This educational technique involves the use of radiation sources fabricated from several chemical fertilizers containing different amounts of potassium. In the present study, the educational approach was evaluated using five disk-shaped radiation sources fabricated using a compression and formation method from five chemical fertilizers containing potassium from 5 to 50%. The technique was evaluated by examining the relation between radiation strength (radioactivity count emitted from the fabricated source) and potassium amount of the raw fertilizers based on 10- and 1-minute integration times. Radiation strength was directly related to potassium amount, indicating that the substance emitting radiation must be potassium.

1. Introduction
Many materials on earth contain naturally occurring radioisotopes such as ^{40}K, ^{232}Th, and ^{238}U that release radiation. However, many people do not realize that such naturally occurring radioisotopes exist. Therefore, radiation education is important to improve understanding of the existence of natural radioisotopes and radiation. Chemical fertilizers containing potassium are often used for this purpose in educational courses on radiation. Naturally occurring potassium consists of three isotopes: ^{39}K, ^{40}K, and ^{41}K; the ^{40}K naturally emits a 1.33-MeV beta particle and a 1.46-MeV gamma ray.

A previous study described development of a compression and formation method for fabricating disk-shaped radiation sources from raw materials such as potassium chloride, kelp, chemical fertilizer, and sinter, which contain naturally occurring radioisotopes. The fabricated natural radiation sources were examined for educational use in a radiation protection course. The results indicated that the natural radiation sources were easy-to-use educational tools for demonstrating the principles of radiation protection relating to time, distance, and shield thickness.

In the earlier study (Kawano 2010), the compression and formation method was applied to 13 commercially available chemical fertilizers containing different amounts of potassium. The fabricated disks are natural radiation sources, referred to as chemical fertilizer radiation sources or fertilizer sources. The suitability (size, weight, solidness, and smell) of the 13 fertilizer sources as educational tools for radiation education was examined. Results indicated that all of the fertilizers but one could be used as natural radiation sources. The one exception was too fragile to fabricate into a disk using the compression and formation method. The radiation strength (count rate) and potassium content of the other 12 fertilizers were measured, and the relation between potassium content and radiation count was examined. Results showed that a linear relation existed between the radiation count emitted from the fertilizer sources and the percentage of potassium in the chemical fertilizers. This linear relation demonstrates that the count rate corresponds to the percentage of potassium. Therefore, a new educational technique using the fertilizers can demonstrate that the radiation emitted from the fertilizers can be attributed to the potassium contained in the fertilizer sources. However, these results were obtained by compiling all of the data obtained by 12 fertilizer sources; in some cases the relation between radiation count and potassium percentage was inverted. This inversion might be caused by differences in components other than potassium, nitrogen, and phosphorus. Therefore, to demonstrate that the substance emitting radiation is potassium, the selection of fertilizer source material is very important.

In the present study, five brands of chemical fertilizers containing different amounts of potassium were selected as source materials and fabricated into disks using a compression and formation method. As before, the relationship between radiation count emitted from the fertilizer sources and the percentage of potassium in the fertilizers was investigated based on 10- and 1-minute integration times. The results showed a linear relation with no inversions, indicating that the radiation count was directly proportional to the potassium content.

Thus, this educational technique is useful for explaining that the substance emitting radiation in chemical fertilizers is potassium. The results also show that the radiation sources fabricated using chemical fertilizers can be handled safely and easily in radiation courses.

2. Chemical fertilizer radiation sources

2.1 Chemical fertilizer

Nitrogen, phosphorus, and potassium are three important elements for plants. Chemical fertilizers generally contain all of these elements. As a chemical fertilizer radiation source and application for radiation education, potassium is very important because naturally occurring potassium consists of three isotopes: ^{39}K, ^{40}K, and ^{41}K, of which ^{40}K emits a 1.33-MeV beta particle at a rate of 89% and a 1.46-MeV gamma ray at a rate of 11%. Because the half-life of ^{40}K is long (1.28×10^9 years), the chemical fertilizer radiation source is suitable for use as an educational radiation source.

In the present study, five commercially available chemical fertilizers were purchased for producing chemical fertilizer radiation sources, which contain different percentages of potassium, a small proportion of which is radioactive potassium-40.

These chemical fertilizer brands were sold for use in the family garden. Consequently, radiation sources fabricated from the chemical fertilizers can be handled safely, since they are only pressed and formed before use in the classroom.

The five brands of chemical fertilizer were called “sulfate of potash,” “phosphorus and potash,” leaf and vegetable,” “organic chemicals,” and “bulbaceous fertilizer” (words used in their brand names). The amounts of the main elements in these fertilizers are summarized in Table 1, obtained from ingredient labels on the fertilizer packages. The “sulfate of potash” brand contained 50% potassium out of, which was the maximum amount of potassium in the five fertilizers. The “phosphorus and potash” brand contained both phosphorus and potassium at 16% and 15%, respectively. The other three fertilizers contained all three main components. The five fertilizers each contained five different potassium percentages: 50%, 15%, 10%, 8%, and 5%.

Table 1. Percentages of nitrogen, phosphorus, and potassium in five brands of chemical fertilizers.

Fertilizer	Content (%)*		
	Nitrogen	Phosphorus	Potassium
(1) Sulfate of potash	*	*	50
(2) Phosphorus and potash	*	16	15
(3) Leaf and vegetable	12	8	10
(4) Organic chemicals	8	8	8
(5) Bulbaceous fertilizer	3	7	5

*Based on “Ingredient labels” on the packages

2.2 Method of fabricating chemical fertilizer sources

For fabricating chemical fertilizer radiation sources from the chemical fertilizers, about 20 g of fertilizer were micronized with a mortar. This fertilizer powder was placed into a cylindrical stainless steel form and then pressed. The fabrication setup is shown in Fig. 1. The cylindrical stainless form had an inner diameter of 35 mm and a height of 30 mm. A hydraulic hand pump (Osaka Jack Co. Ltd., Model TW-0.7) and a jack (Osaka Jack Co. Ltd., Model NT20S12.5) were used to compress the fertilizer powder in the form with a force of approximately 160 kN to produce disk-shaped radiation sources from the fertilizers.

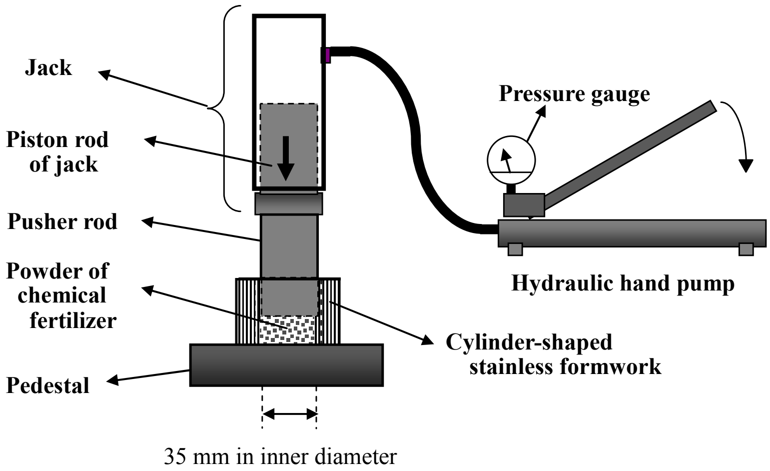

Fig. 1. Fabrication setup for compressing chemical fertilizer powder into disk-shaped radiation sources

3. New educational technique

3.1 Evaluation of new educational technique using five fertilizers

The five fertilizers containing various percentages of potassium were used as raw materials for fabricating radiation sources. The radiation strength of the fabricated fertilizer source was estimated by the radiation count emitted from the fertilizer source. Radiation counts were measured using a GM survey meter (Aloka TGS-146), and were compared with the percentage of potassium among the three main fertilizer elements. The element percentages could be obtained easily from ingredient labels attached to the fertilizer packages. The radiation count was derived by subtracting the background count (measured without the fertilizer source) from that of the fertilizer source. To measure radiation strength, a fertilizer source was placed at the center of the surface of a GM survey meter probe. The measurements were obtained in integration mode. An integration time of 10 minutes was used for both source count and background count, and net count rates at one minute were calculated.

The results obtained are shown in Fig. 2(A)-(C). Figure 2 (A) represents a typical result, in which radiation count is plotted as a function of percentage of nitrogen. It is clear to see that the count does not correlate with nitrogen percentage, because no linear relationship is present. Figure 2(B) shows similar results between count rate and phosphorus fraction. In both Fig. 2(A) and 2(B), radiation counts of about 300 cpm and 120 cpm were obtained from samples with zero nitrogen (A) and zero phosphorus (B), which is explained in Fig. 2(C). Consequently, neither the nitrogen nor phosphorus in the fertilizer sources emits radiation. Figure 2(C) shows the relation between radiation count and potassium percentage. The counts for the five brands of chemical fertilizer were distributed around the line representing the correlation between radiation count and potassium percentage. These experimental results show that radiation strength linearly increases with potassium percentage of the radiation source, meaning that the substance emitting radiation must be the potassium present in the raw fertilizers. Thus, the radiation counts of 300 cpm and 120 cpm observed at zero nitrogen and zero phosphorus, respectively, also result from potassium.

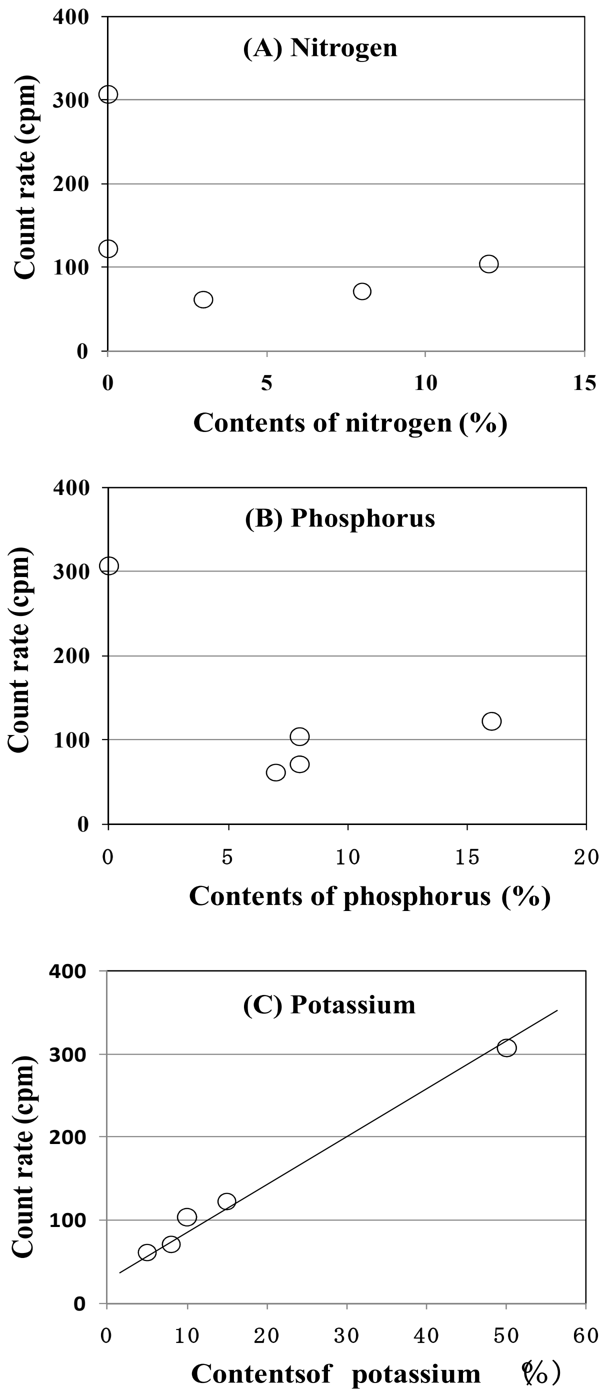

Fig. 2. Relation between component percentage and radiation count obtained with 10-minute measurements

3.2 Evaluation of new educational technique with one-minute measurements

Five fertilizer sources were fabricated from the five chemical fertilizers selected as raw materials. The educational method proposed in a recent study was applied using the five fertilizer sources. Results demonstrated that radiation was emitted from the potassium in the chemical fertilizers. However, this result was based on 10-minute integration time, which may be too long for an actual educational radiation course because younger students may not stay focused for multiple ten-minute measurements. For this reason, the integration time for a single measurement should preferably be one minute or less to allow collection of many data points without loss of student focus. Therefore, further experimental studies were conducted based on one-minute integration times for simulation of an actual educational radiation course.

For the studies using a one-minute integration time, the measurements were repeated ten times. Each set of measurements were conducted for all five fertilizer sources. The results are shown in Fig. 3 and are similar to those shown in Fig. 2(C) based on ten-minute integration time. The average radiation counts for one minute plotted as a function of potassium percentage exhibited a linear relation with a standard deviation of less than 10%, which indicates that count rate reflects the potassium content.

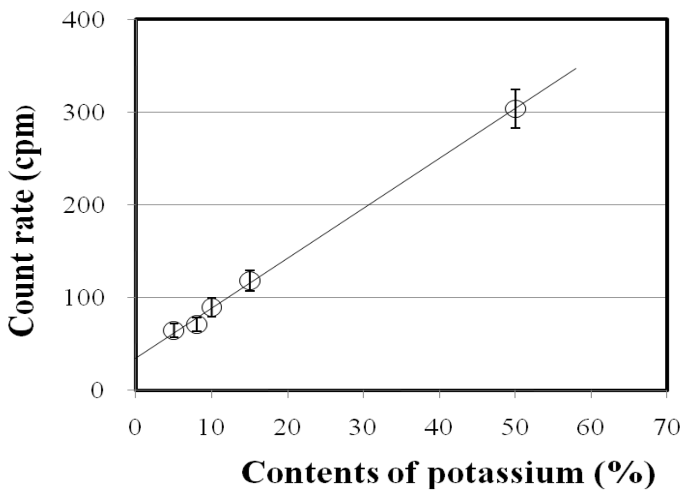

Fig. 3. Relation between potassium percentage and radiation count obtained with one-minute measurements.

4. Conclusions

In a previous study, a new educational tool was proposed for understanding that potassium in fertilizer emits radiation. In the present study, that tool was evaluated from a practical view. Five radiation sources were fabricated from fertilizers containing different amounts of potassium and the relation between radiation count (based on one-minute integration time) and potassium fraction was examined. Results demonstrated that a linear relation existed between radiation count and potassium percentage, confirming that potassium is the source of the radiation from the fertilizer.

This approach can be used to teach students enrolled in a radiation educational course that chemical fertilizers can emit radiation. The experimental portion of the course can utilize natural radiation sources that allow students to handle naturally occurring radioisotopes and recognize that radiation can be emitted from natural materials. This experience helps students understand that people are irradiated continuously with low levels of natural radiation emitted from materials containing natural radioisotopes.

Acknowledgements

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Organisation of pilot modules of the newly developed European Radiation Protection Training Scheme ERPTS

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Abstract

The ENETRAP II Project (European Network on Education and Training in Radiation Protection) aims at harmonizing the Education and Training in Radiological Protection RP e.g. by reaching harmonization of the requirements of RP experts and officers within Europe. These "Reference Standards will reflect the needs of the RPE and the RPO in all sectors where ionising radiation is applied. Therefore a remodelled modular European Radiation Protection Course has been developed in the ENETRAP first phase project. One major goal of the present project is to monitor the effectiveness of the proposed methodologies by organising pilot sessions of selected training events within Work package 8. The courses are designed for radiation protection professionals such as Radiation Protection Experts (RPE) and Radiation Protection Officers (RPO) according to the agreed standards and include On-the-Job OJT Training as key element. Domains to be chosen are occupational radiation protection in Nuclear Power Plants, Radioisotope Training in Non-nuclear industry and Research, and Specifities of Waste Management and Decommissioning. In this paper the outcome of the pilot sessions as far as available is evaluated and the results summarised. Preliminary improvements are recommended for further performance.

Work management to optimise occupational radiological protection at nuclear power plants

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Abstract

In order to provide guidance to the nuclear community on the best practices of work management in nuclear power plants, the Information System on Occupational Exposure (ISOE), sponsored by the OECD Nuclear Energy Agency and the International Atomic Energy Agency, established an international expert working group to update a report on this topic, previously issued in 1997.

The report, *"Work Management to Optimise Occupational Radiological Protection at Nuclear Power Plants"*, was published in 2009. It provides practical guidance based on the operational experience within the ISOE programme in the key areas of work management to optimise occupational radiation protection, including: Regulatory aspects; ALARA management policy; Worker involvement and performance; Work planning and scheduling; Work preparation; Work implementation; Work assessment and feedback; and Ensuring continuous improvement. The specific aspects of work management applicable to each of these areas are illustrated by practical examples and case studies arising from ISOE experience. This paper summarizes and describes the key messages of the report.

Introduction

Occupational exposures at nuclear power plants worldwide have steadily decreased since the early 1990s. Regulatory pressures, technological advances, improved plant designs and operational procedures, ALARA culture and information exchange have contributed to this downward trend. However, with the continued ageing and possible life extensions of nuclear power plants worldwide, ongoing economic pressures, regulatory, social and political evolutions, and the potential of new nuclear build, the task of ensuring that occupational exposures are As Low As Reasonably Achievable (ALARA) continues to present challenges to radiation protection professionals.

Since 1992, the Information System on Occupational Exposure[1] (ISOE), jointly sponsored by the OECD Nuclear Energy Agency and the International Atomic Energy Agency, has provided a forum for radiological protection professionals from nuclear

[1] For more information, see ISOE web site: www.isoe-network.net

power utilities and national regulatory authorities worldwide to co-ordinate international co-operative undertakings for the radiological protection of workers at nuclear power plants. The ISOE objective is to improve occupational exposure management at nuclear power plants by exchanging relevant information, data and experience on methods to optimise occupational radiation protection.

Key to effective occupational exposure management has been the widespread understanding of the need for careful planning and execution of refuelling and maintenance outages. This approach, referred to as *work management*, stresses the importance of approaching jobs from a multi-disciplinary team perspective, and of following jobs completely through all stages from conception to post-job follow-up.

In order to provide guidance to the nuclear community on the best practices of work management, ISOE established an international expert working group to update a report on this topic, previously issued in 1997. This recognises that while work management is no longer a new concept, continued efforts are needed to ensure that good performances, outcomes and trends are maintained in the face of current and future challenges.

The report, *"Work Management to Optimise Occupational Radiological Protection at Nuclear Power Plants"*, was published in 2009 (NEA, 2009). It provides practical guidance based on the operational experience within the ISOE programme in the main areas of work management. The specific aspects of work management applicable to each of these areas are illustrated by practical examples and case studies arising from ISOE experience.

Principles of work management

Work management is a comprehensive and iterative approach to work. The philosophy of work management is a continuous loop that consists of planning, preparation, implementation, assessment and follow-up in order to make the overall work progressively optimised and using a multi-disciplinary team approach involving all relevant stakeholders (see Figure 1). Feedback is a key component, and such feedback should be obtained both locally and globally. Assessment and feedback is the final stage of work and, at the same time, the first stage of the process. However, work management is also forward looking. Therefore, recognising the constant evolution of many parameters that are included in the above topics, such as ongoing technological advances, as well as using past and current lessons to not only inform future work but also future design and operations, the report closes with a chapter on "Ensuring Continuous Improvement".

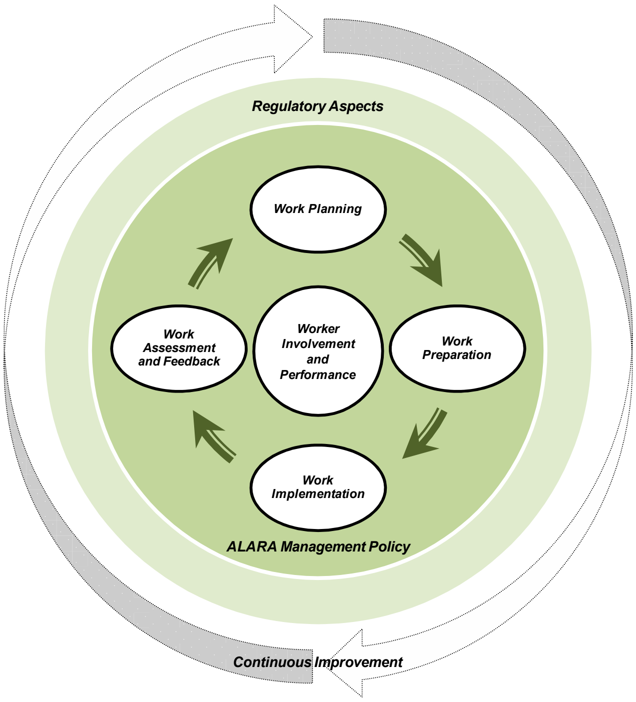

Figure 1. Work management elements and their iterative nature.

Regulatory aspects

While it is the licensee's duty, in the first instance, to ensure that a particular operation is safe from the perspective of nuclear safety and radiological protection, this must be done within the applicable regulatory framework.

National regulatory frameworks are based on radiation protection principles and standards developed at the international level. They aim to secure the maintenance and improvement of safety at civil nuclear installations through regulations addressing nuclear safety, and ensure the protection of workers, public and environment from ionising radiation through regulations addressing radiation protection. Such regulation provides for an effective radiological protection infrastructure which includes a "safety culture" shared by those with protection responsibilities from workers through to management.

In the field of safety, there is an effort to develop and implement "performance based" plant maintenance rather than prescriptive pre-scheduled maintenance. This typically allows reductions in maintenance volume and therefore occupational exposure. In the field of radiation protection, specific rules can be introduced to foster the optimisation of radiation protection. In addition to the regulatory framework, utilities can develop their own radiation protection internal rules, integrating operational restrictions for the management of individual and collective doses.

ALARA management policy

The ALARA approach consists in always questioning whether the best has been done in the prevailing circumstances, and whether all that is reasonable has been done to reduce doses (ICRP 2007).

In order to spread the ALARA "way of thinking" amongst all levels of the management chain, from the company President to the worker on the floor, it is necessary to set up and structure dedicated ALARA programmes that make explicit the goals and objectives of the utility regarding optimisation of radiation protection. The responsibilities associated with the implementation of the ALARA programme should be clearly distributed among the various management levels and work specialisations. The creation of ALARA Committees or other types of specific ALARA organisations are a key element, forming "meeting points" between the main actors in ALARA implementation. This favours their involvement in the ALARA programme as well as the common elaboration of ALARA plans. Finally, plant management must be willing to support, in policy and budget, a multi-disciplinary team approach to plan, schedule, implement, and follow-up jobs.

Worker involvement and performance

ALARA cannot be achieved without worker involvement. It is the worker that is exposed, and it greatly depends on the worker himself to reduce the exposure. The involvement of workers at all levels is one of the most important aspects of an effective work management programme. By engaging the worker in the task being performed, the worker is more likely to be motivated to perform the job to the best of his/her abilities, and this will be reflected in lower job doses as well as in higher job quality. To ensure the full involvement of workers, conditions should favour the creation and continuation of such involvement. It should also implicate workers at all the stages of a job (planning, scheduling, preparation, implementation, follow-up) and assure that there is a mechanism for matching individuals and their skill levels with appropriate tasks.

It is also important to improve worker performance for ALARA implementation. This requires an appropriate level of education and training to ensure that workers possess the correct tools and competencies. Involvement of all levels is also necessary: senior and mid-level management, job foreman, shift supervisors, etc. Good communications between different levels of the hierarchy and among the different disciplines should be a management priority.

Finally, top management must also be committed to this process and favour a structure that encourages and takes into consideration the feedback of workers. Worker incentive programmes will help to improve and maintain worker motivation and involvement, and should pay for themselves in terms of savings in time, dose and costs, and in job quality.

Work planning and scheduling

The planning stage is an essential period within which to implement work management actions and optimise radiation protection.

Work activities must be carefully planned to ensure that radiological protection is optimised. Work planning and scheduling should integrate radiation protection criteria and use feed-back experience and benchmarking to ensure that the most effective approaches are implemented. Planning must recognise not only the sequence of job steps, but also their relationship and their multi-disciplinary nature. The location of job planners can be optimised by centralising all appropriate workers (planners, engineers, schedulers, etc.), thus fostering and facilitating interdisciplinary communications. In addition, the proper scheduling of jobs to co-ordinate the use of services, scaffolding, installed shielding, water shielding in pipes and tanks, etc., and the use of scale models for planning purposes (as well as training and worker orientation) contribute to the efficient use of resources.

Particular attention should also be paid to the optimisation of outage duration. Key issues in the selection of work include the use of realistic assumptions when deciding upon the necessity for performing work, the selection of only those jobs which are "necessary" to the safe and efficient running of the plant and the implementation of a tight but not rushed schedule to reduce the risk of rework.

The scheduling of jobs in relation to each other, the identification of potential work interferences and hazards in the work zone, and the identification of dose intensive jobs are critical to the optimal use of resources and job success.

Finally, the planning stage should also integrate actions for the preparation of personnel, such as pre-job briefings or mock-up training, in order to improve worker performance and reduce occupational exposure.

Work preparation

The success of work greatly depends on the quality of the preparation. Work preparation in the context of this report covers all activities considered or performed before and during a job in order to prepare the site and the work crew. It addresses factors affecting the source term, the duration of work and the number of workers exposed.

Source term removal, decontamination, reduction by shielding or continuous control are effective for achieving dose rate reductions. Various tools and equipment that support work implementation are also appropriate. It is important to take advantage of these techniques as part of work preparation since many effective methods have been developed and a great deal of experience has been accumulated. Finally, support tasks such as optimisation of the work schedule and job co-ordination in the work area are also key components of work preparation.

Work implementation

The work implementation phase refers to the actual performance of the work and to those actions taken during this time which affect or facilitate the work.

There are several areas where work management can effectively contribute to lowering dose as well as time and cost. These includes organisational aspects such as the presence of radiation protection personnel and specific procedures and technical aspects such as remote monitoring and access control systems. Efficient work process control will help to assure that the objectives set during the work planning phase are met. The reduction of transit exposure and unnecessary dose will be facilitated by providing workers with sufficient radiological, plant and job specific information. Finally, the collection of feedback information will assist in real-time work management and facilitate the preparation of future work.

Work assessment and feedback

The philosophy of work management is a continuous loop that consists of scheduling, planning, implementing, assessing, following-up, making modifications as per lessons learned and repeating the process for the next job to be undertaken, thus making the work cycle progressively optimised and in line with current technological developments. "Assessment and feedback" is the final stage of work and, at the same time, the first stage of the continuous loop.

In a generic approach, two levels of information may be necessary to provide complete feedback on work implementation: the "internal" level, which consists of an analysis of in-plant performances, and the "external" level, which will provide national and/or international data favouring the exchange of new ideas and allowing the plant to assess its position with regard to other plants of the same type.

In terms of post-job review, it is essential to have a multi-disciplinary team conduct the review and to include as much direct input from the workers, including contractors, as possible. The follow-up of recommendations and lessons learned should then, ideally, be performed by the same multi-disciplinary team which conducted the post-job review.

Normally, follow-up will lead directly into the next implementation of the operation under consideration such that a certain closure (job conception, scheduling, planning, implementation, assessment and follow-up, job modification as per lessons learned, scheduling, planning, etc.) occurs and the job becomes progressively optimised and appropriately modified to keep up with current technological developments.

The lessons-learned, both good practices and areas for improvement, should be collected in a diligent manner, and exchanged not only with the work team but also with colleagues at the plant, industry and international levels. RP managers should recognise all available information sources and use them effectively as well as share their own information and experience.

Finally, work management implementation should be audited periodically to assure that it is functioning properly.

Ensuring continuous improvement

While work management is an iterative process, it is also forwarding looking, seeking continuous improvement and continuous vigilance to ensure and maintain a high level of radiation protection. Such improvements therefore seek to incorporate, through information and experience exchange, lessons learned and ongoing technological advances to not only inform future work activities, but also in the longer term, new design, new build and new operations to ensure that doses are maintained ALARA.

In addition to experience exchange through programmes such as ISOE, there is a range of new technologies in various fields relevant to exposure reduction. These include technologies addressing source term reduction, decontamination, and mechanisation, automation and remote monitoring. The development and further application of such technologies should be considered in light of the radiation protection issues that will become important in the future, including exposure reduction in newly constructed or newly designed plants (of potentially increasing importance), large-scale modification works expected to be needed in association with aging and lifetime extension of nuclear reactors, and reactor decommissioning.

Conclusion

Safety and radiation protection are the most important factors for the safe operation of nuclear power plants. Experience in occupational radiation protection has shown that radiation protection measures should be adopted in all phases of the nuclear power plant life cycle, from design to operation to decommissioning. This not only allows source term removal or reduction as part of design, but also consideration of how exposure reduction methods or procedures can be most effectively implemented during operation.

Many methods that can be considered by all those with a role in occupational radiation protection at nuclear power plants have been described in this report. This multi-disciplinary, practical experience in work management, based on lessons drawn from many years of nuclear power plant operations, in addition to approaches that are still under development or will be realised in the future, are important elements in the optimisation of occupational radiation protection and for ensuring continuous improvement in the face of current and future challenges and opportunities.

Acknowledgement

The authors would like to thank all members of the ISOE Working Group who contributed their knowledge and experience to the elaboration of this report.

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Challenges on the radiation protection optimization of medical staff in interventional radiology and nuclear medicine: the ORAMED project

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Abstract

The development of an up-to-date radiation protection system for medical staff working with radiations requires data on field (type of radiation, energies, scattering materials...), exposure (time, activities, position of operator..) and protective devices (barriers, glasses, gloves and aprons) which are only partially available in the radiation protection routine practice. Local high exposures in interventional radiology (IR) and nuclear medicine (NM) are due to the closeness of medical staff to the direct and scattered field, in the first case, and to the direct handling of radionuclides, in the second.

Many studies in IR have shown that the doses can vary a lot, even for the same type of procedure. The routine monitoring of the extremities (hands, forearms and legs) is difficult to be performed thus only data for whole body, fingers and wrists are reported.

Moreover, eye lens dose is rarely estimated, even if there is some evidence that cataract is an increasing effect in exposed population. In NM the main topic is the skin dose, but is generally unknown which part of the hand receives highest doses and which is the dose distribution in the hand itself. Indeed, the use of unsealed sources with high

activities and beta emitters in therapeutic NM can worsen the situation for the involved personnel.

These aspects are studied within the ORAMED (Optimization of RAdiation protection of MEDical staff) project, funded by EU-EURATOM FP7, studies in which a series of European laboratories and hospitals participate. The project is subdivided in 5 work packages: extremity and eye lens dosimetry in IR and cardiology (WP1); development of practical eye lens dosimetry in IR (WP2); optimisation of the use of active personal dosemeters in IR (WP3); extremity dosimetry in NM (WP4); and training and dissemination (WP5). In the present work the state of the art of the main tasks performed in WP1, WP2 and WP4 is briefly summarized.

Introduction

The ORAMED project, funded through the FP7, was set up to optimize radiation protection in Interventional Radiology and Cardiology (IR/IC) and Nuclear Medicine (NM) focusing the attention to the dose to the extremities and eye-lens.

The published studies on the doses to the medical staff [Donadille et al. 2008, Ginjaume et al. 2008, Kim and Miller 2009, Martin 2009, Vanhavere et al. 2008, Vano et al. 2008, Zorzetto et al. 1997] report a large variability of the estimated values due to the patient scattering, the followed procedure (for the IR/IC), the kind of radionuclide (for NM), the skill of the operators and the radiation protection devices used (for both) and all of them underline the need of a radiation protection optimization. Moreover, regarding eye lens doses, there is some evidence that cataracts can be produced at lower doses level [Junk et al 2008] and the ICRP itself [ICRP 2008] is moving to this direction.

In IR/IC procedures the medical staff is likely to receive significant radiation doses to their hands and parts of their body not covered with protective equipment (as legs and forearms) because physicians are close to X-ray field. In NM practices the dose to the hands is the main issue, especially when unsealed sources are used. About eye-lens the recent evidences of cataracts at lower doses pointed out the lack of an optimized dose assessment for such organs. $H_p(3)$ is seldom used in the routine radiation protection practice: when required, it is usually deduced from $H_p(0.07)$ and $H_p(10)$ evaluation.

The ORAMED project (http://www.oramed-fp7.eu) started in the beginning of 2008 and will run for 3 years. Within the project a coordinated measurement program in selected European hospitals is presently on-going. The measurement campaign is accompanied by a series of numerical simulations of the most representative workplaces/procedures to determine the main parameters that influence the extremity and eye lens doses. The project is structured in 5 work packages (WPs) all devoted to investigate the medical staff doses in IR/IC and NM. In WP2 $H_p(3)$ operational quantity has been investigated and a suitable dosemeter responding in terms of $H_p(3)$ has been produced. In the present paper the state of the art of WP1 (extremity and eye lens dosimetry in IR/IC), WP2 and WP4 (extremity dosimetry in NM) is summarized.

Material and methods

WP1- Extremity and eye lens dosimetry in IR/IC

The objective of WP1 is to obtain a set of standardized data on doses for staff in interventional radiology and cardiology with the aim of optimizing medical staff radiation protection. The coordinated measurement program in some selected major European hospitals, established in the first year of the project, is still in progress. The procedures to be followed were chosen on basis of a retrospective study from hospital data on the frequency of procedures and the respective KAP (kerma air product) values. The final list includes 3 cardiac procedures- cardiac angiographies (CA) and angioplasties (PTCA), radiofrequency ablations (RFA), pacemaker implantations (PM) - and 5 general interventional diagnostic and therapeutic examinations - angiographies (DSA) and angioplasties (PTA) of the lower limbs (LL), of the carotids (C) and renal (R), embolisations and endoscopic retrograde cholangiopancreatographies (ERCP).

The measurements have been performed according to a protocol supplied to each hospital allowing the homogenization of the collected data. Parameters such as the KAP values, the radiation protection equipment used by the staff and the staff position during the practice have been noted down in order to correlate them with the evaluated doses.

TL dosemeters (LiF:Mg,Cu,P) were used for the measurements. Eight TLDs were sealed in small plastic bags and taped on the parts of the body to be monitored: 2 for the ring fingers, 2 for the wrists, 2 for the unshielded part of the legs (about 5 cm below the lead apron) 1 between the eyes and 1 near the left/right eye, depending if the tube is on the left/right side of the doctor respectively. The measurements during preliminary tests showed that these positions are the most exposed. A previous intercomparison exercise, in which TLDs were irradiated to ^{137}Cs and X-ray beams on the ISO slab phantom, assured that all partners could evaluate comparable doses in terms of $H_p(0.07)$.

Due to the complexity of establishing the main parameters influencing the doses in IR/IC, it was decided to perform a numerical analysis using anthropomorphic models with MCNPX Monte Carlo transport code [Pelowitz 2005]. A first simulation intercomparison was performed in order to work on a common basis for the numerical campaign. The input simulates a typical irradiation scenario: the anthropomorphic MIRD-ORNL [Snyder et al. 2008] model was modified extracting arms from the body structure and arranging forearms and hands in a way that is more similar to that expected for the cardiologist during the insertion of the catheter. A second model, representing the patient, is laying in front the physician that stands at the patient’s femur. Eyes, a thyroid collar, a lead apron of 0.5 mm thickness and the Image Intensifier tube have been added in order to better simulate the real situation (Figure 1). A simulation protocol has been established after a preliminary test that demonstrated the need for variance reduction techniques - MCNPX DXTRAN spheres [Briesmeister 2000]- especially for the small tallies (hands and eyes).

Since the number of the simulations that was finally agreed was too high it was decided to perform the whole analysis (called sensitivity analysis) on three simplified phantoms, a cylinder, (for IR practice in the head) and two phantoms representing the thorax and the whip (for the different IR/IC procedures of those anatomical regions) and to adopt detailed models only to check some parameters (as shielding) or for particular tests.

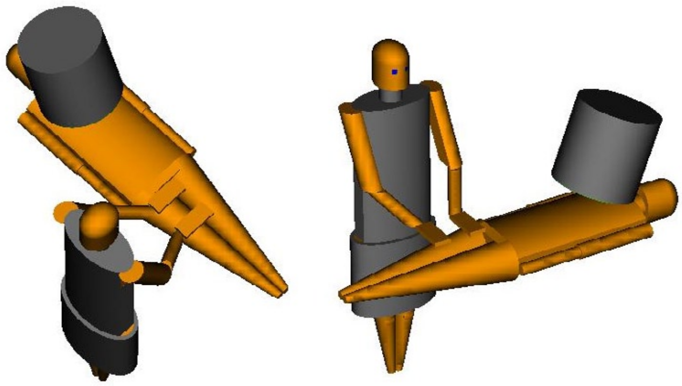

Fig. 1. The modified ORNL-MIRD type models employed in the IR/IC simulation.

WP2- Eye lens dosimetry and $H_p(3)$ dosemeter development

The objective of WP2 is to investigate the theoretical framework within which the operational quantity $H_p(3)$ is defined and the posibility to develop a dosemeter responding in terms of the operational quantity itself.

Until now $H_p(3)$ conversion coefficients were not reported in the official recommendations and the only available data were calculated by GSF (now Helmoltz Institute) for a 30 x 30 x 15 cm^3 4 element ICRU tissue-equivalent slab phantom [Till et al 1995]. The operational quantity should be defined in a phantom able to reproduce the interaction and scattering properties of the part of the body considered. This is the reason why the 4 element ICRU tissue-equivalent slab phantom (representing the thorax) is not well suited to represent the human head. For instance, a preliminary study allowed to investigate if $H_p(3)$ could be defined in a cylindrical phantom of 20 cm diameter and 20 cm height. The shape and the dimensions of this model were defined on the basis of the morphology of the head of the ORNL-MIRD anthropomorphic phantom, which was created according to the characteristics of the standard man expressed by ICRP 23 [ICRP 1975] and ICRP 89 [ICRP 2003]. The calculated $H_p(3)/K_{AIR}$ for slab and cylindrical phantoms, for monoenergetic photons and different angles were compared to the reference ICRP limiting quantity eye-lens equivalent dose $H_{T(eye\text{-}lens)}$ [ICRU 1998].

A cylindrical PMMA calibration phantom, with the same dimensions of the theoretical model, filled with water, was developed for the following dosemeter type-testing (Figure 2). A series of MCNP simulations was performed in order to determine the backscatter properties of the proposed model [Mariotti and Gualdrini 2009]. Mono-energetic and ISO narrow beam spectrum series at different incident angles and distances from the phantom lateral surface were considered. The Monte Carlo simulations were validated through a series of measurements of the kerma in air with and without the corresponding plastic phantom using a small volume ionization chamber. The characterized calibration phantom was used in the rest of the work.

Starting from these studies the dosemeter development was planned. A series of Monte Carlo simulations was employed to determine which of the available plastic materials was more suitable for the expected 3 mm soft tissue attenuation related to the $H_p(3)$ definition. The chosen material was used to provide the encapsulation of the TL

MCP-N (LiF:Mg,Cu,P) chip. The study of the complete dosemeter (TL + plastic encapsulation) response with respect to the imparted $H_p(3)/K_{AIR}$ was carried out through a series of Monte Carlo simulations at various incident angles. The developed prototype is being tested at CEA laboratories at ^{137}Cs and standard RQR X-ray beams on the cylindrical phantom. The characterized dosemeters will be used in the selected hospitals in order to evaluate the eye-lens doses for IR/IC procedures.

Fig. 2. The cylindrical calibration phantom for eye-lens operational quantity dosemeter testing.

WP4- Extremity dosimetry in NM

The objective of WP4 is to evaluate extremity doses and dose distributions across the hands of the medical staff working in nuclear medicine departments. To achieve this, an extensive measurement and simulation program has been started.

A measuring working plan was established following a defined protocol: the labelling and administration of radiopharmaceuticals for diagnostics (^{99m}Tc and ^{18}F) and therapy procedures (Zevalin and DOTATOC labelled with ^{90}Y) were taken into account (other radionuclides as ^{32}P, ^{177}Lu or ^{153}Sm were also considered). The procedures were selected according to their frequency in the collaborative hospitals.

To measure the skin dose across the hands, special gloves were designed with high sensitivity thermoluminescent dosemeters (TLD) specific to beta and gamma radiation placed at a minimum of 11 different positions on each hand (Figure 3). The TLDs inserted in the plastic envelops could reproduce adequately the quantity $H_p(0.07)$. For each procedure a minimum number of measurements was required. Each procedure had to be performed by two operators and in two hospitals assuring an adequate evaluation of the intrinsic variability in the expected doses.

An initial comparison of the different TLDs used by the WP4 participants was performed with ^{137}Cs and ^{85}Kr sources in order to establish a common basis for the measurement campaign in the different nuclear medicine departments.

In order to check the different parameters contributing to the dose and to the dose reduction to the operators (position with respect source, adopted shieldings etc..), it was decided to perform a series of numerical simulations (called sensitivity analysis) reproducing the most common scenarios encountered during the measurements. 2 different scenarios for injecting and 3 different ones for labelling have been selected: handling of a syringe, handling of a vial directly with or without forceps. Hand phantoms made of paraffin have been produced by SMU and scanned using a CT. Voxel models of the phantoms were created by IRSN and employed as input for the sensitivity analysis (Figure 4). The numerical models were validated through a series of measurements done using the paraffin phantom in laboratory in controllable conditions. The sensitivity analysis is ongoing.

Fig. 3. The operator's gloves equipped with dosemeters for the dosimetric study in WP4.

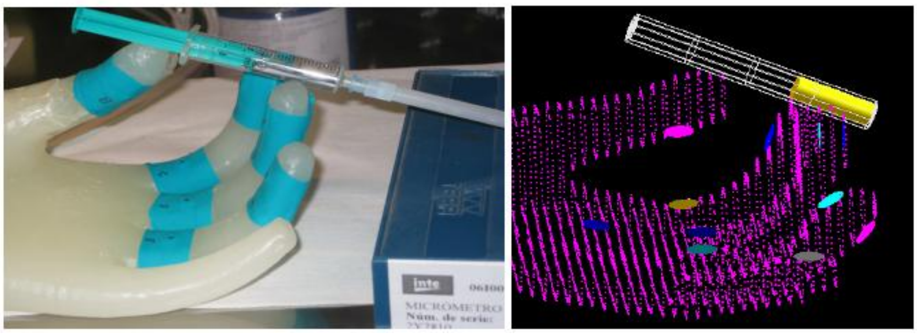

Fig. 4. The paraffin hand used in validation and its voxel model representation.

Results

The measurements and simulation campaigns are in progress; therefore the presented results are only preliminary. Here only few examples of the analysis are reported, a complete investigation of the outcomes will be presented at the final ORAMED workshop.

WP1

For the measurements part, up to now 797 interventional procedures (377 cardiac, 277 general angiographies and 143 Endoscopic Retrograde CholangioPancreatographies, ERCP) in 33 European hospitals have been performed.

The measured KAP values vary from 0.46 Gycm^2, recorded in PM procedure, to 942 Gycm^2 in embolisation. Generally high KAP values are encountered when large number of images/frames is acquired. The KAP values present high distribution even within the same type of procedure, which shows the complexity of the procedure, the variability in the techniques and the influence of the skill of the physicians. The median doses for general angiography procedures are usually higher than for cardiac and ERCP and the left side of the operator is the more exposed to the scattered radiation from the patient (Figure 5).

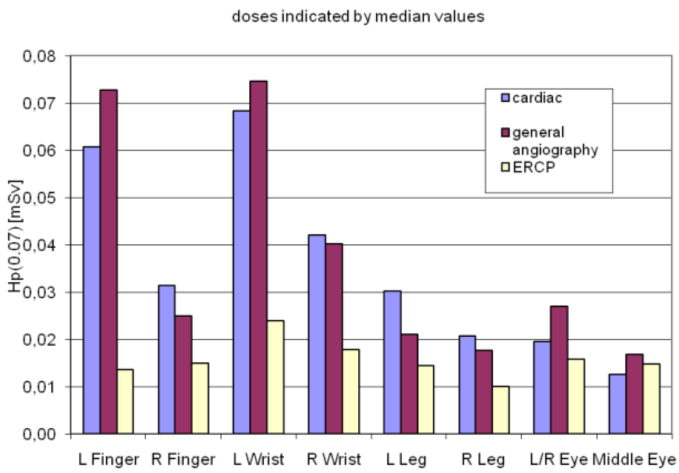

Fig. 5. Mean Hp(0.07) values in mSv for cardiac, genaral angiography and ERCP procedures.

The maximum dose that was recorded was at the left finger in a CA and PTCA procedure (2.08 mSv, KAP=380 Gycm^2). The maximum eye dose (1.28 mSv) was recorded in the embolisation with the highest KAP value.

For CA and PTCA procedures, the fingers received slightly higher dose than the wrists, and the legs higher than the eyes. For the RFA procedures, the finger and wrist doses are higher than the leg and the eye ones as in the CA and PTCA procedures.

For the PM interventions the $H_p(0.07)$ values are generally low due to the short fluoroscopy time and the absence of image acquisitions, although values up to 0.4 mSv/procedure were recorded in the finger region. In those cases the hands were very close or sometimes even inside the beam. In PM implantations the operator's position can be closer to the patient compared to the other interventional procedures so it is not easy to use the protective equipment of the table and ceiling.

In DSA, PTA of lower limbs and carotids the finger and wrist doses are higher than the leg ones. In the embolisations, the wrist doses are higher than the fingers which are in agreement with Whitby and Martin [2005]. Finally the doses recorded in the ERCP interventions are small varying from 0 to 0.03 mSv.

About the sensitivity analysis the tube voltage was changed from 60 to 110 kVp and filtration from 3 to 6 mm Al and from 0 to 0.9 mm Cu. Calculations were performed for different beam projections: PA, LAO and RAO (30°, 60°, 90°), caudal and cranial (20° and 40°) combinations of them. A total of 3300 calculations for the simplified study and at least 220 detailed ones were performed. For all projections, the results showed that doses received by the operator decreased with increasing tube voltage and filtration. When the slab phantom was used the maximum doses for all tallies were observed for the LAO90 beam projection and the minimum ones for the RAO30. In the case of the cylindrical head phantom the maximum doses for all tallies were observed for the CRAN40 beam projection and the minimum ones for the RAO90 and RAO30. As expect the increasing beam aperture increases also the doses in particular for the wrists and the hands, for example, for the slab phantom the doses to the hands and wrists are reduced more than 3 times when the field size changes from 40 to 20 cm in diameter. The effects of the tube voltage and of the filtrations on the doses depend strongly on the irradiated part of the body and the beam projection.

WP2

The simulations demonstrated the validity of the performed analysis on $H_p(3)$ definition. In figure 6, $H_p(3)/K_{AIR}$ calculated for monoenergetic photons employing the cylindrical model and the 30 × 30 × 13 cm^3 4-element ICRU tissue-equivalent material are compared with the $H_{T(eye\text{-}lens)}/K_{AIR}$. As it can be seen the operational quantity defined in the cylindrical phantom is able to reproduce better the ICRP limiting quantity; a complete analysis can be found in an ENEA report [Mariotti and Gualdrini 2009]. The conversion coefficients for $H_p(3)$ were calculated in Kerma approximation, assuming the energy of the secondary charged particles deposited locally. A parallel investigation considering the electron transport was performed by CEA, using the Penelope code [Salvat et al. 2006] and demonstrated, as expected, the overestimation of the conversion coefficients calculated in the kerma approximation at 3 mm depth in soft tissue for energies above 1 MeV [Daures et al. 2009].

The dosemeter development was based on a series of Monte Carlo simulations. In Figure 7 the response of the complete (capsule + TL chip) RADCARD™ dosemeter, normalized per imparted $H_p(3)$, are presented at normal incidence angle for monoenergetic photons for different encapsulation material.

WP4

At the moment, 635 measurements coming from 32 nuclear medicine departments across Europe have been introduced in the database. Results are from diagnostic and therapeutic procedures and include both phases: preparation and injection.

To compare measurements from different departments, measured doses were normalized to the manipulated activity. A large spread of results, due to the influence of many factors, such as the type of protections used and the experience of the technicians is observed. The complete statistical analysis of the results is under progress.

From the 11 measuring points across the hand, it has been observed that the most exposed positions are the tip of the index finger and the thumb, usually of the non dominant hand.

The data analysis is ongoing, the preliminary comparison showed that for ^{99m}Tc administration the maximum dose is 1.5 (mSv/GBq) at index tip. This is the lowest registered maximum dose being 3.7 (mSv/GBq) and 11.2 (mSv/GBq) the evaluated maximum index tip doses for the ^{18}F and ^{90}Y respectively. These values change for the labeling case: 2.0 (mSv/GBq) for ^{99m}Tc 4.4 (mSv/GBq) ^{18}F and 32.0 (mSv/GBq) for ^{90}Y.

Usually labelling delivers higher doses than the administration of the radiopharmaceutical because the activities manipulated for labelling are higher than those manipulated for the administration of the radiopharmaceutical and some of the labelling steps are performed with an unshielded source whilst the administration is usually performed with a shielded syringe.

The impact of placing the routine monitoring dosemeter at a different position than the one corresponding to the maximal hand dose has been estimated by computing dose ratios between the position corresponding to the maximum in the hand (usually the index tip) and the position normally used for routine monitoring (usually the base of the index or the ring finger). The calculated ratios vary from 1 to 7 depending on the radionuclide and the procedure.

About the simulations the established protocol has the scope of evaluating changes in extremity doses varying different parameters. The outcome of these calculations will be used to define practical guidelines. To verify the consistency between measurements and simulations, several measurements were done in laboratory conditions: in Figure 8 ^{18}F injection with unshielded syringe measurements on paraffin hand are compared with the same data obtained with the computational voxel model. The agreement between the data is satisfactory. The sensitivity analysis is in progress.

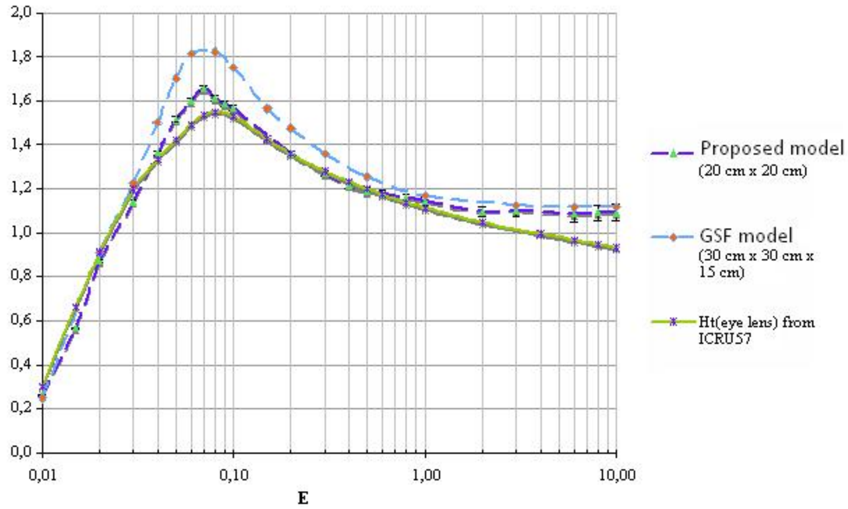

Fig. 6. $H_p(3)$/KAIR in slab (GSF) and cylindrical (proposed model) phantoms compared with $H_{T eye-lens}$ for normal incident monoenergetic photon beams.

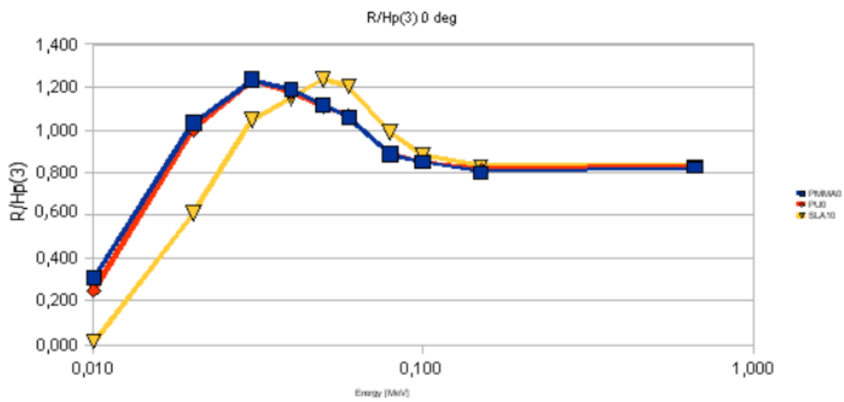

Fig. 7. Study of the response of the RADCARD dosemeter prototype - R/H_p(3) at normal incidence: comparison between PMMA, PU and SLA capsules.

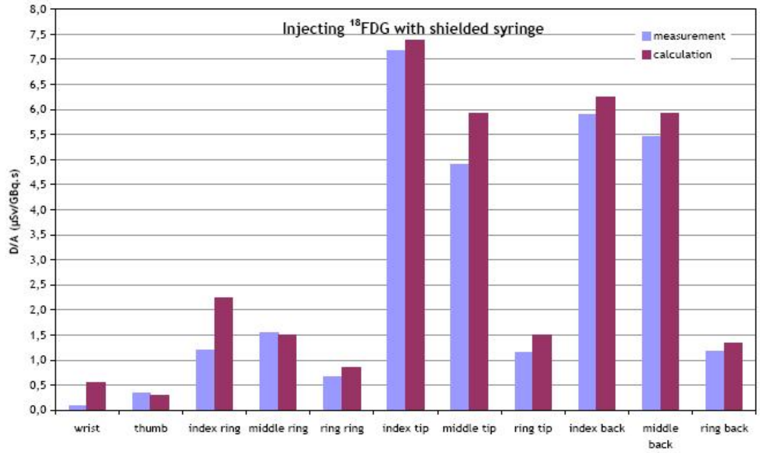

Fig. 8. Validation of the numerical simualtion: comparison between the results obtained with measurements in laboratory and the simualtion of FDG injection with unshielded syringe.

Conclusions

The ORAMED project aims at developing methodologies for better assessing and reducing exposures to medical staff. In the present paper the activities related to the working groups 1,2 and 4 have been summarized.

WP-1 - Measurement and calculation of extremity and eye lens doses in interventional radiology and cardiology: a set of standardized data on doses for staff in interventional radiology and cardiology have been collected. A series of recommended radiation protection measures to optimize staff protection will be prepared on the basis of the outcomes of the wide numerical simulation campaign that is ongoing.

WP-2 - Development of practical eye lens dosimetry in interventional radiology: the operational quantity has been investigated, producing a new set of conversion coefficients for $Hp(3)/K_{AIR}$, a new calibration phantom was adopted and the first prototype of the dosemeter, responding in terms of $H_p(3)$, was developed. The dosemters will be supplied to IR/IC medical staff of a selected number of hospitals in order to assess eye-lens doses for radiation protection optimization.

WP-4 – the optimization of the extremity dosimetry of medical staff in nuclear medicine: a set of doses across the hand for administration and labelling of radiopharmaceuticals have been registered for a selected number of European hospitals. The data allowed to study the maximum dose and to determine the relationship between that maximum and the dose registered in the usual position where the routine dosemeter are worn. The simulations are still in progress, and they will be used to determine the effect of parameters influencing the radiation protection in those medical practices.

ORAMED project final workshop

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Increased extremity doses for staff in the preparation and administration of beta-emitters and PET nuclides in nuclear medicine

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Abstract
Due to the continuous increase in extremity doses in nuclear medicine departments, the Swiss Federal Office of Public Health (SFOPH), in its capacity as the supervisory and regulatory authority dealing with ionising radiation, has investigated the reasons for this and introduced measures aimed at their reduction. Since the mid-90s, the SFOPH has observed a continuous increase in extremity doses of persons occupationally exposed to radiation in nuclear medicine departments. Whereas only a few persons in the 50 facilities had accumulated an extremity dose greater than 25mSv/year in 1996, ten years later there were almost 10 times more. This development was all the more worrying because it is assumed that the extremity dosimeters used actually record only a fraction of the extremity dose effectively accumulated, especially when dealing with beta-emitters. In facilities with high patient throughput and frequent therapeutic uses, it is therefore possible that the dose limit (500mSv/y) is occasionally exceeded. The SFOPH looked into this radiation protection problem and since 2006 has undertaken audits in the relevant nuclear medicine departments to determine the reasons for the increased extremity doses, to optimise doses as far as possible and to order corresponding actions. It was established that there is considerable optimisation potential not only in the preparation of radiopharmaceuticals but also in their application, such that a marked reduction of extremity doses is possible. However, this partly depends on investment in systems, which automate dose-intensive manipulations, such as the loading and activity determination of injection syringes or even the injection into patients. When staff are aware that even a brief manipulation with unscreened therapeutic doses could result in high extremity doses, they generally use the shielding aids more conscientiously and adopt work plans and procedures that optimise radiation protection.

Introduction
The Swiss Federal Office of Public Health (SFOPH) is responsible for enforcing the radiological protection legislation [1] in Switzerland. In its role as the supervisory and licensing authority in radiological protection, the SFOPH monitors the radiation doses of persons occupationally exposed to radiation, of patients and the general population. Over the last 10 years the SFOPH has observed a continuous increase in extremity

doses for persons occupationally exposed to radiation in the approximately 50 Swiss nuclear medicine departments monitored. Fig. 1 shows the trend in extremity doses (greater than 25 mSv and greater then 50 mSv per year) between 1989 and 2006. Four years ago this led the SFOPH to focus on this topic in order to determine the causes and to initiate possible measures for the reduction of doses.

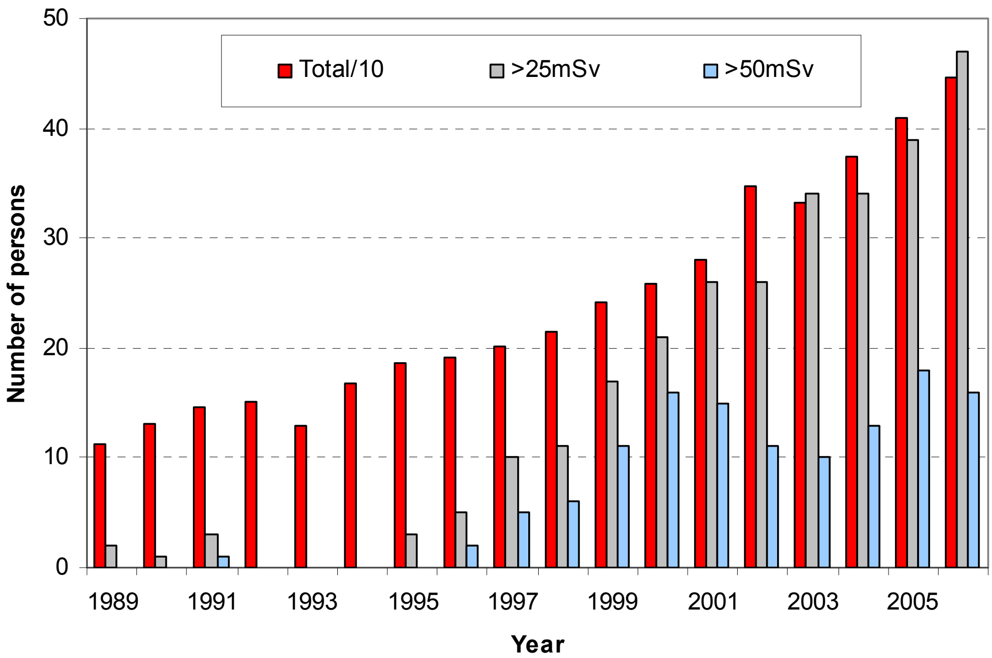

Fig. 1. Trend of extremity doses in nuclear medical facilities from 1989 – 2006.

The main raison for this trend is the growing use of new radiopharmaceutical products for therapeutic use (see Fig. 2) and the increase of departments that carry out PET examinations. Ten years ago there were 5 PET units in operation, nowadays there are already more than 20. Consequently, the use of F-18 has also correspondingly increased.

In a first step all the services concerned were audited in order to determine the current radiation protection practice and in which areas the radiation protection could be further improved. The findings from these audits are intended to be used in a second step to elaborate and publish recommendations for radiation protection and training programmes for the personnel concerned.

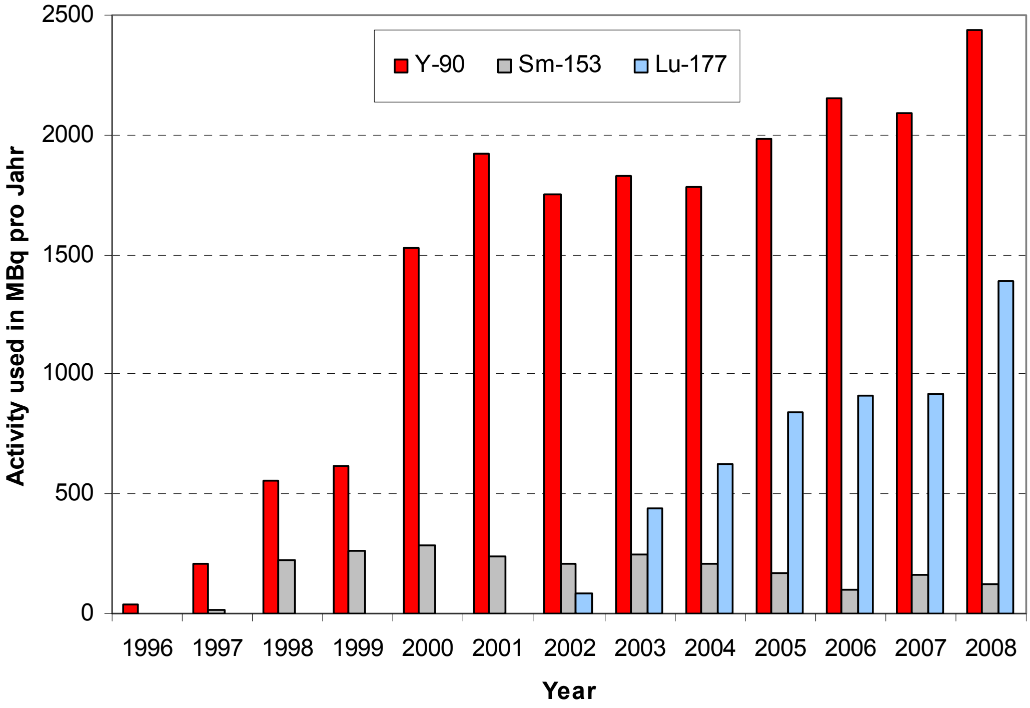

Fig. 2. Use of beta emitters for therapeutic purposes 1996-2008.

Material and methods

Nuclear medicine departments to be audited were selected on the basis of the use of F-18 and beta emitters (Y-90, Lu-177, Re-186, Sm-153, etc.). Data for increased extremity doses (greater than 10mSv/year) were consulted as a further criterion using the Swiss Central Dose Register. A total of 30 departments that handle beta emitters and/or PET nuclides were identified. The remaining 20 nuclear medical institutes work predominantly with Tc-99m, where increased extremity doses are rarely observed. In the audit, the personnel who are responsible for the organisation and monitoring of the radiation protection in the facility were interviewed. In addition, specialist staff (physicians, radiographers, chemists, laboratory technicians), who carry out preparation work in the isotope laboratory and administer radiopharmaceuticals to patients, were consulted as needed. In the planning of the questions for the audit, emphasis was placed on the 4 following topics:

1 Organisation of Radioprotection

Monitoring of radiation doses by means of external dosimetry (extremity dosimeters and whole body dosimeters) is an important instrument for reviewing working procedures. However, this is only valid if the dosimeters are worn consistently and in the right position. Investigations in a service with a high volume of Y-90 have shown that the highest doses are accumulated on the left palm of the hand (for right handed individuals). These results were also confirmed in a study [3] carried out under contract for the SFOPH (Fig. 3) by the Lausanne University Institute of Applied Radiation Physics (IRA).

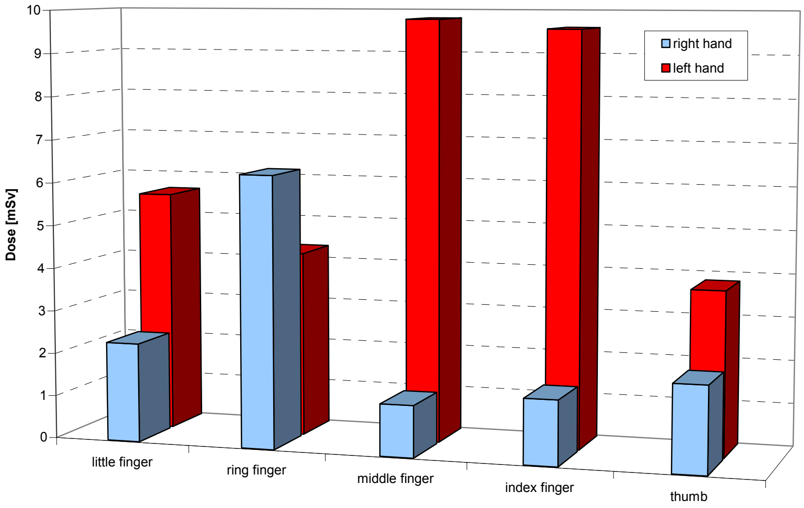

Fig. 3 Doses to fingers of the hands in the course of a preparation of a Zevalin® application (right handed individual).

The radiation protection expert in each service has the task of quickly clarifying increased or even regular doses, and to demonstrate how they can be avoided or reduced. If unusually high extremity doses are recorded in a department, then the persons in question must be quickly notified in order to clarify how this could have occurred. An important aspect is also the organisation of the distribution of the radiation doses. In this regard, unavoidable dose-intensive work should be distributed among as many staff as possible. Having said that, it must also be ensured, that the staff have adequate working routines so as to quickly and safely carry out these procedures.

2 Radiation protection training and continuous professional development

Persons occupationally exposed to radiation must have the relevant technical knowledge and competence, i.e. must at least be aware of radiation protection rules and dose-optimised work practices. In order to maintain this technical knowledge, the person who is responsible for radiation protection in the department must regularly organise continuing training for themselves and their colleagues. Prior to introducing new applications, all working steps with radioactive substances have to be planned and practiced from the viewpoint of reduced dose work techniques. Personnel who mainly work with conventional radionuclides (Tc-99m) used in nuclear medicine must be instructed that when manipulating beta emitters and F-18 there is an increased risk of accumulating high hand doses.

3 Equipment

Isotope laboratories used for the preparation of radiopharmaceuticals and administration rooms must be equipped in such a way that radiation sources are shielded as much as possible in order to reduce the dose to the personnel. Suitable containers must be available for the storage of radiopharmaceuticals and radioactive waste ([4] Ordinance for the use of unsealed radioactive substances). When manipulating radiation sources, short distances and easy access must enable fast and safe work.

4 Aids and auxiliaries for radiation protection

When using beta emitters, suitable Plexiglas shielding must be present and consistently employed (Fig. 4 and 5). When using F-18 for PET diagnostics, thick-walled lead or tungsten shielding has to be used in order to reduce the dose rates (Fig. 6). Vials containing radioactive substances or syringes, into which a radiopharmaceutical is aspirated, must be consistently shielded when manipulated. Manipulations lasting a few seconds without shielding can lead to doses of several mSv to the hand. Direct contact with unshielded syringes and vials should be avoided at all costs; indeed, should the situation arise, then remote handling devices such as gripping arms and tweezers should be employed. Because even small splashes can cause considerable skin doses, it is vital to wear gloves for all manipulations. As according to [2] latex gloves provide only inadequate protection from contamination, vinyl or nitrile gloves should be worn when handling beta emitters with high specific activity. With high patient turnover and extensive nuclide use, increased extremity doses can generally only be avoided if suitable automatic dispensers are utilised for filling syringes and administering the radiopharmaceuticals.

Fig. 4. Vial Plexiglas shielding for beta emitters.

Fig. 5. Syringe shielding for beta emitters.

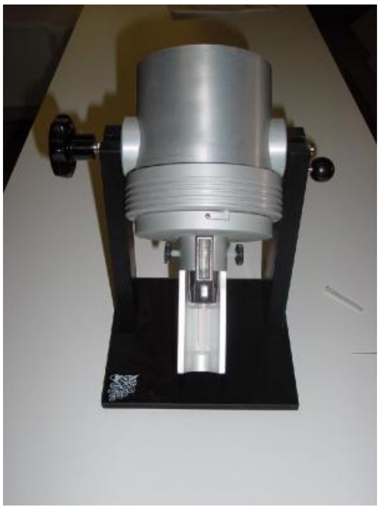

Fig. 6. Lead shielding for F-18.

Results

The radiation protection audits in the nuclear medicine departments generally aroused a high interest and were considered to be useful. The persons responsible for radiation protection and their colleagues are well aware that repeated use of beta emitters and PET nuclides leads to increased accumulated doses and especially extremity doses. Moreover, specific measurements of hand doses on the fingertips have demonstrated that the doses recorded by ring dosimeters, especially when working with beta emitters, do not reflect the maximum accumulated radiation doses. It can be assumed that the effective maximum hand doses are higher by a factor of 3 to 5. When comparing the monthly accumulated radiation doses with the therapies carried out, it could be further established that high values often correlate with individual beta therapies. An analysis and review of the radiation protection situation in the various departments yielded primarily the following two findings:

1 Organisation, training and continuous education

Audits in the different services have shown that the personnel and the responsible persons are well aware of the problem of increased extremity doses. However, training in this field must be further improved. The working procedures are partially configured from experience of handling conventional nuclides such as Tc-99m and the work places are correspondingly designed. There is a lack of awareness that unshielded beta emitters, even when manipulated for a few seconds, can already induce extremity doses of several mSv. Contrary to widespread belief, the time factor plays a minor role compared with shielding (a 10mm Plexiglas shielding reduces the dose rates by a factor of 1000). Discipline in regard to wearing extremity dosimeters is generally good. It is not universally known on which part of the hand the highest doses are to be expected, and hence where the dosimeters should be worn. The “radiation protection culture” in the services is generally well understood but can still be improved. Repeated high doses for staff are partially taken for granted and consistent optimisation is not sought after. In contrast, dose-intensive work is generally distributed as well as possible among the available staff, taking into consideration adequate training and procedures.

2 Facilities and resources

In nuclear medicine services where beta emitters are only used occasionally, the work places are often not set up in such a way as to enable the handling of radioactivity over the shortest possible distances. Moreover, ready-for-use radiopharmaceuticals are often directly syringed out of the suppliers’ shielded containers, which do not ensure adequate shielding above the top. Generally, remote handling devices such as tongs and tweezers are mainly employed when handling vials and syringes (Fig. 7). However, the temptation to intervene directly with ones hands for tricky work is great and often occurs even without thinking. Attaching and removing needle tips and three-way stopcocks (Fig. 8) is often carried out by hand. Even with attached shielding, needle tips are insufficiently shielded. Although there are some devices for this, their use is mostly felt as too laborious.

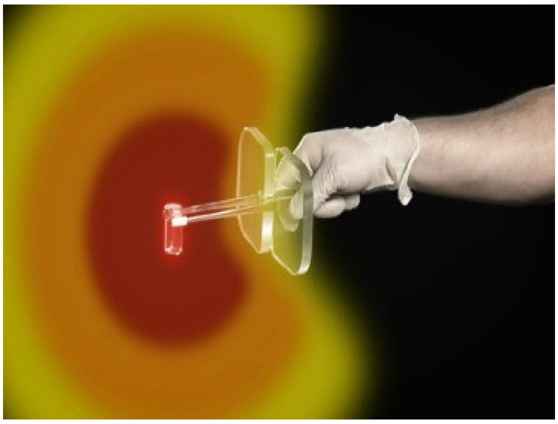

Fig. 7. Tweezers with Plexiglas beta shielding.

Fig. 8 Attaching the tip to a 3-way stopcock.

Prior to the administration of the radiopharmaceutical to the patient, the charged activity in the syringe has to be measured and verified in an activimeter [4]. As the syringe for this has to be removed from the shielding and this procedure is often carried out several times for an accurate dosage, this manipulation is dose-intensive. For this reason particular care must be taken to ensure that the arrangement of the workspace and the location of the activimeter allows fast working. If the activimeter is suitably calibrated, then the measurement of the activity of F-18 can also be carried out with the shielding in place. In some facilities syringes can also be filled automatically. However, practice-proven systems are expensive and also require a lot of space. For this reason devices of this type are primarily installed in services where the purchase and installation can be financially justified by a high patient turnover. Recently however, simpler, cheaper systems have also become available on the market. Nevertheless they still have to be proven in practice before they can find widespread use. A facility that has high turnovers of Y-90 and Lu-177 and consequently also carries out costly labelling, employs radiation protective gloves for some manipulations. Measurements have demonstrated that with these lead gloves - which are also used in interventional X-ray diagnostics - the hand doses can be reduced by a factor of 3 to 4. The methods of administration, depending on the application, are often very different in various institutes even within the same application. This presents a considerable potential for optimisation in this work procedure. If long-term administrations are also carried out in some hospitals with manual injections, they are carried out elsewhere automatically with syringe perfusions. A newly developed PET infusion system, developed in one of the services, actually allows a fully automatic dosing and injection. Following the audit of the SFOPH, binding measures for optimising radiation protection were agreed with the responsible persons. In addition to organisational aspects, such as intensifying the internal training and continuous education or verifying regularly how dosimeters are worn, these measures also include in-depth investigations by the service for the potential optimisation of specific dose-intensive work procedures and appropriate possibilities for assessment are submitted to the supervisory authority.

Discussion

With the SFOPH conducting audits, awareness about increased extremity doses was raised. Accordingly, tangible improvements were successfully introduced in the majority of facilities. The success of these efforts is also shown by the fact that between 2007, when the audits began, and 2009 the extremity doses in nuclear medical services have decreased by 23% (annual doses greater than 25 mSv) and 31% (annual doses greater than 50 mSv). An evaluation of whether individual working procedures can be optimised regarding radiation protection of the extremities, can however not be exhaustively made during an audit. For this reason it is intended to make a detailed examination of the operational procedures in various services, as has also been recommended in a statement by the Federal Commission on Radiological Protection and Monitoring of Radioactivity (CRP) to the SFOPH [6]. In order to detect and record possible errors and their effects, the maximum finger doses should be measured with TLDs on the finger tips during critical manipulations (Fig. 9). The work procedures could also be filmed in order to subsequently analyse the causes of the accumulated doses. Automated dosing and injection systems offer a high capability for reducing the hand doses. Experiments in a PET Centre with a fully automatic doser/injector have shown that a dose reduction by a factor of greater than 10 is possible [5].

However, proven systems are rare and expensive and are only available in institutes that have high numbers of patients and corresponding financial means. More cost efficient systems have still to be tested in depth and must provide evidence that the quality of the dosing or the injection is at least as good as in a manual procedure.

Fig. 9. TLD for the determination of the maximum dose to extremities.

Conclusions

Developments in nuclear medicine show that applications involving PET nuclides and beta emitters will probably increase further. For this reason, additional measures must be provided for the training and continuing education of the staff in order to avoid any further increase in extremity doses. The SFOPH will therefore produce and publish a continuing education DVD, which will illustrate optimised radiation work procedures when handling beta emitters and PET nuclides. Furthermore, a recommendation for radiation protection is intended to be drawn up for this topic and practical continuing education courses will be offered. In addition, the question of whether automatic systems should be made mandatory for dose-intensive work procedures will be examined.

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Morphological dependence of lung counting efficiency for female workers

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Abstract

In vivo lung monitoring of female workers is routinely performed using calibration coefficients calculated with a male thoracic mannequin since no female model exists. More appropriate calibration coefficients can be obtained using numerical models. In this work, flexible 3D Mesh and NURBS (Non Uniform Rational B-Splines) geometries were considered to design representative female thorax. Lung counting efficiencies were simulated for typical germanium detectors and the parameters of their morphological dependence were defined.

A library of 24 different 3D female models was created representing the most common female breasts with various cup sizes (A to F) and chest girths (85 to 120). Monte Carlo simulations were then achieved to investigate the chest girth and cup size effects on the counting efficiency. It was shown that for the 59.54 keV Am-241 gamma ray, the counting efficiency decreases of about 15% between the 85A and the 85B phantoms. Moreover, a 55 fold decrease in efficiency was observed at 22 keV between the 120C and the 85C phantoms.

An equation was developed, involving simple physical assumptions, which defines any counting efficiency as a function of chest girth, cup size and a reference efficiency curve. Morphology-dependent parameters were calculated in order to estimate the efficiency curve of any female subject (any breast size and morphology) if a reference efficiency is provided. Furthermore, the developed equation was able to describe the relation between the calculated female efficiency and the Livermore calibration data. It was found that the simulated 85A efficiency curve is in close agreement with the calibration measurements performed with the Livermore and its first extra-thoracic plate. Since this agreement depends notably on the chosen counting position for females, it was also shown how to transform calibration measurements performed with other thoracic plates.

This work enables a better assessment of the *in vivo* calibration coefficients improving the female workers monitoring.

Introduction

After an intake of radionuclides, *in vivo* counting and bioassays are the only two available techniques enabling the assessment of the retained activity (ICRP 1988). To correctly estimate the contamination using *in vivo* spectrometry, counting systems accurate calibration is required. The latter is typically done using anthropomorphic physical phantoms (ICRU 2003). However, there is no female torso phantom even though the female morphology can significantly influence the count. Hence, Monte Carlo (MC) calculations and numerical models of the human body are used to obtain realistic calibration factors, compensating for the absence or poor realism of physical phantoms (de Carlan *et al.* 2007, Hunt *et al.* 1998, Kramer *et al.* 2009)

The first numerical models (mathematical phantoms) represented the human complex anatomy by simple equations. Later, the voxelized phantoms were introduced in radiation protection to offer a more realistic representation than the mathematical phantoms (Zankl *et al*. 1988) and the ICRP has recently released reference male and female voxel models (ICRP 2009). Recent developments in 3D formats and associated tools enabled the design of more flexible anthropomorphic 3D models. Indeed, Mesh or NURBS formats can be easily manipulated and transformed to obtain various representative postures or morphologies (Xu *et al.* 2007, Lee *et al.* 2007).

Here we address the question of the morphological dependence of counting efficiency curves for *in vivo* lung monitoring of female workers. For this purpose a library of 24 female torsos, representing the most common breast sizes and morphologies, was designed using Mesh and NURBS formats. MC calculations of the counting efficiencies of a typical Germanium counting system were achieved for all phantoms. Next, a simple analytic formula was derived to describe the morphology effect on counting efficiency. Finally, a practical example is given to show how an experimental reference calibration curve, obtained with the Livermore phantom, can be transformed to provide efficiency corrections for most common breast morphologies.

Material and methods

This work first presents the Monte Carlo calculations carried out with the designed phantoms. Then, it focuses on the parameterization of the morphological dependency of the counting efficiency. The definition of an analytic formula is given to describe the relation between efficiency curves obtained for each female phantom. This equation is validated using Monte Carlo simulated data and tested with experimental efficiencies obtained with the Livermore phantom.

1. Simulation of in vivo lung counting measurements

A flexible female library of 24 torso models, representing most common breast morphologies, was created starting from the ICRP Adult Female Reference Computational Phantom (ICRP AF-RCP) (ICRP 2009). When creating the female thoracic 3D models, breast cup size variation was achieved by adipose tissue adding according to plastic surgery recommendations (Turner and Dujon 2005). Realistic variation of the chest girth (from 85 to 120), internal organs volumes, compositions and densities were also achieved. For the 85 and 90 chest girth phantoms, internal organs volumes were the same. For the 100, 110 and 120 chest girth phantoms, these volumes were modified taking into account height dependant correlations (Clairand *et al.* 2000).

Counting efficiency variation with energy was calculated for the 24 different models to identify the morphology effect, namely cup size and chest girth (i.e. internal organs resizing), over the count. Chest wall thickness variation with cup size proved to have exponential attenuation effect over the count as the Am-241 ray at 59.54 keV suffered a counting efficiency loss about 15% when comparing the 85A simulation and the one of the 85B phantom. On the other hand, chest girth variation proved to have greater effect since a factor of about 55 was noticed at 22 keV between the counts with the 85C phantom and the 120C one. This is mainly due to the fact that internal organs volume was changed with chest girth. More details concerning the library creation methodology and Monte Carlo MCNPX simulation results can be found in a previous publication (Farah *et al.* accepted).

2. Parameterization of the morphological dependence on counting efficiency

Counting efficiency is given in terms of source activity and the count of the detectors and is directly related to radiation attenuation by body tissue. Therefore, if phantoms' geometry changes, this attenuation also varies. This work aims to develop an equation that enables a correct and reliable estimate of this variation.

The series of created phantoms have different chest wall thicknesses (cup size and chest girth) and modified internal organs volumes (for the 100-110-120 models). To correctly relate two simulated counting efficiencies, the observed morphology changes are taken into account. In the case of a difference in chest wall thickness, exponential radiation attenuation is involved. The latter is highly dependant on crossed tissue nature. For simplification reasons, the adipose tissue mass attenuation coefficient was considered to represent the whole suffered attenuation by chest structures.

Hence, the new efficiency would be written as follows:

$$\varepsilon' = \varepsilon \times e^{(\mu_{adipose} \times chest \quad thickness)}.$$

In the case of internal organs volumetric differences, it is the loaded organ's volume that is involved. As shown by Farah (Farah et al. 2010), detectors' counting efficiency varies with the inverse of the lung volume in the case of volumetric uniform activity distribution. Hence, the ratio of the reference lung volume and the new one should also be taken into account and the new efficiency would be written as follows:

$$\varepsilon' = \varepsilon \frac{V}{V'}.$$

The equation relating two counting efficiencies is then of the form:

$$\varepsilon_2(E) = \varepsilon_1(E) c_1 e^{-c_2 \mu(E)} \quad (1)$$

Where ε_1 represents the reference counting efficiency, ε_2 the unknown efficiency and E the energy. The parameter c_1 should vary with the inverse of the lungs volumes and c_2 should take into account chest-wall thickness variation. Finally, $\mu(E)$ is the mass attenuation coefficient for ICRU-44 adipose tissue (7 compounds) (ICRU 1989) obtained with the online NIST-XCOM tool (Berger *et al.* 2005).

This developed equation was first used to reproduce the simulated experiments. However, it remains of limited interest if not applied to experimental measurements. Thus, the next step was to investigate the possibility of using calibration measurement as the reference efficiency of Equation (1).

Results

1. Application to simulated experiment

Equation (1) is applied to the MC simulated efficiency curves, taking the 85A model efficiency as the reference. The parameters c_1 and c_2 are found with a linear regression, considering the variables $Y=\ln\left(\frac{\varepsilon_{phant}}{\varepsilon_{ref}}\right)$, $X = \mu(E)$ and writing $Y = \ln(c_1) - c_2 X$.

When applied to the 85 and 90 (chest girth) models, the lung volume is the same as for the 85A model, hence the c_1 parameter is directly set equal to 1; this is not the case for the three biggest girths. Table 1 resumes the c_1-c_2 obtained values for all the 85 chest girth models starting from the 85A simulated efficiency.

Table 1. Values of the c_1-c_2 parameters and associated R^2 for the 85 chest girth and B to E cup sizes with the 85 A model being the reference.

Phantom	Parameters		
	c_1	c_2	R^2
85 B	1	0.856	0.9982
85 C	1	1.179	0.9967
85 D	1	1.467	0.9995
85 E	1	1.867	0.9928

These parameters are then used in Equation (1) and the resulting efficiency is compared to the MC simulated values. The maximum difference between the MC efficiencies and those obtained with equation (1) is a practical indicator of the validity of the proposed model. For the 85 E phantom, the highest relative error between the simulations' counting efficiencies and the calculated ones is about 6.4% and is observed for the lowest energy photons at 15 keV. For the 90 chest girth models the highest difference was found to be about 6% for the 90E phantom at 15 keV. Moreover, for the 100, 110 and 120 models, MC simulated efficiencies and the calculation results do not differ by more than 5%. Figure 1 compares the MC and calculated counting efficiency curves for 85B, 90C, 100D, 110E and 120F models.

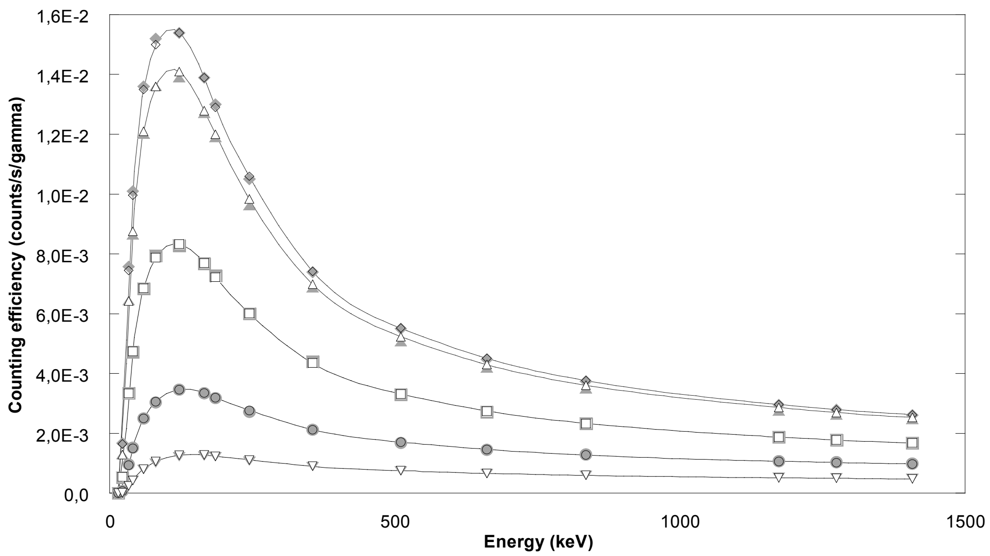

Fig. 1. Counting efficiency variation with energy. Comparison between MC simulated values (empty markers and solid lines) and the equation calculated values (filled markers) for the 85B model (diamonds), 90C (triangles), 100D (squares), 110E (circles) and 120F (inverted triangles); 85A simulation being the reference.

2. Application to measured experiment

In order to use the developed equation for reproducing experimental calibrations, the existing morphological differences were first identified and then their effect over the count included to transform the Livermore (Griffith *et al.* 1978) calibration curve to a MC simulated one.

The Livermore measurements carried out at the AREVA NC reprocessing plant were used for this study. The results included 5 different associated plates (P0 – P4) of thicknesses ranging from 1.8 to 4.3 cm. The Livermore lungs volume (loaded organs) is also different from the one of the female models. Moreover, detectors positioning induces significant variations in the lungs covered regions since it was not the same in both cases. Nevertheless, to a good approximation, moving the detectors away from the source is equivalent to changing the volume of the source without moving the detectors; this is taken into account with the same constant (c_1).

The c_1 - c_2 parameters of Equation (1) were calculated taking the Livermore calibration data as the reference efficiency and the 85A MC efficiencies as the unknown value. The obtained c_1 - c_2 values are given in Table 2 along with the associated R^2 coefficients describing the linear regression quality. According to this table, the best fit of the 85A efficiency is obtained with the Livermore P4 plate measurement being the reference.

Table 2. Values of the c_1-c_2 parameters and associated R^2 obtained to estimate the 85A efficiency from the various Livermore measurements and associated plates.

Livermore plate measurement	Parameters		
	c_1	c_2	R^2
P0	0.963	1.194	0.964
P1	0.990	0.184	0.206
P2	1.045	-0.140	0.192
P3	1.197	-0.999	0.982
P4	1.295	-1.910	0.990

Using the c_1 and c_2 parameters found in the P4 case, the 85A MC counting efficiency is reproduced with a maximal relative difference of 8.7% for the Eu-152 ray at 40 keV. It is hence now possible to transform the P4 plate measurement to any of the 24 female phantoms of the library as shown in figure 2.

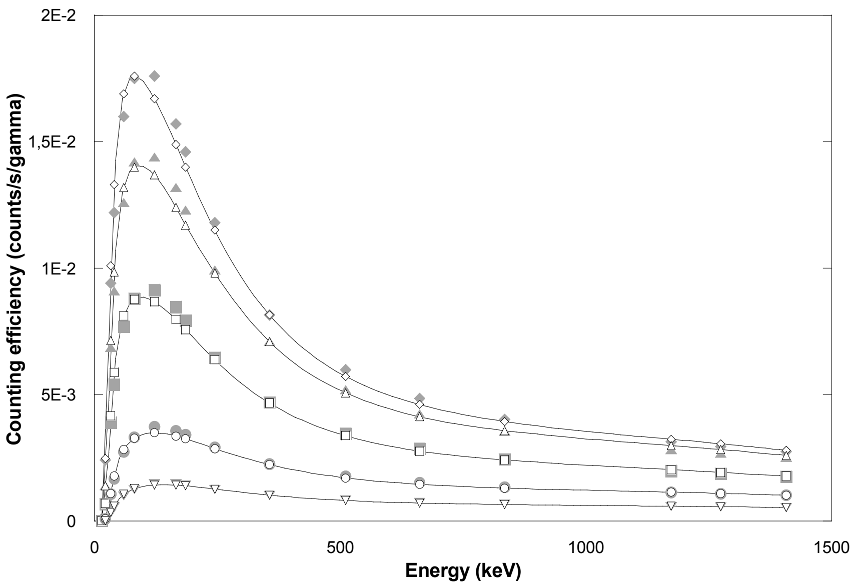

Fig. 2. Counting efficiency variation with energy. Comparison between MC simulated values (empty makers and solid lines) and the equation calculated values (filled markers) for the 85B model (diamonds), 90C (triangles), 100D (squares), 110E (circles) and 120F (inverted triangles). P4 measurement being the reference.

Discussion

Equation (1) uses the adipose tissue mass attenuation coefficient, which is tabulated and easily available, and two parameters. This part focuses on the study of these parameters' dependency with volumetric change (case of c_1) and chest wall thickness variation (c_2), the main implicated morphological factors.

For the 100, 110 and 120 models c_1 is expected to be close to the ratio of the reference lung volume (2.3 L) and the modified lung volume (2.99 L). Since the three biggest chest girths have the same lungs volume, they should have the same c_1 constant regardless of the cup size and its value should be about 0.77. The c_1 value for the 100 chest girth (0.73) is acceptable; this is far from being the same for the 110 and 120 phantoms (0.53 and 0.34 respectively). However, detectors positioning significantly alter these values since the 2 cm distance to the skin could not be systematically used for all models to prevent any collision with breast structures. As a consequence, the solid angle covering lungs changes and c_1 cannot exactly represent the ratio of lung volumes.

However, considering the small difference between calculated and MC efficiencies, it can be deduced that the model given by Equation 1 is useful, as long as we consider that obtaining a counting efficiency with an uncertainty of 5-10% is suitable.

To give a practical relation for the choice of c_1 and c_2, we examined the correlation of these parameters for the 100 to 120 chest girth. Figure 3 shows c_1 as a function of c_2 and demonstrate a linear tendency. This correlation is of particular interest since it enables the calculation of the counting efficiency as a function of female worker morphology. It enables the calculation of the c_1 value if a good estimate of c_2 is available, *i.e.* an estimate of the cup size variation. For example, this equation can be used in the case of a 105D chest girth where no numerical model was created.

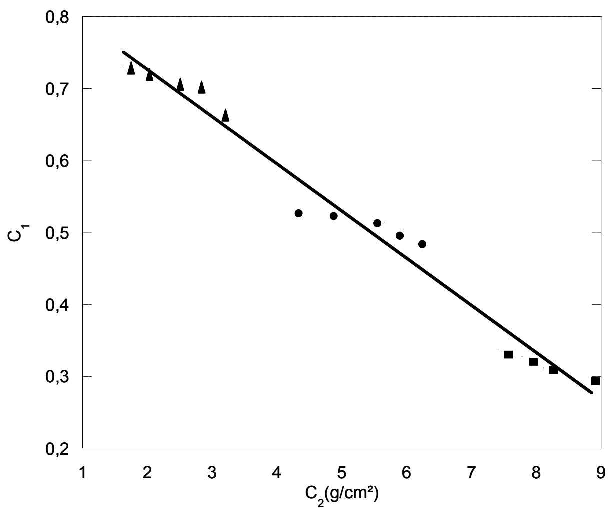

Figure 3. c_1 variation with c_2 for the 100 (triangles), 110 (circles) and 120 (squares) chest girths and associated cup sizes; linear fit plotted by Microsoft Excel 2008™: y = -0.0656 x + 0.8574, R^2=0.9728.

Conclusions

In this work, a new generation of anthropomorphic phantoms using Mesh and NURBS 3D flexible formats was created to optimize the monitoring of workers with internal contamination risks. The study focused on the lung monitoring of female worker for whom no physical calibration phantom is available.

For a typical 4-Germanium detectors *in vivo* counting system, counting efficiency curves were calculated for 24 typical female morphologies, taking into account different chest girth and cup sizes. The MC results are satisfyingly explained by a simple equation relating a reference and the unknown efficiency. The developed equation enabled a good estimate of the counting efficiency for every possible female subject and would be of great interest especially when no numerical model is available.

Acknowledgements

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ALARA – Education for personnel involved in the plant modification process

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Abstract

In order to achieve optimal worker protection at a nuclear power plant the radiological protection aspects should be considered in an early stage of the plant modification process. It is of utter importance that staff involved in the planning of and in the plant modification process itself is well informed on how the ALARA-principle can be applied. Today no specific ALARA-education exists for the staff involved in these tasks. A development of suitable education and information material for the personnel in question has been initiated at Forsmark NPP.

The work to develop an education package "ALARA for Personnel Involved in the Plant Modification Process" has been started by assessing the present level of knowledge and understanding as well as the need and interest for education. An enquiry was sent out to the department managers involved in plant modification process to spread in their organizations.

The percentage of received answers was less than 15% of the number of personnel involved in these tasks. This clearly indicates that the ALARA-principle has not been among the highest priorities in this area. About 50% of the personnel that answered the enquiry were interested in obtaining more knowledge and practical examples on how to apply the ALARA-principle early in the planning stage of plant modifications.

There is an interest and demand for more knowledge in the area but only among a limited group of the personnel. Our goal is to make ALARA-principle more commonly known and to make it an every-day tool used in all the possible steps in the plant modification process.

Reduction of dose around a storage pool by changing the position of BWR irradiated control rods

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Abstract

BWR control rods are irradiated into the reactor by the neutron flux and consequently activation reactions are produced in the materials composing the rod, mainly in stainless steel components and impurities. The dose produced by this activity is not important inside the reactor, but it has to be taken into account when the rod is withdrawn from the reactor and stored into the storage pool for irradiated fuel of the plant at a certain depth under water. The neutron activation has been modelled with the MCNP5 code, based on the Monte Carlo (MC) method. The pool containing hanger devices with irradiated control rods has been also modelled with the MCNP5 code. Doses potentially received by plant workers in the area surrounding the pool edges as well as in a platform moving over the water surface have been also calculated with the same MC code. In previous works, all these models have been validated. Results of the activation model proved that the rod handle is the most irradiated part of the control rod. Inverting the position of the rod into the storage pool with the handle at a deeper position under water should be a suitable method to reduce the dose out of the pool. In this work, the MC models are applied to verify the expected reduction of dose when the irradiated control rod is hanged in an inverted position into the pool.

Introduction

Control rods are activated by neutron reactions into the reactor. The activation is produced mainly in impurities contained in stainless steel as well as in the elements composing the steel alloy. The activity so generated will produce a dose around the rod, not important while it is inside the reactor, but it has to be taken into account when the rod is withdrawn from the reactor.

Activation reactions have been simulated with the MCNP5 code (X-5 Monte Carlo Team 2005), based on the Monte Carlo (MC) method and the number of reactions calculated can be converted into activity. When rods are withdrawn from the reactor, they must be stored into the storage pool for irradiated fuel of the plant at a certain depth under water. The storage is disposed to hang up 12 rods in a hanger device specially designed for this goal. Several hangers can be installed at various positions throughout the pool close to the walls. Doses potentially received by plant workers in the area surrounding the pool edges as well as in a platform moving over the water surface should be calculated to assure their adequate protection. Spent fuel elements are stored at the bottom of the pool while controls rods are nearer the surface. Therefore, doses out of the pool are mainly due to control rods.

In previous works (Ródenas et al. 2010a, 2010b), a simplified model was developed to estimate the activity generated as well as the dose around the storage pool. Afterwards, a detailed model of the control rod was developed (Ródenas et al. 2010c) considering all its components: handle, boron tubes, gain, and central core, dividing the rod into 5 zones to take into account the different axial exposition to neutron flux into the reactor. A further MC model was developed to estimate doses produced at points corresponding to these 5 zones. By comparing these doses with experimental measurements near an irradiated BWR control rod, it was validated the model developed. It was also validated the model developed simulating several groups of twelve control rods stored in the irradiated fuel pool of the NPP to calculate dose rates around the pool (Abarca et al. 2009).

Results of the activation model proved that the rod handle is the most irradiated part of the control rod. In this work, the developed MC models has been applied to verify the expected reduction of dose when the irradiated control rod is hanged in an inverted position into the pool.

Material and methods

Activation

The activity generated in the control rod depends on reaction cross sections, neutron spectrum, neutron flux distribution, concentration of precursors of each radionuclide, irradiation time and control rod history. After withdrawing the rod from the reactor, activities decrease with time and disintegration constants.

It is necessary to estimate the interaction rate Q (reactions /cm^3-s):

$$Q = C \int \Phi(E)\, \sigma(E)\, dE \qquad [1]$$

for each reaction, being C a normalization factor (at/barn-cm) depending on the target concentration; $\Phi(E)$ the neutron flux (n/cm^2-s); and $\sigma(E)$ the microscopic cross section of the reaction (barn).

On the other hand, for each j-isotope generated, a matter balance can be done:

$$dN_j/dt = Q_j - \lambda_j N_j \qquad [2]$$

integrating, the concentration (nuclei/cm^3) of j-isotope is obtained (being t_i the irradiation time):

$$N_j(t) = (Q_j/\lambda_j)\,(1 - \exp(-\lambda_j t_i)) \qquad [3]$$

For a cooling time (rod out of the reactor) t_e the concentration N_j becomes:

$$N_j(t) = (Q_j/\lambda_j)\,(1 - \exp(-\lambda_j t_i))\,\exp(-\lambda_j t_e) \qquad [4]$$

and multiplying by λ_j to obtain activity:

$$A_j(t) = Q_j\,(1 - \exp(-\lambda_j t_i))\,\exp(-\lambda_j t_e) \qquad [5]$$

It is a volumetric activity (Bq/cm^3). To obtain the total activity it is necessary to multiply by the cell volume. The maximum activity will be the asymptotic value, Q_j, considering an irradiation time very long (~∞) and neglecting the cooling time.

Major activation reactions are produced in steel mainly in gains and handle and mostly in some alloy components and impurities. Reactions considered and isotopes produced are listed in Table 1. Nevertheless, only those isotopes emitting gamma rays and with half-lives greater than 60 days have interest from the point of view of dose calculation. Consulting disintegration schemes (JANIS database 2005), they remain the following: Mn-54, Sc-46, Co-60, Zn-65, Nb-94, Ag-108m, Ag-110m, Eu-152, Eu-154, and Hf-178.

Table 1. Activation reactions produced in stainless steel of control rods.

N14 (n, p) C14	Fe54 (n, p) **Mn54**	Co59 (n, γ) **Co60**	Zn64 (n, γ) **Zn65**	Ag109 (n, γ) **Ag110m**
Al27 (n, γ) Al28	Fe54 (n, γ) Fe55	Ni60 (n, p) **Co60**	Mo92 (n, γ) Mo93	Eu151 (n, γ) **Eu152**
Cl35 (n, p) Cl36	Ni58 (n, α) Fe55	Cu63 (n, α) **Co60**	Nb93 (n, γ) **Nb94**	Eu153 (n, γ) **Eu154**
Cl37 (n, 2n) Cl36	Ni58 (n, γ) Ni59	Ni62 (n, γ) Ni63	Ag107 (n, γ) **Ag108m**	Hf177 (n, γ) **Hf178**
Ti46 (n, p) **Sc46**				

Activation model

Monte Carlo models developed (Ródenas et al. 2010a, 2010c) to assess the activity generated in control rods of a BWR are based on a detailed geometry that includes an axial division of the rod to consider the different periods and lengths of insertion during its permanency in the reactor core, so that the movement of control rods during reactor operation can be taken into account in the activation assessment. This detailed model can be seen in Fig. 1 drawn with SABRINA (Van Riper 2003) and VISED® (Carter and Schwarz 2005).

The interaction rate Q is calculated by MCNP using F4 tally and FM4 (tally multiplier card), which provides data for the reactions included in the calculation.

Dose rates around the control rod have been calculated and compared with measurements in order to validate the activation model (Ródenas et al. 2010c).

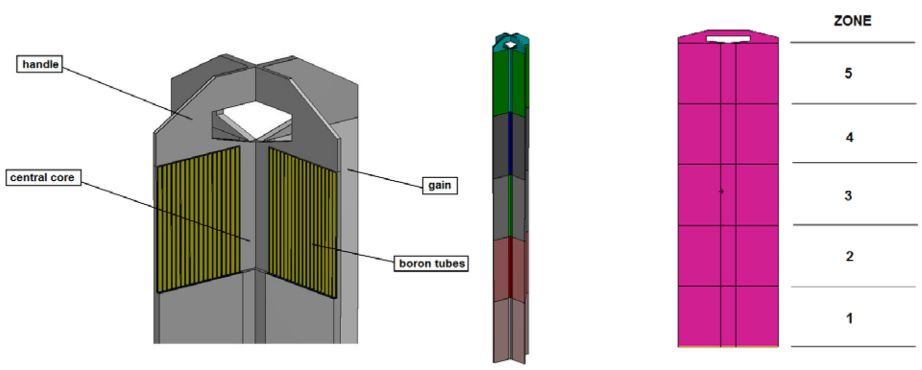

Fig. 1. Components of the control rod model and axial division of the model.

The neutron spectrum used in calculations is based on a mean value of the neutron flux throughout the reactor core, modified by a different probability in each zone that considers the control rod history. Fluxes have been obtained by means of the CASMO code (Knott et al. 1995) for a GE14 fuel element with 20 GWd/t of burnup and 40% of void fraction without control.

Pool Model

The spent fuel storage pool considered in this work has hexahedra shape, 11.375x5.258 m section and 13.1 m depth. The concrete walls and bottom of the pool have 1 m thickness. There are 30 cm of air over the upper level of water. Spent fuel elements are stored at the bottom of the pool while activated control rods are stored over them at about 3 m from the surface. The ground plan and elevation of the pool are represented in Fig. 2, where it can be seen the position of spent fuel elements and hangers for control rods.

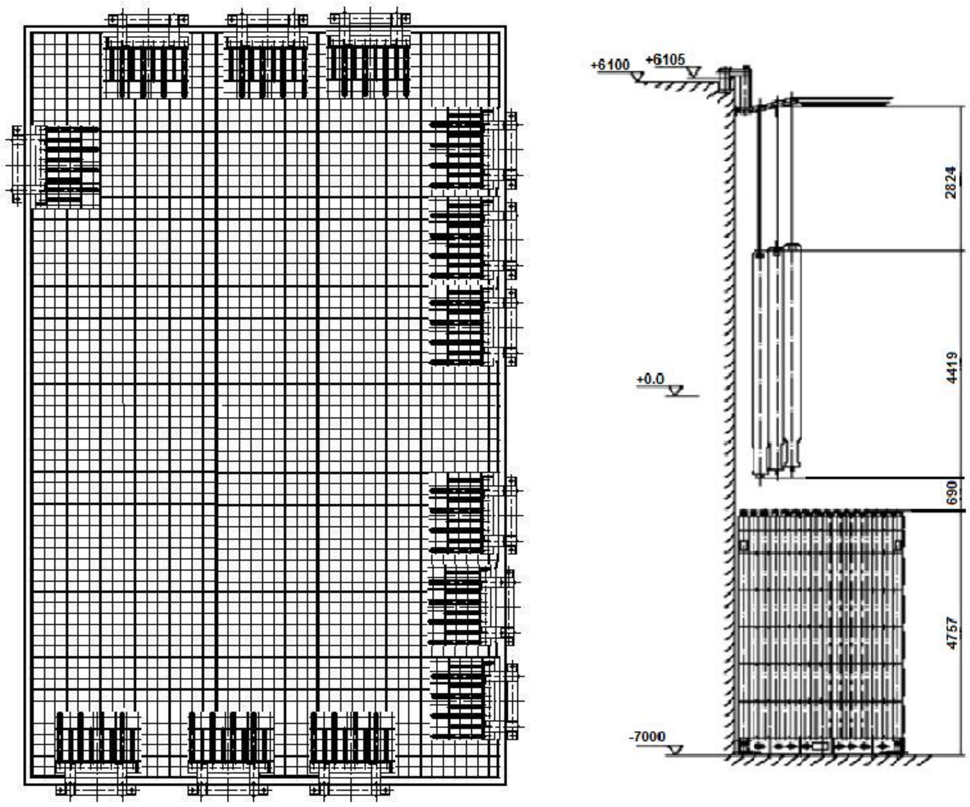

Fig. 2. Position of control rods hangers in the pool.

The pool has been modeled by a lattice with control rods in appropriate positions and the other cells filled with water. A VISED® (Carter and Schwarz 2005) scheme of the pool MC model can be seen in Fig. 3. It is shown in this figure, concrete walls, water filling the pool, hangers with control rods, and the air zone above the water where doses shall be calculated. Fuel elements either rod control hangers are not modeled as they do not practically interfere with the dose calculation.

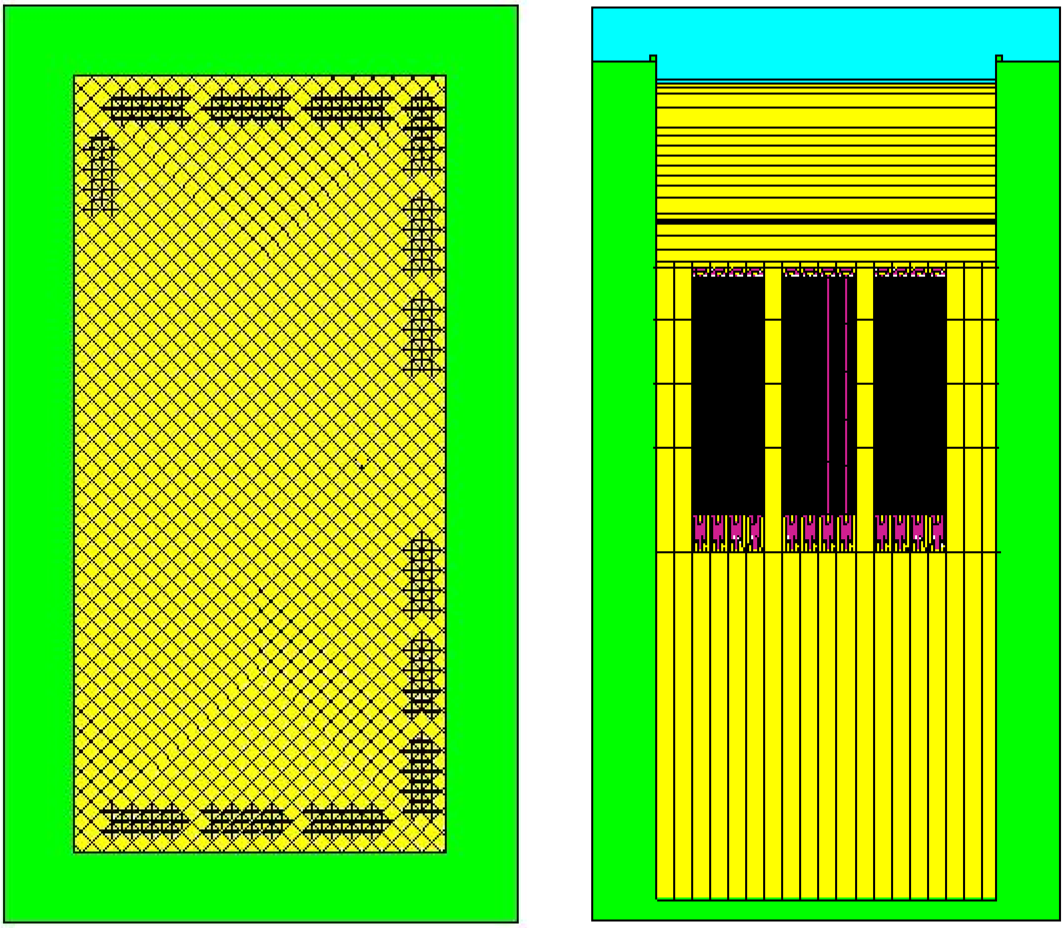

Fig. 3. VISED® scheme of the ground plan and elevation of the pool model.

Source Model

A cylindrical volume source with cell rejection option has been programmed. Furthermore, a defined volume distribution should be used in combination with the CEL variable to sample uniformly throughout the interior of a cell.

Each element of control rod presents different activity. Therefore, each cell in the source must have different photon emission and probability. To do it possible, the FCEL option has been used for energy specification in the source definition card. A repeated structure format shall be used in the CEL card to simulate sources in all the lattice elements occupied by control rods stored in the pool.

Variance reduction techniques

The main problem in the developed model is that very few photons reach the pool water surface. Therefore, statistics is very poor and large uncertainties are associated with results. To improve statistics, the water volume over the control rods is divided into several slabs with increasing importance values. They can be seen in Fig.3 right side.

An important reduction of computing time can be achieved using the SSW/SSR technique. In this technique, a first simulation containing the Surface Source Write (SSW) card is run. Those photons emitted from the source (activated control rod) that pass through a defined plane are recorded in a file with all its characteristics (energy, direction, etc.). This surface source file is used for subsequent MCNP calculations where the input deck contains the Surface Source Read (SSR) card. In the second and subsequent simulations, the number of particles can be strongly increased, but the source is simplified as it is just a plane and the recalculation of all interactions before arriving to this plane is avoided. Therefore, statistics can be improved with slowly increasing computer time.

Dose reduction model

The activation model previously validated (Ródenas et al. 2010c) has been applied to a BWR control rod obtaining the activities listed in table 2, where it can be seen that the maximum activity for all nuclides considered is located in the handle. An expected result because the handle is always into the reactor core while the other parts of the rod can be more or less introduced. Therefore, rotating 180º the rods into the storage pool with handles at a deeper position under water should reduce the dose out of the pool.

Table 2. Total activity (Bq) in the different parts of a control rod.

	Mn-54	Co-60	Nb-94	Ag-108m	Ag-110m	Eu-152	Eu-154	Hf-178	Sc-46	Zn-65
Handle	7.55E+11	7.82E+13	8.51E+07	3.54E+09	1.07E+11	2.97E+10	8.05E+08	6.34E+10	3.88E+07	3.59E+11
Core 1	9.78E+10	2.95E+12	5.28E+06	1.26E+08	4.55E+09	1.08E+09	3.22E+07	2.80E+09	4.39E+06	1.34E+10
Core 2	1.54E+11	4.85E+12	7.11E+06	2.27E+08	1.03E+10	1.77E+09	5.40E+07	5.97E+09	7.91E+06	2.28E+10
Core 3	2.10E+11	6.21E+12	9.35E+06	2.95E+08	9.51E+09	2.30E+09	6.71E+07	7.57E+09	1.10E+07	3.05E+10
Core 4	1.70E+11	6.98E+12	1.03E+07	3.05E+08	1.12E+10	2.51E+09	8.21E+07	7.38E+09	8.87E+06	3.18E+10
Core 5	1.72E+11	6.58E+12	8.37E+06	2.91E+08	1.22E+10	2.42E+09	7.18E+07	7.74E+09	1.01E+07	3.06E+10
Gain 1	1.57E+11	5.95E+12	8.01E+06	2.79E+08	9.94E+09	2.20E+09	6.30E+07	5.38E+09	7.78E+06	2.74E+10
Gain 2	2.72E+11	1.00E+13	1.23E+07	4.53E+08	1.58E+10	3.72E+09	1.06E+08	9.37E+09	1.39E+07	4.74E+10
Gain 3	3.27E+11	1.29E+13	1.61E+07	5.75E+08	2.01E+10	4.77E+09	1.34E+08	1.16E+10	1.65E+07	6.05E+10
Gain 4	3.47E+11	1.30E+13	1.69E+07	5.81E+08	2.00E+10	4.82E+09	1.35E+08	1.15E+10	1.74E+07	6.20E+10
Gain 5	3.09E+11	1.30E+13	1.61E+07	5.85E+08	1.88E+10	4.84E+09	1.35E+08	1.11E+10	1.54E+07	6.01E+10
Tubes	1.07E+12	2.83E+13	4.04E+07	1.31E+09	4.62E+10	1.01E+10	3.02E+08	2.68E+10	1.41E+07	1.36E+11

Results and discussion

The same MC model used for dose calculations around the pool (Abarca et al. 2009) can be applied to verify the dose reduction obtained when BWR control rods are hanged in an inverse position into the pool. It is just necessary to change the cells corresponding to control rods in order to represent the new position, as shown in Fig. 4 where the zone corresponding to control rods has been enlarged. It can be seen the handle in the opposite position than in Fig. 3. The rest of the model doesn’t change and the same variation reduction techniques are applied.

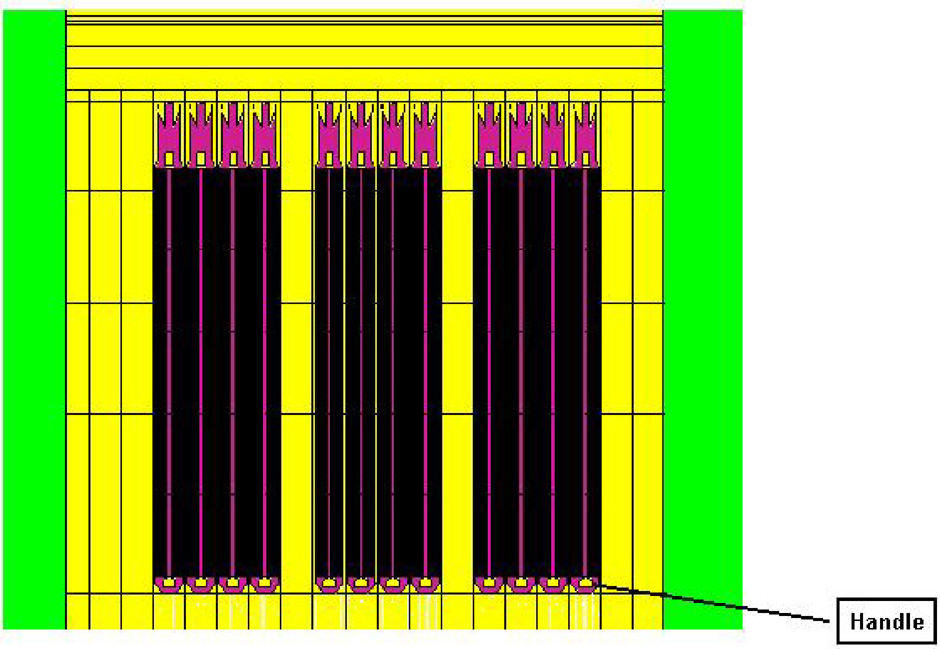

Fig. 4. VISED® scheme of the elevation of the pool model with control rods rotated 180º.

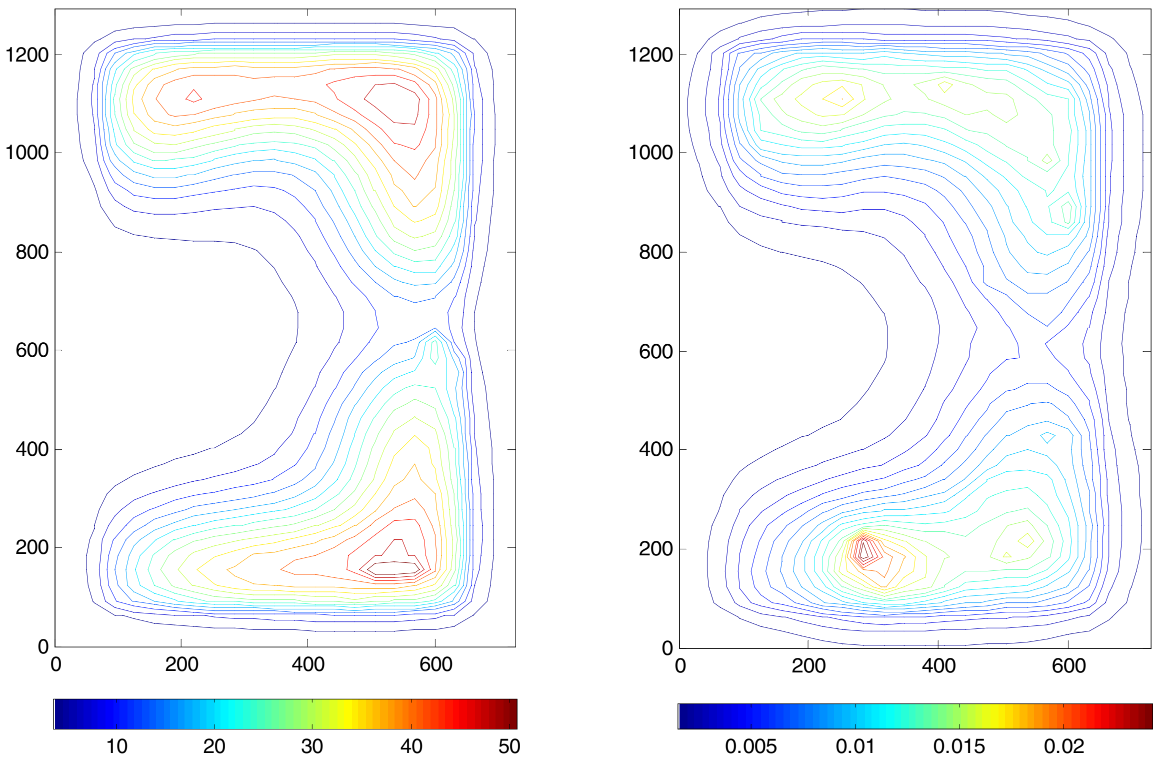

Fig. 5. Dose rate (μSv/h). Left: normal position. Right: reverse position.

Again a F4MESH tally has been used to determine dose rate in a mesh in air over the piscine, the zone 1 m above the free water surface and 1 m beyond piscine walls marked in cyan blue in Fig. 3.

Calculated dose rates (μSv/h) are represented in Fig. 5. At left side for control rods in the normal position, while at right side they are rotated 180º. Uncertainties associated with calculated doses are lower than 10%, except around the red lines in the right graph where it is about 25% due to the fact that few particles reach the water surface from these points.

Really the number of particles reaching the zone where dose rates are calculated is again a problem, even worst than for normal position as photons have a longer pathway when they are started from the handle where the activity is the largest.

In any case, as it can be seen in the isodose maps represented in Fig. 5 the dose rate over the pool can be reduced more than 2,000 times changing the storage position of control rods. A comparison between dose rates (μSv/h) calculated for both control rod positions is done for 5 points: maximum, meaning the point where the dose rate is maximum for normal position of control rods; middle of the pool; and the four corners of the pool (right lower, right upper, left lower, and left upper). In table 2 these values are listed together with the dose reduction and the ratio between doses at normal and inverse position.

Table 3. Dose rate and dose reduction in selected points over the pool.

	Dose rate (μSv/h)			
Calculation points	**Normal Position**	**Reverse Position**	**Dose Reduction**	**Ratio**
Maximum	50.00	0.0161	49.979	2500
Middle of the Pool	7.58	0.0037	7.576	2048
Right Lower Corner	22.15	0.0092	22.141	2408
Right Upper Corner	21.47	0.0089	21.461	2412
Left Lower Corner	13.61	0.0043	13.606	3165
Left Upper Corner	29.38	0.0102	29.369	2880

Dose ratios are represented in Fig. 6 where it can be seen the huge reduction of dose for the area surrounding the pool obtained when the irradiated control rods are stored in a position rotated 180º. In this area, it is more probable the presence of operators.

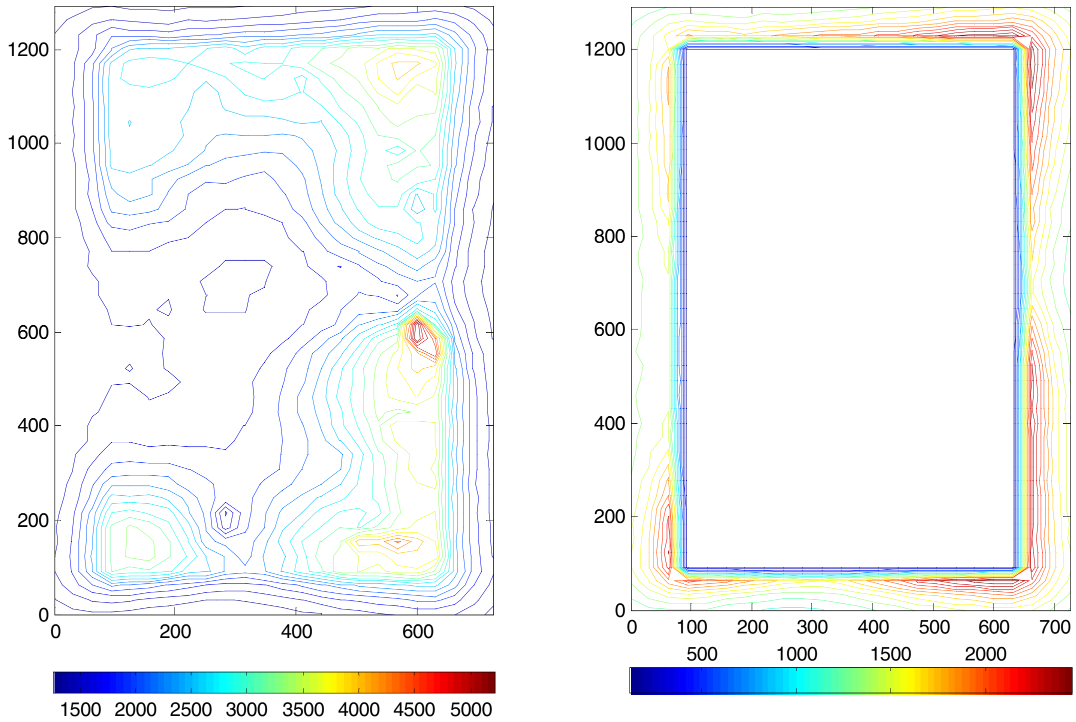

Fig. 6. Dose rate ratio. Left: over the pool. Right: around the pool.

Conclusions

The activation of control rods due to neutron irradiation during its stay into the reactor core is more important for the rod handle as in BWR it is always into the core while the other parts of the rod can be more or less introduced. Therefore, an inversion of the position of control rods stored into the irradiated fuel pool will produce an important reduction of the dose around the pool.

Monte Carlo models developed and validated to assess activity produced in control rods and dose rates in the area surrounding the storage pool have been applied to the case of a BWR power plant, considering the usual position of the control rods and the reverse position into the storage pool.

Results show an important reduction of dose rate, by a factor of 2000 approximately, in the zones where operators are usually working.

Therefore, it is very advisable to introduce the appropriate changes in the hanger devices in order to achieve this important dose reduction.

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Periodic review and update of the company ALARA program – Continuous improvement in the field of radiation protection

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Abstract

The presentation will show how continuous improvement in the field of radiation protection may be achieved by actively follow up the results from the existing company ALARA program at Forsmark NPP and depending on this updating the program for the next period. Our ALARA-program consists of the following parts:

– Collective and individual doses to the personnel
– Releases of radioactive substances to the environment
– Organizational acceptance of the ALARA principle.

By trending and monitor performance it is possible to focus on the most significant areas. In the presentation examples will be given from Forsmark NPP in all of the above mentioned areas.

ISEMIR: a new international system for improving occupational radiation protection in medicine, industry and research

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Abstract

The Information System on Occupational Exposure in the Medicine, Industry and Research (ISEMIR) was set up by the IAEA in January 2009 to focus on very specific topical areas where radiological protection stakes for the workers are not trivial, and where there are still pending issues and deficiencies. The first two selected areas are:

– Interventional cardiology, and
– Industrial radiography.

2009 – 2011 is a test period for checking the ability of the system to cope with its main objectives, as stated in its terms of reference: "To help to improve radiation protection programmes in medicine, industry and research, and in particular:

– To facilitate the implementation of ALARA practices and effective exposure management;
– To internationally benchmark specific task-related occupational exposures in order to identify good practices as well as gaps; and
– To define follow up actions to address identified gaps and disseminate lessons learnt; To contribute to minimizing the likelihood of accidents, e.g. by identifying pre-cursors, user feedback and experience."

This presentation will present the organisation of the system (advisory group, secretariat, working groups) and the first one and half years of experience. In particular it will describe what the Working Group on Interventional Cardiology, which will have already met three times, has performed and achieved, including the results of 3 simple questionnaires dealing with radiation protection practices and occupational exposures addressed to individual interventional cardiologists, chief interventional cardiologists, and radiation protection regulatory bodies. A total of 329 responses from 75 countries were received across the 3 surveys. Finally the complementarities and relationships with other international systems (e.g. UNSCEAR and ISOE) and networks (such as ALARA networks) will be addressed.

Problems of the radiation protection and health effects monitoring in Ukrainian radiation workers

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Abstract

Nuclear energy sector in Ukraine includes 50,000 workers of the 15 power and research units, uranium mines, radioactive waste storages, staff of the Chernobyl zone and the “Shelter” object. Creation of a centralized registry with dosimetry and health data is essential and such understanding exists at the governmental and local levels. However, de facto this work is initiated slowly due to lack of budgetary funding. An analysis was performed of the existing sources of health and dosimetry data information. Two surveys defined the general status of dosimetry monitoring and number of occupationally exposed workers. The local data sources will be used including individual data from the local medical-sanitary departments, dosimetry shops and regional registries for radiologists. Cancer statistics of sufficient quality of case identification and pathological data could be obtained by linkage from the National cancer registry. The extended health data and dosimetry including external irradiation and transuranium elements are available for staff of the “Shelter” object at the RCRM. The existing experience of French and other international registries as well as of the State registry of exposed after Chernobyl, the Clinical-epidemiological registry, and dosimetry databases gained in the Research Center for Radiation Medicine from the prospective follow-up studies could help for practical implementation of the nuclear workers registry. Several tasks are foreseen that could be successfully implemented with international cooperation: a survey of professions, types of jobs and radiation qualities; development of qualification criteria and accreditation procedures for personal dosimetry services; pilot study of medical registry of occupationally exposed workers. Establishment of a new multi-thousand cohort for both prospective and retrospective biomedical and epidemiological studies will allow more precise estimated of the low dose effects of ionizing radiation.

EPR: Comparative approach of the French and Finnish regulatory reviewing process and optimization of radiation-protection at the design phase

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Abstract – Introduction

Following the assessment of the EPR[1] (European Pressurized Reactor) preliminary safety analysis report in France, the purpose of this paper is to present a comparative approach of the French and Finnish reviewing process.

The overall picture drawn in this occasion is dedicated:

- to remind the history of EPR (from the decision to implement studies in the 90's to the French and German cooperation and finally to the construction of a unit in Finland and another one in France);
- to compare French and Finnish safety evaluation systems: in France, the safety authority in charge of the authorization process is not directly linked to its technical support which leads the technical instruction. In Finland, the safety authority is in charge of the evaluation of safety analysis. In this process Technical Support Organizations (TSO) can be requested for example in some comparative calculations.
- to present the dose targets (calculated reference doses) planned by the nuclear operators in the design phase as well as the global radiation-protection optimization process. In France, for example, EDF performed a detailed optimization analysis on selected tasks known to have a major contribution to the annual average collective dose (thermal insulation, logistics, valve-maintenance,

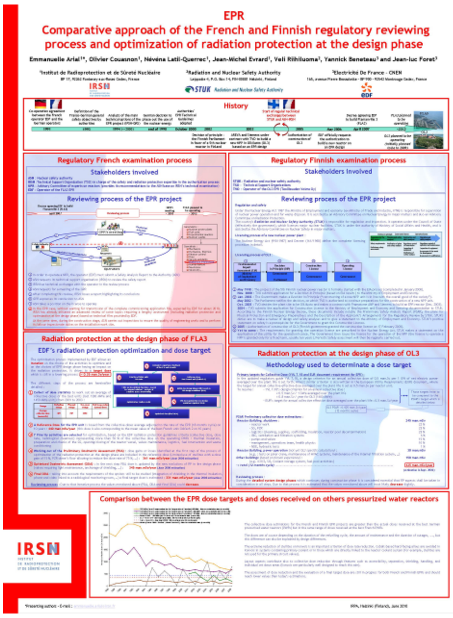
[1] EPR appears now with the sign TM (for trademark) when applied to business relationships. It is a distinctive sign used by AREVA to signify that the EPR is an unregistered trade mark (a mark used to promote or brand goods) which identifies the EPR as a product of AREVA.

opening/closing of the vessel, preparation and checks of steam generators, on-site spent fuel management and waste management). The optimization process is set in France on an iterative method. In Finland the optimization of annual collective dose has to be described in a separate topical report. In every phase of system descriptions the radiation-protection aspects have to be taken into account to meet the requirement stated in specific regulatory guides.

- as a conclusion, to draw a comparison between the EPR collective dose target and doses received on other pressurized water reactors that are close to the EPR design (Konvoi of German design, "best French units", …).

This paper has been jointly written by the French operator (EDF), the French TSO (IRSN), and the authority of nuclear safety in Finland (STUK).

It allows to summarize more than 15 years of partnership and studies, focusing on radiation-protection, in the design phase of the EPR.



History

At the beginning of the 90s, the basic safety provisions were defined and assessed of the basis of a French and German cooperation (Figure 1):

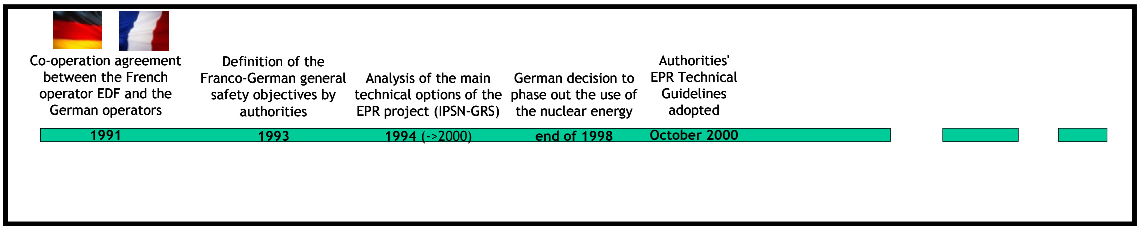


Figure 1. French and German cooperation.

From 2002, on the decision of principle to build a new NPP in Finland (based on an EPR concept), a new kind of cooperation started between French and Finnish stakeholders (figure 2):

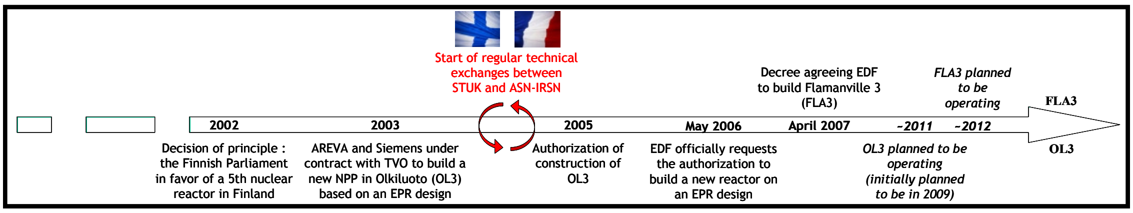


Figure 2. French and Finnish cooperation.

Regulatory examination process

In France:

In France the main stakeholders involved in the reviewing process are:

- ASN: Nuclear safety authority;
- IRSN: Technical Support Organization (TSO) in charge of the safety and radiation protection expertise in the authorization process;
- GPR: Advisory Committee of experts on reactors (provides its recommendations to the ASN on the basis of IRSN's technical examination);
- EDF: Operator of the Flamanville 3 EPR.

The reviewing process from the authorization decree to the operating licence for the Flamanville 3 NPP is described in the figure 3:

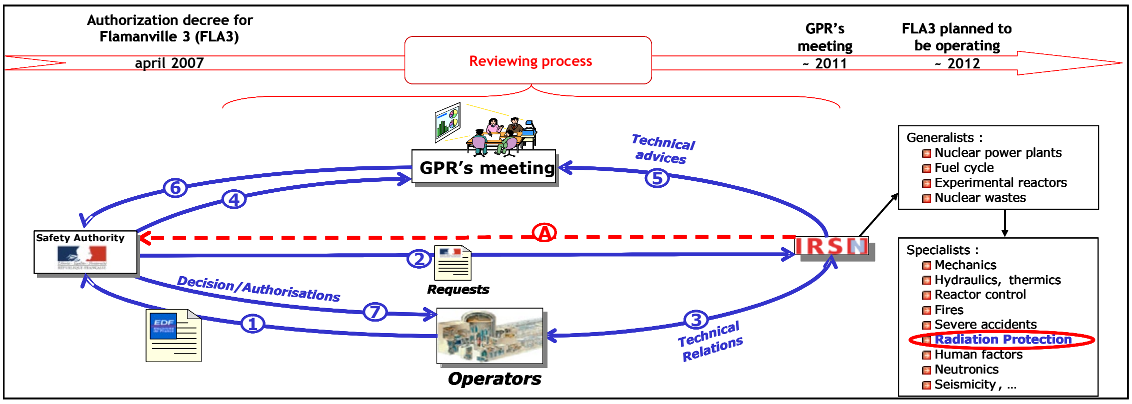


Figure 3. French reviewing process.

The main steps of this reviewing process are the following ones:

1. In order to operate a NPP, the operator (EDF) must submit a Safety Analysis Report to the Authority (ASN);
2. ASN requests its technical support organization (IRSN) to review the safety report and requests the GPR to express its opinion on the conclusions of this review 4;
5. After completing its review based on technical exchanges with the operator 3, IRSN issues a report highlighting its conclusions;
6. GPR expresses its conclusions to ASN;
7. ASN takes a position on the licence to operate;

Ⓐ In the Flamanville 3 NPP case, without waiting for transmission of the complete commissioning application file, expected by EDF for about 2010, IRSN has already initiated an advanced review of some topics requiring a lengthy assessment (including radiation protection and optimization at the design phase) likely to have potential impact on the design. This review is based on technical files provided by EDF.

At the same time, during the construction stage, ASN carries out inspections to ensure the quality of engineering works and to perform its labour inspectorate duties on the installation work site.

In Finland :

In Finland the main stakeholders involved in the reviewing process are:

- STUK: Radiation and nuclear safety authority;
- TSOs: Technical Support Organizations;
- TVO: Operator of the OL3 EPR (Teollisuuden Voima Oy).

Under the Nuclear Energy Act (1987) the Ministry of Employment and economy (ex-Ministry of Trade and Industry, KTM) is responsible for supervision of nuclear power operation and for waste disposal. It is assisted by an Advisory Committee on Nuclear Energy in major matters and also an Advisory Committee on Radiation Protection.

The country's Radiation and Nuclear Safety Authority (STUK) is responsible for regulation and inspection. It operates under the Council of State, which licenses major nuclear facilities. STUK is under the authority of Ministry of Social Affairs and Health, and is assisted by the Advisory Committee on Nuclear Safety in major matters.

The Finnish licensing process of a new nuclear power plant is described hereafter in the figure 4:

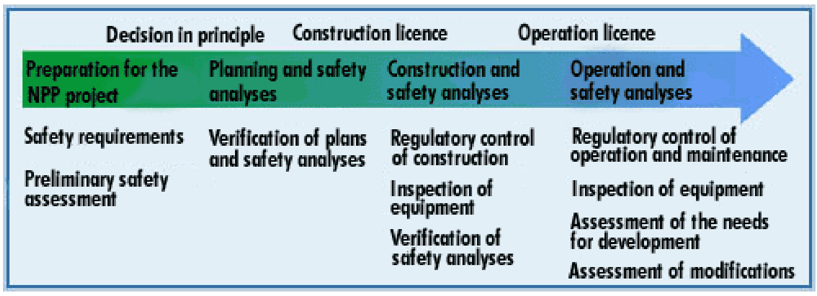


Figure 4. Finnish licensing process.

The Nuclear Energy Act (990/1987) and Decree (161/1988) define the complete licensing procedure in detail. The licensing process of OL3 is described on the figure 5 hereafter:

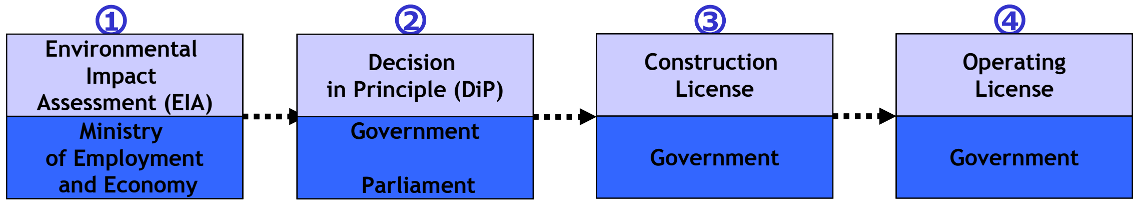


Figure 5. Licensing process of OL3.

① May 1998: The project of the 5th Finnish nuclear power reactor is formally started with the Environmental Impact Assessment (EIA) process (completed in January 2000) ;
Nov. 2000: TVO submits application for a Decision in Principle (based on EIA results) to the Ministry of Employment and Economy ;

② Jan. 2002: The Government makes a Decision in Principle ("constructing of a new NPP unit is in line with the overall good of the society") ;
May 2002: The Parliament ratifies the decision, on which TVO is authorized to continue preparations for the construction of a new NPP unit ;
Oct. 2003: TVO decides the plant site to be Olkiluoto and makes a contract with Framatome ANP and Siemens to build an EPR reactor (Dec. 2003) ;
2004: TVO submits the application for Construction License to the Ministry of Employment and Economy and the licensing documentation to STUK. According to the Finnish Nuclear Energy Decree, these documents include notably the Preliminary Safety Analysis Report (PSAR), the plans for Physical Protection and Emergency Preparedness and the Description of the Applicant's Arrangements for the Regulatory Review by STUK. STUK's duties are to make a review of design and safety analyses and to give a statement to the Ministry of Employment and Economy. STUK's positive statement on safety is a prerequisite for the Government to grant the Construction License ;

③ 2005: Authorization of construction of OL3 (Finnish government granted the construction license on 17 February 2005).

④ Yet to come: The requirements for granting the operation license are prescribed in the Nuclear Energy Act. STUK makes a statement on the application of the utility for the operation license. The Government then issues the license for the operation of the NPP (the license to operate a NPP is granted only for a fixed term, usually ten years). Periodic Safety assessment will then be regularly carried out.

Radiation protection at the design phase

In France (FLA3) : EDF's radiation protection optimization and target dose

The optimization process implemented by EDF allows iteration on the choice of the activities to optimize and on the choices of EPR design phase having an impact on the radiation protection. It drives to a target dose (< 0.35 man.Sv/year) which is still in a lowering process.

The different steps of the process are presented on figure 6.

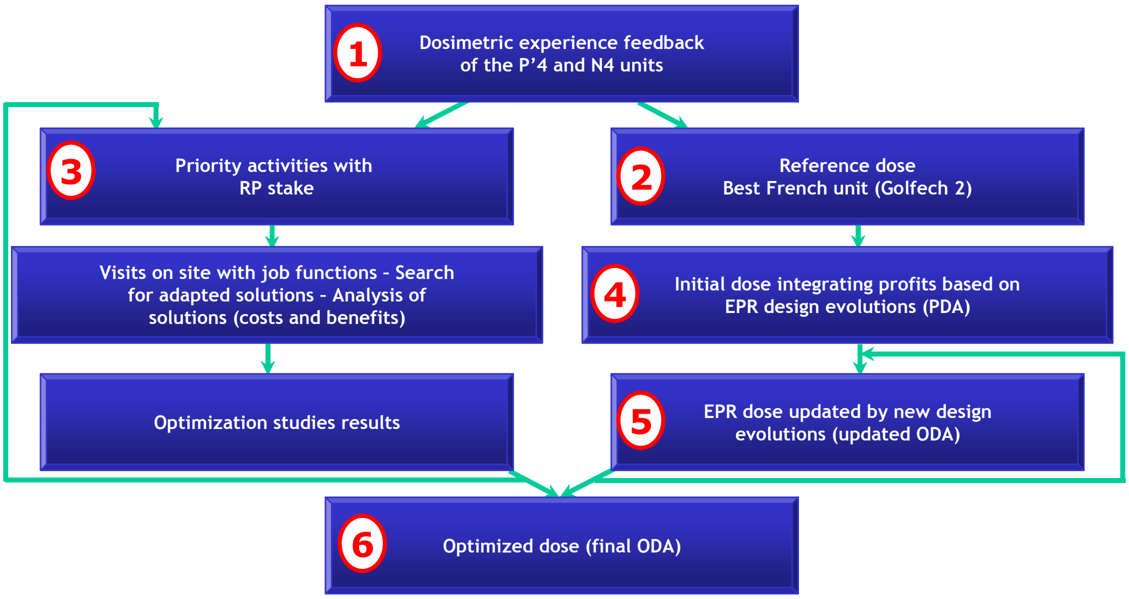


Figure 6. optimization process implemented by EDF.

① The first step aims to collect the dose statistics to work out an average of collective doses of the best units (last 1300 MWe and 1450 MWe units) from 2001 to 2003 :

Table 1. average of collective doses of the best units from 2001 to 2003.

Outage	Refueling outage	Planned outage	Ten-years outage	Unit in operation (annual dose)
Average collective dose (man.mSv)	329	527	1352	89

② A Reference Dose for the EPR unit is issued from the collective doses average adjusted to the rate of the EPR (18 months cycle) on 10 years : 440 man.mSv/year (this dose is also corresponding to the mean value of the best French unit Golfech 2 on 10 years);

③ Seven Priority activities are selected for optimization, based on the EDF radiation protection guidelines criteria (collective dose, dose rate, radiological cleanness) representing more than 50 % of the collective dose on the operating PWRs: thermal insulation, preparation and checks of the Steam Generators, opening/closing of the reactor vessel, valves maintenance, logistics, fuel intervention and waste conditioning;

④ The Preliminary Dosimetric Assessment (PDA) is carried out to integrate in the reference dose the dose gains or losses identified at the first step of the process of optimization of the radiation protection at the design phase (Limited use of stellites with a dose gain of 15 %, pressurizer dome's floor allowing to reduce the dose rate of 75 %, …). As a result, the PDA leads to a collective dose value of 361 man.mSv/year (estimation in June 2008);

⑤ Following an Optimized Dosimetric Assessment (ODA); the PDA dose is updated by the new evolutions of RP in the design phase (valves requiring light maintenance, anchorage of shielding,…). The ODA leads to a collective dose value of 345 man.mSv/year (estimation in June 2008);

6 Eventually, a final ODA is carried out by taking into account the requirements of the options still to be studied (phone and video linked to a radiological monitoring room,…). The final target dose is estimated to be 331 man.mSv/year (estimation in June 2008). However, the official collective dose target is yet set to 350 man.mSv/an (from the version of the Preliminary Safety Analysis Report of 2005).

Due to that iterative process the values mentioned above (PDA, ODA and final ODA) could decrease slightly before the end of the process.

In Finland (OL3): Methodology used to determinate a dose target

As in France, primary targets for collective dose have been set forth .

In the updated regulatory guide YVL 7.18, a design criterion for an annual collective dose of 0.5 man.Sv per 1 GW of net electric power averaged over the plant life is set forth (< 0.8 man.Sv / year for OL3 - 1600 MWe).

Almost similar criterion is also written in the European Utility Requirements (EUR) document, where the target for annual collective effective dose averaged over the plant life is set as 0.5 man.Sv/year per reactor unit.

Those targets have to be compared to the OL3 Preliminary Safety Report (PSAR) target of 0.425 man.Sv/year on 12 months cycle which is detailed below.

<u>PSAR Preliminary collective dose estimations :</u>

Reactor Building, shutdown :	*245 man.mSv*
- reactor work	22 %
- SG, PZR	23 %
- logistics[2]	23 %
- I&C, ventilation and filtration systems	6 %
- pumps and valves	15 %
- management, operations team, health physics	10 %
- NDE, hydraulic tests	1 %
Reactor Building, power operation[3] :	*30 man.mSv*
Other buildings[4] :	*150 man.mSv*
-> total (12 months cycle, estimation in September 2006) :	*425 man.mSv/year*

During the detailed system design phases which continues during construction phase it is considered essential that RP aspects shall be taken in consideration in all steps. Due to that process it is estimated that the values mentioned above will most likely decrease slightly, before the end of the process.

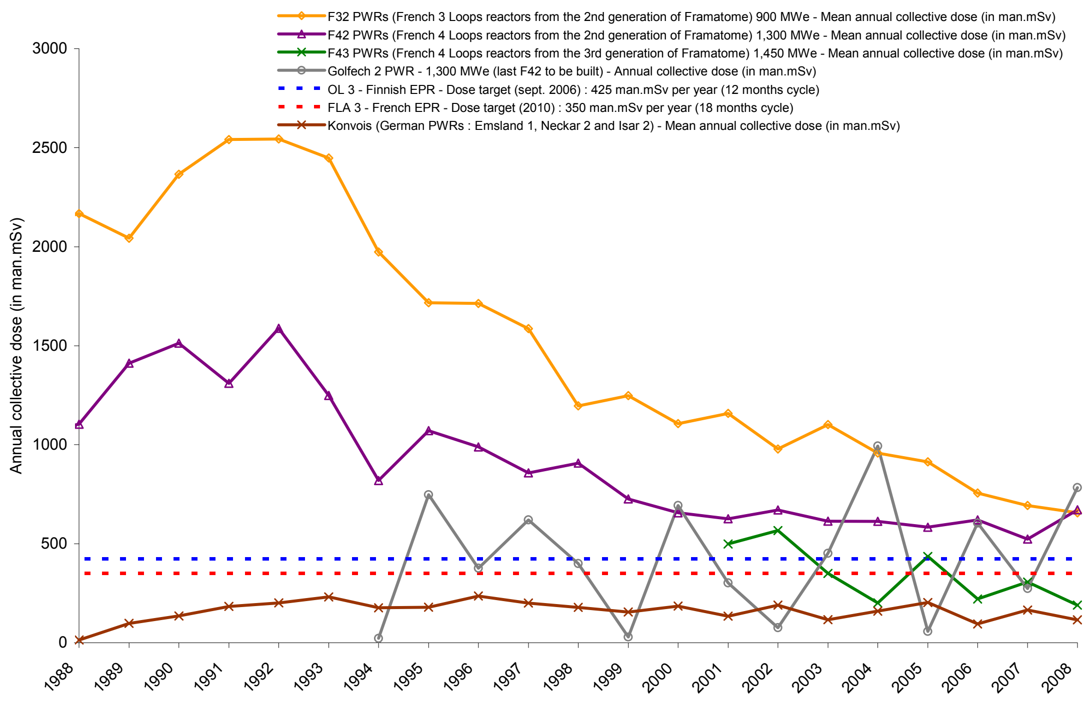
[2] Cleaning, supplies, scaffolding, insulation, reactor pool decontamination.
[3] Not yet OL3 specific calculation (e.g. : tests on polar crane, maintenance of HVAC systems, …).
[4] Based on Konvoi experience (e.g. :CVCS, SIS, Coolant storage system, fuel pool activities).

Specific EPR design in comparison with Finnish and French existing PWR

One of the main specific options in the EPR design is the « two-rooms » concept : this concept is devoted to allow interventions in the reactor building when operating to prepare the outage. It consists in a red zone collecting the leakages of the primary circuit (inaccessible zone) and a green zone (the other part of the RB accessible) which is required to have a total dose rate < 25 µSv/h (and a neutrons DR < 2.5 µSv/h).

Conclusion: comparison between the EPR dose targets and doses received on others pressurized water reactors

The graph shown hereafter gives a good overview of the annual collective dose benchmarking for French PWR compared to the EPR's targets.



Graph 1. Annual collective dose benchmarking (in man.mSv) for French PWR (first 900 MWe excepted) between 1988 and 2008 compared to the EPR's targets (source : ISOE[5] database).

The collective dose estimations for the French and Finnish EPR projects are greater than the actual doses received at the best German pressurised water reactors (PWRs) but in the same range of doses received at the best French PWRs. The doses are of course depending on the duration of the refuelling cycle, the amount of maintenance, the duration of outages, ..., but this difference can also be explained by design differences.

The extreme reduction of stellites on Konvois is an important criterion of dose rate reduction. Cobalt-based hard-facing alloys are avoided in Konvois in systems

[5] Information System on Occupational Exposure

containing primary coolant or in those which are directly linked to the reactor coolant system (for example, stellite is not used for the primary circuit valves).

Layout aspects contribute also to collective dose reduction through features such as accessibility, separation, shielding, handling, and individual set down areas (Konvois are particularly well designed to reach this aim).

The assessment of dose reduction and the evaluation of a final target dose are still in progress for both French and Finnish EPRs and should reach lower values than today's estimations.

Individual doses monitoring for external exposure during the transportations of nuclear fuel bundles performed by Nuclear Fuel Plant Pitesti

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Abstract

Nuclear Fuel Plant (FCN) is a facility that produces nuclear fuel bundles CANDU-6 type for CANDU nuclear power plant based on natural and depleted uranium. The transportation of nuclear fuel bundles from FCN to Cernavoda Nuclear Power Plant is performed by FCN authorised trucks (about 340 km between the two locations). FCN has the activity of transportation and the vehicles authorised by Romanian regulatory body National Commission for Nuclear Activities Control (CNCAN). In addition each transfer of nuclear material is authorised by CNCAN.

About 15 nuclear fuel transportations were performed in year (20 wooden crates with 720 nuclear fuel bundles, containing 15 tons of natural uranium dioxide corresponding with 14 tons of natural uranium). From 1996 until now 120 transportations of nuclear fuel bundles were occuring.

A crew is trained and authorised for these kinds of activities. The components of crew are monitored everytime for external exposure. Different type of instrumentations was used for conforming with radioprotection programme of transportation of radioactive material national and international norms. The doses registered are reported each year to CNCAN.

Compared with the registration level (RL) of doses imposed by CNCAN norms, the doses received during transportation don't exceed this value. Starting with 2009 each occupationally worker is monitored with TLD and electronic devices. Due that the workers from the transportation crew are exposed to the ionising radiations category A, the annually estimated dose that can be received is around 6 mSv per year. Nobody of participants to the transportation of nuclear fuel bundles reach this value during 14 years of activity.

1. Introduction

Nuclear Fuel Plant (FCN) is a subsidiary of National Society NUCLEARELECTRICA SA (SNN-SA), a company that has in subordonance another nuclear facility Cernavoda Nuclear Power Plant. FCN is a facility for manufacturing of the nuclear fuel bundles CANDU type with 37 elements, based on *natural uranium* (0.711% U-235) and *depleted uranium* (a small quantity with 0.25% U-235 and 0.52% U-235). FCN is located at 130 km from Bucharest in Mioveni town on the nuclear site that belonging FCN, Institute for Nuclear Researches (INR) and Nuclear Agency and Radioactive Wastes (AN&DR).

The annual production is about 10,000 fuel bundles CANDU type that means about 200 tons of natural uranium in UO2. The depleted uranium is processing in campaigns only at the starting of a new unit from Cernavoda NPP.

The production of the plant was about 5,600 nuclear fuel bundles. In 2007 the Unit 2 of Cernavoda NPP was commissioned and for this reason the capacity of the plant was doubled at about 10,000 fuel bundles per year. This means that the quantity of uranium throughout in the plant stored and processed was doubled and normally the number of nuclear fuel bundles transportations to Cernavoda NPP.

The personnel working in FCN is about 430 people, and the activity is continuously and about 20 people are involved in packaging, final inspection, loading, movement, transferring and monitoring the shipments from FCN to Cernavoda NPP.

The distances from FCN and Cernavoda NPP for delivery the nuclear fuel bundles is about 340 km.

2. Nuclear national framework

The regulatory body for control of nuclear activities is National Commission for Control of Nuclear Activities (CNCAN). The legislation in this field contains law, norms, regulations, orders, etc.

The Romanian law issued in nuclear field is Law no. 111/1996 for deployment, regulation, licensing and control of nuclear activities, republished in 2006 [1]. The law is following in each nuclear area, upon the specific of activity, by orders, norms, guides, etc.

For FCN the group of norms with direct application to the specific activity are: *norms for radiological safety, norms for radioactive mining, norms for quality management, norms for physical protection, norms for nuclear safeguards, norms for management of the radioactive wastes*, ***norms for transportation of radioactive material.***

3. Nuclear plant licenses

The regulatory body issued for FCN during the period 2010-2012 several licensees that cover the activities in nuclear field using radioactive material. The documentations sending by FCN are based on **Norms for Radiological Safety** and **Norms for Radioactive Mining** requirements**:**

a) **Possesion, Using, Handling, Processing of the nuclear raw material, Producing of the nuclear fuel bundles, Temporary Storage and Supplying**
b) **For Personnel Dosimetry Laboratory CNCAN** has issued an accreditation for 2008-2011
c) **For Transportation of Radioactive Material FCN** is authorized by CNCAN for the period 2009-2014, one of activity being nuclear fuel bundles [1], [2].
d) **For Quality Management System CNCAN** has issued an authorization during the period 2008-2010 [1].

4. Short history of transportation of nuclear fuel bundles

The activities for Radioactive Material Transportation starting from 1996 and until now 126 shipments of nuclear fuel bundles was occur. Three conveyaces formed by trucks with two trailers was used for this activity.

From 1996, over 126 transportations of nuclear fuel bundles (1.4 millions kg of natural and depleted uranium) to Cernavoda NPP were performed, over 88,000 nuclear fuel bundles CANDU-6.

All the activities for radioprotection, dosimetry, physical protection, safety and security related with transportation of radioactive material are under Quality Management System.

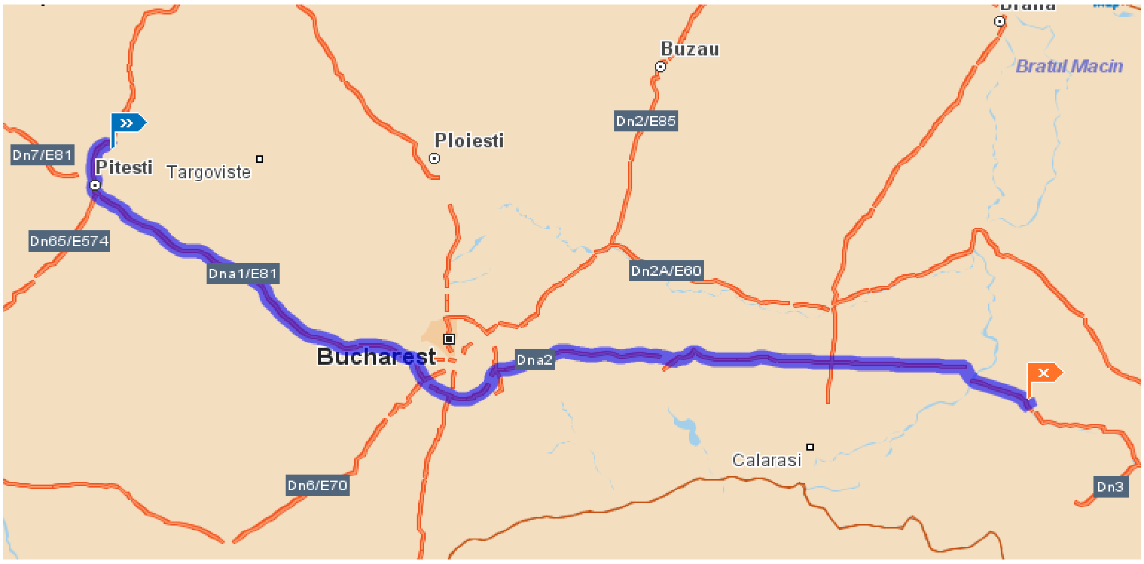

Fig. 1. The itinerary of transportation for nuclear fuel bundles.

5. Performing the transportation of nuclear fuel bundles

The consignor and carrier is FCN and the consignee is Cernavoda NPP.

For radiological safety purpose a Radiation Protection Programme (RPP) for nuclear fuel bundles transportation was developed in the **Quality Management System (QMS)** part of **Integrated Management System (IMS)** in FCN. The principles for elaborating and implementing the RPP for radioactive material transportation in FCN are detailed in **Manual of Radiological Safety** (MSR).

Additionally, the transportation is escorted and physical protection assuring by guard team, special trained for this type of transportation.

From FCN several teams are dedicated to the activity for sending the nuclear fuel bundles to Cernavoda NPP.

5.1. The transportation team, is formed by :

- Transportation chief ;
- consellor ;
- driver 1 ;
- driver 2 ;
- mechanical worker ;
- health physicist.

5.2. The team involved in packaging, labelling and placarding

5.3. The team for loading the truck

5.4. The team for surveillance the loading

Fig. 2: the truck for radioactive material transportation.

6. Description of the final product and package

The final product for FCN is the nuclear fuel bundle CANDU-6 type with 37 elements based on natural and depleted uranium.

The main characteristics of nuclear fuel bundle CANDU-6 type are:

Table no. 1. Description of the package.

The description for package with nuclear fuel bundles		
Package with nuclear fuel bundles	**1.**	**Nuclear fuel bundle CANDU 6** (natural uranium and depleted uranium).
	2.	**Polyethilene foil for wrapping each nuclear fuel bunlde.** Each nuclear fuel bundles is wrapped in polyethilene foil and is streching with a plastic ring. Over the third layer is putting a polyethilene foil
	3.	**Collective from polystiren and plaques separated made from cartoon**
	4.	**Container for transportation (wooden crate)** The wooden crates are made from wood

7. Description of package, storage and discharging the wooden crates

7.1. Parameters of package [7] are described in Table no. 2.

Parameter	Unit	Wooden Crate
Height	M	**0.6**
Width	M	**1.108**
Length	M	**1.146**
Diameter	M	**-**
Surface area of package	m2	**3.92**
Volume of package	m3	**0.76**
Cross section of a package (external)	m2	**0.68**
Cross section of a load (external) m²	m2	**10**
No. of package per load, large truck	-	**20**
Loads per day, large truck	d-1	**0.05**

7.2. The wooden crate with 36 nuclear fuel bundles is the package for transportation. The package is authorised by CNCAN [1] and it is stored temporary in the storage for fresh nuclear fuel bundles (DCNP), storage provided with sprinkles and under video surveillance

7.3. The loading of wooden crates in the truck is the responsibility of FCN. Four persons are involved in this activity (loading team). The discharging of wooden crates is the responsibility of the Cernavoda NPP employees. The Fig. 6 is showing an image of these discharging.

Fig. 3. discharging of wooden crates.

8. Organization for nuclear safety in FCN and transportation of radioactive material

The top management of FCN is focused for the major priorities of plant to quality control, nuclear safety and environmental protection. The system applied by management is **Integrated Management System (IMS)** that has until this time three components: **Quality Management System**, **Environmental Management System**, **Labour Health and Safety System** all based on international standards: ISO 9001:2005; ISO 14001:2005; ISO 18001:2006. According a special place for nuclear safety, the management has proposed for a new organizational chart of FCN.

Taking in account the priority of safety starting with 01 October 2007, FCN has a new organisation chart creating a new department dedicated to nuclear safety. So, it was created the Nuclear Safety Department (DNS) reporting to FCN Director.

The main DSN tasks are to apply the FCN policy, the SNN-SA nuclear safety policy and the Romanian nuclear safety policy. These are:

- To protect employees
- To protect visitors and contractors
- To protect population and public
- To protect environment
- To assure security and safety for nuclear material during the transportation
- To train the participants of the radioactive material transportation
- To inform the public

9. Radiological safety/radioprotection

9.1. Exposure of the participants at transportation

It is obviously that in the condition of doubling the capacity of the plant the doses monitoring of personnel involved in nuclear fuel bundles transportation for comparison became important for radioprotection programme [6].

All the participants at the nuclear fuel bundles transportation are from category A of exposure and they have an estimation dose of 6 mSv/year to be received.

During the transportation, the nuclear material is presented in itemized form so only the gamma dose rate and dose is important for evaluating the exposure to the ionising radiation.

1. External Exposure. The doses arising from the presence of nuclear material based on natural uranium are not high. The monitoring is performed with TLD (Termoluminiscent Dosimeter) PANASONIC type. Changing of TLDs, measurement, interpretation of doses, assessment, irradiation of TLDs, and reports to sanitary authority and CNCAN are done by DNS, Laboratory for Personal Dosimetry.

FCN Laboratory for Personal Dosimetry has the accreditation from CNCAN [1] for this type of activity. The results obtained from an experience of 125 nuclear fuel bundles transportation from FCN to Cernavoda NPP is presented in Table no. 3

Table no. 3. Doses measured during one transportation.

No.	Participants	Minimum value of Dose (µSv)	Maximum value of Dose (µSv)	Medium value of Dose (µSv)
01	Transportation chief	0.9	6.4	2.46
02	Consellor	1	6.2	2.52
03	Driver 1	6	18.9	7.99
04	Driver 2	3.8	13	6.13
05	Mechanical Worker	1	1.3	1.11
06	Health physicist	1	6.2	1.62

Compared with the registration level (RL) of doses imposed by CNCAN norms [8], the doses received during transportation don't exceed this value.

2. Internal Exposure. Being a facility where nuclear material is processed and handled in bulk form, taking in account the contamination hazard of air and surfaces, the internal exposure is very important and must be treated with carefully. These is reflected in contamination measurements of the packages and vehicles, contamination which can cause the releasing of uranium particles in airborne

3. Contamination Monitoring. Direct measurements (total contamination) and indirect measurements (loose contamination) on the surfaces of fuel bundles, wooden crates, trailers are performed. For these reason the contamination of packages and trailers is done after cleaning, at each shipment. The measurements for all contamination registered is under 0.1 Bq/cm^2

Table no. 4. The value of surfaces contamination

No.	Nuclide	Dose (mSv/a)/(Bq/cm²)	Surface activity leading to 1 mSv/a (Bq/cm²)	Medium surface activity measured	Fraction
01	U depleted	$4.8x10^{-01}$	2.39	0.00156	0.065%
02	U natural	$6.2x10^{-01}$	1.61	0.00177	0.109%

9.2. Public exposure times and distances

Occupancy times and distances for members of the public were allocated to the steps in the transport operations for each package type where the exposure of members of the public is likely/ possible.

The occupancy times will not be limited by working hours as in the case of workers; in one instance they are in excess of 90 hours/year. Members of the public will always be at a greater distance from the transport (10 meters in medium) operations than will transport workers for each step. Their exposures are, therefore, expected to be much lower than that of the transport workers. However, the dose criteria applicable to members of the public are generally lower and, therefore, their exposures should be assessed to determine if they are more limiting in some situations.

In this case the evaluation of the values for dose rate at the contact of the truck is relevant. The Table no. 5 show us that this dose rate at the maximum level don't imply a dose rate over the background at 10 meters from the vehicles.

10. Conclusions

1. The doses received during one transportation are very low and don't have the significance on the total exposure of the participants ;
2. Due the very low of contamination of wooden crate and of the surfaces of vehicle the internal doses have negligible values ;
3. The doses received by public during the transportation are very low in the accountancy of the dose effective limit for population .

11. References

[1] Law no. 111/1996 on the safe deployment, regulation, authorization and control of the nuclear activities, republished in 2006,

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[5] INTERNATIONAL ATOMIC ENERGY AGENCY, The Management System for the Safe Transport of Radioactive Material, IAEA Safety Standards Series No. TS-G-1.4, IAEA, Vienna (2008).

[6] INTERNATIONAL ATOMIC ENERGY AGENCY, Radiation Protection Programmes for the Transport of Radioactive Material, IAEA Safety Standards Series No. TS-G-1.3, IAEA, Vienna (2007).
[7] INTERNATIONAL ATOMIC ENERGY AGENCY -TECDOC-1449 Radiological aspects of non-fixed contamination of packages and conveyances, IAEA Vienna 2005
[8] NSR-06 – Norms for Individual Dosimetry, issued by CNCAN, 2002

Six years of radiation protection of operators in the General Electric FDG-radiopharmaceutical facility at the Joint Research Centre of Ispra

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Abstract

The Joint Research Centre of Ispra, one of the research Sites belonging to the European Commission, Directorate General JRC, was created in the late '50s, in order to steer European research on nuclear industry. It currently hosts numerous nuclear facilities, some of which are maintained in operation, while others were shutdown in past years or are currently being decommissioned.

Since 2003, a Radiopharmaceutical facility has been set up at the JRC Cyclotron Laboratory, in collaboration with General Electric Healthcare, for the production and commercialization of [^{18}F]FDG. This Radiopharmaceutical Laboratory has been the first FDG production facility to be officially authorized by the Italian Health Ministry and has been operating since 2003 without any interruption.

During these years, some upgrades in the capacity and maximum activity produced by the Lab have been performed. Two different beam lines and one irradiation vault have been devoted to FDG production in order to increase the production capability.

The paper, thus, is a review of Radiation Protection activities performed in the Lab during its years of operation and in particular the distribution of the dose related to the activity and the daily personal dose trend, subdivided between FDG production and Quality Control (QC), will be shown and discussed. Further, a recent internal contamination dose assessment campaign will be presented and the results discussed.

Introduction

Radionuclides played and still play an ever greater role in Medicine and especially in Nuclear Medicine, which is based on the use of radiolabelled tracers. Against this backdrop, a close collaboration between JRC and General Electric Healthcare started in 2003 to set up the first commercial [^{18}F]FDG production facility in Italy. The entire production system can be subdivided into three main parts: the cyclotron itself, under the direct responsibility of JRC; the target, in which ^{18}F is produced; the two FDG production laboratories, both of which are under the responsibility of GE Healthcare. To achieve satisfactory operational conditions, all these components required more than a year of fine tuning: in March 2004 the first FDG vials were commercialised. Since then, the system has undergone two major target updates: both its capacity and the irradiation current were increased and with them the activity produced. Until May 2005, ^{18}F production was on average between 80 and 100 GBq per run: it increased to a maximum of up to 180 GBq after the first upgrade and of up to about 300 GBq per run after the second upgrade in April 2008. Production may be run more than once per day. Of course, with the increase in the activity produced, the dose to the exposed personnel also increased, not only during the normal working activities, but also during ordinary or extraordinary maintenance.

Two operators staff the laboratory: one in the FDG synthesis room (Clean room) and one in the Quality Control (QC) room. They can receive dose during any type of target maintenance performed in an irradiation room, a target foil exchange, accidents and normal operation. This last contributor is then subdivided into dose from the Clean room - where FDG is synthesised - and dose from the QC room - where the biological analyses are performed.

About once a month, it is necessary to change the foils of the target because further deterioration would lead to impurity releases that may affect FDG synthesis. Before April 2008, target foil maintenance was performed on Fridays, in the late afternoon. Starting in April 2008, new Radiation Protection requirements have been introduced to further reduce exposure of the personnel. First, the target exchange is performed on Sundays nights, at 10 p.m., when decay after the last irradiation, which took place on the previous Friday, has reduced the dose rate inside the irradiation room and the target itself. Second, a short briefing (during which the type and length of work are concisely reviewed), the mapping of the dose rate distribution inside the irradiation room (to identify the different hot spots close to the working position and advise the personnel involved recommending, when necessary, the use of suitable shielding), and preparing the tools before starting operation (to avoid spending time unnecessarily in the irradiation room resulting, again, in an effective reduction of exposure) have become standard practice. Also, the yield of the FDG synthesis modules (i.e. the FDG quantity produced per unit activity) was improved, thus resulting in less activity being left in the modules after each run. Further changes were introduced inside the QC room. The ambient dose inside the room was reduced shielding the bin used for disposable items such as pipette tips, vials, etc., the plastic pipe of the High Pressure Liquid Chromatographer (HPLC) – which carried a large amount of activity – and a pH meter.

During the production of [^{18}F]FDG, the operators incorporate minute amounts of ^{18}F due to trace levels in ambient air and slight contamination of the tools. For this reason, γ emission at 511 keV from the body tissues of the operators slightly increase during the FDG production week. The scale and time dependence of this phenomenon was studied by measuring the internal body radioactivity at JRC-Ispra's Whole Body Counter (WBC) Lab. The actual level of internal radioactivity together with the identification of the nuclide and information on its distribution - incorporation time etc. - allows an assessment of internal dose and the associated long-term risk to the workers.

The aim of this work is to show monitoring methods, how the operators' dose changed over time with the amount of activity produced and also to provide a quantitative estimate of the impact of Radiation Protection interventions.

Material and methods

To better identify the areas in which Radiation Protection interventions would be most effective, dose contributions were separated in 5 categories: routine, scheduled target exchange, ordinary maintenance, extraordinary maintenance and incidents.

- *Routine operation*: this is the dose to the operators from FDG handling. This dose is received in the Clean and QC rooms. In routine conditions operators do not need to access the target room.
- *Scheduled target exchange*: this is the monthly target exchange. It may be performed by the Clean room operator or the QC room operator.
- *Ordinary maintenance*: it is performed, every year, in three well identified time periods, and involves maintaining the targets (foil and O-ring exchange, helium leak test) and the synthesis modules.
- *Extraordinary maintenance*: it is performed when beam line or target malfunctions occur.
- *Incidents*: in this context, "incident" indicates unforeseen circumstances that should present inside Clean or QC room.

All daily dose data were measured with electronic personal dosimeters (EPD. Model DMC 2000 S, MGP Instruments, France). EPD data are normally read out with the associated time stamps any time the operator leaves the controlled area but are reported as day by day aggregates. However, since a calendar day typically spans two different shifts (i.e. previous night and following evening), to assign dose contributions to the correct categories, it was key to cross-check the time data with the Radiation Protection Register in which the activities performed during all shifts could be found. The dose breakdown is reported in pie charts separately for the QC and Clean room.

The trend of the routine dose inside the Clean room (where it depends on the total amount of activity produced) and the QC Room (where it depends on the quota of [^{18}F]FDG sent to analysis) was analysed next. Before April 2008 the volume of FDG sent to analysis was 1000 μL, with an average activity of 1.5 GBq (1.5 MBq/μL); after the upgrade it was 600 μL, with an average activity of 2.5 GBq (4.2 MBq/μL).

Daily dose vs. time was investigated for both the QC and the Clean room before and after the target upgrade of April 2008. In both cases n dose points were studied with two different techniques: the t-test specific for unequal sample size and unequal variance (Zimmerman W.D., 1997) and simple linear regression (Douglas C.M. *et al.*, 2001) with a least-square method. Linear regression yielded the slopes of linear fits, which were indicated with β. Confidence intervals for β were derived under the hypothesis of a normal distribution of the errors: this allowed the construction of the interval as the product of the (1-γ/2)-th quantile of the Student's t-distribution with ($n-2$) degrees of freedom and the estimated standard errors. γ was chosen to be equal to 0.01, so that the confidence levels were equal to 99.5%.

In the case of the Clean room, further considerations were possible. First, the activity vs. time data were fitted separately before and after the target upgrade. Then, the daily dose normalized to the activity produced was also studied as a function of time, again keeping track of the time at which the target was upgraded. It is important to specify that in this analysis only the productions between Tuesday and Friday were taken into account. In fact, for data consistency, Monday productions were excluded because the activity to which data would have been normalised would have been artificially low due to the extended decay period (to the previous Friday) inside the modules.

For the campaign of internal dose assessment both operators underwent WBC examination daily from Monday to Friday directly after the end of the production shift, and in the morning of Saturday and Sunday. In this last case, the measurements were performed to investigate the variability of the ^{18}F level with time. During the WBC examination, qualitative and quantitative assessments of internal radioactivity were conducted with a direct method. Both scintillation NaI(Tl) and HPGe detectors were used. A brief description of the characteristics of both detectors is presented in Table 1. The NaI detector provided information on the whole body activity, whereas HPGe detectors were employed for lungs counting. Comparison of the results so obtained yielded the nuclide distribution inside the body.

The geometry of the measurements is shown in Fig. 1. For the assessment of ^{18}F

Table 1. Selected characteristics of three detectors employed in WBC laboratory of Ispra site.

Detector Type / Code	Position	Energy Range [keV]	Efficiency at 511 keV	Energy Resolution at 511 keV
NaI(Tl) / det. 1	stomach	50 – 2500	$5.55 \cdot 10^{-3}$	43.46 keV
HPGe / det. 2	left lung	10 – 800	$4.01 \cdot 10^{-4}$	1.06 keV
HPGe / det. 3	right lung	10 – 800	$3.63 \cdot 10^{-4}$	1.88 keV

Fig. 1. The geometry of the WBC measurements on GE Healthcare staff.

internal activity, the 511 keV gamma line originating from positron annihilation was used. Body and lung counting was performed simultaneously for an acquisition time of 45 minutes during each examination. The acquired gamma spectra were stored on a dedicated PC and then evaluated to obtain the levels of internal radioactivity. A long time background measurement for all three detectors was acquired during the weekend before the WBC scans to correct for environmental background. Data were collected for 180000 s with the not contaminated Livermore phantom placed in front of the detectors to simulate the presence of the human body. The net peak areas at 511 keV were measured and corrected for environmental background. MDA calculations were performed using the method described by Currie (Currie, 1968). To calculate the operators' dose, additional assumptions were necessary:

- Since the exact time of intake was not known, the most conservative hypothesis (i.e. the intake occurred at the beginning of the night shift) was assumed. Indeed, intake occurred probably throughout the night and with

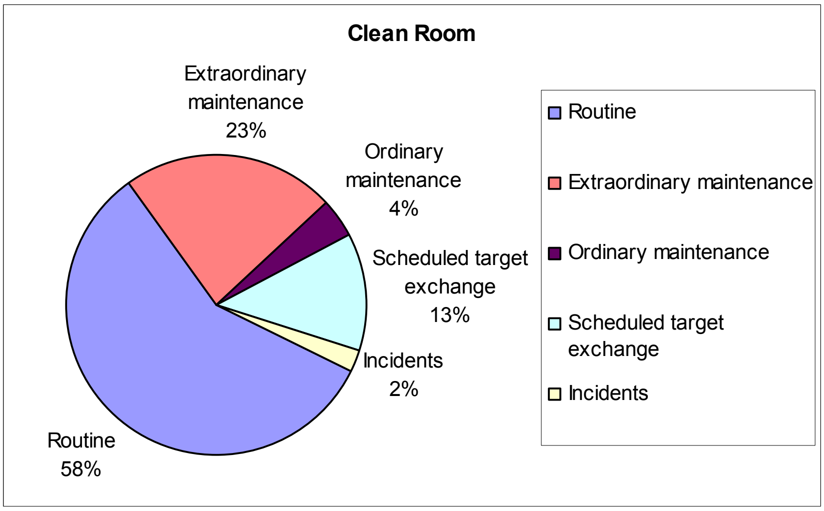

Fig. 2. Attribution of the total dose in the Clean room in the period from September 2007 to January 2010.

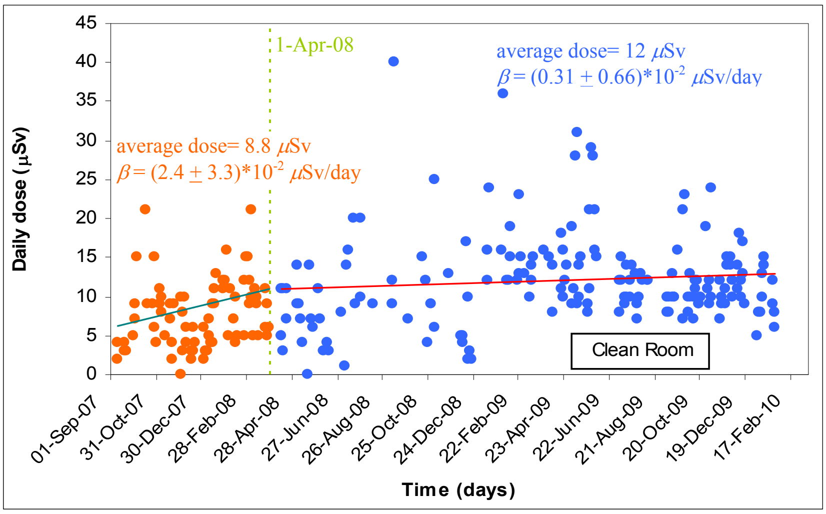

Fig. 3. Daily dose vs. time inside Clean room before and after the target upgrade of April 2008.

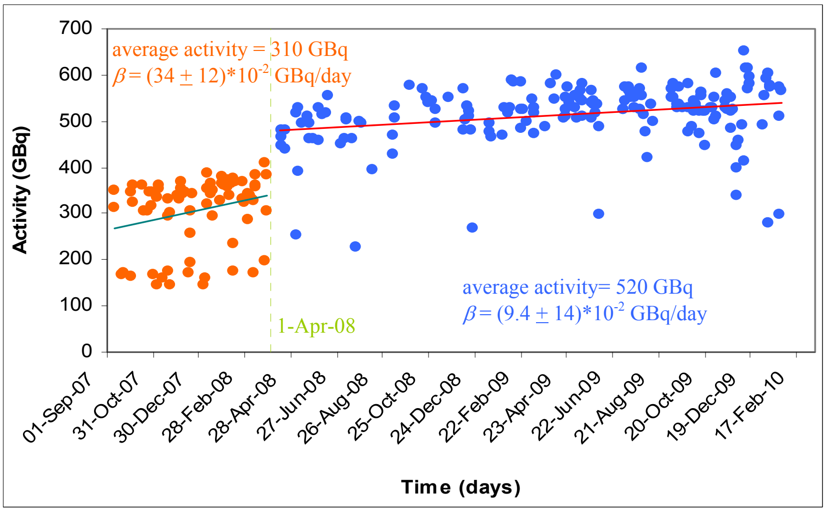

Fig. 4. Total activity produced during a night shift as a function of time.

different rates, depending on air concentration and task performed.

- The ^{18}F distribution was taken as "whole body contamination" according to results of WBC examination and in agreement with literature data (Calandrino et al, 2009).

Finally, the dose coefficient $9.30 \cdot 10^{-11}$ Sv/Bq was used to calculate the committed effective dose to the operators. This value is given by Italian law (D.Lgs. 230/95 All. IV Tab. IV.1) for ^{18}F intake by inhalation and whole body distribution of contamination.

Results

Clean Room

The attribution of the total accumulated dose in the period September 2007 – January 2010 is presented in Fig. 2: routine operation provided the greatest contribution (58%); extraordinary maintenance (23%) and scheduled target exchange (13%) are the second and third source of dose, whereas ordinary maintenance and incidents accounted for only 4% and 2%, respectively.

Fig. 3 shows the daily dose as a function of time before and after the target upgrade. Before April 2008, the slope ($\beta = 2.4 \cdot 10^{-2}$ μSv/day) is positive but not statistically different from zero (confidence interval: $-1.0 \cdot 10^{-2} \div 5.7 \cdot 10^{-2}$ μSv/day). After April 2008, β is still positive ($0.31 \cdot 10^{-2}$ μSv/day) and still not different from zero with statistical significance (confidence interval: $-0.36 \cdot 10^{-2} \div 0.97 \cdot 10^{-2}$ μSv/day). Before the target upgrade the average daily dose is 8.8 μSv; after it is 12 μSv. The t-test indicates that the difference in the averages is statistically significant with p-value = $3.7 \cdot 10^{-5}$.

The average daily activity produced before April 2008 was $3.1 \cdot 10^{2}$ GBq; after April 2008 it was $5.2 \cdot 10^{2}$ GBq (Fig. 4). The activity produced increased over time with statistical significance only before April 2008: $\beta = 34 \cdot 10^{-2}$ GBq/day (confidence interval: $22 \cdot 10^{-2} \div 46 \cdot 10^{-2}$ GBq/day); after April 2008, $\beta = 9.4 \cdot 10^{-2}$ GBq/day, (confidence interval: $-4.6 \cdot 10^{-2} \div 23 \cdot 10^{-2}$ GBq/day), which is statistically significant only to a confidence level of 89%.

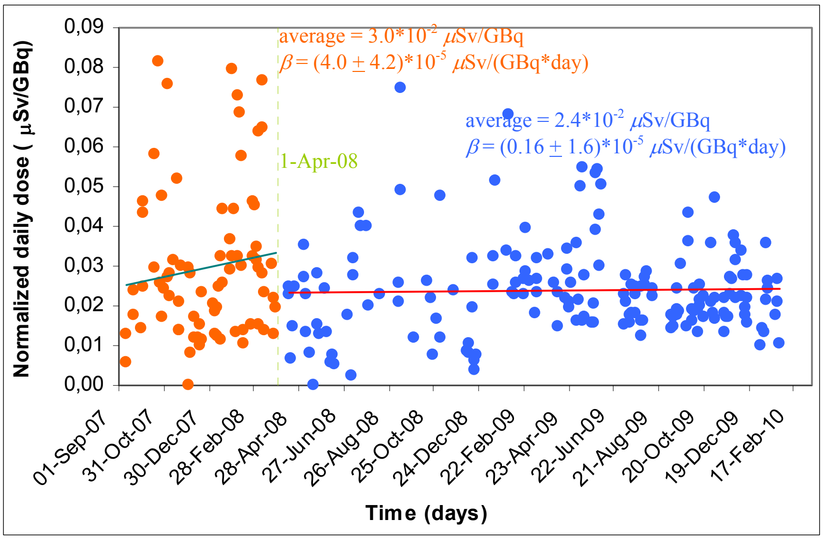

Fig. 5. Daily dose normalised to the total amount of activity produced in the previous working day vs. time.

Fig. 5 shows the dose normalised to the total activity produced as a function of time. Before April 2008 the increase is not statistically significant ($\beta = 4.0\cdot10^{-5}$ μSv/(GBq·day), confidence interval: $-0.26\cdot10^{-5}\div8.2\cdot10^{-5}$ μSv/(GBq·day)); after the target upgrade the slope of the curve is still not statistically different from zero ($\beta = 0.16\cdot10^{-5}$ μSv/(GBq·day), confidence interval: $-1.4\cdot10^{-5} \div 1.7\cdot10^{-5}$ μSv/(GBq·day)). Furthermore, the mean values of the two datasets are different: $3.0\cdot10^{-2}$ μSv/GBq and $2.4\cdot10^{-2}$ μSv/GBq, respectively – with a p-value = $5.8\cdot10^{-3}$.

Quality Control Room

The distribution of the total accumulated dose in the period September 2007 – January 2010 is presented in Fig. 6: routine, again, provided the greatest contribution (58%); scheduled target exchange (17%) and ordinary maintenance (16%) are the second and third source of dose, whereas extraordinary maintenance and incidents together accounted for less than 10%.

Fig. 7 shows the daily dose as a function of time: before April 2008 the average daily dose was 8.3 μSv and the graph presents a non statistically significant increase ($\beta = 5.6\cdot10^{-3}$ μSv/day, confidence interval: $-2.2\cdot10^{-3} \div 3.3\cdot10^{-3}$ μSv/day); instead, after April 2008, the mean daily dose was higher – 12 μSv – with β not different from zero ($\beta = -3.8\cdot10^{-3}$ μSv/day, confidence interval: $-12\cdot10^{-3} \div 4.1\cdot10^{-3}$ μSv/day).

Internal Dose assessment by WBC

The internal dose assessment of both operators was performed based on the whole body count results. Sample spectra are shown in Fig. 8. For both operators, Fig. 9 shows the changes of internal body activity due to ^{18}F, whereas Fig. 10 presents the calculated internal dose. The results of the dose calculations are presented in Fig. 10.

Discussion

The main contribution to the operators' total dose comes from routine, everyday work. Incident doses are, in practice, negligible, whereas scheduled target exchange, ordinary and extraordinary maintenance contribute in a different way depending on the operator. For the Clean room operator doses due to extraordinary maintenance are roughly one fourth of the total, but only about 8% for the QC operator. This is explained by the fact that these interventions are usually performed by the Clean room operator because he is the more experienced of the two and the dose rate inside the irradiation room and close to the target is usually much higher than during normal and scheduled maintenance. Since shielding has already been optimised, further reduction of this dose contribution can be only achieved via fast interventions and thorough planning in preliminary briefings.

The target upgrade resulted in a jump in the activity produced (Fig. 4), but not in a jump in the dose to the operator (Fig. 3). After the upgrade, the average activity production increased by about 70% (520 / 310 ≈ 1.7), while dose to the operator did not, remaining constant at about 12 μSv per day, a 40% increase over the average of the previous period (12 / 8.8 ≈ 1.4). This might be explained with the increased yield of the FDG synthesis modules. These considerations are also evident in Fig. 5, where it is

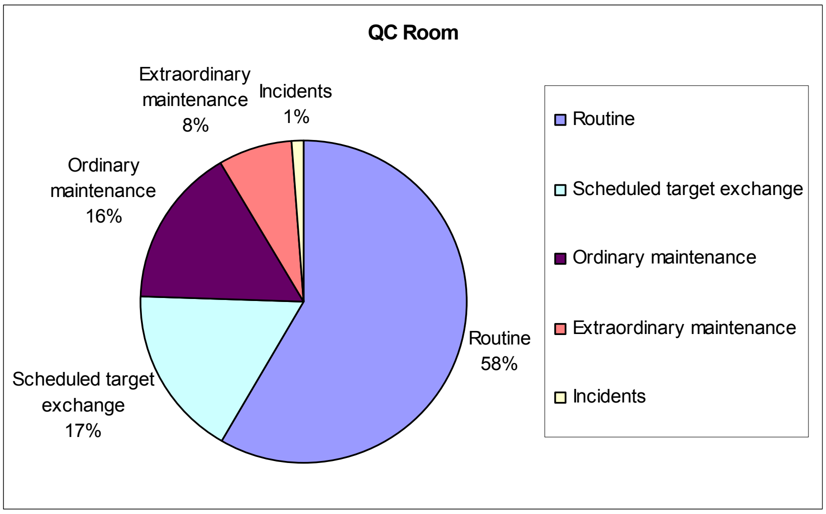

Fig. 6. Attribution of the total dose in the Quality Control room during the period September 2007 – January 2010.

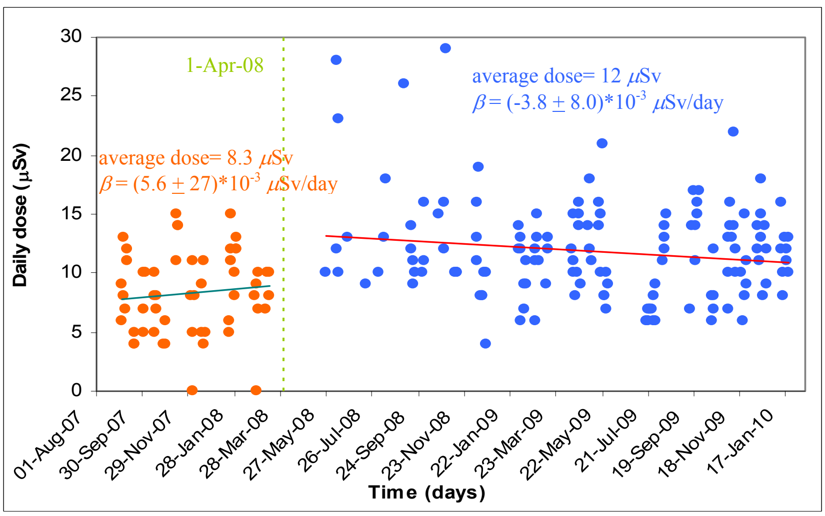

Fig. 7. Daily dose distribution vs. time inside Quality Control room.

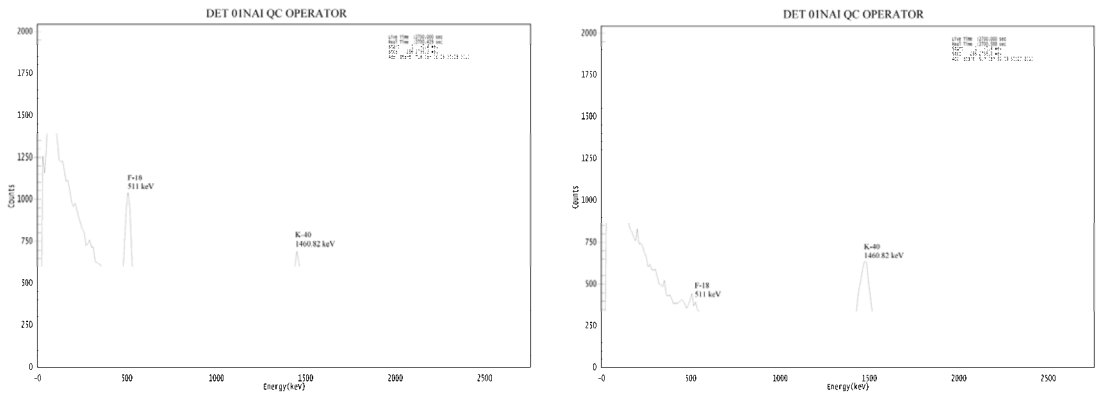

Fig. 8. The "highest" (left) and "lowest" (right) gamma ray spectra of QC operator body radiation.

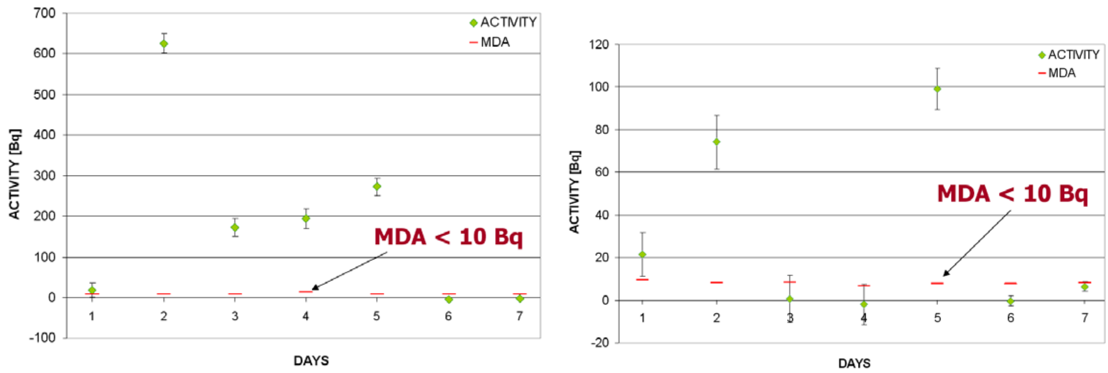

Fig. 9. The ^{18}F whole body activity of both, the QC – (left) and the production – (right) operators during the following days of survey.

shown that, after the target upgrade, the dose per unit activity suddenly drops by 20% ($2.4{\cdot}10^{-2}$ / $3.0{\cdot}10^{-2}$ = 0.80).

A jump in the dose to the QC room operator was also expected as the activity delivered to the room increased by about 1000 MBq, i.e. 1000 MBq / 1500 MBq ≈ 66%. Fig. 7 shows that the daily dose increased by about 40% (12 / 8.3 ≈ 1.40), and therefore less than expected from the increase in activity delivered. This might be explained by the additional shielding placed in the room.

The improved familiarity of the operators with the handling techniques are expected to show in the time trends of the normalised dose. To date, any improvements have not achieved statistical significance.

The values of measured body activity (Fig. 9) and also of the related dose received (Fig. 10) clearly show that, during FDG production, both operators incorporate just a minute amounts of ^{18}F, but the QC operator is a few times more exposed than the other. However, in both cases the doses are still at least two orders of magnitude less than the average dose due to natural background. During the weekend, when no production is performed, the observed activity was smaller and below MDA. It was possible to explain the maximal values of the activity measured during the production period to special events, such as more FDG produced or in vials checked by QC operators (this is the case of day 2); or later-than-usual begin of production due to machine related problems (day 5).

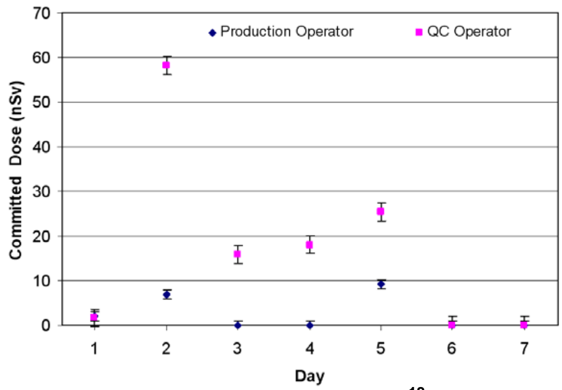

Fig. 10. Values of committed effective dose due to ^{18}F intake for both operators.

The WBC internal contamination assessment campaign was performed recently for the first time and its result is valid for the last period only. There are no reference values from the past. However, since there were no significant changes in protection against spread of ^{18}F contamination in the past and the amount of produced radioactivity used to be lower, it can be expected that the levels found in the present survey to also be representative of past internal dose.

Conclusions

The main contributor to operator dose at JRC-Ispra's FDG production facility is daily routine operation. Production malfunctions, in spite of potentially exposing personnel to the highest dose rates, in practice contribute to total dose only up to a few percent.

The target upgrades resulted in a significant increase of the activity produced (+70%). However, this increase did not result in a similar increase in the dose to the operators. In the Clean room, dose increased only by 40%, probably due to the improvement in the chemical synthesis modules. The activity delivered to the QC room increased by 66%, but dose increased only by 40%; this is likely due to the introduction of additional shielding.

The values of committed effective doses obtained from the WBC measurement campaign are very small. They confirm that ^{18}F internal contamination during routine FDG production night shifts results in negligible health hazards for both of the operators. This conclusion can be extended to all operators working in standard conditions.

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Radiation protection organization in the Joint Research Centre of Ispra

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Abstract

The Joint Research Centre of Ispra, one of the research Sites belonging to the European Commission, Directorate General JRC, was created in the late '50s, in order to steer European research on nuclear industry. It hosts numerous nuclear facilities, some of which are maintained in operation, while others were shutdown in past years, namely: two research nuclear reactors, hot cells facilities, radiochemical laboratories, one Cyclotron (still in operation), facilities for studies on fissile material (in operation), and some facilities for the treatment and storage of liquid and solid waste (in operation). The JRC accounts for 21 nuclear licences, 14 Controlled Zones and 12 main Surveilled Zones, on its Ispra Site. This paper will discuss the organization which has been developed, during the years, and put in place to guarantee maximum safety in all radiation work activities at the JRC-Ispra: the Radiation Protection Sector -which belongs to the "Nuclear Decommissioning Unit"-, the Radiation Protection Assistance Contract, and the role of Consultants and other External Contractors. The internal work authorisation processes, together with the work activities' preparation, assistance and reporting will be discussed: methods for the internal dissemination of experience and good practices will be reviewed. Some Laboratories give internal services to the Radiation Protection Sector: the Dosimetry Laboratory, the Whole Body Count Laboratory, the instruments' Calibration Laboratory, the Electronics Laboratory: their roles and functions will be presented. The Medical Service and the Radiotoxicological Laboratory give external support to the Sector: their role will be discussed. Emergency preparedness and response at the JRC will be reviewed, with emphasis on internal Emergency Squads training and integration. A brief outline of Radiation Protection permanent training organization at the JRC, and a summary of Radiation Protection archival methods will also be presented.

Support service of radiation protection in JRC Ispra (Italy) using the methodology applied in Spanish nuclear power plants

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Abstract

In the Joint Research Centre (JRC) of Ispra (Italy), there are different nuclear facilities, which are most shutdown and in a pre-decommissioning phase:

- Experimental dismantling reactors
- Research facilities to be dismantled
- Hazardous Liquid Waste Treatment Area
- Waste Storage and Decontamination Area
- Cyclotron, for F-18 production (still in use)

Since 2006, a Service Provider Company (LAINSA) and an Engineering company (IBERDROLA) have collaborated on support services of Radiation Protection in the JRC-Ispra. These companies have managed to introduce the operational radiation protection model applied in Spanish Nuclear Power Plants into the JRC.

The support service consists of the development of radiation protection assistance and technical consultancy, related with the following topics:

1. Assistance during Safety & Maintenance / Decommissioning Activities
2. Assistance during Nuclear Transport
3. Assistance in specific activities (SAS tents, glove boxes, hot cells)
4. Routine Activities
5. Electronic Laboratory
6. Whole Body Count / Dosimetry/ SIT+LMR Laboratories
7. Documental Activities, technical consultancy, formative actions

During the development of this project, a high quantity of radiological surveillances (surface contamination, air sampling and dose rate), spectrometry analyses, technical reports and RP procedures have been carried out, responding to the necessities of the JRC.

More than 200 radiation and contamination detectors (fixed and portable) are managed in the E-Lab. Routine operational controls, repairs and training courses are the main activities in the Laboratory.

Other RP services have been started during 2009, concerning the technical support to the Whole Body Count (WBC) and to the Dosimetry and Calibration Laboratories (SIT).

From the work performed in the Centre, excellent results are being achieved by a highly qualified team of professionals with different technical profiles, having a high ability for adapting to the changing conditions of the JRC policies.

Introduction

LOGISTICA y ACONDICIONAMINTOS INDUSTRIALES SAU (LAINSA), with IBERDROLA INGENIERÍA Y CONSULTORÍA, S.A, exposes in this proceeding the most significative work areas concerning radiation protection assistance (RPA) carried out from 2006 in the JRC-Ispra (Joint Research Center).

RPA activities have been divided in:

1. Assistance during Safety & Maintenance / Decommissioning Activities
2. Assistance during Nuclear Transport
3. Assistance in specific activities (SAS tents, glove boxes, hot cells)
4. Routine Activities
5. Electronic Laboratory
6. Whole Body Count / Dosimetry/ SIT/LMR Laboratories
7. Documental Activities, technical consultancy, formative actions

Methodology has consisted of assistance carrying out surface contamination, air sampling and dose rate measurements, fulfilling radiological surveys; routine controls; checking RP procedures at works with radiation and contamination risk; management of electronic instrumentation (i.e. contamination monitors, dose-rate monitors…); assistance in environmental measurements laboratory; assistance in SIT - Dosimetry - WBC laboratories.

Material and methods

Staff organization

LAINSA-IBERDROLA staff is composed by:

IBERDROLA (3)	LAINSA (20)
Site Manager (1)	Team Leader (3)
Team Leader (Deputy Site Manager) (1)	RP Technicians (8+1)
Team Leader (1)	Electronic Technicians (2) WBC/Dosimetry/SIT+LMR Technicians (3+1)
	Archive Support (1) Documental activities (1)

Team Leaders coordinate activities at works, being the interface of JRC Supervisors with RPA Technicians, communicating changes and difficulties, being proactive and developing RP procedures and technical reports.

LabEl Technicians assist electronic laboratory instruments and other equipments fixed on nuclear facilities, carrying out repairs and performance tests.

WBC/SIT/Dosimetry/LMR Technicians carry out activities such as supporting maintenance of whole body count detectors, calibrating electronic instruments and TLD dosimeters, verifying nuclear instrumentation, analysing samples and software.

Planning and distribution of RP Technicians must be flexible; therefore, movement of technicians between areas is important for delivering a right service.

1. Assistance during Safety & Maintenance / Decommissioning Activities

Decommissioning is the last stage in the life of a nuclear facility. JRC-Ispra is in a state of functional shut-down, and decommissioning activities are planned in the future. Safety & Maintenance activities cover efficiency tests to instrumentation (ventilation, radiation monitors, HVAC equipment…), absolute filters change…

Up to now, pre-decommissioning activities have been carried out in nuclear facilities, consisting mainly in nuclear transport, POCO (Post-Operational Clean-Out) waste management and radiological characterisation for waste inventory. Furthermore, waste management area (Area 40) has built several facilities for pre-treatment and measurements of radioactive waste.

RP Assistance, carried out by RP Technicians consists of verifying Personal Protective Equipments (PPE) by workers, prescribed in the Radiation Work Permit (Scheda di Lavoro): protective gloves, protective coverall, overshoes, mask, filter.

RP Technician will perform radiological surveys before carrying out works, including results in the corresponding *Scheda di Lavoro*. These surveys consist of measuring dose rate in work areas, removable surface contamination and air sampling. In the case of contaminated areas, a transit point will be included.

RP Technician, in special work conditions, will follow procedures assisting during dress/undress, controlling transit points, SAS tents, using radiation and contamination monitors.

Fig. 1. Radiological control of surface contamination with a Berthold LB124.

2. Assistance during Nuclear Transport

RPA during nuclear transport in the JRC-Ispra has consisted of:

- Measurements of removable surface contamination before opening empty casks
- Measurements of removable surface contamination of nuclear packs
- Measurements of dose-rate before transport
- Estimation of transport index

3. Assistance in specific activities (SAS tents, glove boxes, hot cells)

RPA in specific activities includes SAS tents, glove-boxes and hot cells. Ventilation and filtering are basic parameters for ensuring confined areas for working (engineering controls)

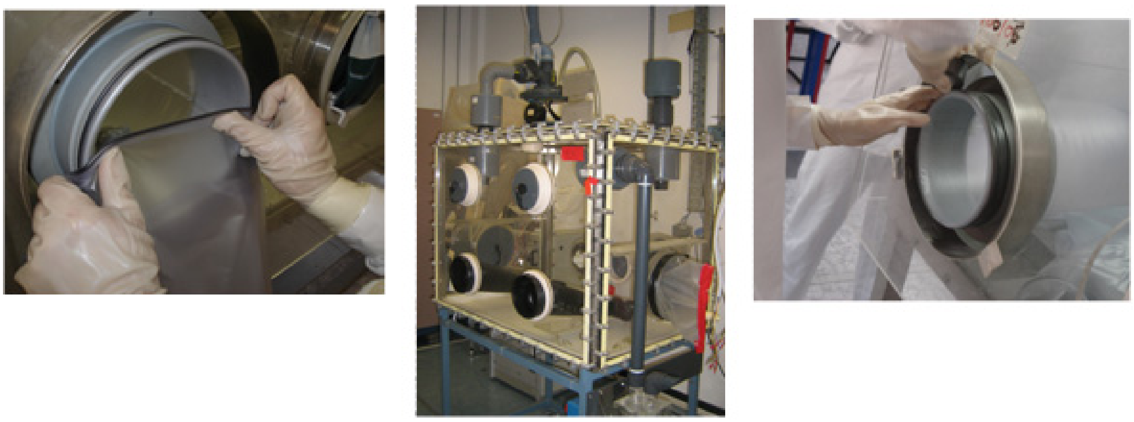

Fig. 2. Assistance to glove-boxes: sleeve and glove change.

Some activities require setting up SAS tents for radiation control, avoiding any Spreads of contamination (i.e. mechanical cutting, handling of radioactive materials, decontamination, clearance …).

Fig. 3. Assistance in a Hot Cell (ATFI).

4. Routine Activities

Some activities carried out for JRC-Ispra is the routine checking, defined by post-operational licensees and RP Sector: periodic removable surface contamination, dose rate measurements, air samplings, routine performance of nuclear instrumentation, environmental dosimeters change. Specific forms have been developed for each Controlled Area of each facility, identifying routine control points (figure 4).

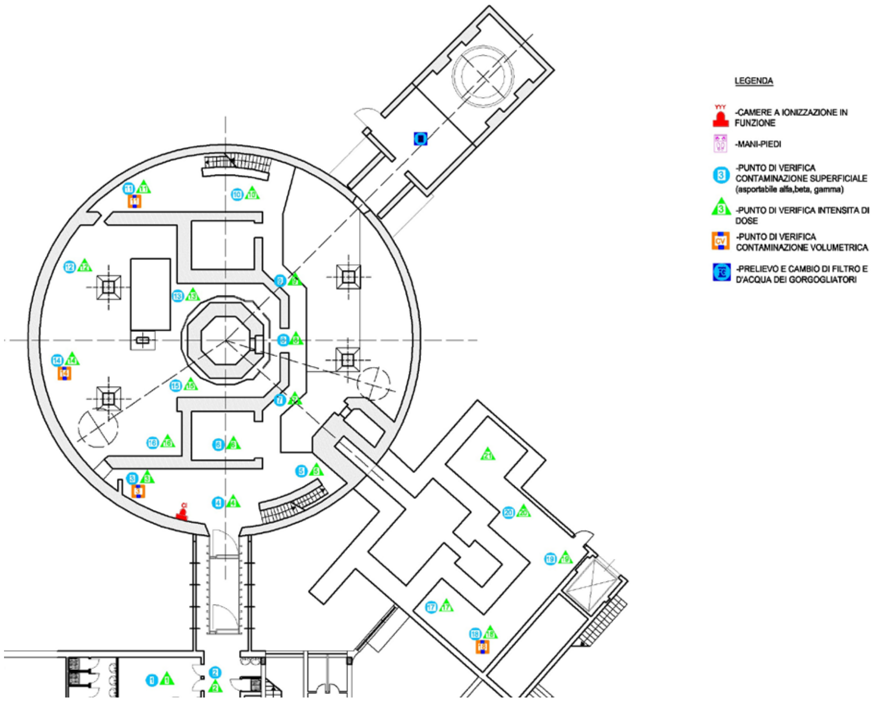

Fig. 4. Signaling controls to carry out trough simbols (Autocad 2007).

Fig. 5. RP Technicians carry out measurements in established points.

5. Electronic Laboratory

LabEl Assistance consist in repairing and verifying the right functioning of measurement equipments, with periodicity stated in Italian norms and JRC procedures. Some activities of LabEl have been listed:

a) Routine performance checkings
b) Electronic Calibrations
c) Repairs
d) Software development
e) Formative actions (i.e. courses)

Contamination monitors (Berthold LB124) are checked every day with alpha and beta/gamma radioactive sources, for verifying that efficiencies are in a valid range, checking that mylar is in good state and battery status. Furthermore, contamination on the contamination monitor is also surveyed. A check-up of state of hand-foot monitors is also carried out each week.

6. Whole Body Count / Dosimetry/ SIT + LMR Laboratories

RP assistance to external laboratories has been a request of Radiation Protection Sector during 2009, including previous assistance to LMR laboratory (environmental laboratory).

7. Documental Activities, technical consultancy, formative actions

Some procedures have been developed for several areas, being approved by the JRC-Ispra:

- Set up Transit Point
- Dress/Undress PPE
- Clearance
- Routine controls
- RP assistance in SAS tents

Radiological surveys forms have been developed for including removable surface contamination, air sampling and dose-rate measurements, including ALARA studies. A technical report in a Technical Gallery, used for channelling pipelines from discharge of hot cells. Radiation levels are showed in figure 5.

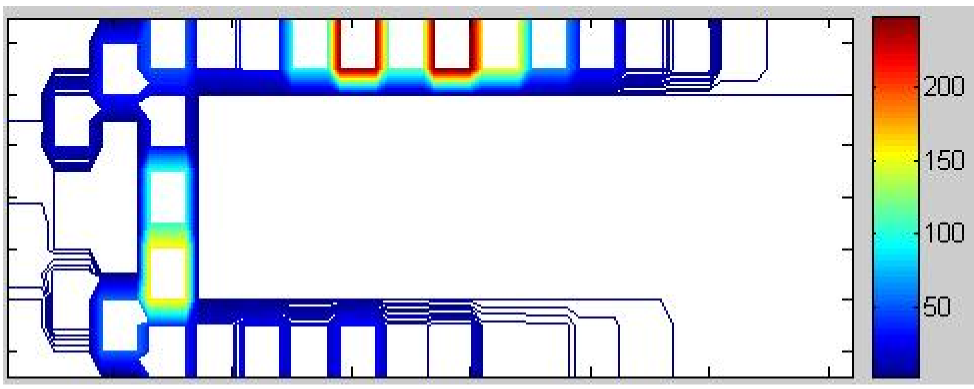

Fig. 6. Visual distribution of effective dose in a Technical gallery in μSv/h (Matlab 7.0).

Another report has been the development of a program for estimation of absorbed doses for staff during production of F-18.

Input	
	min
Tempo dalla fine del irragiamento	2000
>300 min	
Tempo in punti:	
C1	2
C2	2
C3	3
C4	3
C5	2
C6	2
C7	2
C8	1

Output	
	microSv
Dose assorbita	108,93
Dose assorbita nei punti	
C1	0,75
C2	1,34
C3	4,12
C4	11,80
C5	4,48
C6	8,84
C7	50,29
C8	27,31

Fig. 7. Program for evaluating dose to workers in Cyclotron facility.

Conclusions

During the development of this project, a high quantity of radiological surveillances (surface contamination, air sampling and dose rate), spectrometry analyses, technical reports and RP procedures have been carried out, responding to the necessities of the JRC.

Other RP services have been started during 2009, concerning the technical support to the Whole Body Count (WBC) and to the Dosimetry and Calibration Laboratories (SIT) and Formative Actions.

Effective doses received by staff have been lower than 1 mSv/worker/year.

From the work performed in the Centre, excellent results are being achieved by a highly qualified team of professionals with different technical profiles, having a high ability for adapting to the changing conditions of the JRC policies.

The pilot study of the radioactive aerosol particle size distribution in the air from the uranium mine Rožínka, Czech Republic

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Abstract

The radiological importance of the radioactive aerosol particles depends not only on the kind of radionuclide and its chemical form but also on the aerodynamic properties of the particles mainly on their size. Our aim was to determine the aerosol particle size distribution connected to uranium and its daughter products in the environment of the Uranium Mine Rožínka (Czech Republic) – the last place where the uranium is mined in the Czech Republic – in dependence on conditions of the mine and the ore mill. We also tried to estimate the percentage of radon emanation from the aerosol. The 6-stages cascade impactors were repeatedly placed in three places where the maximum dustiness was expected – close to the workers at the crushing plant, at the face of the shaft and at the collecting site at the end of the ore chute. Gamma spectrometric analysis was performed using HPGe detectors to determine the activities of radionuclides on each collection substrate from individual stages and on the back-up filter. The results of gamma spectrometry were used to evaluate the particle size distribution for individual radionuclides in terms of the activity median aerodynamic diameter (AMAD) and the geometric standard deviation (GSD). The first results from all places show that the distributions are slightly bimodal with the boundary under 0.4 µm and that AMAD is about 6 – 7 µm and GSD about 3 in agreement with the literature. The total activity concentrations were in an order of magnitude 10^{-3} – 10^{-2} Bq/m^3. The ratio of the activity concentration of the radon daughter products (^{214}Bi, ^{214}Pb) to the ^{226}Ra ranges around 0.5. Approximately 20 – 30% activity is attached to particles with aerodynamic diameter (AP) > 10 µm and more than 50% activity to particles with AP > 4 µm; about 10 – 20% of the activity is attached to aerosol particles with AP < 0,39 µm.

The characteristic of long-lived radionuclides in the uranium mine atmosphere in Dolní Rožínka in the Czech Republic

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Abstract

Miners of uranium mines are exposed by three ways: the irradiation of external gamma rays, the dose which is caused by the inhalation of RnDP (radon decay products), and also long-lived products of the uranium series. The presence of ^{232}Th and its decay products is not taken into account.

The derived limit for intake of the mixture of long-lived radionuclides which emit alpha particles is 1850 Bq per year according to the Decree valid in the Czech Republic.

The measurement for studying the mixture of long-lived radionuclides in the atmosphere of the last uranium mine (Rožná I) in Central Europe is realized as a part of the SUJ 200407 project. The mine is situated in central Moravia which is in the varied series of Moldanubic, represented mainly by biotitic up to amphibolites gneiss. This is affected by strong migmatization.

Measurements were performed in two typical workplaces underground (in the forefield, at the filling of corf) and one workplace in the crushing plant. The concentrations of radon and decay products (EER – equilibrium equivalent of radon) were also measured. The concentration and the properties of aerosol particles and the climatic parameters were also registered.

The mean concentration of long-lived radionuclides in the case of workplace A was 0.34 Bq·m^{-3} and in the case B lower - 0.06 Bq·m^{-3}. The mean concentration of radon was 5700 Bq·m^{-3} and the mean concentration of RnDP was 320 Bq·m^{-3} in forefield A. The low factor F (the ratio of EER and radon concentration) is caused by intensive ventilation. All parameters vary strongly influenced by the type of work operation.

Introduction

The total dose (E) for miners in a uranium mine consists of three parts: the effective dose from external photon irradiation (E_{ext}), the effective dose which is caused by the inhalation of radon daughter products ($E_{\mathrm{int},Rn}$), and the effective dose which is given by the inhalation of compound of the long-lived alpha radionuclides that are members of uranium series ($E_{\mathrm{int},alpha}$).

$$E[mSv] = E_{ext} + E_{\mathrm{int},Rn} + E_{\mathrm{int},alpha}$$

The concentration of long-lived radionuclides in the mine's atmosphere is influenced by many factors. The level of concentration is affected by concentration of long-lived radionuclides in ore and by working activity. The decrease of the concentration is affected by the intensity of ventilation and by the level of artificial humidification during individual working processes.

The results of personal dosimetry [1] in the case of uranium miners indicate that the intake of long-lived radionuclides contributes to about one half of the total effective dose. The results of personal dosimetry over the past five years are presented in Figure 1.

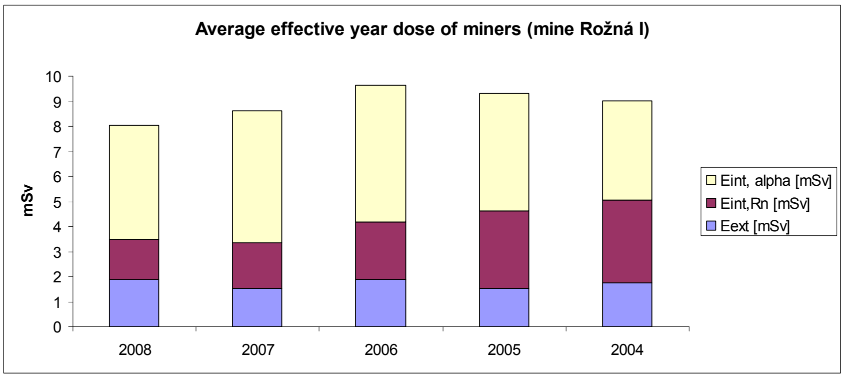

Fig. 1. Effective Yearly Dose for Uranium Miners in the Rožná mine.

The radiotoxicity of long-lived radionuclides which are members of the uranium family in the mine's atmosphere do not depend only on the total concentration of this compound, but also on the chemical form of radionuclides, on the composition of long-lived members of the uranium family, and on the aerodynamic characteristics of aerosol particles that carry attached radionuclides.

There were two underground workplaces (the stope and the place where the ore is placed in the mine truck) and one surface workplace chosen where one could anticipate the higher level of concentration of long-lived radionuclides. The surface place was the crusher house. This was the selected point beside the ore crusher with a permanent presence of a crew.

The size of distribution of aerosols particles should be divided [2] into three modes: the Aitken nuclei mode has a size range from 0.003 to 0.08 μm, the accumulation mode has a size range from 0.08 to 1 μm, and the coarse particle mode has the size range from 1 to 40 μm.

Material and methods

The radon daughter products were sampled on an open-face filter (glass fiber). The alpha activity of the filter was determined with help of the MAAF alpha spectrometer.

The concentration of radon was measured with help of instantaneous sampling of air put to scintillations chambers. The obtained sample was measured using a LUK device.

The climatic condition was determined with help of the Commeter climatic probe. The wind velocity was measured by the TESTO thermal probe.

The determination of concentration and size distribution of non-active aerosol was realized by CPC 3025 and SMPS 3071 (TSI). A setting was used that allowed to measure aerosols of diameters between 15 nm and 750 nm.

The determination of concentration of long-lived radionuclides in the mine atmosphere was carried out with help of two types of open-face filters (glass fiber and membrane filters). The determination of this activity was estimated by the MAAF semiconductor spectrometer and Canberra multichannel alpha-spectrometer.

The detailed sampling of active aerosol was performed by the Andersen 10-stages ambient cascade impactor model 2110K. The intervals of classification of aerodynamic aerosol diameters are listed in Table 1. The impactor was operated at 7 $l \cdot min^{-1}$ flow rate

Table 1. Series 210-Aerodynamic Equivalent Particle Diameter at 50% Collection Efficiency.

Impactor stage	**Cut-point diameter**
1	18 μm
2	11 μm
3	4.4 μm
4	2.65 μm
5	1.7 μm
6	0.95 μm
7	0.53 μm
8	0.32 μm
9	0.16 μm

A new method was developed for the determination of activity of individual stages of the impactor. This method is based on the interaction of alpha particles with sensitive surfaces of KODAK LR115 foil. The surface of detecting foil is covered by another foil, reducing the energy of alpha particles to needed energy (1-3 MeV).

Results

Most of the results were collected at the two places on the 21st floor (about 1100 m under the Earth's surface) in the Rožná I mine. These places were selected for their higher concentration of uranium ore.

The concentration of activity in the stopes was determined with help of the described instruments. The next tables summarize these results.

Table 2. The concentration of radon and EER in the A stope.

A Stope				
Time	**Concentration of radon [$Bq \cdot m^{-3}$]**	**EER [$Bq \cdot m^{-3}$]**	**F (= EER/concentration of radon)**	**Unattached fraction of EER**
16:00	4210 ± 132	682 ± 42	0.16	0.039
19:00	3166 ± 111	213 ± 25	0.07	0.183
0:15	11288 ± 206	295 ± 26	0.03	0.022
9:15	4182 ± 122	711 ± 46	0.17	0.014

Table 3. The concentration of radon and EER on the B stope.

Stope B			
Time	**Concentration of radon [Bq·m^{-3}]**	**EER [Bq·m^{-3}]**	**F (= EER/concentration of radon)**
15:25	1256 ± 87	57.1 ± 8.7	0.05
17:33	444 ± 53	38.9 ± 5.2	0.09
23:15	913 ± 74	22.2 ± 3.5	0.02
0:40	1085 ± 79	38.3 ± 5.2	0.04
2:40	334 ± 52	33.1 ± 4.6	0.10
7:26	1107 ± 79	53.8 ± 5.4	0.05

One can see that at both workplaces there was a high changing of air. The relative humidity on the stopes was about 90%, the average temperature was 23°C, and the mean air velocity was 0.21 m·s^{-1}.

The concentration of long-lived alpha activity was determined with help of two independent parallel open-face filters. In the case of the A stope, the concentration was 0.34 Bq·m^{-3}, in the case of the B stope, it was 0.06 Bq·m^{-3}.

The additional analyses were made using filters collected at the B stope. Filters were covered by foils, allowing the detection of alpha particles with different energies. One foil allows the detection of alphas emitted by ^{234}U, ^{230}Th, ^{226}Ra, and ^{210}Po and the second foil allows the detection of alphas emitted only by ^{210}Po. The detection foil was Kodak LR115. The advantage of this method is a relatively long duration for measurement. In the case of filters from stope B, a concentration of ^{210}Po was determined of about 3 mBq·m^{-3} and the total concentration could be estimated as 10 mBq·m^{-3}. So, the ratio between the before-radon and after-radon branch is 3.4.

The influence of self-absorption was not included; it is one of the possible reasons for differences.

The next table summarizes our results.

Table 4. The summary and comparison of results by various methods of evaluation.

Workplace	**Method**	**Concentration**	**AMAD**	**Concentration of nonactive aerosol**
		Bq·m^{-3}	µm	particle.cm^{-3}
B stope	Grab sampling, glass-fiber filter, 17 l/min	0.07		
B stope	*covered by foil to detect only Po*	0.003		
B stope	Grab sampling, membrane filter, 19 l/min	0.05		
B stope	*covered by foil to detect only U+Th+Ra+Po*	0.01		
B stope	Sum of results obtained by the Andersen impactor application – filters were covered by a system of foils (Po detected)	0.01	4,2	
B stope	TSI			19000 – 47000
Typical result after grab sampling in the case of uranium mines dosimetric service monitoring		0.1		
Typical result after use of personnel dosemeter		0.2		

The sampling with help of the cascade impactor was performed in the presence of the crew in the workplace. The volume of the sample was 4.1 m^3. The next graph documents the size distribution of airborne ^{210}Po.

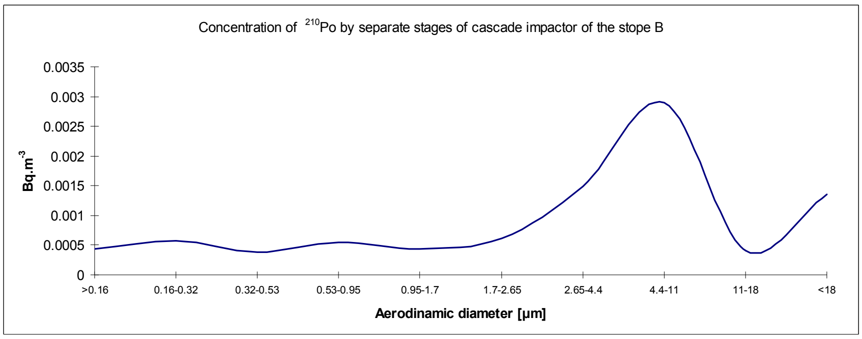

Figure 2. The results of evaluation of the cascade impactor.

The gained size distribution of active aerosol could determine the AMAD of about 4.2 μm and GSD of about 4.75. These results are preliminary and they will be confirmed with help of further measurements.

The determination of concentration and size distribution of non-active aerosols was performed in the B stope. The samples were taken during working activity, except for explosions. The sampling time was set to five minutes. The next graph shows the concentration of aerosol particles during separate working processes.

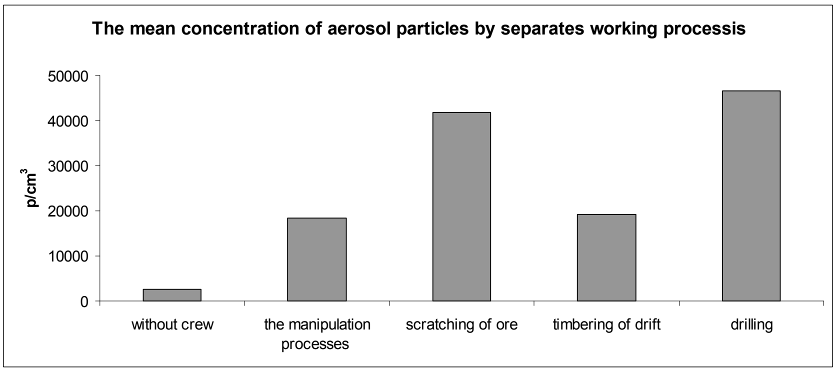

Figure 3. The typical concentration of aerosols particles by differents working proceses.

The typical size distribution during drilling is demonstrated in the next graph.

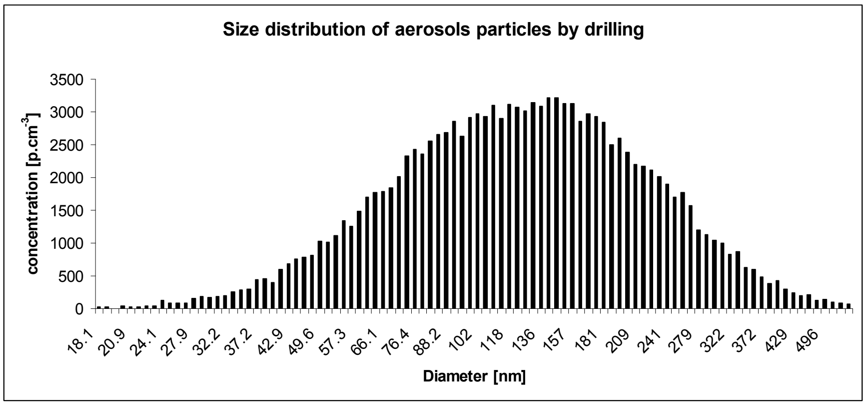

Figure 4. The typical size distribution of aerosols particles by drilling.

Discussion

To determine one of the important parts of the effective dose for uranium miners, it is necessary to know the detailed information about long-lived aerosol particle radioactivity.

With help of the TSI analyzer, it was determined that the size distribution of non-active aerosol is in the range from 15 to 750 nm. The maximum distribution is in the 80 nm area. The parameters were set in this way so that the aerosol coarse mode was not registered.

A new method was developed for the determination of activity of separate stages in the cascade impactor. It is necessary these methods confront and confirm new approach.

The coarse mode is observable with help of the cascade impactor. The maximum cut-point is 18 μm. The maximum size distribution of active aerosol is in the range from 1 to 10 μm.

The high concentration of aerosol particles causes the low unattached fraction of radioactive particles, of course.

Conclusions

The mean concentration of non-active aerosol particles was $4.5.10^4$ particles·cm^{-3} through drilling. The maximum concentration was $1.35.10^5$ particles·cm^{-3}. These values correspond with the published values by [3], there values less than 2.10^5 particles·cm^{-3} where registered. Working processes and the structure of rock influence the concentration and the size distribution of aerosol particles intesively.

The total concentration of long-lived alpha-emitting radionuclides varies of course with the concentration of these radionuclides in mining ore.

Acknowledgements
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ROBOSCAN: an advanced method and system that ensures a total radioprotection of the operators working with mobile vehicle scanners

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Abstract

The radiological scanning of vehicles containers is a non-intrusive inspection method capturing the interest of many current researches. It is realized by using a gamma ray source (Co^{60}, Cs^{137}, Ir^{192}) or a linear accelerator emitting a well collimated radiation beam. The radiation beam penetrates the inspected vehicle and reaches a system of detectors. An electronic system converts the radiation image into a corresponding visual image displayed on a high-definition monitor. The image can be saved and archived after visual inspection.

The detection system and the radiation source are mounted on a special designed vehicle that was approved for travelling on public roads. The vehicle is robotized and is remotely operated from a command centre placed outside of an exclusion area protected by infrared barriers. Any intrusion into the exclusion area is reported to the command centre and triggers the automated turn-off of the radiation emission.

Due to the fact that during the scanning operation the operator is situated outside the exclusion area, the radiation dose that he receives is situated within the limit of the natural background.

The paper presents the technical achievements, the radiological safety measures and the operation of ROBOSCAN 1M, a scanning system that implements the concept of total radiological protection of the operator. Such systems are currently used by custom officers on the Romanian borders for inspecting the vehicles and containers entering the European Union. Using personnel radiation monitoring equipment we proved that the equivalent dose is under the limit for the public. The obtained values are close to the radiation natural background and confirm the theoretical estimations and the efficiency of the radiological protection provided by the ROBOSCAN concept implementation. ROBOSCAN has won in 2009 the Grand Prix of the “Salon des Inventions” in Genève, and the “WIPO Best Inventor 2009 Award” at the International Warsaw Invention Show.

Introduction

The increasing growth of terrorist activities, contraband and illegal transportation drives the demand for high performance security systems. Special attention needs to be paid to the area of international cargo transports, which is usually the preferred vector for the contraband with guns, radioactive and explosive materials, narcotics and other forbidden stuff, easy to be hidden inside large volume of goods. The daily increasing volume of transported cargo makes impossible to physically inspect all suspicious transports.

The nonintrusive inspection method and system, that radiographies containers, vehicles and train carriages without having to break seals, open containers or physical control may be used for the scanning of vehicles, to create a radiography, that can be evaluated and from which to result the nature and the quantity of the transported merchandise, to track down smuggling attempts or illegal transports of forbidden or undeclared products (drugs, explosives, weaponry, etc.), for antiterrorist protection, by scanning all vehicles that have access in restricted areas, like airports, maritime and fluvial harbours, border crossing points, access to secure buildings, military bases, etc.

With the purpose of nonintrusive control several scanning methods are known, for which the following radiation sources:

- Gamma radiation sealed sources, double encapsulated radioactive material like: Cobalt, Cesium, etc.
- X-ray generators or linear accelerators of X-ray, gamma radiation and neutrons.

The nonintrusive inspection system principle requires the irradiation of a detector area, linearly placed in front of a thin fan-shaped curtain of collimated radiation through which the scanned object is relatively moved. The detectors' electrical signals are analogically/digitally processed, to generate, line by line, radiography to be displayed on a PC monitor. The relative movement of the scanned object is realized by moving the object relative to a fixed scanner, or by moving the scanner relative to a fixed object. The operation of the entire system is realized from a control cabin, placed close to the scanner, cabin for which extensive radiation shield is mandatory.

This method has the drawback that it exposes the operators to the professional irradiation risk.

Currently, several nonintrusive scanning systems are known that include the technologies presented previously. One of these is the mobile imaging system MIS with gamma radiation, manufactured by the MB Telecom-Ltd., Romania, is presented in fig.1.

MIS, like all other mobile scanning systems, that are presently known, have the operator's cabin mounted on the chassis, exposing the crew of the system to the professional and accidental irradiation risks.

The operating of the known systems is very complicated, needing a minimum three person per shift crew, operator, driver and external supervisor, the last having the responsibility to direct the traffic of the vehicles that are to be scanned in the scanning area, as well as to prevent intrusion in the exclusion area, where the danger of irradiation exists.

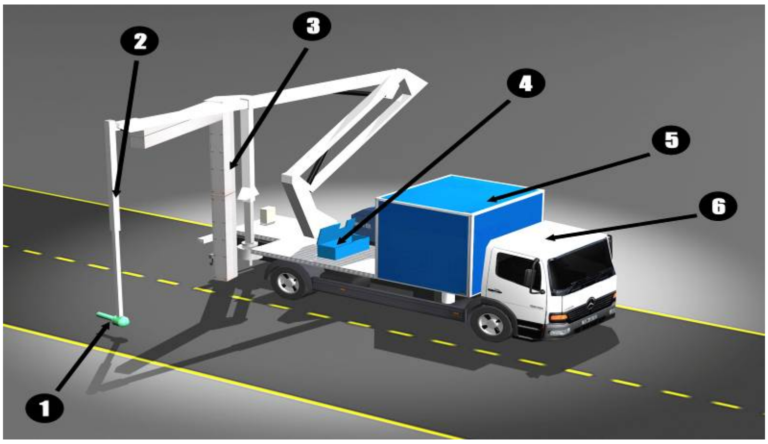

Fig.1. Mobile Imaging System (MIS): 1. Radioactive source in typ A package; 2. Source boom; 3. Detector boom; 4. Transport and repository container; 5. MIS command centre cabin; 6. Driver's cabin.

Material and methods

In accordance with ICRP 60 Recommendation and the IAEA Basic Safety Standard (BSS) the practice of radiographic non-invasive inspection is justified and requiring the optimization of radioprotection and the dose limitation.

This optimisation of radioprotection and dose limitation in the practice of radiographic nonintrusive inspection is implemented in ROBOSCAN - nonintrusive inspection method and system, has been patented in European Union (2006), Romania (2007), Eurasia (2008), Russian Federation (2008) and United States of America (2008).

The subject of the ROBOSCAN invention is the realization of a nonintrusive inspection method and system, that eliminates entirely the professional irradiation risk, by removing the operators cabin (the control centre) from the exclusion area and eliminating the need of a driver and external supervisor, by automation and remote operation of all processes deployed in the exclusion area and the limitrophe area.

The vehicle to be scanned has access in the exclusion area through an automated traffic management subsystem that automatically commands the functioning of the barriers and of the entry/exit semaphores. The vehicle is placed in a marked spot, before its driver leave the exclusion area (where there is the irradiation risk), then the protection of the exclusion area is activated, followed by the initiation of the scanning process by remote commands to the mobile scanning unit and the source, when the radiation source is activated and the slow and constant motion movement of the mobile units is started. This unit is moving rectilinear and uniform on parallel trajectories framing the scanned vehicle. The movement of the mobile unit is automatically controlled by electronic and informatics modules, connected to the control centre in a local area network, through radio modems, centre from which they receive commands, and towards which they send in real time status information and dedicated data. The stopping of the scan is performed automatically in the following situation, when the detector boom has passed the extremity of the scanned vehicle and the detectors receive the maximum level of radiation, at the end of the programmed scan length, when the protection limiter of the movement is triggered, when the protection of the exclusion area has been breached, when the proximity sensor has been triggered indicating dangerous distance between the detector boom and the scanned vehicle, when obstacles close to the guiding paths have been automatically detected by sensors placed on the mobile units. The stopping of the scanning process can be manually commanded by the operator in any moment. During this stage of the process, the image resulted from scanning the vehicle is displayed on the operator's monitor and at the end of the stage the protection of the exclusion area is automatically deactivated, and the vehicle may leave the scanning area. The mobile units move back to the start position and the scanning cycle may be restarted. The system includes also a mobile control centre, that is placed outside of the scanning area and that remotely manages all processes, including a subsystem for acquisition, processing, storage and displaying of scanned image. The system also includes an exclusion area protection subsystem, an automated traffic management subsystem and a computer management subsystem.

The automated traffic management subsystem is endowed with some barriers and traffic lights commanded by radio, directly by a dedicated software application and the exclusion area protection subsystem is made out of some motion detection active sensors, a control module for the sensor's status and an emergency automated radiation source shutdown module in the case that the exclusion area has been breached.

Results

In the first implementing variant ROBOSCAN 1M, presented in fig.2 "in transportation mode" and in fig.3 in scanning position, the major components are:

- Mobile Scanning Unit (MSU)
- Mobile Control Centre (MCU)
- Hydraulic Slow Motion System (HSMS)
- Source Boom
- Detector Boom
- Imaging System
- Automatic Protection of the Exclusion Area (APEA)

Fig. 2. View of ROBOSCAN 1M in transport mode.

The components of the unit (HSMS, source boom, detector boom) are assembled on a truck chassis, resulting MSU on extremely versatile and mobile robotized system.

An essential difference between all competitor systems and ROBOSCAN 1M is that doesn't need a driver to control the truck's movement (drive direction, sense, steering, brakes, vehicle's parameters, etc.). These functions are managed by a "driverless" subsystem that controls all the commands and parameters of the truck.

Fig. 3. View from the right of the ROBOSCAN 1M in scanning position.

In scan mode the operator initiates the scan procedure just by touching the corresponding virtual button, on the touch screen display, inside the MCU and the process is executed in a fully automated sequence, providing to the operator real time data by graphic animation and voice messages.

The MCU, organized inside a caravan that during scanning process is placed outside of the exclusion area and during transport, is towed by the MSU (see fig.4 and fig.2). The MCU manages all the peripherals and subsystems of the inspection system (imaging system, APEA, Graphic User Interface-GUI) in a fully automated process. In scan mode, all processes are managed by the software applications on a Wireless Local Area Network (WLAN).

The truck's transmission is modified by adding a dedicated HSMS capable of engaging direct or reverse drive with very low speed, continuously variable in the range of 0.1m/s up to 1.2 m/s. The operator can define any time the scan length and speed of the scan process. The slow motion in scan mode is initiated by the operator by simply touching an intuitive icon on the Graphic User Interface (GUI) on a touch screen display.

The return of the scanner to the initial position is automatic and the software application receives real time data concerning the position of the MSU inside the scanning lane. The MSU complies with international legal limits for axle load and dimensions, and doesn't need special license or authorizations to access the public roads.

In ROBOSCAN 1M the source boom content double encapsulated Cobalt 60 radioactive source, with the activity (A) of 37,0GBq in special shielded protection manufactured in steel and tungsten. The sizing of the shielded protection is made in accordance with instructions and standards for type A package (7), in accordance with IAEA categorisation of the radioactive source (8), with Co 60 source, in practices of industrial radiography sources, for $A=3{,}7\times10^{-2}$ TBq and dangerous sources for Cobalt 60 (D) $D=3\times10^{-2}$ TBq, the risk $A/D=1{,}24$. In accordance with (8), for $A/D<10$ the Categorisation of radioactive source corresponding for category 3.

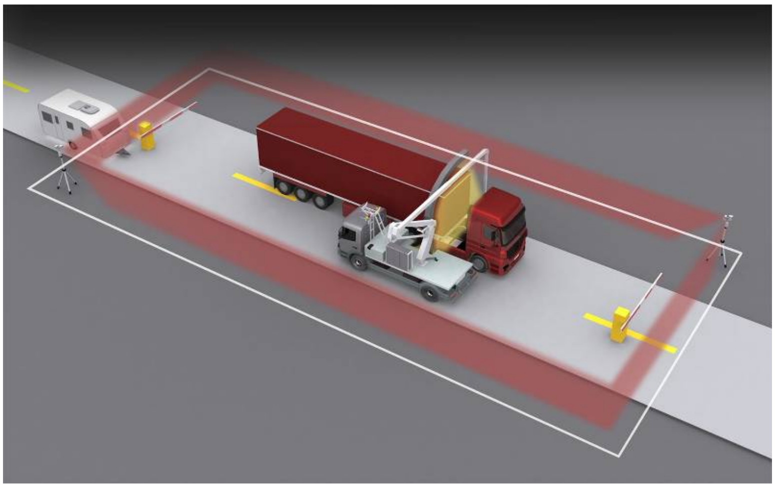

Fig. 4. View of the ROBOSCAN 1M nonintrusive inspection system, placed within the exclusion area.

The Cobalt 60 source is carried by a steel boom assembly which is displayed and stowed by hydraulic cylinders in automated sequences.

In Cobalt 60 shielding a fan-shape cut with an angular opening of 80 degrees is made, in order to collimate a radiation curtain at a width of approximately 18 cm on the detector areas, having placed the radiation source at a distance of five meters from the detector boom. The activation of the source is made through a pneumatic actuator system.

The system used, ensure the automatic retreat of the radioactive capsule in order to stop the radiation, if the actuator would be defective. The activation of the source is signalled acoustically and optically in order to warm the operator and any of the presence of radiation in the exclusion area.

The detector array is made of a special shape and structure of welded aluminium. The ensemble of the source boom and detector boom is automatically deployed and stowed in few minutes, by simply touching an intuitive icon, on the GUI.

The booms have been optimized as shape and structure in order to minimize their own weight and to maximise their performances.

Imaging system content the subsystem for image acquisition, designed to collect, process and analyze different data from the radiation detectors in order to generate the radiography of the scanned object.

The operator can apply on this image different filters and processing algorithms in order to improve the penetration or image quality, just touching intuitive icons on the GUI. The primary image is processed in a proprietary format and can be exported in bmp, jpeg, or other conventional formats (see fig. 5).

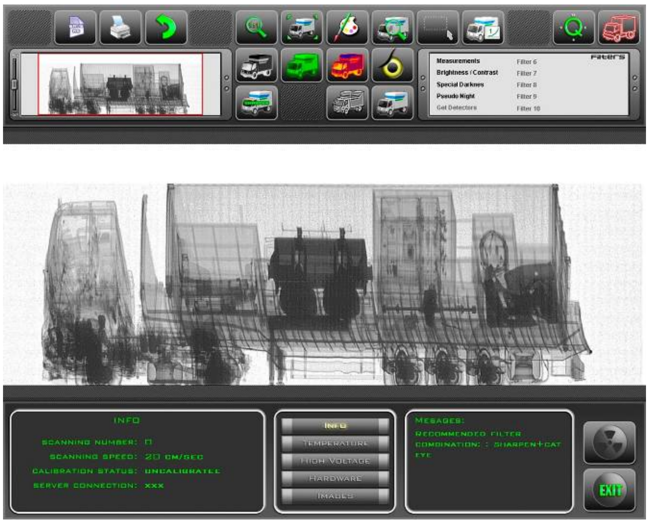

Fig. 5. View of radiography of the scanned truck in imaging software application.

The imaging software application provides multiple choices filters in order to allow optimal image processing like: Low/High Penetration filter, Cat eye, Owl eye, pseudo colour filter, Quick Filter, Pseudo Night and Special Darkness. All filters can be applied on the full image, or only on defined areas of interest.

The command and control software application is displayed on 2 monitors and manages the HSMS, video surveillance, perimeter protection, barriers remote operation and the automation of the scanning process.

All documents related with the transport, as well the front image of the scanned truck, the scanned image and the under view image of the inspected vehicle are saved in complex folder, having as ID the license plate of the inspected vehicle and the date/time stamp. In case of container inspection the container ID is introduced or automatically recognized and used as the folder ID. All commands and status of the sub-systems are registered in a “black-box” can be remotely administrated over a secure internet connection.

The exclusion area protection subsystem is an active radiological protection subsystem, which operates directly the source to automatically shut it down in the case that the exclusion area has been breached. The active sensors of the exclusion area protection subsystem, are placed in groups of two in the extremities of a diagonal of the exclusion area and angled 90 degrees one with the other, they create a virtual barrier two meters high and forty meters long, enough to limit a rectangular surface of maximum 40mx40m (L+30m) (L- is the length of scanned vehicle).These sensors are permanently radio connected with the control centre where they send an alarm signal in case of a breach of the infrared barrier. This signal automatically closes the source and activates a text, vocal and graphical message on the graphical interface of the operator’s software application, indicating the breached side. The subsystem is designed to work in difficult weather conditions like rain, snow, wind, dust, extreme temperatures, etc.

The subsystem for protection of the exclusion area is deactivated to allow the entry/exit in/out of the exclusion area synchronized with the working times of the barriers and when the driver of the inspected vehicle has left the area, the subsystem is reactivated.

The automated traffic management subsystem manages the barriers and the traffic lights placed at the entry and exit in the scanning lane in order to control the access of the vehicles that are inspected. This subsystem is controlled automatically by the operator's software application. On the operator's graphical interface live status info are displayed in real time. The commands and status are sent throughout some corresponding interfaces and radio modems.

The positioning of the mobile control centre outside of the exclusion area, as well as the elimination of the need for a driver during scan, eliminate all risks of radiation exposure and makes possible the reduction of operating crew from minimum three per shift necessary to any existing similar systems, to only one person per shift.

Discussion

Delimitation of the exclusion area, protected by IR barrier is made by the limited dose criteria bellow 0,1µSv, according with the ICRP and IAEA BSS. The resulting estimated dimensions are 40m by (L+30) m, where L is the scanned vehicle length.

The Co^{60} radiation source is protected by a multiple-barriers safety system, with interlocks and special keys accessed only by authorized operator.

According with ISO 3999 requirements, when the scanning is finished, the radiation source is protected by automatically returning it in docking safety position or interrupted due to any unauthorized intrusion in exclusion area or at any malfunction of electrical or pneumatically system.

When scanning process is finished the source boom is folded in transporting mode automatically by the operator command so the source is introduced in ROBOSCAN shielded box and automatically is turned on.

The driver's cabin radiation dose rate, due to the presence of radiation source in transporting mode is less than 0,1 µSv/h.

Any unauthorized intrusion to radiation source or at ROBOSCAN vehicle is turning on the alarm and triggers the guarding security personnel response.

Those measures fulfil the IAEA Security of Radioactive Sources requirements and ROBOSCAN User Manual contains the operating instructions specific to "Security Level B measures" covering all requirements for radiation sources classified in category 3.

The design of ROBOSCAN covers all radiological safety measures according with IAEA BSS and also with Romanian National Regulations, elaborated in accordance with 96/29 Euratom Directive.

The ROBOSCAN has obtained in 2007 the Radiological Safety License (ASR) from Romanian Regulatory Body (National Commission for Nuclear Activities Control).

Five ROBOSCAN 1M are currently used by customs officers on the Romanian border for inspecting the vehicle and containers entering the European Union. Using personnel radiation monitoring equipment we proved the individual equivalent dose for operators is under the limits for the public.

The obtained values are close to the radiation natural background and confirm the theoretical estimation and efficiency of the radiological protection provided by the ROBOSCAN concept practical implementation.

ROBOSCAN has won in 2009 the Grand Prix of the "Salon des Inventions" in Genève, and the "WIPO Best Inventor 2009 Award" at the International Warsaw Invention Show.

Fig. 6. WIPO AWARD for Best Inventor for the invention of ROBOSCAN and The Grand Prix of the "Salon des Invention" - Genève 2009.

Conclusions

Total radiological protection of the operator in scanning with ROBOSCAN system was previously demonstrated. An advantage it represents the reduction of the crew requirement to one single person per shift and the elimination of occupational radiation exposure for end user. Fully robotic automated operations, is key advantages for elimination of the risk of accidental irradiation of the possible intruders in the exclusion area.

The limitation of equivalent dose rate < 0,1µSv at the border of controlled area (exclusion area) is key advantage for radiation exposure of public and environment.

The capability of ROBOSCAN to a later analysis of the functioning parameters and/or operation sequences, by implementing a "black-box" facility, is key advantage for evaluation of custom officer activity and for inspection of users' activity by Regulatory Enforcement.

The optional extend the inspection capability by adding radioactive materials detection, under view video images and documents examination, in a single control procedure, increasing the performance of security control using ROBOSCAN.

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Special shielding solutions for the ITER neutral beam test facility

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Abstract

The Neutral Beam Test Facility (NBTF) will be located in Italy, near Padua, and will be devoted to the testing phase of the Neutral Beam Injector system (NBI) for the International Thermonuclear Experimental Reactor (ITER). NBTF includes two facilities that will be used to test in the same time both the main components of the final system and the whole system. NBTF includes two facilities that will be used to test in the same time both the main components of the final system and the whole system. The facilities are named respectively Spider (the small one) and Mitica (the whole system) and are located inside two distinct shielded bunkers. In each bunker important penetrations for the ventilation, the power supply and the auxiliary systems were arranged, providing that the safety of workers and population is guaranteed.
Tunnels in the underground region, with relatively large dimensions, were needed for both MITICA and SPIDER bunkers in order to allow personnel access for inspection and maintenance of cables and ducts.

Each of these shielding weaknesses needed specific safety analyses that were carried out both making reference to the scientific literature and performing ad hoc calculations.

In the paper the approaches applied to the different safety concerns due to the wall penetrations are considered and described. Reference analyses and computer calculations are reported together with the relevant results.

The outcomes in terms of neutron flux and associated dose data are compared with the regulatory limits and the project constraints, showing that positions and local shielding solutions for the penetrations allow the same level of safety for workers and population as that allocated by the main shielding walls.

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Introduction

ITER (International Thermonuclear Experimental Reactor) is a joint international research and development project that aims to demonstrate the scientific and technical feasibility of fusion power. ITER is a tokamak, in which strong magnetic fields confine a torus-shaped fusion plasma. The device's main aim is to demonstrate prolonged fusion power production in a deuterium-tritium plasma.

The experimental Neutral Beam Test Facility (NBTF) under construction in Padua (Italy) will be the testing station for the ITER NBI systems, hydrogen/deuteron particle accelerators based on negative ion technology operating at 1 MeV and 40 A.

The neutral beam test facility (NBTF) is composed by two main facilities: The facilities are named respectively SPIDER (Source for Production of Ion of Deuterium Extracted from Rf Plasma - ion source only) and MITICA (Megavolt ITER Injector Concept Advanced - the main system). Both injectors accelerate negative deuterium ions with a maximum energy of 1 MeV for MITICA and 100 keV for SPIDER, and a maximum beam current of 40 A for both experiments. The safety assessment for the design status of NBTF defined since 2007 was concluded at the beginning of 2008 by issuing a preliminary Site Specific Radiation Safety Report (SSRSR) [1]. Design adjustment and modification were proposed after SSRSR and require new safety evaluations and assessments.

The current work describes the activities performed in the frame of the F4E Grant F4E-2008-GRT-011-01(PMS-H.CD) to adapt the design of the components and the infrastructures for the NBTF. In the following the process for taking the final decision related to the wall shielding and penetration requirement also for the ventilation system is shown and the results are presented and discussed.

Main shielding walls

The general layout for the main buildings of the facility has been defined during year 2008. The main dimensions of the MITICA and SPIDER hall have been fixed. The working procedures and the needing for the staff shifts have been indicated by the research managing. According to this arrangements the main shielding walls have been defined in materials and dimensions taking into account the parametric assessments outlined in a specific report [1]. The shielding assessment was performed for the two main halls, for the different operation phases, considering the neutron radiation produced during D-D operations for some D beam energy scenarios. Neutron attenuation for standard and borated concrete was considered to determine the walls dimensions. In Table 1 and in Table 2 annual dose limits and constraints that have been used for the design are recalled.

In Table 2 the constraints proposed for the design of the NBTF facility were simply stated multiplying the limits by a safety factor of 0,5. The hourly constraints are obtained by considering 2000 hours of working time during each year.

Reference material for the shielding is the standard concrete. The borate concrete has been considered too expensive and regarded as a possible solution only in case space needing would be the main issue. A dose rate constrain of 0.25 µSv/h for the areas adjacent to the MITICA bunker was considered in order to meet the legal requirements for permitting free access even to non exposed workers into those areas.

Table 1. Dose constraints according to the NBTF RPP.

Project Guidelines	
Project guideline for annual individual worker doses	5 mSv/a
Project guideline for individual dose per shift	0.5 mSv/shift
Collective annual worker dose target averaged over life time of plant	0.5 pers-Sv
Interim ALARA Guidelines	
ALARA threshold for dose rates	100 µSv/h
ALARA threshold for collective worker dose to operate and maintain a system for a year.	30 pers-mSv
ALARA threshold for collective worker dose for a task performed less often than annually.	30 pers-mSv
Note, an 'ALARA threshold' is a level that triggers a formal ALARA assessment during the ITER design phase. This does not imply that ALARA reviews will not be performed when the design is below the thresholds.	

Table 2. Limits and constraints for the Radiation Workers.

Categories	Individual Effective Dose Limits (mSv/ year)	Annual Constraints (mSv/ year)	Hourly Constraints (µSv/ h)
POPULATION	1	0,5	0,25
CAT. B Radiation Workers	6	3	1,5
CAT. A Radiation Workers	20	10	5

In Table 3 the attenuation coefficient and the thickness calculated for the walls of the MITICA Bunker are recalled for the specific dose constraint in the adjacent areas, considering the 5 m reference distance from the MITICA vessel.

Table 3. Minimum wall thickness for the MITICA bunker.

	Attenuation coefficients	Standard concrete thickness (cm)
BL VESSEL	1,09E+06	174
BS VESSEL	1,71E+05	151
NB front end	1,22E+06	176
NB rear end	4,12E+04	133

A thickness of 176 cm of ordinary concrete is then required for the areas around the BL vessel and the front end of MITICA; minor thicknesses are acceptable for the portion of the bunker walls in the areas of the BS vessel and of the rear end. Therefore the following minimum thicknesses of ordinary concrete are recommended for the MITICA bunker walls:

- 180 cm from the floor and for an eight of 3 m from the MITICA symmetry axis in the front end and BL vessel areas,

- 155 cm from the floor and for an eight of 3 m from the MITICA symmetry axis in the BS vessel area,
- 135 cm from the floor and for an eight of 3 m from the MITICA symmetry axis in the rear end area,
- the upper part of the MITICA wall, over 3 m from the MITICA symmetry axis, could have a thickness reduced by 30% of the lower one

The evaluation for the shielding of the ceiling has been done in the previous analysis using the approach suggested by NCRP [2] and considering that the main neutron diffusion is due to the atmosphere above the bunker roof. According to this method the needed attenuation required to allow free access to non radiation workers is:

$5{,}0\ 10^2 / 0{,}25 = 2{,}0\ 10^3$

and the related minimum roof thickness is 95 cm of standard concrete. This thickness could be reduced by a 20% factor in the area of the ceiling corresponding to the BS vessel.

Considering that the simulation was performed with a simplified geometry and that not all the detailed structure of the MITICA surrounding was implemented into the model a more conservative approach was adopted in the final design. A 180 cm wall all around the facility was planned and a reduced thickness, of the order of 1 m, was established for the lateral walls in the dome area around the power supply line only. Due to special requirements for the connection of some auxiliary systems (HV and other components) in the upper rear part of the bunker, the wall thickness was reduced accounting for the needed room, meeting meanwhile the recommendation for the minimum shielding thickness reported above.

The SPIDER facility will be devoted to the testing phase of the source section only. The source will operate at a maximum voltage of 100 kV. This testing phase will take place in another area of the same building; thus an independent shielding is required. Even in this case a 0.25 µSv/h dose rate constraint was considered and the consequent shielding thickness for the SPIDER bunker were calculated. Results are shown in Table 4.

For the SPIDER bunker the following minimum thicknesses of ordinary concrete are therefore recommended:

- 120 cm from the floor and for an eight of 3 m from the SPIDER symmetry axis in the front end and BL vessel areas,
- 95 cm from the floor and for an eight of 3 m from the SPIDER symmetry axis in the BS vessel area,
- 80 cm from the floor and for an eight of 3 m from the SPIDER symmetry axis in the rear end area,
- the upper part of the SPIDER wall, over 3 m from the SPIDER symmetry axis, could have a thickness reduced by 30% of the lower one

Table 4. Minimum wall thickness for 100 keV energy.

	0.25 µSv/h
BL VESSEL	116
BS VESSEL	93
NB front end	118
NB rear end	76

For the same level of access a 90 cm ordinary concrete ceiling is needed as well. This thickness could be reduced by a 20% factor in the area of the ceiling corresponding to the BS vessel.

Also in this case the final design was defined according to more conservative considerations and some wall thickness was adjusted where needed only.

Dose evaluation through the walls penetrations

In the MITICA bunker some access-ways are needed to permit entry of personnel and equipment as well as some ducts for cables, radio-frequency (RF) waveguides and ventilation system. Radiation protection assessment for these wall penetrations has been made in order to define dimensions and possible additional shielding where needed.

Utility ducts are much smaller than personnel access penetrations (whose cross-sectional dimensions are about 1 meter by 2 meters, "door-sized"), typically no larger than 0.2 by 0.2 m. Often the utility penetrations are partially filled with cables and other items, and even cooling water in pipes.

Two general rules are advised for the penetrations of an accelerator shielding:

- A penetration should not be arranged so that the primary radiation field is aimed directly toward it.
- For any maze, the sum of the wall thickness between the source and the "outside" should be equivalent to that which would be required if the labyrinth was not present.

The first step in designing penetrations and ducts is to parameterize the reflections of the primary and secondary radiation field. These reflections can be treated through the use of reflection or albedo coefficients.

The MITICA tunnel for the passage of cables and cooling pipes underwent different modifications and more than one configuration was proposed and evaluated in order to meet the radioprotection requirements. In the following the two main solutions are reported along with some consideration and evaluation performed making reference to the related experimental studies available in the literature.

Typical methods for addressing the attenuation of radiation by penetrations involve the use of the results of calculations and semi-empirical formulas derived from Monte Carlo codes. These can be used for both curved and rectilinear labyrinths. Such formulas permit the dose rate calculation at the exit of the MITICA tunnel [4, 5, 6, 7, 8].

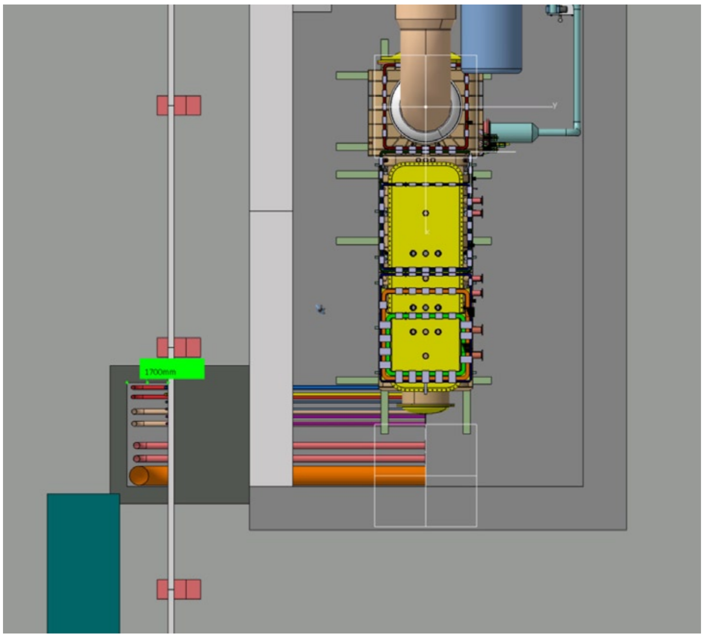

Figure 1. MITICA duct for the passage of cables and cooling pipes.

The tunnel lead to a service room whose dimensions are 10 m x 15 m and which is not accessible during MITICA operations; moreover no worker access is allowed inside the tunnel, and in the area around the tunnel exit during beam-on operations.

The annual average of the dose rate due to three interim campaigns and one final test without shielding walls was assessed in a previous work through Monte Carlo simulation with MCNP Code [3]. In the following the main results are summarized:

(Dose rates in Sv/h)	100 cm	500 cm
From the VESSEL (highest value)	5.41	0.89
From the front end	7.38	1.21
From the rear end	0.28	0.026

Conservatively, the highest of these values was considered (7.38 Sv/h at 100 cm from the front end). Knowing the dose rate at a certain distance from the source, the dose rate at any distance and for any geometrical angle from the beam can be assessed with simple point kernel equations of the form:

$$H(d,\theta) = r^{-2} \cdot H_\theta e^{-\frac{d(\theta)}{\lambda}}$$

which combines the inverse square law and an exponential attenuation through the shield. This equation applies to a point source of a specified radiation where:

- $H(d,\theta)$ is the dose equivalent at depth (d), angle (θ) in the shield
- H_θ constant that may be described as the dose equivalent extrapolated to zero depth in the shield, and corresponding to angle (θ), at unit distance from the point source

- r distance from the source to the point of interest outside the shield
- λ effective attenuation length for dose equivalent through the shield

Over a limited range of shield thicknesses, the radiation transmission could be approximated by an exponential function. For shields of thickness less than 100 g/cm^2, the value of λ changes with increasing depth in the shield because the more easily absorbed (''softer'') radiations are attenuated more rapidly. This process is often described as ''spectrum-hardening''. For the MITICA bunker only ordinary concrete shielding were planned (around 2.35 g/cm^3), with thickness of the order of 100-200 cm. The shielding is therefore always thicker than 100 g/cm^2. For neutrons of maximum energy equal to about 3 MeV (as those coming from D-D reaction), one has $\lambda = 15$ cm.

The source dependence in a tunnel of cross sectional area, A, can be approximated by the following expression:

$$H_1(\delta_1) = \frac{r_0^{\ 2}}{(\delta_1 + r_0)^2} H_0(R) \qquad r_0 = 0.4$$

where:

$$\delta_1 = \frac{d_1 - R}{A^{1/2}},$$

R is the distance from the beam axis (in this case R=0) and r_0 is a fitting parameter. $H_1(\delta_1)$ is the dose equivalent at distance δ_1 in the first leg as measured from the mouth of the passageway in "units" of the square root of the cross sectional area of the first leg. $H_0(R)$ is the dose equivalent at the mouth the determination of which has been previously discussed. Over the domain of $0 < \delta_1 < 9$ the expression is accurate within ± 10 per cent, sufficiently accurate for radiation protection purposes.

Second and successive legs of such "rectilinear" penetrations change the situation dramatically, mainly by modifying the spectrum of the transmitted neutrons. The following recursive expression, reported in NCRP [9, 10], adequately describes this curve, where, δ_1 is the distance in the i^{th} leg measured in "units" of the square root of the cross-sectional area of the i^{th} leg:

$$H(r_i) = \left(\frac{e^{-\frac{r_i}{0.45}} + 0.022 \cdot A_i^{1.3} \cdot e^{-\frac{r_i}{2.35}}}{1 + 0.022 \cdot A_i^{1.3}} \right) \cdot H_{0i} \quad i^{th\ leg}\,(i > 1)$$

The MITICA tunnel is composed by two legs. The first one is an horizontal duct at a level under the MITICA hall floor that goes from the hall to the area outside of the shielding wall. The second leg is a vertical duct that connects the end of the first one to the service room.

Configuration 1. The first configuration refers to a rectilinear 2 legs duct whose cross section is about 0.9 m (height) x 4.6 m (width). After the primary shielding, the tunnel cross section increases up to 2 m x 4.6 m.

For calculation purposes, (even if the first leg is made by a two separate parallel ducts whose height is 0.9 m) a single duct 1.5 m high was considered. This assumption is quite conservative since with these dimensions the cross sectional area of the tunnel is 6.9 m^2 (4.6 m x 1.5 m) rather than 4.14 m^2(4.6 m x 0.9 m).

Configuration 2. This configuration is different from the previous one only for the presence of a concrete shelf whose aim is to attenuate the radiation directly produced by the calorimeter. A concrete shelf with variable thickness (ranging from 0.5 to 1 m), 1.5 m long and 4.6 m wide was designed. Not all the primary neutron beam intercepts the concrete shelf; conservatively, it was assumed that half of the beam is attenuated by the shield, and half passes through the tunnel without suffering any collision (this assumption is confirmed by geometric consideration from CAD drawings).

Results of the calculation are summarized in the table below (Table 1).

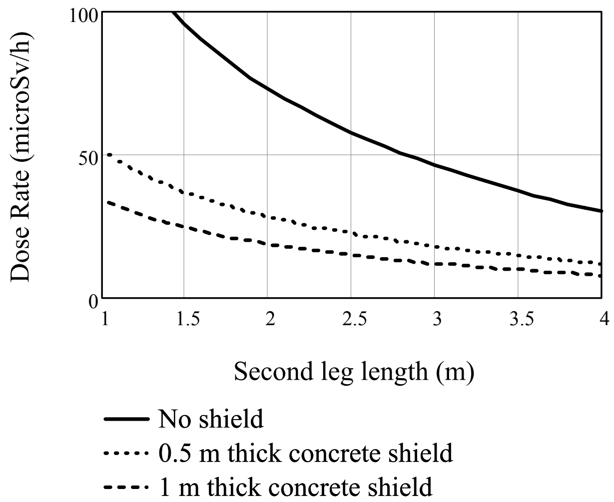

Table 1. Dose rate at the exit of the tunnel according to the different configuration and to different shelf thickness as a function of the second leg length. In the present case, the second leg length is about 3.5 m.

At the exit of the tunnel, i.e. at a distance of 3.5 m from the first leg end (tunnel length), the dose rate is:

$$D_{no\ shield} \approx 30\ \mu Sv/h$$
$$D_{0.5m\ concrete\ shield} \approx 12\ \mu Sv/h$$
$$D_{1m\ concrete\ shield} \approx 8\ \mu Sv/h$$

The presence of a concrete shield reduces the dose rate at the exit of the tunnel by a factor of 3-4, depending on its thickness. In this phase it was thus decided not to take into account a concrete shield, whose design may reveal unpractical and expensive, and whose contribution to the dose reduction is quite poor.

Further analysis was done to assess the dose rate beyond the tunnel exit and the general result showed that neutrons "disappear" rather rapidly because their energy spectrum is heavily dominated by thermal and near-thermal neutrons in all “legs” after the first. The contribution to the dose rate was found to be almost negligible and no additional shielding were designed.

The configuration for the MITICA ventilation ducts consists in Z-shaped ducts. In order to meet the air ventilation requirements, the sectional area of the duct should be about 0.7-0.8 m^2 per single duct. Since the primary shied thickness is 180 cm, the single sections will have a dimension L not bigger than 0.9 m. The number of duct is still to be determined.

The mouth of the duct A is 8.5 m away from the axis. The dose rate considered in the calculations is the highest dose rate obtained during final phase from Monte Carlo simulations. 5 meters away from the calorimeter, the dose rate during the final phase is 2.38 microSv/h and at the mouth of the duct A this value goes down to 0,82 microSv/h.

Z-shaped cylindrical ducts with 3 sections were considered. Different calculations were performed, implementing different duct sectional area sizes and different section length. Dose rate at the exit of the reference duct was calculated to be of the order of few microSv/h (μSv/h) or even lower.

Other penetrations such as an entrance, cable supply, exhaust ducts and pipes go through walls in MITICA facility. As already showed, penetrations are generally designed as bent structures in order to decrease leakage radiations (streaming) through them.

Generally the ducts designed for the NBI consist in long pipes which host cables of different kinds. The ducts leave the vessel from underneath going horizontally towards the primary shield. The ducts penetrate the main concrete shield and emerge with a vertical section from the ground of the MITICA building, adjacent to the bunker where workers are supposed to work/pass/walk during operation. The dose rate values obtained considering Z-shaped ducts are all negligible (< 1 pSv), and even the dose rates obtained with straight ducts are very low (less than 1 μSv/h), the last configuration was thus preferred due to its simplicity. Exit dose rate as a function of the diameter was also evaluated, showing that a duct diameter up to 20 cm yield dose rate acceptable at the exit of the passage. When the duct diameter exceed 0.8 m, the dose rate at the exit is no more negligible and can be as high as 10 μSv/h.

Shielding design for the High Voltage line

The design of the high voltage line is a key issue in the definition of the MITICA shielding solutions. Different changes respect to the first configurations were performed, according to the specific needs and structural requirements. In the following, the different solutions will be briefly described along with some shielding data obtained from analytical calculations.

In the first configuration a long concrete duct (whose dimensions are approximately 6 m x 11.3 m) follows the transmission line for the entire length of the pipe. The thickness of the lateral concrete walls is variable: the lateral primary shields of the transmission line can be sized down to 1.5 m (respect to the primary shield, 1.8 m). The concrete floor is 1 m thick.

This configuration turned out to be quite unpractical; severe shielding requirements were needed given the great sectional area of the duct. In fact, the streaming of neutrons through the duct imposes important shielding thickness on the sides and the roof, with a consequent too heavy concrete load.

A second configuration was then evaluated. The shield of the transmission line is reduced respect to the previous one. The left-lateral wall was brought closer to the transmission line reducing the concrete weight. The right lateral shield thickness was progressively reduced: the initial 1.8 m were sized down to 1.5 m and to 1.2 m as the transmission line gets away from the source approaching to the side of the building.

A wide concrete septum “strangling” the line was designed. Related calculations showed the possibility of avoiding the concrete roof. The septa dimensions are (thickness x width x height): 1 x 7 x 5 m and 1 x 1 x 5 m. Introducing these shields, the streaming of neutrons beyond the septa is allowed only through the void slot produced by the septa themselves.

A further optimization was then applied at the end of the transmission line, where the control room is designed (and where operators and workers are supposed to be without restrictions).

A small maze was introduced in order to bring inside the bunker connection cables, a concrete elbow was located in front of the control room adjacent to the facility and a smoothed concrete wall follows now the transmission line out of the building.

Conclusions

Room shielding for high current NB facilities were designed considering the main wall penetrations. Neutron diffusion throughout tunnels and ducts was studied with an empirical approach, based on the application of approximate formulas. The result shows that such a simplified approach is viable in assessing radiation diffusion through the shielding wall penetrations avoiding the use of specific time consuming Monte Carlo calculations.

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Effective dose to staff from interventional procedures: estimation from single and double dosimetry

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Abstract

In the Academic Medical Centre (AMC) in Amsterdam numerous interventional procedures are performed by cardiologists and radiologists. During these interventional procedures the physicians are exposed to ionising radiation. From 2004 on the exposure of the physicians was measured by means of two personal dosemeters. One dosemeter was worn outside the lead apron and an additional one inside the lead apron. The study was set up to determine the added value of a second dosemeter worn inside the lead apron. The doses measured on both sides of the lead apron were used to estimate the effective doses of the physicians using one algorithm for single dosimetry and two algorithms for double dosimetry. With the algorithm for single dosimetry the effective doses ranged from 0.11 up to 0.85 mSv in four weeks while with the double dosimetry algorithms, the effective doses ranged from 0.02 mSv up to 0.47 mSv. The statistical analysis revealed no significant differences between the effective doses estimated with the algorithms for single and double dosimetry. It was concluded that the effective doses cannot be estimated more accurately when two dosemeters are used instead of one.

Introduction

In the Academic Medical Centre (AMC) in Amsterdam numerous interventional procedures are performed by cardiologists and radiologists. During these interventional procedures the physicians are exposed to ionizing radiation. Over the years the doses of the physicians performing interventional procedures were monitored by means of a single dosemeter worn outside the lead apron. Since the doses measured outside the lead apron increased and the doses of several physicians exceeded 20 mSv in one year, it was decided to provide all interventional physicians with an additional personal dosemeter to estimate the exposure under the lead apron as well.

In publication 85 ICRP recommended the use of two personal dosemeters (double dosimetry) to estimate the effective doses of physicians performing interventional procedures more accurately[1]. Despite the ICRP recommendations, double dosimetry is not common practice in the Netherlands. Recently the Dutch Commission on Radiation

Dosimetry (NCS) released a code of practise for personal dosimetry of professionals wearing a lead apron[2]. In this code of practise the use of a single personal dosemeter outside the lead apron is recommended. In addition to this, conversion factors to calculate effective doses from the measurements are recommended.

In this context it was decided to estimate the effective doses of the physicians in the AMC as recommended by the NCS code of practise and compare the results with the effective doses estimated with two algorithms for double dosimetry. The purpose of the study was to quantify the differences in effective dose when using single or double dosimetry. Moreover the advantages and disadvantages of single and double dosemeter were reviewed.

Material and methods

Starting 2004 the exposure of eleven physicians (eight radiologists and three cardiologists) was monitored by means of two personal dosemeters. All physicians were experienced to perform interventional procedures. To receive consistent measurements during all interventional procedures the two dosemeters were worn in a special designed holder that was fixed to the lead apron (Fig. I). The holder with the two personal dosemeters was worn breast-high. The two dosemeters were replaced every four weeks on the same day, while the holder remained fixed to the lead apron during the whole study.

The eleven physicians used lead aprons of 0.25 mm lead equivalent thickness at 100 kVp (Medical Development and Technology B.V., Hilvarenbeek, Netherlands; Scanflex Medical AB, Täby, Sweden). The physicians also used thyroid collars of 0.50 mm during the interventional procedures.

The effective doses were calculated by means of three different algorithms (Table 1). First, the effective doses were calculated using the conversion factor from the NCS code of practice. The conversion factors were used for the doses measured outside the lead apron. Additionally the effective doses were calculated with two algorithms for double dosimetry: the NCRP algorithm[3] and an algorithm described by Clerix et al.[4].

Results

The mean doses measured outside the lead aprons ranged from 0.53 mSv up to 4.24 mSv in four weeks while the mean doses under the lead aprons ranged from 0.01 mSv to 0.18 mSv in four weeks (Table 2).

The effective doses estimated with the NCS conversion factor were higher than the effective doses calculated with the two algorithms for double dosimetry. The effective doses estimated with the NCS conversion factor ranged from 0.11 up to 0.85 mSv (mean 0.42 mSv). The effective doses calculated with the Clerinx algorithm ranged from 0.05-0.47 mSv (0.29) and the effective doses calculated with the NCRP algorithm varied between 0.02 mSv and 0.17 mSv (mean 0.10) (Fig. 2).

The regression analyses revealed a linear relation between the effective doses calculated with the NCS-single algorithm and the effective doses estimated with the two algorithms for double dosimetry (ANOVA, $p<0.05$). The regression coefficient between the effective doses for the NCS and the NCRP algorithm was 4.19 and for NCS and Clerinx 1.50.

The effective doses calculated with the NCRP algorithm were plotted against the effective doses calculated according to the NCS algorithm (Fig. 3). The r-squared (R^2) was 0.86. A similar plot was made for the effective doses calculated with the Clerinx algorithm and the NCS algorithm (figures 4, $R^2 = 0.81$).

A Two-Related-Sample test was performed to compare the effective doses calculated according to NCS and the effective doses according to double dosimetry (NCRP and Clerinx). For the tests the effective doses according to NCS were divided by the regression coefficient from the relation between the NCS algorithm and the algorithms for double dosimetry (4.19 for NCS/NCRP and 1.50 for NCS/Clerinx). The two-related-sample tests revealed no significant differences between effective doses calculated from single dose measurements outside the lead apron and effective doses calculated from double dosimetry (Wilcoxon, $p>0.05$).

Discussion

The effective doses of the physicians performing interventional procedures in the AMC differed based on the algorithm used. With the NCS algorithm the effective doses were higher than effective doses calculated with the double dosemeter algorithms. According to the NCS code of practice the doses measured outside the lead apron were used to estimate the effective doses of the physicians in the AMC. To estimate the effective doses the NCS conversion factor for single dosimetry was used. The conversion factor is the same for all exposure circumstances: different tube voltages and variation in irradiation geometry. As NCS describes the approach is save and the effective dose will not be underestimated. This underlines the idea that the NCS algorithm overestimates the effective dose. Even the algorithms for double dosimetry can still lead to an overestimation of the effective dose. Although the probability is smaller and there is even a risk of underestimation.

Conclusions

The purpose of the study was to determine whether the doses measured inside the lead aprons provide additional information that is necessary to estimate the effective doses more accurately. In this study, the added value of measurements under lead aprons could not be demonstrated. The statistical analysis revealed no significant difference between effective doses calculated with the algorithm for single dosimetry and the algorithms for double dosimetry.

Table 1. The three algorithms used to calculate the effective doses (E) were H_{under} is the dose measured under the lead apron and $H_{outside}$ the dose measured outside the lead apron.

Name	Algorithm
NCS	$H_{outside}$ / 5
NCRP	$E = 0.5\ H_{under} + 0.025\ H_{outside}$
Clerinx et. al.	$E = 1.64\ H_{under} + 0.058\ H_{outside}$

Table 2. The number of measuring periods (N), the median doses and the mean doses of 4-weekly measurements outside and under lead aprons.

Physician	N	Median doses (mSv/4-weeks)		Mean doses (mSv/4-weeks)	
		Outside apron	under apron	Outside apron	Under apron
1	4	3.58	0.17	3.41	0.18
2	54	1.02	0.11	2.46	0.13
3	8	2.09	0.07	2.12	0.07
4	9	3.56	0.15	3.65	0.14
5	59	1.40	0.06	1.54	0.07
6	59	3.09	0.13	3.42	0.15
7	42	2.56	0.15	2.82	0.16
8	57	4.16	0.10	4.24	0.13
9	59	0.41	0.01	0.53	0.01
10	58	0.63	0.01	0.74	0.03
11	58	1.04	0.04	1.14	0.06
Total	**454**	**1.43**	**0.06**	**2.12**	**0.09**

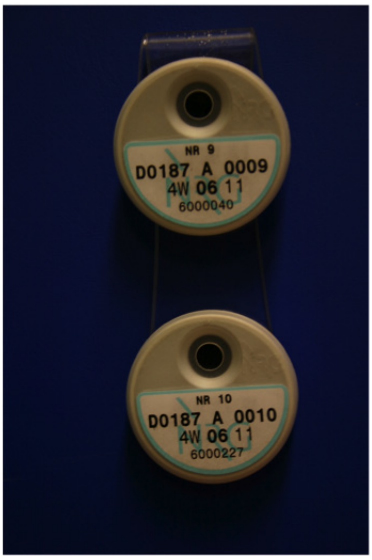

Fig. 1. The special designed holder with the two personal dosemeter.

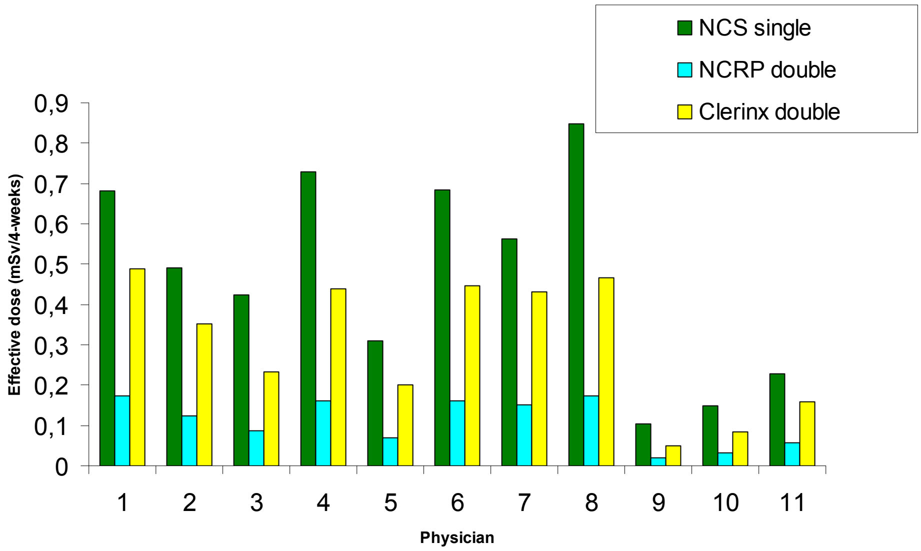

Fig. 2. Effective doses (mSv/4-weeks) of the physicians. The green bars represent the effective doses of the physicians by means of the doses measured outside lead aprons (NCS single). The blue, and yellow bars represent the effective doses by means of doses measured outside and under lead aprons (NCRP double and Clerix double).

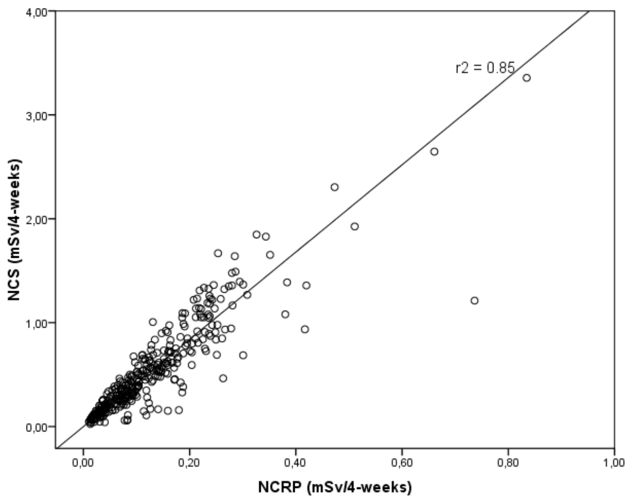

Fig. 3. Effective doses (mSv/4-weeks) according to the algorithm of NCS (single dosimetry) compared to the effective doses according to NCRP (double dosimetry).

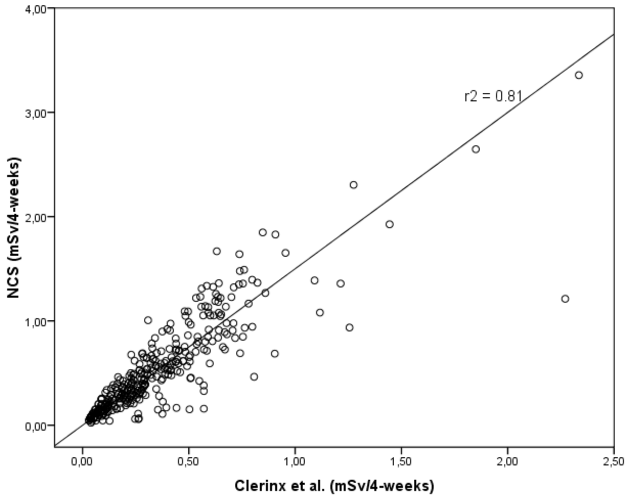

Fig. 4. Effective doses (mSv/4-weeks) according to the algorithm of NCS (single dosimetry) compared to the effective doses according to Clerinx et al. (double dosimetry).

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Exposure levels of workers during some surgical procedures

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Abstract

The use of fluoroscopy during surgery is quickly growing up, involving more and more surgical specialties (neurology, urology, cardiology operating rooms, etc.). Such radiological interventional procedures often require long fluoroscopy times, sometimes longer than 30 minutes. It is thus critical the radiation protection of the whole staff (surgeon, nurse, radiographer, anaesthetist), but it is very difficult to evaluate dose.

Worker exposure dose level is thus evaluated almost entirely on an empirical basic. This study is performed with respect to four surgical specialties: urology, orthopaedy, cardiology and radiology (angiography). Exposure levels for each procedure type (e.g. leg, femur, hip for orthopaedy) were measured, collecting data from at least 10 patients for every procedure.

In order to measure average procedure exposure level, each member of the surgical staff was given a TLD dosemeter to be used only for that kind of procedure.

The collected results are also an useful tool to have *a priori* estimation of dose levels for new procedures and to perform classification of workers.

Introduction

It is well known that radiation dose to the staff during all kind of surgical interventions involving X-ray machines is a quite difficult problem, due to:

1) The rapidly spatially varying radiation field;
2) The worker changing his position during the procedure;
3) Many other quantities changing during the exposure (collimation, distance, tube position, x-ray pulse duration...).

Moreover, present-day X-ray machines have automatic exposure controls, thus making more difficult to monitor exposure parameters.

We are usually able to estimate staff dose, with simple measures and the approximate number of patients per month, but a such rough estimate is sometimes too few in order to properly assess workers classification. A lot of work is in progress in order to give more precise values(1), our contribution is to directly measure *in situ* the dose for each member of the staff and for each procedure.

Material and methods

Our choice is to use the same type of dosemeter that is normally used for routine monitoring of that workers, that is, two TLD100 detectors (manufactured by RadCard in Krakow, Polska), the first filtered by 1,5 mm Aluminum. Reader is the automatic *Harshaw 6600* and the whole system is calibrated to X-ray response (using the facility of ENEA in Bologna, Italia). Wrist dosemeters also use TLD100 detectors but each dosemeter has a single detector, and we read them using the *Harshaw 5500* automatic reader. The whole system follows EU recommendations(2).

The surgeons (they are usually two, the first performing the intervention and the second, sometimes a student, helping him) were given each three dosemeters, the first to wear under the protective apron, the second to wear over the apron, and the wrist dosemeter. The remaining members of the staff (nurse, radiographer, anesthesist) were given each only two dosemeters, as the hand monitoring is not useful.

We thus prepared a lot of sets of dosemeters. Each set is composed by 20 whole body dosemeters (10 for the staff plus 10 for background measurement) and 7 wrist dosemeters (2 for the surgeons plus 5 for the background, that we must measure twice as we have different readers). Each set was used for a single procedure, e.g. homerus in orthopaedy: each dosemeter was given to the owner, and collected back after the intervention. Dosemeters of the homerus set were given again when another patient was scheduled for the same procedure, even if members of the staff changed. The dosemeter in this way belongs to a procedure and not to a worker.

A single set has to measure the maximum number of patients, in order to average different operators, with different skill levels, and different fluoroscopy times, due to pathology or difficulty of each intervention. We planned to collect at least 10 patients for each procedure. Some months were spent for each speciality (urology, cardiology, etc.) collecting the maximum number of data, but in some cases it was not possible to collect more than 4 patients, as the procedure is not so common, and it is not useful to wait longer times to keep signal to background ratio as high as possible.

The complete list of procedures, patients collected, mean fluoroscopy time is given in Table 1. Procedures are chosen where staff dose evaluation is more difficult in a theoretical way.

All procedures shown were performed in the different departments of the AOU Careggi University Hospital in Firenze, and data were collected in 3 years from 2008 through 2010.

To assess individual effective dose values we used the Rosenstein-Webster algorithm(3) to weight the dosemeters under and over apron, that is: effective dose = reading over apron times 0.025 + reading under apron times 0.5. Hand doses are the Hp(0,07) readings of the wrist dosemeter.

Table 1. Procedures and number of patients used for these measurements.

Specialty	Procedure	Patients	Mean fluoroscopy time (minutes.seconds)
Urology	Pig-tail positioning	5	4.05
Urology	Ureterorenoscopy	6	4.42
Urology	Lithotripsy	4	1.02
Urology	Renal percutaneous	3	23.17
Orthopaedy	Hip	15	1.20
Orthopaedy	Shoulder	8	1.43
Orthopaedy	Femur	6	2.32
Orthopaedy	Leg	6	1.03
Cardiology	Valvuloplasty	2	16.18
Cardiology	Patent Foramen Ovale	3	20.12
Cardiology	Carotid Stenting	2	8.48
Cardiology	Atrial Septal Defect	2	25.36
Cardiology	Coronarography	6	10.23
Radiology	Embolization	1	17.24
Radiology	Flebography	3	14.54
Radiology	Lower Abdomen	4	15.00
Radiology	Cholangiography	2	14.12

Results

Figure 1 shows the data normalizing them to 100 patients, to have realistic exposure of a member of the staff. Data uncertainty is very high, about 0,1 mSv or 20%, mainly due to background subtraction. Another uncertainty source, that cannot be shown, is the difference among each patient,

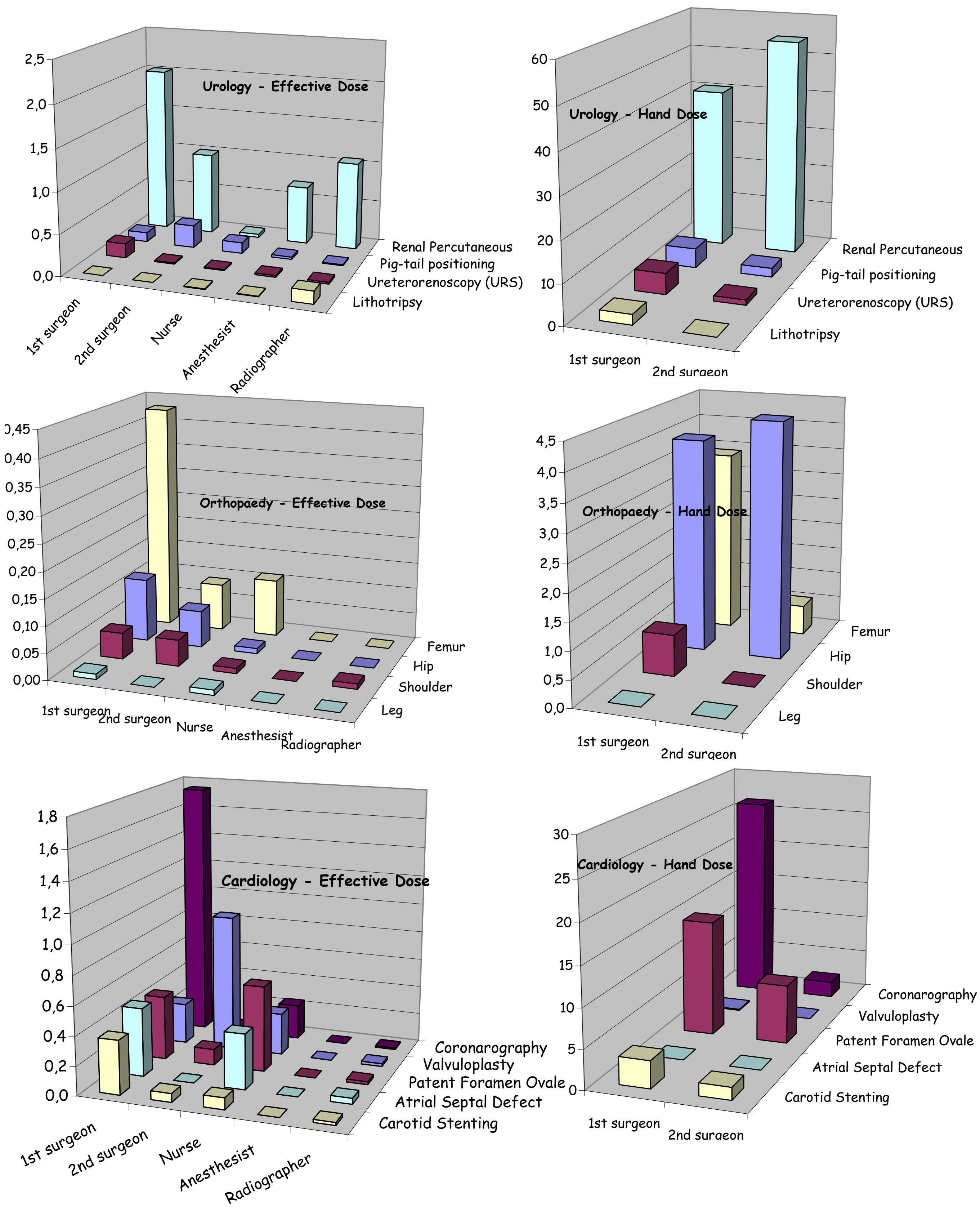
Discussion

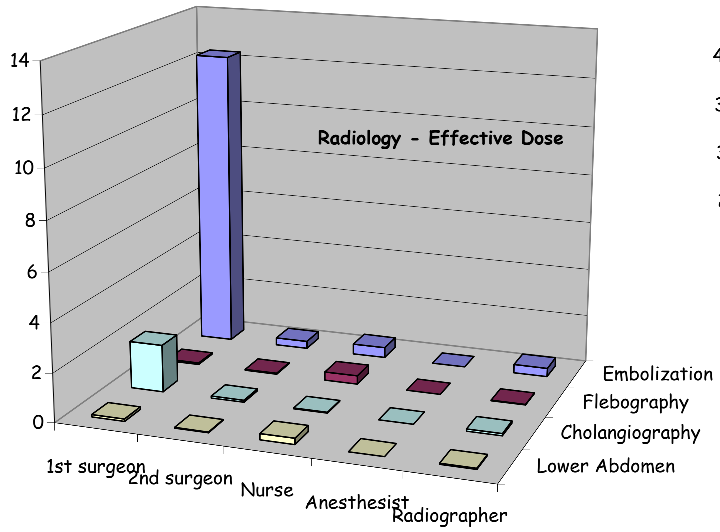
All dosimetric data agree with the rough estimates we had in advance, but are much more precise even with the very high uncertainty. Some values looking somewhat strange (like the dose to the nurse in the femur procedure, much higher than the other procedures) are due to the quite different position assumed by the operator.

Conclusions

We performed direct measurement of staff dose for several surgical procedures. The measuring method is good, but time consuming, even if results are now acquired and there is no need to perform again such measurements. Data collected are useful for workers classification and, as we have already tested, to show to the same workers during training courses, and this is the most important feature. We noticed, while collecting data, that workers are very often not enough trained in radiation protection, and a training course is much more effective if the own dose levels, and not general literature data, are shown.

Fig. 1. Results. Vertical scale is in mSv, and data are normalized to 100 patients.





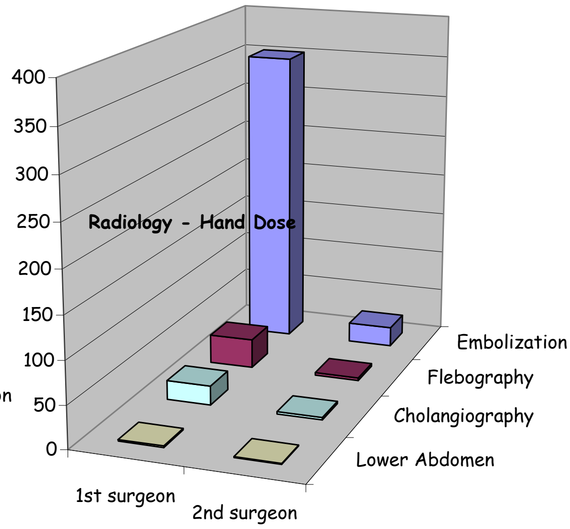


Table 2. Same data as in figure 1.

Specialty	Procedure	Effective dose in mSv					Hand dose in mSv	
		1st surgeon	2nd surgeon	Nurse	Anesthesist	Radiographer	1st surgeon	2nd surgeon
Urology	Pig-tail positioning	0.1	0.3	0.1	0.0	0.0	4.9	2.2
Urology	Ureterorenoscopy	0.2	0.0	0.0	0.0	0.0	5.3	1.3
Urology	Lithotripsy	0.0	0.0	0.0	0.0	0.2	2.5	0.0
Urology	Renal percutaneous	2.0	1.0	0.0	0.7	1.1	40.7	54.9
Orthopaedy	Hip	0.1	0.1	0.0	0.0	0.0	4.0	4.4
Orthopaedy	Shoulder	0.1	0.1	0.0	0.0	0.0	0.8	0.0
Orthopaedy	Femur	0.4	0.1	0.1	0.0	0.0	3.4	0.6
Orthopaedy	Leg	0.0	0.0	0.0	0.0	0.0	0.0	0.0
Cardiology	Valvuloplasty	0.3	0.9	0.3		0.0	0.2	0.0
Cardiology	Patent Foramen Ovale	0.4	0.1	0.6		0.0	14.8	7.5
Cardiology	Carotid Stenting	0.4	0.1	0.1		0.0	3.5	1.6
Cardiology	Atrial Septal Defect	0.5	0.0	0.4	Not present	0.0	0.0	0.0
Cardiology	Coronarography	1.8	0.0	0.2		0.0	26.4	2.1
Radiology	Embolization	12.3	0.3	0.4		0.4	352.1	23.3
Radiology	Flebography	0.1	0.0	0.4		0.0	34.8	3.3
Radiology	Lower Abdomen	0.1	0.0	0.3		0.0	1.8	1.1
Radiology	Cholangiography	2.0	0.1	0.0		0.1	22.7	2.7

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Staff doses in cardiological interventional radiography

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Abstract

The aim of the study has been to assess the radiation dose to personnel involved in heart X-ray assessments (cardioangiography assessments, CA), as well as some cardiology procedures (such as PTCA, EF). In the study the dose exposure has been defined for each assessment and procedure. Three different types of dose meters have been used for dose assessments. Dose measurement protocols were also examined laboratory conditions at STUK. The study was done by STUK in cooperation with two university hospitals and one procedural unit in a central hospital.

Radiation doses were measured (in head, upper arm, hand and foot/ankle) for approximately 350 different procedures. The dose for the doctor performing the procedure varied significantly depending on the length and difficulty of the operation: the average measured dose for the procedures at the upper arm level was about 50 microSv. The highest measured dose was 576 microSv. The average hand dose was about 150 microSv. The highest dose to the hand was 1093 microSv, to the foot 1003 microSv, and to the head 196 microSv. The average dose for the nurses at the upper arm level during a PTCA-procedure was 19 microSv, and the maximum dose was 199 microSv. The largest measured hand dose for a nurse was 492 microSv. According to the measurements the effective dose for the doctor was about 1 microSv / procedure and maximum dose was 12 microSv. The annual dose limits will not exceeded under normal working conditions. However, the dose limits for hands and feet could be exceeded.

The doses received by the people involved in the procedures can be quite high when compared to the annual dose limits. The appropriate use of radiation shields is necessary. Radiation meters with displays, or meters suitable for direct reading are good for observing working conditions. The information obtained in the study can be used in the hospitals to improve procedural methodologies, planning off training, as well as interventional cardiology.

Radiation doses to occupationally exposed personnel working with radioiodine and technetium

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Abstract

The I-131 and Tc-99m content in the thyroid of staff members working with this radionuclides has been measured in some Departments of Nuclear Medicine performing therapy and diagnosis of thyroid disease in Poland . The measurements were performed with portable detection unit for „in situ" measurements of radioiodine and technetium.

All individuals actively working with iodine and technetium show measurable amounts of this isotopes in their thyroids. The average measured activity in the thyroid of the nuclear medicine staff was found to be equal at average 600 Bq within the range from 50 Bq to 70 kBq. The average and range of I-131 activity measured in thyroids for all medical units were: 1500 Bq, (100 Bq – 70 kBq Bq), 400 Bq, (30 Bq – 3000 Bq), 150 Bq, (50 Bq – 1000 Bq) for technical staff, nuclear medicine staff and hospital services staff respectively. There is no apparent correlation between the measured I-131 levels and risk categories. Nevertheless the technical and nuclear medicine staff show higher I-131 thyroid level comparing to hospital services staff.

Base on results of measurements, the Effective Dose Equivalent for particular person due to inhalation of I-131 and Tc-99m was calculated.

Calculated average Effective Dose Equivalent for particular exposed person is below 50 per cent of 20 mSv/year.

Introduction

In 1997 the Central Laboratory for Radiological Protection set up programme "The Laboratory for monitoring of radioiodine in thyroid for population in emergency situation". The main goal of this program was to establish monitoring assembly and develop risk assessment methods for internally contaminated people with I-131 in the event of a nuclear accident or radiological emergency. This Laboratory programme takes advantage of unique opportunity for testing monitoring devices and dose estimation methods on the base of measurements of activity of radioiodine and technetium in thyroid of occupationally exposed workers.

Material and methods

The monitoring assembly of the Laboratory consists of two independent measuring units:

1. Stationary Unit for measuring I-131 and Tc-99m with low limit of detection,
2. Portable Unit for "in situ" measurements of I-131 and Tc-99m. It has been mainly foreseen for fast screening population in radiological emergency situation, or for monitoring occupationally exposed people far away from Laboratory.

The measurements of iodine and technetium content of occupationally exposed personnel were performed with portable detection unit (prod. Canberra-Packard) (Fig.1.), which is consist with scintillation detector NaI(Tl) (size 76 x 76 mm, resolution 9%) - battery-powered, portable tube base Multichannel Analyzer Canberra UniSPEC, paired with the notebook computer and Genie-2000 Basic Spectroscopy Software.

In the period 2008 – 2009 the measurements of I-131 and Tc-99m content in the thyroid of staff members working with radioiodine and technetium has been measured in some Nuclear Medicine Units performing therapy and diagnosis of thyroid disease in Poland. The measurements were performed with portable detection unit.

The counting configuration for monitoring personnel was identical to that used in the calibration procedure. Typically, detector set at a neck - to - detector distance of 10 or 15 cm, using a 300 seconds counting time. The background was measured with detector placed 15 cm away from the availably RSD neck phantom, prior to or just following the count performed on the person. The measurements were performed in selected as low as possible background places. The MDA for mobile unit ranges from 10 – 50 Bq at the time measurement of 300 seconds and depends on background condition in particular units.

The measured personnel can be divided into some categories according to internal contamination risk to unsealed sources of I-131 and Tc-99m:

1. Technical staff mainly performing routine diagnostic investigation,
2. Nuclear medicine staff (physician, nurse) working with in vivo administration of I-131 or Tc-99m to patients,
3. Hospital services staff (orderlies, cleaners) performing auxiliary activities to the patients (cleaning of the rooms, changing of bedclothes).

Results

The measurements of radioiodine content in the thyroid were performed in six medical units that use I-131 for therapy and diagnosis of thyroid disease. About of one hundred exposed persons were investigated. The results of measurements are presented in Table 1.

All individuals actively working with iodine show measurable amounts of the radioiodine in their thyroids (Fig.2.). The average measured activity in the thyroid of the nuclear medicine staff was found to be equal at average 600 Bq within the range from 30 Bq to 70000 Bq. The average and range of I-131 activity measured in thyroids for all medical units were: 1500 Bq, (100 Bq – 70000 Bq), 400 Bq, (30 Bq – 3000 Bq), 150 Bq, (30 Bq – 1000 Bq) for) for categories 1, 2, 3 respectively. Nevertheless the 1 and 2 categories show higher I-131 thyroid level comparing to category 3.

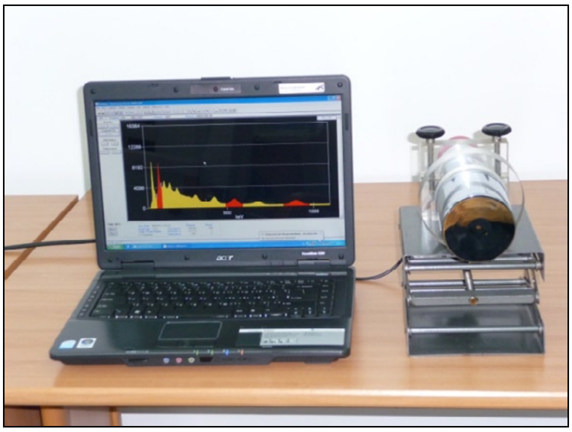

Fig. 1. The portable unit with scintillation detector NaI(Tl) for measurement of I-131 and Tc-99m.

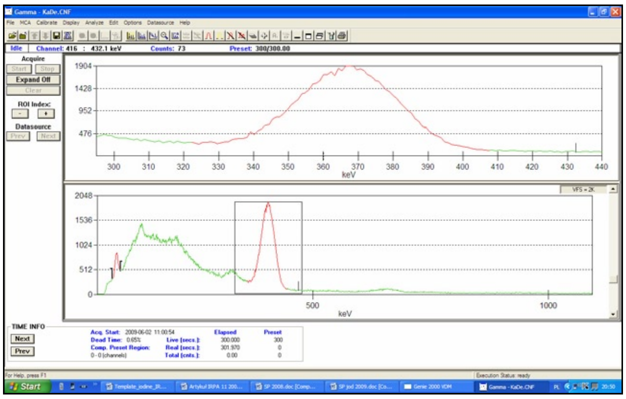

Fig. 2. Spectrum of I-131with photopeak of 364keV collected at thyroid of exposed worker in Medical Unit.

Table 1. The I-131 content and effective doses assessment for personnel of Nuclear Medicine Units.

Medical Unit No.	Category	Range of I-131 content in thyroid [Bq]	Mean value of I-131 content in thyroid [Bq]	Effective dose equivalent from inhalation of I-131 [mSv]	Per cent of occupational exposure limit
1	1	50 – 100	70	0.38	2 %
	2	50 – 70	60	0.35	
	3	50 – 60	60	0.35	
	All	50 – 100	65	0.37	
2	1	60 – 80	70	0.38	2 %
	2	50 – 75	60	0.35	
	3	60 – 80	70	0.38	
	All	60 – 80	70	0.38	
3	1	70 – 400	250	1.50	4 %
	2	50 – 180	120	0.70	
	3	60 – 100	80	0.40	
	All	50 - 400	150	0.80	
4	1	60 – 200	150	0.80	4 %
	2	60 – 220	180	1.05	
	3	70 – 180	120	0.70	
	All	60 – 220	150	0.80	
5	1	80 – 70000	23000	125.00	250 %
	2	170 – 35000	1000	6.00	
	3	100 – 5000	2300	12.50	
	All	80 – 70000	9000	50.00	
6	1	70 – 200	150	0.80	4 %
	2	100 – 200	170	1.00	
	3	80 – 180	135	0.78	
	All	70 – 200	150	0.80	

Conclusions

The results of I-131 content in the thyroid of staff members working with radioiodine in four Departments of Nuclear Medicine do not show any correlation between the measured I-131 levels and risk categories. The averages of I-131 thyroid contents calculated for the particular medical unit item differ remarkably. These differences do not necessary depend on I-131 usage in the particular medical unit but rather on its specific and complex work conditions, staff training and so on.

Base on results of measurements, the Effective Dose Equivalent for particular person due to inhalation of I-131 was calculated with somewhat a conservative assumption that I-131 thyroid content remains constant during the whole year. For the occupational exposure limit of 20 mSv it gives the reference I-131 thyroid level is equal to 7 kBq. Calculated average Effective Dose Equivalent for particular medical is below 50 per cent of 20 mSv/year.

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Occupational exposures from increased use of F-18 FDG in Denmark

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Abstract
In Denmark, the number of nuclear medicine examinations using F-18 FDG has increased greatly in recent years. Theoretically, a corresponding increase in radiation dose to staff in departments performing PET scans might be expected. We have studied annual and semiannual staff doses over three years (2006-2008) and conclude that despite a 270% increase in the number of F-18 FDG investigations over the period, the increase in collective dose to workers is relatively small.

Introduction
Since the national integrated cancer pathways were implemented in the Danish Health Care System in 2007 the number of investigations using F-18 FDG has increased considerably. 6.000 investigations were performed in the year 2006 and in 2008 the number had increased to almost 16.000, meaning that in 2008 13 % of all Danish nuclear medicine investigations used F-18 FDG. Table 1 shows the total number of investigations in nuclear medicine departments and the number of investigations using Tc-99m and F-18, respectively. Figure 1 shows the evolution of the number of PET examinations over the last 9 years.

Table 1. Number of investigations in the nuclear medicine departments.

Year	2006	2007	2008
Total Number of investigations in nuclear medicine departments	108055	120160	121100
Number of investigations using Tc-99m	84200 (78 %)	92160 (77 %)	87290 (72 %)
Number of investigations using F-18 FDG	6093 (5.6 %)	9916 (8.2 %)	15449 (13 %)

There is a concern that the increased use of F-18 FDG is going to affect the doses to the staff in the nuclear medicine departments at the hospitals (Ennow et al **2009**), due to the high energy of annihilation photons and the greater activities administered to patients compared to examinations using Tc-99m. Factors that influence the radiation dose to staff are the activity administered per patient, the number of patients, the methods of dose dispensing and dose administration, and the amount of time spent in close proximity to patients. (Elliott **2009**)

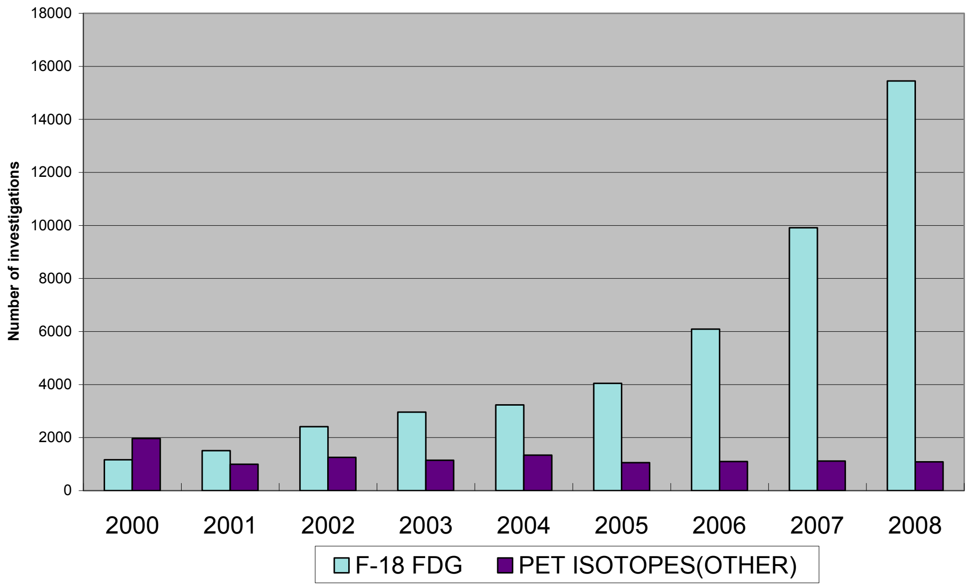

Fig. 1. Number of PET investigations from 2000 to 2008.

Material and methods

The National Institute of Radiation Protection has studied the trends and distribution of doses to staff in the relevant departments in the period 2006 to 2008 by use of the national dose registry. The dose measurements were performed by either TLD or film badges for personal dosimetry. The exposure of the entire staff was assessed by calculating the collective effective dose (ICRP 103 **2007**) every 6 months for all nuclear medicine departments

Values for the annual numbers of PET examinations were obtained from the mandatory annual reports from the nuclear medicine departments of examinations and treatments performed. Over the studied period, Denmark had twenty hospitals with nuclear medicine departments, three of them with cyclotron facilities.

Results and discussion

The study showed that the collective dose to staff increased over the study period in the departments that produce F-18 in cyclotrons, while remaining roughly constant, or even decreasing, in the rest (see figure 2). However, the number of workers receiving annual doses above 1 mSv has increased, especially the number of workers receiving more than 4 mSv/yr. The number of individuals with annual doses exceeding 5 mSv increased from 1 in 2006 to 4 in 2007 to 13 persons in 2008, all employed in departments producing F-18. Figure 3 shows the distribution of annual doses to the staff exceeding 1 mSv over the study period.

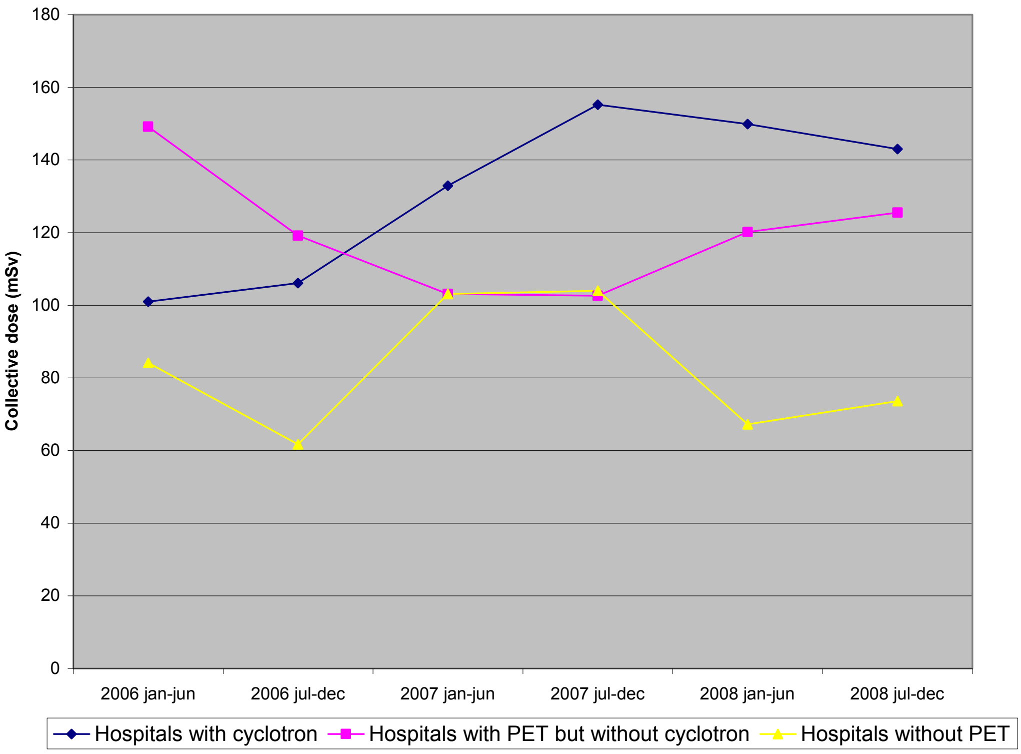

Fig. 2. Collective doses in the nuclear medicine departments from 2006 to 2008.

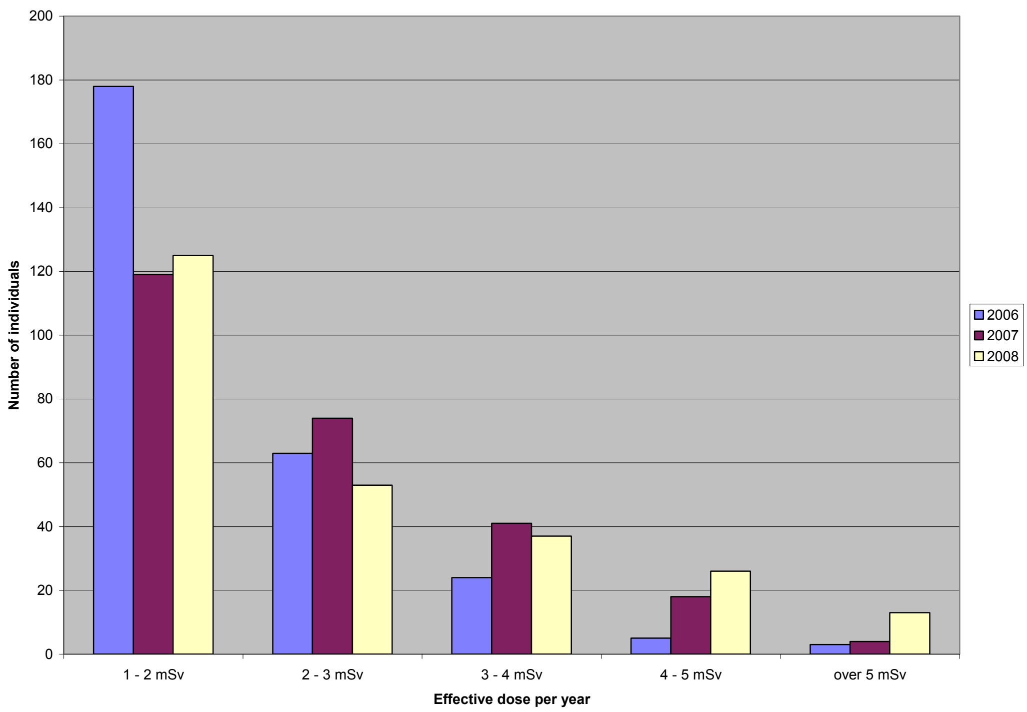

Fig. 3. Number of annual doses exceeding 1 mSv.

Conclusions

The expected raise in doses to the personnel handling F-18 FDG was not seen in this study. It is therefore concluded that the PET facilities have a well established radiation protection programme, and the personnel has been well educated and well prepared for this increase in investigations using F-18 FDG.

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Control of radiation protection and occupational radiation exposure doses of medical staff in Ukraine

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Abstract
According to Ukrainian legislation in radiation protection it is necessary to provide the workplace and individual dose monitoring of occupational radiation exposure. In Ukraine the legislative dose limits for occupational exposure correspond to Basic Safety Standards (IAEA BSS-115). The best way to estimate radiation protection is to evaluate results of individual dose monitoring of occupational exposure. The main purpose is to get information about realistic doses for different groups of staff, compare it with dose limits (category A), estimate the radiation protection of workplaces, inform Regulatory Bodies (in Ukraine these are Ministry of Health and State Committee of Nuclear Safety) about results, and suggest further measures for reduction of radiation exposures. Individual monitoring of occupational exposure for medical staff in Ukraine is carried out since 1979 by Central Laboratory of Radiation Hygiene of Medical Staff at Grigorev Institute for Medical Radiology. Now more than 6500 medical workers from departments of radiation therapy, nuclear medicine and diagnostic radiology from all regions of Ukraine are covered by quarterly TLD-monitoring. In this work, the dynamics of radiation-hygienic parameters during 1980 – 2008, which characterized the state of radiation protection for different professional groups of medical staff, is analyzed. 92 – 98 % of medical staff annual doses are less than 2 mSv. The most exposed groups of medical staff are handlers of radioactive substances and radio-manipulation nurses in radiotherapy, as well as the personnel which is involved in interventional radiology. In some years the average annual doses in these groups were more than 5 mSv. Every year the annual reports with radiation-hygienic statistical analysis of occupational exposure are generated for each hospital, region of Ukraine, different professional groups. These are sent to each hospital and to National Regulatory Bodies for analysis of radiation protection.

The new method of distinguishing static exposure of individual TLD dosemeters

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Abstract

It is not possible to distinguish between cases of static (against source) and dynamic exposure of thermoluminescence dosemeters (TLD) evaluated in readers with photomultiplier tubes. Budzanowski et al (2006, 2007) demonstrated that using TL reader with CCD camera it is possible to qualitatively identify the cases of static exposure of TL detector covered with an non-uniform filter. Kopec et al, (2010) developed and tested a method, which can be used for automatic detection the cases of static exposure in dosimetric RADOS badge with TL (MCP-N) detectors. Detectors were readout in a novel type of thermoluminescent reader with CCD camera developed at IFJ PAN.

Introduction

In thermoluminescent (TL) dosimetry it is not possible to identify the cases of static exposure against the source, because the signal is detected and integrated by a photomultiplier tube.

Budzanowski et al. (2006, 2007) demonstrated qualitatively that it is possible to identify the cases of static exposure of TL detector covered with a non-uniform filter and evaluated using the new generation of TL readers with CCD camera. Kopec et al (2010) developed a quantitative method of distinguishing static exposure, which can be applied in routine dosimetric service.

Material and methods

The routine personal thermoluminescent RADOS dosemeter comprises four thermoluminescence (TL) detectors. Three detectors are held under Al filter. In this work the standard RADOS personal dosemeter was modified by replacing Al filers with 1 mm thickness Pb or Cu filters with a centrally placed 1 mm diameter hole (Fig. 1).

Fig. 1. The standard dosimetric badge (RADOS) with modified filters made of Pb and Cu with 1 mm diameter hole.

MCP-N (LiF:Mg,Cu,P) TL detectors (4.5 mm diameter and 0.9 mm thick) were applied in the badge. The dosemeters, placed on a 30 x 30 x 15 cm^3 PMMA water phantom. The modified RADOS badges with MCP-N detectors were irradiated on using X-rays and Cs-137. Dosemeters were irradiated to doses Hp(10) = 5, 10 and 20 mSv, in order to verify if the effect is observed at the current dose limits for occupational exposure in Europe. TL detectors were evaluated using laboratory TLD reader with PCO SensiCam CCD camera, developed at the Institute of Nuclear Physics (Fig. 2).

Fig. 2. TLD reader with CCD camera developed in Insitute of Nuclear Physics PAN.

The analysis of image

The typical pictures collected by the TLD-CCD reader are shown in Fig. 3, where the increased intensity of TL light under the uncovered central part of TLD can be observed in case of static exposure.

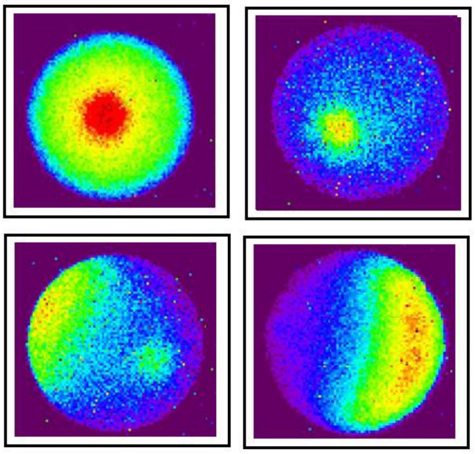

Fig. 3. Two- dimensional TL images obtained for static exposure to 40 keV X-rays. Images are collected for four angles of incidence (0, 30, 45 and 60 degrees). Dosemeters were exposed on a standard PMMA phantom to the dose of 20 mSv.

The images of intensity and were processed using median filter and threshold to removing the background emission outside the detector. The measured level of individual pixels intensity were grouped in form of histogram. The difference between the pixel number with the end intensity, $level_{end}$, and the position of pixel with the number with the most probable intensity, $level_{max}$. was defined as the "coefficient of static exposure" (Fig. 4.):

$$w_s = level_{end} - level_{\max}$$

The value w_S is used to distinguish between static and dynamic exposure. It was found empirical that for all analyzed cases of dynamic exposure (yellow histogram) w_S<=23 (in 256 scale).

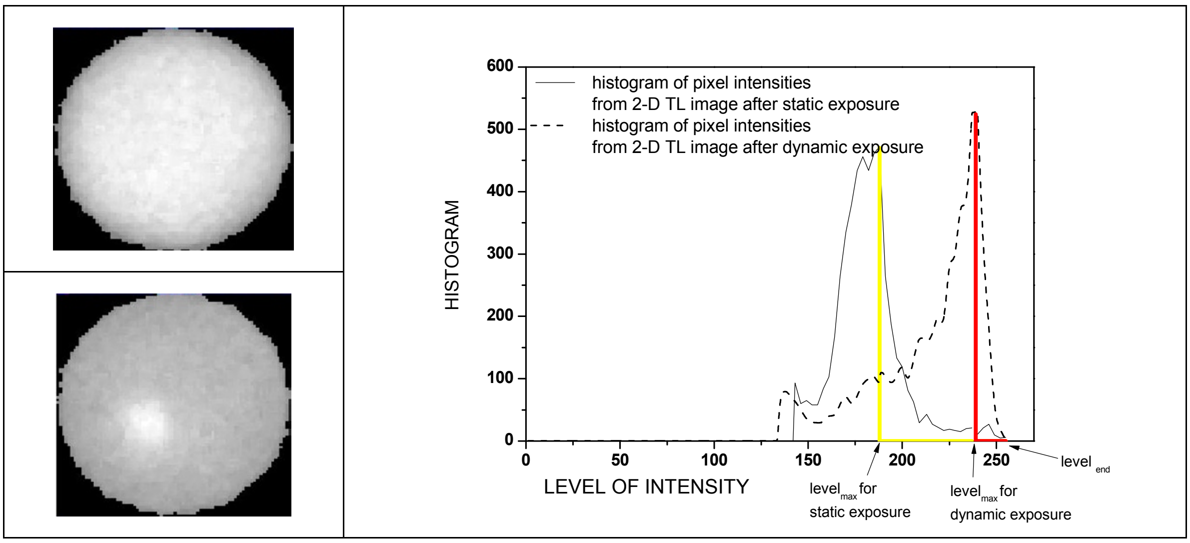

Fig. 4. Left: Images from CCD camera in 256 gray scale intensity of MCP-N detector covered with the 1 mm thick Pb filter with a hole 1 mm in diameter hole after dynamic (up) and static (down) exposure to N40 X-rays, D = 20 mSv. Right: Histograms of pixel intensities. (Kopec, 2010).

Results and discussion

The maximum possible thickness of the filter was limited by the geometry of the RADOS badge to 1 mm. The identification of static exposure is particularly important for doses close or exceeding the limits in radiation protection. Therefore all irradiation in this work were performed for dose range between 5 and 20 mSv For photon energies under 118 keV the proposed method is effective (Fig. 5).

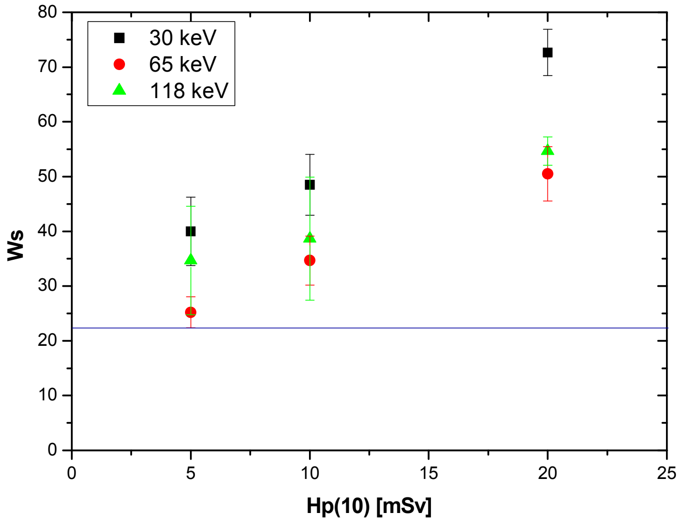

Fig. 5. Dose response of the MCP-N detectors evaluated in TLD reader with CCD-camera for mean energy X-ray 30 keV, 65 keV and 118 keV.

Conclusions

Using the existing setup i.e. using the modified RADOS badges (using the 1 mm thickness Pb filter with 1 mm central hole), MCP-N TL detectors, IFJ PAN TL reader with CCD camera and the analyzing software makes it possible to distinguish the cases of static exposure down to doses of 5 mSv and to X-rays energies up to 200 keV (approximately). The method is based on evaluation of the factor, which can be evaluated from the histogram of pixel intensities. Further work is planned to introduce the system into the routine individual monitoring by the LADIS IFJ dosimetric service.

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Occupational radiation exposure in Poland based on results from the accredited dosimetry service at the IFJ PAN, Krakow

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Abstract

Dosimetry service based on thermoluminescence (TL) dosimetry started its activity at the Institute of Nuclear Physics (IFJ) in 1965. At that time dose measurements were done only for 250 IFJ`s workers. In 2002 the new Laboratory of Individual and Environment Dosimetry (LADIS) was formally established at the IFJ and got accreditation according to EN-PN-ISO/IEC 17025 standards.

The service is based on standard thermoluminescent detectors MTS-N (LiF:Mg,Ti) and MCP-N (LiF:Mg,Cu,P), known worldwide. The TL detectors are readout in automatic DOSACUS or RE2000 (Rados Oy, Finland) readers. The Laboratory provides dosimetric service for individual dosimetry in terms of the personal dose equivalent Hp(10) and Hp(0,07) for photon and neutron fields, within the range from 0.1 mSv to 1 Sv; and environmental dosimetry in terms of the ambient dose equivalent H*(10) within the range from 30 μSv to 1 Sv, and also air kerma Ka within the range from 30 μGy to 1 Gy. Dosimetric service is currently performed for ca. 3200 institutions over the whole Poland. Exposure periods are on a quarterly and monthly basis. Measurements are performed mainly for oncology centres, hospitals, dentists, research institutes, food-, light-, fuel- industry, state border officers, police, antiterrorist troops, prisons, museums, and many other sites where X- or gamma-ray systems are used for inspection.

The paper presents results of statistical evaluation of more than 400 000 quarterly effective dose measurements performed in years 2002-2009. The dose records were divided according to a technical or medical institution and a type of measurement performed (individual dosimetry or environmental dosimetry). The results showed that more than 85 % of measurements were below 0,1 mSv/quarter which means that there was no occupational exposure. The remaining 15% was above 0,1 mSv/quarter with cases reaching even several hundreds of mSv/quarter. The cases exceeding the annual limit will also be discussed in this paper.

Introduction

The first thermoluminescent dosimeter was developed at the IFJ PAN and used for routine measurements of doses for the IFJ's workers in 1965. These dosemeters were based on LiF:Mg, Ag and CaF in Teflon. In 1970's new LiF:Mg, Ti dosemeters (MTS-N, TLD Poland), sintered in the form of pellets, were introduced in individual and environmental dosimetry. From 1986 MTS-N has been replaced with the new highly sensitive MCP-N (LiF:Mg,Cu,P), also developed at the IFJ. The first 25 years of TL dosimetry at the IFJ was described in details by Tadeusz Niewiadomski (Niewiadomski, 1994).

Previously, in the Polish Atomic Law only three polish institutions were explicitly mentioned in the text concerning film dose measurements, therefore the IFJ was not able to do any official dose control for workers from outside of the Institute.

In 2000 the new Polish Atomic Law (Act of Parlament, 2000) was issued in accordance with the European regulations and the paragraph related to the dosimetry services was changed. As a result, currently only accredited laboratories are able to get permission for dose measurements of ionizing radiation in Polish institutions where staff are exposed to such radiation.

In 2002 the LADIS laboratory was created and on 31st December it got accreditation after preparation of all documents.

In Poland, the reference levels were specified in the Ordinance of Ministry of Health: "On the conditions on the safe application of ionizing radiation for all types of medical exposure", on 25th August 2005. According to the Polish Atomic Law the individual dosimetry is obligatory for category A workers while for category B workers the workplace environment monitoring is sufficient. However, most of the radiation safety inspectors decide to apply individual monitoring for category B workers as well, for safety reasons. In Poland, the head of an institution is responsible for summarizing annual doses received by his staff and for contact with the central register in the Polish Atomic Agency. The dosimetry service is not obligated to calculate doses annually but only for periods agreed with clients. The Polish Atomic Law permits the monitoring periods to be not longer that three month, so practically 100% of clients order dosimetry service on a quarterly basis. Equivalent dose for the whole body, $H_P(10)$, should be less than 20 mSv a year, according to the agreed dose limits for occupational exposure (cat. A).

Material and methods

Standard, commercially available MTS-N (LiF:Mg,Ti) detectors are used for individual monitoring, while highly sensitive MCP-N (LiF:Mg, Cu, P) detectors are used for workplace environmental measurements. An individual dosimeter badge used for measurements is a standard RADOS Oy one with three Al filters and four MTS-N pellets. For environmental monitoring a RADOS Oy badge without filter is applied. Dosemeters are readout in automatic RE1 and RE2000 (RADOS Oy) TL readers in standard recommended conditions.

The laboratory has developed a data base based on Microsoft Access. The data base contains all necessary information about client's details, dosmeters numbers, and quarterly doses. The LADIS service has got accreditation for measuring $H_p(10)$ in mSv in the range from 0,1 mSv up to 1 Sv, and K_{air} in mGy respectively from 0,03 mGy to 1 Gy (Polish Accreditation Centre). Detailed information about methods used in the laboratory is provided on the website (http://dawki.ifj.edu.pl).

Database

The rapid increase of the number of customers from 250 in 2001 to 30000 in 2010 stimulated the development of the dedicated DosBaz database. The database was built in 2004 using the MS Access platform. The content of the data structure was elaborated according to the EUR 14852 EN recommendations. The DosBaz allows for complete processing of the data, including preparation of the final certificate, reporting to establishment authorities, preparation of the statistics etc.

At the beginning of 1990's dose values for people under the radiation control were stored on the manually updated, individual dose record sheets. After introducing personal computers into the service the dose records were stored semi-automatically on EXCEL worksheets. The main aim of the development of the DosBaz was to decrease the workload associated with the process of preparation of all documents and to facilitate the flow of the documents and information within the service and between the service and the customer. In addition, the database enables the service administrator to have fast access to the data related to the customer, monitored workers, and measured doses. The database has been developed taking into account the *Technical recommendations for monitoring individuals occupationally exposed to external radiation* (EUR 14852) (Christensen,1994) . The *Recommendations* have formulated major milestones in all aspects of modern, European individual monitoring service including dosimetric concepts, requirements for personal dosemeters, their type and performance testing, aspects of management and administration, and the quality assurance system. For statistical purposes the classification (coding) of data is necessary. The coding includes defining work activities specific for the given employee, recognising occupational category of individual workers and identifying sites with the risk of exposure.

Results and discussion

Statistical analysis was performed on ca. 400 000 quarterly recorded occupational radiation doses. Few examples are discussed.

According to the LADIS classification of occupational exposure, the controlled radiation workers were divided into seven main categories:

- Interventional Radiology
- Oncology Centres
- Nuclear Medicine
- Dental
- Industrial Radiography
- Veterinary Radiology
- Radiology

These categories were compared in terms of the radiation doses received by workers. Figure 1 presents whole body dose measurements performed using individual dosemeters - measurement operational value $H_p(10)$. Figure 2 depicts the results of environmental measurements - measurement value K_{air}. The results reflected the fact that occupational exposure from Nuclear Medicine made a prominent contribution to the exposure values for all medical uses of radiation. Also, in the industrial radiography and veterinary sector, occupational doses were higher than the average occupational dose. Depending on the monitored population, usually the majority of the workers did not receive any occupational dose (e.g. Dental uses).

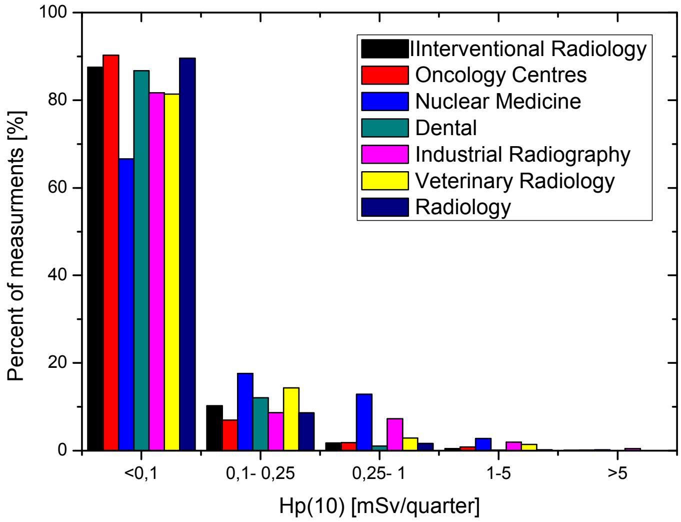

Fig. 1. Percentage of quarterly measurements of photon (gamma, X-rays) doses for different kinds of occupational radiation exposure as a function of dose ranges $H_p(10)$ using the DI-02 dosemeter.

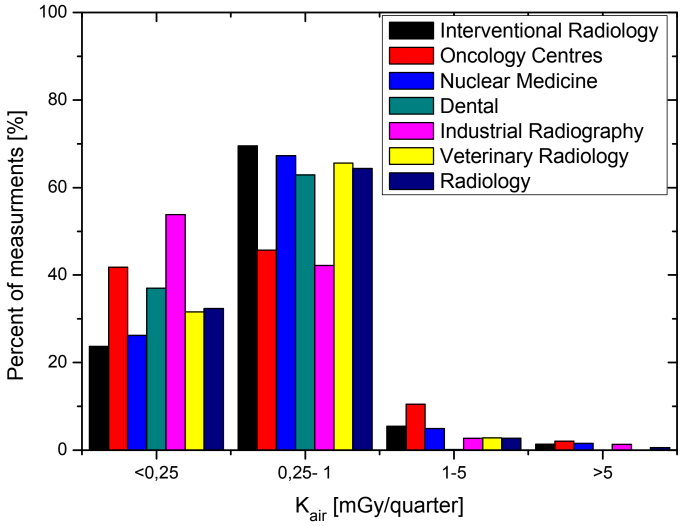

Fig. 2. Percentage of quarterly measurements of photon (gamma, X-rays) doses for different kinds of occupational radiation exposure as a function of dose ranges K_{air} with the environmental dosemeter.

The comparison of ten highest dose values between medical and industrial use of ionizing radiation shown that doses in industrial sectors were much higher than doses in medicine (Figure 3).

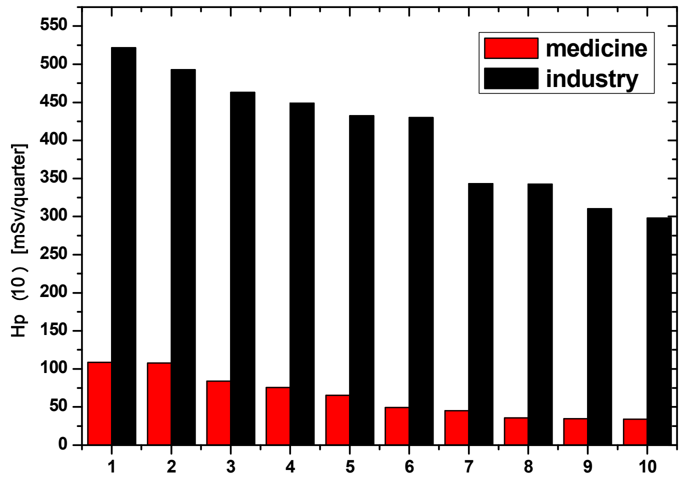

Fig. 3. Ten highest Hp(10) doses measured in medicine and industry with the DI-02 dosemeter.

Conclusions

65% to 90% of quarterly recorded $H_p(10)$ doses were below 0,1 mSv, which is on the level of the natural environmental radiation doses. The natural radiation was the main factor contributing to the exposure dose and occupational exposures made a very small contribution to the average exposure dose of the national population. The results showed that dose levels between 0,1 - 5 mSv/quarter were the most frequent in nuclear medicine, veterinary and industrial radiography sectors. Comparing the medical and industrial radiation uses it can be concluded that the highest doses were registered for industrial applications of gamma and X-rays. These occupational doses were over the dose limit.

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Finger doses in Poland in the view of the extremity ring dosimetry results of LADIS Dosimetric Service Kraków

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Abstract

The Laboratory of Individual and Environmental Dosimetry (LADIS) of the Institute of Nuclear Physics (IFJ), Kraków, applies MTS-N (LiF:Mg,Ti) thermoluminescence (TL) detectors to individual monitoring of Hp(0.07) in photon radiation fields.

The TLDs are inserted in a plastic ring holder (internal name PI-01, Fig. 1) that can be adjusted to any finger size. One standard TL pellet, 4.5 mm diameter and 0.7 mm thickness, is placed in a bar coded holder under a plastic cover, 0.4 mm thick. The dosemeter's behaviour during various sterilisation processes, used at the clients' hospitals, has been tested with positive results.

The LADIS laboratory has been performing extremity ring dosimetry measurements for more than eight years. This procedure has been accredited since 2002, according to the quality system based on PN-EN ISO/IEC 17025 standards, for Hp(0.07) personal dose equivalent within the range of doses from 0.1 mSv to 1 Sv. The paper presents the results of evaluation of extremity ring measurements performed by LADIS in years 2002-2009. More than 50 thousand measurements were performed for more than 300 institutions in Poland on a quarterly basis. Radiation fields represented exposure situations for staff in hospitals, as the users of our ring dosemeters were mostly from interventional radiology and nuclear medicine units.

As the results showed, about 70% of Hp(0.07) doses were below 1 mSv/quarter, however, some very high doses were also registered. We found doses exceeding the dose limits and these cases will also be presented. The clients were classified according to their medical activity and the levels of measured doses were found for these categories.

The laboratory participated with positive results in the international intercomparisons of ring dosemeters organized in 2007 and 2009 by the European Dosimetry Group EURADOS, in order to verify performance of different extremity dosemeters used in Europe.

Introduction

In order to assess occupational exposure at the workplace in therapeutic and diagnostic radiology as well as nuclear medicine it is necessary to monitor external radiation.

The LADIS laboratory provides dosimetric service for individual dosimetry in terms of the personal dose equivalent $H_p(10)$ and $H_p(0,07)$ for photon and neutron fields and environmental dosimetry in terms of the ambient dose equivalent H*(10) [1].

In extremity dosimetry we use ring dosemeters to measure radiation exposure (Fig. 1). The laboratory has been performing extremity ring dosimetry measurements for more than eight years. This procedure has been accredited since 2002, according to the quality system based on PN-EN ISO/IEC 17025 standards, for Hp(0.07) personal dose equivalent within the range of doses from 0.1 mSv to 1 Sv. This paper presents results of evaluation of extremity ring measurements performed by LADIS in years 2002-2009. More than 50 thousand measurements were performed for more than 300 institutions in Poland on a quarterly basis. Radiation fields represented exposure situations for staff in hospitals. The users of our ring dosemeters were from interventional radiology, nuclear medicine units, oncology centres, and radiology [2,3].

Materials and methods

The dosimetry service (LADIS) of the Institute of Nuclear Physics (IFJ), Kraków, applies MTS-N (LiF:Mg,Ti) thermoluminescence (TL) detectors (developed and produced at the Institute of Nuclear Physics PAN) for individual monitoring of Hp(0.07) in photon radiation fields. The TLDs are inserted in a plastic bar coded ring holder under a plastic cover, 0.4 mm thick. Each of TL pellets have a 4.5 mm diameter and 0.7 mm thickness. Ring dosemeters can be adjusted to any finger size [4].

Measurements carried out in the Laboratory of Individual and Environmental Dosimetry IFJ PAN using ring dosemeters type PI-01 are based on the process of thermoluminescence.

For dose measurements we use TLD RADOS Dosimetry System. All detectors are annealed and read in an automatic RADOS hot-gas reader system. Post-irradiation annealing at 75°C for 10 minutes is applied to detectors. TLDs are put in TLD RADOS badges which are placed in a cassette and then read in RADOS RE1 and RE 2000 readers, using the routine read-out procedure. During the readout TLDs pellets are lifted by vacuum needle into the measuring chamber inside the reader, which is filled by nitrogen heated to 300°C (1.5 s pre-heat and 11 s read-out time). The emitted light is measured with photomultiplier; the signal from the photomultiplier tube is consequently amplified and eventually measured with the measuring electronics. The computer obtains information about the number of pulses counted in each pellet. The obtained figure is proportional to the cumulative dose on the detector. By comparing it with the number of counts from the calibration detector (irradiated with a dose of a known value), we can calculate the dose for the given dosemeter [4].

Results

Years of experience allow the laboratory to analyze the levels of doses received by workers in medical institutions cooperating with LADIS. Our analysis indicated that most of the doses received by individuals were on the level of the natural radiation background, but some of them exceeded the dose limit (Fig. 3).

The percentage of overdoses was relatively small, however, looking at individual cases the risk of irradiation was noticeable. Therefore constant monitoring continues to be necessary.

The highest extremity doses came from measurements taken in the nuclear medicine and interventional radiology (Fig. 3).

In 2007, the Laboratory of Individual and Environmental Dosimetry IFJ PAN participated in an international intercomparison of ring dosemeters organized by the European Dosimetry Group EURADOS. More than twenty centers conducting dosimetric extremity measurement operating in Europe took part in this project. In 2009 the LADIS laboratory participated in Eurados intercomparison for the second time ("EURADOS Intercomparison 2009 for extremity dosemeters"). The LADIS laboratory participated in both international intercomparisons with positive results.

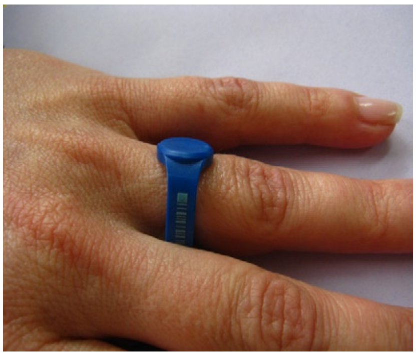

Fig 1. Bar coded ring dosemeter.

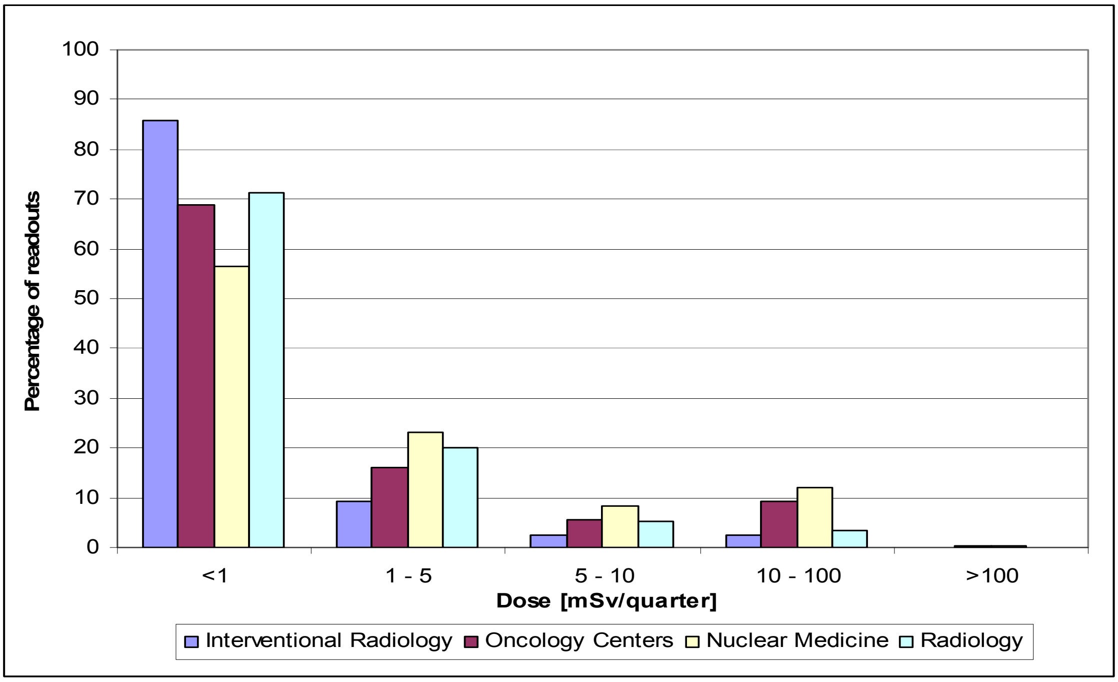

Fig. 2. Summary of extremities dose measurements performed in LADIS.

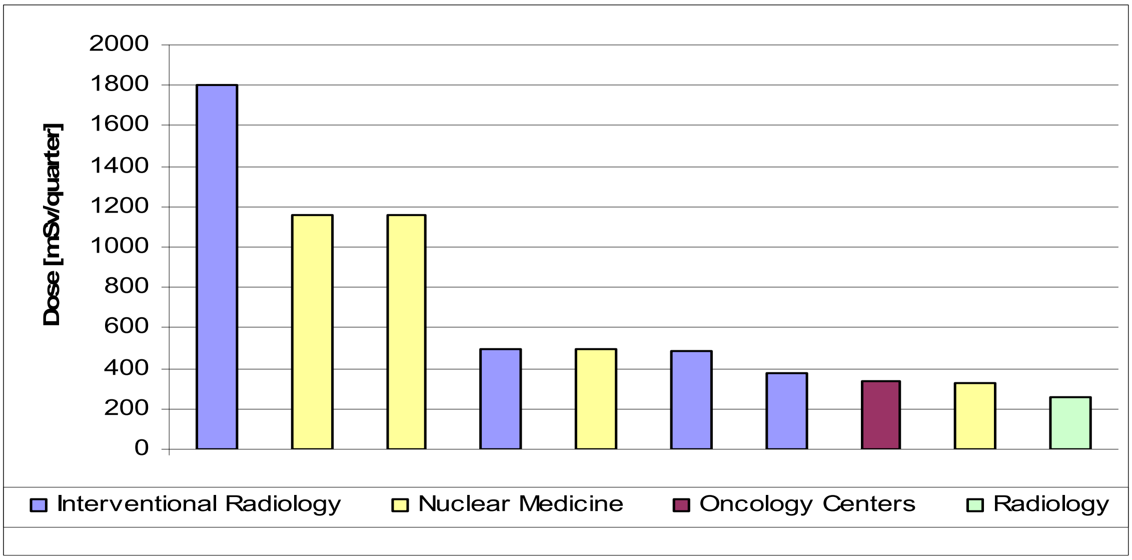

Fig 3. Top ten highest extremity doses.

Discussion and Conclusions

Investigation of cases of overdose led to a conclusion that they were caused by lack of caution, for example leaving dosemeters in the radiation field instead of keeping them on the finger. Consequently, the dose was absorbed by a dosemeter alone but not by a monitored person. Another reason of getting raised doses was incorrect application of radiation shields. Nevertheless, these results still indicate that some degree of risk is present, therefore regular monitoring remains necessary.

70% of all extremity doses in Poland are on the level of the natural radiation background. Dose levels between 1 – 10 mSv quarterly are the most frequent in nuclear medicine and interventional radiology (Fig. 2). The highest dose level above 100 mSv is observed in nuclear medicine. The highest readings of ring dosemeters, exceeding the dose limits, are observed in interventional radiology and nuclear medicine (Fig. 3).

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Study of deterministic and Monte Carlo simulation methods for neutron and photon dosimetry at the Royal Surrey Hospital radiotherapy facility

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Abstract

The application of linear accelerators in the treatment of cancerous cells has become increasingly common in the last few decades. During this period the operational beam energies have increased to a point where the dose from unwanted photoneutron production has to be considered. This can add an additional complexity to calculating the dose to the patient being treated and also the potential dose rate external to a radiotherapy bunker. To this end the 10 MeV radiotherapy linear accelerator at the Royal Surrey County Hospital has been used as a template to calculate the photon and neutron dose rates at the entrance of a bunker and to discuss the advantages between the uses of two different dose assessment methodologies.

Calculation of the radiation beam path through the bunker and doses at the entrance have been performed using traditional deterministic methodologies and compared against Monte Carlo model simulations.

The deterministic method examined photon beam energies of 6 MeV and 10 MeV, which yielded entrance dose rates of 0.34 and 0.27 μSv/hr, respectively. However, the use of correction factors, such as scattering coefficients, used in this method placed doubt on the accuracy of the results. The Monte Carlo method was used to simulate the radiotherapy bunker of the Royal Surrey County Hospital to examine the neutron dose at the bunker entrance, yielding a dose of 8 x 10^{-11} neutron/Gy. Further Monte Carlo simulations were conducted with the bunker configuration modified through the addition of shielding and neutron moderators to demonstrate further dose reduction options available in the bunker design.

The advantages of the deterministic and Monte Carlo methods are discussed, concluding that Monte Carlo analysis offers an improved radiation protection tool for the calculation of doses and dose rates where complicated geometries and materials must be considered.

Neutron production

Medical equipment, such as linear accelerators (linacs) are an integral tool in the fight against cancer. Linacs utilise either electrons or photons to destroy cancerous cells in patients. Above threshold energies interactions of these particles with certain materials can result in the production of neutrons. These interactions are categorised as electroneutron (e,e'n) and photoneutron (γ,n) reactions, respectively.

The first opportunity a photon has to produce neutrons occurs in the treatment head of a linac. Figure 1.1 displays the treatment head of a linac at the Royal Surrey County Hospital (RSCH) where electrons have already been accelerated through a waveguide. Subsequently, the electrons have travelled through a flight tube onto an x-ray target resulting in photon production. The cone created by the primary collimators intrudes onto the edges of the target and defines the largest available field size. Collimators are typically constructed out of lead with a steel casing, tungsten, copper alloys or depleted Uranium (DU); ideal materials for the production of photoneutrons at energies as low as 6 MeV. Photons then pass through the beam flattening filter exiting with a uniform dose distribution; monitored by a set of ionisation chambers. The wedge filters are removable and allow the production of a gradient across the x-ray field. Wedges can have a very complex shape in order to produce the required dose distribution. Finally, photons pass through a moveable collimator where the width of the radiation field is minimised and exit through a perspex window.

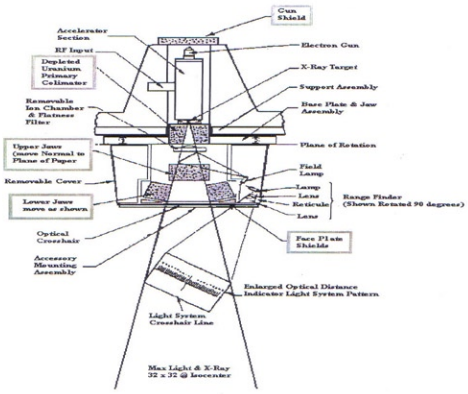


Fig. 1. Treatment head of the new linac at the Royal Surrey County Hospital.

The treatment head consists of several sources of neutron contamination including the walls of the vacuum chamber, the x-ray target, filters, DU collimator, the walls of the waveguide and light localisers (NCRP 79).

Once clear of the treatment head neutron production can arise through photon interactions with the air between the linac and the patient, the tissue in the patient and the bunker itself. Interaction with the bunker can cause further photoneutron interactions if it is lined with a high-density material for photon shielding, such as steel or magnetite. Neutrons can easily scatter of these materials with little loss in energy resulting in neutron escape from the bunker if there is no shielding door.

Photoneutron reactions occur above threshold energies that are dependent upon the cross section of element that the photon interacts with and the photons kinetic energy. When a reaction takes place it is possible for more than one neutron to be produced; therefore the photoneutron production cross-section is the sum of σ(γ,1n), σ(γ,2n) and σ(γ,3n)… etc. with σ(γ,1n) usually contributing significantly (Chin 1999). The only known exceptions to this are the σ(γ,1n) for the odd-Z, even-A nuclides ^{6}Li,

^{10}B and ^{14}N (NCRP 79). The total photoneutron yield cross section, σ(γ,xn), is given by σ(γ,1n) + 2σ(γ,2n) + 3σ(γ,3n) + … etc. The (γ,1n) and (γ,xn) cross-sections for ^{12}C and ^{16}O are shown in Figure 1.3.

Effects on neutron spectra in accelerator head

When a linac is energised the amount of shielding around the target can serve to produce a collimated beam of x-rays. The neutrons produced inside the treatment head are approximately isotropic and penetrate the shielding in all directions. Heavy metals, such as tungsten and lead, are the most common materials used for shielding in the head in conjunction with amounts of copper and iron present in the bending magnets. It should be noted that the majority of photon shielding is in the forward direction as this is where photons are emitted. These materials offer no shielding against neutrons and can facilitate photoneutron interactions. The only significant mechanisms of neutron energy loss from these heavy elements are inelastic scattering and (n,2n) reactions.

Figures 2 (Howerton 1958) displays the neutron interaction cross-sections of tungsten and lead. Shown are the inelastic, elastic and (n,2n) cross sections in the energy region of interest for linacs. It can be seen that inelastic scattering dominates energy losses at lower energies whilst at higher energies loss is primarily due to (n,2n) reactions. However, inelastic scattering can only occur at energies above the lowest excited state of the shielding material. The lowest excited states for some shielding materials are given in Table 1 (NCRP 79). Following an inelastic collision there is an energy loss leading to the excited state of the nuclei; theses nuclei de-excite to their ground states through the emission of photons. The energy loss from this collision cannot be determined but there is a minimum energy loss that equals the energy of the lowest excited state (NCRP 79).

Fig 2. Neutron interaction cross section in tungsten and lead as a function of neutron energy (Howerton 1958).

Table 1. Inelastic scattering thresholds (NCRP 79).

Element	Atomic Mass Number	Abundance %	1st Excited State (MeV)
Pb	206	25.1	0.803
	207	21.7	0.570
	208	52.3	2.61
Fe	54	5.8	1.41
	56	91.7	0.847
W	182	26.4	0.100
	183	14.4	0.047
	184	30.6	0.111
	186	28.4	0.123

In the (n,2n) reactions the emerging neutrons have less energy than the parent neutron. This increases the number of low energy neutrons that will be captured. The sum of the (n,2n) and inelastic cross sections is of the order 1 to 2 barns for the lead, tungsten and iron. Elastic scattering also occurs with theses materials but the energy loss is negligible. However elastic scattering increasing the average path length of the neutrons in the shielding material and therefore increases the probability of further inelastic or (n,2n) reactions occurring.

Room shielding

The majority of linacs are situated in concrete bunkers where design has been centred on the photon hazard. Photons are typically more penetrating than neutrons for the energy ranges in which a linac operates (IPEM 75). The average neutron energy from such machines is never in excess of 1 MeV. This means that adequate concrete shielding for photons should be adequate for neutrons as well due to the hydrogen content in concrete. However, neutron capture is likely to occur in the concrete leading to an increase in the absorbed dose near the walls due to the release of gamma rays.

The neutron field in the concrete bunker is considered to be a combination of neutrons coming directly from the source Φ_{dir}, which follows an inverse square law, a scattered component Φ_{sc}, which is approximately constant throughout the room, and thermal neutrons as a result of neutrons being scattered several times and have attained thermal equilibrium Φ_{the}, which is approximately constant throughout the room (McGinley et al, 1995). This is summarised as the total epithermal neutron flux, Φ, i.e., all neutrons with energy greater than cadmium cut off, 0.41 eV, being $\varphi_{total} = \varphi_{direct} + \varphi_{scattered} + \varphi_{thermal}$

Where: $\varphi_{direct} = \dfrac{aQ}{4\pi\pi^2}$ $\quad \varphi_{scattered} = 5.4\dfrac{aQ}{S}$ $\quad \varphi_{thermal} = 1.26\dfrac{aQ}{S}$

Q is the number of fast neutrons, S is the inside surface area of the treatment room in cm^2, a is the fractional flux transmission of the head shielding (typically 0.85 for tungsten shielding and 1.00 for lead shielding), and R is the distance from the source to the detector in cm.

Maze design

Many linac rooms incorporate a maze design in order to reduce the photon dose rate to a safe level at the exit of the bunker. This enables the room to be constructed without a door allowing easy movement of radiation workers and patients to and from the linac. This is partially a cost saving mechanism as it avoids spenditure on shielded doors and interlocks but also decreases the time required to transport patients and thus increases the number of treatments that can be performed.

Neutrons can easily scatter through large angles with only a minimal loss of energy so the maze needs to be designed with the maximum number of scatters reasonably practicable. The neutrons can also generate capture gamma rays as that can travel around the maze and have been shown to contribute up to half of the dose rate at a maze entrance.

Photons

Photons are produced when energetic electrons interact with the X-ray target material, usually tungsten in the treatment head. The interaction of the photons with matter is random and covers a range of mechanisms that is dependent upon the photons energy. These mechanisms are Raleigh scattering (below 10 keV), the photoelectric effect, Compton scattering (30 keV to 10 MeV) and pair production (5 – 50 MeV). These mechanisms are well understood and will not be discussed further.

The deterministic method

The deterministic method is a straightforward approach to calculate particle transport using formulae. It is relativity fast and does not use random numbers, allowing it to be performed without the use of sophisticated computers.

The method can been used to calculate the photon dose equivalent at the entrance of a radiotherapy bunker. This is by far the most common method used by radiation protection professionals in hospitals and will be outlined below.

Method

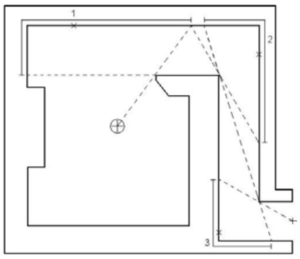
Photon scattering down a maze will come from two main sources – scattering from the patient and the walls of the treatment room. With this method scattering from the floor and ceiling is not considered. Photons will leak from the treatment head however and must be accounted for. The first factors that need to be determined when undertaking scatter calculation is the paths that the photons can take to exit the maze and the areas from which they can scatter.

This is achieved by analysis of the bunker design. Figure 3 shows the layout of the newly commissioned radiotherapy bunker at RSCH. The dose rate, D_Q, received at a point Q from scatter at P is given by, $D_Q = \frac{D_P aA}{d^2}$, where D_P is he dose rate at P, A is the scattering area, d is the distance between the points, and a is the appropriate scatter coefficient.

The first scatter from the patient must include a factor L to account for leakage from the treatment head. This means that the dose rate, D_1, reaching the first wall from scatter in the patient is given by, $D_1 = \frac{D_0}{d_1^2}(a_1A_1 + L)$

Manufacturer guidance dictates that leakage from the head should be less than 0.1% of the main beam dose rate. Therefore a factor of 0.001 is used as a worst-case scenario.

The determination of the scattering areas and distances involves an analytical approach using lines of sight. Whilst photons can feasibly scatter several times around the treatment room and still exit down the maze, it is only those that scatter the minimum number of times that will really contribute to the dose at the maze entrance. Therefore the areas can be determined by working out where on the walls a photon can hit for each scatter and still make it out of the maze in the minimum number of scatters (four in this case, including that in the patient). Figure 3 shows the scattering areas and paths that have been identified for this calculation. The first scattering area, A_0, is the maximum beam size at the surface of the patient (40 cm by 40 cm = 0.16 m^2)



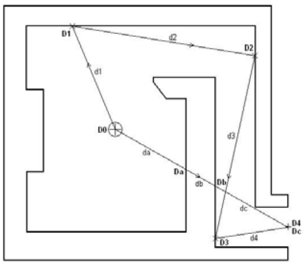


Fig. 3 and 4. Scattering areas and beam paths of photons in RSCH bunker, respectively.

The dotted lines represent the lines of sight used to define the three scattering areas on the treatment room walls. While area one could feasibly be extend to cover most of the left hand wall, the distance travelled by the photons scattering from such points would reduce their energy sufficiently to justify their exclusion from the calculation. The crosses on Figure 3 mark the centre points of the scattering areas. The beam path that scatters from these points has been taken as the average scattering distance for use in the calculation. The path is shown in Figure 4.

The dose at the entrance to the maze will also include a component from the radiation that has scattered in the patient and subsequently penetrated the bunkers secondary shielding. The dose follows the line d_a, d_b, and d_c in Figure 4.3. The total dose at the entrance to the maze will be the sum of the radiation scattered down the maze (D_4) and that penetrating the wall (D_c).

The dose rate reaching the maze entrance from scatter down the maze is given by $D_4 = \frac{D_3 A_3 a_3}{d_4^2}$ where, $D_3 = \frac{D_2 A_2 a_2}{d_3^2}$ etc.

The values of a_0 for the 6 MeV beam have been taken from IPEM Report 75 Appendix VI table VI.2. The corresponding 10 MeV beam value is taken from NCRP Report 144 (Figure 4.14). These values have been normalised to a scattering area of 0.16 m^2. The scatter coefficients used to describe scatter at the walls comes from IPEM Report 75, Appendix VI Figure VI.10. This value assumes equal angles of incidence and scattering and that the walls are made of ordinary concrete. Table 2 shows all the necessary areas and distances of the RSCH bunker.

Table 2. Data for scattered dose calculations.

Photon Beam Energy (MeV)	Parameter					
	D0 (Gy/hr)	A0 (m2)	A1 (m2)	A2 (m2)	A3 (m2)	a0
6	360	0.16	42.28	38.27	29.27	0.00240
10	360	0.16	42.28	38.27	29.27	0.00037

	Parameter						
	a1	a2	a3	d1 (m)	d2 (m)	d3 (m)	d4 (m)
6	0.022	0.022	0.022	4.50	8.10	7.15	3.30
10	0.022	0.022	0.022	4.50	8.10	7.15	3.30

In order to calculate the dose through the wall the tenth value layer must be determined.

The tenth value layer, the distance at which photon intensity is reduced to a tenth of its initial value, must be calculated for the maze shielding. The shielding at the

RSCH is composed of a concrete and magnetite homogeneous mixture; the constituent elements of this mixture are listed in Table 3. The percentage weights of the materials are accurate to within 2 – 3%.

Table 3. Elemental analysis of Magnetite concrete at RSCH.

Element	Weight kg/m3	% Weight
Ca	237	16.6
O	1205	30.5
Si	115	2.9
Al	24	0.6
Fe	2143	54.3
H	20	0.51

The chemical composition was determined to be $Ca_{14}Si_2Fe_{51}O_{30}AlH$. Using a photon cross-section database (XCOM) the mass attenuation coefficient for the compound was determined for a 6 MeV and a 10 MeV beam to be $2.9\text{x}10^{-2}$ and $2.7\text{x}10^{-2}$ cm^2/g, respectively. Tenth layer values of 20.03 and 21.30 cm for the 6 and 10 MeV beams, respectively was calculated using ln10/mass attenuation coefficients.

If the dose rate incident on the inner side of a wall n TVL's thick is D_a then the dose rate D_b coming out of the other side is $D_b = D_A \times 10^{-n}$.

The thickness of the wall along the beam path (d_b) is 120 cm. This means that n is approximately 6.0 for 6 MeV and 5.6 for 10 MeV.

The dose rate, D_c, reaching the maze entrance, a distance d_c from the outside of the wall is then given by, $Dc = \frac{D_A \times 10^{-n}}{d_c^2}$ where, $D_A = \frac{D_0}{d_a^2}(a_0 A_0 + L)$

Results

The dose rates from scatter down the maze (D_4) and leakage through the wall (D_c) are summed to give the total dose rate expected at the maze entrance, D_{TOT}. The results for 6 MeV and 10 MeV beams are summarised in Table 4. The total neutron dose at the end of the maze was calculated to upto four times greater than the photon dose.

Table 4. Dose rates at maze entrance.

Energy (MeV)	D4 (µSv/hr)	Dc (µSv/hr)	DTOT (µSv/hr)
6	0.3397	2.2×10-3	0.3422
10	0.2600	5.6×10-3	0.2656

Discussion

The deterministic method produced a dose rate at the maze entrance of 0.34 µSv/hr for the 6 MeV beam and 0.27 µSv/hr for the 10 MeV. Assuming a 40 hr week, over a 5-day period, a dose rate of 0.014 and 0.011 mSv per week, respectively.

The results show that the higher the initial photon beam energy the lower the dose rate at the maze entrance. This suggests that the higher energy photons are able to penetrate the shielding more effectively, thus resulting in less scatter. This does not imply that the photons will escape through the shielding as the magnetite/concrete mixture has TVL's great enough to attenuation the beam.

The method however relies up on a series of approximations and correction factors to complete the calculations, such as: leakage from the treatment head is assumed to be no worse than 0.1 % of the beams dose rate, there is an average distance travelled by the photons, floor and ceiling scatter is negligible, the tenth value layers rely upon knowing an accurate elemental analysis of the shielding material., the scattering coefficients used are only defined for a water phantom and concrete.

Monte Carlo method

The Monte Carlo method uses random numbers to solve complex problems, as opposed to the deterministic method. It uses simulations to mimic real world process that can occur inside radiation bunkers and produces an expected value from a random variable.

The simulation charts a particles life history within the designed system, from its birth until it ends either being captured or it leaves the system. This is a very complex process due to the probability distribution that governs particle behaviour. For a neutron produced in a linac its emission is isotropic, therefore its direction and location is a function of probability, this leads to random interaction locations within the bunker and so whether the particle is captured or scattered (elastically or in-elastically) is random, hence the term random walk. (Chin 1999).

The Monte Carlo process is then a system that samples statistically the probabilistic distributions governing individual probabilistic events (Chin 1999).

Many Monte Carlo codes have been developed for general and specific purposes. The simulation for this work was created using a general Monte Carlo code for tracking the life of neutrons, photons, electrons, or coupled neutron/photon/electrons, within a bunker.

Original bunker configuration

The total physical dimensions of the radiotherapy bunker were 11.9 x 9.7 x 3.7 m^3. The main linac room is 6.9 m in width and 8 m long and has primary corridor 11.35 m long and 2 m wide. The main room also has a baffle that protrudes 1.7 m inwards with a nib on its end at an angle of 50°. The primary shielding mainly attenuates photons that are initially directed in that field of view as they are emitted from the treatment head and is 1.5 m thick. The bunker entrance is on the other side of the secondary shielding and has no door in order to facilitate the transport of patients in and out the treatment room as mentioned in Section 2.3. This leaves a gap 2 m wide from which neutrons can escape.

Modification of the bunker

Separate simulations were run with the bunker modified to include two baffles in the treatment room, boron lining along the bunker entrance and a boron door. Figure 5 displays the layouts and relative thickness are the materials added to the bunker.

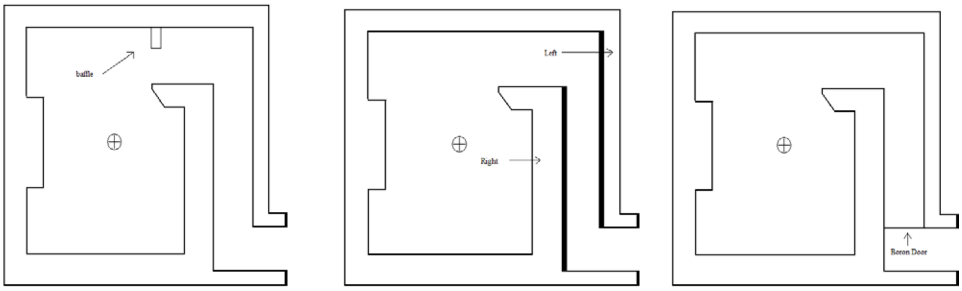


Fig 5. Location of added baffle, boron absorbers and door, respectively.

Results and discussion

The original bunker design, which is in operation at the RSCH, yields a neutron dose of 8 x 10^{-11} per neutron/Gy at the maze entrance for neutrons with an initial energy of 1 MeV. The introduction of the baffle shows a reduction in dose of 40%. This reduction in the neutron dose is due to the increase in surface area of the maze and the effective addition of an extra corner. The increase in surface area acts to minimise the total neutron flux. The fast neutrons have a greater probability of undergoing additional scattering, which in turn lowers their energy and leads to an increase in the number of thermal neutrons. This increases the probability of capture in the bunker. The addition of further corners in the maze result in additional shielding from concrete and thus the hydrogen content is increased.

Addition of boron lining to the bunker walls, reduced dose by 39 % at the entrance. The addition of boron to both walls increases the probability of absorbing thermal neutrons, which accounts for the lower dose.

Addition of a boron door results in a neutron dose rate reduction of 56 % for neutrons in the 1 MeV range. This is due to the characteristics of boron described above and also because the door acts as barrier to scatter the neutrons back into the main treatment area. This increases the average number of scatters that the neutrons will undergo and thus increase the probability of neutron capture within the bunker shielding or the boron door. The dose still delivered to the maze exit is most likely a result of the high-energy neutrons that were able to penetrate the door and then become thermalised and secondary photon emissions from neutrons that were captured by the boron. The addition of the door appears to be very effective but ultimately this must be balanced between the effectiveness of moving the patient in and out of the treatment room, the risk of door interlocks failing and the presences of a partially irradiated door accessible to the public.

Conclusions

The deterministic method produced a dose rate at the entrance of the maze of 0.34 µSv/hr for the 6 MeV beam and 0.27 µSv/hr for the 10 MeV. However, this value does not take into account the neutron dose at the end of the bunker that is stated to be four times larger than that of the photon dose. The results were gained through the use of several correction factors and approximations, such as the floor and ceiling scatters being ignored, an average path length for the beam to travel along and the use of scatter coefficients that only exists for water phantoms and concrete. This means that the results from this method may not be accurate and have to be viewed with caution.

Using Monte Carlo simulations a neutron dose of 8 x10^{-11} neutrons/Gy resulted at the entrance to the radiotherapy bunker. The addition of a baffle resulted in a reduction of dose of 40 % due to an increase in surface area and hydrogen content. This resulted in a greater number of scatters and an increase in the neutron energy loss per collision.

The addition of boron to the maze wall was found to reduce the neutron dose by up to 39 %. This was accomplished due to the large capture cross section of boron for neutrons around thermal energies. Higher energy neutrons scattered off this material towards the maze entrance relatively unaffected by the boron. Scattering neutrons off moderating materials before they reach the boron could increase the effectiveness of the boron by reducing neutron energies to thermal levels

A boron-lined door was then introduced to the bunker, having the effect of reducing the dose by 56. The position of the door along the maze resulted in the scatter of high-energy neutrons back into the treatment room whilst thermal neutrons were captured. The dose still received at the maze entrance is likely to be combination of very high-energy neutrons that were able to scatter through the door and that of secondary photons that were emitted during the capture of neutrons within in the door. Although this simulation resulted in the lowest dose it would probably not be used as it would slow the movement of patients in and out the treatment room.

The simulations demonstrate the effectiveness of using a method such as Monte Carlo. By statistically sampling the probabilities of how a large number of neutrons will interact an accurate process can be simulated. This method may require significant input initially to write the code and collect the inputs needed to create the simulations, such as the shielding constituents and maze design. Once this is complete the simulations do not require the use of correction factors and approximations, producing accurate results with an associated error. The method also allows different variables to be studied in the bunker, such as the neutron flux, and can be modified relatively easily to simulate alternative geometries and shielding.

In relation to the radiotherapy bunker the accuracy and ability to adapt inherent to the Monte Carlo code makes it a far more useful tool than the deterministic method. This accuracy is at the initial cost of the time needed to construct the code but does negate the need to search through literature for scatter coefficients and other approximations.

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Practical implications of the RELID (Retrospective Evaluation of Lens Injuries and Dose) project in the radiation protection of medical professionals

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Abstract

An international study on Retrospective Evaluation of Lens Injuries and Dose (RELID) was carried out under the coordination of the International Atomic Energy Agency (IAEA) with the collaboration of the Latin American Society of Interventional Cardiology (SOLACI). Two surveys were performed during regional congresses of SOLACI and included an evaluation of potential radiation-induced lens opacities in a cohort of interventional cardiologists (IC), nurses and technicians working in cardiac catheterization laboratories, as well as a control group of non-medical professionals whose eyes were never exposed to ionizing radiation. Retrospective eye dose calculations were made by combining data from experimental scatter dose measurements in standard catheterization laboratories with answers to occupational and work-practice details from questionnaires and oral interviews of all participants. Personal dose records were requested to the participants. A total of 116 exposed individuals were screened. A control group of 93 non-exposed persons was included. Posterior sub capsular opacities were found in a 38% of cardiologists and in a 21% of paramedical personnel as compared to 12% of controls. Personal dose records values were practically not available in most of the cases. From the questionnaires fulfilled by the participants, cumulative professional life-time equivalent doses to the lens until several Sv were estimated. The use of personal protection devices reported by the some of the subjects with higher doses was scarce or never used. These findings demonstrate the necessity to improve the education in radiation protection for interventional medical and paramedical professionals especially in the aspects affecting occupational doses and on the importance of using ocular radiation protection tools. Personal dosimetry methodology for these professionals should be revisited and the need of the proper use of personal dosimeters should be highlighted.

Risk of occupational radiation-induced cataract in medical workers

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Abstract

The cataract is the most common degenerative opacity of the crystalline lens, developing with aging. Physical agents causing the cataract may be as follows: infrared, ultraviolet and microwave non-ionizing radiation, electric power and ionizing radiation. Other risk factors of cataract are the following: alcohol abuse, tobacco smoke, systemic diseases (endocrine and metabolic), impaired eye circulation, ocular pressure, refractory eye abnormalities (myopia), blood pressure, heart conditions, low food antioxidant level. The study involved 3240 medical health workers in Serbia, who used to work within and beyond the area of ionizing radiation. The annual periodical-preventive controls of health workers included the eye examination as well. After visual acuity measurement, the lens was examined by retroillumination method (red reflex) and using the biomicroscope. In case of impaired visual acuity or presence of cloudy lens, the examination was performed in mydriasis. X-ray radiation may be a significant cofactor of cataract in radiological technicians. It is an occupation-related disease and may, along with other impairments, contribute to diagnosis of chronic radiation syndrome as occupational disease. Absorbed doses in exposed subjects with cataract, measured by thermoluminescent personal dosimeter (TPD) ranged from 2.64 (min.) to 48.10 (max), mean value: 7.58 ± 4.78 mSv / 5 years (1.59 ± 30 mSv / year), and were not different from mean absorbed annual doses in the exposed subjects without cataract (1.63 ± 1.45 mSv / 1 g), being below maximally tolerated dose of 100 mSv / 5 years 920 – 50 mSv annually). It is evident that chronic exposure to low doses is not the cause of occupational radiation cataract as an independent occupational disease. The effects of radiation to biological DNA material failed to be proven on cataract development, it was rather related to type of radioactive emission, mode and type of radiation, working burden and working conditions as well as type of job.

Setting up a whole body counting system in Portugal

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Abstract

We have recently set up a Whole Body Counter (WBC) at the ITN in Portugal (type Canberra Accuscan). This system is currently prepared to respond to any type of radiological emergencies adequately, if any should happen in Portugal. Also, in the near future, we intend to start a series of routine measurements of workers both inside and outside the ITN.

The WBC contains a moveable Canberra HPGe type GC 2520 detector, and is calibrated with the help of the GENIE 2K, and the ABACOS softwares. We use a calibration source consisting of a cocktail of several radionuclides, with photon energies within the range [55, 1900] keV. For the calibration, we use a Canberra RMC-II phantom in two different types of measuring geometries: the thyroid and the whole-body – both in BOMAB phantom type equivalent calibration arrangements. The calibration is checked on a daily basis, as well as the background.

This is being complemented by Monte-Carlo simulations. The simulations are being performed using the State-of-the-art MCNPX and PENELOPE codes and recent comparisons of calculations with measured results are extremely satisfactory.

Finally we have established collaborations on the European level to participate in future inter-comparisons, as well as in joint-projects, in the framework of EURADOS, with gradual harmonization with EURADOS and EC recommendations in mind.

In this talk we will discuss the actual and future projects associated with the applications of this equipment in the framework of Portugal's Radiological Protection policies, as well as present in some detail the results we have obtained so far.

Internal dosimetry at the Institute of Atomic Energy POLATOM in Poland

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Abstract

Nuclear Centre Świerk is the main and the biggest nuclear centre in Poland, where Institute of Atomic Energy POLATOM, the Andrzej Soltan Institute for Nuclear Studies and Radioactive Waste Disposal Unit are located in the same site. Institute of Atomic Energy operates a nuclear reactor and there is also the Radioisotope Centre which manufactures radiopharmaceuticals and other unsealed radiation sources. Waste Disposal Unit collects and treats radioactive waste of low and moderate activity from the whole country.

Monitoring of the internal exposure of employees is performed by Radiation Protection Measurement Laboratory (LPD) of the IAE, which is the only laboratory of this type in Poland accredited by Polish Centre for Accreditation. LPD provides also monitoring of the internal exposure for employees from other institutions and sometimes serves as a public service in case of post accident contamination.

LPD uses a whole body counter and thyroid counter for *in vivo* measurements and performs *in vitro* measurements using flow counters and liquid scintillation analyzers. All measurement results are gathered in data base which contains data since 1986.

This paper shortly describes the procedures used in the laboratory and lessons from intercomparisons. Results of monitoring in 2008 are discussed for the group of 378 persons subjected to the measurements with whole body counter, 128 persons examined with thyroid counter and 122 persons using *in vivo* radiochemical. Occupational contamination with iodine ^{131}I was registered in 21 cases by whole body counter and in 57 cases by thyroid counter. Radiochemical measurements showed 9 cases of contamination (5 with ^{32}P, 4 with tritium, 2 with ^{90}Sr and 1 with ^{35}S). Contamination with ^{137}Cs was registered in 53 cases. This isotope is still present in the environment in Poland after the Chernobyl accident.

Introduction

Nuclear Centre Świerk is the main and the biggest nuclear centre in Poland. It is located 30 km from Warsaw. Institute of Atomic Energy POLATOM, Andrzej Soltan Institute for Nuclear Studies and Radioactive Waste Disposal Unit are located there.
Institute of Atomic Energy operates a research nuclear reactor and a part of it is the Radioisotope Centre which manufactures radiopharmaceuticals and other unsealed

radiation sources. Employees who are working in these units are exposed on internal contamination.

Monitoring of the internal exposure of employees is performed by Radiation Protection Measurement Laboratory (LPD) of the IAE POLATOM, which is accredited by Polish Centre for Accreditation according to the norm ISO/IEC 17025. LPD provides also monitoring of the internal exposure for employees from other institutions and sometimes serves as a public service in case of post accident contamination. All measurement results are gathered in special data base which contains data since 1986.

Other fields of LPD activity are environmental measurements and calibration of dosimetric equipment. LPD is the only one Polish laboratory offering the full range of internal dosimetry measurements.

Measurements methods used in LPD

In vivo measurements methods

There are two stations for *in vivo* measurements in LPD – whole body counter and thyroid counter.

Whole body counter in Świerk is the only one of this type of equipment in Poland. Around 400 measurements per year are performed there.

Whole Body Counter is placed in a special room which walls are built from 20 cm of iron, 5 mm of lead, 2 mm of cadmium and 1 mm of cooper. The equipment used for the measurements is HPGe detector with multichannel analyzer, both produced by SILENA. Measurements are done in chair geometry, time of one measurement is 20 minutes, BOMAB phantom is used for calibration. Whole Body Counter with BOMAB phantom during calibration is presented in figure 1.

Fig. 1. Whole Body Counter in LPD with BOMAB phantom during calibration.

The thyroid counter used for measurements in LPD contains the NaI(Tl) detector produced by CANBERRA and multichannel analyzer Tukan8k produced by Andrzej Soltan Institute for Nuclear Studies [Guzik et al. 2006].

LPD is using own-construction thyroid water phantom for calibration. The water phantom is a cylinder shaped Lucite (PMMA) vessel (128 mm diameter, 165 mm high), with a cover and two small (13 cm^3) cylinder vessels inside. During the measurements, the bigger vessel is filled with distillated water and the small ones, which simulate the thyroid lobes, are filled with the reference solution of radioisotopes. Construction of the phantom makes it possible to move the small vessels in two directions (horizontal and vertical). The whole phantom can rotate on its base and certain angle of rotation can be fixed. There is also a possibility to use the vessels of different volume.

During calibration measurements the whole spectrum of the emitted radiation is measured to determine the ratio of the counting rate in the main energy peak and in the Compton scattering band. The result of calibration is the ratio between detection efficiency and the depth of the ^{131}I source in water thyroid phantom. This method allows to detect the cases of non-typical distribution of the iodine in the patient's neck and to estimate the associated uncertainties of the measured activity. [Ośko et al. 2007A, Ośko et al. 2007B]

The thyroid counter used in LPD is presented in figure 2 and thyroid water phantom is presented in figure 3.

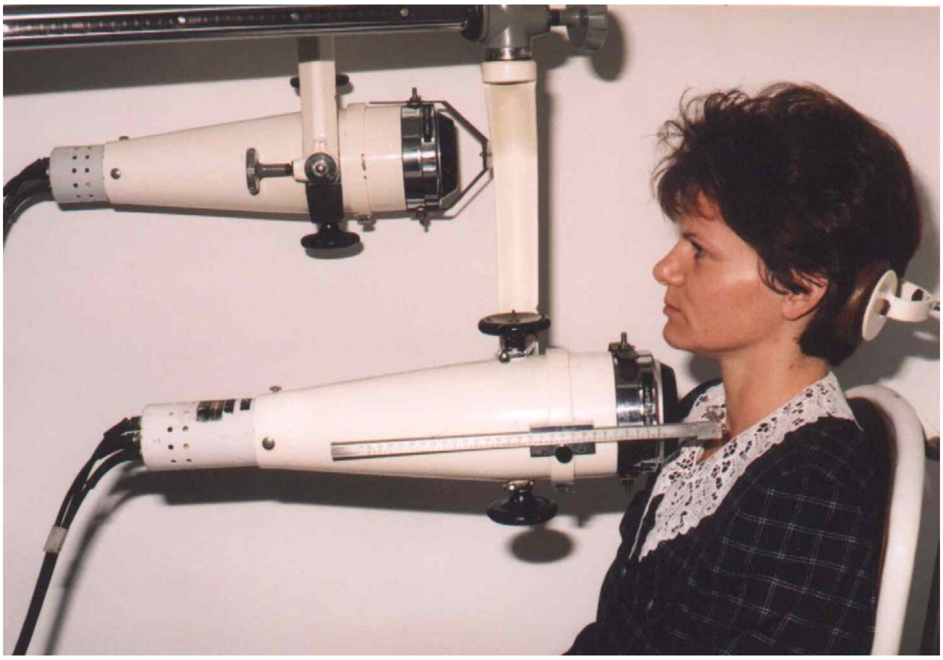

Fig. 2. Thyroid Counter in LPD.

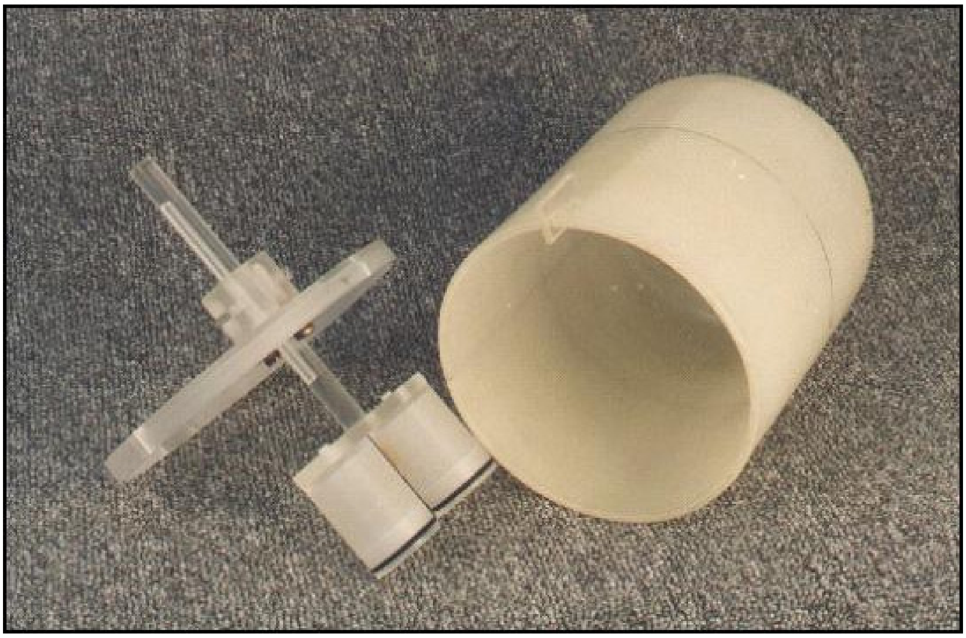

Fig. 3. Thyroid water phantom used for calibration of thyroid counter in LPD.

In vitro measurements methods

Radiation Protection Measurement Laboratory performs *in vitro* measurements to determine:

- contents of radioactive beta isotopes: sulfur ^{35}S, phosphorus ^{32}P, carbon ^{14}C, tritium ^{3}H and strontium ^{90}Sr (procedures accredited by PCA);
- total alpha and beta activity (not-accredited procedures);

in human urine.

The activity of ^{35}S, ^{32}P and ^{14}C in urine is determined by direct measurement in Liquid Scintillation Analyzer (LSC). Definite volume of daily collection of urine is mixed with scintillator. Next beta radiation activity of cocktail is being measured in applicable energetic channel of the counter, which depends on determining element.

The activity of tritium ^{3}H in urine is also measured in LSC. Before measurement the sample must be properly prepared. Urine is discoloured by active carbon, sodium carbonate (Na_2CO_3) and sodium thiosulfate ($Na_2S_2O_3$) are added, next a distillation is carried out. The definite volume of distillate is mixed with scintillator and beta radiation activity is measured with LSC.

Chemical preparation of a sample is necessary to determine strontium ^{90}Sr activity in urine. At the first stage deposit is precipitated in the presence of oxalic acid, with oxalates strontium ^{90}Sr and yttrium ^{90}Y are co-precipitated. After deposit mineralization, the sample is transformed to liquid form. The laboratory determines activity of ^{90}Sr indirectly, the measure of the activity of ^{90}Y, it is possible because ^{90}Sr and ^{90}Y are always in equilibrium. As a result of multi-stage extraction the extract is received and then evaporated on the measuring plate. Next the activity of deposit is determined by measuring system with the proportional flow counter.

The method of determining the total beta activity in urine consist of co-precipitation, in orthophosphoric acid environmental, alkaline earth phosphates. Next the forming deposit is mineralised in muffle furnace to ashes and beta radioactive nuclide of ashes is measured with system with proportional flow counter.

The method of determining the total alpha emitters in urine consist of co-precipitation with calcium phosphate. After mineralization of deposit, the hydrolysis is carried out. Alpha emitters are reduced by sodium sulfite and adsorbed on glass filter. Alpha emitters are eluted from filter by using hydrochloric acid, then sample is evaporated to dryness and after dissolving in nitric acid, sample is transferred to measuring plate. The measurement of deposit activity is made with the proportional flow counter.

The measurements of activity of ^{35}S, ^{32}P, ^{14}C, ^{3}H and ^{90}Sr and total alpha and beta activity in urine are required in order to estimate the value of effective dose.

All measurements data are recorded in special data base and information about registered effective doses are send to National Atomic Energy Agency.

Intercomparisons

As accredited laboratory LPD shall validate developed methods to confirm that the methods are fit for the intended use. One of the techniques which can be used for the determination of the performance of a method is interlaboratory comparison.

Each year LPD participates in international comparison of *in vitro* measurements organised by French association PROCORAD (Association for the Promotion of Quality Control in Radiotoxicological Analysis).

Unfortunately LPD is the only one Polish laboratory which is accredited in whole body measurements and one of two accredited in thyroid measurements (the second is Central Laboratory for Radiological Protection (CLOR)). This is the reason why the national comparisons of *in vivo* measurements are not organized in Poland.

The last international comparison in which LPD was participated was organised in 2002 by International Atomic Energy Agency. The results of this comparison have not been published until now.

LPD organized comparison for thyroid measurements for two laboratories – itself and CLOR in 2008.The measurements has been performed for two thyroid phantoms – LPD water thyroid phantom and RSD thyroid phantom. The results of this comparison are presented in figure 4 [Pliszczyński et al. 2008].

As it was shown, all measurements results were within the uncertainties range ± 25 % as well as for thyroid water phantom and for RSD phantom.

LPD is going to participate in international comparisons organized by IRSN, France. The comparison for thyroid counters will be organized in 2011 and for whole body counters in 2012.

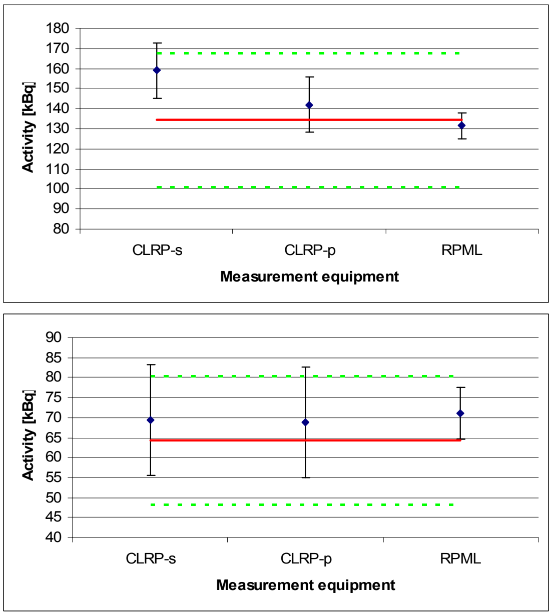

Fig. 4. The measurements results of iodine ^{131}I activity in water phantom (upper) and in RSD phantom(lower). The solid red line represents the standard sample's activity and the dashed lines – the probable error ± 25 % [Pliszczyński et al. 2008].

Measurements results in 2008

The monitoring of internal occupational exposure in 2008 contained:

- 378 measurements with whole body counter of group of 268 persons,
- 128 measurements with thyroid counter of group of 15 persons,
- 122 *in vivo* measurements using radiochemical methods of group of 102 persons.

Occupational internal radiation contamination was registered in 185 cases. All registered radionuclides are presented in table 1.

Table 1. Occupational internal contamination registered in LPD in 2008.

Radionuclide	Measurement method	No. of Cases
^{131}I	thyroid	57
^{131}I	whole body	21
^{32}P	in vitro	5
tritium	in vitro	4
^{90}S	in vitro	2
^{35}S	in vitro	1
total beta	in vitro	95

Contamination with ^{137}Cs was registered in 53 cases. These cases are not presented in table 1 because this is not occupational contamination. ^{137}Cs is still presents in the environment in Poland after the Chernobyl accident. The activity of caesium in human body depends on place of living and diet – especially mushrooms cause the increasing of ^{137}Cs activity.

Iodine activity registered with whole body counter is not used for calculation the effective dose. These measurements are treated as preliminary. In that case patient is sent for thyroid measurement and after that effective dose is calculated.

In all 57 cases of iodine contamination using thyroid counter effective doses were higher then 0.01 mSv, in 11 of them were higher then 0.1 mSv. The highest annual dose for one person was 1.53 mSv which equals 7.65 % of annual effective dose limit for employee category A. For other persons it was below 1 mSv.

Effective dose calculated after *in vitro* measurements were higher then 1 µSv only in 4 cases. The highest registered dose was 4.83 µSv which equals 0.02 % of annual effective dose limit for employee category A.

Conclusions

Radiation Protection Measurements Laboratory provides monitoring of internal occupational exposure for a numerous group of employees from Polish nuclear sites where the sealed radiation sources are used. The level of contamination registered in 2008 was very low comparing with annual effective dose limit. The highest level was registered for employees who work in sector of radioiodine production but it was still much below the limits.

The monitoring results showed that work conditions in Polish nuclear sites are properly kept in according to principles of radiation protection. The results obtained in earlier years also confirmed this. The internal contamination is rare occurrence and even if they are registered the dose level is very low.

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Designing and using a veterinary megavoltage X-ray facility

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Abstract
Until now cancer in companion animals is being treated in the Clinic for Companion Animals of the Faculty of Veterinary Medicine, Utrecht University, mainly by surgery and/or chemotherapy. The Faculty is currently building a megavoltage X-ray facility. In April 2010 treatment with radiotherapy using a linear accelerator will be started. The use of linacs in veterinary medicine is increasingly becoming more common and therefore the building of more structural shieldings in the near future is foreseen in Europe. The design of the structural shielding for the veterinary radiation therapy facility is based on the NCRP 151 (2007). Based on concrete, a shielding design goal (P) of 10 μSv/y, as prescribed by the national radiation control agency, a workload (W) at 1 m from the X-ray target of 17000 Gy/y (91% 6 MV and 9% 10 MV), a use factor (G) of 0.25 and an occupancy factor (T) of 0.25 for the residential environment, the primary barrier has been calculated at 150 cm concrete. The design of the maze without a special entrance door, is relative long and has multiple legs. Because of this only small companion animals (i.e. cats and dogs) can be treated. As the endpoint energy of the accelerator is below 10 MV the presence of photoneutrons and neutron capture gamma rays are disregarded. After presenting calculations of the primary and the second barrier locations, the design of the structural shielding was approved by the national state radiation control agency. Building the veterinary radiation therapy facility and purchasing the linear accelerator has been licensed in January 2009. As soon as the accelerator has been made operational, a preliminary survey shall be carried out to ensure that radiation exposures to the installation engineer and personnel near the facility do not exceed the applicable shielding design goal. Once the accelerator has been made completely operational and an initial calibration has been completed, a complete radiation survey shall be conducted.

Beta doses from handling UO_2 pellets at a nuclear fuel factory; Monte Carlo based simulations and TLD measurements

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Abstract

Aim: The objective with this project was to establish beta radiation doses to workers (skin and eye doses) from UO_2 pellet production work in a uranium fuel factory in Sweden.

Material and methods: The beta energy spectra emitted from UO_2 pellet plates were obtained by Monte Carlo (MC) simulations using the MC-code MCNP5, yielding beta fluxes (from Pa-234m decay) at different positions and distance from the source. Also, the dose-rate distributions in soft tissue were calculated in order to obtain skin and eye lens doses. Validation of the MC-calculations were made by TL-based field measurements, where thin TL-discs of (i) LiF-PTFE; thickness 0.13 mm, and (ii) LiF-discs; TL-active thickness 0.04 mm, were used as follows: TL-discs were stacked in PMMA cylinders with a 8 mm deep hole holding the dosimeters and then exposed for 48 hours at various distances above source centre simulating workers hand and eye positions. In addition, the legal personnel badge dosimeters (LiF-discs, thickness 0.9 mm), with one closed position (3 mm plastic cover) to measure (gamma) deep dose and one open position for shallow dose, were exposed at several distances from the source. To assure good quality in the TL dose data, each dosimeter type was calibrated in beta fields from Sr-90/Y-90 at 0, 30 and 60 degrees angle of incidence at known doses (Hp(0.07) at the National Standard Laboratory at SSM in Stockholm.

Results: The MC-calculated beta dose distribution agreed well with field measurement data. Both at positions for workers hands and head/eyes, the beta radiation clearly dominate the contribution to Hp(0.07) over the gamma contribution. The badge dosimeters severely underestimate the shallow beta doses due to their location and orientation but also add dose to the closed position TL-disc which will then overestimate the deep dose. To obtain good beta dosimetry for the workers, i.e. for extremity and eye lens doses, thin TL-dosimeters (e.g. 0.13 mm thickness) in thin plastic bags are suggested.

Review of the constraints of the effective dose levels at OKG NPP

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Abstract

The maximum level of the effective dose a worker that at a Swedish NPP accumulates during a year or over a five year period is regulated by the Swedish Authorities. In order to meet up to these standards, the NPP at Oskarshamn in Sweden (OKG AB), introduced constraints in the 1990s. These values were mainly based on the standards and limits set by the authorities and not so much the conditions at the NPP or up-to-date research in the field.

Recently a new electronic personal dosimeter system was introduced at OKG AB. It is now possible to determine both the effective dose and the dose rate in intervals as low as one minute and this can be followed online. It opens up for a whole set of possible evaluations, one being the trending of doses and dose rates. This is valuable information for setting our own standards regarding constraints of the effective dose/dose rate levels. The review of the constraints of the effective dose and dose rate levels at OKG AB has been carried out taking into account the trending parameters of the station, estimations given from the electronic personal dosimeter system, latest research results, PR policy at OKG AB, α-value of ALARA and limitations set by the Swedish Authorities.

Low dose radiation-induced non targeted effects – How the changing paradigm impacts radiation protection of biota?

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Abstract

Ever since the grudging acceptance that non-targeted effects (NTE) can be measured in unirradiated cells or distant progeny of irradiated cells, the discussion has raged about the relevance of these effects for radiation protection and risk assessment. Arguments will be made in this presentation, that NTE may call into question radiation effects pillars such as the LNT model. They may also have relevance to wider mechanisms in cancer biology, population ecology and evolutionary biology concerning process of selection, the transmission of heritable traits, the relevance of “social” interactions between cells, organisms and populations and the mechanism by which cells/organisms respond rapidly to environmental stress. This presentation will also argue that a key consequence of findings in NTE biology is that at any given level of organization, from gene to ecosystem – communication of stress signals and heritability of stress adaptations provide the bridges linking one hierarchical level to the next and enable the rapid propagation of change triggered by stress at one level, resulting in change at a higher (or lower?) level. This addresses a major problem in evolutionary biology because while the molecular mechanisms of natural selection are fairly well understood a major knowledge gap exists in translating mutational drift at the level of the individual cell to natural selection at the ecological level where sociobiological factors are so important. The existence of the mechanisms discovered in the NTE field provides a glimpse of a major way that evolution could be regulated through communicated signals between cells, individuals, and populations, These control and optimize responses at the level of the population and coordinate the emergence of exquisitely tuned systems which can adapt rapidly to micro or macro environmental change. Radiation protection strategies for biota mean that an understanding of the driving mechanisms is vital.

The activities of the IAEA in developing standards on radiological protection of the environment

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Abstract

Radiological protection of the environment is an issue that has been intensively discussed for more than one decade. A great deal of progress has been made during this time regarding the capabilities: (i) of estimating the uptake of radionuclides by flora and fauna in different habitats and ecosystems; (ii) in the development of dosimetric models that allow the calculation of internal and external exposures for a wide range of terrestrial and aquatic organisms; and (iii) in investigating and analysing the effects of radiation exposures to biota.

This paper gives an overview of the IAEA's activities in this field and it describes the interactions with UNSCEAR, the ICRP and other national and international bodies and institutions with regard to the work on analysis and evaluation of exposures to flora and fauna. The current status of the discussion on the integration of environmental protection into the radiation protection system is also summarized. Results of case studies are presented that assess the radiological impact to flora and fauna due to discharges of radionuclides to terrestrial and aquatic environments.

Finally, the paper discusses the factors, assumptions and considerations that are regarded as important for risk characterisation and decision making in relation to radiation exposures to biota. It explores the remaining challenges related to the integration in regulating and controlling dischargeable waste.

Introduction

For a long time, the management of radionuclides entering the environment was in general based on the assessment and evaluation of the radiological impact to humans. However, in the last decades, with an increasing awareness of environmental issues, concerns were raised that flora and fauna might be affected by radionuclides released into the environment; it is now considered necessary to demonstrate that the environment is appropriately protected against ionizing radiation (ICRP, 2007).

Considerations on protection of the environment from ionizing radiation started in the 1960s and 1970s (IAEA, 1961, 1975, 1979; IMO, 1972), one important reason for this interest was the practice of disposing of low-level radioactive waste into the oceans.

Concerns rose that this could harm marine flora and fauna, which initiated detailed investigations on the effects of ionizing radiation to biota.

This original scientific interest has been accompanied by new and developing international policies and agreements. Important milestones were the Convention on the Prevention of Marine Pollution by Dumping of Wastes and Other Matter (London Convention) (IMO, 1972) and the Declaration of the United Nations Conference on Environment and Development of 1992 (UNCED, 1992), which stimulated the development of policies and approaches to specifically address impacts of radioactive substances on non-human species

Nevertheless, for a long time risk assessment and management of radionuclides released into the environment were based on radiological impacts to humans alone. In 1990, the International Commission on Radiological Protection (ICRP) (ICRP, 1991), the view is held that — as is also the case in the 1977 recommendations (ICRP, 1977) — the environment implicitly is afforded adequate protection by application of standards that provide for adequate protection of humans. The main problem is related to situations where man is absent or not exposed as in e.g., unpopulated areas or in the sea. In the meantime, this indirect approach is considered insufficient (e.g., Strand and Larsson, 2001) and in the most recent ICRP recommendations (ICRP 2007), the development of a framework is proposed to explicitly demonstrate radiological protection of the environment. The IAEA is following progress in this area and is exploring the need for those cases where it could be considered necessary because of an actual radiological risk, as well as the form and content of any improved international standards.

Current status

Objectives of radiation protection of the environment

The objectives of the radiation protection of humans are to avoid deterministic effects and to limit stochastic effects to individuals (e.g., IAEA, 1996; ICRP, 2007). However, the objectives for the protection of the environment are more complex. The protection of individual organisms is not an issue, approaches in environmental protection always target protection on the population and community levels. Therefore, there is a consensus that the objectives of protection of the environment are related to the conservation of species, the maintenance of biodiversity and the protection of habitats, communities and ecosystems (IAEA, 1999, 2002, 2006; ICRP, 2003, 2007, 2008).

Scientific background

Much work has been done in recent years to develop methodologies to assess and evaluate exposures to flora and fauna, taking into account both the terrestrial and aquatic environments (Figure 1). The European Union has launched two projects, FASSET and ERICA (Larsson, 2004, 2008) in order to develop a framework for assessment and evaluation of radiological impacts to biota from radionuclides discharged to, or already existing in, the environment. In 2005, the ICRP established a Committee dedicated to Protection of the Environment with the aim of setting up a framework for assessment and evaluation of exposures to biota. This work has been supported by the approaches developed during the EU projects FASSET and ERICA.

Due to the enormous variability of nature, it is impossible to take all species into account. For this purpose, within FASSET and ERICA a set of Reference Organisms has been defined representing terrestrial, freshwater and marine ecosystems. With the same intention, ICRP developed, in accordance with the system for the protection of humans from ionizing radiation, a set of 12 Reference Animals and Plants (RAPs) covering plants and animals, and different habitats (ICRP, 2008).

During these activities, models and databases were developed that allow for the:

- estimation of activity concentrations in reference organisms from measured or predicted radionuclide concentrations in soil, water or sediments;
- estimation of internal and external doses to aquatic and terrestrial; and
- setup of relationships between exposure and adverse effects to animals and plants.

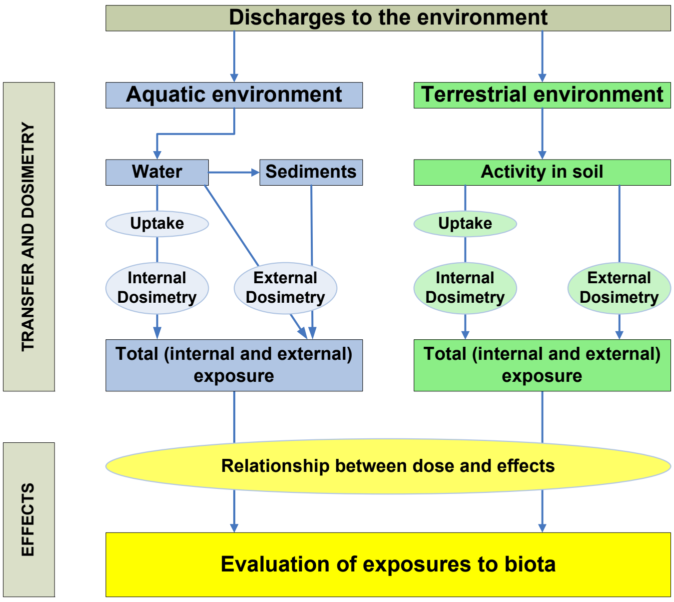

Fig. 1. Overview of the steps to estimate and evaluate exposures to biota in the aquatic and terrestrial environments.

A key issue in considerations on protection of the environment is the evaluation of exposure in relation to effects. However, whereas the objectives of the radiation protection of humans are to avoid deterministic effects and to limit stochastic effects to individuals, the objectives for the protection of the environment are related to the population and community level; as formulated in the ICRP 2007 recommendations, the targets are conservation of species, and to protect habitats, communities and ecosystems.

For this purpose, ICRP has derived a set of Derived Consideration Reference Levels (DCRL) for the 12 RAPs; which usually cover one order of magnitude (Fig. 2). The DCRLs represent bands of doses that are associated with no or very little adverse effects.

Within the EU-funded project PROTECT (Andersson et al., 2009), and on the basis of a statistical analysis of dose-effects relationships, screening levels were derived for vertebrates, invertebrates and plants of 2, 200 and 70 µGy h^{-1} respectively, and a generic screening value of 10 µGy h^{-1} was derived. Due to the approaches used during their derivation, they represent values that can be used to screen out exposure conditions of no concern. These screening values are generally in agreement with the DCRLs.

DCRLs do not represent dose limits, they should be considered as zones of doses at which a more detailed evaluation of the situation would be warranted. Therefore, DCRLs should be applied in a similar way as reference levels for existing and emergency exposure situations.

For this evaluation of the exposure conditions, factors should be taken into account as, e.g., the type of exposure situation (planned, existing, emergency), the size of area that is affected, the time period for such exposures, the fraction of a population of a species that is exposed to such dose levels, the appropriateness of the database used for the dose estimation, and the degree of precaution that is needed for the assessment.

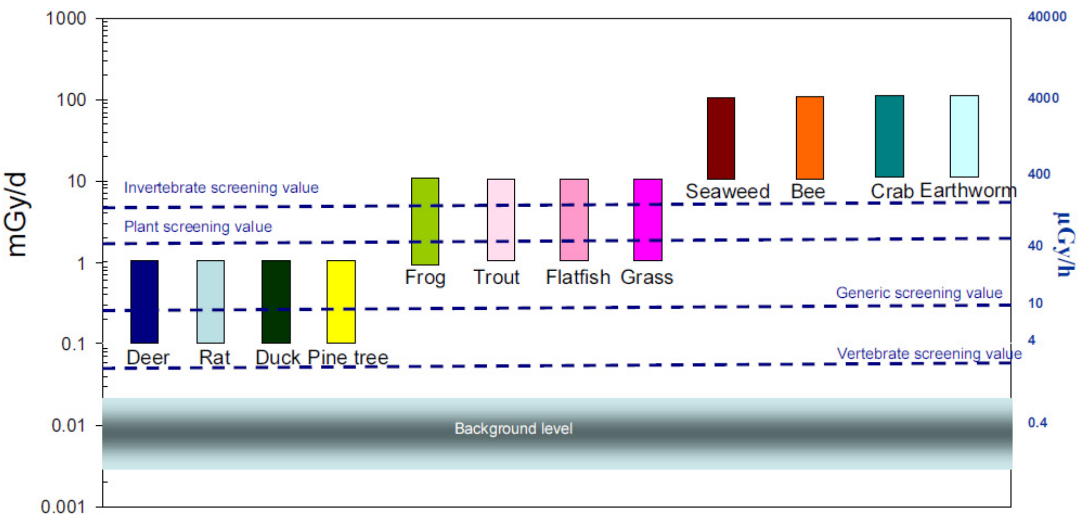

Fig. 2. Comparison of Derived Consideration Reference Levels (DCRLs) for the Reference Animals and Plants (RAPs) as proposed by the ICRP (2008). The figure also gives generic and organism-group specific screening levels as derived within the PROTECT programme for vertebrates, invertebrates and plants, respectively (compared to the generic value, there is less confidence to the organism-group specific values; Andersson et al., 2009).

Examples of exposures to biota as derived from case studies

During recent years, a number of studies were performed to estimate doses to biota under different field conditions. Some results are summarized in Table 1. It should be noted, however, that this list is not complete. The selection given is intended to cover a wide range of contamination situations. Very high exposures are only observed during the early phase of a nuclear accident as illustrated by the examples given for the Techa River, Chernobyl or Kyshtym (Kryshev et al., 1998; Sazykina et al., 1999).

Recently estimated doses to biota in the Chernobyl exclusion zone vary widely as the contaminations vary. The peak values refer to a few highly contaminated areas.

The exposure of biota in Finnish lakes contaminated by the Chernobyl fallout depends on the habitat, i.e., exposures in the water column are low, whereas organisms living on or in the sediments are still exposed at levels of a few µGy h^{-1} due to the persistent contamination of lake sediments with ^{137}Cs.

Table 1. Exposures to biota as estimated in selected studies.

Site and Year	Source of contamination	Radionuclides involved,	Organism	Exposure (µGy h^{-1})	Reference
Techa river, 1951	Nuclear weapons industry		Fish	up to-25000	Kryshev et al., (1998), Sazykina et al. (1999), Brechignac (2003)
Kyshtym, 1957/1958	Nuclear weapons industry		Pines Fish	up to 90000 up to 1000	
Chernobyl 1986	Chernobyl accident,		Pines, rodents	up to 10000	
Chernobyl exclusion zone, 2005	Chernobyl accident, 3 selected sites	^{90}Sr, ^{137}Cs, $^{239/240}$Pu, γ-dose rate=2 – 31 µGy h^{-1} ^{137}Cs in soil: 7-100 kBq kg^{-1}	Small mammals	12-810	Beresford et al, 2008
Lakes, Finland, 2003	Chernobyl fallout	^{134}Cs, ^{137}Cs, ^{90}Sr	Fish, waterplants		Vettiko and Saxen, 2008
			Water	0.03-0.85	
			Sediment surface	2-3	
Loire river, 1999	Chinon Nuclear Power Plant	^{3}H, ^{14}C, ^{131}I, $^{134/137}$Cs	Various plants and animals	$1*10^{-6}$ -$5*10^{-4}$	Berwesford and Howard, (2005)
Norway 2003	Oil and gas rigs	$^{226/228}$Ra, ^{210}Pb, ^{210}Po	Fish, molluscs, phyto- and zooplankton	$5*10^{-3}$ -$7*10^{-2}$	
La Hague, 1996	La Hague reprocessing plant	^{3}H, ^{14}C, ^{60}Co, ^{90}Sr, ^{106}Ru, $^{129/131}$I, $^{134/137}$Cs, $^{238/239/240/241}$Pu, ^{241}Am	Crustaceans, mollusks, round and flat fish, algae	0.05	Chambers et al. 2005
OSPAR region (2005)	Nuclear industry	^{3}H, ^{99}Tc, ^{137}Cs, $^{238/239}$Pu	Macroalgae, crustaceans, vertebrates	10^{-4} - 10^{-1}	OSPAR (2008)

Exposures to biota due to routine discharges of radionuclides, as reported for the Loire River and La Hague, are low. In the latter case, this is also due to strong sea currents at this part of the English Channel, i.e., within a distance of 500 m from the discharge point, the concentrations decrease by a factor of 100 000 (Chambers et al., 2005). Exposures to biota reported for the vicinity of oil and gas rigs are low. In OSPAR (2008), dose rates to marine flora and fauna were estimated; in 2005, the highest estimated dose rates due to radionuclides discharged by the nuclear industry were in the order of 0.1 µGy h^{-1}.

The results indicate that doses to biota above a level of 10 µGy h^{-1} may hardly be achieved under planned exposure situations, since those dose rates are only possible for contamination levels that do not comply with the 1 mSv a^{-1} dose limit for public exposure.

IAEA activities on the development of Safety Guides and Safety Standards in relation to protection of the environment

The IAEA has been active in this field since a long time (e.g., IAEA, 1975, 1988, 1992). One important step during the last decade refers to the Conference "Protection of the Environment from the Effects of Ionizing Radiation" which was held in Stockholm, Sweden in 2003 in cooperation with the United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR), the European Commission (EC) and the International Union of Radioecology (IUR). The main objective of the Conference was to stimulate the development of an international policy on the protection of the environment from the effects of ionizing radiation (IAEA, 2005).

The findings of this Conference provided the basis for the 2005 "IAEA Plan of Activities on the Radiation Protection of the Environment: An International Action Plan on the Protection of the Environment against the Detrimental Effects Attributable to Radiation Exposure". This plan was developed in cooperation with UNSCEAR, the ICRP, OECD/NEA, the EC and a number of other International organizations and was approved by the IAEA's Board of Governors and General Conference in September 2005. The Plan of Activities is targeted at:

- Fostering the cooperation of relevant international organizations in considering radiation protection of the environment;
- Supporting IAEA Members States in their efforts to protect the environment;
- Elaborating how protection of the environment may be integrated into the system of protection from adverse effects of ionizing radiation.

The main recent activities related to these issues are summarized in Table 2. An important step was achieved in 2006 when the IAEA published the Fundamental Safety Principles as these include explicitly the protection of people and the environment, present and future, against radiation risks (IAEA, 2006). It is important to note that the Fundamental Safety Principles are binding for IAEA activities and constitute a key document for the development of IAEA safety standards for protection against adverse effects of ionizing radiation.

Currently, the BSS (Basic Safety Standards for Protection against Ionizing Radiation and the Safety of Sources) (IAEA, 1996) are under revision; the current draft includes requirements that, subject to national decisions, protection of the environment needs to be taken into account during:

- registration and licensing of activities;
- setting of discharge limits for nuclear facilities; and
- optimization of existing and emergency exposure situations.

These issues will be considered in three IAEA safety guides that are currently under development:

- The first is dedicated to guidance on the application of the Fundamental Safety Principles and the requirements of the revised BSS in relation to protection of the public and environment in planned, existing and emergency exposure situations. This is essential to ensure a consistency of approaches that include an integrated consideration of the radiation protection of the public and the environment. It should be noted that the radiological impact to biota under existing and emergency situations is considered as one factor – among many others – during the optimization process, when planning remediation or mitigating actions.

- A further safety guide is under development to give detailed guidance on the assessment of the radiological impact to the environment arising from authorized discharges to terrestrial or aquatic environments. The preparation of a radiological environmental impact analysis is a key component for demonstrating radiological protection of the environment: For this purpose, a graded approach is proposed in order to ensure that the efforts dedicated to safety are commensurate with the radiation risks. The guide will facilitate the development of a standardised approach; it will promote a common understanding of the process, definitions and methodologies, and it will consider environmental aspects in all stages of the life cycle of a facility.
- Furthermore, the IAEA will integrate the requirements formulated by the Safety Fundamentals and the revised Basic Safety Standards during revision of the safety guide on the regulatory control of radioactive releases (IAEA, 2000).

These activities are accompanied by efforts to stimulate the scientific exchange on protection of the environment: In 2009, IAEA launched the programme Environmental Modelling for Radiation Safety (EMRAS II), which is intended to act as a scientific forum to stimulate model development and testing to improve capabilities for assessments of radiological impacts to the public and the environment from radioactivity in the environment. The programme is scheduled to run until 2011. A total of 9 Working Groups were setup within EMRAS II, three of which work on issues related to protection of the environment:

- Working Group 4 “Biota Modelling” focuses on testing and comparison of dosimetric and radioecological models being applied to assess exposures to biota.
- Working Group 5 “Wildlife Transfer Coefficient Handbook” focusses on the collection, analysis and evaluation of transfer parameters being used for the estimation of exposure to biota
- Working Group 6 “Biota Dose Effects Modelling” concentrates on the analysis of the effects of ionizing radiation to flora and fauna, the derivation of dose effect relationships and the impact of ionizing radiation on populations.

For many years the has IAEA advised the Convention on the Prevention of Marine Pollution by Dumping of Wastes and Other Matter (London Convention) (IMO, 1972) and the Convention for the Protection of the Marine Environment of the North-East Atlantic (OSPAR Convention) on questions related to radioactivity in the marine environment and the possible radiological impact to man as well as to marine flora and fauna.

Table 2. Summary of recent IAEA activities in relation to protection of the environment.

Item	Activity	Remark
Safety Principles	Safety Fundamentals, Principle 7: “People and the environment, present and future, must be protected against radiation risks”	IAEA (2006)
Safety Standards	Draft of revised BSS: Consider Protection of the Environment during • Registration and licensing • Setting discharge limits • Optimization in existing and emergency exposure situations	Draft sent out in January 2010 to Member States for Comments
	Implement requirements related to protection of the environment as defined in the Safety Fundamentals and the new BSS in new Safety Standards on: • Regulatory Control of the Releases of Radioactive Material from Facilities and Activities • Radiological Environmental Impact Analysis for Facilities and Activities • Radiation Protection of the Public and the Environment	Development initiated in 2010
Scientific Basis	EMRAS: Environmental Modelling for Radiation Safety:	
	• EMRAS I (2003-2007), working group on ○ Model validation for biota dose assessment	http://www-ns.iaea.org/projects/emras/
	• EMRAS II (2009-2011): working groups on ○ "Biota Modelling" ○ "Wildlife Transfer Coefficient" Handbook ○ Biota "Dose Effects Modelling"	http://www-ns.iaea.org/projects/emras/emras2/
Advisory Work	Giving advice in questions related to radioactivity in the oceans and resulting exposures to man and biota to: • Convention on the Prevention of Marine Pollution by Dumping of Wastes and Other Matter (London Convention) • Convention for the Protection of the Marine Environment of the North-East Atlantic (OSPAR Convention)	

The development of safety standards as specified above requires the integration of issues related to environmental protection into the system for radiation protection of humans. A scheme of an approach for protection of humans and the environment is shown in Figure 3. For all three exposure situations (planned, existing and emergency) the assessment of exposures to both humans and biota start with the radionuclide concentrations in the environment. Exposures are calculated for reference persons and reference animals and plants respectively. Decisions needing to be made in relation to human exposure are guided by comparison with dose limits and constraints for planned exposures and with reference levels for existing and emergency exposure situations. Exposures to biota may be evaluated by comparison with Derived Consideration Reference Levels (DCRLs), taking into account the specific conditions of the exposure situations.

If the releases of radioactivity lead to exposures of the public that comply with the dose limits, it is very unlikely that there is any concern with regard to the exposure to biota at the same location. The situation is more complex if the exposure to biota needs to be evaluated for unpopulated areas. Even in such a case, the radiological impact to biota cannot be considered in isolation since on a planet with a population of nearly 7 billion people also the (unpopulated) environment is at least to some extent always part of the human habitat — or it could become so in the future — and therefore needs to be explored as a actual or potential source of exposures to humans.

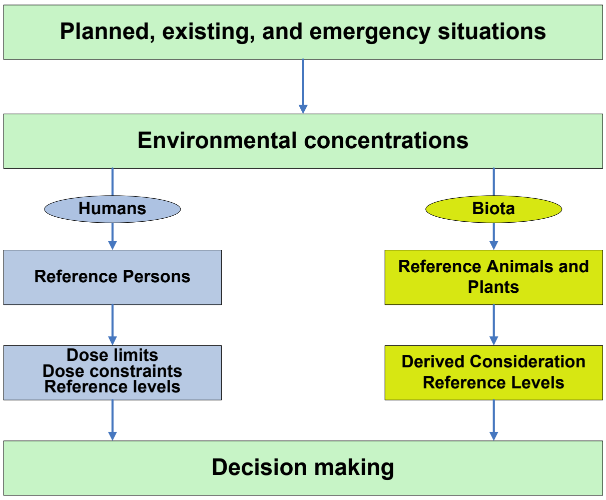

Fig. 3. Scheme for estimating and evaluating exposures to humans and biota (according to ICRP, 2008).

Discussion and conclusions

Radiological protection of the environment is a topic that has been intensively discussed in the last years. Much progress has been made on the development of methodologies to estimate exposures flora and fauna and to evaluate exposures to biota with regard to adverse effects induced by ionizing radiation. Derived Consideration Reference Levels derived by the ICRP for chronic exposure represent bands of dose that indicate the need for a detailed consideration of the exposures and its circumstances. For the most sensitive group of References Animals and Plants, this zone ranges from 4-40 µGy h^{-1}. This agrees very well a screening value of 10 µGy h^{-1} to be applied to identify exposure conditions of no concern for biota.

Doses to biota above a level of 10 µGy h^{-1} may hardly be achieved under planned exposure situations. If doses to humans and biota are assessed for the same habitat, it is difficult to imagine that conditions leading to such dose rates for biota could comply with the 1 mSv a^{-1} dose limit for public exposure.

The development and application of International Safety Standards to ensure protection of the public and environment is a key activity of the IAEA. Currently special emphasis is given to integrate issues related to environmental protection into the system for radiation protection of humans in a manner that is coherent with the ICRP Recommendations and the input of the IAEA Member States.

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Dose rates to freshwater biota in Finnish lakes

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Abstract

In recent years there has been growing international interest in the assessment of doses and risks from ionising contaminants to biota. Incremental dose rates to biota in freshwater ecosystems in Finland mainly resulting from exposure to the Chernobyl-derived radionuclides ^{137}Cs, ^{134}Cs and ^{90}Sr were estimated using the ERICA Assessment Tool developed within the EC 6th Framework Programme. Data sets consisting of measured activity concentrations in fish, aquatic plants, lake water and sediment for three selected lakes located in a region with high ^{137}Cs deposition were applied in the assessment. The selected lakes are among those having the highest activity concentrations found in Finland and therefore represent the highest exposure to biota in freshwater ecosystems affected by the Chernobyl fallout. The dose rates to most species studied were clearly below the screening level of 10 μGy h^{-1}, indicating no significant impact of the Chernobyl fallout on these species. However, the possibility of higher dose rates to certain species living on or in the bottom sediment cannot be excluded based on this assessment. In addition, dose rates from ^{210}Pb and ^{210}Po to selected organisms were calculated. Dose rates from these radionuclides were negligible in comparison with the screening level. In the ERICA Tool the parameter "occupancy factor", defining the fraction of time the studied organism spends in a given habitat, can have a considerable impact on the dose rate estimates. The values set for this parameter should be as realistic as possible with respect to the use of the habitat of the studied organisms.

Introduction

In recent years there has been growing international interest in the assessment of doses and risks from ionising contaminants to biota (Andersson et al. 2008; ICRP 2003, 2007, 2009; Larsson 2008). Several models are now available that enable the assessment of radiological risk to biota (Beresford et al. 2008a; Vives I Battle et al. 2007). In this study one of the models, the ERICA Assessment Tool (Brown et al. 2008), was applied to estimate dose rates to biota in freshwater ecosystems in Finland.

The ERICA Tool allows the estimation of dose rates for terrestrial, freshwater and marine ecosystems for a set of default reference organisms or, alternatively, user-defined organisms. The Tool includes databases on radionuclide transfer (Beresford et al. 2008b; Hosseini et al. 2008) and dose conversion coefficients (Ulanovsky et al.

2008) enabling dose calculation to be performed from input data on radionuclide concentrations in biota and/or environmental media such as soil or water.

The aim of this assessment was to estimate incremental dose rates to biota from the late 1980s to the 2000s, as affected by the Chernobyl fallout in 1986. In addition, dose rates from ^{210}Pb and ^{210}Po to selected organisms were calculated.

Material and methods

Study sites and the data

Selection and characterisation of the study sites have been described in Vetikko and Saxén (2010). Existing data sets consisting of measured activity concentrations of ^{137}Cs, ^{134}Cs and ^{90}Sr in fish, aquatic plants, lake water and sediment for three selected lakes located in a region with high ^{137}Cs deposition were applied in the assessment. The selected lakes Päijänne, Siikajärvi and Vehkajärvi are among those having the highest activity concentrations found in Finland and therefore represent the highest exposure to biota in freshwater ecosystems affected by the Chernobyl fallout. Sampling, sample treatment and analyses of the different sample types have been described in Saxén et al. (1996) and Saxén and Ilus (2008). Activity concentrations of ^{137}Cs, ^{134}Cs and ^{90}Sr in perch (*Perca fluviatilis*), pike (*Esox lucius*), bream (*Abramis brama*), lake water and sediment for Lake Päijänne are presented in Table 1. For Lakes Siikajärvi and Vehkajärvi, activity concentrations of ^{137}Cs in perch, pike, water horsetail (*Equisetum fluviatile*), water lily (*Nymphaea candida*), yellow water lily (*Nuphar lutea*), lake water and sediment are provided in Table 2.

The mean activity concentration of ^{210}Po in lake water collected in 2007 from several lakes was 0.0019 Bq kg^{-1} and that of ^{210}Pb 0.0032 Bq kg^{-1}. The corresponding activity concentrations in roach (*Rutilus rutilus*) sampled in 2005 from Lake Päijänne were 1.45 Bq kg^{-1} f.w. for ^{210}Po and 0.24 Bq kg^{-1} f.w. for ^{210}Pb.

Application of the ERICA Assessment Tool

The ERICA Assessment Tool, version 1.0 (April 2008, available from: http://www.project.facilia.se/erica/download.html) was used for estimating dose rates to biota. The ERICA Tool allows the input of site-specific measured activity concentrations in biota and/or environmental media at Tiers 2 and 3 of the three tiers in the model (Brown et al. 2008). In this assessment Tier 2 was used. Best estimate values of measured activity concentrations in biota, lake water and/or sediment were used as inputs at Tier 2, as recommended by Brown et al. (2008). For the geometries and dimensions of the biota, default reference organisms were used in the model, so that for the fish species the reference organism selected was "pelagic fish" and for aquatic plants it was "vascular plant".

In case of missing activity concentrations in sediment, the model uses concentrations in water and a distribution coefficient (Kd, L kg^{-1}) to calculate the concentrations in sediment needed for estimation of the dose rate. Default Kd values provided by the Tool were used in such cases.

The ERICA Tool contains a parameter "occupancy factor", which defines the fraction of time an organism spends in a given location in its habitat. For aquatic ecosystems, the possible locations are the water surface, water column, sediment surface and sediment. As a default in the model, the occupancy factor for vascular plants is entirely assigned to the sediment surface to maximise the dose rate (Oughton et al. 2008). For pelagic fish the default occupancy factor is 1 for the water, which means that fish only occupy the water column. In this study alternative occupancy factors were also used to provide lower and upper estimates of the dose rates.

The Tool performs dose rate calculations from the input data by applying dose conversion coefficients (µGy h^{-1} per Bq kg^{-1} fresh weight; see Ulanovsky et al. 2008) and weighting factors for various components of radiation (10 for alpha, 3 for low energy beta and 1 for beta, gamma as default). A default uncertainty factor of three at Tier 2 was used in the model to account for uncertainties in the assessment method (Oughton et al. 2008).

Table 1. Activity concentrations of ^{137}Cs, ^{134}Cs and ^{90}Sr in fish (Bq kg^{-1} fresh weight) and water (Bq l^{-1}) sampled in 1986, 1988, 1998 and in 2000 - 2006 (mean values), and in sediment (Bq kg^{-1} dry weight) sampled in 1997 from Lake Päijänne (Saxén et al. 1996). Activity concentrations in fish are for edible parts.

Sample	Sampling time	^{137}Cs	^{134}Cs	^{90}Sr
Perch	August 1986	1700	870	5.5
	December 1986	2200	1070	5.5
	1988	2250	625	5.5
	1998	346	-	-
	2000 - 2006	270	-	-
Pike	August 1986	970	500	0.27
	December 1986	1600	720	0.27
	1988	4100	1300	0.5
	1998	376	-	-
	2000 - 2006	250	-	-
Bream	August 1986	1400	660	5.0
	December 1986	1800	900	5.0
	1988	1080	310	5.0
	1998	212	-	-
	2000 - 2006	110	-	-
Water	August 1986	1.65	0.77	0.024
	December 1986	0.6	0.2	0.02
	1988	0.322	0.089	0.017
	1998	0.05	-	-
	2000 - 2006	0.025	-	-
Sediment	1997	18 530	-	-

At Tier 2 the results are given as total, internal and external weighted whole body absorbed dose rates (Brown et al. 2008). The total (internal and external summed) dose rates are compared directly to the selected screening dose rate to enable assessment of the risk to biota from ionising radiation (Brown et al. 2008). A default screening dose rate of 10 µGy h^{-1} provided by the Tool and suggested by Andersson et al. (2008), Beresford et al. (2007) and Garnier-Laplace et al. (2008) was used in the model.

Table 2. Activity concentrations of ^{137}Cs in fish and aquatic plants (Bq kg^{-1} fresh weight), in water (Bq l^{-1}) and in sediment (Bq kg^{-1} dry weight) sampled from Lakes Siikajärvi and Vehkajärvi (Saxén and Ilus 2008). Activity concentrations in fish are for edible parts.

Lake	Sample	Year of sampling	^{137}Cs
Siikajärvi	Perch	2003	4710
	Pike	2003	3610
	Water horsetail	2003	1720
	Water lily	2003	350
	Yellow water lily	2003	1100
	Water	2003	0.290
	Sediment	2003	14 942
Vehkajärvi	Perch	1998	1558
		2000 - 2004	2070
	Pike	1998	2300
		2000 - 2004	2134
	Water horsetail	2003	633
	Water lily	2003	248
	Yellow water lily	2003	264
	Water	1998	0.332
		2002 - 2003	0.304
	Sediment	2003	18 396

Results and discussion

Dose rates from ^{137}Cs, ^{134}Cs and ^{90}Sr to perch in Lake Päijänne varied from 0.05 to 0.6 µGy h^{-1} (Vetikko and Saxén 2010). For pike the dose rates were between 0.05 and 1 µGy h^{-1}(Vetikko and Saxén 2010). The maximal dose rate for perch was observed in December 1986, followed by a small decrease in 1988, and declined to one tenth in 1998 and in the 2000s (Figure 1). For pike the maximal dose rate was shown in 1988, followed by a decline to a level almost two orders of magnitude lower in 1998 and in the 2000s (Figure 1). This temporal variation in the dose rates was associated with the changes in the activity concentrations in fish and water from August 1986 to the 2000s. The missing contributions of ^{134}Cs and ^{90}Sr to the dose rates in 1998 and from 2000 to 2006 may also have influenced the lower dose rates then observed.

For bream in Lake Päijänne the dose rates varied from 0.1 µGy h^{-1} in the 2000s to 15 µGy h^{-1} in August 1986 (Figure 2). The result for bream was based on activity concentrations in water and on the default Kd parameter provided by the ERICA Tool, not on measured data for the sediment surface, as the only available data on activity concentrations in sediments of Lake Päijänne were from the year 1997. Therefore a more site-specific assessment for the year 1986 was prevented and the exceeding of the screening dose rate to bream could not be confirmed.

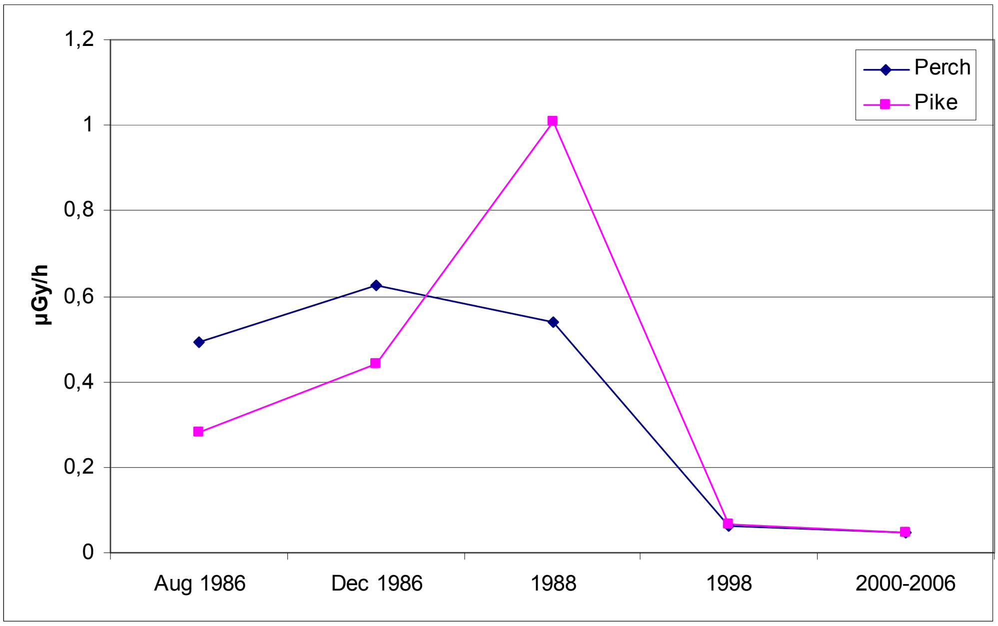

Fig. 1. Absorbed dose rate (µGy h^{-1}) from ^{137}Cs, ^{134}Cs and ^{90}Sr to perch and pike in Lake Päijänne (Assumption: Occupancy factor 1 for the water column was used for both fish species.).

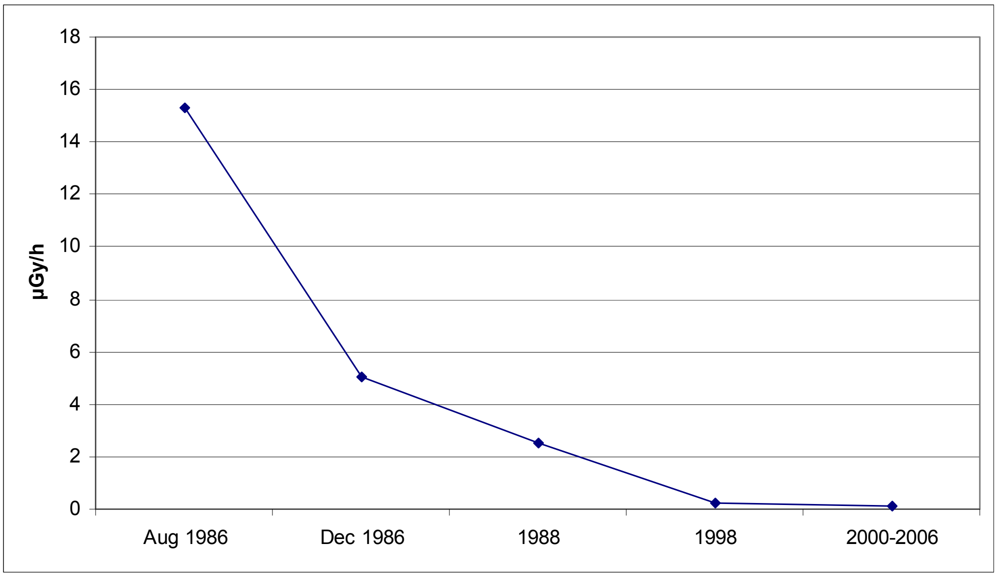

Fig. 2. Absorbed dose rate (µGy h^{-1}) from ^{137}Cs, ^{134}Cs and ^{90}Sr to bream in Lake Päijänne (Assumptions: Occupancy factor 0.8 for the water column and 0.2 for the sediment surface were used. Default values for the Kd parameter were used so that the Kd was 137 000 L kg^{-1} for Cs and 2000 L kg^{-1} for Sr.).

In the short term after the deposition, short-lived nuclides also caused a radiation dose to biota. However, this was mainly external exposure during a short period and therefore not significant for the incremental dose (Vetikko and Saxén 2010).

For perch and pike the absorbed total dose rate consisted almost entirely of the internal dose rate, whereas for bream the external dose rate was more pronounced due to exposure to radiation from the bottom sediment reflected by the parameter "occupancy factor" that for bream was assumed to have a component related to the surface sediment in addition to the water column (Table 3).

Table 3. External, internal and total dose rates (µGy h^{-1}) from ^{137}Cs, ^{134}Cs and ^{90}Sr to fish in Lake Päijänne in August and December 1986, calculated using Tier 2 of the ERICA Tool (Vetikko and Saxén 2010).

Sample	Sampling time	Occupancy factor	External dose rate	Internal dose rate	Total dose rate
Perch	August 1986	Water column = 1	1.09E-3	4.92E-1	4.93E-1
	December 1986	Water column = 1	3.32E-4	6.24E-1	6.24E-1
Pike	August 1986	Water column = 1	1.09E-3	2.80E-1	2.81E-1
	December 1986	Water column = 1	3.32E-4	4.39E-1	4.40E-1
Bream	August 1986	Water column = 0.8 Sediment surface = 0.2	1.49E1	3.94E-1	1.53E1
	December 1986	Water column = 0.8 Sediment surface = 0.2	4.54E0	5.16E-1	5.06E0

^{137}Cs accounted for 62-75% of the total dose rate to perch and pike, while 24-38% was due to ^{134}Cs (Vetikko and Saxén 2010). However, for bream ^{137}Cs accounted only for 45% in August 1986 when the contribution of ^{134}Cs to the dose was 55%. Later ^{137}Cs exceeded ^{134}Cs as a contributor to the dose. The contribution of ^{90}Sr to the dose rate of all fish species was only 0.02-0.7% (Vetikko and Saxén 2010).

Dose rate from ^{210}Po and ^{210}Pb to roach in Lake Päijänne was 0.045 µGy h^{-1} (Assumptions: Occupancy factor 1 for the water column was used.).

For Lakes Siikajärvi and Vehkajärvi, two sets of occupancy factors for fish and aquatic plants were used to provide lower and upper estimates of the dose rates (Vetikko and Saxén 2010). Dose rate from ^{137}Cs to perch in Lake Siikajärvi varied from 0.8 to 1.3 µGy h^{-1}, depending on the occupancy factor (Figure 3). For pike the dose rates were between 0.7 and 1.1 µGy h^{-1} (Figure 3). The corresponding values for perch of Lake Vehkajärvi were 0.3 and 1.6 µGy h^{-1} and those for pike 0.4 and 1.7 µGy h^{-1} (Figure 4). These dose rates were slightly higher than those for Lake Päijänne due to higher activity concentrations in fishes and water in Lakes Siikajärvi and Vehkajärvi.

The absorbed dose rate also to aquatic plants varied according to the values set for the occupancy factor (Vetikko and Saxén 2010). It should be noted that the lower values of the dose rates are based on the assumption that the whole plant is located in water, therefore excluding the exposure of roots in the sediment and resulting in unrealistically low dose rates (Figure 3, 4). The upper values of the dose rates, on the other hand, represent the maximum, whereas more realistic exposure of plants will be obtained between the two values.

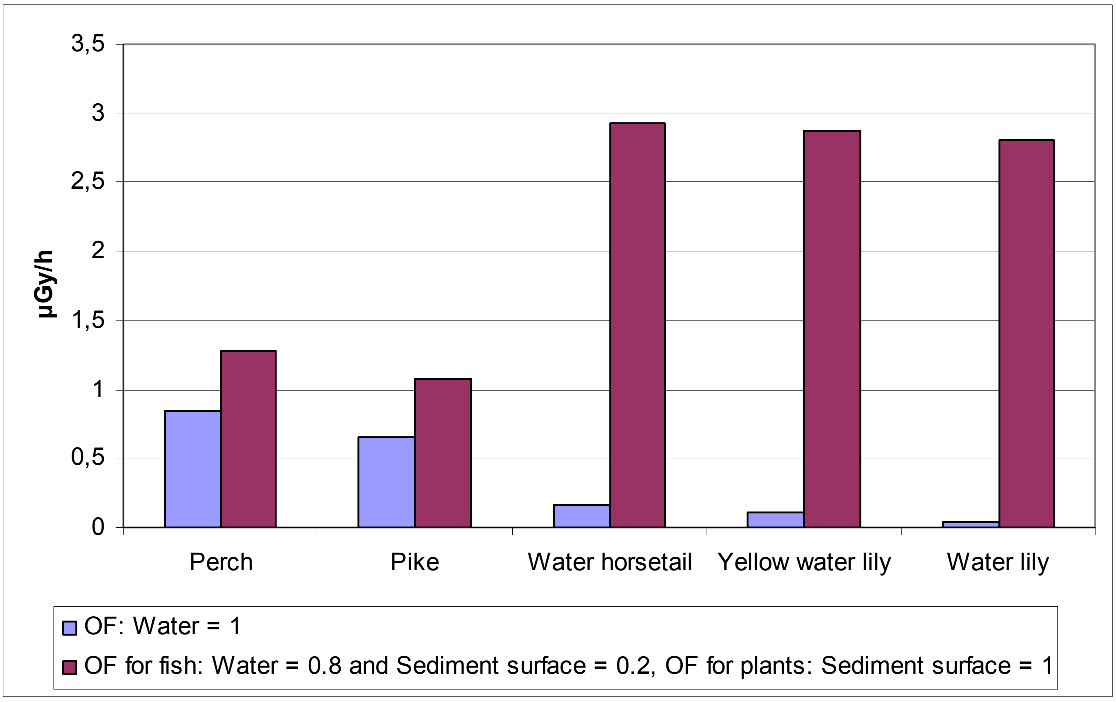

Fig. 3. Absorbed dose rate (µGy h^{-1}) from ^{137}Cs to fish and plant species in Lake Siikajärvi in 2003 (OF = occupancy factor).

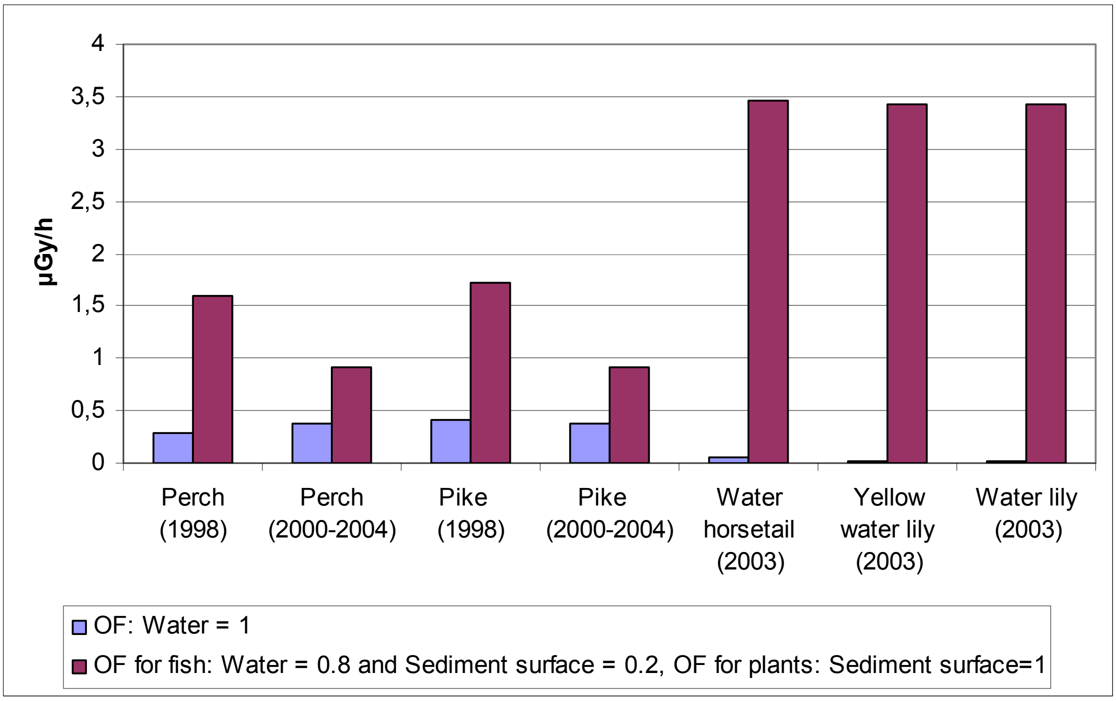

Fig. 4. Absorbed dose rate (µGy h^{-1}) from ^{137}Cs to fish and plant species in Lake Vehkajärvi (year of sampling given in brackets; OF = occupancy factor).

The roots of aquatic plants can receive higher dose rates via external exposure to contaminated sediment compared to the other plant parts located in the water column or on the water surface. For example, Nedveckaite et al. (2007) reported a ten times higher dose rate to the roots of submerged hydrophytes (0.044 μGy h^{-1}) compared to the above-sediment plant parts (0.0044 μGy h^{-1}) from ^{54}Mn, ^{60}Co, ^{137}Cs and ^{90}Sr in the cooling pond of Ignalina nuclear power plant in Lithuania.

Conclusions

The dose rates from ^{137}Cs, ^{134}Cs and ^{90}Sr to most species studied were clearly below the screening level of 10 μGy h^{-1}, indicating no significant impact of the Chernobyl fallout on these species. However, the possibility of higher dose rates to certain species living on or in the bottom sediment cannot be excluded based on this assessment.

The screening dose rate was exceeded for bream in Lake Päijänne in August 1986 due to external exposure to radiation from the sediment surface. However, the dose rate to bream did not differ from the levels of natural background radiation generally reported in the literature (see Zinger et al. 2007). This leads to the assumption that the likely risk to bream from ionising radiation in August 1986 was also minimal.

In the ERICA Tool the values set for the occupancy factor can have a considerable impact on the dose rate. The occupancy factors should be as realistic as possible with respect to the use of the habitat of the studied organisms.

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Effects of radioactive contamination on plant populations and radiation protection of biota

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Abstract
An assessment of the state of plant and animal populations inhabiting polluted territories and the analysis of mechanisms of their adaptation to adverse environmental conditions undoubtedly have general biological importance. Consequently, studies that examine biological effects on non-human biota in natural settings provide a unique opportunity for obtaining information about the potential biological hazard associated with radioactive contamination. Nevertheless, up to now there is a distinct lack of quantitative data on the real long-term biological consequences of chronic radiation exposure lasting a long period of time. Actually, few studies exist that are directly relevant to understanding the responses of plant and animal populations to radioactive substances in their natural environment. The results of long-term field experiments in the Bryansk Region affected by the Chernobyl accident and in the Semipalatinsk Test Site, Kazakhstan that have been carried out on different species of plants are discussed. Although radionuclides cause primary damage at the molecular level, there are emergent effects at the level of populations, non-predictable solely from knowledge of elementary mechanisms of the pollutants' influence. Plant populations growing in areas with relatively low levels of pollution are characterized by the increased level of both cytogenetic disturbances and genetic diversity. Radioactive contamination of the plants environment activates genetic mechanisms, changing a population's resistance to exposure. However, in different radioecological situations, genetic adaptation to extreme edaphic conditions in plant populations could be achieved with different rates.

Introduction
The majority of abiotic stress studies performed under controlled laboratory conditions does not reflect the actual situations that occur in the field. Therefore, to understand effects of contaminant exposure properly we must pay attention to what is actually going on in natural conditions. Field studies are particularly useful for assessing long-term biological effects induced by chronic low dose-rate and multi-pollutant exposure at contaminated sites. Although radionuclides and heavy metals cause primary damage at the molecular level, there are emergent effects at the level of populations that are not predictable solely from knowledge of elementary mechanisms of the pollutants'

influence. Up to now we have known little about responses of plant and animal populations to environmental pollutants in their natural environments. These data gaps imply that the protection of the environment from ionizing radiation will require more experimental data related to effects of chronic low-level exposure to radionuclides at the population level. Previously completed and ongoing field studies that have been carried out in Laboratory of Plant Ecotoxicology, RIARAE in different species of wild and agricultural plants are briefly summarized in Table 1. A wide range of radioecological situations and climatic zones have been covered in frames of this work. To illustrate the main findings, two field studies are discussed here in more detail.

Table 1. Field studies on wild and agricultural plants.

Species	Site & Time	Assay and/or endpoints
Winter rye and wheat, spring barley and oats	10-km ChNPP zone (12-454 MBq/m^2), Ukraine, 1986-1989	Morphological indices of seeds viability, cytogenetic disturbances in intercalary seedling root meristem (Geras'kin et al. 2003a)
Scots pine, couch-grass	30-km ChNPP zone (2.5-27 μGy/h), Ukraine, 1995	Cytogenetic disturbances in seedling root meristem (Geras'kin et al. 2003b)
Scots pine	Radioactive waste storage facility, Leningrad Region, Russia, 1997-2002	Cytogenetic disturbances in needle and seedling root meristems (Geras'kin et al. 2005)
Scots pine	Bryansk Region radioactively contaminated in the Chernobyl accident (0.6-3.5 μGy/h), Russia, 2003-2009	Cytogenetic disturbances in seedling root meristem, enzymatic loci polymorphism, abortive seeds (Geras'kin et al. 2008; Geras'kin et al. 2009a)
Crested hairgrass	Semipalatinsk Test Site (0.7-36 μGy/h), Kazakhstan, 2005-2008	Cytogenetic disturbances in coleoptiles of germinated seeds (Geras'kin et al. 2009b)

Material and methods

In 2005-2008 seeds of crested hairgrass (*Koeleria gracilis* Pers.) were collected from six sites of the Semipalatinsk Test Site (Kazakhstan). Radiation background at the sites and specific activity of the main dose-forming radionuclides in soil samples were measured. Doses to generative organs of crested hairgrass were calculated. Squashed slides for cytogenetic analysis were prepared of coleoptiles (2-5 mm in length) of germinated seeds. In every slide, all ana-telophase cells (4800-11900 ana-telophases in 30-90 slides) were scored to calculate frequency of aberrant cells. Detailed description of methods used is given in (Geras'kin et al. 2009b).

To study biological effects in Scots pine (*Pinus sylvestris* L.) populations experiencing chronic radiation exposure, six test sites were chosen in the Bryansk region radioactively contaminated as a result of the Chernobyl accident. Pine cones were collected in autumns of 2003-2007. Specific activities of radionuclides in soil samples were measured, and doses to the pine trees' generative organs were estimated. Aberrant cells were scored in root meristem of germinated seeds in ana-telophases of the first mitosis. The method of isozymic analysis of megagametophytes was used for an estimation of genetic variability in populations of Scots pine. Five enzymatic loci (GDH, LAP, MDH, DIA, and 6-PGD) were studied in endosperms of the seeds collected in 2005. Detailed description of materials and methods used can be found in (Geras'kin et al. 2008; Geras'kin et al. 2009a).

Results and discussion

In the Semipalatinsk Test Site (STS), 116 atmospheric and ground-surface explosions for nuclear and hydrogen bomb testing were carried out between 1949 and 1963. A study of crested hairgrass (*Koeleria gracilis* Pers.) populations, a typical wild crop for Kazakhstan, showed that the frequency of cytogenetic disturbances in coleoptiles of germinated seeds increases proportionally to the dose absorbed by plants (Fig. 1). The agreement between findings from four years of study (2005-2008), different in weather conditions, suggests the leading role of radioactive contamination in an occurrence of cytogenetic effects. Severe disturbances of single and double bridges as well as laggard chromosomes contribute mainly to the observed cytogenetic effect (Geras'kin et al. 2009b).

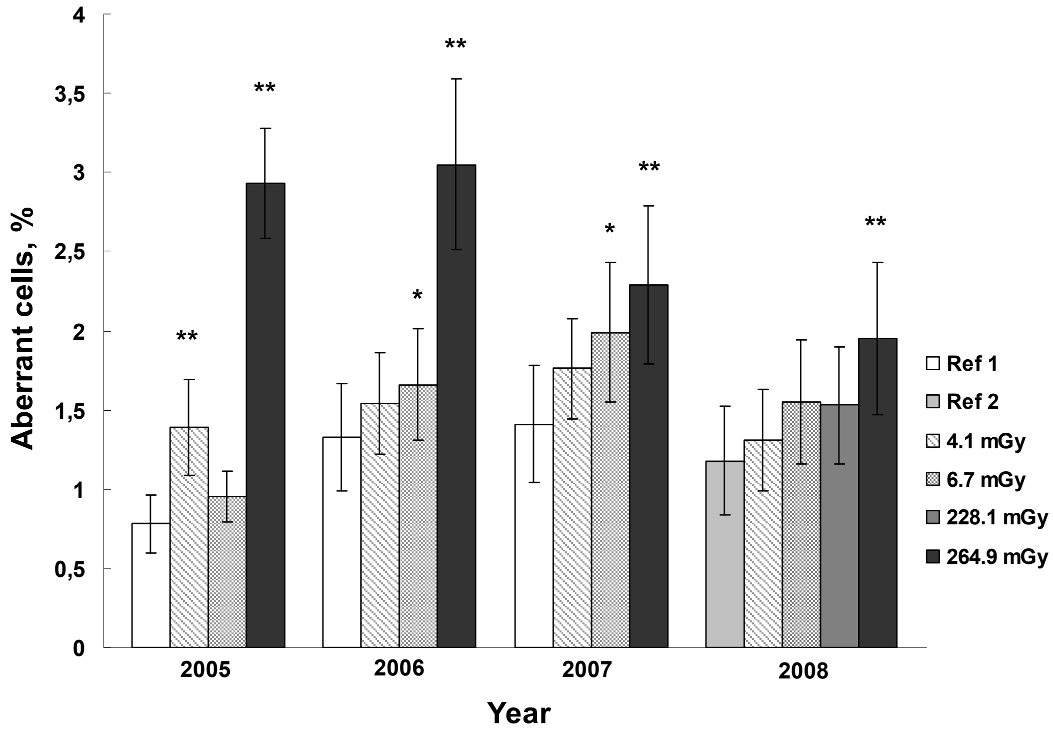

Fig. 1. Frequency of aberrant cells in coleoptiles of germinated seeds of crested hairgrass collected in the Semipalatinsk Test Site, Kazakhstan in 2005-2008 in dependence on annual dose absorbed. Ref1 and Ref2 are the reference sites in 2005-2007 and 2008, respectively. Significant difference from the corresponding reference site: * - p < 0.10; ** - p < 0.05.

Dose rate in the epicentre of nuclear tests amounts to 36 µGy/h, which is more than 3 fold of the predicted no-effect dose rate of 10 µGy/h derived in the EC ERICA project (Andersson et al. 2009). It is, however, well below the threshold for statistically significant effects (100 µGy/h) derived at the FASSET Radiation Effects Database analysis (Real et al. 2004). It is not surprising, than, that in the STS study there are found significant cytogenetic effects in crested hairgrass populations but no morphological alterations. Thus, the finding obtained are in agreement with the benchmark values proposed in the FASSET and ERICA projects to restrict radiation impact on biota.

Forest trees have gained much attention in recent years as nonclassical model eukaryotes for population, evolutionary and ecological studies (Gonzalez-Martinez et al. 2006). Because of their potential to affect many other species, any responses to selection pressures that are exerted on such keystone species as forest trees are especially important to quantify. The low domestication, large open-pollinated native populations and high sensitivity to environmental exposure make conifers almost an ideal species for the study of environmental effects of radioactive contamination.

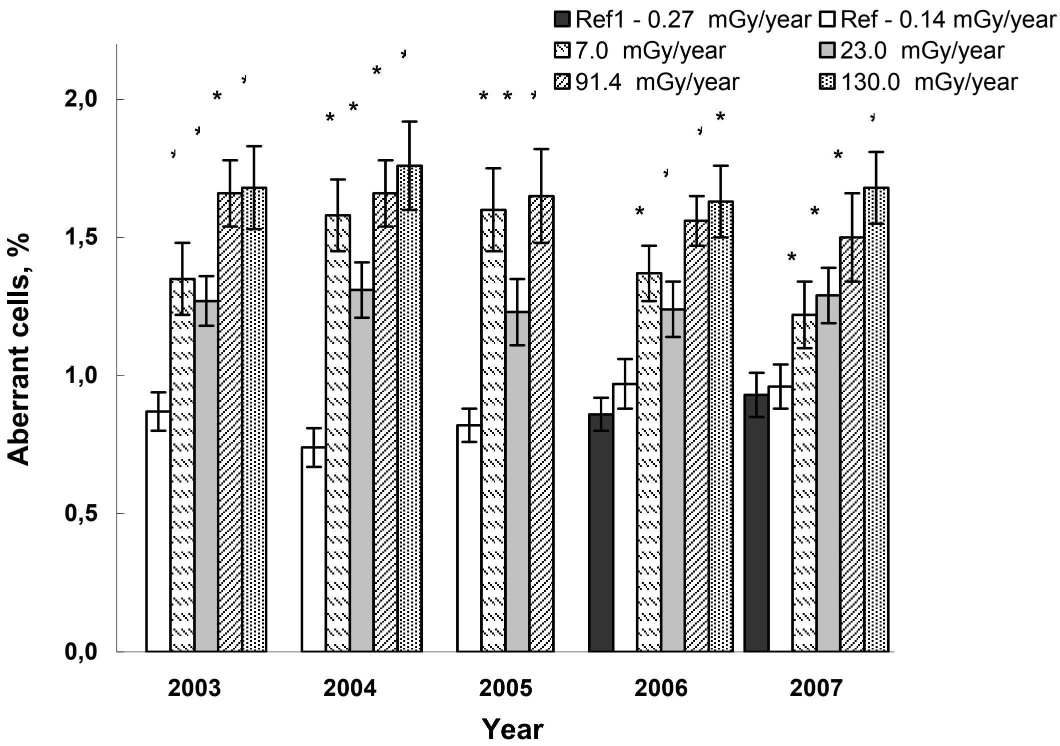

Fig. 2. Aberrant cells in root meristem of germinated seeds from Scots pine populations collected in the Bryansk Region of Russia, 2003-2007. Ref and Ref1 are the reference populations. * - significant difference from the reference populations, p < 0.05.

In Fig. 2, the 5-year results of long-term (2003-2009) study of cytogenetic effects in Scots pine populations growing in the Bryansk region radioactively contaminated as a result of the Chernobyl accident are presented. Populations under investigation have not shown any significant difference between years, so our results are robust and replicable over time. Aberrant cell frequency in root meristem of germinated seeds collected from these populations significantly exceeds reference level and shows statistically significant correlation to specific activity of ^{137}Cs, the main dose-forming radionuclide, in pine cones during all five years of study. Although there is a tendency for aberrant cells occurrence to increase with dose absorbed by the pine trees' generative organs, it is not always significant. Compiled with data from other our studies (Geras'kin et al. 2003a; Geras'kin et al. 2003b; Geras'kin et al. 2005), these findings indicate that an increased level of cytogenetic disturbances is a typical phenomenon for plant populations growing in areas with relatively low levels of pollution.

Absorbed doses in generative organs of pine trees are assessed with a dosimetric model (Fig. 2). In 22 years after the ChNPP accident, the annual doses are about thirty times below dose rate of 10 mGy/day proposed by IAEA (International… 1992) as safe for terrestrial plants. On the other hand, dose rates for two most contaminated sites are exceed the ERICA generic predicted no-effect value of 10 μGy/h (Andersson et al. 2009). From this comparison we can suppose that radiation exposure at the study sites is strong enough to induce cytogenetic but not morphologic disturbances in the exposed populations. Indeed, aberrant cell frequency in root meristem of germinated seeds collected from experimental populations significantly exceeded reference level during all five years of study (Fig. 2). It should be noted that, in the STS study, a wide range of doses from 4 to 265 mGy absorbed by the plants was studied, and dependence of cytogenetic effects on dose was revealed. On the contrary, in the Bryansk region, the range of doses absorbed by the pine trees at the study sites is much narrower; this could be the reason for an absence of statistically significant increase of biological effect with the dose absorbed in some years of observations.

It is becoming increasingly clear that cytogenetic disturbances detected in our studies might only be tip of an iceberg, reflecting global structural and functional rearrangements induced by radiation in exposed populations. An increase in mutation rate can affect the population genetic structure by producing new alleles or genotypes, and thereby has ecologically relevant effect. Alterations in the genetic make-up of populations are of primary concern because somatic changes, even if they lead to a loss of some individuals, will not be critical in populations with a large reproductive surplus. To analyze whether an exposure to radionuclides causes changes in population genetic structure, we evaluated frequencies of three different types of mutations (null allele, duplication and changing in electrophoretic mobility) of enzymatic loci in endosperm and embryos of pine trees from the studied populations. It is found that chronic radiation exposure results in the significant increase of total occurrence of enzymatic loci mutations. In particular, frequencies of mutations for loss of enzymatic activity increase with a dose absorbed by generative organs of pine trees (Fig. 3).

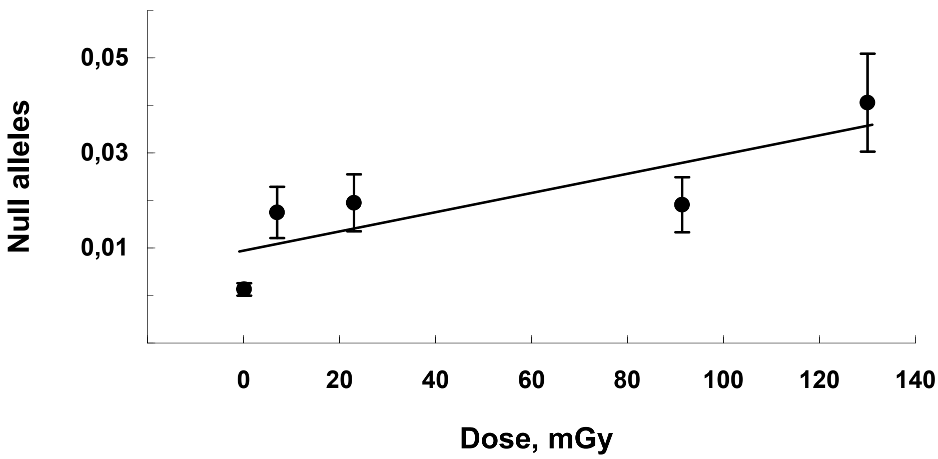

Fig. 3. Frequency of null alleles in enzymatic loci of endosperms (2005) in dependence on annual dose absorbed by generative organs of pine trees.

There are plenty of theoretical interpretations of evolution, but what is important is to see what happens in practice. Mutations in plant or animals are not necessary bad events when they do not adversely affect the population fitness. Mutation is one of the mechanisms that maintains genetic variation within a natural population and thus enables that population to cope with an adversely changing environment. Indeed, phenotypic variability in the exposed pine tree populations, estimated via Zhivotovsky index (Zhivotovsky 1980), significantly exceeds the reference level and increases with dose absorbed by generative organs of pine trees (Fig. 4).

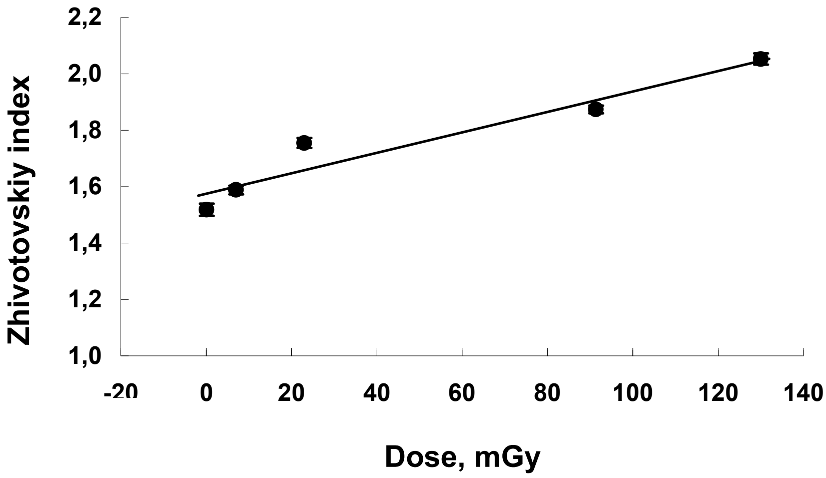

Fig. 4. Phenotypic variability estimated via the Zhivotovsky index in dependence on annual dose absorbed by generative organs of pine trees.

A decrease in heterozygosity within individuals has been associated (Theodorakis 2001) with decreased resistance to diseases, decreased growth rates, and decreased fertility. This would suggest that variations in individual heterozygosity may affect population growth and recruitment. The observed heterozygosity in pine tree populations at the radioactively contaminated sites is essentially higher then the expected one and increases with dose absorbed by generative organs of pine trees (Fig. 5).

From the data presented we can conclude that the relationship between radioactive contamination and genetic variability provides evidence of adaptation which optimizes the physiological response of a population to environmental changes. Keeping in mind all the data mentioned, it could be concluded that a high level of mutation occurrence is intrinsic for descendants of pine trees in the investigated populations, and genetic diversity in the populations is essentially conditioned by radiation exposure. So, in spite of their low values, dose rates observed can be considered as a factor able to modify genetic structure of population. Furthermore, an increased genetic diversity within the population of keystone species is likely to be positively correlated with increased species diversity of the depended community (Whitham et al. 2006).

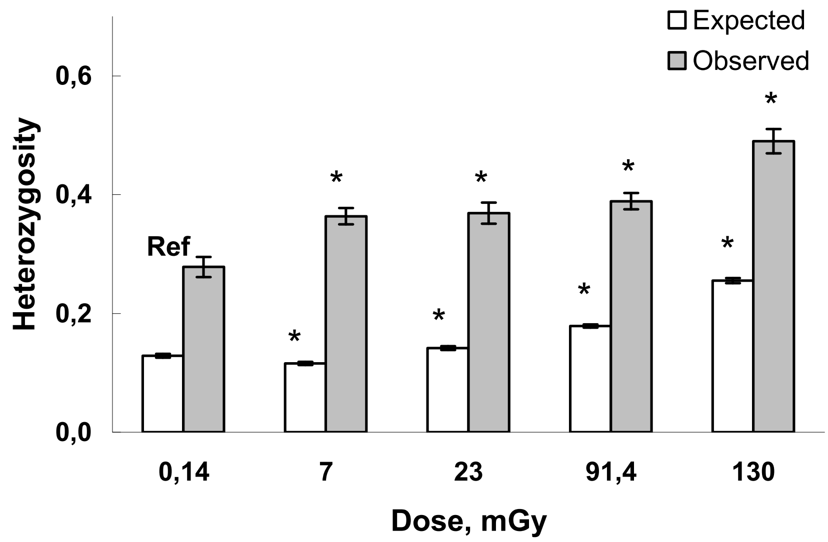

Fig. 5. Heterozygosity in endosperms of Scots pines in dependence on annual dose absorbed by generative organs of pine trees. Significant difference from the Ref population: * - p < 0.01.

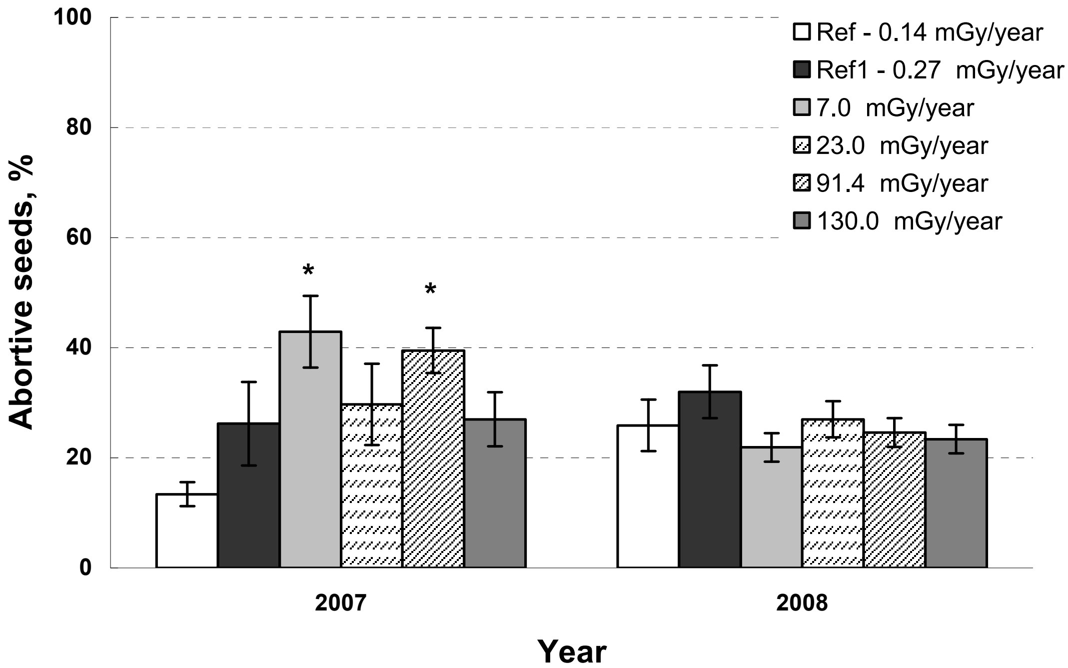

Fig. 6. Portion of abortive seeds in chronically exposed Scots pine populations. * - difference from Ref is significant, p < 0.05.

Although great progress has been made in understanding the nature of mutations, too little is yet known about the way in which mutations can lead to effects at the level of an organism and population. The effect of severe conditions on an organism is often considered to eliminate individuals. However, the alternative effect is to change the number of offspring produced by individuals without killing them. The plasticity of plants, and the fact that their reproductive organs are usually the terminal points of a branching structure, means that they tend to respond to environmental stresses by variation in reproductive rate without death. It is true that a much larger number of seeds are produced than developed into adult plants, and that the changes in frequency of the different genotypes occur due to a greater death of some genotypes than others. This is a form of response to selection, and a very powerful one (Valladares et al. 2007). From the results gathered a question arises: what could be an effect of high mutation rates revealed in our study on a reproductive potential of pine trees? From Fig. 6 it is clear that chronic exposure within the range of dose rates studied has virtually no effect on reproductive ability of pine trees from the exposed populations.

Conclusions

A basic level of concern within a newly developing system for radiological protection of the environment is a population. Of special importance in this context are studies on plant and animal populations inhabiting sites with contrasting levels and spectra of radioactive contamination. Special attention should be paid to population-level effects such as radioadaptation, changes in sexual, age and genetic structure of populations, since knowledge of elementary mechanisms of the radionuclides' impact is insufficient to predict them. Corresponding studies are likely to increase in importance as the rate at which we change the environment worldwide continues to accelerate. The findings presented here clearly indicate that plant populations growing in areas with relatively low levels of pollution are characterized by an increased level of both cytogenetic disturbances and genetic diversity. Concordant responses between changes in population genetic structure and elevated levels of cytogenetic damage provide evidence that the population genetic changes are influenced by exposure to radionuclides. Effects of contaminants on genetic diversity within a population are important because the level of genetic diversity affects a population's ability to adapt or the rate of adaptation to changes in the environment. Therefore, the amount of genetic variation within a population can influence its relative susceptibility to extinction. Finally, in spite of the wealth of information collected so far, much more still remains to be explained in order to fully understand the basis of plant populations' adaptations to a harmful environment.

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Unusual damaging effects of low radiation: Model experiments with protozoa and invertebrates

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Abstract

The importance of peculiar disturbing effect of low radiation, observed previously by Bychkovskaya et al in protozoa and in tissues of rats (http://irbb.ucoz.ru) was studied in the model experiments on aquatic invertebrates water flea *Daphnia magna*. During these experiments we were able to show that the probability of animals' death increases even at a dose of 0.1 Gy which is a thousand times lower than predicted as lethal dose (100 Gy). The process was registered for the greater part of daphnia's lifetime. The frequency of death of irradiated animals did not increase with dose increasing from 0.1 to 20 Gy.

The negative effect of low radiation (0.01 Gy) we also observed on the aquatic unicellular ciliates *Spirostomum ambiguum*. Modal experiments of the laboratory population of *Spirostomum* showed that these protozoa are highly sensitive to gamma-irradiation by spontaneous motor activity (SMA) criterion. We observed the reliable effect of SMA decrease right after the irradiation in doses 1 sGy. It was approximately on the similar level (about 40% below control) in a broad variety of doses. These changes were mass, did not dependent on dose in a wide range and were transmitted to the descendants of irradiated cells. Changes had been inherited not only through the vegetative reproduction of organisms, but also through the sexual reproduction (water flea daphnids). Due to the fact that the cellular dose-independent effects induced by low doses were found on the different organisms (including rats), we assumed that these effects might be widespread in nature. Molecular-genetic mechanism of these changes still remains unknown; mass character and non-linear nature of these changes can not be attributed to the mutations. Generally, they can be classified as epigenetic changes which attract more and more attention.

This work is directed to the ecological problem. We have witnessed and detected the decline of the viability and functional activity of aquatic organisms in the interval of doses that are lower than considerably dangerous ones.

Introduction

In the plural studies of the subjects of various species (amoebas, ciliates, cells of different low regenerating tissue of rats) we found (Bychkovskya 1986; Bychkovskya et al 2006) the peculiar negative cellular effects of low radiation which do not fit the general concept of stochastic nature of radiation. One of the typical manifestations of these effects was the high prolonged increase of the possibility of damage and cells death compared with the control level. Unlike the widely studied genotoxic changes, these effects: i) are mass; ii) are already registered at the doses which are very low for the subjects; iii) do not increase with further increasing of radiation dose; iv) inherited; v) may be caused by exposure to the nucleus as well as the cytoplasm of cells (experiments on amoeba).

In this paper, the possibility of this unusual form of damage was confirmed in experiments conducted at new subjects – unicellular and multicellular aquatic organisms. Let's consider these materials.

Material and methods

Ciliates *Spirostomum ambiguum* Ehrbg were cultured in laboratory tubes with separated dechlarinated water at a temperature of 20±2^{0}C. The ciliates were fed nutritional yeast once a week. It is known that ciliates refer to highly radio resistant organisms. $LD_{50/30}$ is around 1000 and more of Gray (Choppin et al 1995). Spirostomum were irradiated in masscult in tubes with 6 ml of water for "Gamma-cell" radiator (Canada, ^{60}Co; 0.15 Gy/min) at doses 0.01, 0.1 and 1 Gy and "Issledovatel" radiator (Russia, ^{60}Co; 36 Gy/min) at dose 20 and 50 Gy at 28±2^{0}C as an additional negative factor. The control groups were in the same conditions as experimental, but without irradiation. The effect of irradiation was estimated by reduce the ciliates' motor activity in the experimental groups. These changes were measured immediately just after irradiation and at 2, 4 and 7 days after that. To do this, the ciliates from irradiated groups and from the control groups were placed individually in special plastic plate with holes of 5 mm in diameter and 1-2 mm in depth. Motor activity of each ciliate was observed under the microscope. At the eyepiece of the microscope were did two lines crossing each other perpendicularly. A quantitative measure of each ciliate's motor activity was the number of intersections of the lines for 1 min. It was held for 5 series of observations at each dose with coding of the samples. The effect was manifested not only in reducing the SMA of irradiated ciliates, but also in the appearance of essential disorders of motion as a function of reaction – "death prognosticators" – convulsive twitching of the body, changes in moving patterns from the normal lineal to abnormal twirling and rolling, backward movements or total body immobility. Additionally, we took in the account the number of individuals (organisms) with motion pathologies. The frequency of pathological movement forms for ciliate was measured in % of the total investigated ciliates.

In the experiments we also used aquatic invertebrates' water flea *Daphnia magna*. They were born from females of the same age and irradiated on the first day after birth. They were irradiated for 5 individuals in biological tubes with 6 ml of water for "Gamma-cell" radiator at doses 0.1 and 1 Gy and "Issledovatel" radiator at dose 20, 100, 250 and 600 Gy. Based on literature data, radiation doses around 10^2 Gy are lethal for the class Crustacea (Choppin et al 1995). Control animals were in the same conditions as experimental, but were not irradiated. Then, the control and irradiated groups were cultured in the laboratory glasses for 5 individuals per 100 ml of separated dechlarinated water at a temperature about 20±2^{0}C. Daphnia were fed algae suspension once a day at a rate of 0.1 ml (density of 600–1000 million cells per ml) per 100 ml of water. Water in the experimental and the control glasses was refreshed every week; daphnia were transferred using a plastic pipette. In these conditions daphnids' lifetime were not much longer than 3 months.

Animals' death had been checked every day for 30 days. We presented the combined data collected in the 4th (embryonic period), 14th and 30th (reproductive period) days after the irradiation. We estimated the number of surviving daphnids as % of their original number or % from control groups for the corresponding period of observations. It was held for 4 series of observations at each dose.

The reliability assessment of the experimental data was carried out by the nonparametric Fisher test.

Results and discussion

Experiments with ciliates

Consider Table 1, which demonstrated the data of ciliates' SMA in control and at various periods after γ-irradiation at doses of 0.01–50 Gy (absolute value).

Table 1. Values ciliates' SMA (in absolute value) in control and at doses of 0.01–50 Gy at various periods after γ-irradiation.

Dose, Gy \ Days	0	2	4	7
	[1]*M±m*			
0 (Control)	2.1±0.2	1.8±0.1	1.8±0.1	1.9±0.1
0.01	1.4±0.2*	1.2±0.1*	1.2±0.3*	1.3±0.1*
0.1	1.4±0.1*	1.1±0.2*	1.0±0.2*	0.8±0.2*
1	1.5±0.2*	0.9±0.2*	1.0±0.1*	1.1±0.3*
20	1.1±0.3*	1.0±0.1*	1.1±0.1*	0.9±0.1*
50	1.5±0.1*	1.0±0.1*	1.0±0.1*	0.9±0.3*

[1] In the control, in each dose and at each period studied of 60 ciliates.
*** p≤0.05**

The effect is caused by exposure to a dose of 0.01 Gy and has not run-up even with 5 thousand times dose increase. Table clearly demonstrates the detected non-linear effects of radiation and underlines the effectiveness of low doses. Figure 1 presents the data for variations in the experimental ciliates' SMA in percentage of the corresponding control with standard deviation for the percentages at various periods after irradiation.

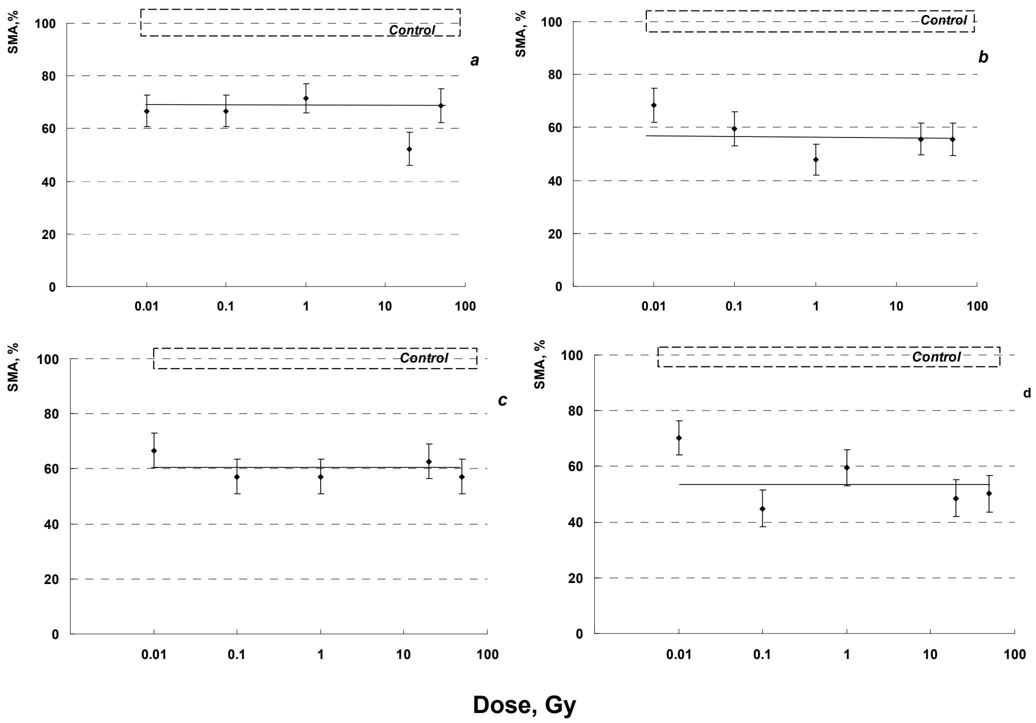

Fig. 1. The changing in ciliates' SMA (in % relative to controls) after γ-irradiation at dose of 0.01-50 Gy immediately (a) and in 2 (b), 4 (c) and 7 (d) days.

Figure 1 shows that the significant decrease ($p<0.05$) in SMA is approximately the same at all times (immediately and in distant periods after exposure) and at all doses. This effect already taken place by exposure at 0.01 Gy. The effect was recorded on the 7th days after irradiation. During that time the ciliates replaced about 3-4 generations (average length of the cell cycle was about 2 days). It follows that the effect can transmitted to descendants of irradiated cells.

The same change pattern was identified while we calculated the frequency of occurred pathological forms. Figure 2 shows these data. If in control groups pathology occurs very rarely or never recorded; but in the experiment it is expressed at each dose. It is seen that increasing the dose does not influence the degree of damage.

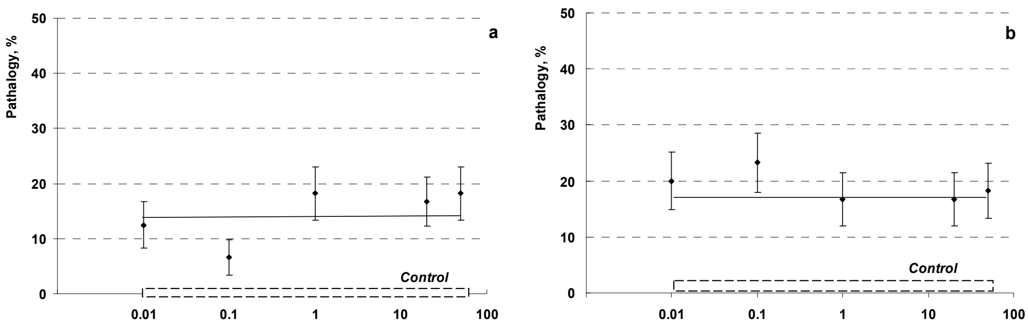

Fig. 2. The pathological forms of ciliates' movement (in %) after γ-irradiation at doses of 0.01, 0,1, 1, 20 and 50 Gy immediately (a) and in 7 (b) days.
In controls, in each dose and at each period studied for 50 ciliates.

This data shows that changes of motion function occur after low radiation and are transmitted to descendants of irradiated cells in vegetative reproduction of protozoa. We observed noticeable deviations from control group in all doses. It shows the massive nature of the changes of the described type.

Experiments with daphnids

Consider figure 3 which shows the survival of daphnia at different times in the control and after irradiation at doses from 0.1 to 600 Gy, expressed as % of the original number of animals (60 animals for each curve).

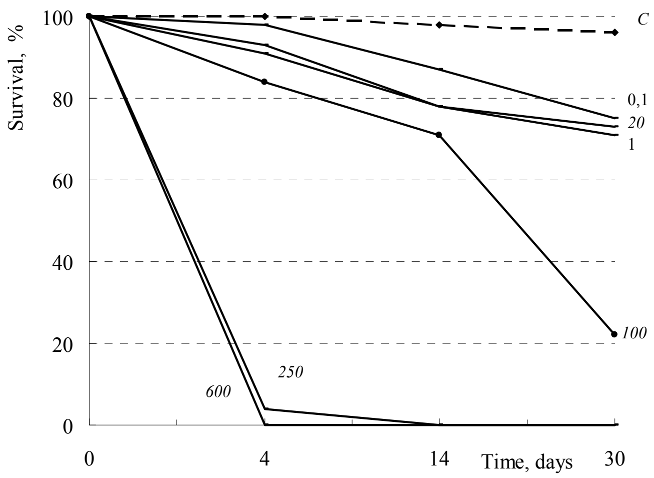

Fig. 3. Survival of daphnids during 30 days in the control (*C*) and after exposure at doses of 0.1, 1, 20, 100, 250 and 600 Gy in % of their original number (60).

There are 4 categories of curves which respectively show the evidence of the daphnids viability in the control (*C*) and after irradiation at doses from 0.1 to 20 Gy, at 100 Gy ($LD_{80/30}$) and at absolutely lethal doses – 250 and 600 Gy ($LD_{100/16}$ and $LD_{100/4}$). Fig.3 shows that even at the lowest of doses – 0.1 Gy, as well as low for subject doses of 1 and 20 Gy there is a marked deviation from control. The effect does not increase with increasing dose in this dose range and by the end of the period of observation at all doses the effect is about 20%. At the same time, unlike the actions in higher doses, there is a constant extension in the time of the death of individuals, rather than the death of the population in the relatively early time.

In the follow up experiments we studied the changes of daphnids' survival on the 30th day because the maximum differences from the control after the irradiation in relatively low doses were achieved at that time. In these experiments we discovered the possibility of inheritance of the studied forms of damage in the descendants of irradiated animal. Figure 4a shows the evaluation of survival of daphnids-parents at 30th day after the irradiation at doses of 0.1, 1 and 20 Gy.

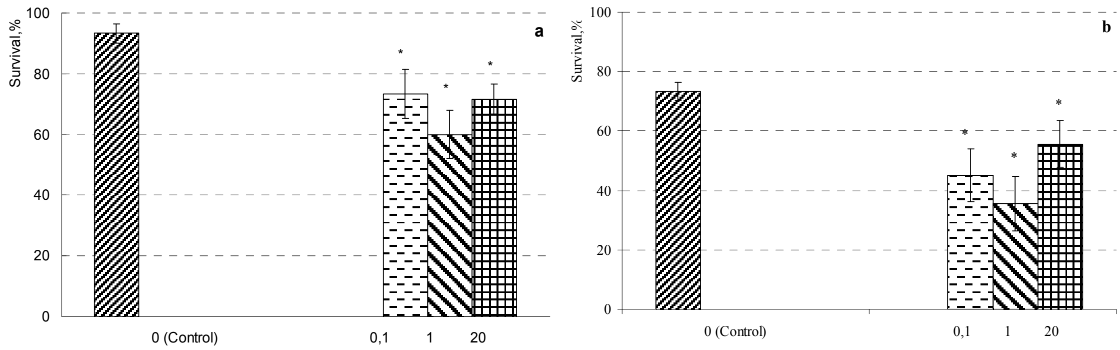

Fig. 4. The survival of daphnids-parents (a) and daphnids-descendants F_1 (b), % of the original number of the 30th day of observation with a standard error at p <0,05.
In experiments (a) up to 60 and in experiments (b) up to 45 animals in each dose and in the control.

In these experiments, we see almost the same results as in the first case (fig.3). The survival of daphnids-parents, irradiated at low for the subject doses (including 0.1 Gy), is significantly lower than in the control, and the effect does not increase even with a very large increase in radiation exposure (at 200 times).

These results enabled us to proceed to study the viability of the 1st generation offspring born from the daphnids-irradiated. The data is presented in Figure 4b. Likewise, we see that in this case the survival of experimental animals is lower than in control, and that the effect does not increase with the increasing doses.

Significant deviations from control observed for daphnids-parents and for unexposed descendants of the first generation show the mass nature of the disruption of the described type. However, we haven't seen valid deviations between control and experimental groups for second and third generations of daphnids.

Conclusions

Similar dose-independent inheritable mass damage was found in different species after low radiation. The results support the idea of the universal nature of this unusual form of reaction (Bychkovskaya 1986, Bychkovskaya I.B. et al 2006). Innovative materials are obtained showing that the phenomenon can manifest itself in violation of important physiological functions (experiments on ciliates) (Sarapultseva 2008) and on the organism level of integration (experiments on daphnids) (Sarapultseva et al 2009). In the experiments on daphnids the inheritance of the effect in F_1 confirmed previously observed effect on rats (Bychkovskaya et al 2006). It also showed the elimination of damage in subsequent generations.

The features of this effect, including unusual for radiobiology dose-independent character, cannot be related to mutations. In general this phenomenon can be related to a category of epigenomic alterations (Jablonka et al 2008).

This research can be applied to the low radiation risks' assessment.

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Impact assessment of elevated levels of natural/technogenic radioactivity on wildlife of the North – INTRANOR

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Abstract

Arctic and boreal regions are often considered to be vulnerable to exposures from contaminants and therefore merit special attention in relation to the application of environmental impact assessment methodologies. The INTRANOR project has focussed specifically on environmental assessments for radiation exposure through application of existing methodologies and their adaptation to quantify transfer, exposure and effects in Boreal/Arctic ecosystems. Non-parametric statistical methods have been applied in order to estimate the threshold dose rates above which radiation effects can be expected in vertebrate organisms. The effects considered in the analyses include morbidity, reproduction, and life shortening and the approach has drawn upon data collations pertaining to databases on effects of chronic low-LET radiation exposure. In addition, industrial areas contaminated by uranium mill tailings and radium production wastes, in the Komi Republic, Russia, were selected as suitable study sites to study further the effects of exposure to radiation under boreal conditions. Dose–effect relationships have been established for natural *Vicia cracca L.* populations inhabiting this area. The various endpoints considered include chromosome aberration frequency in seedling root meristem, frequency of embryonic lethal mutation in legumes, germination of seeds and survival rate of sprouts of seeds. Analyses of data have allowed a benchmark to be established below which no decrease in reproductive capacity could be observed. Other work performed within the project includes the collation of data in relation to naturally occurring radionuclides and application of existing methodologies to characterise background radiation exposures. These dose-rates used in conjunction with dose rates known to have specific biological effects on individuals/populations may be a suitable means of contextualising the exposure attributable to enhanced dose-rates arising from human activities.

Introduction

At a European regional level, methodologies to assess the impact of exposure to ionising radiation on flora and fauna in European temperate and Arctic environments have been developed in two European collaborative projects "FASSET - Framework for Assessment of Environmental Impact" (Larsson et al., 2004) and "EPIC - Environmental Protection from Ionizing Contaminants in the Arctic" (Brown et al., 2003) respectively. These studies were superseded by the project "ERICA - Environmental Risk from Ionising Contaminants: Assessment and Management" wherein risk assessment methodologies have been developed and issues relevant to decision making in the context of the management of environmental impacts of radioactivity have been addressed (Larsson, 2008). Of particular relevance to the Arctic is the project EPIC which provides a number of the foundation stones that are prerequisite in the process of developing a robust assessment methodology. However, the development of the EPIC framework was curtailed at a point that did not incorporate risk characterisation or concomitant management options. With this in mind, the central rationale behind the INTRANOR project was to build upon the recent advances in environmental impact assessments, as detailed in the abovementioned research programmes, with focus on adapting the systems for Arctic/boreal environments, developing the risk characterisation component of the analysis and testing the assessments for actual situations. Some of the activities in the project are presented below.

Further development of Environmental Impact Assessment methodologies

The Environmental Impact Assessment methodologies outlined above are constructed around the concept of Reference organism, definitively specified in Larsson (2004) as '*a series of entities that provide a basis for the estimation of radiation dose rate to a range of organisms which are typical, or representative, of a contaminated environment. These estimates, in turn, would provide a basis for assessing the likelihood and degree of radiation effects'*. The approach was designed to be compatible with the methodology adopted by the International Commission on Radiological Protection, ICRP (2008) and as such some of the geometries that had been proposed for the ICRP's "Reference Animals and Plants" were used as defaults in the ERICA Approach. Reference organisms were defined and used for the derivation of geometric relationships between radiation sources and organisms, as well as for considerations of the dosimetry of both external and internal exposure (further discussed by Ulanovsky et al., 2008). The concept of reference organism also includes the consideration of transfer to the plant or animal and occupancy of the organism at locations within their habitat. With regards the former, this is achieved through the application of default concentration ratios, based on literature review or the use of suitable analogues or models, in order to derive body concentrations from media concentrations (e.g. see Hosseini et al., 2008). Occupancy factors, i.e. the time spent by an organism at a particular location within its habitat, have often been defined to maximise the dose-rate. A further intended use of reference organisms was that they could be used for pooling some of the effects data generated for a range of species. In addition, the selection of reference organisms was made with the intention of making it possible to address most protected species within Europe (and therefore be of applicability to envisaged requirements for environmental impact assessment). In view of the fact the ICRP are in the process of developing a data set that are references in the truest sense, the organism lists in EPIC and ERICA might be more appropriately referred to as representative organisms (Pentreath, 2009) which implies an application within site specific situations much in the same way as the ICRP suggest using the representative person with habits typical for those of a small number of (human) individuals who are most highly exposed within a given assessment.

Work conducted in the INTRANOR project has led to the conclusions that : Although the EPIC reference organism suite would seem to provide us with a reasonable starting point in selecting region-specific organisms for the estimation of doses and radiation impact on wildlife of the North, there are severe limitations in the existing organism suite. In particular, the fact that in many cases the reference data either do not exist, as is the case for freshwater reference organisms, or are few (much of the transfer data) means that it is a moot point whether they should be regarded as reference points at all. This of course is immediately mitigated if we consider the EPIC suite to be ‘representative’ (as oppose to ‘reference’) organisms. Several years of additional work leads to the clear observation that the transfer data in ERICA are far superior to those collated in EPIC and, in many cases, the EPIC data have been incorporated into the much larger ERICA datasets. Furthermore, ERICA allows derivation of transfer and exposure for a much broader suite of radionuclides and provides an indication of uncertainty for overall exposure estimates through the collation of detailed statistical information for concentration ratios. Additionally, with regards transfer there appears to be little justification, in most cases, for using smaller Arctic/boreal specific data sets in lieu of more comprehensive global data sets. In fact, following the arguments of Sheppard (2005), not drawing on larger generic data sets (as typified by ERICA) may be counter-productive. In the few named examples where studies have been undertaken to demonstrate a significant difference between Arctic/boreal and temperate data (see Brown et al., 2004a), then work should be done to provide these alternative values for use within an Environmental Impact Assessment. A cursory examination of the dosimetric models for ERICA also suggests that they may be suitably applied to the Arctic. This reflects the fact that the ERICA dosimetric models are based on highly generic organism categories, e.g. pelagic fish, bird etc. (Table 1) and it is therefore non-problematic to illustrate that such types of biota will be present in Arctic/boreal regions (and in most ecosystems for that matter).

For the sake of simplicity it seems reasonable to recommend, in relation to conducting environmental impact assessment in the Arctic, using the organism suite applied in the ERICA approach. This might be supplemented by a representative organism suite for application in the Arctic/boreal environment where this is deemed necessary. The system for performing an impact assessment for an Arctic site would therefore involve :

- the use of ICRP Reference Animals and Plants and guidance to allow, inter alia, site specific assessment to be more readily compared to other assessments.
- the use of the ERICA integrated approach (Larsson, 2008) supported by the ERICA Tool (Brown et al., 2008) to provide screening tools and generic

data sets to perform the assessment. The ERICA dataset might be regarded as providing data for 'representative organisms'. In this way the approach might be considered as a management tool to be applied for authorisations/compliance, for example, and as providing a pool of information to allow a more comprehensive assessment where clearer links can be made to (and relevant data derived for) actual species of interest

- If required, to collate and undertake a site specific investigation drawing on tools available in the ERICA Tool, e.g. defining site specific geometries and occupancy factors, conducting probabilistic model runs etc. Furthermore, site specific activity concentration and transfer data should be collated and for the transfer data Bayesian methods might be implemented in refining the information (e.g. see Barrera et al. 2007), i.e. allow the assessor to utilise the more comprehensive datasets available from ERICA in tandem with newly acquired site specific information.

Table 1. Boreal/Arctic Representative organisms for application in an Environmental Impact Assessment – from the ERICA Tool (Brown et al., 2008) - in italics within brackets = the corresponding ICRP Reference Animals and Plants, for which the ERICA Tool uses the proposed ICRP geometries as default.

Freshwater	Marine	Terrestrial
Amphibian *(frog)*	(Wading) bird *(duck)*	Amphibian *(frog)*
Benthic fish	Benthic fish *(flat fish)*	Bird *(duck)*
Bird *(duck)*	Bivalve mollusc	Bird egg *(duck egg)*
Bivalve mollusc	Crustacean *(crab)*	Detritivorous invertebrate
Crustacean	Macroalgae *(brown seaweed)*	Flying insects *(bee)*
Gastropod	Mammal	Gastropod
Insect larvae	Pelagic fish	Grasses and herbs *(wild grass)*
Mammal	Phytoplankton	Lichen and bryophytes
Pelagic fish *(salmonid/trout)*	Polychaete worm	Mammal *(rat, deer)*
Phytoplankton	Reptile	Reptile
Vascular plant	Sea anemones/true corals	Shrub
Zooplankton	Vascular plant	Soil invertebrate (worm) *(earthworm)*
	Zooplankton	Tree *(pine tree)*

Statistical analyses of dose-effects data

There have been recent efforts to derive predicted no effects dose-rates for wild-life based on the construction of species sensitivity distributions (SSD) that are in turn constructed from effective Dose-rate 10 % (EDR_{10}) data from individual experiments (Garnier-Laplace et al., 2008, Andersson et al., 2009). Although such methods are well established in the ecotoxicological sciences (see EC, 2003), their application within the field of radiological protection is relatively new and not without deliberation. One important limitation is the fact that the criteria used in the selection of appropriate data for construction of the SSD are extremely strict which results in the loss of a large dataset that may have great utility in informing the derivation of appropriate dose-rate benchmarks. For this reason, alternative methods for analysing dose-effects data have been explored within the INTRANOR project. Non-parametric statistical methods have been applied in order to estimate the threshold dose rates above which radiation effects can be expected in vertebrate organisms by Sazykina et al. (2009). The effects considered in the analyses include morbidity, reproduction, and life shortening and the approach has drawn upon data collations pertaining to databases on effects of chronic low-LET radiation exposure. Radiation thresholds dose-rates in vertebrate animals subjected to chronic low-LET exposure were estimated to be 2.1 x 10^{-4} Gy/day, 4.1 x 10^{-4} Gy/day and 1.1 x 10^{-3} Gy/day for the endpoints of morbidity, reproduction and life-shortening respectively. This means that that the generic screening value suggested by Andersson et al. (2009), based on SSD analysis corresponds to the lowest level for morbidity effects in mammals based on the non-parametric analyses of data proposed in INTRANOR by Sazykina et al. (2009). Further work may be required to establish whether additional uncertainty factors should be applied to generic benchmarks to account for the (perceived) greater sensitivity of Arctic systems to impacts of radioactivity.

Estimating the impacts of ionising radiation on natural plant populations

The work described above by Sazykina et al. (2009); Andersson et al. (2009) and others essentially involves an assimilation of data for dose-effects relationships for many different species of organism. In many cases of plants and animals this relationship between exposure and effects is poorly characterised or unknown. Many of the available data pertain to laboratory studies and there are few published data reporting the occurrence of biological effects in natural populations of plants and animals located within areas with enhanced concentrations of naturally occurring radionuclides, In this light, experiments have been conducted on herbaceous vegetation, Tufted vetch (*Vicia cracca)* within a boreal environment contaminated with radionuclides from ^{238}U and ^{232}Th decay chains to investigate the effects of different dose regimes (Evseeva et al., 2009). The following endpoints were selected in this study:

- chromosome aberration frequency in seedling root meristem,
- frequency of embryonic lethal mutation in legumes,
- germination of seeds and
- survival rate of sprouts of seeds.

The latter three parameters were selected as they reflect reproduction capacity in the studied organisms. A site near the Vodnyi settlement within the Komi Replublic was selected as a suitable field station. Detailed descriptions of the study area as well as results of previous radioecological investigations are presented by Geras'kin et al. (2007). The contamination of the site has been caused by storage of the uranium mill tailings and radium production wastes.

Doses to *V. cracca* plants from areas contaminated with radium production wastes, exceeded the natural background level (0.0007– 0.001 Gy) by a factor of 1.4–900. The main dose forming radionuclide was ^{226}Ra. The study of relationships between biological effects observed and weighted absorbed dose for *V. cracca* seeds showed that nonlinear models fit experimental data better than linear ones. The relationship between the frequency of the chromosome aberration in seedlings' root tip cells and the absorbed dose was found to be quadratic (Figure 1).

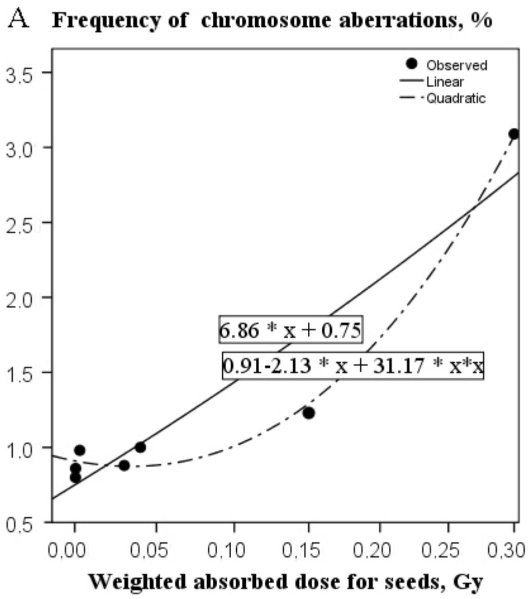

Fig. 1. Plot of models showing the relationship between chromosome aberration frequency in meristematic root tip cells of seedlings and weighted absorbed dose for seeds (From Evseeva et al., 2009).

The exponential model provided the best result in describing the empirical dependence between the absorbed dose and both the germination capacity of seeds and the survival rate of sprouts of *V. cracca*. No significant relationship (p_F>0.05) was found between the embryonic lethal mutations frequency and the absorbed dose. For *V. cracca* plants inhabiting areas contaminated with uranium mill tailings and radium production wastes, a weighted absorbed dose for the aboveground part of plants of 0.2 Gy (weighting factor for alpha particles=5) during the vegetation period (120 days) can be considered to be a level below which no increase in genetic variability and decrease in reproductive capacity is observable. Evseeva et al.(2009) essentially derive a (weighted) no observable effect dose-rate of 1.66 mGy/day (0.2 Gy over 120 days) for *V. cracca* growing over a site contaminated by enhanced levels of naturally occurring radionuclides.

Background dose rate characterisation

In helping to assess the impacts of radiation exposure on organisms, Pentreath (2002) suggested that only two reference points can be utilised in a practicable way, these being natural background dose-rates and dose-rates known to have specific biological effects on individuals/populations. Building on this, the ICRP has suggested that it would be helpful for the decision-making process if information concerning effects on biota was set out in terms of multiples of the natural background dose-rates typically experienced by each type of Reference Animals and Plants (ICRP 2008). For such a structuring of data to be made, there is clearly a requirement to provide well characterised background dose-rate estimates for the selected Reference Animals and Plants However, whilst data have been collated for terrestrial Reference Animals and

Plants (Beresford et al. 2008) information has not been collated with the express purpose of deriving background dose-rates to aquatic Reference Animals and Plants. Furthermore, there are significant limitations associated with earlier compilations of naturally occurring radionuclides in the context of deriving background dose-rates for aquatic wildlife. For example, no natural radionuclide data for European freshwater organisms were identified in the review of Brown et al. (2004b).

In light of these propositions, information on activity concentrations of naturally occurring primordial radionuclides for marine and freshwater ecosystems have been applied and appropriate dosimetry models used to derive absorbed dose-rates for Reference Animals and Plants. Although coverage of activity concentration data is comprehensive for sediment and water, few, or in some cases no, data were found for some organism groups, for most radionuclides. The activity concentrations for individual radionuclides in both organisms and their habitat often exhibit standard deviations that are substantially greater than arithmetic mean values, reflecting large variability in activity concentrations. The dominating radionuclides contributing to exposure in the Reference Animals and Plants are ^{40}K, ^{210}Po and ^{226}Ra.

The mean unweighted and weighted dose-rates for aquatic Reference Animals and Plants are in the ranges 0.07–0.39 μGy h^{-1} and 0.37–1.9 μGy h^{-1} respectively. Typical results are shown in Figure 2.

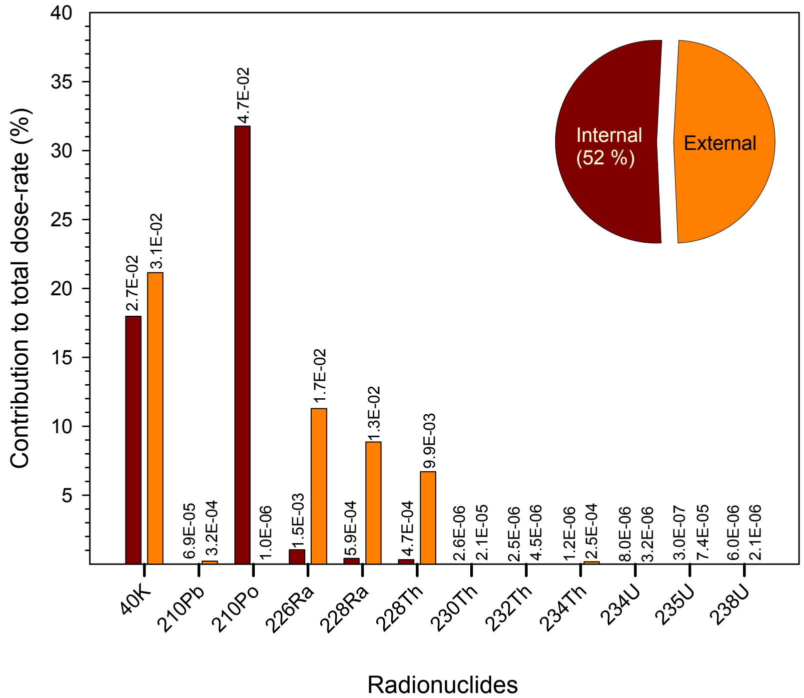

Fig. 2. Contributions of different radionuclides to internal, external and the total unweighted dose-rates for (marine) Flatfish. The vertical-orientated values are estimated internal and external dose-rates. Reproduced from Hosseini et al. (in press).

The data compiled and the dose-rates estimates in this exercise pertain to specific organism types as defined by the ICRP and are not specifically related to Arctic organisms. Further studies are required to determine the differences in exposures of boreal organisms compared to temperate organisms building on previous studies on this theme (Sazykina et al., 2003).

Conclusions

The INTRANOR project has dealt with many aspects relating to the development of a more robust methodology for assessing the impact of ionising radiation on wild-life. The general methodology itself has been reviewed with particular focus on the applicability of representative organisms in the context of boreal/Arctic systems and how these might relate to the reference animals and plants being developed by the ICRP. Dose response data from published literature has also been investigated by considering the applicability of different statistical methods in the derivation of generic benchmarks for application in protection systems. There are clearly deficiencies in relation to well characterised dose-response relationships for many organism types especially under field conditions. With this in mind, experimental studies have been conducted to investigate the dose response of herbaceous vegetation to exposures from naturally occurring radionuclides. This has allowed the shape of dose response relationship to be elucidated and information relating to no observable effects and lowest observable effects doses to be elaborated. Finally work has also been conducted in relation to the characterisation of naturally occurring radionuclides in aquatic ecosystems. By organising the data around the reference animals and plants considered by the ICRP, the data should have direct relevance as a point of reference for contextualising calculated dose-rates for any given impact assessment for ionising radiation.

Acknowledgements

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Testing of linearity assumption of soil-to-plant transfer factors in boreal forest

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Abstract
The use of traditional transfer factors (TF) in radioecological modelling is based on the assumption that element concentrations in plant and soil are linearly related. We tested the validity of the linearity assumption for Mo, Ni, Pb, U and Zn in boreal forest using narrow buckler fern (*Dryopteris carthusiana*) as a model plant. Higher TFs were generally found in lower soil concentrations. A non-linear function was found to explain the relationship between TF and soil concentration significantly better than the linear assumption. The use of traditional TFs may be inadequate for estimating plant concentrations, and new approaches to modelling soil-to-plant transfer are thus needed.

Introduction
In radioecological modelling, the transfer of an element from soil to plants is commonly described by an element-specific transfer factor (TF), which is defined as the plant/soil ratio of the concentration of the element (IAEA 2010). The use of these TFs is based on the assumption that soil and plant concentrations are linearly related. However, this assumption is not valid for essential elements and has also been questioned for non-essential elements (Sheppard and Sheppard 1985; Simon and Ibrahim 1987).

From experience in other fields of science, such as studies on plant nutrition and uptake of heavy metals, it s known that concentration in plants as a function of concentration in soil generally shows a steep increase at low concentrations towards a plateau at high concentrations. Several mathematical functions have been used to describe this behaviour, e.g. the Freundlich equation (Martinez-Aguirre et al. 1997; Krauss et al. 2001; Yaylah-Abanuz and Tüysüz 2009) and the Langmuir equation (Wenger et al. 2002; Han et al. 2006; Redjala et al. 2010).

In this study the soil-to-plant transfers of three essential (Mo, Ni and Zn) and two non-essential (Pb, U) elements were studied in a boreal setting using narrow buckler fern (*Dryopteris carthusiana*) as a model species. Empirical data were used investigate the relationship between plant and soil concentrations. The total element concentrations rather than radionuclide concentrations were measured in this study. Radionuclides and stable isotopes of the same element are considered to behave similarly in ecosystems, and stable isotopes can thus be used for modelling the behavior of radionuclides in the biosphere (IAEA 2010).

Material and methods

The study site was a uranium occurrence in a herb-rich forest *(Oxalis-Maianthemum* type*)* located in Nilsiä, Eastern Finland (N63°04', E27°54') (Fig 1).

Fig 1. Overall view of the study site.

The soil samples were collected from 29 systematically selected sampling points in June 2007. The topsoil was collected to the depth of 100 mm, which was considered to be the rooting depth of the understorey species, within an area of 100 mm x 100 mm. The soil samples were dried at 40 ^{o}C and sieved to diameter fractions < 2mm and > 2mm. The fraction < 2mm was used for analysis.

Plant samples were collected at the same time as soil samples. The collected plant species was narrow buckler-fern (*Dryopteris carthusiana*) which was present at 27 sampling points. The plant samples were divided into root (containing both rhizome and fine roots), petiole and leaf fractions. The plant samples were dried at 60 ^{o}C before analysis.

Inductively coupled plasma-mass spectroscopy (ICP-MS) measurements were carried out in the laboratory of Labtium Ltd. in Espoo, Finland providing pseudototal concentrations of Mo, Ni, Pb, U and Zn after nitric acid digestion (EPA 3051) in microwave oven. An estimate of mobile fraction of these elements in the soil samples was obtained by ICP-MS analyses after 1 M Ammonium acetate (NH_4Ac, buffered at pH 4.5) leach. Detection limits for Mo, Ni, Pb, U and Zn were 0.02; 0.3; 0.05; 0.01 and 0.4 mg/kg, respectively. If the concentration of an element was below detection limit the value of detection limit divided by two was used in calculations (EU 2008).

All the results were corrected to represent dry matter content (dw). TFs based on the IAEA (2010) definition for U, Mo, Ni, Pb and Zn were calculated separately for root, petiole and leaf as follows:

$TF_{t,\,p}$= concentration (dw) of Mo, Ni, Pb, U or Zn in plant part p (p= root, petiole or leaf) / total concentration (dw) of Mo, Ni, Pb, U or Zn in soil
$TF_{m,\,p}$= concentration (dw) of Mo, Ni, Pb, U or Zn in plant part p (p= root, petiole or leaf) /mobile concentration (dw) of Mo, Ni, Pb, U or Zn in soil

To investigate the relationship between TFs and soil concentrations, the data were fitted with non-linear functions. In this paper, we described results based on a Langmuir-tupe function. The Langmuir equation is of the form:

$$C_p = a\, b\, C_s / (1 + b\, C_s), \qquad (1)$$

where a and b are experimentally determined constants, C_s is the concentration in soil, and C_p is the concentration in the plant. To allow fitting with data that suggested a slowly accumulating function at high soil concentrations, we added a linear term to the Langmuir equation and got what we call the Langmuir+ equation:

$$C_p = a\, b\, C_s / (1 + b\, C_s) + c\, C_s \qquad (2)$$

To study the relationship between TF and element concentration in soil, the following equation was derived from the Langmuir+ equation:

$$TF = C_p/C_s = a\, b / (1 + b\, C_s) + c \qquad (3)$$

Results

Soil concentrations

The medians and ranges of soil total and mobile concentrations of Ni, Pb, U and Zn are shown in Table 1. For Mo the majority of measured mobile concentrations were under the detection limit and thus only the range is shown in Table 1. Zn was the most abundant of these five elements followed by Pb, Ni, U and Mo, respectively, whether soil total or mobile concentration was considered.

Table 1. Medians and ranges of total and mobile Mo, Ni, Pb, U and Zn concentrations in soil (mg kg^{-1}).

	Mo	Ni	Pb	U	Zn
Total	0.83 (0.17-32)	9.0 (4.3-26)	8.7 (5.0-120)	1.6 (0.46-110)	27 (14-74)
Mobile	<0.02-0.15	0.68 (<0.3-7.7)	2.6 (0.56-17)	0.33 (0.04-66)	3.1 (0.72-12)

Soil-to-plant transfer factors (TFs)

A trend towards higher TFs at lower soil concentrations was systematically seen in the data for both essential and non-essential elements. Data for Pb are shown as an example in Fig. 2.

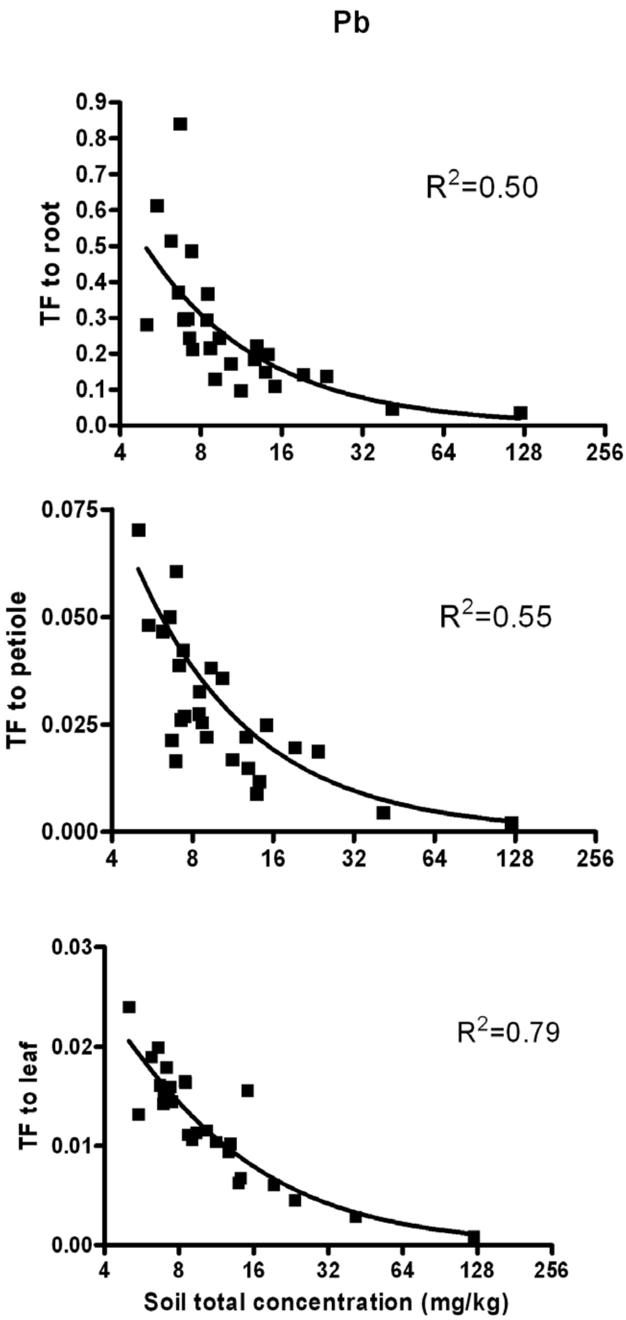

Fig. 2. TFs of Pb for different plant parts (root, petiole, leaf) as a function of soil total concentration. The line shows the fit with equation (3) (see Materials and Methods).

Non-linear and linear model

When equation (3) was fitted with the data, the goodness of fit (R^2) ranged from 7.68E-06 to 0.94 (average 0.44). The R^2-values for Pb are shown as an example in Fig. 2. In all cases the fits with the non-linear function were better than fits with linear assumption (constant TF), with produced R^2-values below 5.5E-16 (average 2.0E-17).

Discussion

Lack of linearity between soil and plant concentrations was clearly shown in this study for both essential and non-essential elements, in agreement with previous results (Sheppard & Sheppard 1985; Simon & Ibrahim 1987; Cook et al. 1994). Our results indicated that TFs are high at low soil concentrations and decrease towards a constant value at high soil concentrations. Therefore, the use of constant TFs may be justified at high soil concentrations but may lead to underestimation of plant concentrations in case of low soil concentrations (Martinez-Aguirre et al. 1997). For example, possible leaks from final disposal of spent nuclear fuel, will in all likelihood lead to low soil concentrations.

Empirically determined TFs, generally show very large variation, and published TF values therefore include a lot of uncertainty (Higley & Bytwerk 2007). Examination of the non-linear functions found in this study (Fig. 2) suggests that part of the variation (which has been traditionally assumed to be random) may in fact be systematic variation with soil concentration. Therefore, examining soil-to-plat transfer as a non-linear function might greatly improve the accuracy of predictions concerning plant concentrations.

Conclusions

The soil-to-plant transfer of both essential (Mo, Ni and Zn) and non-essential (Pb, U) elements is clearly non-linear. New approaches to model the soil-to-plant transfer are needed to improve the accuracy of model predictions.

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Application of ellipsoid geometry in dose assessment of forest plants

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Abstract

Posiva is implementing a deep repository for spent nuclear fuel in Olkiluoto, Finland. A site-specific safety case is being produced, and data has been acquired for the assessment models. In the dose assessment of other biota, international approaches are applied. However, determining geometry parameters for the variability of forest plants is a challenging task, not to mention coupling of the geometry to plant physiology. First, it is well known that physical dimensions of plants differ widely between plant species. Secondly, dimensions of certain plant species are dependent on growth conditions (e.g. weather, soil fertility), biological age and developmental stage of plant (e.g. many annual herbs have their highest biomass in largest extent during late summer). Furthermore, plant parts have different physiological tasks, and they should be taken into account as well (e.g. photosynthetically active green leaves compared to lignified brown stems). In this paper we illustrate problems in plant geometry using some key plant species of boreal forests and propose alternative geometries for assessment use.

Introduction

Posiva is implementing a deep repository for spent nuclear fuel in Olkiluoto, Finland. A site-specific safety case (Posiva 2008) is being produced, and data has been acquired for the assessment models (Hjerpe et al. 2010). In the dose assessment of other biota, international approaches are applied.

In all generic approaches (Copplestone et al. 2001, Brown et al. 2003, Larsson et al. 2004, Beresford et al. 2007, ICRP 2008) the basis of the dosimetric modelling is to select reference organisms that are represented by target geometry (a phantom). It has been widely acknowledged that identification of actual species or groups of species is helpful, if not even necessary, to derive the geometrical and radionuclide transport properties. However, selection of such species or group does not refer to any particular species or group of species (e.g. ICRP 2008).

Furthermore, the reference organisms can be considered to be typical of the environment "in the sense that one might expect to find them there" (ICRP 2008) or at least in the sense of being representative of the environment (Copplestone et al. 2001, Beresford et al. 2007). This is supposed to be also the meaning of the Finnish regulatory requirement to assess "typical radiation exposures" (YVL 8.4, YVL E-5).

Concerning the representatives of forest plants, roots, meristem, buds and/or seeds have been considered as the dosimetric target (Copplestone et al. 2002, Brown et al. 2003, Larsson et al. 2004), or "whole" plants have been used, as apparently in (Beresford et al. 2007). Whereas in these approaches the geometry is defined as an ellipsoid, the ICRP approach (ICRP 2008) considers pine tree as an ellipsoid for estimating the internal doses but for external doses uses a homogeneous canopy layer, and in (Larsson et al. 2004) buds and meristem of shrubs and trees are located in a canopy. Respectively ICRP also uses an ellipsoid for a spike of Wild grass, but its meristem as homogeneous layer on the ground surface. In all these approaches the organisms are assumed to be homogeneous with a density of 1 g/cm³ and in an infinite water medium (ensuring sufficient secondary photon transport). The composition of the reference organism is best described to be a four-component tissue substitute (ICRP 2008).

However, determining geometry parameters for the variability of forest plants is a challenging task, not to mention coupling of the geometry to plant physiology. First, it is well known that physical dimensions of plants differ widely between plant species. Secondly, dimensions of certain plant species are dependent on growth conditions (e.g. weather, soil moisture and fertility), biological age and developmental stage of plant (e.g. many annual herbs have their highest biomass in largest extent during late summer). Growing density and competition between plant species (or between individual species) affect also size and shape of single species. Furthermore, plant parts have different physiological tasks, and they should be taken into account as well (e.g. photosynthetically active green leaves compared to lignified brown stems).

In this paper we illustrate problems in plant geometry using some key plant species of boreal forests and propose alternative geometries for assessment use.

Material and methods

Selection of representative plants and defining their dimensions

Due to a large number of plant species and large variation in plant dimensions between different species it is impossible to cover dimensions of single plant species in dose assessment. A practical approach would be using of functional plant groups from which some typical key species are selected for detailed examination. However, two main problems still remain. First, what are the requirements and justifications for selection of key species from plant groups? Secondly, how to handle a huge variation inside plant species?

Plant species groups have been used in mapping of the nutrient status of the vegetation on Olkiluoto Island in 2005 (Tamminen et al. 2007). Plant groups were also used in determining biomass and chemical composition of the vegetation on the intensive forest monitoring plots at Olkiluoto in 2008 (Salemaa & Korpela 2009). The latter included six functional plant groups (some exemplars):

1) Evergreen dwarf shrubs (*Vaccinium vitis-idaea*)
2) Deciduous dwarf shrubs (*Vaccinium myrtillus*)
3) Lower herbs (*Maianthemum bifolium, Oxalis acetosella* and *Trientalis europaea* (shoots die annually), and *Linnaea borealis* (perennial shoots)

4) Ferns (*Equisetum sylvaticum*, *Dryopteris carthusiana*, *Gymnocarpium dryopteris*, *Pteridium aquilinum* (shoots/leaves die annually)
5) Grasses (both perennial and annual leaves, *Deschampsia flexuosa*)
6) Mosses (lower parts die gradually, *Pleurozium schreberi, Hylocomium splendens*)

In this study selection of key plant species was based on earlier vegetation inventories at Olkiluoto (Tamminen et al. 2007, Salemaa & Korpela 2009), frequency of plant species and their importance in food chains. Hence bilberry (*Vaccinium myrtillus*) was selected to represent deciduous dwarf shrubs, chickweed wintergreen (*Trientalis europaea*) lower herbs, wavy hair-grass (*Deschampsia flexuosa*) grasses and red-stemmed feather-moss (*Pleurozium schreberi*) mosses. All the selected plant species belong to a group of ten most common plant species in Southern-Finland (Reinikainen et al. 2000). Red-stemmed feather-moss also is commonly used as a bioindicator species in environmental studies. Scots pine and Norway spruce dominate forests on Olkiluoto Island (Saramäki & Korhonen 2005). Because appearance of Scots pine differs clearly from that of Norway spruce (i.e. tree crown forms separate part in the older trees), pine was selected to represent tree species in this case study.

Selection criteria for the reference organisms include ecological niche, intrinsic radiosensitivity, radioecological sensitivity, distribution (e.g. presence year-round), suitability to research and monitoring and protected status (IAEA 2010). It seems that the criteria are quite well in accordance with the key species we have selected for this study. However, intrinsic radiosensitivity of the selected species is an uncertain issue, especially regarding long-lived radionuclides characteristic to releases from deep repositories.

Determination of ellipsoid geometries was attempted to base on literature where, however, only plant heights were available. Therefore some estimates were created for plant width and length. For bilberry a demonstrative sampling was carried out. Three samples of bilberry in different developmental stages (i.e. age of plant) were subjectively collected and the dimensions measured (Fig. 2). Some results based on expert judgment during Olkiluoto Biosphere Description 2009 process (Haapanen et al. 2009) were also included. In addition, a few applications of determination ellipsoid geometries for certain plant species were proposed and discussed.

Evaluation of impacts on doses to plants

In this contribution, the effect of varying sizes of ellipsoids representing plants has been preliminary studied using the ERICA Assessment Tool (Beresford et al. 2007), version 1.0, May 2009. Default values have been used except for the organism dimensions (Table 1) and the concentration ratios, which were taken from Helin et al. (2010). The activity concentration in the soil was taken from the biosphere assessment simulations, corresponding the maximum exposure in terrestrial systems (case Sh4Q-C, dried area of Mäntykarinjärvi Lake, year 12020; Hjerpe et al. 2010). As the needed C-14 concentration in air was not easily extractable from the simulation data, it was omitted here, similarly to Pd-107 and Sn-126 not supported by the ERICA Assessment Tool. Due to the pathway, contribution of C-14 is believed not to invalidate the results below.

The contribution of the two other omitted radionuclides would be insignificantly small anyway.

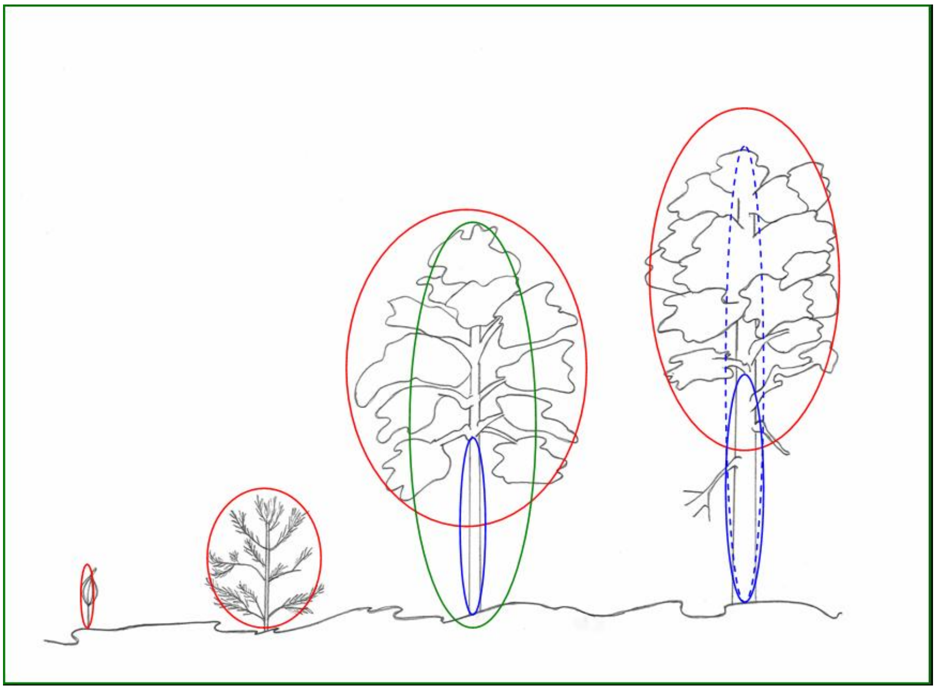

Fig. 1. Examples on Scots pine heights in different developmental stages. From left to right: young seedling (1-year-old, height 0.1 m), advanced seedling (<10 yrs, h= 5 m), mature pine tree (100-yrs-old, h= 22 m) and old mature pine tree (150-yrs-old, h= 25 m). Drawing: A. Hamari /Metla.

Fig. 2. Examples on bilberry of different developmental stages growing in mesic heath forest during winter season. Maximum dimensions (width x length x height, cm) from left to right: 18 x 10 x 25 (A), 6 x 4 x 18 (B) and 25 x 30 x 35 (C).

Results and discussion

Plant dimensions and ellipsoid geometry

A variety of plant dimensions are presented in Table 1. Typically only plant heights were found in literature. Those heights represent different growth conditions of plants, e.g. *Trientalis europaea* appears in wide range of habitats from xeric heat forests to herb-rich forests, and *Deschampsia flexuosa* and *Vaccinium myrtillus* from sub-xeric to herb-rich heat forests (Reinikainen et al. 2000). *Pleurozium schreberi* is the most common plant species in Finland and has the largest range of habitats among forest mosses (Reinikainen et al. 2000). In many cases plants reach their highest dimensions in the most fertile sites.

The age of plant (i.e. developmental stage) determines also plant size. Illustrative figures are shown for *Pinus sylvestris* and *Vaccinium myrtillus* (Table 1, Figs. 1 and 2). Some examples on determination of ellipsoids for pine are also demonstrated (Fig. 1). There are two options: consider tree crown and stem below crown separately, or consider the whole tree. ICRP (2008) also suggested that tree crown could be analysed as a one-level sheet.

Table 1. Dimensions (cm) of selected key plant species.

Species	Class/source	Width	Length	Height
Pinus sylvestris	1-year-old seedling	3[1]	3[1]	10[1]
	<10 yrs old	200[1]	200[1]	500[1]
	<100 yrs old	500[1]	500[1]	1200[1]
	150 yrs old	800[1]	800[1]	1000[1]
	BSD2009, crown [2]	100	100	475
	BSD2009, stem [2]	10	10	475
Vaccinium myrtillus	BSD2009[2]	15[2]	15[2]	20[2]
	Mesic heath forest	6 – 25[3]	4 – 30[3]	18 – 35[3]
Trientalis europaea	Literature	?	?	5 – 20[4]
Deschampsia flexuosa	BSD2009[2]	50	50	40
	Literature	?	?	30 – 70[4]
Pleurozium schreberi	BSD2009[2]	2[2]	3.5[2]	2[2]
	Literature	?	?	5 – 10[5]
	Part of population [6]	60[6]	80[6]	5 – 10[5]

[1)] Hypothetical dimensions for crown of tree including part of stem
[2)] Haapanen et al. 2009; crown= crown of tree including part of stem, stem= stem of tree below crown
[3)] Measured samples, see Fig. 2
[4)] Hämet-Ahti et al. 1986, Mossberg & Stenberg 2005
[5)] Jahns 1980
[6)] E.g. on stone surface

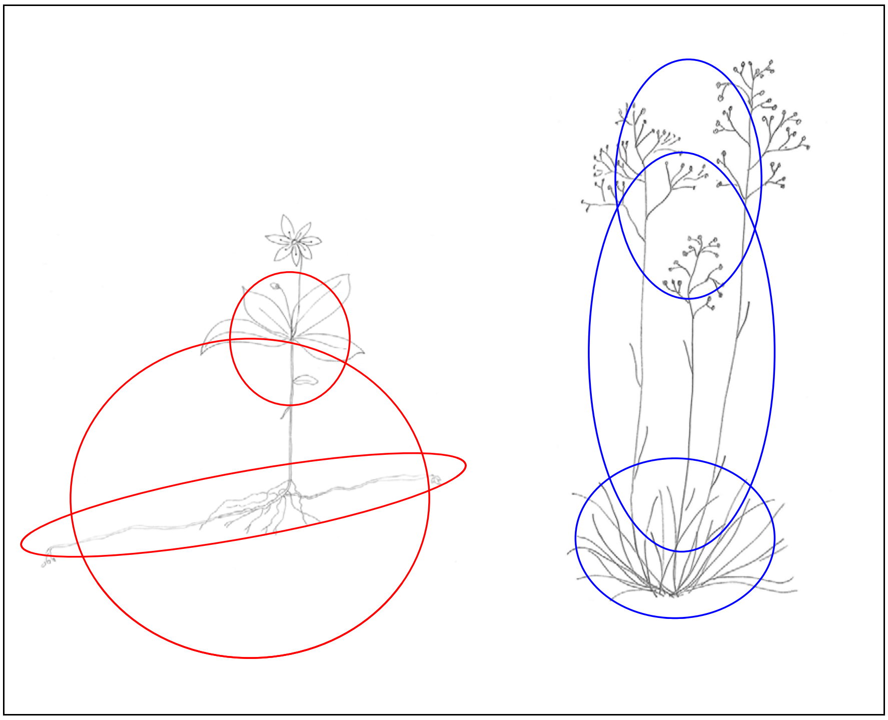

Fig. 3. Examples of ellipsoid geometry for *Trientalis europaea* (with rhizomes, on the left side) and *Deschampsia flexuosa* (perennial grass, on the right side). Drawing: A. Hamari /Metla.

Fig. 4. Red-stemmed feather-moss (*Pleurozium schreberi*) population on the left side and few individual mosses on the right side (Photos: L. Aro / Metla).

Challenges in determination geometries for species that have plant parts with distinctively features are presented in Figure 3. *Trientalis europaea* grows annually one to three superficial rhizomes which should be included in ellipsoid geometry. *Deschampsia flexuosa* is a typical example of perennial grasses with narrow leaves growing from the base and inflorescences with stems. In the case of this grass ellipsoids can be considered separately for inflorescences and leaves, or consider the whole plant which may be difficult due to the unbalanced biomass distribution in the grass.

Many plant species grow in dense populations, such as *Pleurozium schreberi* (Fig. 4). In this case ellipsoids can be determined for single moss or for a part of moss population. If the approach of single moss is used then neighbouring effect of other mosses should also be considered in dose assessment.

Implications for dose assessment

By keeping the other parameters constant and varying the size of *Vaccinium myrtillus* according to Table 1 (18·4·6, 20·15·15, 25·10·18, 35·25·30 cm), the dose conversion coefficients (DCC) for internal low beta or external radiation do not change, but those for internal beta/gamma radiation increase 0.1 to 130% from the smallest to the largest ellipsoid. For the two nuclides contributing most to the dose rate, Cl-36 and I-129, the increase is 1 and 17%, respectively.

Concerning the total absorbed dose rate to the different sizes of *Vaccinium myrtillus*, similar trend is observed but in more modest extent: compared to the smallest, the dose rate to the largest ellipsoid is 1% higher.

Similarly, for the *Pleurozium schreberi* the DCC for internal beta/gamma radiation and the total dose rate are higher for the part of the population (10·60·80 cm) compared to the ellipsoid representing an individual (2·3.5·2 cm; see also Fig. 4): The increase in the DCC is 0.4 to 190% (Cl-36 4%, I-129 27%) and the increase in the total dose rate is 4%, mostly explained by the contribution of chlorine and iodine.

It needs to be noted that these results are for the latest Posiva's assessment case producing highest exposure to the terrestrial biota (see above), and not for unit activity concentrations most commonly applied with the biosphere part decoupled from the rest of the safety assessment.

Anyway, even if the uncertainty from the organism size in the total dose rate remains much smaller than the notoriously large overall uncertainties in the biosphere assessments, there is an effect – provided that the ellipsoidal geometry assuming a phantom density of 1 g/cm³ and a substitute-material composition (ICRP 2008) is valid. For the external exposure the size does not matter, in this case, but the variation in the DCC of internal radiation infers that the geometry might have a role. At least, if the ellipsoids are used to represent also plants of varying actual shape, it appears that appropriate ellipsoid dimensions could be obtained only by careful scaling and bearing in mind the composition of the phantom and the role of the combined effect of concentration ratio and weight in the case of internal exposure. However, for screening purposes the simple ellipsoids and their casual dimensioning seem to produce robust enough results, provided that the screening limits are then set appropriately.

Conclusions

In the selection of the representative species and estimating their size should be done in the context of the site conditions. Furthermore the life stage, or age, of the plant should be taken into account. This means that ideally the sizes should be weighted with site type and plant age distributions of the area of the dose assessment, which actually is out of any reasonable resources. However, despite of large variation in plant dimensions and consequently in ellipsoids derived from them, differences in calculated doses remained quite minor in comparison with the overall uncertainties in the dose assessment. However, to improve the overall confidence to the assessment, some site-specific measurements of the selected reference species should be carried out.

It seems that the approach of using ellipsoids to represent plants in the dose assessment is reasonably robust especially for screening purposes. Still, in a graded approach to assessment, the level of conservativeness (overestimation of the doses) should decrease and realism increase with subsequent tiers. In the case of plants, there appears to be little justification for the use of the geometry – showing that size does not matter in a context of a dogmatic use of fixed geometries does not self-validate the approach; it only shows that the output is not sensitive to the dimensions. As the discussion is lacking at least from the main level documentation of the recent methodological descriptions, it is difficult to find confidence on that using an ellipsoidal geometry also for plants, of which many actually looks very different in shape, would be more conservative than a more characteristic geometry (e.g. a plate or a set of slabs). This is especially when the question comes to scaling the plant's physical dimensions to correspond the ellipsoid of uniform 1 g/cm³ matter substitute.

Even though the doses to biota are expected to remain small in the case of deep geological repositories (at maximum in the order of 10^{-3} μGy/h to terrestrial organisms in the plausible calculation cases of Posiva (Hjerpe et al. 2010), i.e. three orders of magnitude below the screening value recommended by Beresford et al. (2007)), further work on the dose assessment for plants seems useful at least to improve the conceptual basis and confidence to the approach. It is required that the assessment shall overestimate the consequences, but it shall not be overly conservative (YVL 8.4, YVL E-5); a better conception of the degree of realism of estimating doses to plants would have its use.

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Assessment of critical doses for reproduction and survival of cultivated plants

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Abstract

The aim of the study is to estimate radiation doses that are critical for sustainability of agrocenosis. Available information on dose dependences in such umbrella endpoints as reproductive potential, survival, morbidity, alterations in morphological and biochemical processes, genetic effects in crops, vegetables, fruit trees, etc are gathered from papers issued mainly in Russian scientific press during last 50 years. Data are maintained as database in MS Access that contains about 7000 entries; the work is ongoing. Quantitative data obtained from different sources are transformed in relative units, which makes possible analysing unified arrays referred to specific endpoints. As critical, there are considered doses producing 50% changes of biological effect at acute impact, or dose rates resulting in 10% changes at chronic exposure of plants. There are three main exposure situations for plants: acute irradiation of seeds, acute and chronic exposure of vegetating plants. Critical doses and dose rates are assessed from dose-effect dependences constructed with data sets, referred to indexes of reproduction and survival. It is found that data on survival collected so far are rather insufficient to estimate critical dose (rates) for species of cultivated plants. From the available information, the predicted no-effect doses and dose rates for agrocenosis are estimated basing on reproduction endpoint. They range within 67÷80 and 15÷17 Gy at acute exposure of the most radiosensitive species in dormant and vegetation periods, correspondingly, and 3÷10 mGy/h at chronic exposure of vegetating plants. The estimates obtained are going to be improved with further development of the database and treatment approaches.

Introduction

In recent years many efforts have been undertaken to develop a system of radiation protection for non-human biota (ICRP 2003, 2009). Agrarian ecosystems are of special concern from the viewpoint of establishing safe levels of radiation impact on the environment. On one hand, their contamination can affect human health via radionuclide uptake with food. On other hand, agroecosystems are ones of the most sensitive to a number of environmental impacts including ionising radiation (Agricultural radioecology 1991). While the existing system of radiation standards (ICRP 2007) is effective in providing radiation protection of human and restricts radionuclides content in food chains, there are no guidelines on setting any limitations to directly protect agrarian ecosystems from negative effect of radiation. The aim of this work is to develop methods for an assessment of critical doses and dose rates that can result in significant radiation-induced effects in agrocenoses. This is realized on an example of cultivated plants which are one of the main components of agroecosystems since cultured plants not only contribute essentially to food production but also fulfil an important ecological function in agrarian biocenosis.

Material and methods

Available information on radiation-induced effects in cultured plants are being gathered from papers issued mainly in Russian scientific press during last 50 years; the work is ongoing. Data are maintained as database in MS Access that now contains about 2000 records from 134 original sources. There are available about 7000 entries which are pairs of "exposure level – biological effect" data accompanied with information on experimental design, species and its life stage, exposure parameters and conditions. Most information collected concern radiation effects in crops, vegetables and roots, as well as legumes – 39%, 22% and 17% of all entries, correspondingly.

Biological responses are grouped under umbrella endpoints of reproduction, survival, morbidity, morphological changes, biochemical changes, and (cyto)genetic effects. Only data on reproduction and survival endpoints are presented here. The first group pools traits of biological productivity (seed and straw mass, dry and wet biomass, number of plants per 1 m^2, etc) and reproductive potential (portion of fertile pollen seeds, fertility/sterility, number or portion of male and female flowers, etc). The later group indexes are germination rate (in field or in laboratory), survival of germinated seeds or plants by certain time/day, etc. To stanardize the data sets, biological responses are expressed as percentages of corresponding controls, which makes possible analysing unified arrays formed of data obtained from different original sources.

Reproduction and survival data are estimated for three possible scenarios of radiation impact on cultivated plants: acute exposure of seeds, acute or chronical exposure of plants during vegetation. Following an approach proposed by (Garnier-Laplace et al 2006, 2008), the dose giving 50% change in observed effect (ED_{50}) is considered as the critical dose for acute exposure, and the critical dose rate for chronic exposure is defined as the dose rate resulting in 10% change of observed effect (EDR_{10}). Dose(rate)-response dependences are reconstructed for individual species using the linear model as the simplest approximation. An assessment of uncertainties in estimated critical doses is illustrated in Fig. 1 on an example of data on reproduction in wheat after seeds' acute exposure. The main effect is fitted with a linear dose-effect function. For every dose value, a 95% confidence interval (CI) for predicted response is calculated (Draper & Smith 1981). A CI width is not constant but increases along with shifting up or down from the median dose. Dose dependences for the upper and down margins of CIs (dotted lines in Fig. 1) are fitted with linear-quadratic model, and their intersections with the 50% effect level (dashed line in Fig. 1) are found. Thus, for the example in Fig. 1 the ED_{50} = 182.3 Gy, 95% CI – 167.3÷202.8 Gy. In several cases the linear-quadratic fit of the upper CI margin always is over the 50% effect level. Then, a linear model is used to fit the CI margins; these cases are marked below.

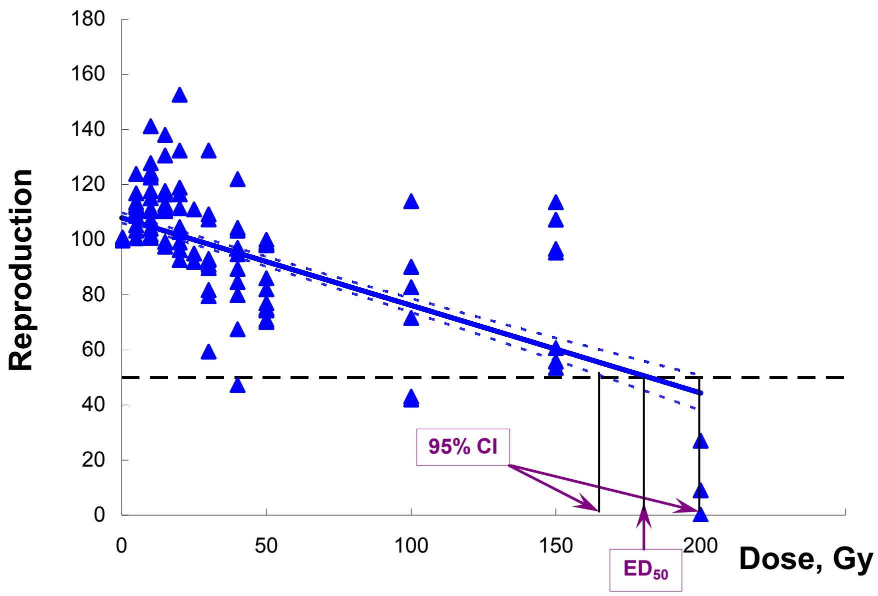

Fig. 1. Standardized data (% of control) on reproduction in wheat after seeds' acute exposure, and estimation of the ED_{50} and its uncertainty.

Results

Acute exposure of seeds

Information on acute exposure of seeds, bulbs, tubers, and other planting stock for cultivated plants is respectively well represented in Russian scientific literature because of a period of radiobiological researches when a pre-sowing irradiation was considered as a possible mean to increase harvest, resistance to diseases and other economical benefits for farming. In the database, 1124 records (4028 pair entries) refer to this exposure scenario which is about 67% of data used to reconstruct dose dependences. A short description of corresponding data collected so far including maximum dose used in original works and numbers of records and paired entries for crops, legumes, vegetables and other cultivated plant species are presented in Tables 1 for reproduction and survival endpoints. Fig. 2 illustrates data on reproduction in crops and their linear fittings. For most species, a linear model fits the data well ($p<5\%$).

On reproduction, carrot appears to be the most radioresistent (ED_{50}=1244.6 Gy). The radiosensitive species are onion and potato as an acute exposure of tubers and bulbs before germination to estimated critical doses of 33.8 Gy and 66.6 Gy, correspondingly, would result in the 50% loss of their productivity. Legumes are also quite sensitive, since bean reproduction reduces twice after 57.0 Gy of acute seeds exposure. Crops show intermediate radiosensitivity with ED_{50}=180-350 Gy.

On survival, carrot is most resistant again (ED_{50}=902.5 Gy). Cotton plants are also very resistant at seeds irradiation (ED_{50}=629.7 Gy). Wheat and corn show the level of radiosensitivity (ED_{50}=222-230 Gy) similar to that on reproduction. There are obtained the very low estimated critical doses for barley and rye (the 95% CIs are within 14÷24 Gy); it is because of very little information taken from a single paper only, and a narrow dose range studied – only up to 5 Gy, which is completely insufficient. For other cultivated plants data volumes on survival indexes are also very limited (N<10, Table 1) which often results in poor fitting (p>10%).

Table 1. Critical doses ED_{50} for reproduction and survival at acute exposure of planting stock.

Species	Reproduction					Survival				
	D_{max}, Gy	N	N_p	F	ED_{50} (95% CI), Gy	D_{max}, Gy	N	N_p	F	ED_{50} (95% CI), Gy
Crops										
Barley	200	20	84	0.5	NE	5	6	12	5.9**	16.5 (13.6÷20.8)
Wheat	200	26	126	86.0***	182.3 (167.3÷202.8)	400	18	70	74.7***	229.8 (211.9÷250.5)
Rye						5	6	18	6.9*	23.0 (21.9÷24.0)[l]
Corn	100	8	55	14.2***	349.5 (317.5÷387.1) [(l)]	300	18	45	71.3***	221.9 (205.4÷241.6)
Legumes										
Bean	50	4	20	48.8***	57.0(51.5÷65.2)					
Peas	80	4	20	3.3	177.2 (146.8÷213.2)[(l)]	80	9	48	68.3***	80.6 (74.2÷89.2)
Soya	2	2	4	169.0***	NE					
Vegetables and roots										
Carrot	1500	2	14	39.4***	1244.6 (1108.9÷1435.4)	2000	16	140	97.5***	902.5 (834.9÷976.0)
Cucumber	400	16	58	41.9***	375.2 (339.8÷423.0)	400	8	36	0.4	NE
Onion	5	5	30	5.0*	33.8 (28.9÷40.4)[(l)]	10	2	14	1.6	30.2 (21.3÷46.0)[(l)]
Tomato	30	76	182	15.4***	NE					
Pepper	18	8	16	32.4***	NE					
Beet	50	6	26	0.6	NE	50	6	54	0.4	240.9 (181.8÷323.5)
Potato	30	16	67	6.7*	66.6 (56.2÷79.9)[(l)]	20	2	9	0.1	118.0 (45.9÷NE)[(l)]
Others										
Cotton	300	82	290	140.7***	462.1 (421.9÷519.7)	800	36	114	14.0***	629.7 (520.1÷838.5)
Sunflower	200	36	102	56.7***	444.6 (378.0÷655.5)					
Tobacco						150	3	11	53.0***	94.7 (86.9÷103.3)

D_{max} – maximum dose; N – number of records; N_p – number of entries presenting pairs of "exposure level – biological effect" data; [(l)] – a linear model is used to fit the CI margins; NE – value could not be estimated; goodness-of-fit on the Fisher test: * - p<10%, ** - p<5%, * - p<1%.**

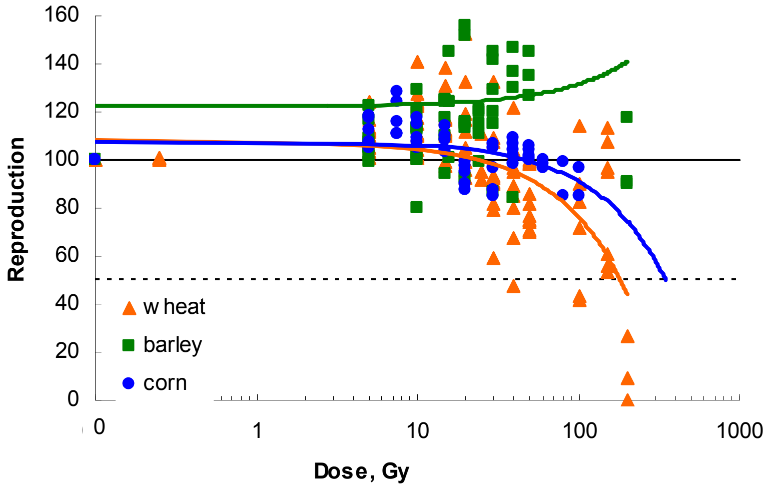

Fig. 2. Standardized data (% of control) on reproduction in crops after acute exposure of seeds and their fittings with a liner model.

For several species (barley, soya, beet, sweet pepper, and tomato at reproduction analysis, and cucumber at survival analysis) critical doses could not be estimated because of a) dose ranges of negative radiation effect are not covered in the works available, or b) large data variance and poor fittings (p>10%).

Despite respectively large in number, data on radiation effects in plants after acute exposure of seeds and other planting stock are not well suited for a purpose of a critical radiotoxicity values estimation, since original researches were mainly aimed at studying hormetic effects of radiation and often do not cover doses producing negative effects in plants. Thus, a better reproduction or lower survival in comparison to a control is shown by about 37% of the database entries refered to the scenario of seeds exposure. Stimulation could reach up to 500%. Although from the viewpoint of developing safety standards at radiation exposure of biocenoses, biological effects of low doses are of special interest, however, a correct interpretation of this kind of data is possible only when detailed information on radiation effect in the whole dose range – from stimulating to harmful – is available.

Acute exposure of plants during vegetation

Studies of radiation effects at total or partial exposure of plants during ontogenetic development are not very large in number. There are certain limitations for such researches such as an availability of special equipment at γ-fields or in greenhouse experiments, providing homogeneous radiation fields, dose and dose rate meausrements, etc. At the moment, reproduction in the database is presented by 330 pair entries that mainly refer to radiation effect in crops (Table 2); survival was studied only in one work with cherry tree cuttings' irradiation. Study design in these works was more homogeneous than at acute seeds exposure. Mostly, γ-sources are used, and dose ranges studied are similar. Radiosensitivity of plants much depends on a stage of ontogenesis at the moment of irradiation, but unfortunately, the available data are not enough to analize dose dependences for an every development stage. Radiation resulted in decreasing of reproduction and survival in all studied species excluding apple tree, where acute exposure of pollen produced a better ability of trees to set fruits and seeds.

For reproduction data, the linear fitting appears good for all species (Table 2). The less sensitive species is potato (ED_{50}=45.8 Gy) in the first and second generation after parent plants' acute exposure at different stages of ontogenesis. Cotton plants also shows high resistence with the estimated critical dose of 36.8 Gy. In crops variety, an assessment of doses leading to 50% loss of productivity gives the similar values (Fig. 3, Table 2). The most and the less radiosensitive are barley (95% CI = 14.6÷16.6 Gy) and rye (95% CI = 25.7÷37.5 Gy), respectively.

Table 2. Critical doses ED_{50} and dose rates EDR_{10} for reproduction and survival at acute and chronic exposure of vegetating plants.

Species	Acute exposure					Chronic exposure				
	D_{max}, Gy	N	N_p	F	ED_{50} (95% CI), Gy	DR_{max}, mGy/h	N	N_p	F	EDR_{10} (95% CI), mGy/h
Reproduction										
Crops										
Barley	30	46	112	112.1***	15.6 (14.6÷16.6)	388.8	11	46	8.2***	227.3 (173.3÷310.7)
Oats	45	10	40	18.7***	27.8 (24.4÷32.5)					
Rye	35	5	10	17.8***	30.4 (25.7÷37.5)	0.3	1	6	0	NE
Wheat	60	31	94	46.4***	18.3 (16.4÷20.4)	75.0	32	108	21.9***	6.8 (3.3÷10.0)
Others										
Apple	90	9	26	4.3**	NE					
Black-currant						127.0	5	11	7.4**	5.9 (0÷25.7)
Cherry						1.5	2	6	22.6***	NE
Cotton	20	5	8	34.9***	36.8 (32.3÷48.2)					
Grape						16.0	3	9	1.5	3.2 (0÷12.2)$^{(I)}$
Peas						33.3	7	33	16.0***	8.6 (5.2÷11.5)
Potato	30	23	40	45.0***	45.8 (37.8÷42.7)	123.0	2	16	19.9***	11.9 (1.9÷21.1)
Survival										
Cherry	30	9	2	2.3	57.1 (40.3÷84.1)$^{(I)}$					
Peas						500.0	2	10	32.7***	158.6 (125.0÷190.7)
Rye						0.3	1	6	3.3	0.24 (0.16÷0.35)
Wheat						0.2	4	12	7.9**	NE

DR_{max} – maximum dose rate. Other designations and marks are the same as in Table 1.

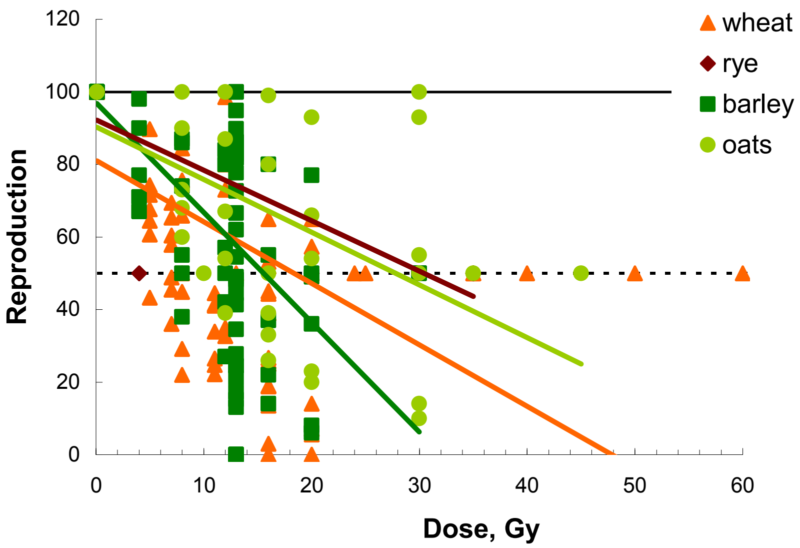

Fig. 3. Standardized data (% of control) on reproduction in crops after acute exposure of vegetating plants and their fittings with a liner model.

Chronic exposure of plants during vegetation

Works on chronic exposure of vegetating plants are the most scarce. In the database, only 63 and 7 records refer to reproduction and survival, respectively, for this exposure scenario (Table 2). Experimental design varies: plants' exposure in γ-field, soil contamination with radionuclides in controlled conditions or growing plants at territories contaminated in radiation accidents. Dose and dose rates range up to 4-5 orders of magnitude – from 10 mGy to 20 Gy, and from 1 μGy/h up to 500 mGy/h. But, the case of chronic exposure is the most interesting in terms of establishing the critical radiotoxicity values. Data on reproduction indexes gathered for cultivated plants are shown in Fig. 4. To fit dose rate – effect dependence, a linear model is used as before. The estimated critical dose rates giving the 10% decrease in biological response are presented in Table 2. In most species, chronic exposure decreases reproduction and survival (Fig. 4). Stimulating effect is observed in rye and cherry trees on reproduction, and in wheat on survival, but data volumes in these cases are limited.

For reproduction endpoint, the least value of EDR_{10}=3.2 mGy/h is obtained for grape in which the number of fruiting brunches decreased after two-year growing at γ-field. In crops, the critical dose rates for wheat (EDR_{10}=3.3÷10 mGy/h) and barley (EDR_{10}=173.3÷310.7 mGy/h) differ by a factor of 30. Possible reasons could be different study designs and incomplete dosemetric information. Thus, wheat plants were exposed in γ-field while barley data refer to both external irradiation in γ-field and mixed exposure from ^{90}Sr-contaminated soil. In the last case, internal doses were not estimated. The other species are close in radiosensitivity and the estimated critical dose rates range from 5.9 to 11.9 mGy/h (Table 4).

Survival indexes are taken from 3 papers, but neither of them gives a good example of a study on chronic radiation effect in vegetating plants. For rye, data on a germinating ability of seeds collected in August, 1986 from the 10-km zone of the Chernobyl NPP are used. The estimated critical dose rate of EDR_{10}=0.24 mGy/h is very low. To understand this data properly, it should be taken into that the parent plants not only experienced chronic radiation with dose rate dramatically changing during the whole period of vegetation, but also they were acutely irradiated in the initial period after the Chernobyl accident (May-June). At the moment of harvesting, contamination density of soils with ^{137}Cs at the study sites reached 800 MBq/m^2. So, this case should not be used to estimate derived radiation levels of chronic exposure. For peas, data are considered on the 24-day survival of seedlings after 12-hour imbibition of germinated seeds in ^{90}Sr solution, and EDR_{10}=158.6 mGy/h is obtained. However, duration of radiation impact is too short in comparison to life time to consider these data as useful for establishing critical radiotoxicity values.

In total, there is a lack of acceptable information on effects of chronic radiation in cultivated plants. The data available are few and scarce, methods and study designs are very different, dosemetric information is incomplete. A part of studies do not contain calculated and measured doses and dose rates. In some works, only data on radioactive contamination are presented (radionuclide activity in soil or plant tissues, contamination density, etc). Internal exposure is rarely taken into account at dose assessments.

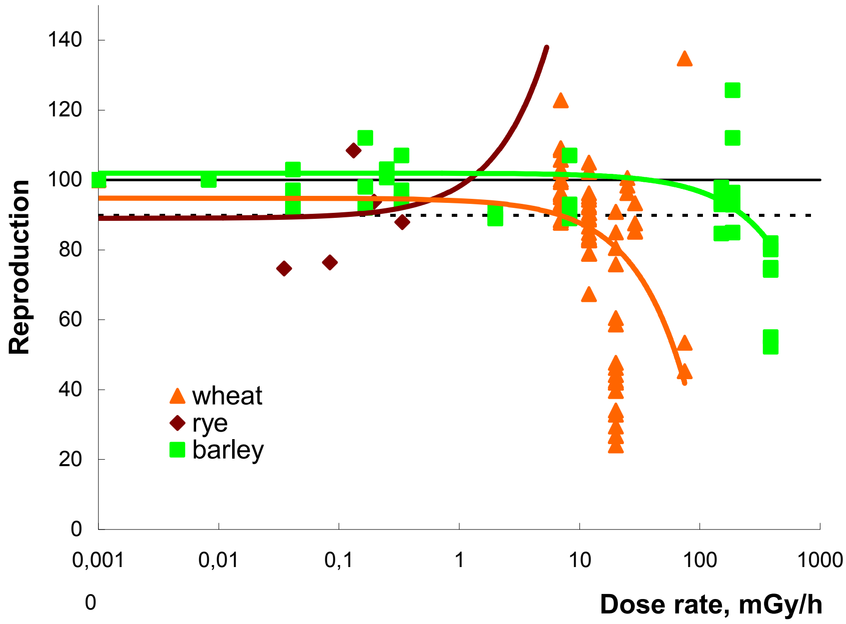

Fig. 4. Standardized data (% of control) on reproduction in crops after chronic exposure of vegetating plants and their fittings with a liner model.

Discussion

The estimated critical doses and dose rates can be used to derive the predicted no-effect dose (PNED) and dose rate (PNEDR). The lowest short-term ED_{50} and long-term EDR_{10}, which characterize radioresistance of the most sensitive species on the most affected umbrella endpoint, can be considered as the first approximation for the PNED and PNEDR in argocenosis. However, the quality and quantity of the primary data influence the safety of the estimated levels. Datasets collected from the original sources are very heterogeneous both in terms of materials and methods used and the data volumes available. To provide a certain confidence in the PNED and PNEDR values, an additional requirement of a minimal number of records N≥10 is applied to select the 'well-defined' values of ED_{50} and EDR_{10}. In Table 3 the minimal ED_{50} and EDR_{10}'s are presented that fullfill the agreed requirement for the three radiation scenarios.

Table 3. Critical doses and dose rates for the most sensitive species for different scenarios of radiation impact on cultivated plants, and the corresponding PNED and PNEDR values.

Exposure scenario	Reproduction	Survival	PNED(R)
Acute exposure of seeds	ED_{50} = 182.3 (167.3÷202.8) Gy at wheat seeds exposure ED_{50} = 66.6 (56.2÷79.9) Gy at potato tubers exposure	ED_{50} = 221.9 (205.4÷241.6) Gy at corn seeds exposure	170÷200 Gy for seeds 67÷80 Gy for other planting stock
Acute exposure of vegetating plants	ED_{50} = 15.6 (14.6÷16.6) Gy for barley	No data	15÷17 Gy
Chronic exposure of vegetating plants	EDR_{10} = 6.8 (3.3÷10.0) mGy/h for wheat	No data	3÷10 mGy/h

Minimal ED_{50} and EDR_{10} are chosen from Tables 1 and 2 following the requirement of N≥10.

From the above consideration it is obvious that data on survival collected so far are rather insufficient to estimate the critical radiotoxicity values. Radiosensitivity of two umbrella endpoints analysed (reproduction and survival) could be compared only for the case of acute seeds exposure (Table 3). According to the findings of this work, reproduction is more sensitive criteria than survival. It is well known that half-lethal doses often result in the full loss of productivity for most domesticated plants (Agricultural radioecology 1991). In terrestrial plants, the dose reducing survival by 10% is roughly equivalent to the dose reducing the yield by 50% (UNSCEAR 1996). (UNSCEAR 2008) also considers that reproductive changes are a more sensitive indicator of radiation effect than mortality. So, reproduction endpoint is used to derive PNED(R)s in this study. Consequently, at acute exposure of the most radiosensitive species in dormant and vegetation periods the PNEDs for agrocenoses amounts to 67÷80 and 15÷17 Gy, correspondingly, according to the data gathered so far. At chronic exposure, the PNEDR is within 3÷10 mGy/h.

There is a number of uncertainties that influence the estimates derived. They emerge from pooling field and laboratory based studies, different ontogenetic phases, various biological effects under one umbrella endpoints, etc. There are also some shortages in dose (rate) determination and data statistical confidence for certain datasets. Typically, the safety factors method is applied to assure a security of finally derived safe levels of radiation impact.

Authorized international organisations proposed guideline dose rates below which a population level effect in wildlife is unlikely to be induced. According to the last recommendations (UNSCEAR 2008) chronic dose rates of less than 0.1 mGy/h for terrestrial communities and acute exposure of non-human biota at doses below about 1 Gy would be unlikely to have significant effects. The PNEDR equal to 10 μGy/h was derived with the ERICA Integrated Approach (Garnier-Laplace et al. 2008). At this, mammals are regarded as the most sensitive of all non-human organisms. Taking into account that cultured plants discussed are commonly more resistant than mammals as well as the further lowering of the estimated PNED(R) due to their dividing by a safety factor, the findings of this work can be considered non-contradictory to the existing guidelines.

The estimates presented here are the first derivation of PNED(R)s for agrarian biocenoses. They are going to be improved with further development of the database and methods for data treatment. In particular, other endpoints are expected to be analysed, and different fitting models, including logistic, would be applied.

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Preliminary results on Cernavoda NPP operation impact on terrestrial and aquatic biota

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Abstract

Recently, the awareness of the vulnerability of the environment has increased and the need to protect it against industrial pollutants has been recognized. The concept of sustainable development, requires new and developing international policies for environmental protection. (Protection of the environment from the effects of ionizing radiation. IAEA-TECDOC-1091. International Atomic Energy Agency, Vienna.). As it is recommended in "Cernavoda Unit #2 NPP Environmental Impact Assessment it is Cernavoda NPP responsibility to conduct an Ecological Risk Assessment study, mainly to assess the impact of Nuclear power plant operation on terrestrial and aquatic biota. Long records from normal operation of Cernavoda Unit 1, wind pattern, meteorological conditions, and upgread source terms data were used to evaluate areas of interest for environmental impact, conducting to a circle of 20 km radius around mentioned nuclear objective. The screening campaign established tritium level (because Cernavoda NPP is a CANDU type reactor, and tritium is the most important radioisotope evacuated in the environment) in air, water, soil and vegetation, focusing the interest area on particular ecosystem. Using these primary data it was evaluated which are the monitored ecological receptors and which are the measurement endpoints. This paper presents the Ecological Risk Assessment at Cernavoda NPP technical requirements, and the preliminary results of evaluating criteria for representative ecosystem components at Cernavoda NPP.

Update of impacts of the Chernobyl accident: Assessments of the Chernobyl Forum (2003–2005) and UNSCEAR (2005–2008)

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Abstract

The accident at the Chernobyl NPP in Ukraine in 1986 was the most severe in the history of the nuclear industry, causing a huge release of radionuclides over large areas of Europe. Its radiological consequences were recently revisited by the UN Chernobyl Forum (2003-2005) and UNSCEAR (2005-2008).

For the first time environmental impacts were considered in detail, including radioactive contamination of terrestrial and aquatic environments, application and effectiveness of countermeasures and effects on biota. Updated dosimetric data were presented for more than half a million of emergency and recovery operation workers, about 100 million inhabitants of the three most affected countries, Belarus, the Russian Federation and Ukraine, and for 500 million inhabitants of other European countries. The average effective dose of the emergency and recovery operation workers was estimated to be about 0.12 Sv. An exception is a cohort of several hundred emergency workers who received high radiation doses; of whom 28 persons died in 1986 due to acute radiation sickness.

The majority of the six million residents of the 'contaminated areas' in Belarus, Russia and Ukraine received relatively minor radiation doses which are comparable with the natural background levels. However, those who were children at the time and drank milk with high levels of radioactive iodine received during the first few months after the accident particularly high doses to the thyroid. Since early 1990s there was the dramatic increase in thyroid cancer incidence among those exposed to radioiodine at a young age. Also in 1990s there was some increase of leukaemia in most exposed workers.

Apart from those two health effects revealed in the two different population cohorts as the result of 20-year epidemiological observations there was, until 2007, no clearly demonstrated increase in the somatic diseases due to radiation. There was, however, an increase in psychological problems among the affected population, compounded by the social disruption that followed the break-up of the Soviet Union. In particular, the UN Chernobyl Forum concluded that after a number of years, along with

reduction of radiation levels and accumulation of humanitarian consequences, severe social and economic depression of the affected regions and associated psychological problems of the general public and the workers had become the most significant problem to be addressed by the national authorities.

Introduction

The Chernobyl accident was the most severe in the history of the world nuclear industry. At night of 26 April 1986, Unit 4 of the Chernobyl nuclear power plant, located 130 km to the north-east of Kiev, the capital of Ukraine[1], was destroyed by two powerful explosions in the reactor core. The Chernobyl NPP was equipped with four RBMK reactors with a graphite moderator, a thermal power of 3200 MW and an electrical power of 1000 MW each. The explosions were caused by gross breaches of the operating procedures by staff and technical inadequacies in the safety systems (INSAG 1993). As a result of the explosions, highly radioactive core fragments were ejected onto the site. The hot graphite exposed to air caught fire and burned for 10 days.

During that time period, radioactive substances were ejected from the burning reactor and spread by winds under changing weather conditions over Europe, principally Belarus, Ukraine and Russia. No more than 20% of the radioactive discharge spread beyond Europe (De Cort et al. 1998).

More than 400,000 emergency and recovery operation workers, including army, power plant staff, local police and fire services, were involved in containing and cleaning up the accident in 1986-1987. Later, the number of registered “liquidators” rose to about 600,000.

About six million people live in areas of Belarus, Russia and Ukraine that are ‘contaminated with radionuclides’ due to the Chernobyl accident (above 37 kBq m^{-2} or 1 Ci km^{-2} of ^{137}Cs)[2]. Amongst them, about 400,000 people lived in more contaminated areas – classified at the time by Soviet authorities as ‘areas of strict radiation control’ (above 555 kBq m^{-2} or 15 Ci km^{-2} of ^{137}Cs). Of this population, 115,000 people were evacuated in the spring and summer of 1986 from the area surrounding the Chernobyl power plant (designated the “Exclusion Zone”) to non-contaminated areas. Another 220,000 people were relocated in subsequent years (UNSCEAR 2000).

The consequences of the Chernobyl accident were widely discussed at the Kiev conference in 1988 (IAEA 1989), pursuant to the results of the IAEA Chernobyl project (IAG 1991), and at the conferences marking its 10th anniversary (Karaoglou et al. 1996; IAEA 1996). The health consequences were analysed comprehensively by UNSCEAR in its 1988 and 2000 reports (UNSCEAR 1988, 2000). In 2005-2008 UNSCEAR has undertaken further assessment of the Chernobyl environmental and health consequences and recently published the resulting report (UNSCEAR 2010).

Because of continued contradictions in the interpretation of the Chernobyl accident consequences, the IAEA initiated in early 2003 establishing the Chernobyl Forum aiming to retrospectively assess the environmental and health consequences of the accident and to advise the Governments of Belarus, Ukraine and the Russian

[1] Up until 1991, Belarus, Russia and Ukraine were parts of the USSR.

[2] In the mapping of the deposition, ^{137}Cs was chosen because it is easy to measure long-lived radionuclide, and it is of radiological significance.

Federation on future actions, such as environmental remediation and special health care as well as research activities.

The Forum participants were eight United Nations organisations (IAEA, WHO, UNDP, FAO, UN-OCHA, UNEP, UNSCEAR[3] and The World Bank) as well as the competent authorities of Belarus, Russia and Ukraine. The Forum was created as a contribution to the United Nations' ten-year strategy for Chernobyl, launched in 2002 with the publication of *Human Consequences of the Chernobyl Nuclear Accident – A Strategy for Recovery* (UNDP and UNICEF 2002). The Chernobyl Forum and subsequent conference were chaired by Dr. Burton Bennett, Radiation Effects Research Foundation, Japan.

Forum reports on environment (IAEA 2006) and health (WHO 2006) were prepared by corresponding expert working groups and approved by consensus at the last Forum meeting in April 2005. The reports were presented and discussed during the International Conference entitled "Chernobyl: Looking Back to Go Forwards" organised by the IAEA in Vienna in September 2005 (IAEA 2008).

In November 2005, the United Nations General Assembly considered a report on efforts to promote recovery in areas affected by the Chernobyl legacy and adopted a resolution A/60/L.19 (UN 2005) in which, *inter alia*, noted consensus reached among members of the Chernobyl Forum regarding assessment of the accident consequences and future actions.

This paper presents the recent findings of the Chernobyl Forum and UNSCEAR on radiological health and environmental consequences of the Chernobyl accident.

Environmental Consequences (IAEA 2006, UNSCEAR 2010)

Radionuclide release and deposition

Major releases of radionuclides from Unit 4 of the Chernobyl NPP continued for ten days following the April 26 explosion. These included radioactive gases, condensed aerosols and a large amount of fuel particles. The total release of radioactive substances was about 14 EBq, including 1.8 EBq of iodine-131, 0.085 EBq of ^{137}Cs, 0.01 EBq of ^{90}Sr and 0.003 EBq of plutonium radioisotopes. The noble gases contributed about 50% of the total release. The most up-to-date estimates of the amounts released are similar to those of the UNSCEAR 2000 Report (UNSCEAR 2000), except for the refractory elements, which are now about 50% lower (Kashparov et al 2003).

More than 200,000 square kilometres of Europe received levels of ^{137}Cs above 37 kBq m^{-2}, (De Cort et al. 1998). Over 70 percent of this area was in the three most affected countries, Belarus, Russia and Ukraine. The deposition was extremely varied, as it was enhanced in areas where it was raining when the contaminated air masses passed. Most of the strontium and plutonium radioisotopes were deposited within 100 km of the destroyed reactor due to larger particle sizes.

Many of the most significant radionuclides have decayed away. The releases of radioactive iodines caused great concern immediately after the accident. For the

[3] International Atomic Energy Agency (IAEA), World Health Organization (WHO), United Nations Development Programme (UNDP), Food and Agriculture Organization (FAO), United Nations Environment Programme (UNEP), United Nations Office for the Coordination of Humanitarian Affairs (UN-OCHA), United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR).

decades to come ^{137}Cs will continue to be of greatest importance, with secondary attention to ^{90}Sr. Over the longer term (hundreds to thousands of years) the plutonium isotopes and americium-241 will remain, although at levels that are not significant radiologically.

Environmental transfer

In the early months after the accident, the radionuclide levels of agricultural plants and plant-consuming animals was dominated by surface deposits. The deposition of iodine-131 caused the most immediate concern, but the problem was confined to the first two months after the accident because of ^{131}I decay. The radioiodine was rapidly absorbed into milk leading to significant thyroid doses to people consuming milk, especially children in Belarus, Russia and Ukraine. In the rest of Europe increased levels of ^{131}I in milk were observed in some southern areas, where dairy animals were already outdoors.

After the early phase of direct deposit, uptake of radionuclides through plant roots from soil became increasingly important. Radioisotopes of caesium (^{137}Cs and ^{134}Cs) were the nuclides which led to the largest problems. The radiocaesium content in foodstuffs was influenced not only by deposition levels but also by types of ecosystem and soil as well as by management practices. In addition, ^{90}Sr could cause problems in areas close to the reactor, but at greater distances its deposition levels were low. Other radionuclides such as plutonium isotopes and ^{241}Am did not cause real problems in agriculture, both because of low deposition levels and poor availability for root uptake from soil.

In general, there was a substantial reduction in the transfer of radionuclides to vegetation and animals in intensive agricultural systems in the first few years after deposition, as would be expected due to weathering, physical decay, migration of radionuclides down the soil, reductions in bioavailability in soil and due to countermeasures. However, in the last decade there has been little further obvious decline, by 3-7 percent per year.

Currently, ^{137}Cs activity concentrations in agricultural food products are generally below national and international action levels. However, in some limited areas with high radionuclide deposition (parts of the Gomel and Mogilev regions in Belarus and the Bryansk region in Russia) or poor organic soils (the Zhytomir and Rovno regions in Ukraine) milk may still be produced with ^{137}Cs activity concentrations that exceed national action levels of 100 Bq kg^{-1}. In these areas countermeasures and environmental remediation may still be warranted.

The uptake and retention of ^{137}Cs has generally been much higher in semi-natural ecosystems than in agricultural ecosystems, and the clearance rate from forest ecosystems is extremely slow. The highest levels in foodstuffs continue to be in mushrooms, berries, game and reindeer.

Levels of radionuclides in rivers and lakes directly after the accident fell rapidly and are now generally very low in water used for drinking and irrigation, although the radiocaesium levels in the water and fish of some closed lakes have fallen only slowly. Levels in seawater and marine fish were much lower than in freshwater systems.

The deposition in urban areas could have initially given rise to a substantial external dose. However, due to radioactive decay, wind, rain and human activities, including traffic, street washing and cleanup, surface contamination by radioactive materials and air dose rate has been reduced significantly in inhabited and recreational areas since 1986, Fig. 1.

At present, in most of the settlements subjected to radioactive contamination as a result of Chernobyl, the air dose rate above solid surfaces has returned to the background level predating the accident. But the air dose rate remains elevated above undisturbed soil in gardens and parks in some settlements of Belarus, Russia and Ukraine.

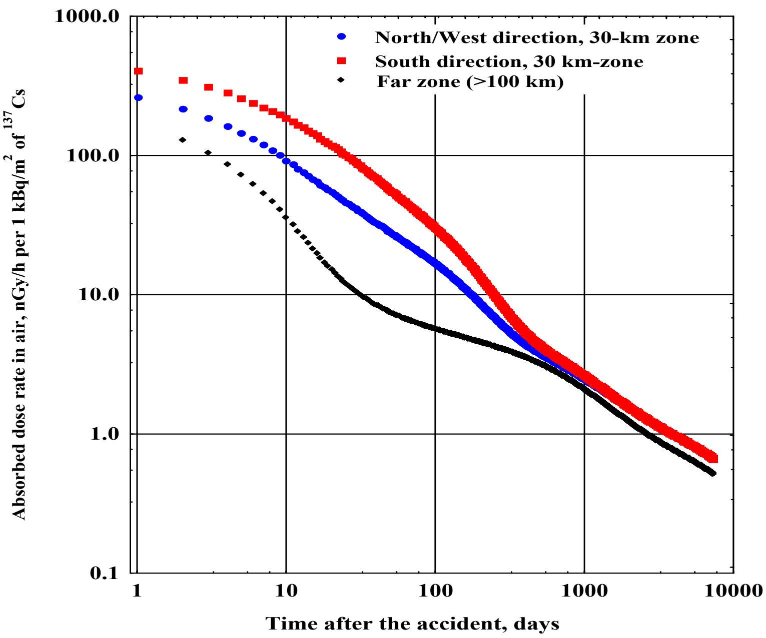

Fig. 1. Dynamics of standardised dose rate in air over undisturbed soil after the Chernobyl accident in different geographical areas (Golikov 2006).

Environmental countermeasures

The Soviet authorities introduced a wide range of short- and long-term environmental countermeasures to mitigate the accident's consequences. The countermeasures involved huge human, financial and scientific resources.

Decontamination of settlements in the affected regions of the USSR during the first years after the Chernobyl accident was successful in reducing the external dose when its implementation was preceded by proper remediation assessment. However, the decontamination has produced a disposal problem due to the considerable amount of low-level radioactive waste that was created.

In the first few weeks, management of animal fodder and milk production (including prohibiting the consumption of fresh milk) would have helped significantly to reduce the doses to the thyroid due to radioiodine. However, wide implementation of early countermeasures in the former Soviet Union was flawed, because timely advice was lacking, particularly for private farmers. Many European countries changed their agricultural practices and/or withdrew food, especially fresh milk, from the supply chain, and, in Poland, iodine prophylaxis was promptly organized; these actions generally reduced thyroid doses in those countries to negligible levels.

Over the months and years after the accident, the authorities of the former Soviet Union introduced an extensive set of agricultural countermeasures. These helped to reduce the long-term exposures from the long-lived radionuclides, notably radiocaesium. During the first few years, substantial amounts of food were removed from human consumption because of concerns about the radiocaesium levels, especially in milk and meat. In addition, pasture was treated, and clean fodder and caesium binders were provided to livestock, resulting in considerable reductions in dose.

In addition, countermeasures were instigated to reduce exposures from living and working in forests and using forest products. They included: restrictions on access; restrictions on harvesting of forest foods, such as game, berries and mushrooms; restrictions of the gathering of firewood; and alteration of hunting practices.

Early restrictions on drinking water and changing to alternative supplies reduced internal doses from aquatic pathways in the initial period. Restrictions on the consumption of freshwater fish from some lakes also proved effective in Scandinavia and Germany. Other countermeasures to reduce the transfer of radionuclides from soil to water systems were generally ineffective.

Radiation-induced effects on plants and animals

Radiation of radionuclides released from the accident caused numerous acute adverse effects on the plants and animals living in the higher exposure areas, i.e., in localized sites at distances up to about 30 kilometres from the release point. Outside this area, no acute radiation-induced effects in plants and animals have been reported.

The response of the natural environment to the accident was a complex interaction between radiation dose, radiosensitivity and recovery of the different plants and animals. Both individual and population effects caused by radiation-induced cell death have been observed in biota inside the 30-km area as follows:

- Increased mortality of coniferous plants, soil invertebrates and mammals; and
- Reproductive losses in plants and animals.

Following the natural reduction of exposure levels due to radionuclide decay and migration, biological populations have been recovering from acute radiation effects. As soon as by the next growing season following the accident, population viability of plants and animals had substantially recovered as a result of the combined effects of reparation and repopulation from less affected areas. A few years were needed for recovery from major radiation-induced adverse effects in plants and animals.

Genetic effects of radiation, in both somatic and germ cells, have been observed in plants and animals during the first few years after the Chernobyl accident. Different cytogenetic anomalies attributable to radiation continue to be reported from experimental studies. Whether the observed cytogenetic anomalies in somatic cells have any detrimental biological significance is not known.

The recovery of affected biota in the exclusion zone has been facilitated by the removal of human activities, e.g., termination of agricultural and industrial activities. As a result, populations of many plants and animals have eventually expanded, and the present environmental conditions have had a positive impact on the biota in the 30-km area. Indeed, this area has paradoxically become a unique sanctuary for biodiversity.

Radiation doses to exposed population groups (IAEA 2006, UNSCEAR 2010)

The following population categories were exposed from the Chernobyl accident:

- Emergency and recovery operation workers who worked at the Chernobyl power plant and in the exclusion zone after the accident;
- Inhabitants evacuated from abandoned areas of Belarus, Russia and Ukraine; and
- Inhabitants of areas with radioactive fallout, who were not evacuated.

With the exception of the on-site reactor personnel and the emergency workers who were present near the destroyed reactor during the time of the accident and shortly afterwards, most of recovery operation workers and people living in the contaminated territories received relatively low whole-body radiation doses, comparable to background radiation levels accumulated over the 20 year period since the accident (Table 1). For comparison, the annual average effective dose from natural background radiation is 2.4 mSv.

Table 1. Summary of updated dose estimates for the main population groups exposed from the Chernobyl fallout

Population group	Size (thousands)	Average thyroid dose in 1986 (mGy)	Average effective dose [b] in 1986-2005 (mSv)	Collective thyroid dose in 1986 (man Gy)	Collective effective dose [b] in 1986–2005 (man Sv)
Recovery operation workers (1986-1990) [a]	530	NA	117 [a]	NA	61 200
Evacuees (1986)	115	490	31	57 000	3 600
Inhabitants of areas of strict radiation control [c]	216	NA	61	NA	13 100
Inhabitants of contaminated areas [c]	6 400	102	9	650 000	58 900
Inhabitants of Belarus, Russia (19 regions) and Ukraine	98 000	16	1.3	1 600 000	125 000
Inhabitants of distant European countries [d]	500 000	1.3	0.3	660 000	130 000

NA – Data not available.

a Effective dose estimates for the workers include only the doses from external irradiation.

b Effective dose estimates are the sum of the contributions from external and internal irradiation, excluding the thyroid dose.

c The contaminated areas were defined in the former Soviet Union as areas where the ^{137}Cs levels on soil were greater than 37 kBq/m^2 and areas of strict radiation control - where the ^{137}Cs levels were greater than 555 kBq/m^2.

d All the European countries except the three republics, Turkey, countries of the Caucasus, Andorra, and San Marino.

Compared to the UNSCEAR 2000 report: (a) dose estimates have been updated for a larger number of the Belarusian, Russian, and Ukrainian recovery operation workers (510,000 instead of 380,000), and new information is presented on the Estonian, Latvian, and Lithuanian recovery operation workers; (b) thyroid dose estimates have been updated for the Belarusian and Ukrainian evacuees, and new information is presented for the Russian evacuees; (c) the estimation of thyroid and effective doses has been expanded from 5 million to 100 million inhabitants of the three republics; and (d) thyroid and effective dose estimates have been updated for the 500 million inhabitants of other European countries.

The highest doses were received by emergency workers and on-site personnel, in total less than 1,000 people, during the first days of the accident, ranging up to 20 Gy, which was fatal for some of the workers. The doses received by recovery operation workers, who worked for short periods during four years following the accident ranged from less than 10 mSv to more than 1,000 mSv, although about 85% of the recorded doses were in the range 20–500 mSv, with an average of about 117 mSv.

The collective effective dose to the 530,000 recovery operation workers is estimated to have been about 60,000 man Sv. This may, however, be an overestimate, as conservative assumptions appear to have been used in calculating some of the recorded doses.

Effective doses to the persons evacuated from the Chernobyl accident area in the spring and summer of 1986 were estimated to be about 30 mSv on average, mostly from external gamma radiation, with the highest dose of the order of several hundred mSv.

The high thyroid doses among the general population were due almost entirely to drinking fresh milk containing ^{131}I in the first few weeks following the accident. Figure 2 presents the estimated average thyroid dose to children and adolescents in 1986. The average thyroid dose to the evacuees is estimated to have been about 500 mGy (with individual values ranging from less than 50 mGy to more than 5,000 mGy). For the more than six million residents of the contaminated areas of the former Soviet Union who were not evacuated, the average thyroid dose was about 100 mGy, while for about 0.7% of them, the thyroid doses were more than 1,000 mGy. The average thyroid dose to pre-school children was some 2 to 4 times greater than the population average. For the 98 million residents of the whole Belarus and Ukraine and 19 oblasts of the Russian Federation, the average thyroid dose was much lower, about 20 mGy; most (93%) received thyroid doses of less than 50 mGy. The average thyroid dose to residents of the other European countries was about 1.3 mGy.

The collective thyroid dose to the 98 million residents of the former Soviet Union was some 1,600,000 man Gy. At the country level, the collective thyroid dose was highest in Ukraine, with 960,000 man Gy distributed over a population of 51 million people. At the regional level, the highest collective thyroid dose of about 320,000 man Gy was to the population of the Gomel oblast, corresponding to an average thyroid dose of about 200 mGy.

As far as whole body doses are concerned, the six million residents of the contaminated areas of the former Soviet Union received average effective doses for the period 1986–2005 of about 9 mSv, whereas for the 98 million people considered in the three republics, the average effective dose was 1.3 mSv, a third of which was received in 1986. This represents a minor increase over the dose due to background radiation over the same period (50 mSv). About three-quarters of the dose was due to external exposure, the rest being due to internal exposure. About 80% of the lifetime effective doses had been delivered by 2005.

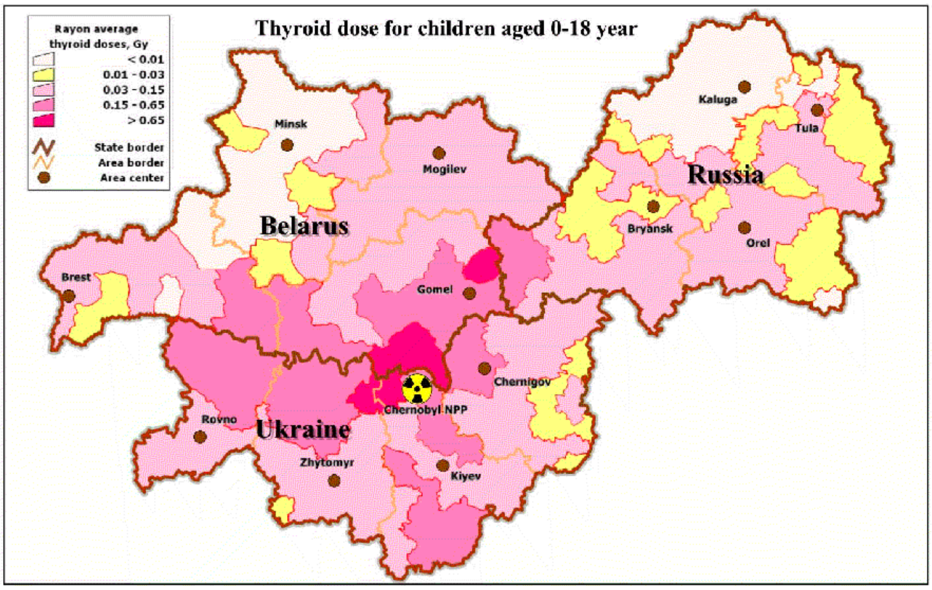

Fig. 2. The estimated district-average thyroid doses to children and adolescents living at the time of the accident in the most affected regions of Belarus, Russia and Ukraine (Russian National Medical and Dosimetric Registry 2006).

However, about 150,000 people living in the most contaminated areas received an effective dose of more than 50 mSv over the 20-year period. For the population of about 500 million in other countries of Europe, the average effective dose is estimated to have been 0.3 mSv over this period. The collective effective dose is estimated at about 125,000 man Sv to the combined populations of Belarus, Ukraine and the relevant parts of Russia, and about 130,000 man Sv to population in the rest of Europe.

Early health effects (UNSCEAR 1988, 2010)

A total of 237 emergency workers were initially examined for signs of ARS that was verified in 134 of these individuals. Of these 134 patients, 28 died within the first four months, their deaths being directly attributable to the high radiation doses.

The dominant exposures were external irradiation of the whole body at high dose rates and beta irradiation of the skin. Internal contamination was of relatively minor importance, while neutron exposure was insignificant. Underlying bone marrow failure from the external whole body irradiation was the major contributor to all the deaths during the first two months.

Each patient with bone-marrow syndrome of Grade III–IV usually also had serious radiation damage to the skin that aggravated other conditions. Skin doses exceeded bone marrow doses by a factor of 10–30, and many ARS patients received skin doses up to 500 Gy. Radiation burns to the skin were felt to be a major contributor to at least 19 of the deaths and significantly increased the severity of the ARS, especially when skin burns exceeded 50% of the body surface area and led to major infections. After 50–60 days, if the skin was not healing, a number of patients received skin graft surgery. In addition, the leg of one patient was amputated more than 200 days after the accident, gastrointestinal syndrome was seen in 15 patients and radiation pneumonitis in 8 patients.

There were no cases of ARS among the general public, either among those evacuated or those not evacuated. This is consistent with the assessment of the radiation exposures, which showed that the whole body radiation doses to members of the general public were much lower than the well-known dose thresholds for ARS.

Late health effects (UNSCEAR 2010, WHO 2006)

ARS survivors

Among the 106 patients surviving ARS, haematopoietic recovery occurred within a matter of few months. However, recovery of the immune system took at least half a year, and complete normalization several years. Cataracts, scarring and ulceration are important ongoing problems in the ARS survivors. Between 1990 and 1996, 15 ARS survivors with extensive skin injuries underwent surgery. Most ARS survivors had suffered functional sexual disorders up to 1996; however, 14 normal children were born to survivor families within the first five years of the accident.

The major health consequences from the radiation exposure of the ARS survivors remain the skin injuries and radiation-induced cataracts. The current nature and severity of the skin injuries depend on their severity during the early period. Patients who had suffered first-degree skin injuries displayed various levels of skin degeneration, ranging from slight smoothing of the skin surface to more pronounced changes. However, over longer periods, the slight changes disappeared almost completely. With third- and fourth-degree injuries, there were areas of scarring, contractures, and radiation-induced ulcers. However, since the early 1990s, microsurgery techniques have significantly reduced the problems of radiation-induced ulcers.

Many of the patients who suffered moderate or severe ARS, developed radiation-induced cataracts in the first few years after the accident, with a strong correlation between the grade of ARS and cataract prevalence.

Over the period 1987–2006, 19 ARS survivors died for various reasons. As time progressed, assignation of radiation as the cause of death has become less clear.

The follow-up of the ARS survivors indicates that: the initial haematological depression has recovered substantially in many patients; there remain significant local injuries; there has been an increase in haematological malignancies; and the increase in other diseases is probably largely due to ageing and other factors not related to radiation exposure.

Thyroid cancer

A substantial increase in thyroid cancer incidence has occurred in the three republics (the whole of Belarus and Ukraine, and the four most affected regions of Russia) since the Chernobyl accident among those exposed as children or adolescents. Amongst those under age 14 years in 1986, 5,127 cases (under age 18 years in 1986, 6,848 cases) of thyroid cancer were reported between 1991 and 2005 (Ivanov et al 2006). A substantial fraction of those cases was caused by internal exposure of thyroids of local residents with incorporated radioiodine.

Figure 3 demonstrates that in Belarus, after the Chernobyl accident in 1986, thyroid cancer incidence rates among children under age 10 years increased dramatically and subsequently declined, specifically for those born after 1986 (see 1996–2005). This pattern suggests that the dramatic increase in incidence in 1991–1995 was associated with the accident. The increase in the incidence of thyroid cancer among children and adolescents began to appear about 5 years after the accident and persisted up until 2005. The background rate of thyroid cancer among children under age 10 years is approximately 2 to 4 cases per million per year.

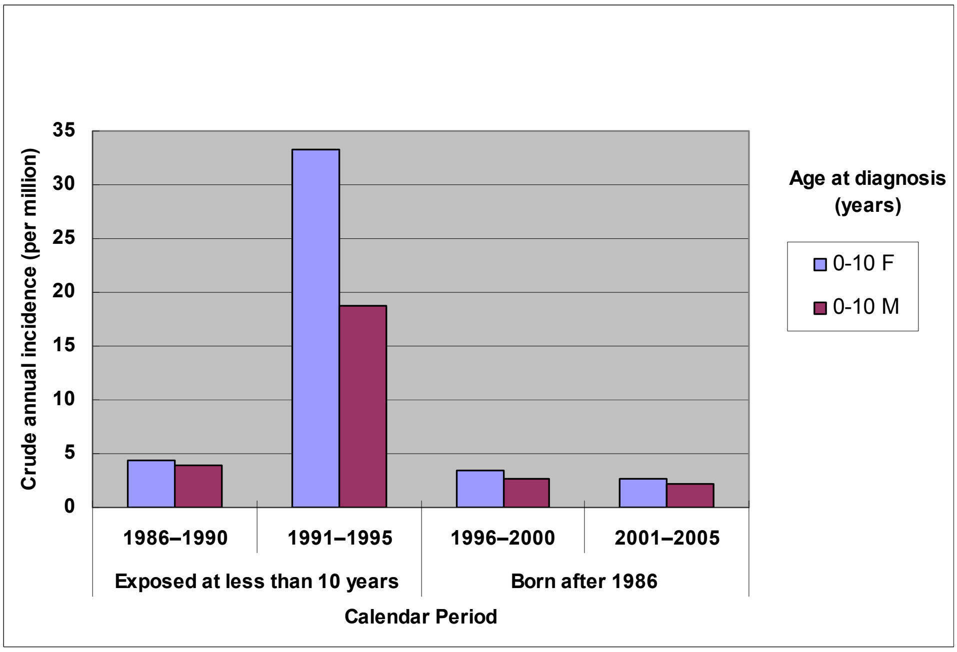

Fig. 3. Thyroid cancer incidence rate in Belarus for children under 10 years old at diagnosis.

This increase has been confirmed in several case-control and cohort studies that have related the excess incidence of thyroid cancer to the estimated individual doses due primarily to the radioiodine released during the accident. There is little suggestion of increased thyroid cancer incidence among those exposed as adults in the general population.

Evidence has also emerged since the UNSCEAR 2000 Report indicating that iodine deficiency might have influenced the risk of thyroid cancer resulting from exposure to the radioactive isotopes of iodine released during the accident (Cardis et al 2005).

Leukaemia, Solid Cancers and Circulatory Diseases

Given the level of doses received, it is likely that studies of the general population will lack statistical power to identify radiation-induced risk of leukaemia, although for higher exposed emergency and recovery operation workers an increase may be detectable. The most recent studies suggest a two-fold increase in the incidence of leukaemia between 1986 and 1996 in Russian emergency and recovery operation workers exposed to external dose of more than 150 mGy. Since the risk of radiation-induced leukaemia decreases few decades after exposure, its contribution to morbidity and mortality is likely to become less significant as time progresses.

There have been many post-Chernobyl studies of leukaemia and cancer morbidity in the populations of 'contaminated' areas in the three countries. So far, there is no convincing evidence that the incidence of leukaemia or cancer (other than thyroid) has increased in children, those exposed *in-utero*, or adult residents. It is thought, however, that for most solid cancers, the minimum latent period is of the order of 10 years or more – and it may be too early to evaluate the full radiological impact of the accident.

There appears to be some recent increase in morbidity and mortality of Russian emergency and recovery operation workers caused by circulatory system diseases. Incidence of circulatory system diseases should be interpreted with special care because of the possible indirect influence of confounding factors, such as stress and lifestyle (smoking, alcohol consumption, etc.)

Cataract

Clinically significant cataracts developed in some of the ARS survivors exposed to high radiation doses. In addition, the recently completed Ukrainian–American Chernobyl Ocular Study (Worgul et al 2007) indicates that lens opacity arising in the recovery operation workers is related to the dose received. A key finding was that the data were not compatible with a dose–effect threshold of more than 0.7 Gy, and that the lower boundary of the estimated dose threshold was close to the current dose limit for the lens of the eye, i.e. 150 mSv, although this needs to be tempered by consideration of the uncertainties in the dosimetry.

It should be noted that the main findings pertain to subclinical cataracts: 96% of observed cases were Grade I opacities. Whether or not some fraction of the radiation-associated Grade I opacities eventually progress to become more severe vision-disabling cataracts remains to be resolved.

Psychological and mental health problems (Chernobyl Forum 2006, WHO 2006)

Stress symptoms, depression, anxiety (including post-traumatic stress symptoms), and medically unexplained physical symptoms have been reported in Chernobyl-exposed populations. The studies also found that exposed populations were more likely to report subjective poor health than were unaffected control groups. In general, the social context in which the Chernobyl accident occurred makes the findings difficult to interpret because of the complicated series of events unleashed by the accident, the multiple extreme stresses and culture-specific ways of expressing distress.

In addition, individuals in the affected populations were officially categorized as "sufferers", and came to be known colloquially as "Chernobyl victims". This label, along with the extensive government benefits had the effect of encouraging individuals to think of themselves fatalistically as invalids. Thus, rather than perceiving themselves as "survivors," many of those people have come to think of themselves as helpless, weak and lacking control over their future.

Other findings of the Chernobyl Forum (Chernobyl Forum 2006, IAEA 2008)

The Chernobyl Forum also considered social and economic consequences of the Chernobyl accident that are out of scope of this paper. The Forum also elaborated practical Recommendations to the Governments of Belarus, the Russian Federation and Ukraine on health care and research, on environmental monitoring, remediation and research and those for economic and social policy that can be found elsewhere (Chernobyl Forum 2006, IAEA 2008).

Conclusions

From this paper based on 20 years of studies, it can be concluded that although those exposed to radioiodine as children or adolescents and the emergency and recovery operation workers who received high doses are at increased risk of radiation-induced effects, the vast majority of the population need not live in fear of serious health consequences from the Chernobyl accident.

Most of the workers and members of the public were exposed to low level radiation comparable to or, at most, a few times higher than the natural background levels, and exposures will continue to decrease as the deposited radionuclides decay or are further dispersed in the environment. This is true for populations of the three countries most affected by the Chernobyl accident, Belarus, the Russian Federation and Ukraine, and all the more so, for populations of other European countries. Lives have been disrupted by the Chernobyl accident, but from the radiological point of view, generally positive prospects for the future health of most individuals involved should prevail.

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Lessons learnt from an accidental release of 45 GBq ^{131}I in Fleurus, Belgium

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Abstract

On August 22, 2008, a cloud of ^{131}I was released from the Institut des Radioelements (IRE), a producer of medical isotopes located in Fleurus, Belgium. The peak of the release took place in the weekend of 23-24 August and was followed by a smaller, continuous release which lasted for several weeks.

Several measures were taken after notification of the release to the Federal Agency of Nuclear Control, the Belgian nuclear safety authority. The nuclear emergency plan was activated on a N2 level. Under this emergency plan many environmental samples were taken and analysed with the purpose of verifying whether there was no risk for the population to obtain a significant dose via the food chain. Air samples were taken to confirm that no significant persistent release of ^{131}I was still occuring. This air monitoring was performed just after notification of the release and during major replacement work of the filter systems of the isotope production plant.

The Belgian Nuclear Research Centre SCK•CEN collected and analysed many of the environmental samples, performed the air monitoring and participated in the evaluation cell of the National Crisis Centre. In the course of these activities and in the period after the immediate crisis, some points were discovered that could be improved. Most of these findings dealt with:

- acquisition and measurement capacity of environmental samples
- Public communication
- separation of different tasks in the field
- communication between experts
- availability of expert knowledge

This paper elaborates on the main aspects of the SCK•CEN participation in the emergency response of this incident and draws important lessons to dealing with similar situations in the future.

Introduction

In nuclear emergency management a relatively low number of real incidents and accidents are available for training and improving the emergency plans. Therefore nuclear emergency management relies relatively heavily on theoretical exercises, based on calculations and hypotheses. A real incident or accident may provide a strong confrontation with the existing emergency plans, not only for the technical part, but also including the high pressure from the authorities, public and press.

On August 22, 2008, a cloud of ^{131}I was released from the Institut des Radioelements (IRE), a producer of medical isotopes located in Fleurus, Belgium. The peak of the release took place in the weekend of 23-24 August and was followed by a smaller, continuous release which lasted for several weeks.

The release was followed by the activation of the Belgian nuclear emergency plan, under which e.g. many environmental samples were taken and analysed to see whether there was any radiological risk for the population.

The Belgian Nuclear Research Centre SCK•CEN collected and analysed many of these environmental samples and performed several other duties in the nuclear emergency plan. In the course of these SCK•CEN activities and in the period after the immediate crisis, several points were discovered that could be improved in the working of SCK•CEN. This will be discussed in the following chapters.

Description of the incident

The release of ^{131}I started on August 22, 2008. The total release was 45 GBq ^{131}I. The peak of the release took place in the weekend of 23-24 of August and was followed by a smaller, continuous release which went on for several weeks. The release is shown graphically in figure 1.

The cause of the release was the mixing of three different waste streams from small waste tanks (two of 50 l and one of 23 l) into one main waste tank (2700 l). This caused a chemical reaction that produced I_2 gas. The total source term in the main tank was estimated to be 37TBq. The main part was absorbed by the filters, but 0.1 % was released through the stack of the installation.

Mixing of the different waste streams was common practice. However, this time all the three small waste tanks were transferred at the same time onto the main waste tank, while normally this was done one small waste tank at a time. Furthermore the main waste tank was originally almost empty (200 l), so the remaining concentrations were higher than usual. Simulations showed that certain chemical reactions that would produce I_2 were likely to happen.

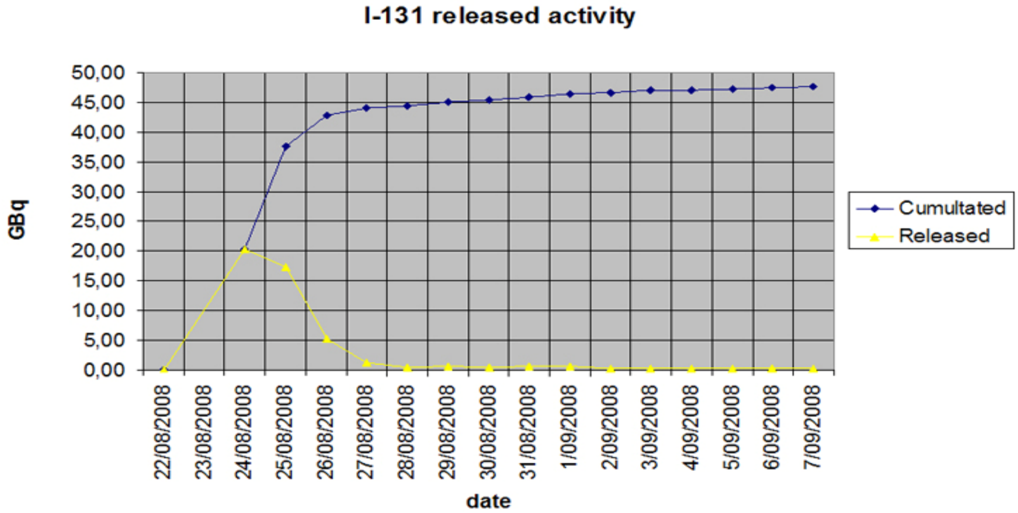

Figure 1. Released and cumulated activity of ^{131}I from the IRE incident [FANC1].

The incident remained unnoticed during the weekend of 23-24 August. It was observed on Monday 25 August by the safety engineer of IRE, who started his duty. The Belgian authorities (Federal Agency of Nuclear Control, FANC) were notified of the release at 17:30 Monday afternoon. Earlier on Monday IRE had already notified the TSO, Bel V [FANC1].

Simulations of the incident

On its own initiative SCK•CEN performed calculations with several dispersion models and decision support systems. These included Noodplan, a home-made model that is used by most Belgian nuclear facilities for emergency planning, HOTSPOT, a simple dispersion model, and RODOS, a decision support system developed with support of the EC.

The calculations with the different models showed a good agreement with each other and with the results of the measurements that were performed. Figure 2 shows just as an example results from RODOS calculations. The obtained results have not been analysed in detail and will not be further discussed in this paper.

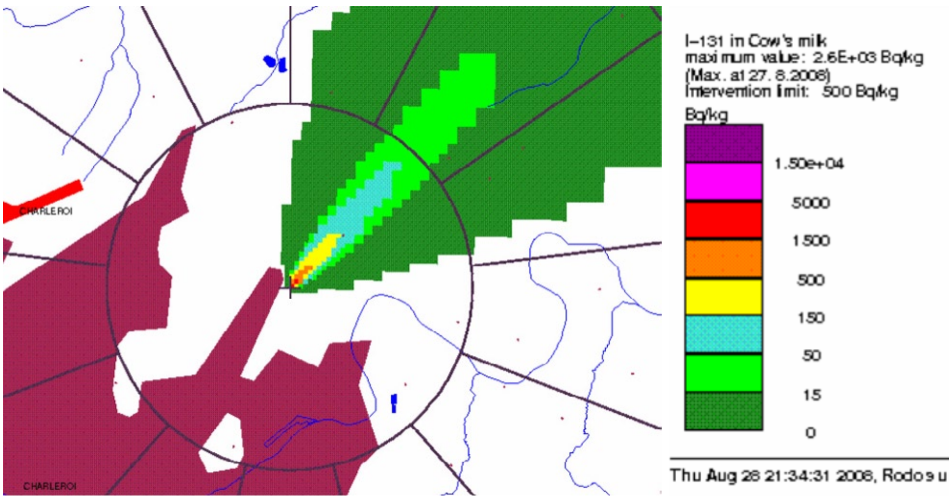

Figure 2. Results of a RODOS calculation of a release of 45 GBq ^{131}I.

The Belgian nuclear emergency plan and the role of SCK•CEN

Figure 3 shows schematically the structure of the Belgian nuclear emergency plan. The national crisis centre CGCCR consists of several cells which have their specific functions.

COFECO is chaired by the Minister of Interior Affairs or its representative and is responsible for taking decisions how an accident should be dealt with. It is supported by several cells for advise: ECOSOC for advice on social-economic consequences of countermeasures; INFOCEL for advice on communication to the public, CELEVAL for advice on radiological consequences of the accident and countermeasures. CELEVAL manages the measurement cell that performs measurements in the field and operates the TELERAD system for direct measurements of radioactivity.

By Royal Decree SCK•CEN provides a radiological expert to CELEVAL, a local measurement coordinator and a measurement team for in-situ measurements and sampling.

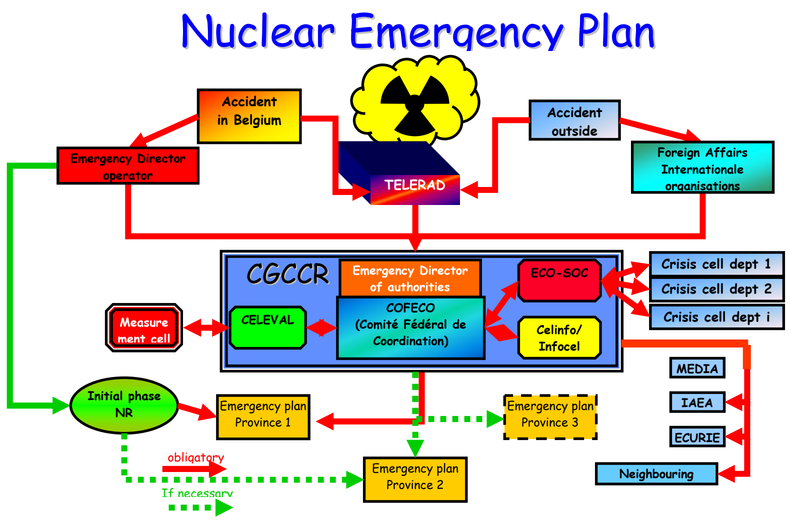

Figure 3. Structure of the Belgian nuclear emergency plan.

Measurements that were performed

On Wednesday 27 August the first three samples of grass were taken in the close vicinity of IRE by FANC. These samples were analysed at the Scientific Institute of Public Health ISP/WIV and the results were made available to the authorities on Thursday 28 August. They showed contaminations that could result in countermeasures for the food chain.

After the national nuclear emergency plan was activated on Thursday 28 August in the late afternoon, SCK•CEN was asked to prepare for the next day an intervention team for measurements and sampling. Instructions would follow the next day.

On Friday 29 August SCK•CEN sent out two teams, one for performing direct measurements and one for sampling. Two teams were sent since no instructions were received yet and it was not clear what would be asked by the authorities. Upon arrival at IRE, instructions were obtained to perform at specific locations air sampling and to perform in a quarter of a circle doserate measurements of the ground contamination. At 15:00 the coordinates were received of the locations were samples of grass or vegetables should be taken. These were five locations that were visited consecutively. In function of the availability, samples of grass, cornleaves and lettice were taken. The upper leaves of the corn were sampled to take the maximum contamination.

The samples were taken at the end of the day to SCK•CEN for sample preparation and measurement by gamma-spectrometry. Measurements were performed during the evening and night in order to provide the results to the authorities at the beginning of the next day.

Similar sampling and measurement campaigns were held at Sunday 31 August, Tuesday 2 September, Wednesday 3 September and Thursday 4 September. At the end of each day samples were taken to SCK•CEN in order to provide measurement results at the beginning of the next day. On Thursday 4 September an air sample was taken on active charcoal.

On Monday 1 September and Tuesday 2 September SCK•CEN performed, together with the University of Liège, a thyroid measurement campaign on the population of Lambusart, a village northeast of IRE. This campaign is reported in a separate paper [van der Meer et al.].

On Friday 5 September it was decided to do another air sampling campaign, since at that day an additional filter group was placed between the waste tank and the stack.

Figure 4. Ground contamination measurements (left) and air sampling (right).

Figure 5. Sampling of vegetables in local gardens.

Countermeasures that were taken [FANC2]

As mentioned beforehand, the incident remained unnoticed during the weekend of 23-24 August. It was observed on Monday 25 August by the safety engineer of IRE, who started his duty. The Belgian authorities (Federal Agency of Nuclear Control, FANC) were notified of the release at 17:30 Monday afternoon. Earlier on Monday IRE had already notified the TSO, Bel V.

On Tuesday 26 August inspectors of FANC and Bel V visited IRE and ordered not to start any new production batch. Mobile monitoring systems were installed around the installation.

On Wednesdat 27 August FANC notified the IAEA and rated the incident on level 3 of the INES scale. FANC also collected grass samples in the close neighbourhood of the plant.

On Thursday 28 August results of the measurements became available, indicating that at certain locations the limits for countermeasures for the food chain could be passed. This triggered the activation of the national nuclear emergency plan on 17:00 on the level U2, implying no direct risk for the population but potential countermeasures for the food chain. A recommendation was issued that the population should not use vegetables or fruit from their own garden nor should use rain water in the region between 0-5 km northeast of IRE. Selling milk produced in the same region was also prohibited.

On Friday 29 August the sale of milk was again allowed, based on measurements on milk carried out on behalf of the Federal Food Agency. The other countermeasures remained in force.

On Saturday 30 August new measurement results were available. Based on the results, the affected region was reduced from 0-5 km to 0-3 km, while rainwater could be used again. The recommendation not to consume vegetables and fruit from the own garden remained.

On Saturday 6 September the remaining countermeasures were lifted, based on the results of all the measurements performed in the course of that week. Furthermore the emergency level was reduced from level 2 to level 1.

On Friday 12 September the national nuclear emergency plan was lifted.

Possible improvements

Many aspects of this incident can be used to improve the working of the nuclear emergency plan. In this paper we will mainly limit ourselves to propose improvements to the duties of SCK•CEN in the framework of the Belgian nuclear emergency plan as described above.

Separation of different tasks in the field

Before the IRE incident it was always assumed during exercises that one measurement team would be able to perform all actions in the field. Very soon it became clear that it was very cumbersome to send a team in the field with a combination of tasks. Different tasks like performing direct doserate measurements and sampling interfere quickly with each other and are best seperated, unless very little samples have to be taken. Separation of tasks also improves considerably the flexibility of the teams for answering to specific requests from the authorities.

Three different measurement teams have been defined after the incident: an intervention team responsible for direct measurements (doserate, ground contamination, air activity sampling), a sampling team for biological sampling (grass, vegetables, milk, water, ..) and an AGS team (Aereal Gamma Survey). The latter is supposed to operate from a helicopter or airplane, but at present it is operated by car.

Due to the fact that the intervention team and AGS team rely partly on the same expertise, SCK•CEN may at a certain moment not be able to provide experts for all teams in case of a protracted incident that requires measurements for e.g. several weeks.

Acquisition and measurement capacity of environmental samples

The IRE incident was a relatively small incident that contaminated only a limited area. Nevertheless it resulted in a serious demand on the capacity of the sampling team and the gamma-spectrometry measurement lab.

A larger incident in Belgium would require more resources from SCK•CEN. Both sampling and measurement capacity should be increased in that case.

A more optimized use of sampling and measurement teams seems possible. During the IRE incident the sampling locations and types were based on the collective results of the previous day resulting in a discontinuous sampling/measurement process (sampling in afternoon, measurements over night, analyses next morning + definition new sampling locations/types, new sampling in afternoon, ...). A more continuous sampling/measurement strategy in combination with the use of pure logistic teams for e.g. the transport of samples could already result in an optimization of sampling/measurement capacity.

Not all sampling teams had during the incident access in the field to software for the optimization of the route between a set of sampling locations (typical 10-20 locations). A general procedure on the transfer of measurement locations/type and measurement results was missing.

Sampling capacity was already increased by establishing a dedicated sampling team. Further expansion of the capacity is envisaged by increasing the number of sampling teams. This is only feasible in case the pool of persons that have sampling experience is increased. Expansion of this pool is envisaged by the participation of more people in the regular sampling campaigns in the framework of the surveillance of the Belgian territory. Additionally written procedures for sample taking have been developed that were lacking during the IRE incident. This will provide a better guarantee that samples will be taken in a similar way.

For several types of samples (mainly vegetation) the preparation of the samples will prove to be a bottleneck rather than a gamma-spectrometry measurement. At present preparation of samples deals with samples from the surveillance programme, that have no or very little radioactive contamination. Preparation rooms for these samples should not be used in case of contaminated or suspected samples. A dedicated room should be established. In case of the IRE incident an ad hoc solution was found by using a room with a ventilated box in a controlled area, but this room will not be sufficient for large quantities of samples.

Once prepared, samples can be measured in a rather short time since they are either sufficient active to give a good signal or will not give a significant signal from which the onclusion will be drawn that the activity is below a certain limit. However, clear measurement procedures still have to be developed for this case.

More exotic contaminations like beta or alpha contaminations will prove to be much more difficult to deal with, since the preparation time for these measurements are much longer than for gamma-spectrometry. Again specific measurement procedures have to be developed for these contaminations.

Public communication

It was inevitable that teams in the field would encounter representatives of the media. Although the communication about the incident should go exclusively via the national crisis centre, the field teams were sometimes forced to answer at least partially to specific questions. In a similar way the people whom were asked to provide their vegetables had many questions and also these questions should be answered by the teams in an adequate way in order to guarantee a continuous public support for the sampling campaign.

So although the principle of crisis communication demands one clear voice of the authorities and therefore demands a central spokesperson, the people in the field are exposed to questions from the media and should have adequate means/training to respond to these questions, either by providing clear answers or referring to a central spokesperson.

Communication between experts

Several types of communication could be distinguished:

- communication between experts for passing through information. This was mainly important for the evaluation cell of the national crisis centre. A summary of what was discussed during the day had to be given to the expert for the next day. This was done either in the form of a written report or orally by phone
- specific questions from the evaluation cell had to be dealt with by measurement experts. E.g. the request for a thyroid measurement campaign had to be supported by estimates of the detection limits of the available equipment and the amount of people that could be measured. This type of questions were handed over to a central scientist at SCK•CEN and then discussed internally with the relevant experts. An answer was provided in the form of a short note to the SCK•CEN radiological expert
- communication with the measurement team was dealt with by the measurement cell coordinator, an official function executed by SCK•CEN that takes care for the transfer of information between the federal evaluation cell and the measurement team. Coordinates of sample locations and type of samples are typical data to communicate to the measurement team.

Communication was often performed by cellular phone. This worked well, but in case of a major crisis it is envisaged that the cellular phone networks will be overloaded and cannot be used. Therefore extra phones have been purchased that function on a secure and reliable network.

Availability of expert knowledge

Both for making available an expert for the evaluation cell of the national crisis centre and for providing an expert for coordination of the measurement team, SCK•CEN has established a duty cycle. Each expert fulfills a duty of one week and hands it then over to the next expert.

It was noticed that the establisment of such a duty cycle is not a warrant that during a real crisis the expert functions can be fulfilled 24 hours per day and 7 days per week. The duty cycle is only a warrant that SCK•CEN will be able to provide an expert the moment an emergency call comes in for the next 8-12 hours. An expert shift had not been established.

During the incident SCK•CEN provided an expert for each evaluation cell meeting, based on the availability within the pool of experts at SCK•CEN. The decision who would participate in a meeting was normally taken one or two days before that meeting took place, but it cannot be denied that this method of working implied a certain form of improvisation.

The same was true for the establishment of the measurement teams. This was done on a daily basis and subject to the demand from the national crisis centre. Since only the coordinator of the measurement team is in a duty cycle, but not the team members themselves, here the level of improvisation tended to be higher than for the evaluation cell expert.

It has been recognised that the number of available experts should be enlarged in several domains: radiological experts, sampling team members, measurement analysts and experts (both for in the field and at the labs). However, in view of the limited resources available for this enlargement it should also be recognised that a certain level of improvisation will always be present.

Conclusions

The IRE incident did not cause any radiological harm to the general public. It is an interesting case to provide some lessons learnt for the emergency preparedness organisations in Belgium and possibly abroad.

The tasks that had to be executed in the field required one person (or team) for one job. Too many different tasks divide the attention people can devote the these tasks and imply a risk for making mistakes.

Sampling procedures have to be improved in order to be able to take more samples in a uniform way. Dedicated rooms for active samples should be present where many samples can be processed.

Gamma-spectrometry will not provide a bottleneck in measuring samples. However, due to the long preparation time alpha and beta measurements may require dedicated measurement procedures for large numbers of active samples.

Public communication is an important tool to provide the necessary support of the public for the measurement campaigns and countermeasures. During the sampling campaign SCK•CEN encountered full support of the public to provide samples from gardens.

Communication between the experts was satisfactory, but should be performed in a more standardised way by e.g. providing meeting reports after each day.

Availability of sufficient experts remains a difficult problem for relatively small organisations in a small country. Improving procedures will solve this partly, but a certain reliance on improvisation will remain.

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[FANC1] http://www.fanc.fgov.be/nl/page/radiological-incident-at-ire/827.aspx (2010-04-09)

[FANC2] http://www.fanc.fgov.be/nl/page/incident-ire-fleurus-2008/930.aspx (2010-04-09, in Dutch)

[van der Meer et al.] Klaas van der Meer, Anne-Laure Lebacq, Eric Boogers, Sven Boden, Greet Verstrepen, Filip Vanhavere, Jantine Schröder, Johan Camps, Carlos Rojas Palma, Lieve Sweeck, Tanja Perko, Isabelle Majkowski, Catrinel Turcanu, Benny Carlé, "Thyroid measurement campaign after an accidental release of 45 GBq ^{131}I in Fleurus, Belgium", this conference

How user involvement improves decision support: Experiences from the EURANOS project

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Abstract

The EURANOS (European Approach to Nuclear and Radiological Emergency Management and Rehabilitation Strategies) project aimed to increase the coherence and effectiveness of nuclear and radiological emergency management in Europe including the rehabilitation of contaminated areas. Basis of this project was the establishment of an effective working platform of emergency management institutions, Research and Technological Development (RTD) institutes, end-users and other stakeholders. Within the EURANOS project, 17 different demonstrations were conducted. The objective of these demonstrations was to apply a certain product in the operational environment of an emergency centre and to check its performance either in particular exercises or in their daily use. As a result of a demonstration feedback to the development team was provided, highlighting requirements for the further development of the product – if necessary. A second key pillar in the project was the intensive involvement of the RODOS Users Group (RUG) in the research activities. In particular all work packages related to the further improvement of the RODOS system have been observed by the RUG. In some of the work packages, the RUG was involved from the beginning and participated even in the design documents for the software development. Both mechanisms, the demonstrations and user involvement, guaranteed that the end user's perspective was considered throughout all phases of the EURANOS project.

Introduction

Since the Chernobyl accident, considerable effort has been made in improving emergency management in Europe following nuclear incidents or accidents. However, many of these methods and information technology (IT) tools developed either nationally or under the EC's 4th and 5th Framework Programme are still not fully operational or not disseminated all over Europe. As a consequence, the EURANOS (European Approach to Nuclear and Radiological Emergency Management and Rehabilitation Strategies) project was initiated to increase the coherence and effectiveness of nuclear and radiological emergency management in Europe including the rehabilitation of contaminated areas. From the beginning it was obvious that such a task can only be accomplished successfully when Research and Technological Development (RTD) institutes, end-users and other stakeholders work together focusing on methods and tools that could be used operationally all over Europe.

The overall goal of the project was the enhancement of the technical, methodological and strategic approaches for national and cross-border emergency management and rehabilitation in Europe, what could progressively lead to the establishment of a European Platform for emergency management and rehabilitation strategies.

This could only be achieved with a large number of partners and when those responsible for nuclear or radiological emergency management and rehabilitation strategies within their countries and the Research Institutes developing methods, IT tools and strategies work closely together. Within the EURANOS project, 50 partners (17 national emergency management organisations and 33 research organisations) from 22 countries established a comprehensive working platform that covered all parts of Europe (see Figure 1).

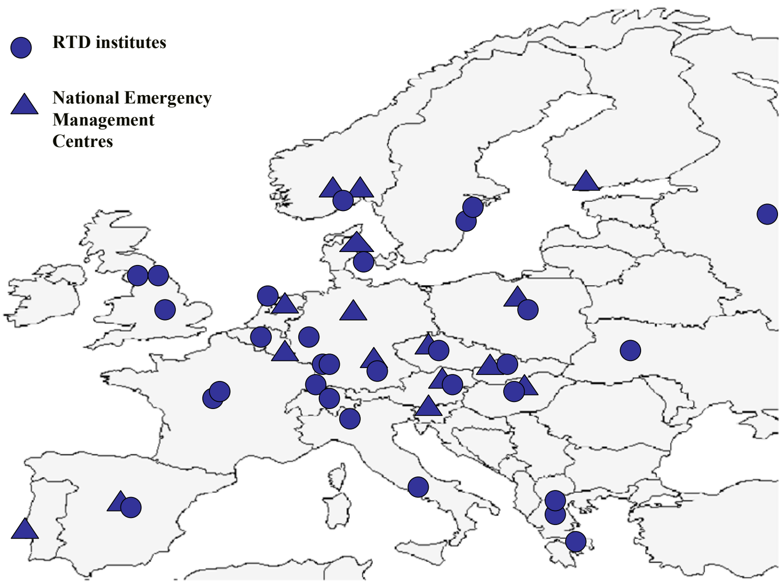

Fig.1. European coverage of the EURANOS project.

The work program of the EURANOS projects was structured in such a way that end products of research activities were continuously demonstrated by the operational community. This provided immediate feedback on the operation and usefulness of new methods, IT tools strategies and guidance (see Fig. 2).

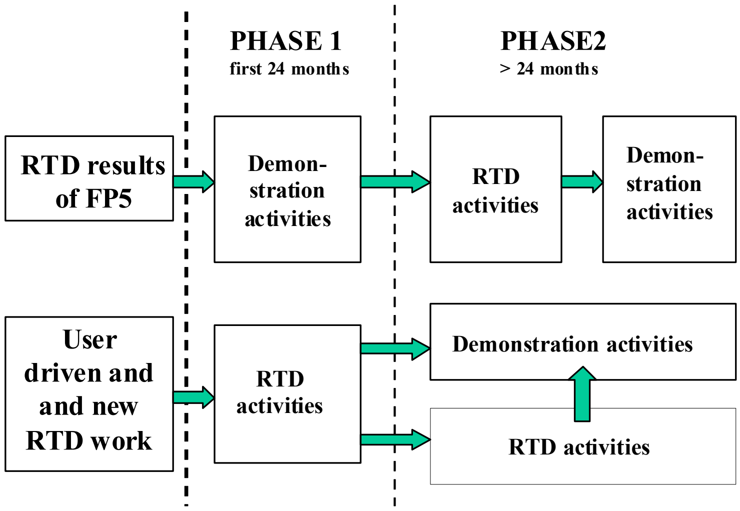

Fig. 2. Structure of the interaction between RTD and demonstration activities within phases 1 and 2 of EURANOS.

The objective of the demonstrations was to - using the feedback from expected future users - identify which of the tools, methods and approaches need further development and refinement. The scope of the demonstration activities in the EURANOS project encompasses activities in the various RTD work packages. In each of these work packages, the tools (in some cases specific elements of the tools), methods and approaches have been demonstrated under conditions which reflected as close as possible future operational uses of the tools demonstrated.

As one of the aims of EURANOS was to improve the operational use of the RODOS (Real-time On-line Decisions Support) decision support system (Ehrhardt, 2000), the RODOS Users Group (RUG) has been established as a discussion and interaction forum for co-ordinating and managing activities related to the demonstration and enhancement of the RODOS system and for providing essential feedback to the developers.

This combination of the user interaction via the demonstration projects and via the RUG as forum was one of the pre-conditions for the success of the EURANOS project.

Demonstration Projects

Demonstration activities were subdivided into two phases. In the first two years of the project (phase 1), either products from former research programs were demonstrated or products that were accomplished in the beginning of the various RTD activities of EURANOS. In the second phase, demonstrations focused on those produced realised within the EURANOS project only.

The demonstrations carried out within Phase 1 focused on the RODOS system. In particular the rather complex user interaction with the system was criticised in light of the continuously decreasing manpower and resources in emergency centres. Therefore, the use of the system has to be guaranteed also with a limited number of operators not deeply familiar with the RODOS system.

In the second phase, demonstrations focused on the usability of the decision aiding component of the RODOS software, small scale exercises dealing with releases other than from a nuclear power plant and the two handbooks on assisting in the management of contaminated food production systems and contaminated areas, which have been developed within the EURANOS project. In these two demonstrations of the generic handbooks not only the final product was exercised but also the process how to establish the best environment for the usage of these handbooks. This comprised the set-up of stakeholder panels which are an integral part of the successful application of the handbooks following a nuclear or radiological emergency. The two handbook demonstrations provided valuable feedback for the further improvement of the final documents available by the end of the EURANOS project. The final demonstration was devoted to the re-engineered RODOS system. The following list provides an overview on the seventeen demonstrations carried out.

Title	Objective of the Demonstration
Probabilistic Estimation of Source Terms from in plant data (SPRINT)	To demonstrate the use of SPRINT within the emergency organisation (plant or site-level as well as national level) to determine the feasibility of this tool for early prediction of the source term and to resolve related issues. To determine the appropriateness and related conditions for operational use of SPRINT To collect feedback from plant operators, TSC staff and decision makers (a/o authorities) in order to produce recommendations on further refinements or enhancements needed.
Networking and processing of on-line data in decision support systems	To verify the operability of the systems and including the compatibility of the formats to enable RODOS system to adopt the data and use it in its predictions: Specifically, the aims of the demonstration are to verify: • Transmission of on-line radiological and meteorological data from monitoring systems • Request and transmission of meteorological forecasts from national weather services • Further processing and adoption of both types of data within RODOS, in particular in the different operating modes.
Visualisation and evaluation of on-line data in decision support systems (RtGraph)	To evaluate RtGraph, RODOS' data visualisation tool for its usefulness and possible uses by emergency support organisations. To provide specialized user's perspective feedback to the development of the RtGraph on possible needs and ways to enhance and optimise the products to achieve the functionality needed.
User interfaces of decision support systems 1 – UI provided with patch 6.0	To assess the adequacy and usefulness of the RODOS user interface that was delivered within the RODOS patch #6.
User interfaces of decision support systems 2 – RODOS Lite	To assess the adequacy, usefulness and robustness of the new RODOS Lite user interface.
Adequacy of the system results and their forms of presentation within the decision making process	To establish to which extent RODOS (or another DSS) fulfils the needs of an emergency centre in providing the information that is needed for the support the decision making during various phases of nuclear emergencies.
Operation of decision support systems in a cluster environment with remote users of different access rights within one country	To demonstrate the use of the RODOS system with a central installation of the system in a national cluster with several external (i.e. remote from the central installation) RODOS users with different roles and rights.

Title	Objective of the Demonstration
Shared use of and data exchange between decision support systems installed in neighbouring countries	To demonstrate the use of the MODEM server for the exchange of information between countries using decision support systems, RODOS and ARGOS in this case. To investigate the benefits and possibilities to enhance the decision support while continuously (i.e. discretely, but in relatively short intervals) obtaining the data for a country where an accident occurred, and using those within RODOS.
Evaluation of the appropriateness of the ASTRID system for source term estimations based on in-plant data	To evaluate if the ASTRID system answers to potential end-users needs in emergency situation.
Small projects	To demonstrate the usability and usefulness of RODOS to support the decision making related with releases other than major NPP accidents.
Testing atmospheric capabilities of RODOS system installations through ENSEMBLE atmospheric dispersion exercises	To explore in RUG or elsewhere the interest in the emergency management community for having access to results for food concentrations, intervention levels, etc, of a similar form to those in ENSEMBLE for air concentration and deposition and second to attract all emergency management organisations to make use of the ENSEMBLE facility and to test the implementation and use of the MATCH model (inside RODOS) in the various countries
Demonstration of the usefulness and usability of the evaluation techniques in RODOS - Web-HIPRE	To demonstrate the usefulness and usability of the Web-HIPRE (Hierarchical PREference analysis in the World Wide Web) Java applet for decision analytic problem structuring, multi-criteria evaluation and prioritisation as part of the RODOS system for the later phase problems
Application of the MOIRA DSS to evaluate rehabilitation strategies for contaminated freshwater bodies at the local or regional levels	Testing the applicability of the MOIRA system for the definition and analysis of a variety of appropriate strategies for the long-term management of contaminated freshwater bodies, for both lakes (local scale) and rivers (regional scale). Testing the validity of the MOIRA system as a tool in the decision making process able to incorporate inputs from different ranges of stakeholders.
Demonstration of the usefulness and suitability of the hydrological dispersion module (HDM)	To assess whether the use of hydrological dispersion model (rivers) is of interest for emergency preparedness and how such a model would enable faster and better decision making at emergency centres. This was exercised with the help of a Polish case study
Generic Handbook for assisting in the management of contaminated food production systems in Europe	The first part of the demonstration aimed at testing the usefulness of the Handbook for developing a management strategy for one or more scenarios involving contamination of the foodchain. The objective of the second part of this demonstration was to assess the value of engaging stakeholders in applying the Handbook.
Generic Handbook for assisting in the management of contaminated inhabited areas in Europe	To test the usefulness of the Handbook for developing a management strategy for one or more scenarios involving contamination of an inhabited area
Basic Evaluation of JRODOS	To evaluate the reengineered RODOS system software, from the perspective of its usability and its robustness. Further to corroborate that the RUG requirements for the reengineered RODOS software are adequately addressed.

A demonstration project consisted of three main steps. First, the demonstration notebook was created which defines the main tasks to be carried out during the demonstrations and identifies the evaluation criteria. The second step was the demonstration itself which was performed either by individual organisations or in clusters of countries (which was the case for the demonstration of data exchange between decision support systems installed in neighbouring countries). Finally, the findings were summarised and evaluated resulting in recommendations from the operational user community how to proceed further with the tool demonstrated.

Key demonstrations related to RODOS were devoted to the revised user interface (RODOS-Lite) and the re-engineering of the RODOS system.

RODOS-Lite

In the first phase, the demonstration of the user interface resulted in a complete restructuring and improvement of the design, layout and IT-basis of the interface. From the beginning, even in the design phase, the RUG was heavily involved and interlinked with the research team. The RUG provided guidance in all aspects related to the operational use of such input forms. They also defined used cases describing the functional specifications needed for their operational application in their emergency centres. The demonstration thus focused on the fact to which extent the requirements set up by the end users had been realised in the final product. This final product, named RODOS-Lite user interface, was and is an integrated part of the RODOS system in all its realisations, either on HP, Linux or, with the re-engineered system, also Windows operating system.

Most of the RODOS users participated in the demonstration. At the end 12 organisations from 9 countries delivered individual reports with detailed information on the demonstration and findings that were forwarded to the software developers for further improvement of the product. In general it was concluded, that the RODOS-Lite user interface met the objectives defined at the start of the development pahse. Furthermore, it is applicable also for those users with little RODOS knowledge or even for fully untrained users.

JRODOS

With the ongoing EURANOS project, the RUG decided to request a fully re-engineering of the RODOS system. This was partly caused by the success of the RODOS-Lite interface demonstrating the capabilities when using state of the art IT technologies and modern design aspects. Therefore, two years before the end of the project, re-engineering of RODOS started focusing on a complete restructuring of the operating system but aiming to leave the simulation models unchanged. The new version should be JAVA based and thus applicable under the Linux and the Microsoft WINDOWS operating systems. The new version was named JRODOS. As RODOS was an important part of EURANOS, the success of the re-engineering of the RODOS system was important for the success of EURANOS in general. To demonstrate and evaluate the re-engineered RODOS system, four months before the end of the project, a demonstration was scheduled with the aim to assess the installation, overall performance and robustness of the new version and to test all functionalities by applying it to exercises or test runs. The assessment was expected to be done also in comparison to existing HP-UX/Linux installations of RODOS.

The number of participants was the highest among all demonstration projects. 15 organisations participated and issued either individual or combined reports with their main findings. There was a clear positive outcome of this demonstration project. Most of the participants concluded that JRODOS would be the version of RODOS that will be used in future in their emergency centres. However, some recommendations should still be considered. In this way, this demonstration highlighted the positive feedback of the users' involvement from the beginning of the RTD project and via the demonstration up to the final product.

RODOS Users Group (RUG)

When starting the EURANOS project, the Management Committee realised quite soon the importance of the end users' integration into the feedback loop – via the demonstrations – but also into the software development projects themselves. Therefore an imitative was started to create the RODOS Users Group in the first year of the EURANOS project. Furthermore, the chair person of the RUG became a member of the EURANOS Management Committee. This assured that feedback from this side was always represented at the highest level of management of the project.

The objective of the RUG during the EURANOS project was to provide RODOS system users with assurances that their experience and their demands was adequately considered. The aim was also to assure that there is a long lasting commitment of the RODOS developers to ensure proper operation and maintenance of the system for the future. To stimulate the feedback expected from the RUG, a strong interaction with the demonstration activities within EURANOS project had been established.

Specific aims of the RUG included:

- To provide a platform through which the members of the RUG can communicate their views, needs and comments and exchange their experience related with all elements of the RODOS system and its use, in particular provide response and guidance on refinements to make the system more user friendly and for any future developments of RODOS.
- To contribute to, to discuss and to approve the evaluation reports of the individual demonstration projects.
- To establish reports on RODOS users' requirements and views to the EURANOS Management Committee, and to provide advice on specific questions of the Management Committee, such as on planned directions for improvements / development.
- To share experience gained while integrating RODOS in the national emergency management arrangements, and to enable RUG members to enhance their own arrangements.
- To identify best practices, to share technical know-how and organisational solutions, software developments and data bases and their implementation, and to provide mutual support, particularly on a regional basis.
- To share practice and solutions related with use of RODOS for training and in exercises.
- To provide a forum through which the members of the RUG can network with each other, independent of the RUG's activities.
- To establish contacts to the User Groups of other decision support systems within Europe (ARGOS and RECASS) and overseas and/or to emergency management organisations not involved in the EURANOS project.
- Strive at reaching compatibility of the RODOS system with other decision support systems.

- To ensure sustainability of the RODOS decision support system after the end of the EURANOS project through the establishment of maintenance procedures and sustainable arrangements between the users and the developers.
- Promote the use of RODOS in Europe.

Members of the RUG participated in all demonstrations related to the RODOS systems. From the beginning, the user interaction with the system was of highest concern for the RUG. In a first step the RUG participated in the development of the RODOS-Lite user interface promoting state of the art IT technologies with its implementation in JAVA (object oriented programming language). The RUG requested functionalities for a direct error management of the user's input and consequent guidance of the user through a series of easy to understand input frames. In a second step, the RUG laid down their requirements for the re-engineering of the RODOS system in a long "wish list", in which priorities for the various requirements were provided, general demands were listed and more detailed requests were developed for a list of 16 use cases, which described typical applications of the system in an emergency centre. Especially important for the RUG were the following topics to be considered in the re-engineering:

- User friendly and intuitive graphical user interface with low training requirements and easy, consistent and neatly arranged user input forms
- Graphical representation of results that meets user requirements (e.g. integration of GIS functionalities, multi-lingual, annotated, etc.)
- Easy system administration and low maintenance costs
- Technical support including hot-line, web page (properly managed, supervised and kept actual, with notification on news etc.) for download both updates, patches, new versions etc. and guides and manuals, FAQ section
- Easy way of integrating external simulation modules in a framework with clearly defined interfaces

The RUG also requested that intermediate prototypes should be released for immediate testing. With this prototyping the RUG assured that new features introduced in one prototype could be immediately tested and feedback from the tests could be used in the development of the next prototype. This feedback loop via the numerous prototypes released within the two years of the re-engineering guaranteed that the development followed the requests and recommendations of the operational end users.

Due to the successful cooperation within the RUG and in between RUG and the RODOS developers it was decided among the RUG members to keep this group alive even beyond the end of the EURANOS project. For the future it is planned that the RUG becomes a part of the emerging European Platform NERIS (see further down).

Conclusions

When evaluating the results of the EURANOS project, one can conclude that the project successfully accomplished most of the objectives defined in its working program. This success is clearly the result of the demonstration projects as focal point for the application oriented RTD activities and the intensive interaction with the end-users organised in the RUG. Both have led to significant modifications of the working programme from the beginning of the project. They initiated in particular the development of the user friendly input interface RODOS-Lite and the complete re-engineering of the RODOS system focusing on modern IT-technology and enhancing the system to be used as an information platform for tools related to emergency management and rehabilitation. In this way, the new concept of involving end users from the beginning in a research project has been the key of success of the EURANOS project and will build a perfect basis for any further development of decision support systems. This success can be also seen in the emerging European Platform on Preparedness for Nuclear and Radiological Emergency Response and Recovery (NERIS) that will be established as part of the IRPA 2010 congress. In NERIS, for the first time research and operational community will form a sustainable platform aiming to sustain the current high achievements and further improve emergency management and response in Europe.

Acknowledgment

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Enhancement of the radiation monitoring and emergency response system in the North-West Region of Russia

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Abstract

Large scale activities aimed at decommissioning of a great number of Naval radiation-hazardous facilities are being implemented in Northwest Russia throughout the last decade. One of the key elements of safe decommissioning is preparedness to response to possible radiological accidents. Therefore, a modern emergency response system in the Murmansk and Arkhangelsk Regions are an obligatory element ensuring protection of population and territories in case of radiological accidents at facilities connected to nuclear submarine decommissioning, spent nuclear fuel (SNF) and radioactive waste (RW) management.

The paper describes the basic results of the international project "Enhancement of the Radiation Monitoring and Emergency Response System in the Murmansk region" implemented in 2005-2008 by the Energy Safety Analysis Centre of IBRAE RAS. The Project was funded by the Northern Dimension Environmental Partnership Support Fund - Nuclear Window. The Government of the Murmansk Region was the customer and beneficiary of this activity. The same project was started in Arkhangelsk region in mart 2009.

Introduction

The key objective of the Project is the cardinal improvement of the Radiation Monitoring and Emergency Response System in case of accidents at radiation hazardous facilities related to NPS dismantlement and RW, SNF management. The project is directed to the increase of emergency response forces and techniques preparedness, the minimization of possible radiation accidents effect, the increase of efficiency of decision-making and the implementation of actions related to population and environment protection. Radiation-hazardous facilities integrated into the radiation monitoring and emergency response system of the Murmansk region are given in Fig. 1.

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Fig. 1. Radiation-hazardous facilities integrated into the radiation monitoring and emergency response system of the Murmansk region.

The Project Objective are:

– creation of Regional Crisis Center(RCC) in Murmansk region and Crisis Center of FSUE SevRAO;
– creation of program and technical system of operative expert support to make decision concerning the personnel, population and areas protection;
– development of existing and creation of new automated radiation monitoring systems (ARMS) at the areas of the enterprise and facility including the mobile systems of radiation survey;
– creation of communication systems and lines for data transfer, collection, processing, storage and representation for emergency response participants at facility, regional and federal levels and creation of operative expert support system of RCC of Murmansk region and FSUE SevRAO on the basis of Technical Crisis Centre(TCC) of IBRAE RAS.

Functional Diagram of Emergency Response System in Murmansk Region is given in Fig. 2. The main activities in the Project framework included the modernization of the existing and creating ARMS for facilities and territories, including mobile radiation surveillance laboratories; establishment of the RCC of the Murmansk region and the Crisis Centre of the FSUE SevRAO; setting up communication systems for transfer, acquisition, processing, storage and presentation of data for participants of emergency response at the facility, regional and federal levels; development of software and hardware systems for expert support of decision-making on personnel, population and environment protection; and establishment of a system for expert support of the created centres by the TCC of IBRAE RAS.

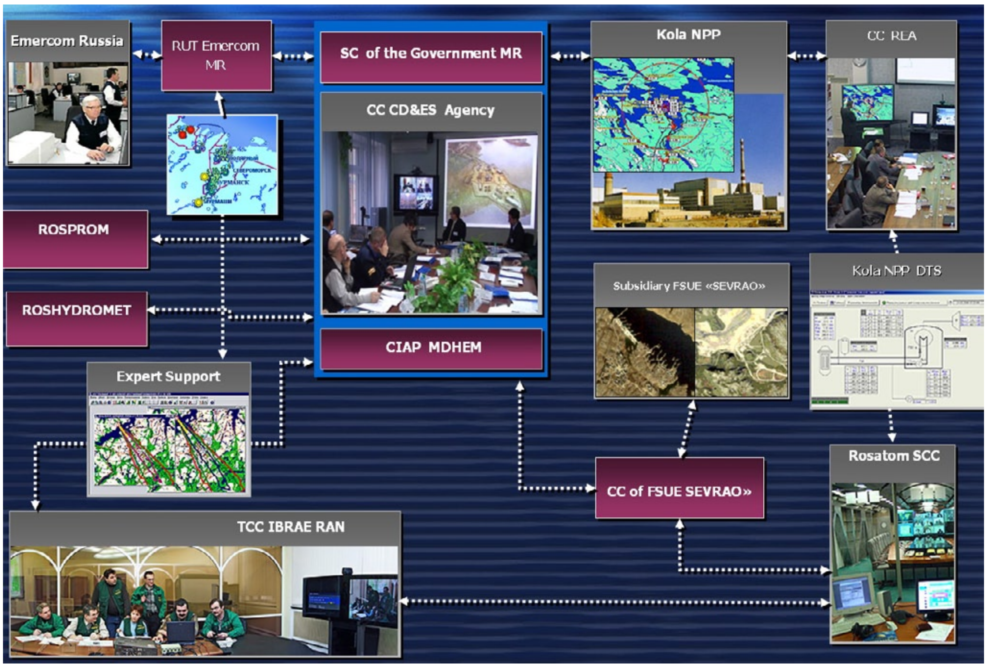

Fig. 2. Functional diagram of emergency response system of the Murmansk Region.

The Regional Crisis Centre(RCC)

The Regional Crisis Centre is created for the information and technical support of the Regional Administration, expert groups and commissions responsible for development and decision making for population and area protection in case of emergency at the nuclear and radiation hazardous facilities. In routine activity RCC carries out the operative monitoring of radiation environment at the local area, planning and control of measures for emergency situation prevention, trials and exercises. RCC includes the Contingency Center of the Government of Murmansk area, Crisis Center of Civil Defense, Emergency and Fire Protection Agency, Data Collection and Processing Center of Murmansk Agency of the Federal Hydrometeorology and Environmental Monitoring Service.

All parts of RCC are equipped with modern equipment and communication channels to organize the information interaction with local enterprises, regional and federal structures of the executive power.

The key tasks of the RCC:

- The information and technical support of the regional administration
- Agencies responsible for decision making under the minimization and elimination of the radiation accident effects;
- Planning and control of measures aimed against the accidents for the region as a whole;
- Current monitoring of key parameters of radiation situation at the overall area of the region;
- Provision of the interaction between participants of emergency response system at local, regional and federal levels;

The key tasks of the Contingency Center:

- Operative information provision about current situation in emergency area to the regional administration, operative cooperation with the Commission on emergency situation in the region, regional and federal executive bodies and enterprises;
- Daily information provision about emergency situation prevention measures to the regional administration, control of potentially dangerous facilities and environment of the region;
- Public information.

The key tasks of the Crisis Center of Civil Defense, Emergency Situations and Fire Protection Agency:

- Information and technical support of the regional commission on emergency situations and the Government of Murmansk area in daily and emergency modes;
- Planning and control of commission and execution of actions against the emergency situations for the region as a whole;
- Operative monitoring of key parameters of radiation situation at overall region including mobile laboratories of radiation survey;
- Support and development of information and program technical resources, provision of communication facilities operation and data exchange.

The Major Tasks of Data Collection and Processing Center:

- Collection, accumulation, processing, analysis, submission and transfer of the radiation situation data of local ARMS;
- Processing of current data about meteorological situation in the region and prognosis
- Provision (in case of the radiation situation); technical support and development of local ARMS; assessment and prognosis of air and water radionuclides transfer including Tran boundary one (together with Information Analytical Center of Rosgidromet).

Crisis Center of SevRAO

Crisis Center of SevRAO is intended for support and information reaction at the level of radiation hazardous facilities located at the area of the enterprise engaged in nuclear fleet decommission, RW and SNF management and the infrastructure facilities remediation. There are three branches in SevRAO, i.e. the former coastal technical bases of the Navy in Andreeva Bay and Gremikha and reactor compartment storage facility in Saida Bay. Crisis Center of SevRAO is formed on the basis of the enterprise in administration in Murmansk. The key tasks of the Center:

- Operative monitoring of main parameters of the radiation situation at the area of the enterprise branches;
- Planning and control of measures against the emergency situations at the branches of the enterprise;
- Assessment of situation, elaboration of recommendations and technical support of Commissions on Emergency Situations of SevRAO and branches in emergency situations;
- Information coordination with Rosatom;

- Information interchange between participants of emergency response system including the interaction with Regional Center of Murmansk area, Contingency Crisis Center of Rosatom and TCC of IBRAE RAS.

Information and Technical Complex of Crisis Centers

Information and Technical Complex of Crisis Centers includes:

– software: data bases, informational systems of radiation hazardous facilities condition, scenario of possible accidents, plans of the personnel and population protection, electronic maps, design simulated systems of prediction and assessment of the radiation situation with radioactive waste release in the atmosphere, water pollution assessment system (coastal water), engineering application programming of radiation dose and contamination assessment, display facilities of radiation monitoring system data;
– hardware: videoconference system, audio video presentation equipment, modern automated working places for the personnel, server and communication equipment, no break power system;
– communication facilities: office automatic exchange, automated warning system, fiberoptic communication lines between Crisis Centers; communication facilities with centers of the Federal level; satellite network for Rosatom data transfer; duplicated communication lines.

The certain set of the program and information facilities are used in the Crisis Centers depending on specific features of work. For example, the systems of assessment and prediction of radiation accident effects to the personnel, coastal sites and controlled areas are installed in the Crisis Centre of SevRAO. Centralized Information Exchange System of Murmansk Hydrometeorology and Environment Monitoring Agency provides the opportunity of operative processing of data concerning the radiation monitoring and prediction of air, water transboundary radioactive waste transfer.

In the Crisis Centre of Civil Defense, Emergency Situations and Fire Protection Agency it is possible to carry out the preparation of recommendations, concerning the population and areas protection measures with help of computer systems.

Murmansk Territorial ARMS

Murmansk Territorial ARMS are designed for online data collection about the radiation situation in Murmansk region, information of the regional and the Federal executive bodies and population. Murmansk Hydrometeorology and Environment Monitoring Agency is assigned by the corresponding legislative acts as the center of operational data accumulation, storage and the primary analysis. Within the framework of the project the works on Murmansk territorial ARMS elaboration including the development of new and updating the existing software of the system, the installation of 23 new automaticdose-rate measuring points at area of the region, the installation of 9 automatic meteorological stations, the installation of modern computer and communication equipment in Murmansk administration of the Federal Hydrometeorology and Environmental Monitoring Service have been carried out.

Location of monitoring stations of the Murmansk Regional Radiation Monitoring System is given in Fig. 3.

Under the initiative of Rosenergoatom Concern the integration of the existing ARMS of the supervised area of the Kola atomic power station in the territorial system of Murmansk region is provided. Simultaneously the staff of the local Crisis centre of the Kola NPP has got the access to the measurement data of the territorial ARMS of Murmansk area.

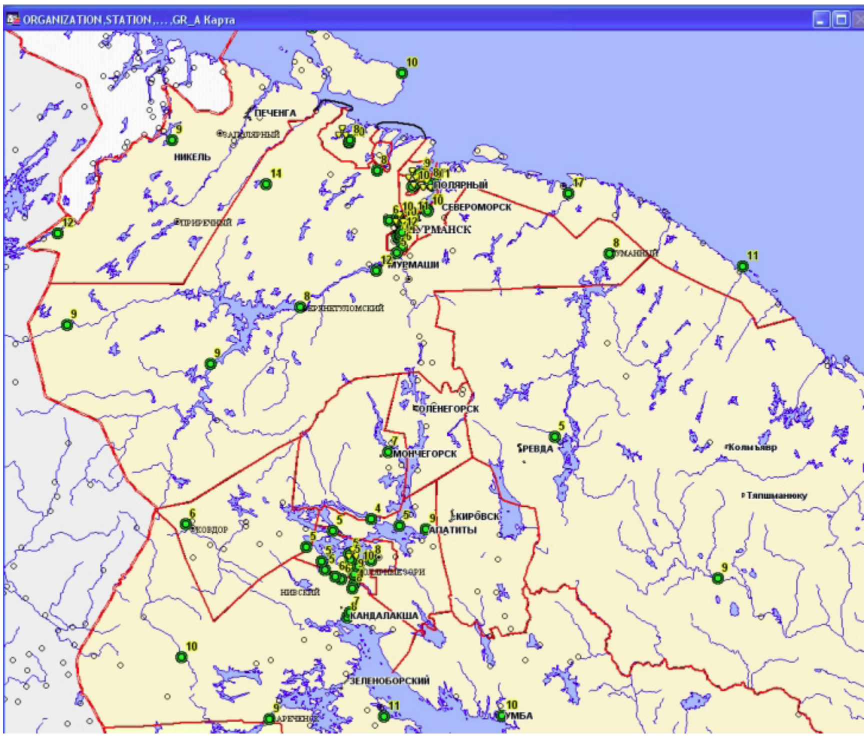

Fig. 3. Location of monitoring stations of the Murmansk Regional Radiation Monitoring System.

Mobile radiation surveillance Laboratories(MRL)

Four mobile radiation surveillance laboratories (MRL) are designed for operative radiation surveillance duties in case of radiological emergencies.

Capabilities: detection and localization of radioactivity sources and contamination; sampling and express-analysis of soil, air and water samples; determination of the characteristics of contamination; mapping of the boundaries of contaminated areas; transmission of measurement data to crisis centers in a real-time mode.(fig.4).

MRL equipment includes: measurement equipment stationary and portable gamma-spectrometer installations, dosimeters, alpha-, beta-, and gamma radiometers, sampling equipment; computer and communication equipment Inmarsat satellite terminal, cellular telephone communications, VHF radio station, GPS satellite navigation system, industrial computer and an auxiliary laptop, photo and video equipment; specialized software; auxiliary equipment vehicle power supply system, including petrol generator, and adaptation of the equipment to climatic conditions; working clothes and deactivation equipment.

Fig. 4. Location Mobile Radiation Laboratory(MRL). A fragment of data transferred from MRL System.

System of scientific and technical expert support

Now the Technical Crisis Centre of IBRAE RAS carries out the scientific and technical, expert support of Contingency Crisis Center of Rosatom, National Emergencies Management Center of the Ministry of Emergencies of Russia, the Crisis Centre of Rosenergoatom Concern, Information Analysis Center of Rostehnadzor, the enterprises, the regional bodies of the emergency situation prevention and elimination

The basic functions of Technical Crisis Centre at the Northwest of Russia are:

– Expert support of the Crisis Centers personnel and development of recommendations on minimization of emergency situation radiation effects for the personnel, population and areas of the region; Scientific, methodical and technical support of actions on forces and emergency response equipment readiness in the course of the exercises and trials included; Scientific, information, methodical and technical support for creation, development and introduction of new hardware and software systems for the support of administrative decisions concerning the protection of the personnel, population and areas in case of emergency situation.

The development of the territorial ARMS is carried out in Murmansk, Arkhangelsk, Kursk, Tver, Kaluga regions and Moscow in the framework of Rosatom, EMERCOM, Rosgidromet and IBRAE RAS cooperation as well as with Russian Federation Subjects agreements. There is a public site (Radiation Situation at Rosatom Enterprises) (www.russianatom.ru) created by Rosatom and IBRAE RAS. (fig 5).

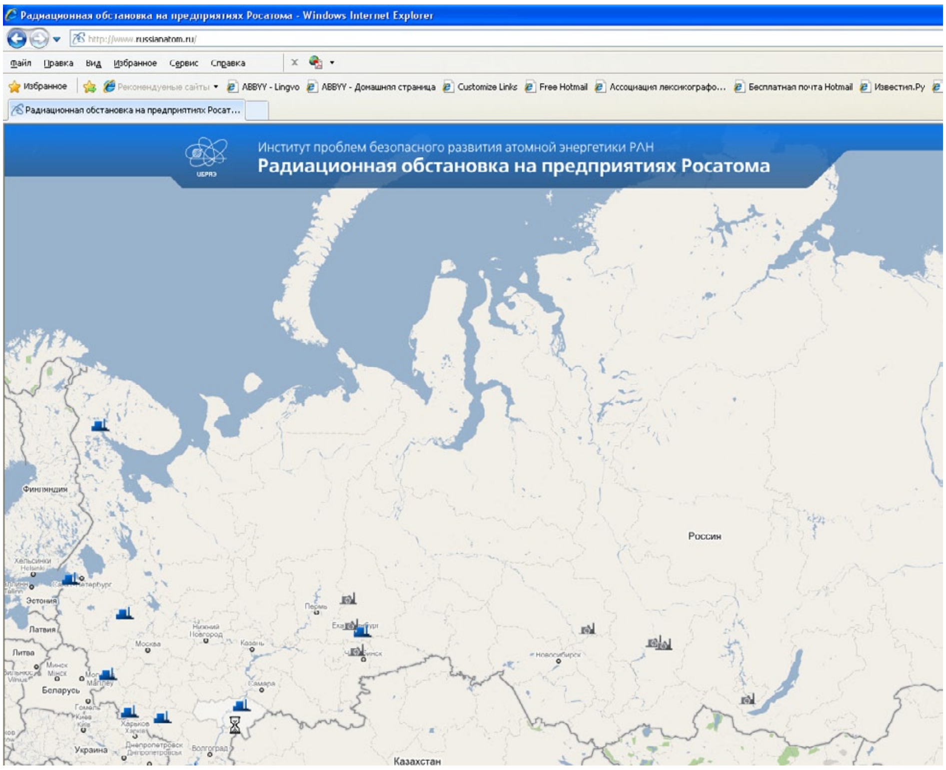

Fig. 5 The Internet Site Screen (Radiation Situation at Rosatom Enterprises).

Command staff exercise Zapolyarie 2007

Exercise "Zapolyarie-2007" in October 2007 demonstrated operability of all created elements and interaction of all participants of emergency response in the Murmansk Region.

All stated objectives of Zapolyarie 2007 Exercise have been reached. The interaction of emergency response system elements of Murmansk region and Murmansk territorial subsystem of the Unified State System of Emergency Situation Prevention and Elimination has been fulfilled. The exercise showed the steady work of all automated systems, communication lines under working in various modes. Data of current situation were operatively transferred to the exercise control body and all participants of the exercise (data from exercise area : branch No.1 of SevRAO, local and territorial ARMS, mobile laboratories). The interaction of all parts of emergency response system of Murmansk region and Murmansk territorial subsystem of the Unified State System of Emergency Situation Prevention and Elimination was completed.

The exercise was acknowledged as very successful by the management of the Murmansk Region and observers of IAEA mission. An IAEA mission, while assessing the emergency response status in the Murmansk region, confirmed a high level of preparedness and recommended the system established for further expansion.

Radiation Monitoring and Emergency Response System Upgrading in Arkhangelsk Region

In March 2009 the Agreement of the similar project implementation in Arkhangelsk region was signed with EBRD. There are some radiation dangerous facilities where works with Spent Nuclear Fuel (SNF) and Radioactive Waste (RW) are carried out. The key facilities are in Severodvinsk, Russia. Such facilities are: Zvezdochka Shipyard, Sevmash Shipyard, and Mironova Gora (radioactive waste storage facility).

Safety measures in the radiation emergency situation of the regional scale are taken at the local level in accordance with the Federal Law No. 68_ФЗ dated December 21,1994 and regulatory documents (the RF Government Decree No. 794 dated December 30, 2003). In case of emergency situation at the radiation dangerous facilities the Administration of Arkhangelsk region is to solve the problems of population protection, information delivery and intercommunication at the interregional level with the Administration of the neighboring Russian Federation Subjects, Authorities of the Federal Districts and Federal structures of the executive bodies. The Emergency Response Commission of Arkhangelsk region renders the decision on counter measures Implementation The system will comply with requirements of the Russian legislation and the international practice of the radiation monitoring and emergency response system organization.

One more key objective of the Project is the provision of the total radiation situation information to the population and public authorities at Arkhangelsk region and the transfer it to the neighboring states in accordance with the international obligations. The Project implementation will allow carrying out the sustained radiation environmental monitoring, the measurements for the short time radiation dangerous works execution period at the enterprises and obtaining data about the radiation doses at the areas contaminated as the result of the radiation accident.

In the framework of the emergency response system development the creation of the local and two site ARMS, the delivery of the MRL is of the highest priority. These systems are the important parts of the modern early warning and emergency response system for the enterprises and the Russian Federation subject. The system is to be integrated with similar system of Murmansk region.

The main scope of Project works will be completed for 20–24 months. It gives six months more for trial operation of subsystems, operators training and systems upgrading. Throughout 30 months the system is to be ready for operation as a whole. The system operation is to be carried out by owner.

Conclusions

NDEP-003 project is incomparable in Russia in terms of the covered territory, the number of radiation hazardous facilities and facilities participating in the international and Russian programs aimed at NS decommissioning and management of SNF and RW. The implementation of the Project provides the Murmansk Region with modern systems of radiation monitoring, informational, analytical and operative expert support of regional authorities in planning and executing activities on protection of the population in case of radiation accidents.

By considering the successful experience in Murmansk region the State Corporation Rosatom management included the works concerning the creation of similar systems in the regions of nuclear and radiation dangerous facilities location in the Federal Target Program (Nuclear and Radiation Safety in 2008 and to 2015).

Nordic Emergency Preparedness (NEP) – A regional concept for emergency planning

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Abstract
The Chernobyl accident 1986 lead to number concrete steps that were taken worldwide aiming at enhancing the preparedness for Nuclear and Radiological Emergencies. One of these steps was the establishments of bilateral agreements and associated practical arrangements between Denmark, Norway, Finland and Sweden on Early Notification and Information Exchange. This in turn lead to an understanding of the feasibility of setting up an organised form of cooperation between the Nordic nuclear and radiation safety authorities on emergency preparedness matters. This cooperation was formally established in the year 1993 and has over the years rendered a regime for regular communication tests, good cross boarder knowledge and understating of the set-up in our countries, detailed arrangements for the practical implementation of the bilateral agreements and a lot more. The presentation will describe the historical background of NEP, show its present agenda, work process and highlight some of results achieved. The overall objective of the presentation is to, by using the NEP cooperation as an example, show the value a good and structured regional cooperation on an institutional level could add to existing international agreements and arrangements.

Incident and Emergency Centre of the IAEA

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Abstract

The Incident and Emergency Centre of the International Atomic Emergency Agency is the global focal point for preparedness, event reporting and response to nuclear and radiological incidents and emergencies irrespective of their cause.

The Centre continuously works to develop standards and guidance for strengthening Member States' preparedness; practical tools and training programs to assist Member States in promptly applying the standards and guidance; and organizes a variety of training events and exercises. Lessons learned from response to past emergencies form the basis for developing such standards, guidance and tools. Together with the experts from the Member States and the ICRP the IEC made efforts to develop the generic and operational criteria for application in preparedness for and response to a nuclear and radiological emergency. A rigorous examination of the response to past emergencies has shown that there is a need for international guidance on taking protective and other response actions and for placing a guidance in a context that is both comprehensive for the decision makers and can be explained to the public. So the criteria developed are accompanied by the plain language explanation. In addition these generic criteria are developed in the way that there application will ensure consistency with the concept of reference levels presented in the ICRP 103.

The Centre evaluates national plans and assists in their development; facilitates effective communication between countries; develops response procedures; and supports national exercises. The Centre provides access to multiple information resources; assess trends that may influence crisis and consequence management plans and response; and develops and continuously enhances methodology for identifying conditions needed for early warning and response. The Centre provides around-the-clock assistance to Member States in dealing with nuclear and radiological events, including security related events through timely and efficient services and the provision of a coordinated international response to such emergencies.

TMT Handbook – Triage, Monitoring and Treatment of people exposed to ionising radiation following a malevolent act

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[7] CLOR Central Laboratory for Radiological Protection, POLAND

Abstract
In the aftermath of the Chernobyl accident European national emergency response plans were focused on dealing with accidents at nuclear power plants. The perception of the increased threat has shifted the focus to being prepared also for malevolent use of ionising radiation. The European Commission through the Euratom Sixth Framework Programme was co-sponsoring a specific targeted research project aimed at producing a practicable handbook for an adequate response to such incidents (TMT Handbook). The handbook gives advice on how to prepare the response for such incidents and how to handle the situation both at the scene of the incident and at the hospitals. Advice on public health interventions including criteria for the long term follow up are also described.

1 Introduction
Until recently European national emergency response plans focused on accidents at nuclear power plants. Several terrorist acts carried out by disaffected groups have shifted the focus to being prepared also for malevolent use of ionising radiation aimed at creating disruption and panic in the society. The casualties of these kinds of acts will most likely be members of the public. The radiation exposure can range from very low to substantial and it could be combined with conventional injuries. It might also be the case that the magnitude of the incident is such that the national response capability is overwhelmed, calling for international assistance.

The European Commission through the Euratom Sixth Framework Programme was co-sponsoring the specific targeted research project TMT handbook. The main objective of this project was to produce a practicable handbook for the effective and timely triage, monitoring and treatment of people exposed to radiation following a malevolent act. The TMT handbook project started with the collection of already published material to identify useful practices and provide a basis on which to develop clear guidance that was consistent with the current knowledge and experience in this field. There are a number of recent publications that address some of the issues raised in responding to malevolent use of radiation. For example, ICRP Publication 96: Protecting people against radiation in the event of a radiological attack [1] focuses on protection criteria for responders and the public. US NCRP [2] provides a review of the consequences and management of terrorist events involving radioactive material. A number of international publications [3-9] including the advice for first responders [10] and for medical response [11] provide useful guidance on the early response and generic treatment options available. In addition to the published articles in journals [12, 13, 14] there are some national guidance and protocols available. However, the information on triage, monitoring and treatment of people is scattered in several documents, especially with regard to a malevolent event scenario. As no single reference source existed on dealing with these issues, the need for TMT handbook was apparent.

In order to produce the TMT Handbook a project consortium was drawn together including the Belgian Nuclear Research Centre (the project coordinator), the Norwegian Radiation Protection Authority, Radiation and Nuclear Safety Authority of Finland, the UK Health Protection Agency, the Central Laboratory for Radiological Protection of Poland and the World Health Organization. Enviros Consulting was acting as the technical secretariat for the project. The TMT handbook was drafted by the consortium members with input from subject-matter experts and circulated for feedback to European emergency response institutions that would play a part in the handling of malevolent acts using radioactive material. These institutions were given a consultation time with encouragement to test and evaluate the handbook content through national emergency response exercises and stakeholder consultations. A workshop was held in December 2008 in Lillehammer, Norway to obtain feedback from the end users on the content, structure and usefulness of the handbook before the final version was produced. The received comments and suggestions helped in improving and harmonising the handbook for the use in European countries.

The TMT handbook contains both general information and detailed recommended actions to be taken at the scene of the incident and in hospitals by specialised response teams in radiation protection, monitoring, dosimetry and medical management. It gives advice on how to plan the response for such incidents and how to handle the situation starting at the scene of the incident going through the response at hospital level and further to public health interventions including criteria for the long term follow up. It also provides guidance on public information and communication strategies. The TMT Handbook is also a useful tool for training purposes. The aim of this paper is to give an overview of the contents of the TMT handbook.

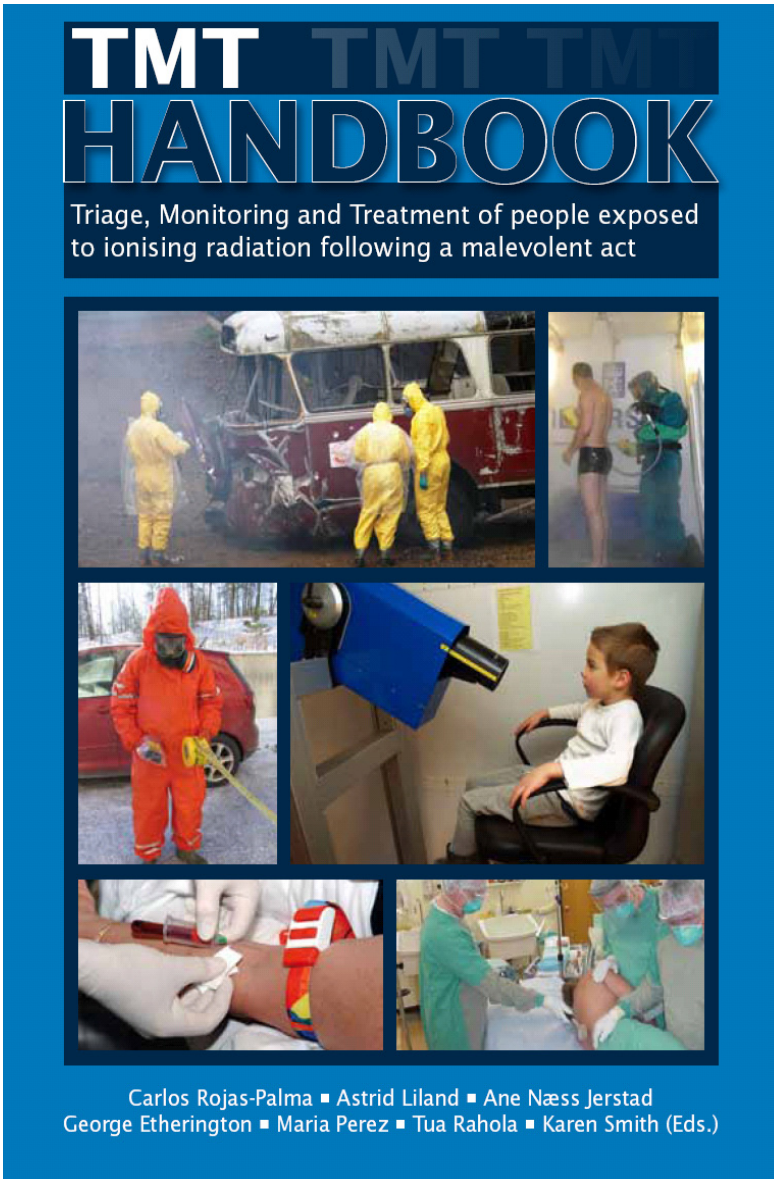

Figure 1. The cover of the TMT handbook.

2 Contents of the TMT Handbook

The content focuses on topics specifically related to the triage, monitoring and treatment necessary to respond to a malevolent act, while ensuring the appropriate protection of responding personnel. It is not intended as a complete check list for first responders, but instructions are given for actions and it can be used in the training of such personnel. Nor is the handbook intended to include exhaustive descriptions of medical treatment of conventional injuries since hospital staff is already trained for this. The handbook includes twelve color-coded chapters, annexes, references and a glossary.

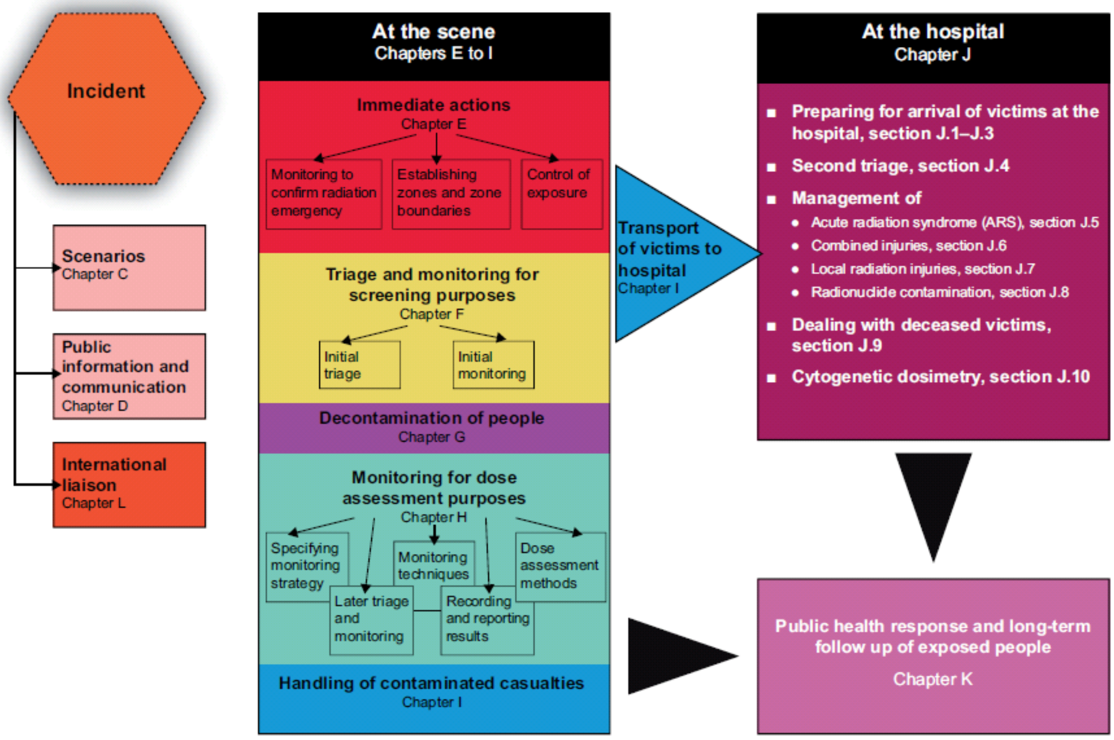

Figure 2. The content of the TMT handbook.

Chapter A introduces the handbook.

Chapter B gives an explanation on the handbook structure and guidance on how to use it. It is recommended to read this carefully before proceeding to chapters E to J. Chapters E-J give practical guidance for incidental and accidental situations. The structure of the handbook is such that the numbered instructions to be used by the incident responders are presented on the left hand pages, linked to relevant supportive information that is presented on the right hand pages.

Chapter C gives a short summary of possible malevolent scenarios. The chapter does not include detailed descriptions of scenarios in order not to facilitate malevolent acts by disaffected groups. The scenarios are not meant to indicate the probability or possibility of any such event actually occurring. Neither should it be assumed that the scenarios described are an exhaustive list of the possible incidents that could occur. It is the responsibility of each country to carry out its own threat assessments as a basis for developing national radiation emergency and response plans.

Chapter D gives general guidelines on public information and communication strategies. Public communication should be considered a key function in any response involving the malevolent use of radiation. For public communication to be credible and trustworthy, the organisation providing it must be seen as open and transparent. To be effective, public information response should be planned in advance. These plans will need to be integrated within the overall planning for managing malevolent acts and should detail the roles and responsibilities to be carried out during the response. It should also be recognised that there exist cultural differences between countries and therefore similar means and techniques for communication may not be effective in all countries, any approach would need to be tailored to the specific situation and location.

Chapter E describes immediate actions specific to the radiological emergencies to be taken at the scene mainly during the first hours after notification or discovery of an incident. These actions include monitoring to confirm a radiation emergency, establishing zones and controlling the exposure situation. In the initial stages of the response, there will be little time to carry out detailed planning of the response, and minimal information on which to base such plans. The actions given in this chapter may be implemented automatically without the need to develop plans that are specific to the incident. It is advised, however, to be familiarised with the handbook content before using it in the field.

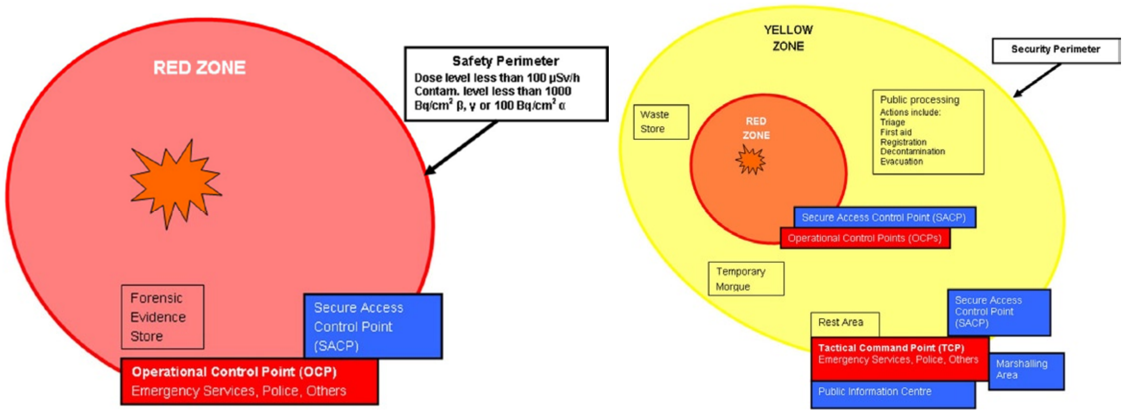

Figure 3. Generic layout of the red and yeallow zones.

Chapter F is dedicated to the best practice for triage and monitoring of people for the purpose of screening. "Triage" is the use of simple procedures for rapidly sorting people into groups based on (a) their degree of physical injury and (b) on actual or potential radiation effects on health, and the allocation of care to these people so as to expedite treatment and maximise the effective use of resources. Conventional trauma triage may be required following incidents involving the malevolent use of radiation or radioactive material in a public place. However, the scope of triage is broader for such incidents and includes a group of actions that can be termed "*radiological triage*". These actions are intended to sort people rapidly into groups depending on actual or potential effects on their health resulting from radiation exposure. In the handbook the objectives of the triage process are presented. The term "*monitoring*" describes the measurement of radiation dose or contamination for reasons related to the assessment or control of exposure to ionising radiation or radioactive material, and the interpretation of the results.

Figure 4. Left: "Lap geometry" for whole body measurement with portable gamma spectrometer. Right: Measurement of ^{131}I in the thyroid. These photographs illustrate the method, but when responding to an incident, additional measures would need to be employed, i.e. the detector would be wrapped in plastic film, and the subject would wear disposable gloves. Photos: STUK.

Chapter G gives practical advice on decontamination of people in the field. In this chapter only the removal of radioactive contamination is considered. People who have only been externally irradiated do not require decontamination. Decontamination should be carried out as soon as possible but does not require the same immediacy as chemical or biological contamination, except in extreme circumstances where the contamination is sufficient to cause deterministic effects. People involved in an incident where radioactive material is present in the environment will be prioritised for decontamination using the procedures detailed in chapter F. Decontamination of injured people will take place either in hospital or adjacent to the incident, depending on the severity of injuries.

Figure 5. Demonstration of decontamination of people who need assistance. Photos: HPA.

Chapter H provides instructions for monitoring of ionising radiation for dose assessment purposes. The TMT Handbook is mainly concerned with individual monitoring, but the other forms of monitoring (e.g. source monitoring, environmental monitoring) also come within the scope of the handbook. Individual monitoring is monitoring using measurements of quantities of radioactive material in or on the body of the individual, or measurements made by dosimeters worn by individual workers. It includes the assessment of radiation doses from the results of such measurements. The main objectives of monitoring are: to quantify absorbed doses to organs and tissues for people exposed to radiation at a level high enough to potentially give rise to deterministic health effects, to provide the dosimetric information that would allow urgent decisions to be made to remove individuals from a source of external exposure, or to remove or reduce contamination on or in the body, to quantify committed effective doses for people with lower levels of internal contamination that could result in an elevated risk of stochastic health effects, to provide dosimetric information that could be used when making decisions on medical treatment and to quantify committed effective doses for people whose exposures are very unlikely to have an effect on health.

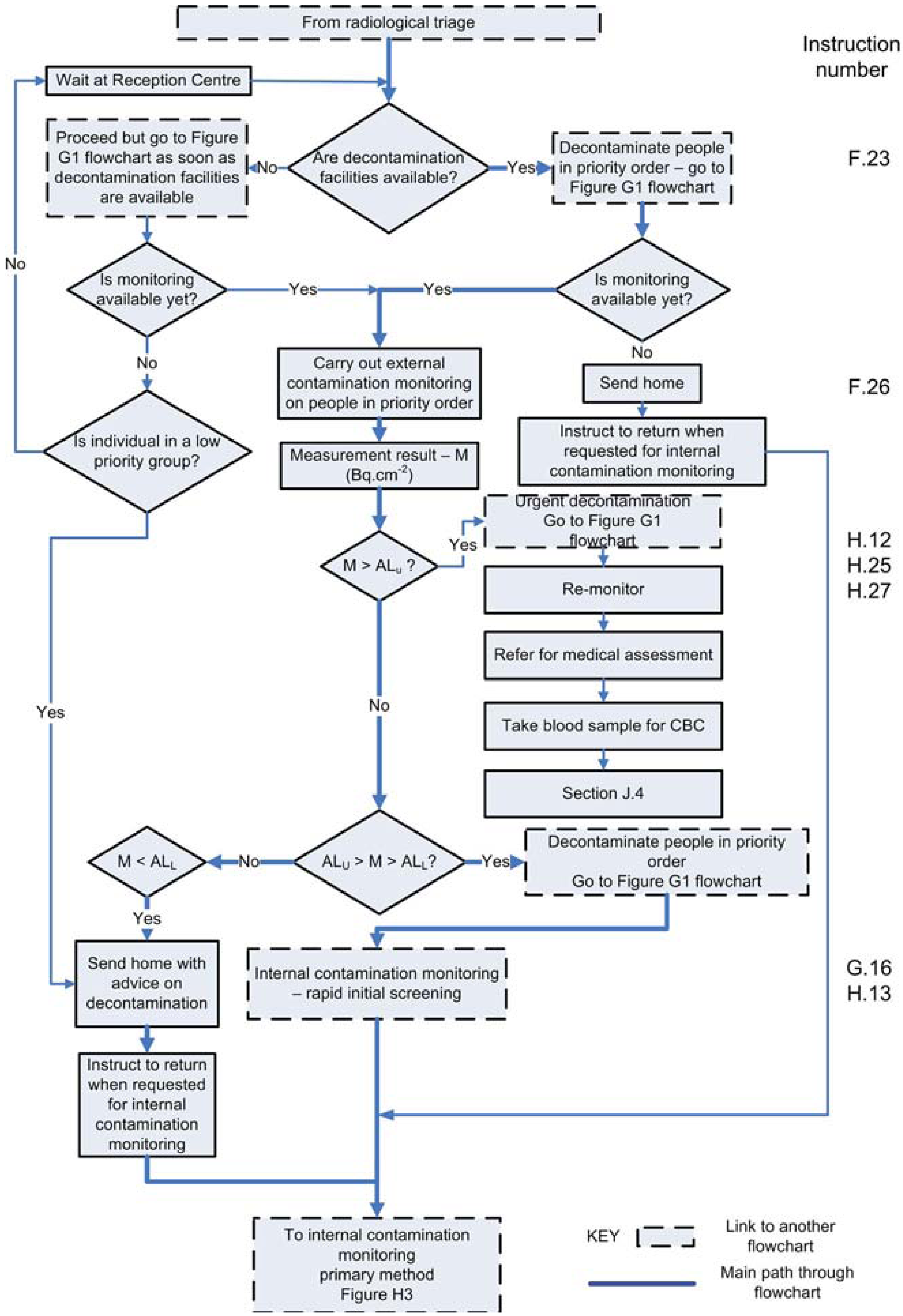

Figure 6. An example of the flowcharts in the handbook. This flowchart shows radiological triage based on results of external contamination monitoring.

Chapter I gives advice on handling of contaminated casualties and transport to hospital.

Chapter J is addressed at medical doctors, nurses and other health-care workers who may have to manage people affected by events involving the malevolent use of radioactive sources and would be responsible for actions to be taken at the first referral hospital level concerning diagnosis, prognosis and treatment and health-care facilities management. The chapter includes the information on management of acute radiation syndrome (ARS), local radiation injuries, combined injuries and internal contamination including management of contaminated patients and decorporation techniques. The chapter highlights the critical links between pre-hospital and hospital response.

Figure 7. While conservative treatment may be indicated for superficial lesions, painful deep ulcerations and necrosis require surgical treatment. These photos show the evolution of tissue necrosis after surgical treatment including artificial skin graft. Photos: courtesy of Percy Hospital and IRSN. For artificial skin graft photo, permission also provided by the IAEA.

Chapter K provides guidance on public health response, including the role of health authorities during the emergency, initial actions to be taken, management of outbreaks of unusual disease attributable to radiation exposure, and criteria for long-term follow-up. During radiation emergencies resulting from malevolent use of radioactive sources, health authorities should make provisions for dealing with a large number of people who may self-report experiencing symptoms or even as asymptomatic patients. Some people may be concerned about consequences of possible exposure to radiation, even if not actually exposed (“worried well”). Prevention and management of psychosocial effects are also addressed in this chapter. Psychosocial impact is one of the terrorism’s chief aims.

Chapter L provides information on existing arrangements based on the international conventions for early notification and assistance in case of radiological and nuclear emergencies. The proper handling of serious incidents or situations where prompt response is warranted in order to mitigate the effects of a perceived hazard, may require resources that challenge the capabilities of a single country. It is therefore important for countries to co-operate in order to better respond to such emergencies. The international assistance arrangements are set up through formal mechanisms such as the IAEA's Emergency Conventions, the WHO's International Health Regulations. These mechanisms provide for coordination of international arrangements, while not necessarily eliminating the need for bilateral or multilateral agreements between countries relating to the information exchange or assistance.

Annexes 1-14 contain a lot of supplementary information and practicable look-up tables for both the planning and the response phase:

Annex 1: Required facilities
Annex 2: Equipment required for radiological triage and monitoring purposes
Annex 3: Forms, questionnaires and information leaflets
Annex 4: Allocation of roles
Annex 5: Interpretation of clinical signs and symptoms
Annex 6: Specifying a monitoring strategy for internal contamination
Annex 7: Later triage and monitoring
Annex 8: Monitoring techniques
Annex 9: Biodosimetry
Annex 10: Action Levels
Annex 11: Sampling of excreta and blood
Annex 12: Management of internal contamination
Annex 13: Look-up tables for the assessment of internal doses
Annex 14: Methodology applied by WHO for developing guidance on health interventions for Chapters J and K of this Handbook

3 Dissemination of the TMT Handbook

At the start of the TMT Handbook project, an information leaflet was sent to all European emergency response organisations with specific functions to plan, coordinate and execute mitigating actions in response to malevolent acts involving ionising radiation. The leaflet informed about the project and encouraged organisations to join as end user. The role of the end users was to give feedback on a draft version of the handbook and, if possible, test the handbook in national emergency response exercises. End users from 16 countries participated in the project and gave valuable input to the draft both by correspondence and by participation at a feedback workshop arranged in Norway after a consultation period of about 6 months. This gave valuable input for adjusting the layout and content for the final, printed version.

A training course based on the TMT Handbook was held in February 2009. The training was directed primarily to representatives of national emergency response organisations with responsibility for first response in emergency situations, hospitals and wider health-care infrastructure, such as public health authorities. The aim of this course was to enable participants to better understand the principles of management of malevolent events involving exposure to radiation, to strengthen national capabilities for planning and response to such events and to encourage participants to promote the incorporation of the TMT Handbook into exercise and training programmes in their countries. The course also provided a platform to identify common challenges and discuss opportunities for harmonised and coherent response strategies within the European Union.

The TMT Handbook was published in April 2009. The electronic version is available on the project's web page: www.tmthandbook.org upon registration. More than one thousand users from 57 countries on five continents have downloaded the handbook and nearly two thousand hard copies have been distributed by the summer 2010.

The TMT handbook will contribute to a harmonisation across Europe of the approaches to triage, monitoring and treatment of people exposed to radiation following a malevolent act. It will be also a tool to build capacity in the region as well as beyond Europe to be incorporated into training programmes at national, regional and global level.

Acknowledgements

This work was partially supported by the Norwegian Research Council and the European Commission Euratom Sixth Framework Programme through grant number FP6- 036497. We are grateful to all European emergency response organisations that agreed to test and evaluate the handbook as end users.

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Norwegian assessment of current national nuclear and radiological preparedness

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Summary

In 2004, the Norwegian Crisis Committee for Nuclear and Radiological Preparedness initiated a project to assess the current level of national preparedness regarding nuclear and radiological emergencies. The purpose of the initiative was to make recommendations on how to further develop preparedness within the Norwegian nuclear and radiological preparedness organisation and to improve current national emergency response planning. The project is expected to be completed during the summer of 2010.

The first phase of this project was finalised in the autumn of 2008 and constitutes a comprehensive hazard assessment. The second phase of the project will address two issues. Firstly, based on foreseen scenarios and relevant consequences from the study in phase one, it will study management of nuclear and radiological emergencies. It will also review the roles and responsibilities of different regional and national authorities and explore the practical implementation of different mitigation efforts. Secondly, resource requirements within the nuclear and radiological preparedness organisation will be addressed.

Phase 1: Assessment of nuclear and radiological hazards

The first phase of the project points out changes in the Norwegian perception of nuclear and radiological hazards over the last few years. It provides an overview of possible nuclear or radiological events which may affect Norway or Norwegian interests abroad, as well as relevant consequences for public health, the environment and other public interests.

The assessment of nuclear and radiological hazards was published in the autumn of 2008 as NRPA Report 2008:11 (in Norwegian).

Phase 2: Adequate management of nuclear and radiological emergencies and resource requirements

Roles and responsibilities

Norwegian preparedness for nuclear and radiological emergencies differs from most other national emergency preparedness systems in Norway. In order to ensure an efficient, rapid and competent management of the early phase of a nuclear event, a national Crisis Committee for Nuclear Preparedness has been appointed. The Committee is authorised to make decisions and order implementation of specific countermeasures in the early phase and ensures good coordination on a sub-strategic level (directorate level).

Phase 2 of the project reviews and clarifies the roles and responsibilities of the governmental bodies involved in the nuclear emergency organisation and addresses the relationship between them.

Dimensioning of Norwegian nuclear and radiological preparedness and crisis management

There is an evident need for a common understanding of the basis for choices made regarding emergency preparedness development. Phase 2 of the project establishes a set of six general dimensioning scenarios qualitatively describing the consequences of a nuclear or radiological incident. These scenarios enable a better prioritising of current needs and have been approved by Norwegian political authorities.

Emergency management timeline and nuclear emergency phases

Current Norwegian nuclear and radiological emergency management adheres to the “acute” and “late” categorisation of emergency phases. These phases are poorly defined in existing emergency plans and the transition between the phases is poorly developed. The police and medical services in particular have a different intuitive understanding of the term “acute” than other governmental bodies involved in nuclear emergency preparedness. A more thorough description of the emergency timeline is therefore recommended.

Necessary management

Phase 2 of the project reviews necessary management of nuclear and radiological emergencies and examines mitigation strategies. Different actions are defined and categorised and practical implementation of different mitigation efforts is explored.

Resource requirements

In the last few years, there has been a positive development concerning warning routines and measurement capabilities. Phase 2 of the project recommends that future preparedness development has emphasis on competence, organisation, plans and exercises.

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Collaborative software for the nuclear emergency management

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Abstract

A lot of expertise and teamwork is needed to understand an evolving nuclear accident, to assess its off-site consequences, and to make optimal recommendations. The working of the various groups involved relies on information exchange and on adequate computer support. In Finland we have been developing a Web platform that makes it easier to exchange information and for all users to be kept up-to-date with the development of the situation. The platform bridges the spatial remoteness of expertise (incl. hardware and software) and provides a consistent user interface to tools that could be used before only with special training. In addition, it provides the different users with their views of the situation and supports the process from source term input, over dispersion and dose assessment, up to the approval of various reports. Although still developed, the platform has been in use in the Radiation and Nuclear Safety Authority of Finland (STUK) and the Finnish Meteorological Institute (FMI) for roughly two years. So far we have received mainly positive feedback.

Introduction

Emergency management of nuclear accidents is a collaborative effort: one team assesses the plant status and makes an assessment of the likelihood and magnitude of a release; another team is in charge of making dispersion calculations with the given release assessment; their results are given to yet another team so that they can plan suitable measurement campaigns; still another team assesses possible health effects and the need for interventions by taking into account the most likely dispersion situation and already available measurement results; and there is a need for coordinating all this effort and communicating to the public, which in turn can only be done successfully if first hand and timely information about the current state of affairs and about likely future developments are readily at hand. Of course collaboration does not stop here but this does roughly picture the collaboration that is needed at STUK and FMI to cope with an event.

Collaboration, however, is not something that happens spontaneously during an event. It has to be planned for and arrangements have to be in place. STUK and FMI have developed therefore a Web application that allows them to manage, view and share the results of dispersion and dose calculations and other information related to nuclear or radiation accidents (Lahtinen et al. 2008, Ammann et al. 2008). The application helps them to produce timely reports of the radiological situation and to plan protective actions. In addition, it keeps record of all relevant user activities and allows the simulation of measurement data.

Ketale collaborative software

The main purpose of the Ketale system is to support the management of nuclear or radiological emergencies at STUK and FMI. The use cases of Table 1 are typical in this context.

Table 1. Typical uses cases of the Ketale system during emergency management.

1.	The duty officer creates an event ID.
2.	FMI issues weather bulletins and recommends the most appropriate numerical weather prediction dataset to be used in subsequent dispersion calculations. FMI also provides weather radar images, met tower measurement readings, and other weather related products.
3.	The radiological consequence assessment group requests trajectories.
4.	The accident assessment group makes a release assessment.
5.	The radiological consequence assessment group requests dispersion and dose calculations for the given release assessment. The group compiles a report with various forecasts of the radiological situation.
6.	The recommendation planning group plans protective actions and compiles a recommendation report.
7.	The management group views and evaluates the various reports.

STUK and FMI work together in this process. FMI operates numerical weather prediction models, provides access to their long-range atmospheric dispersion model and offers other weather related products. STUK, on the other hand, keeps contact to the accident site, makes release assessments, compiles all data into various reports and recommends protective actions. Within these institutes the work is further distributed over different groups, most importantly for our purposes an accident assessment group for release assessments, a radiological consequence assessment group for dispersion calculations, and a recommendation planning group for the planning of protective actions.

Next to the human aspect of collaboration there is a software and hardware aspect as well. Experts use all sort of tools (source term assessment codes, dispersion models, environmental transfer models, dose models, etc.) when making their assessments. Some of these tools work only on particular operation systems and on dedicated hardware. For example the long-range dispersion code runs on a supercomputer at FMI, and the dose model is a traditional single-user Windows program. Users need various communication software (e.g. FTP, email) to transfer results between tools and other users; and they need office tools, GIS, image processing software, and other productivity software for various purposes. An additional complication is the need for authorization and authentication in order to gain access to the various resources. In practice, it can be therefore quite challenging to timely produce the high-quality reports that are required.

In summary the following observations can be made:

- Emergency management is a collaborative effort. An easily accessible and constantly updated audit trail of all activities such as model requests or communications is needed.
- Modelling applications and expertise are distributed (weather forecasts and dispersion predictions by FMI, source term and dose assessments by STUK) and have are integrated.
- Modelling applications run on different hardware and software and often have scientifically motivated user interfaces. The users, however, should not have to care about these peculiarities.
- Modelling applications often produce static images that are hardly suitable for all users in all cases.

The Ketale system was created to streamline the data and information exchange between the various groups involved in the process. The goal was to create an application that integrates the distributed modelling applications and facilitates collaboration and sharing of information. It is a Web application that complies with the following user requirements (Lahtinen et al. 2008):

- Collaboratively multi-user: users work together on a single case and need access to remote resources;
- Ease-of-use: there should be only very few client side requirements; the system should not require any special or advanced computer skills; different users need different views of the system;
- Multi-purpose: emergencies, exercises, training, measurement simulation, comparison exercises, source detection should all be supported; the system should have at least European wide coverage;
- Multi-lingual: Finnish, Swedish, and English should be supported at least as far as products and reports are concerned;
- Reliable: the system should be fit for operational use in emergency centres; this must be assured by constant quality control;
- Open: data import and export should be easy; the system should integrate well with other systems;

Operational experience

Feedback from emergency exercises is the main vehicle to constantly improve the system and make it more apt to the actual process of how emergencies are handled. The most recent operational experience comes from the Loviisa preparedness exercise held in March 2010. This exercise demonstrated quite well the important role that Ketale has today in the emergency arrangements of STUK and FMI. During this exercise 11 Ketale users collaborated in the production of 31 different reports, and many more participants used Ketale to keep track of the accumulation of information. The system was used to exchange 7 different release assumptions, which subsequently were fed into both far-range and near-range dispersion and dose models. The results of these model runs were again available to all Ketale users.

Expectedly, model results changed with new release assumptions; but they also changed according to the chosen numerical weather prediction model or dispersion model. During and after the exercise the contamination pattern that was estimated by the long-range dispersion model was compared with those of three different near-range dispersion models (all calculations were basically done with the same release assumptions and numerical weather prediction dataset) and they all differed to various extents. This raised the problem of how to choose the most suitable weather prediction and dispersion models, or how to unify such disparate results and communicate the underlying uncertainties to other users. Currently Ketale does not offer a solution to this problem.

Although positive in its general tenor, user feedback pointed out several other areas of improvements. Most notably the rather new features of planning protective actions, on-line report editing and the integration of measurement data seem to need additional attention. For the first time the simulation of measurements was available during the exercise. A Ketale user kept all her activities private and feed simulated measurements into STUK's monitoring software, from which it was available to other Ketale users as well. This feature was widely appreciated as it allowed testing the use and usability of monitoring data.

Discussion

In several exercises before the introduction of Ketale it was recognized that the handling of dose and dispersion calculations was quite unsatisfactory and that it took far too long to produce reports that were needed to brief decision makers or their senior advisors.

First of all, most of the modelling applications were not very user-friendly. Models were difficult to use and they often required special operation systems (e.g. UNIX). Their graphical results were mostly static maps with poor geographic details and lacking annotation. The maps also varied with colour, projection, grid size, etc., which made it difficult to compare results from different programs.

Secondly, the modelling applications did not interoperate well or not at all and it was difficult and time consuming to produce reports of the radiological situation. It took typically over an hour to produce a first report with a map of the area of risk and relevant weather descriptions, which was considered too long a time.

Information exchange between FMI and STUK happened mainly by telephone request and FTP transfer of the results. Telephone requests, however, had major drawbacks as lists of phone numbers had to be maintained, messages could be easily misunderstood, and the process worked only if there was a counterpart present on the other side. Luckily FMI operated a 24/7 person-on-duty service, though. In addition there was the problem of how to convey the content of the telephone conversation to other participants.

There were no technical arrangements in place to communicate source terms – that is, data on the amount and nuclide composition of the release – to different dispersion models. Each model had its own ways (and limitations) of dealing with source terms so that the procedures of providing source terms to these models were rather cumbersome to follow.

Another issue that was not sufficiently supported was the planning of countermeasures. It relied on generic GIS software, which was not linked well with other software. For example, though desirable, it was not possible to display model results as a backdrop map when planning intervention areas. Furthermore, manual cut & paste procedures had to be followed in order to get the resulting images into a report.

Eventually, modelling data and reports were not stored in one place but instead were distributed in several places. After the exercise it was difficult therefore to analyze the case.

On the whole, the reliability of the whole process was unknown. Major exercises are arranged quite rarely (about once a year) and accidents can happen at any moment. That is why it is important to regularly check the availability of all tools. Routine tests of such non-automated procedures are quite time-consuming to perform, however, so that they were not often enough made. This left the users often in the awkward situation that they did not know whether their tools will work or not.

Many of the just mentioned shortcomings could be remedied. A trajectory model and a long-range dispersion model, both from FMI, are coupled to the Ketale system as is a dose model from STUK. Results from these models can now be transparently requested by filling in and submitting Ketale forms, the results can be displayed interactively, and suitable portrayals can be added effortlessly to summary reports. The requests appear on a notification page and are accessible to all observers.

The major advantage compared with the situation before the introduction of Ketale is the significant improvement in the time it takes to create reports. It takes now less than 10 minutes to create a summary report containing for example a map of the dispersion area and a textual weather description. Also the quality of graphical outputs is improved. These improvements were achieved by automating some steps that previously had to be done by hand with word processors and image manipulation programs. Ketale’s translation feature is another major benefit. By changing the language settings of the web page it is very easy now to create reports in English in addition to reports in Finnish.

Conclusions

Information exchange between STUK and FMI has mostly been automated. It is now documented and constantly tested. Testing does not only allow increasing the reliability of the preparedness tools and communication channels, but also allows putting a reliability index on the availability of the system. This is useful for quality assurance purposes.

Ketale was found to be very useful in the exercises held so far. The time needed to produce reports for senior advisors or decision makers has decreased significantly, and their quality and consistency has improved substantially. The countermeasure planning page was highly appreciated by the recommendation group.

But Ketale was not only found to be useful, it had also a considerable effect on the emergency preparedness organization and on how the process could be conducted. Prior to the introduction of Ketale, far too much expertise had to be diverted from producing assessments and recommendations to the technical details of the process (how to get data from here to there, how to produce maps, etc.). Formerly STUK needed to maintain trained personnel for the various modelling applications, now this demand has almost vanished. Ketale provides a consistent user interface to the modelling applications and hides all technical peculiarities. Questions like: Where is program X installed? What is the user account? How do I get the results into the report?, do not have to be asked anymore.

The application supports the process from source term input up to the issue of reports. In doing so it preserves a complete audit trail. Data and information exchange is streamlined, transparent, traceable, and routinely tested. Reports are produced much faster than before, they are standardized and better deliberated, and data portrayals are tailored to the needs of the users. Although still developed, the application has been in use in STUK and FMI since 2008. So far we have received mainly positive feedback.

Acknowledgements
There were many more people involved in the project than could be accommodated in the list of authors. We thank the project members K. Alm-Lytz, L. Pöllänen, and H. Lemmelä from STUK and J. Vira, J. Korhonen, M. Visa, M. Aalto from FMI for their valuable contributions. We thank the members of the steering group H. Aaltonen, S. Salomaa, R. Mustonen, and H. Toivonen from STUK and H. Juntti, J. Korhonen, and T. Tourula from FMI.

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Assessing the hydrological impact in nuclear emergencies

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Abstract

The paper presents several aspects believed to be relevant for the integration into the decision support systems for the management of radiological emergencies, of assessment tools addressing surface water contamination. A few exemplary cases are discussed, relating to domestic and international alert exercises and including, in the major league, tools like RODOS and MOIRA, both developed by multinational consortia; and, in the domestic league, solutions trading off scientific appropriateness in grasping the complex phenomenology involved, with computational practicality, mainly revolving around a radiological assessment toolkit, developed as an open-ended software that assemblies requisite source term evaluation, environmental transport, dosimetric diagnose and countermeasure projections. Tools complementarity and synergy; model streamlining; and interfaces comprehensible and palatable for the stakeholders appear to be the key factors paving the way of the aquatic path into the emergency response business.

Introduction

The National Institute of Physics and Nuclear Engineering (IFIN-HH) in Bucharest was the Romanian partner of election for the European consortium that has developed RODOS (Real-time, Online DecisiOn Support System for the management of nuclear emergencies in Europe); In so doing, institute's underlying strategy has rested on:

(i) pursuing a substantive participation in the development and promotion of a reference, mutually agreed upon, and thereby credible and internationally-credited decision support systems in the case in point RODOS and MOIRA (A MOdel-based computerised system for management support to Identify optimal remedial strategies for Restoring radionuclide contaminated Aquatic ecosystems and drainage areas);

(ii) developing, parallelly, domestic knowledge and data bases as well and innovative tools - methods and software, in order to emulate, at various levels of complexity, and get familiar with, the functions embedded in the reference tools; and ensure a meaningful participation in hypothetic incident analysis and periodic national and international drills.

Two decades of tool development and applications have left many practitioners with the feeling that, while the atmospheric path of environmental and human radioactive contamination is confidently marked, the aquatic path is, comparatively, approached with reluctance and treated in much more idiosyncratic ways. The fact may be attributable, at least in part, to the modelling complexities of the transport and dispersion of pollutants in water, called to account for sensitivities to volumes, boundaries, shapes, flows, injection mechanisms, space and time scale of events etc., and the reservations of the emergency responders to confront such complexities. Indeed, the comparative look at the essentials in the surface water dispersion models may result in quite a list of input variables that may be compiled from the variety of minimal problems that challenge the analysis. A typical (yet non-exhaustive) set of problems may read as follows:

(1) A single port of cross-sectional area A_o (m^2) discharges a liquid pollutant of density ρ (kg/m^3) at a volumetric flow rate Q_o (m^3/s) at, or close to, the surface of a water body of depth H (m), density ρ_o (kg/m^3), and featuring a crosscurrent of speed U_a (m/s).

(2) A single port of cross-sectional area A_o (m^2) discharges a liquid pollutant of density ρ (kg/m^3) at a volumetric flow rate Q_o (m^3/s), under an angle θ (deg. of angle) from the horizontal, deep below the surface of a water body of depth H (m) and density ρ_o (kg/m^3). Determine the concentration field.

(3) A multiple port of length L_d (m), consisting of n identical ports, spaced at l metres from each other, each port d metres in diameter, discharges a liquid pollutant of density ρ (kg/m^3) at a volumetric flow rate Q_o (m^3/s), under an angle θ (deg. of angle) from the horizontal, deep below the surface of an ambient water of density ρ_o (kg/m^3), depth H (m), crossflow velocity U_a (m/s), L_r cross-length, and flow rate Q_r (m^3/s). Determine the downstream concentration field.

(4) Determine the steady-state bidimensional concentration field $C(x,y)$ (kg/m^3) resulting from an initial (representative near-field) source concentration C_i (kg/m^3) of a pollutant released at Q_o (m^3/s), of decay constant λ (1/s), in a river with uniform depth H (m), ambient velocity U (m/s), and slope s, knowing that, by virtue of either plume narrowness or river's wide channel, there is no significant plume/bank interaction.

(5) Determine the steady-state bidimensional concentration field $C(x,y)$ (kg/m^3) resulting from an initial (representative near-field) source concentration C_i (kg/m^3) of a pollutant released at a rate of Q_o (m^3/s), of decay constant λ (1/s), occupying a transverse river expanse between y_1 (m) and y_2 (m), in a river with uniform depth H (m), uniform width W (m), ambient velocity U (m/s), cross-current velocity U_a (m/s), and slope s, knowing that, by virtue of either plume's wide aperture or river's narrow channel, there is a significant plume/bank interaction.

(6) Determine the steady-state concentration field $C(x)$ (kg/m^3) resulting from an initial source concentration C_o (kg/m^3) of a pollutant released at a rate of Q_o (m^3/s), of decay constant λ (1/s), in a river with uniform depth H (m),

uniform width W (m), ambient velocity U (m/s), and slope s, knowing that the pollutant is short-lived and/or the conditions are highly unsteady (e.g. in case of an accidental spill, of an important river shear flow or cross-sectional turbulent mixing).

(7) Determine the time-evolving concentration field $C(x,t)$ (kg/m^3) resulting from an instantaneous discharge of M_o (kg) pollutant of decay constant λ (1/s), in a river with uniform depth H (m), uniform width W (m), ambient velocity U (m/s), and slope s.

(8) Determine the steady concentration field $C(x)$ (kg/m^3) resulting from a discharge of initial concentration C_o (kg/m^3) and flow rate Q_o (kg/m^3) pollutant of decay constant λ (1/s), in a marine (tide-featuring) estuary with uniform depth H (m), uniform width W (m), ambient freshwater velocity U_f (m/s), and slope s.

(9) Determine the concentration, assumed to be uniform, C (kg/m^3) resulting from a recirculated discharge of initial concentration C_o (kg/m^3) and flow rate q_o (kg/m^3) pollutant of decay constant λ (1/s), in a small lake, pond or reservoir of volume V (m^3), with a throughput flow rate q (m^3/s).

(10) Determine the time-evolving concentration field, $C(x,y,z;t)$ (kg/m^3), with z measured from the water surface, resulting from a discharge of initial concentration C_o (kg/m^3), flow rate Q_o (m^3/s) and pollutant decay constant λ (1/s), into an open sea advective current of velocity U (m/s).

(11) Determine the time-evolving concentration field, $C(x,y;t)$ (kg/m^3), valid within a water colomn of depth H (m), resulting from an instantaneous discharge of a mass M_o (kg), of pollutant with a decay constant λ (1/s), into an open sea.

Fig.1 presents some accidental releases to surface water as model output examples like: river release, weak interaction with banks; river release, strong interaction with banks; open sea, instantaneous release; estuary, continuous release. The DSS software employed is AIDRAM (Gheorghe and Vamanu, 2003; 2005).

Drills and tools

An international exercise targeting the response to a virtual (simulated) abnormal event with offsite consequences at the nuclear power plant Cernavoda, ConvEX-3 aimed at, essentially, comparing national capabilities to conjointly and consistently address the same occurence in the nuclear emergency business. On the domestic side, Oltenia 07 has tested near-site response capabilities in the case of a transborder accidental release at the Kozloduy NPP, on the Danube river in nearby Bulgaria. Again looking at Cernavoda, Axiopolis 09 challenged, under international monitoring, local authority capabilities to mount countermeasures to an abnormal release, in the order of a limited evacuation of the population. The impact assessment of the (virtual) radioactive release on the hydrological network, especially on the Danube river were looked at as obviously less familiar than the atmospheric release (Vamanu et al., 2010), and so were also the decision support tools called upon to assist in the process.

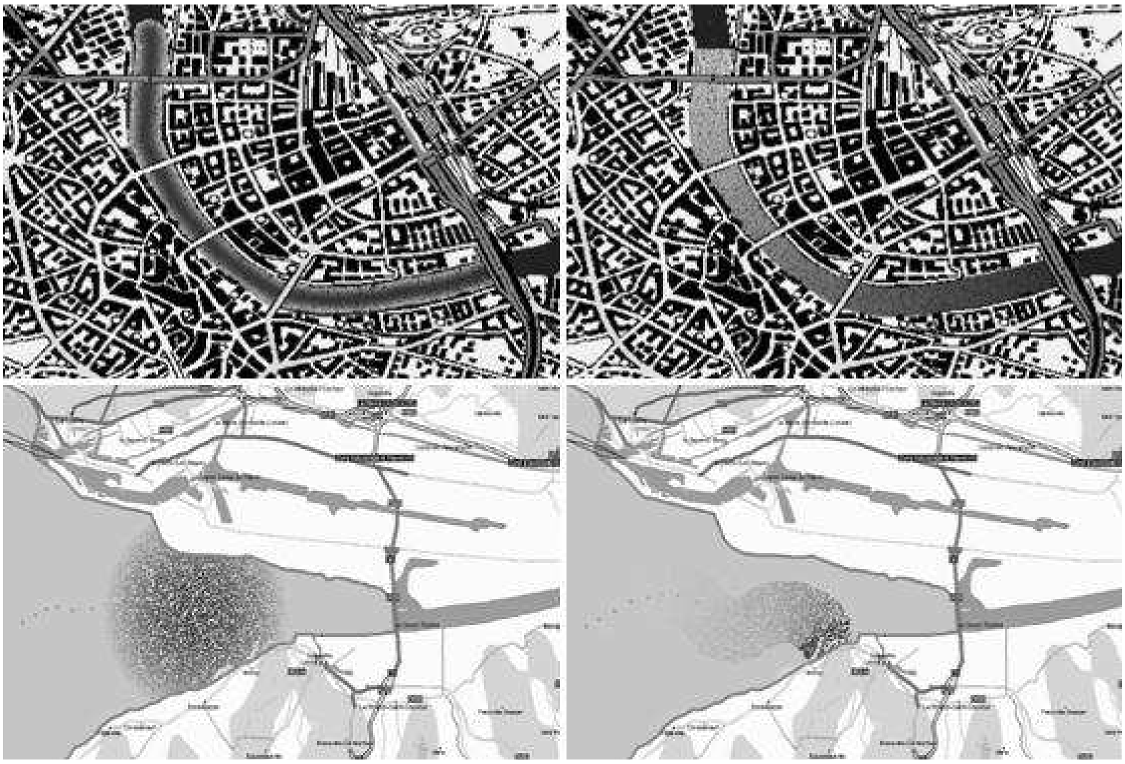

Fig. 1. Accidental releases to surface water as model output examples like: river release, weak interaction with banks; river release, strong interaction with banks; open sea, instantaneous release; estuary, continuous release.

A comprehensive software package developed by a consortium of European research institutions and promoted by EC as a reference DSS, RODOS (Vamanu et al., 2006; Rafat et al., 2006; Raskob and Ehrhardt, 2000) covers the early (1-7 days) as well as the intermediate and long (ingestion) phases (months, years) in the development of an accidental radioactive release, with the health-, environmental- and economic issues properly considered. The RODOS system includes a specialist module known as HDM, covering the relevant transfer processes in the hydrosphere. The HDM facility relies on the Saint-Venant equations for water flow modelling, assuming the flow as one dimensional. MOIRA (Magan and Gallego, 2006), on the other hand, is a management-oriented model to identify optimal strategies for restoring radionuclide contaminated aquatic ecosystems and drainage areas. The system is able to screen and evaluate a variety of strategies for the long-term management of contaminated freshwater bodies. MOIRA models the water transport process by the assessment of the input/output balance of water flowing to and from each reach, subdividing the river in 20 reaches. Both RODOS (RODOS Working Group, 2005) and MOIRA simulate processes of migration of radionuclides to sediment, resuspension and to biota. RODOS was customised by the Romanian (IFIN-HH) team in charge with its implementation only to Cernavoda conditions and was used in ConvEX-3 and Oltenia 07 exercises, whereas MOIRA was customised for all Romanian segments of the Danube and used throughout the entire series of exercises.

The first information requested by the decision makers concerned the estimated time of arrival of the radioactivity peak from ^{137}Cs due to deposition of the radionuclide on and near the Danube river, along with an assessment as to whether or not the concentration in water is harmful. In consideration of this, the assessment was designed in two steps:

(a) The determination of the radioactive peak arrival times, following the deposition of radioactivity carriers on water surface, at several cities along the Danube river. The computation was conducted using the RODOS – HDM (hydrological module) and emphasized the early time estimation of concentration in water for the isotopes ^{137}Cs and ^{90}Sr.

(b) The evaluation of the activity concentration in fish and water in the long run (months, years) due to the deposition and radionuclide migration from the catchment – this deed performed by the MOIRA system.

For the sake of illustration, figures 2 reproduce RODOS-HDM results over ConvEX-3 and Axiopolis 09 drills, regarding the concentration and peak arrival for ^{137}Cs and ^{131}I, respectively, at several major cities downstream the Danube.

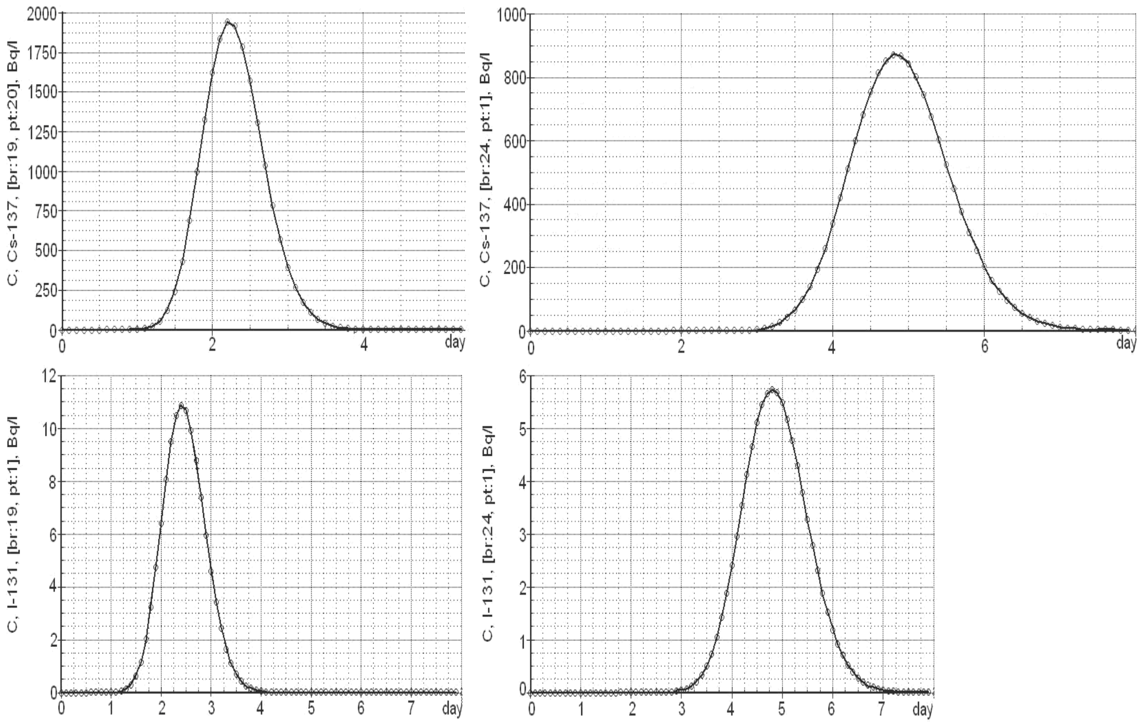

Fig. 2. CONVEX-3 and AXIOPOLIS 09. Peak arrival at several major cities downstream the Danube.

Figure 3 renders findings from MOIRA system runs, concerning ^{137}Cs concentration in water, prey and predatory fish in the first 20 and 40 months after deposition, due to migration of the radionuclide through river catchment and in the absence of any countermeasure. The results are obtained on the Danube segment near the Cernavoda – 20 months; and Kozloduy – 40 months, respectively. Another important result – coming from RODOS - concerned the evaluation of the radiological impact of Tritium – a nuclide expected to abound in releases from CANDU reactors like the one at Cernavoda.

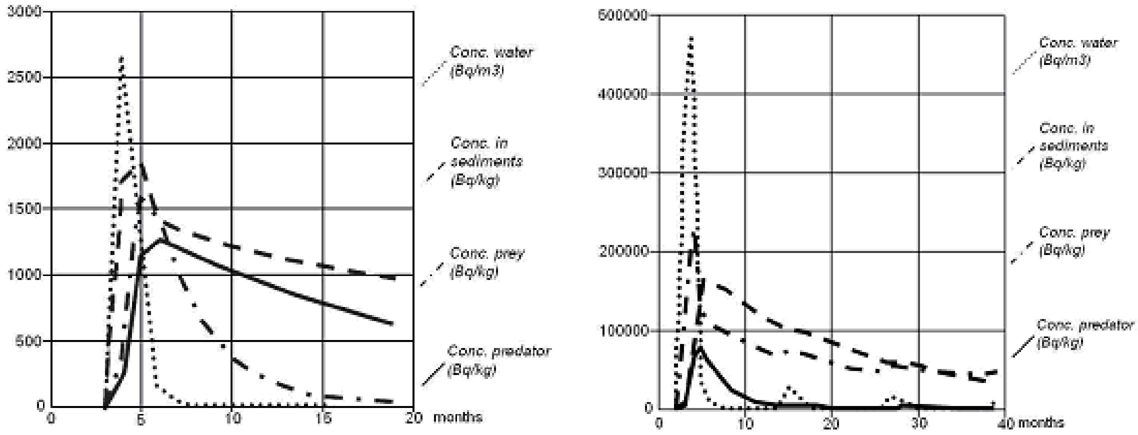

Fig. 3. CONVEX-3 ^{137}Cs concentration – left and, OLTENIA-07 ^{137}Cs concentration - right

Beyond the numbers and graphs however, what the response operatives seemed to appreciate even more during the exercises were the 'verbose' recommendations such as: "Proceed to banning the river water use, yet allow locally-catched fish consumption" – in the case on Conv-EX-3; or "Proceed to banning the river water use and locally-catched fish consumption" – in the case of Oltenia 07.

Such lessons on 'how-relevant-is-what, for whom', resulting from the assistants-operatives interaction added up into the folkore collection of rules of thumb in the emergency support ergonomics, currently open and indeed taken seriously at IFIN-HH.

Conclusions

Drill debriefings have consolidated a number of lessons, conducive towards a correct assessment of the current status and, in principle, to improving the effectiveness of the decision support in nuclear emergencies. Thus, attention was drawn to:

- the effective role and weight that tends to be retained, in actual practice, by Civil Defence and regulatory authorities for decision support systems in the response to abnormal nuclear events; and, in the context, the importance of listening to stakeholders and speaking their tongue;
- the complementarity and productive synergy of various systems - including RODOS, MOIRA and others, developed and customised for national compatibility;
- the merits of employing expert systems to overcome stress and psychological pressure, and avoid confusion or ill-fated decisions, in the early phase as well as in the aftermath of emergencies.

It also becomes increasingly evident that the proper implementation of DSS systems requires considerable volumes of custom environmental and socio-economic data be collected. The operational use may require that data be prepared ahead and updated. In fact, many views nowadays converge on *seeing a DSS as a tripolar, integrated simulation and visualization platform resting on three resources: the models; the data libraries; and the geographic information system (GIS) – all dynamically exchanging information at runtime in both inputing and outputing modes*. And, looming over horizons are the prospects for full 'webification' of the decision support tools.

While the importance of the internationally-accepted reference DSSs – a status to which RODOS aspires and was designed for – cannot and should never be underplayed, *the reality of national authorities and scientific establishments lending a confident ear to their own analytic tools should also be properly recognized*, and indeed be valued as a consolidating, rather than a dissipative, factor.

Concerning, specifically, the value of pursuing water contamination assessments as a consubstantial part of nuclear emergency preparedness and response management the following may stand out for consideration: the operation is definitely necessary, particularly in view of (i) the stealthy character of both environmental nuclide migration and exposure, on the aquatic paths; (ii) the remarkably long time spans involved; (iii) the proclivity of the paths for internal – that is aggravating, exposure. Given the complexities of the modeling, simulation and visualization techniques involved, the key success factor seems however to stay with the advent of a new kind of developer: one who is able to think like a scientist – yet speak like a stakeholder.

Aknowledgments

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SNIFFER: an aerial platform for real time measurements of contamination in the plume phase of a nuclear emergency

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Abstract

AGMS (Airborne Gamma Monitoring System), equipped with passive detectors, are used in many countries to face nuclear or radiological accident that results in a release of radioactive plume; however the AGMS is able to provide quantitative assessment on the radiological situation (land surface contamination level) only when the air contamination due to the passage of the travelling plume has become negligible.

To overcome this limitation, the Italian Institute of Health has developed and implemented a multi purpose air sampling system based on a fixed wing aircraft, for time-effective, large areas radiological surveillance (to face radiological emergency and to support homeland security).

A fixed wing aircraft (Sky Arrow 650) with the front part of the fuselage properly adapted to house the detection equipment has been equipped with a compact air sampling line where the isokinetic sampling is dynamically maintained. Aerosol is collected on a Teflon filter positioned along the line and hosted on a rotating 4-filters disk. A system of detectors allows radionuclide identification in the collected aerosol samples. The correlated analysis of the detectors data provides a quantitative measurement of air as well as ground surface concentration of gamma emitting radioisotopes. Environmental sensors and a GPS receiver logs the sampling conditions and the temporal and geographical location of the acquired data. Acquisition and control system based on compact electronics and real time software that operate the sampling line actuators, guarantee the dynamical isokinetic condition, and acquire the detectors and sensor data. The system is also equipped with additional sampling lines to provide information on concentrations of other chemical pollutants.

Operative flights have been carried out in the last years, performances and results are presented.

Introduction

Main motivations for the research program launched some years ago stem from our experience gained during the Chernobyl accident. Detailed information on what we have learned in that occasion has been already reported elsewhere [Castelluccio et al. 2006,] here we summarize only what is needed to understand the inputs to our program.

Flying during the period beginning of May – mid June 1986 with an AGMS, mounted in an Agusta-Bell 412 helicopter and developed in the follow up of the COSMOS 954 event [Gummer et al. 1980] by Italian Civil Defence (VVFF) and Italian National Institute of Health (ISS), three different situations were found.

Till approximately May the 7th, measurements taken at the same place but at different heights could not match the scale behaviour expected if only contamination at ground were present. Moreover as the helicopter moved along its flight path a continuous increase of the counting rate was detected, showing a clear accumulation of radioactivity on the helicopter fuselage. After each flight this contamination was easily removed through a normal procedure of external washing of the vehicle, subsequent controls shown no residual contamination.

From May the 7th till approximately May the 17th, measurements still did not scale in a proper way at different heights but there was not any more the accumulation of radioactivity on the helicopter fuselage. This could be understood as a result of a still persisting air contamination, but only with fine and ultra-fine radioactive aerosol that due to its mobile Brownian-like nature was not fitted to be accumulated by any surface. Instead the more gross type aerosol, being dragged off with an essentially gravity-like mechanism, did not represent at that time a significant source of air contamination.

Starting from May the 19th (25 days since the beginning of the Chernobyl accident and 20 days since the arrival of radioactive plume in Italy), measurements at different heights scaled as expected and only at that time it was possible to have quantitative measurements of (ground) contamination with the required accuracy.

The lack of quantitative measurements and the ensuing uncertainty in forecasting the propagation of radioactive contamination do not help the emergency management in the most critical phase, i.e. when countermeasures have to be taken in a preventive way and some risk of negative effects is inevitably linked to their enforcement.

Computer based management tools for nuclear emergency (developed since the Chernobyl accident) like RODOS [Raskob et al 2005] and ARGOS [Hoe et al. 2005], support integration of AGMSs but the provided information are of relative use during the plume phase of an accident. On the other hand the measurement of γ emitters concentration in air, extension of the plume, in situ environmental and meteorological parameters would be an invaluable help to forecast transport and dispersion of the plume and ground contamination levels.

During the last years research and development of the SNIFFER system, a new aerial platform instrumented for in-plume contamination measurements have been carried on; the SNIFFER aims to characterize the extension, composition and concentration of the radioactive mixture in the plume, as well as to measure in situ meteorological parameters [Cisbani et al. 1996, Frullani et al. 2004, Castelluccio et al. 2005, Castelluccio et al. 2006, Frullani et al 2008]. Here we report the main features of the SNIFFER system and the results obtained.

Material and methods

SNIFFER payload allows the atmospheric and ground radioactive contaminants and air pollution monitoring on large areas in relatively short time.

The system is mounted on board of a fixed wings aircraft (shown in Fig. 1) and consists of the following main components: a) aerial platform; b) isokinetic sampling unit (probe, suction line and filters subsystem); c) radiation measuring equipments (BGO, Geiger, HPGe and NaI detectors and relative electronics); d) VOC - PAH (Volatile Organic Compounds - Polycyclic Aromatic Hydrocarbons) sampling unit – developed for a program on environmental control on traffic pollutants – not discussed in this paper; e) control and data acquisition subsystem (electronic cards, actuators and sensors).

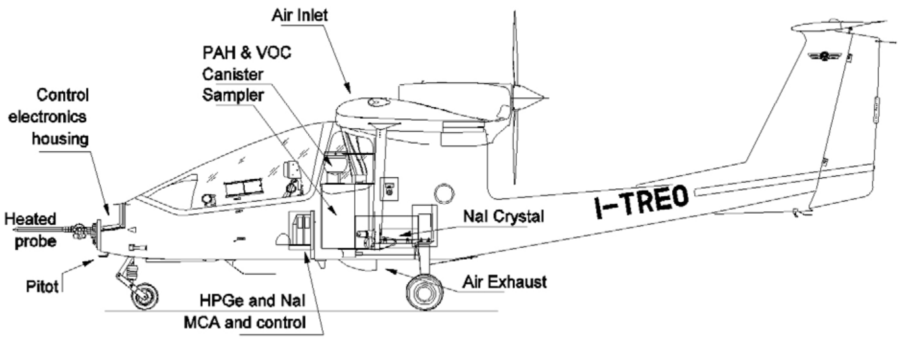

Fig. 1. Installation of the SNIFFER in the Sky Arrow.

The aerial platform complies with the constraints demanded by sampling methodology and operative conditions. The sampling probe is located in a place where aerodynamic perturbation induced by the movement of the platform is negligible. The profile of the front cap of the airplane has been modified to allocate the sampling unit. Safety conditions for the flights are satisfied at an altitude range from some tens of meters to a few kilometres while take off and landing operations are possible in a grass type airstrip of a few hundreds meters (Short Take Off and Landing – STOL type aircraft).

To guarantee the representative and significance of the gathered data, the sampling has to ensure isokinetic conditions, i.e. the inlet walls of the sampler have to be parallel to the gas streamlines and the gas velocity entering the probe has to be identical to the free stream velocity entering the inlet. This is equivalent to the absence of any stream lines deformation in the neighbourhood of the inlet. A failure in the isokinetic sampling may result in a distortion of the size distribution and a misrepresentation of the concentration. To fulfil the isokinetic condition, the sampling line is then provided with a flow regulator (through a valve) operated by an automated control unit that, by means of sensors measuring the relevant environmental parameters, can assure isokinetic sampling. The control software regulates the suction of the air and computes the needed sampled air volume in the current and nominal (STP, Standard Temperature and Pressure) conditions.

The sampling line is essentially a controlled suction line with filters to collect aerosol samples and radiation detectors; its most important subcomponents are (Fig. 2): a) the probe; b) the Shutter (a controlled valve that opens or shuts the line); c) the sampling filters and the filter-case disk; d) the Holder (a small movable box containing a small BGO detector and a Geiger counter); e) the needle valve that permits to maintain the active isokinetic sampling; f) two radiation detectors (BGO and Geiger); g) the Venturi flow meter (not shown in the figure).

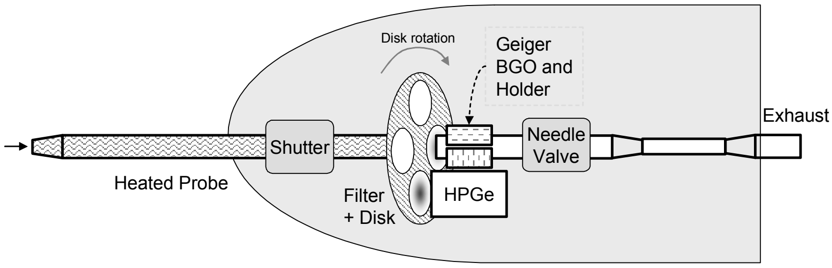

Figure 2. Scheme of the sampling line unit with its components and radiation detectors.

To keep isokinetic sampling condition, the velocity of the entering stream must be adapted to that of the external air (relative speed of the aircraft respect to the air) through a continuous regulation of the inlet flow rate according to the operative and environmental parameters (aircraft speed, pressure and temperature).

The sampling line is therefore operated by the control system that assures isokinetic sampling by means of sensors measuring the relevant environmental parameters. The control software regulates the suction (through the needle valve) according to what the environmental sensors measure.

The radiation measuring system includes four detectors that, under the supervision of the acquisition and control system, allows the quantitative estimation of the radionuclide activities and the determination of the environmental contamination.

A small Geiger detector, having 10 mm external diameter, is mounted inside the Holder box with the entrance mica window in front of the in-line filter. It is powered by the acquisition and control boards and the generated signals are sent to a pulse counter whose contents is periodically read and stored.

A small in-line gamma detector is made of (1 cm^3) BGO crystal, a photodiode and a signal preamplifier. It is located inside the Holder box, next to the Geiger counter, with the sensible window in front of the filter. It provides online information on the presence of radioactive contaminants on the exposed filter (and therefore on the sampled air) but, due to its very small size, it does not have enough energy resolution to permit the identification of the radioactive isotopes.

A high resolution HPGe (High Purity Germanium) detector allows radionuclide identification in the sample collected on the filters. It has been designed in collaboration with Camberra Semiconductor, taking into accounts the constrains imposed by the sampling unit and its aerial platform. The system cooling is assured by a dewar filled up of liquid Nitrogen, providing the proper cooling for up to four hours, compatible with the aircraft autonomy. Due to the crystal size, the detector is not inserted into the sampling line, but is located in such manner that with a 90° rotation of the filter disk it can face the last exposed filter. The dewar is flanged to aircraft fuselage but installed externally to it to minimize its influence on penetration efficiency of the aircraft and to facilitate Nitrogen refilling.

External to the sampling unit, a large volume high sensitivity NaI(Tl) detector ($400 \times 100 \times 100$ mm^3, 17.5 kg in weight), is installed in the rear part of the aircraft, behind the shoulders of the pilot, in correspondence of an opening hole on the bottom of the fuselage.

Environmental contamination in its airborne (particulate) and ground contaminations can be deduced from measurements. Air contamination through the direct measurement of aerosol activity deposited on filter and ground contamination as derived by the large volume high sensitivity detector measurements, taking into account the measured air contamination and a suitable model of the influence of the air contamination and possibly other background contributions on the NaI measurements.

A management unit is dedicated to the control of the devices of the SNIFFER (detectors, sensor and transducers) and to the acquisition of signals coming from them.

This sub-system is made of: 1) two 386-compatible CPU boards (Mesa 4C60 and Mesa 4C28) in PC104 and PC104 Plus standard. The two boards communicate by the respective parallel ports (laplink protocol) in a master-slave scheme. The use of the PC104 offers a reliable operation in a very compact solution, fitting the strong constraints on the available room for the electronics; 2) a PC104 card equipped with a series of Digital to Analog Converters to handle the flow regulation valve; 3) a standard PC104 card (Mesa 4I22) to manage the digital TTL input/output signals of the Shutter, Holder, filter Disk, Canister and Sampler sensors and to drive the regulation of the Geiger high voltage; 4) a custom board consisting mainly of voltage regulators to provide the proper power supplies to several devices; 5) a custom board for signal conversion (sensor specific to TTL and vice versa).

The initialization and final phases are serialized on the two CPUs. Main control is delegated to the 4C60. During the acquisition the 4C60 receives messages from the 4C28 and handle the commands from operator by means of the 3 keys keyboard. Periodically it reads GPS stream data and store them on the on-board flash card. On the other hand, the 4C28 is responsible of the regulation of the flow (via the regulation valve) in order to keep the isokinetic conditions and periodically reads the Geiger's counter. Besides it stops the acquisitions, read data acquired and save them on files at fixed times (defined during the mission plan); eventually it changes the filter (by operator action or at planned intervals).

The interface with the operator (the pilot) consists of a monitor and a small keyboard, these two devices allow to the operator to interact in real time with the acquisition and control system. The operator screen connected to the 4C60 VGA interface is virtually split into two windows: the upper one displays information about the status of the devices; on the lower window scrolling log messages are shown. The verbosity of the visualized messages can be controlled at configuration time. The operator keyboard allows three simple commands: i) start of the acquisition; ii) stop of the acquisition and end of the flight program; iii) change of the filter before the pre-programmed time if an anomalous level of radioactivity has been found.

Results

The instrumented aircraft has got the provisional authorization to flight and after devoted test flights obtained the full certification to perform environmental campaigns, fulfilling the full set of criteria of the Italian Airworthiness Authority. The system obtained also the permission to fly over the Rome urban area in October 2007, in connexion with a campaign devoted also, with different instrumentation, to assess the levels, at height, of some pollutants connected with vehicle traffic.

During these tests, the instrumentation performed quite in agreement with the design specifications. At the end of every flight mission measurement data stored in flash memories were retrieved, analyzed and compared with expectations according to the flight operations for what regards aircraft speed patterns, static pressure differences, flow variations respect to the controlling valve aperture, isokinetic condition automatic regulation, pressure and temperature sensors performances.

The different detectors have been tested and calibrated, using mainly calibrated sealed sources of ^{60}Co and of ^{137}Cs. These sources have been also located very close to the filter position in the suction line to simulate a contaminated filter. Tests were done to check gain stability of the different detectors and connected electronics before and after the flights.

Fig. 3 shows the GPS data during one of the missions. Two heights have been planned on each height, the aircraft has 4 run speeds (60, 70,80, 90 knots aircraft-air relative speed), moving forth and back on a fixed direction.

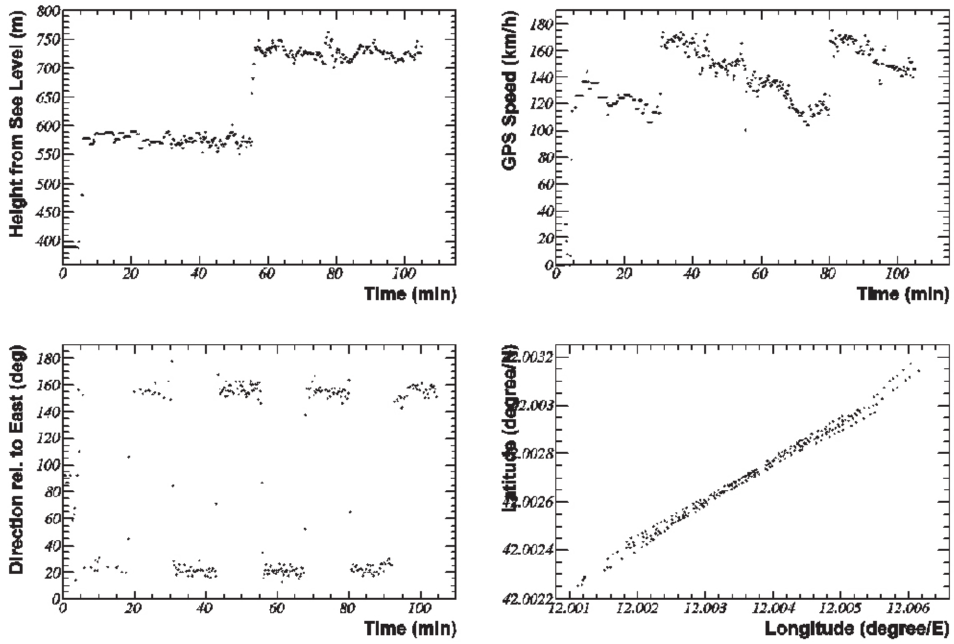

Fig. 3. GPS information from mission 07091415.39, (two heights respect to ground with about 150m variation and four different aircraft speed) that reflects a more complex trend of the air-ground speed registered by GPS module depending on the aircraft flight direction and wind speed.

Fig. 4 shows aerodynamic data from the same mission. Note as static pressure follows aircraft altitude variation during the flight, while dynamic pressure variation reproduces the four different aircraft-air speeds for the two altitude reached during the flight. Comparison with the certified aircraft instrumentation allowed to obtain the proper calibration of the data.

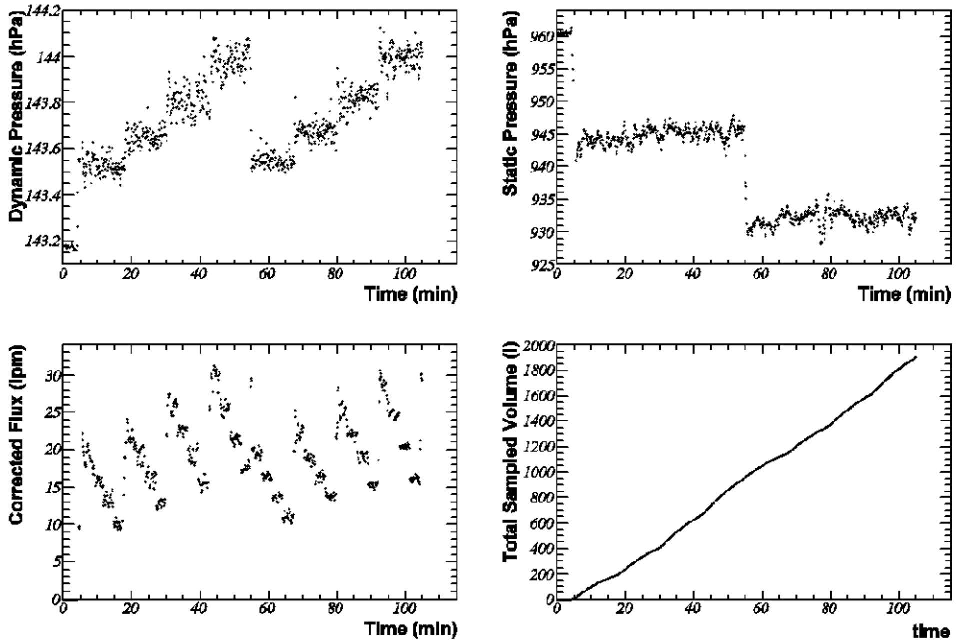

Fig. 4. Static and Dynamic pressures, sampled air flux and total sampled air in mission 07091415.39, see Fig. 3 for a comparison with the correlated GPS data.

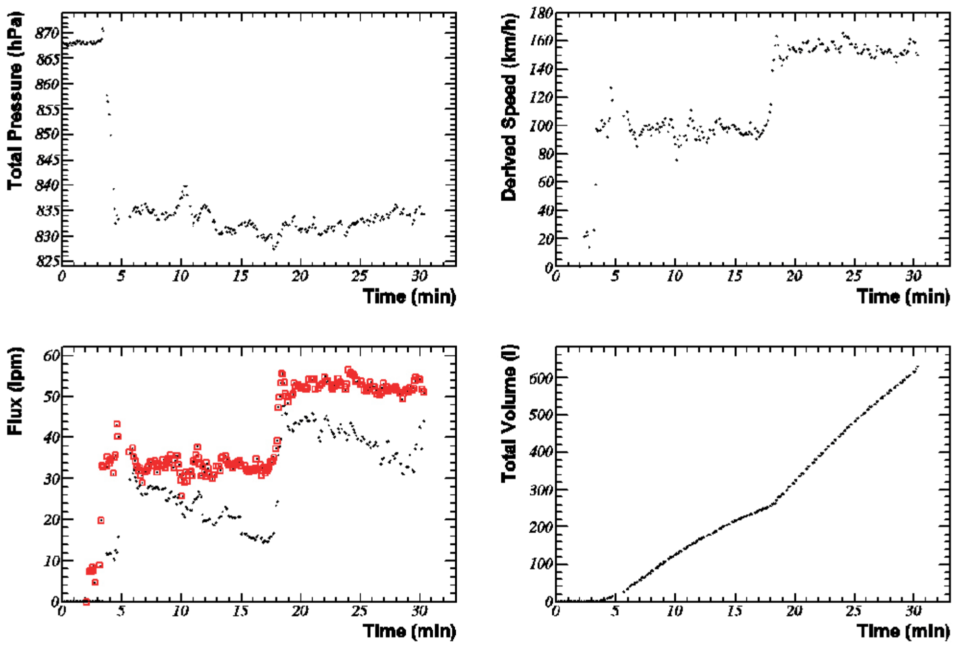

Fig. 5. Calibrated parameters of mission 07100416.52. Data taken at two different speeds at fixed height. Upper-right plot: speed derive from the on-board certified Pitot-Prandtl device; bottom-left plot: the inlet flow isokinetic values (red squares) derived from pressure and temperature sensors are compared to the measured flux (small black cross). The difference shows an uncompensated output impedance.

Data in Fig. 5 have been obtained during a single flight mission at one flight altitude and two different aircraft-air speed. The dynamic pressure data are calculated from calibrated static and total pressure data. The bottom-left plot shows the difference

– due to uncompensated impedance, between the isokinetic optimal flux (used to regulate the needle valve) and the measured one. This difference has been eliminated with a more effective ejector at the end of the line with the result to lower the line impedence

While the filter γ activity characterization is made during the mission, characterizations of the aerosol collected on the filters can be done at the end of the flight mission through detailed offline-analysis.

Different kind of analysis can be carried out, in particular as routine analysis we can mention: 1) gravimetric analysis to evaluate the particle mass collected; 2) SEM (Scanning Electron Microscope) analysis to characterize the morphology and the composition of the particulate; 3) ICP - MS (Inductively Coupled Plasma – Mass Spectroscopy) analysis to characterize metallic-like components in aerosols.

As an example, in Fig 6 a 2000x SEM picture of a filter exposed during a flight along the border of the high traffic ring road encircling Rome is shown. Two different aerosol size distributions (fine and coarse) are clearly visible, as also is shown by the quantitative analysis made and reported in Fig 7.

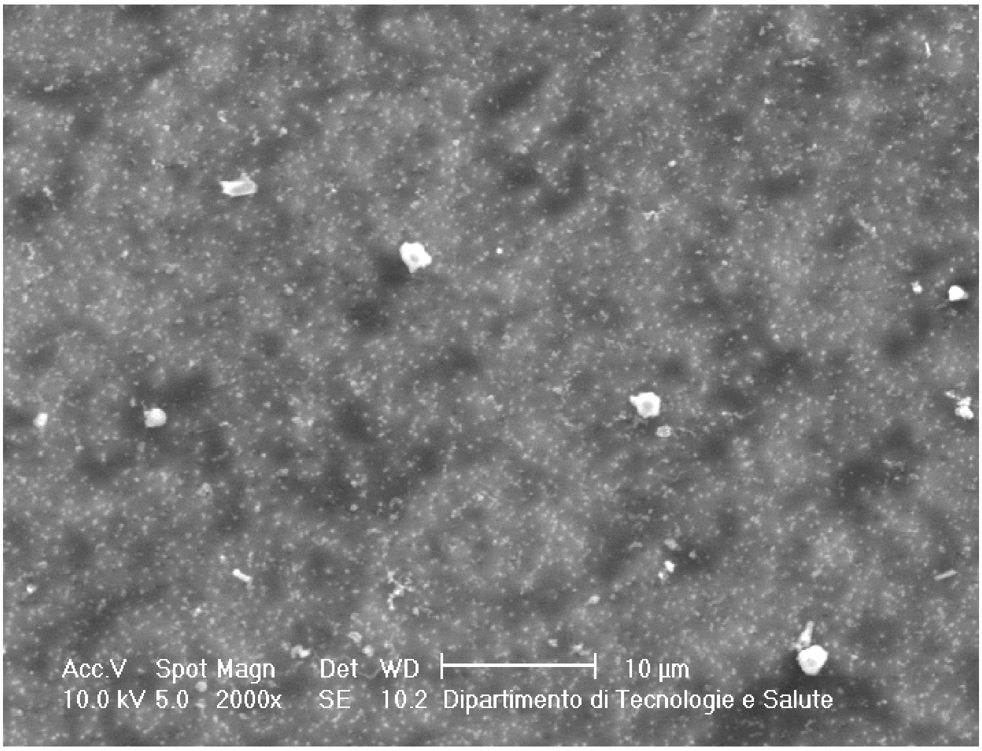

Fig. 6 High enlargement (2000x) SEM picture of an exposed filter during an environmental campaign with SNIFFER. Two different aerosol size distributions can be seen, as demonstrated by the quantitative analysis of Fig 7.

The analysis can be pushed further determining the spectrum of aerosol elemental composition that, in the case discussed, shows a high carbon content in the fine size population while there is a high concentration of silicates in the coarse size population.

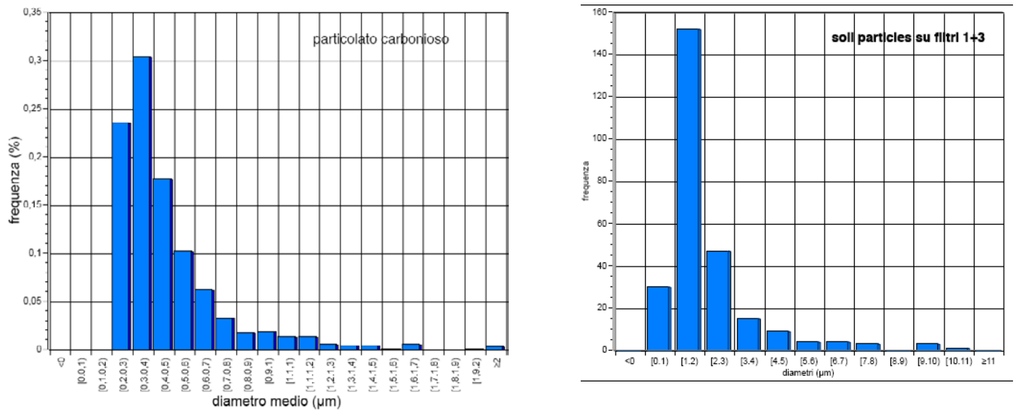

Fig 7. Two populations of aerosol size distribution (fine and coarse) present on an exposed filter, probably attesting different (antropic and natural) mechanisms for their production.

Both such analysis, size distribution and elemental composition, can be a precious element of knowledge in case of an accident to better known the kind of pollutants and the possible pathways for radioactivity transport and migration.

Conclusions and Outlook

A new type of Airborne Gamma Monitoring System has been demonstrated viable, combining a traditional system based on passive (NaI or HPGe) detectors with an active sampling system able to collect air sample and measure their γ emitters composition in real-time.

In this way, the main limitation of the traditional AGMS, that is the inability to provide quantitative measurements on the radioactive contamination during the early phase of an accident, when the plume or its residual fine aerosol content result in contamination dispersed in air, can be overcome. A precious tool for the quantitative radiological assessment can then be available for the emergency management since the early phase of an accident.

The feasibility has been proved for a manned airplane, then useful only in a far field situation. The instrumentation used, for its weight and power requirement can be easily adapted in an UAV (Unmanned Air Vehicle), suitable then to be used even in near field situations. A network of equipped UAVs can give a substantial support in the management even of a severe accident.

The promotion, within an International Cooperative Program, of a project to develop new measurement platforms with the above discussed (and demonstrated) characteristics and performances, represents an important opportunity for research, industry, safety and security.

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SNIFFER: a unique aerial multifunctional platform for large area radiological surveillance and emergency management

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Abstract

The Italian Institute of Health in collaboration with the Ministry of Environment developed and implemented a multi purpose air sampling system based on a fixed wing aircraft, for time-effective, large areas radiological surveillance (to face radiological emergency and to support homeland security) and air pollution monitoring.

The system design has mainly originated from the experience gained during the emergency management of the Chernobyl nuclear power accident (1986).

In order to provide quantitative information on ground as well as atmospheric radioactive contamination (and permit, e.g., reliable plume behavior forecast), the SNIFFER consists of a fixed wings aircraft (Sky Arrow 650 by 3I) properly adapted to house a relatively compact payload equipped with a sampling line that works in isokinetic regime. The line serves the β and γ atmospheric radiation detection system made of: one small BGO scintillator and one Geiger counter providing a near real time information on the air sampled radiation excess and one high energy resolution hyper-pure germanium for radionuclides quantitative identification.

A large NaI scintillator detects the ground and atmospheric radiations; The combination of the information from the above 4 detectors permits the quantitative estimation of both atmospheric and ground contamination.

Two sampling devices devoted to polycyclic aromatic hydrocarbons and volatile organic compounds air quality monitoring are also integrated into the SNIFFER.

A fully automated acquisition and control system actively manage all the payload components.

The whole acquired data are integrated by environmental (temperature, pressure and air density) and geolocation information for a complete characterization of the sampling conditions and their temporal and geographical location.

The system and the results of the first aerial mission will be presented at the conference.

BALTRAD – An advanced weather radar network for the Baltic Sea Region meteorological institutes and authorities

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Abstract
Accurate and timely weather forecasts provide essential information for the authorities by warning them of hazardous conditions (e.g. snow storms or floods). Weather radar systems are one of the tools producing such information because they are capable of monitoring rain, snow, hail, and wind over large geographical areas with high resolution in both time and space. Radiation protection authorities are amongst the potential end-user organizations because radar data can be utilized in making forecasts of the precipitation scavenging of radioactive particles to the ground. The objective of the BALTRAD project is to create a state-of-the-art real-time weather radar network for the Baltic Sea Region that facilitates the harmonised production, exchange and use of radar data. The project (2009–2012) is partially funded by the European Union's Baltic Sea Region Programme 2007–2013. An integral part of the project includes so-called pilots that will demonstrate the value of BALTRAD for end users in the fields of e.g. hydrology, aviation and radiation safety.

Introduction
The objective of the BALTRAD (An advanced weather radar network for the Baltic Sea Region: BALTRAD) project is to create a state-of-the-art real-time weather radar network for the Baltic Sea Region, the so-called BALTRAD system. When completed, the system will facilitate the harmonised production, exchange and use of real-time weather radar data, and contribute to the production of accurate and timely weather forecasts that provide essential information for the authorities by warning them of

hazardous conditions (e.g. snow storms or floods). Generally, BALTRAD delivers value-added precipitation information to improve short-term weather forecasts.

The project, headed by SMHI, was officially started in February 2009 and continues until the end of January 2012. The source of funding is the EU's Baltic Sea Region Programme 2007–2013. The Partnership comprises eight national organizations from seven countries (Sweden, Finland, Poland, Latvia, Denmark, Belarus, Estonia). In addition, there are associated organizations in Denmark, Latvia and Poland which will be contributing in various ways. The project contains a few pilot applications that will demonstrate the value of BALTRAD for end users in the fields of e.g. flood protection, aviation and crisis management (including nuclear accidents).

The technology developed by and for BALTRAD will be proposed as the standard for exchanging weather radar data in the World Meteorological Organization Information System (WIS).

In the following we first describe briefly the project and project's organizational structure. After that we discuss possible uses of the final system from an end-user point of view, the emphasis being on applications related to radiation monitoring and management of nuclear or major radiological emergencies.

BALTRAD project

BALTRAD represents the first dedicated international weather radar networking project funded by the EU. However, the concept being followed has already been proven once before in the NORDRAD network (involving the radars from Norway, Finland and Sweden) some 20 years ago. The original NORDRAD technology is becoming obsolete and it is time to go further. Within BALTRAD it is hoped to create:

– real-time exchange of radar data among all members;
– a common production framework containing harmonized algorithms;
– consistent end-to-end management of quality information;
– an Open Source software system.

The goal is to achieve cutting-edge distributed radar networking (see Fig. 1 for the radars planned to be in the network), i.e. a network where data is exchanged on equal terms and where each institute processes the data according to its local needs.

The work is organized into seven work packages (leader organization given in parentheses):

1. Management and administration (SMHI).
2. Communications (FMI). This work package is dedicated to both internal and external communications.
3. Core network (IMGW). This work includes creating the real-time networking functionality. It is foreseen that it will be based on secure HTTP-based mechanisms with subscription services. The long-term goal is to contribute such mechanisms to WIS for Europe.
4. Data catalogue (EMHI). This work involves managing all data to be exchanged and processed and focuses on creating a database for all metadata and systematic methods for storing data files.
5. Production framework (FMI). This work package deals with developing, collecting, and harmonizing numerous algorithms for improving the quality of raw data and creating products from them.

6. Deployment (SMHI). This involves integrating the outputs of work packages 3-5 into a real-time system suitable for operational deployment. Regular releases will be installed at all Partners. A case log will be prepared containing cases relevant to air traffic, flash flooding, urban hydrology and nuclear or radiological accident scenarios.
7. Pilots (IMGW). This work serves to make the BALTRAD system and its radar-based information available to test pilots and to collect the pilots' feedback to integrate new and improved software and products. In this way the value achieved in a true end-to-end radar network will be demonstrated.

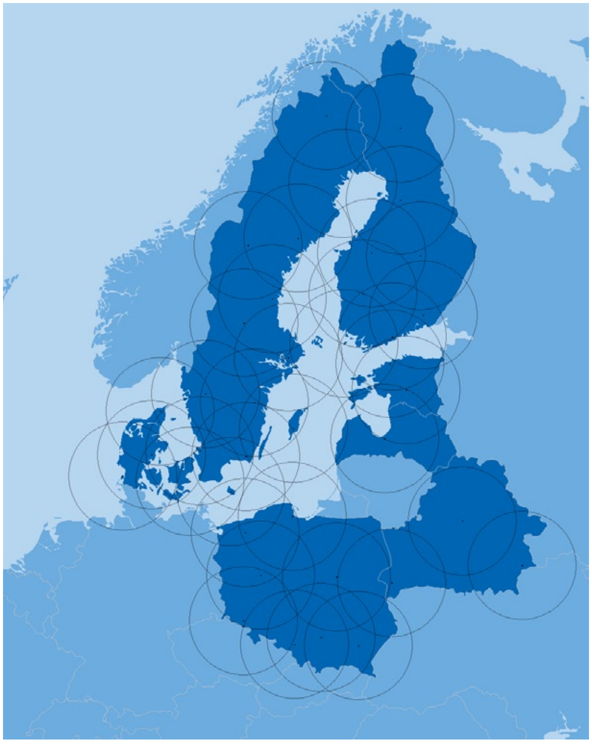

Fig. 1. BALTRAD radars. Radar radius is typically 200 kilometers.

Importance of weather radar data for end-users

Weather radar systems are capable of producing information on rain, snow, hail, and wind over large geographical areas with high resolution in both time and space. This information is of great value for the authorities especially in the context of various disasters, such as accidents involving releases of chemical or radioactive substances to the atmosphere, or in case of severe weather conditions (heavy rain, tornadoes, very stormy wind, air pollution situations…) that may affect the normal functions of the society and people’s everyday life. However, accurate and timely weather data are needed also during normal situation. In aviation, for example, weather radar is the most important tool for monitoring weather for flight planning and air traffic control. The most hazardous phenomena in this context are thunderstorms, heavy rain and hails, and strong wind shear. Generally, precipitation intensity and accumulation, precipitation type, wind velocity and direction are amongst the data that various authorities need to have access to both in emergencies and in routine conditions.

As already mentioned, the BALTRAD project includes a work package (WP7) which is dedicated to end-user views and pilot applications. Radiation protection authorities are amongst the potential end-user organizations because rain, for example, increases deposition and consequently dose rates at the ground level (see Fig. 2). Rain information can be used in planning and directing emergency measurement activities and in interpreting exceptionally high radiation levels during routine conditions, and numerical data (precipitation rates, accumulated precipitation) can be input to consequence assessment models. Heavy rain or otherwise severe meteorological conditions (e.g. very strong wind) may also prevent performing certain types of radiation measurements, such as measurements using airborne platforms (airplanes, helicopters, unmanned aerial vehicles).

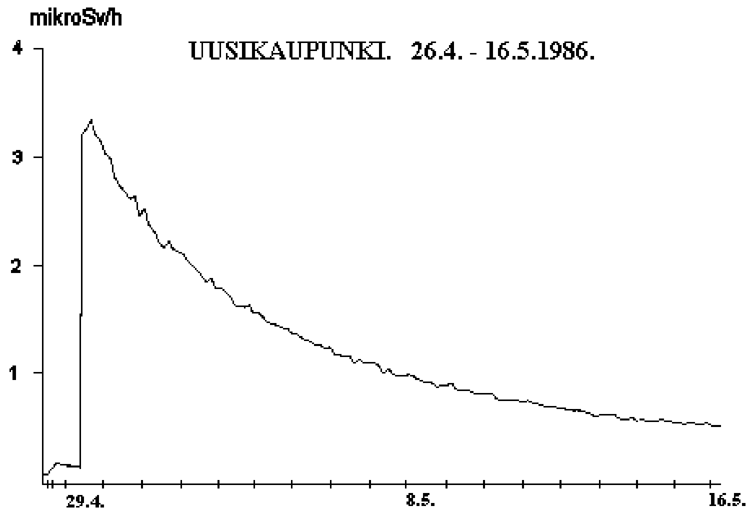

Fig. 2. Effect of rain occurrence on measured dose rates at ground-level during the Chernobyl accident. Uusikaupunki is located on the western coast of Finland.

End-user applications of the Finnish radiation protection authority

Currently STUK utilizes radar data from FMI in two basic ways:

1. The radiation information system (USVA) controlling the national automatic dose-rate monitoring network (ULJAS) receives observation data (given as echo strengths) every ten minutes. These data can be used in routine conditions to help to interpret the increases of dose rate levels caused by natural radioactive substances brought down close to the ground by precipitation.
2. STUK and FMI have developed a system known as KETALE (Fig.3; Amman et al. 2010) for the overall management of meteorological data and dispersion and dose calculation results in radiation emergencies. Radar images – such as the one shown in Fig. 4 – can be downloaded from FMI's systems and included in the situational analysis reports prepared with KETALE but flexible comparison with calculation results (mainly deposited activities or ground exposures) is currently not easy.

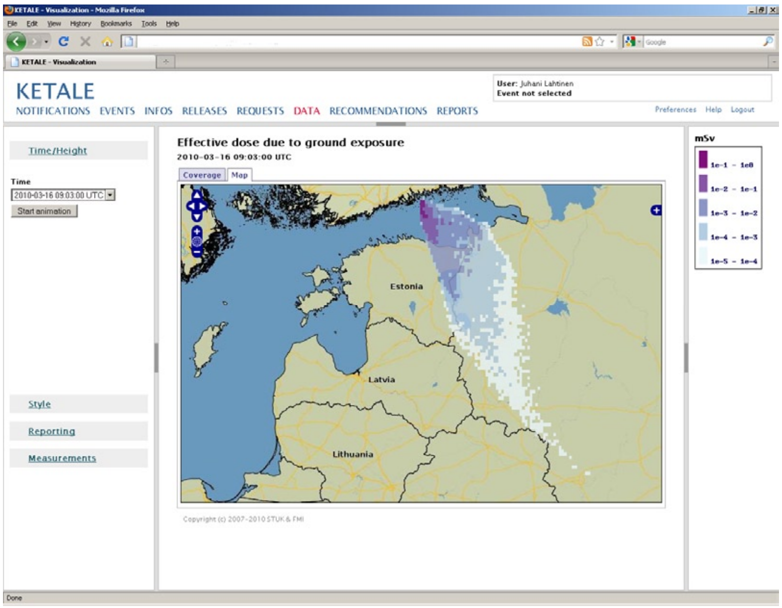

Fig. 3. Example of KETALE output on screen (hypothetical scenario). Radar images available in a proper format could be compared with calculational results using the same map. This would help to assess and interpret the overall situation and to plan further measurement activities.

Fig. 4. Example of radar images accessible to STUK at FMI's restricted web site.

After the completion of the BALTRAD project, STUK would appreciate having access to a flexible, reliable and user-friendly system that could provide both high-quality radar images and numeric data. STUK and many other authorities are interested primarily in precipitation-related ground data (precipitation rate, accumulated precipitation, precipitation type…), although certain other available quantities may be of use in some situations. All data should be given in user-friendly quantities – not as echo intensities in desibels but as e.g. rain intensities in millimeters per hour – and include some index of uncertainty or inaccuracy.

On one hand, numeric data, such as average rain intensities in spatial grid cells, can be utilized in emergencies to correct or estimate so-called precipitation scavenging coefficients (washout coefficients) used in model calculations (see e.g. Ilvonen 2002). On the other hand, graphical images generated from numeric data by e.g. a Web Map Service can be used in KETALE to visually compare the observed spatial rain distributions with the calculated spatial radionuclide-concentration and dose distributions. This kind of comparison helps to assess the overall situation and to plan further measurement activities. As mentioned above, in the USVA system radar data can be used in interpreting observed measurement results.

Conclusions

Weather radars provide meteorological institutes and many other authorities with essential data that can be used both in normal situation and during hazardous conditions, including various accidents and other events involving releases of dangerous substances to the atmosphere. The BALTRAD project aims at creating a state-of-the-art real-time weather radar network for the Baltic Sea Region.

Various end users play an important role in the project by developing pilot applications. Radiation protection authorities are amongst the potential end-user organizations because precipitation, for example, increases deposition of airborne radioactive particles. As an official partner in the project, STUK is responsible for creating applications related to emergency activities and radiation monitoring. It is foreseen that these applications will have close interfaces with the Finnish ULJAS/USVA and KETALE systems, the first being the nation-wide dose-rate monitoring system and the latter being the national platform designed to manage the assessment of environmental consequences of major radiation emergencies.

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The 'NOODPLAN' early phase nuclear emergency models: an evaluation

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Abstract

Over the past decades SCK•CEN has developed a set of models –called the NOODPLAN models– for consequence analysis of atmospheric releases during the early phase of a nuclear emergency. All these models use segmented bi-Gaussian plume dispersion algorithms based on the Bultynck-Malet scheme and are limited to a local scale (50 km from the release point).

The different NOODPLAN models are largely customised to the needs of several external users. Customisation includes e.g. source term determination from stack monitoring, connection to on-site meteorological masts, graphic display of countermeasure areas corresponding to intervention guidelines, special modules with standard accident scenarios and specific calculations at the location of detector points (e.g. corresponding to monitoring network). These models have been primarily designed for the Belgian nuclear power plants, the results being used by the radiological evaluation cell of the Federal Crisis Centre, within the framework of the Belgian nuclear and radiological emergency plan. Additional users include research reactors and other nuclear facilities.

The NOODPLAN models perform real time calculations based on variable on-site measured meteorological data, as well as prognostic evaluations. Following a brief description of the different functionalities of the models, this paper will focus on the evaluation of the different NOODPLAN models. This evaluation will draw on our experiences with the NOODPLAN models from regular emergency exercises and will include an inter-model comparison study with state-of-the-art atmospheric dispersion models integrated into the European decision support system RODOS. The results of this evaluation highlight possible future development required to ensure a fast and reliable decision making during the early phase of a nuclear emergency.

Introduction

In a nuclear or radiological emergency the release of radionuclides in the atmosphere can lead to the external and internal exposure of the surrounding population and the contamination of land and water including food products. To estimate the radiological consequences and to initiate the essential countermeasures during an incidental release historically many organisations have developed their own emergency models or systems. Such models combine an atmospheric dispersion part for the calculation of air concentrations and deposition values with a dose model to evaluate the dose from different pathways. Depending on the complexity of the model, the dose from contributions like cloud shine, inhalation, ground shine, ingestion, re-suspension … is calculated.

The NOODPLAN models are a set of models developed for the radiological evaluation of an incidental or accidental atmospheric releases from specific nuclear sites in Belgium. Off-site radiological evaluations up to a few tens of kilometres from the release point by the nuclear operators are a legal requirement in Belgium. The results of the radiological evaluations are sent to the Federal Crisis Centre for further evaluations and decisions on e.g. countermeasures and measurement strategy. Although the NOODPLAN models are historically well established, a regular evaluation and confrontation with the latest developments in the field of nuclear emergency modelling is required. Over the past decade advanced decision support systems with state-of-the-art atmospheric dispersion and dose models have been developed and are increasingly being installed for operational use in national emergency centres. Examples of the main decision support systems for nuclear emergency management used in Europe are ARGOS (Information Management and Decision-Support System for Radiological-Nuclear Hazard Preparedness and Response) and RODOS (Real-time On-line DecisiOn Support system for off-site nuclear emergency management). The evaluation and confrontation of historical models developed for specific sites like the NOODPLAN models with new and generally applicable models can highlight interesting aspects such as the use of specific required model input, model differences and uncertainties (under specific conditions) and possible requirements on future developments for fast and reliable radiological evaluations.

Description of the models

The NOODPLAN models are based on the historical work of Bultynck-Malet [Bultynck H, Malet L, 1969]. During an extensive measurement campaign in the period 1966-1968 using the meteorological mast at the Belgian Nuclear Research Centre (SCK•CEN) an atmospheric turbulence typing scheme with seven stability categories E1-E7 was developed. This so-called Bultynck-Malet categorization is essentially based on the ratio between the vertical gradient of the potential temperature (8m -114m) and the square of the measured mean wind speed at an intermediate level (69m). During the same measurement campaign the appropriate dispersion parameters for each of the stability classes E1-E7 were experimentally determined from wind vector fluctuations in three dimensions measured by means of an anemometer at a height of 69 m. Based on these measurements the dispersion parameters in respectively the horizontal and vertical direction are analytically expressed in function of the distance (x) as $\sigma_y=Ax^a$ and $\sigma_z=Bx^b$ with A, a, B and b depending on the stability category E1 to E7. E1 corresponds to very stable, E3 to neutral and E6 to very unstable atmospheric conditions. E7 is a special neutral category for high wind speed (≥ 11.5 m/s at a height of 69 m). Bultynck-Malet completed these atmospheric dispersion investigations with tracer experiments. In total 15 experiments were executed in which 5 to 6 kg of fluorescent material/experiment was released by an aerosol generator (aerosol dimensions < 4 μm) from the meteorological mast at a height of 69 m and during a release period of ~50 minutes. Air samples were taken at 3 to 7 different locations/experiment at distances ranging from 100 m to 15 km. These experiments showed no significantly nor systematically difference from the calculated values. In addition Bultynck-Malet compared their atmospheric dispersion results with extensive field experiments executed by Brookhaven National Laboratory in better experimental conditions and for a release height near ground level and at 110 m. Due to the better experimental conditions they found even better agreement with these experimental data. More recently the model was confronted with additional limited tracer experiments for ground releases [Govaerts et al., 1988] and stack releases [Rojas Palma et al., 2004]. In both studies the Bultynck-Malet based model performed as good as Lagrangian models.

It can be summarized that the Bultynck-Malet scheme was experimentally well established for hourly values and varying release heights in regions with a rather high surface roughness of 1 to 3 m, applicable in large parts of Belgium and Europe and from distances of a few hundred meters to a few tens of kilometers from the release point.

Based on the Bultynck-Malet formalism different bi-Gaussian plume and even tri-Gaussian puff [Govaerts et al., 1988] models were developed. The formalism was used as well for the development of short duration discharge models for the evaluation of the consequences of accidental releases in nuclear facilities as well as for long duration discharge models for the assessment of routine discharges. The latter evolved to the Immersion Frequency Distribution Model (IFDM) under current development by VITO (The Flemish Institute for Technological Research) which is widely used to calculate the impact of routine releases from industrial facilities [Cosemans et al., 2000]. The Bultynck-Malet (sometimes also called Mol) parameterization is also part of the different parameterization schemes used in the RODOS system.

To overcome the stationary and homogenous situation inherent to Gaussian plume models the NOODPLAN models discussed here make use of the segmented bi-Gaussian plume approach. This makes it possible to treat time-varying releases and transport conditions and especially changes in wind directions. In this approach the plume is broken up into plume segments corresponding to fixed time periods of 10-minutes. To take into account the difference between the original Bultynck-Malet sampling time (30-60 minutes) and the 10-minute periods used in NOODPLAN a correction factor to the dispersion parameters is applied.

The difference between the different NOODPLAN models is the specific customization for the origin of the release. Because the models were developed historically at different moments in time also some –in general minor- differences exist in the modeling aspects. Currently three versions are existing corresponding to a version specific for the Nuclear Power Plants (NPP's) at Doel, at Tihange and the nuclear facilities in the surrounding of Mol (SCK•CEN, Belgoprocess …). All models have the same structure and comprise an input, calculation and reporting module. The input module is highly customized and allows the selection of a specific release pathway (different stacks, steam generator incidents, releases with fire, ground release …) and determines the relative isotopic composition of the different release groups (noble gases, iodine, βγ-aerosols, α-aerosols, tritium) based on the time of release after reactor shutdown. For some of the models automated import of source term data (directly converted from e.g. stack monitoring as well as ventilation rates) and meteorological data is possible. The calculation module allows for the Doel and Tihange systems to add 3 user defined periods with prognostic data. The output module allows printing and visualizing numerical and graphical results. Direct selection of a color scale for the graphical results corresponding to the intervention limits in Belgium is possible. The Doel and Tihange models have in addition a standard scenario module. This is a simple (non-segmented) bi-Gaussian plume model coupled to a database of over 100 standard accident scenarios/reactor. When, based on the technical information available, the best fitting accident scenario is determined this standard scenario module allows doing a first full assessment in less than 1 minute with on-site meteorological data. The models are used by the operators and the results (scenario files) are sent to the Federal Crisis Centre in case of an emergency (or during exercises).

A polar spatial grid is used with a calculation point every 5° on concentric circles with distances from the release point determined by the arithmetic progression with a ratio of $2^{1/8}$ (for Doel and Tihange) and $2^{1/4}$ (for Mol) starting from 200 m up to ~50 km. This gives a very high calculation resolution near the release point while the total number of calculation points is limited resulting in a very short calculation time. Several hundreds additional calculation points can be defined and stored in a database by the users (e.g. detector points corresponding to the environmental monitoring network and points on pre-defined measurement routes) without increasing the calculation time noticeable.

Results

The models are regularly –in general at least two times a year - used during emergency exercises resulting in an important experience in using the models and interpreting the results. Based on the SCK•CEN experience (as utility and operator of the model at the evaluation cell of the Federal Crisis Centre) the following shortcomings can be identified:

- The restriction of the calculation area to around 50 km on several occasions caused problems and confusion regarding intervention limits especially in the case of an important release of I-131. Easy export of all input data available in the NOODPLAN models to a model with an extended range would be interesting.
- The transfer of calculation results from the operator to the Federal Crisis Centre is slow because old technology is used. For this reason the evolution of an emergency –which can easily be calculated with the NOODPLAN models by adding 10-minute periods without the requirement for performing the original calculations again- cannot always be followed continuously. This problem will be solved by using fast data connections in the future.

- Although the definition of additional (detector) points is possible, a direct and fast relation with measurement predictions or results (e.g. combination of dose rate from cloud and ground shine) is missing.
- Although the interface is rather simple, some specific actions of the user are necessary (e.g. specific action for selection and validation of input). This implies the necessity of a regular training (a limited yearly training session seems sufficient) for adequate knowledge of the software.

Further, an evaluation was made by executing an inter-model comparison with state-of-the-art models. Inter-model comparisons are an excellent tool for the assessment of model differences and the uncertainties related to these differences as shown in [Kok et al., 2005]. In this inter-model comparison the Noodplan results are compared with RODOS (ATSTEP) and RIMPUFF results (RIMPUFF within RODOS or a RIMPUFF stand alone version). ATSTEP is a tri-Gaussian puff model with time integrated elongated puffs and employs modified Karlsruhe-Jülich/Mol dispersion parameters applicable for different types of land-use up to a distance of about 100 km from the release point [Pässler-Sauer, 2007]. RIMPUFF is a Lagrangian mesoscale atmospheric tri-Gaussian dispersion puff model developed by Risø (RIsø Mesoscale PUFF model). For RIMPUFF within RODOS the Carruthers sigma-parameters for near range (similarity scaling), and a sigma parameterisation for medium range calculations are used. For the standalone version of RIMPUFF different parameterization schemes can be used but for the calculations performed here similarity-scaling of atmospheric turbulence and diffusion is used in which the calculation of plume spread is based on the basic physical parameters that governs the atmospheric boundary layer turbulence [Thykier-Nielsen et al., 2004].

Both basic scenarios and full realistic scenarios are included in this study. The basic scenarios assume a constant release and constant meteorological conditions for the duration of the plume passage. Nine basic scenarios are defined, starting from a release condition very close to the tracer experiment conditions of Bultynck-Malet, investigating the effect of release height and atmospheric stability under dry conditions. An additional (tenth) basic scenario is defined to evaluate the effect of rain. The scenarios are defined in Table 1. For the RODOS calculations a Cartesian variable grid is selected up to 40 km from the release point with near the release point a grid resolution of 500 m.

Table 1. Definition of the basic scenarios for the inter-model comparison. As well the Pasquill as Bultynck-Malet stability class is specified.

Basic Scenario	Variable parameters			Common parameters
	H_{eff} (m)	Stability	Rain (mm/h)	
1	72	D/E3	0	Release duration: 60 minutes
2-3	1/160	D/E3	0	Released nuclide: I-131 (100% elemental)
4-6	1/74/160	C/E5	0	Total released activity: 6.10^{14} Bq
7-9	1/74/160	F/E1	0	Wind speed= 5 m/s at 72 m
10	72	D/E3	3	No plume rise, no building effects

The complex or realistic scenarios comprise a calculation of a stack release for the Doel site and a release with fire from the Tihange site. Both scenarios consist of a release of 35 periods of 10 minutes, with variable release rates (noble gases, iodine and aerosols) and real meteorological data as typically available on-site (measurements from mast) including periods with rain as well as important changes in all other meteorological parameters. For this study in addition to the NOODPLAN calculations RODOS (ATSTEP) calculations are performed on a variable Cartesian grid with an extension of 80 km from the release point and for the RIMPUFF stand alone calculations with a constant Cartesian grid of 364x364 points with an extension of ~50 km from the release. The models are used at their full complexity for both the basic and complex scenarios. This means that standard land use and topography data are taken into account for the RODOS and/or RIMPUFF calculations. Also the NOODPLAN models are used in the way they are operationally used (i.e. topography used for the Tihange site, not for the Doel site, differences due to different land use is not taking into account in the NOODPLAN models).

The basic results (Time integrated concentration near ground level) for scenario 1 are displayed in Fig. 1. For this scenario the Noodplan results on the plume axis as a function of distance from the release point are compared with RODOS results (ATSTEP as well as RIMPUFF). Clearly visible is the effect of the polar grid with high resolution near the release point of the 'Noodplan' models resulting in a higher value at the position of the maximum. For the other basic scenarios a comparison of 'Noodplan' with RODOS (ATSTEP) is only made at specific distances corresponding with typical distances used for the implementation of countermeasures like sheltering, iodine prophylaxis or evacuation (at position of maximum, 5, 10 and 20 km from the release point). Results for the time integrated concentration as well as the total effective dose are shown in Fig. 2. The total effective dose includes only cloud shine, inhalation and ground shine integrated for 1 week. For the Noodplan models the ground shine for 1 week is calculated based on the value for 24h and 2 weeks (1 week integration time is not a standard output). In general the agreement between the models is very good (in general within a factor 2). For the value at the position of the maximum, the Noodplan results are nearly always higher compared to the RODOS values. This can again be attributed to a difference in spatial resolution near the release point of the two models. The variation of the ratio of the results seems to be somewhat higher for the location at 20 km, compared to 5 or 10 km. The total effective dose is mainly determined by the inhalation dose. The NOODPLAN results are in general higher. If individual exposure pathways are studied the difference is mainly due to a higher contribution from ground shine by the NOODPLAN models.

Results on dry and wet deposition based on scenario 1 and 10 are summarized in Table 2. The difference between the models for deposition (as well dry as wet deposition) is larger then the differences for the air concentration and total effective dose. Differences of a factor 3 to 4 are found here. The Noodplan models show higher deposition values which is explained by a difference in deposition velocities used by the models.

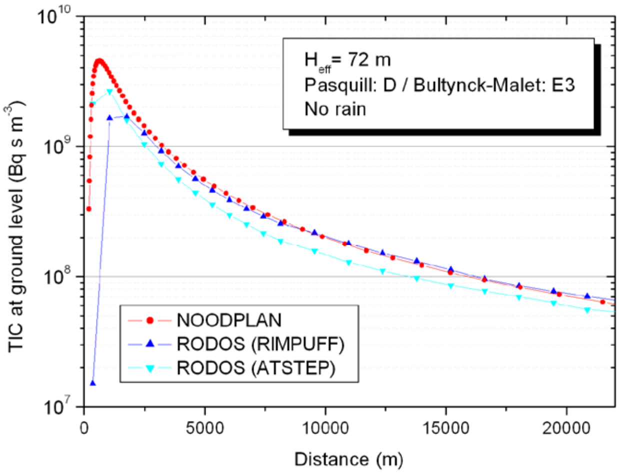

Fig. 1. The Time Integrated Concentration (TIC) near ground level for scenario 1 as a function of distance under the plume axis for the NOODPLAN model compared with the atmospheric dispersion models ATSTEP and RIMPUFF within RODOS.

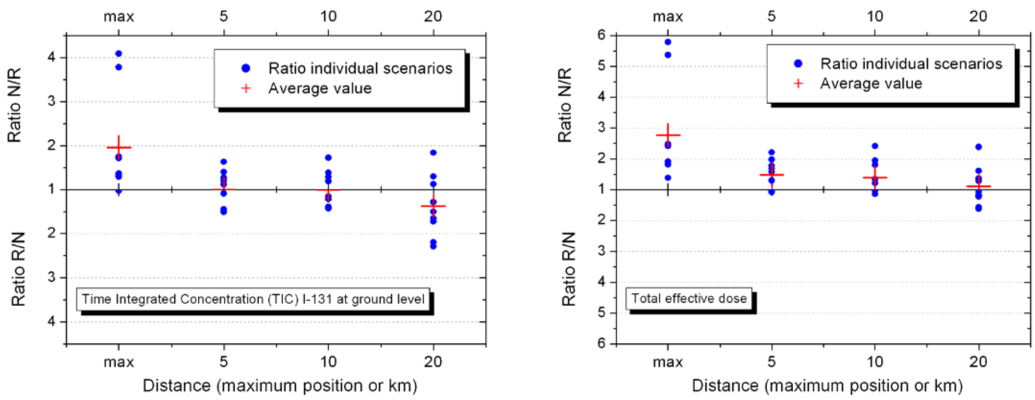

Fig. 2. Results of the inter-model comparison between NOODPLAN (N) and RODOS/ATSTEP (R) for the basic scenarios 1 to 9. On the left site the results of the Time Integrated Concentration are shown, on the right side the results for the total effective dose.

Table 2. Inter-model comparison between NOODPLAN (N) and RODOS/ATSTEP (R) based on scenario 1 and 10 for dry and/or wet deposition of elemental I-131. The dry deposition velocities are derived from the ratio of the deposition to the Time Integrated Concentration at every location. For NOODPLAN this deposition velocity is a constant, for RODOS the value can be dependent on the land use. The difference between the NOODPLAN and RODOS results for the ground contamination is mainly determined by a difference in the deposition velocity (a factor 3).

Distance (max or km)	Dry deposition			Dry+wet deposition
	Ratio N/R	$v_{d,Noodplan}$ (m/s)	$v_{d, Rodos}$ (m/s)	Ratio N/R
max	5.07	1.00E-02	3.39E-03	4.99
5	3.61	1.00E-02	3.51E-03	3.00
10	3.94	1.00E-02	3.52E-03	3.28
20	3.27	1.00E-02	3.97E-03	3.22

For the complex scenarios only a qualitative comparison could be made due to differences in the calculation grids used by the models. For the NOODPLAN results a transformation could be made from the standard polar grid to the Cartesian grid selected for the RIMPUFF calculations by a nearest neighbour algorithm. The results for NOODPLAN and RIMPUFF for the most complex scenario with fire and in a complex region (Tihange site in valley of river Meuse) are shown as an example in Fig. 3. The RODOS (ATSTEP) results look very similar. The results for the Doel scenario (stack release, flat terrain) match even somewhat better.

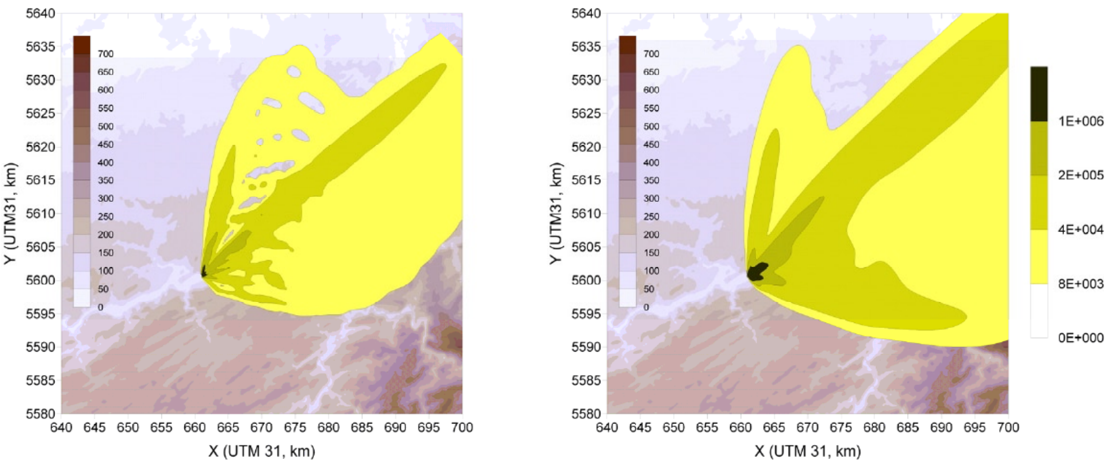

Fig. 3. Time Integrated Concentration (TIC) near ground level of Cs-137 for a release scenario for the Tihange site with fire (35 periods of 10 minutes) for NOODPLAN (left) and RIMPUFF (right) on top of a map showing the topography (height above sea level in meters). The same (color) scale scale for the TIC is used expressed in Bq·s/m^3 in both graphs.

Discussion

Although a good agreement of results in an inter-model comparison is not a guarantee for accurate results due to e.g. common simplifications and assumptions, the execution of inter-model comparisons can identify differences and give confidence in the typical uncertainties present in atmospheric dispersion modelling. Based on this study several interesting observations can be made which highlight possible future developments.

During emergency exercises often the maximal dose or contamination values calculated by the models are used as a starting point for a discussion on the advice of countermeasures. This study clearly illustrates that the maximal value can be strongly connected to grid resolution around this maximum position. A polar grid as used by the Noodplan models has a very high resolution near the release point without having a large number of total grid points. In many scenarios, in which the maximum is within a few kilometres from the release point, this can be seen as an important advantage. However, for specific scenarios with the maximum at several kilometres from the release point the resolution near this maximum will also be limited for the Noodplan models. Most models have fixed grids (or has to be fixed by the user as part of the input data) leading to results in a calculation area and with a resolution which is not always optimal for the interpretation of the scenario under study. The development of routines with flexible and (automated) guidance on grids based on available input and required output seems interesting in this context.

Although the differences in the inter-model comparison in this study for the basic model output (the time integrated concentrations) are small, differences increase clearly for derived parameters like e.g. the ground contamination by dry and/or wet deposition. This increase in difference is only due to the use of different deposition velocities. This shows clearly that good knowledge of basic parameters (such as the deposition velocity) is crucial. Apart from effort in better and more accurate atmospheric dispersion modelling (especially for specific situations like e.g. influence of buildings, urban areas …) a better quantification of basic parameters like deposition velocities seems important. In this respect the confrontation of model results for routine releases with measurements from these discharges in specific regions can be helpful.

Executing inter-model comparisons confronts the user of the models with differences in the required input. The NOODPLAN models are highly customized in a way that all input required by the models is directly related to physical (measured) parameters operationally available on-site. For performing calculations with other models conversions of some parameters have to be done to be able to run these models such as e.g. the parameters for the calculation of the plume rise (specifically for a release with fire). Because the NOODPLAN models are only applicable for the early phase and local scale of a nuclear incident it is interesting to have all input data available for other models. In the new versions of the NOODPLAN models an XML-file with all input data is generated. Concerning the output, a specific module is foreseen to transfer the output files to CSV files for export to a Geographical Information System (GIS).

Although, it is possible in the NOODPLAN models to define a large number of specific calculation points (e.g. detector points) and to display all calculated output for a specific calculation point, fast prediction of or comparison with measurement results is difficult. The development of (scenario preparation) tools for e.g. direct calculation of measurement results in a large number of points (e.g. dose rates, surface contamination, air concentrations) for a number of operationally used detectors would be very interesting.

Simple but highly customized models used by different operators like the NOODPLAN models are under constant evaluation by as well the users (operators), inspectors of the nuclear facilities, and developers. The models are often also used as part of the licensing procedures for the nuclear activities as well as for investigations of routine releases and their impact. In this respect such models are fully complementary to the decision support systems introduced in many European countries the last decade. However, results should be in line and/or differences should be well understood.

Conclusions

This review and evaluation of the NOODPLAN models used as a fast tool to perform radiological impact assessments for the early phase of an accidental atmospheric release from specific nuclear sites in Belgium shows that the results of these models are in line with state-of-the-art nuclear emergency decision support systems installed nowadays by different countries in their emergency centres. The differences found in the inter-model comparison for the basic atmospheric dispersion parameters like air concentrations at ground level are typical smaller than a factor 2-3 even for complex scenarios in large areas of the calculation grid. Derived values such as deposition and dose values differ in general more. This underlines the importance of a good knowledge of the parameters used for the determination of these derived quantities because they determine in a large group of scenarios the main difference between the different models studied here. In addition the polar grid used in the NOODPLAN models showed that the determination of the maximum value of a quantity is less dependent on the gird compared to the grids used for the other models when the location of the maximum is within a few kilometres from the release point. Optimization of grid format, extension and spacing in function of scenario can in this respect be identified as an interesting topic for model developers. The high degree of customization of the NOODPLAN models in combination with (partly automated) input of source and meteorological data validated by the operators during an emergency guarantees the use of as complete and correct as possible data for performing the radiological impact calculations. Because the NOODPLAN models are limited in e.g. range (50 km) and scope (early phase) the export of all input data in a common file structure for use by other models has been identified as an important topic for future development.

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The use of rain-radar data in the early phase nuclear emergency response

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Abstract

Following an atmospheric release of radioactive material during e.g. a nuclear or radiological incident, it is well known that local precipitation can have a large impact on the deposition pattern. Atmospheric dispersion and deposition models use different approaches to take precipitation into account. Moreover, often only limited precipitation information is available. Although rain-radar data are less accurate than rain intensities that are measured directly, they ensure a more detailed coverage of the potentially affected region, which can, in turn, affect the decisions on management options and measurement strategies.

Data from a C-band Doppler rain-radar located at Wideumont and operated by the Royal Meteorological Institute of Belgium were converted to be used with the Risø Mesoscale puff model (Rimpuff). This model was then used to execute several calculations concerning a possible nuclear event with different rain input data ranging from single station to rain-radar data. Results obtained for different parameters, such as deposition and cloud-shine, corresponding to precipitation measurements from weather stations on the one hand, and from rain-radar data on the other hand, are compared.

This paper assesses the importance of the rain-radar data in the early phase of a nuclear or radiological emergency. The resolution and the uncertainties associated with the different available rain data are discussed. Also, a quantitative analysis is presented of the results obtained when different rain input data are used. Finally, to strengthen our conclusion on the added value of rain-radar data in the early phase, a survey on the opinion of several radiological experts on the use of rain-radar data based on a case study with calculated examples will be discussed.

Is one set of intervention levels for radiation emergency planning enough?

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Abstract

A common set of intervention levels is still missing for introducing early (sheltering, evacuation) and late (relocation, resettlement, returning to evacuated areas) protective actions in radiation emergencies. A two-step method has been suggested to derive a consistent set of intervention levels. In the first step, the projected effective doses are calculated for different accidental release scenarios for three defined time periods which corresponds with integration times for sheltering and relocation and the corresponding dose ratios estimated. In the second step, it is necessary to adopt a reference dose level for a certain time period. The relationships between the projected effective doses and the agreed reference level are used to derive a consistent set of intervention levels for early and late protective actions. In this contribution the recommendation of ICRP to limit the annual dose to the public to 100 mSv in the first year is adopted. Based on this reference level, the intervention levels for the early and late protective actions are derived for a variety of scenarios. The influence of different parameters on the variation of results is analysed and discussed with respect to establish appropriate intervention levels. Based on the results, it is further discussed whether it is of advantage to have only one set of intervention levels which covers even unfavourable conditions, or, whether it might be more appropriate to derive more than one set of intervention levels for different scenarios, which enable decision makers for a flexible response according to the accidental situation.

Introduction

An international agreed set of intervention levels is still missing for introducing early (sheltering, evacuation) and late protective actions (relocation and returning to evacuated areas) in radiation emergency preparedness. International organizations (ICRP 2007; IAEA 2009) recommend a reference level between 20 and 100 mSv per year but they do not give guidance how to split this level into discrete values for sheltering, evacuation and returning, which is of greatest importance for practical decisions. For implementing best radiation protection to the affected population, decision makers have to decide on sheltering and evacuation in an event even before radionuclides are released, and on relocation immediately after the passage of a cloud. As there is not much time for consideration of protective actions, intervention levels for each countermeasure have to be agreed in the planning phase. In this paper sets of intervention levels for early and late protective actions are derived, which are in line with each other. It is discussed whether one set of intervention level is sufficient to deal properly with all kind of radiation emergencies or whether few sets of intervention levels might be more appropriate to cover different scenarios.

Material and methods

The recommendation of ICRP to limit the maximum annual dose to the public to 100 mSv in the first year after the accident occurance is adopted. The time dependent curve of the projected effective doses within the first year is analysed for different scenarios. It is assumed that the integration time for early protective actions will be 2 days and for relocation 30 days and 1 year. Accordingly, effective doses are calculated for the three time periods separately: from the beginning of the release until the end of the 2nd day, from 3rd day until the 30th day and from the 31st day until first year. Then, the ratios of the doses for the three time periods are calculated. When a set of dose ratios is adopted for one scenario, the recommended annual dose of 100 mSv/a is split according to the dose ratios to obtain a consistent set of intervention levels for sheltering and relocation. The variation of the dose ratios leads to different sets of intervention levels for sheltering and relocation, dependent on each scenario considered.

Three different kinds of source terms have been defined:

- two complex source terms for accidental releases from a NPP,
- Iodine-131 release, as example for short lived β- and γ radionuclides, and
- Cs-137 release, as example for long lived β- and γ radionuclides.

Six different weather conditions have been assumed: Pasquill stability class B, with wind speed of 3 m/s; Pasquill stability D, with wind speed 6 m/s; and Pasquill stability F, with wind speed 2 m/s. Both dry and rainy conditions have been assumed. When rain has been considered, a wash-out coefficient of 3.5E-04 s^{-1} has been used for calculations. The emission characteristics are: heigh 50 m, duration 2 hours. The dose calculations have been carried out with the Romanian RO-CODE and the German version of RODOS.

The Romanian RO-CODE is software specifically developed by the representative of Regulatory Authority in Romania as an in-house tool for the assessment of possible radiological consequences of nuclear accidents at Cernavoda NPP (Baciu 2004). The computer code uses the Gaussian plume model to simulate the dispersion and the transport of the radionuclides at different distances from the source. The dose coefficients are taken from Eckerman and Ryman (1993). The RODOS code is described in literature (Ehrhardt and Weis 2000; Ehrhardt et al. 2007).

Results

The results indicate that the amount of doses in the first period (the first two days after the accident occurrence) depends significantly on the weather condition. The projected effective doses vary with the stability class and the kind of deposition – dry or wet. But the dose ratios show a different picture: The ratios of the doses in the three periods are fairly independent from the stability class but differ significantly for wet and dry deposition. The results of both models, presented in Table 1, are in a fairly good agreement.

Table 1. Projected effective doses *E* (mSv) and dose ratios P_n/P_2 (n = 1, 2, 3) calculated with RO-CODE and RODOS for the time periods P_1 to P_3, in 5 km distance assuming different atmospheric stability classes (B, D, and F) as well as dry and wet conditions, for all source terms considered.

Conditions	RO-CODE			RODOS		
	P_1	P_2	P_3	P_1	P_2	P_3
NPP source term - dry conditions						
Stability B						
E (mSv)	24.0	6.3	24.3	168.0	47.2	40.9
P_n/P_2	*3.8*	*1*	*3.9*	*3.6*	*1*	*0.9*
Stability D						
E (mSv)	151.2	39.0	149.0	254.6	74.6	64.0
Pn/P2	*3.9*	*1*	*3.8*	*3.4*	*1*	*0.9*
Stability F						
E (mSv)	591.3	158.0	610.0	2070.0	524.0	518
P_n/P_2	*3.7*	*1*	*3.9*	*3.9*	*1*	*1.0*
NPP source term - wet conditions						
Stability B						
E (mSv)	242	397	1282	1222	2470	3330
P_n/P_2	*0.6*	*1*	*3.2*	*0.5*	*1*	*1.3*
Stability D						
E (mSv)	421	541	1759	1008	2063	3620
P_n/P_2	*0.8*	*1*	*3.3*	*0.5*	*1*	*1.8*
Stability F						
E (mSv)	1266	1890	6190	5200	9650	17200
P_n/P_2	*0.7*	*1*	*3.3*	*0.5*	*1*	*1.8*
Cs- 137 – dry condition						
Stability B						
E (mSv)	42.1	13	160	3.0	1.05	9.4
P_n/P_2	*3*	*1*	*12*	*2.9*	*1*	*9.0*
Cs-137 – wet condition						
Stability B						
E (mSv)	72.9	706.1	8351	5.0	67.8	562.3
P_n/P_2	*0.12*	*1*	*12*	*0.07*	*1*	*8.3*
I-131 – dry condition						
Stability B						
E (mSv)	8.39	2.71	0.30	2.8	1.15	0.1
P_n/P_2	*3*	*1*	*0.1*	*2.4*	1	*<0.1*
I-131 – wet condition						
Stability B						
E (mSv)	33.6	142.4	14	5.3	18.4	1.5
P_n/P_2	*0.25*	*1*	*0.1*	*0.3*	*1*	*0.15*

To derive sets of intervention levels, the annual reference dose of 100 mSv has been split according to the dose ratios obtained for the different scenaris (table 2)

Table 2. Derived sets of intervention levels for sheltering and relocation, for all source terms considered (source term, Cs-137 and I-131, in dry and wet conditions.

Conditions	Sheltering (mSv, 2 days)	Relocation (mSv, 1st month)	Relocation/ Resettlement (mSv, 1st year)
Dry deposition			
Complex source term for NPP			
Intervention level	50 mSv/2 days	65 mSv/1st month	100 mSv/1st year
Dose ratio	4	1	2
Cs-137 release			
Intervention level	22 mSv/2 days	30 mSv/1st month	100 mSv/1st year
Dose ratio	3	1	10
I-131 release			
Intervention level	75 mSv/2 days	100 mSv/1st month	100 mSv/1st year
Dose ratio	3	1	0.1
Wet deposition			
Complex source term for NPP			
Intervention level	12,5 mSv/2 days	37.5 mSv/1st month	100 mSv/1st year
Dose ratio	0,5	1	2.5
Cs-137 release			
Intervention level	1 mSv/2 days	10 mSv/1st month	100 mSv/1st year
Dose ratio	0.1	1	10
I-131 release			
Intervention level	30 mSv/2 days	100 mSv/1st month	100 mSv/1st year
Dose ratio	0.3	1	0.15

Concerning the source term of a NPP and assuming dry deposition, about half of the annual dose is received during the first 2 days when the radioactive cloud passes. Concerning wet conditions, the dose in first 2 days is only about one eighth of the annual total dose. What does this mean with respect to emergency management? Many countries have settled an intervention level for sheltering of 10 mSv integrated over 2 days. This means when the dose of 10 mSv is exceed in the first two days and it is raining, then it has to be assumed that also reference dose of 100 mSv for one year might be exceeded and relocation have to be taken into account directly after the cloud´s passage. But a completely different picture is given when it does not rain during the cloud´s passage. The projected dose integrated over 2 days could be significantly higher than 10 mSv but relocation must not be considered because the annual dose of 100 mSv will be not be exceeded.

It is obvious that the results strongly depend on assumption on the radionuclide composition. To get an idea on dose ratios for the different time periods, calculation were carried out for I-131 as representative of a short lived β- and γ radionuclide and for Cs-137 as representative for long lived β- and γ radionuclide. Concerning I-131 about 70 % of the dose is received during the passage of a cloud in the absence of precipitations. Under wet conditions, the dose ratios are vice versa. About 30 % of the dose is received in the first 2 days but about 70 % in the next 28 days.

Opposite results are obtained for Cs-137. About 25% of the annual dose is received in the first 2 days assuming dry deposition which means that inhalation plays a significant role during the passage of the cloud. In the later periods, when direct deposition is the dominant pathway, the monthly exposure declines slowly. Concerning wet deposition, the dose from inhalation is becoming of minor importance. The dominant pathway is direct radiation in each period. Consequenty only about 1/100 of the total annual dose is received in the first 2 days during the passage of the cloud. But when the doses are fairly constant over 1 year, late countermeasures become more relevant than short term ones.

Conclusions

First analysis show that sets of intervention levels vary significantly with the fraction of long-lived and short-lived radionuclide in the source term and with dry or wet deposition. But the dose ratios and consequently the derived intervention levels are fairly independent from the stability class.

Concerning dry deposition, an intervention level for sheltering of 10 mSv per 2 days would make sure that the annual dose of 100 mSv per year will not be exceeded in all examples. The annual dose will even stay below 50 mSv in these cases. This means when the intervention level for sheltering is not met, also relocation must not be recommended too. The derived values are in line with each other.

A different picture has been received for wet deposition. The contribution of the dose received during the passage of a cloud to the total annual dose is significant lower for all sources considered. For I-131, the intervention level of 30 mSv during the first 2 days corresponds with an annual dose of 100 mSv. Concerning releases from NPPs, an annual dose of 100 mSv can be expected when the dose within the first period is expected to be a little more than 10 mSv for wet deposition. But concerning Cs-137, a dose of only 1 mSv during the first 2 days corresponds with an annual dose of 100 mSv. This means that the dose reduction by implementing sheltering will be insignificant in comparison to the annual dose. Consequently, when the dose is below 1 mSv in the first 2 days, then neither sheltering not relocation have to be considered. Above 1 mSv per 2 days, countermeasures to reduce direct radiation have to be taken into account immediately. Otherwise evacuation has directly to be taken into account as protective action for population.

Coming back to the question whether one set of intervention level is sufficient for emergency planning: The answer is complex: a distinction between wet and dry deposition seems to be appropriate due to the differences in results. But on the other hand fast and common decisions can rather be expected when only one set of intervention has been settled. Based on these calculations, it is would be possible to take into account the intervention levels derived for a release at a NPP under wet conditions. It covers all situation analysed, excepting the very special situations when only long lived β- and γ emitting radionuclides are released under wet conditions. In this special situation, evacuation, respectively relocation has to be considered directly.

The objective of this contribution is to address the problem of setting up intervention levels which are in line with each other. It is obvious that additional scenarios and assumptions, like different radionuclide compositions, different deposition rates and washout coefficients have to be analysed to create a good basis for a final conclusion whether one set of intervention levels is appropriate in radiation emergency preparedness.

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Technical considerations for protective action strategy in nuclear emergency using probabilistic accident consequence assessment model

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Abstract

If an accident occurred at nuclear power plant and any plume of radioactive materials was released into the environment, country and local government immediately have to implement protective actions for people around the plant to reduce exposed dose. In order to efficiently implement protective actions, it is important to preliminarily develop the good planning for emergency preparedness and response to a nuclear accident. In particular, it is important to provide the technical guidance for the development of urgent protective actions (e.g. evacuation, sheltering and iodine prophylaxis), based upon a comprehensive threat assessment that takes into account the full range of postulated events for nuclear accidents. The most effective response strategy in nuclear emergency will involve a combination of these protective actions. The ICRP recommends focusing on optimization with respect to the overall strategy, rather than the individual protective action in the ICRP new recommendation, Publication 103. To develop a generic response strategy, probabilistic accident consequence assessment models can be very useful for providing a quantitative basis on discussing the effective emergency plan. Our study shows results of the technical considerations for the overall strategy of implementing protective actions in emergency exposure situation using a probabilistic accident consequence assessment model by Japan Atomic Energy Agency. For postulated accidents with source terms derived from a generic level 2 PSA of the reference plant in Japan, residual doses are calculated in case of implementing combination of these protective actions and then optimization process being guided by reference levels is applied. The preliminary results will provide the insights of technical guidance for the development of these protective actions.

System for the prognosis of the population doses due to emergency atmospheric release from nuclear power plants

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Abstract

A description of the structure and capabilities of the system used for assessment and prediction of the effects of radioactive releases from Rivne Nuclear Power Plant (RNPP) is given. The system includes models for calculation of the atmospheric transport, external and internal doses, as well as module for justification assessment of emergency and urgent countermeasures. The calculations of possible impact of beyond design basis accident for the population within RNPP's supervised area were performed. It is shown that an increase in efficiency of iodine prophylaxis by using the prediction results of the radiation situation can provide a significant reduction of avertable dose to thyroid.

Introduction

In 2007 Ukrainian state enterprise National Nuclear Energy Generating Company (NNEGC) "Energoatom" adopted the concept of a computer decision support system (DSS) in the event of radiation accidents at Nuclear Power Plant (NPP), according to which DSS must have a two-level structure, consisting of two subsystems:

– site (plant) subsystem which placed at each NPP in Ukraine;
– central subsystem which placed in the Crisis Centre of NNEGC "Energoatom".

The main element of the DSS is site subsystem that efficiently should perform decision-making support at an early phase of the accident according to requirements of the Typical emergency plan for NPP in Ukraine.

System of operative analysis of radiation situation due to accidents at NPP in Ukraine (SOARS) was established by the Radiation Protection Institute (Kyiv) for dose calculation and support decisions about countermeasures. SOARS is used at Rivne NPP from 2003, it fully complies with the requirements of Radiation Safety Standards of Ukraine. The paper is devoted to a brief description of its capabilities and prospects for its further improvement.

The generic structure of SOARS

System of operative analysis of radiation situation is intended for calculation the consequences of atmospheric releases from NPP within its supervised area at an early stage of the accident. It includes following main modules:

- module for calculation of atmospheric transport and fallout to the ground surface;
- module for calculation of external doses from radioactive clouds;
- module for calculation of external doses from fallout to the ground surface;
- module for calculation of internal doses due to inhalation;
- module for calculation of internal doses due to ingestion of contaminated food (pilot module);
- module for countermeasures.

Fig. 1 shows the appearance of the main window of the system and a fragment of one of the auxiliary window with summary results of dose calculations and countermeasures for settlements in NPP supervised area.

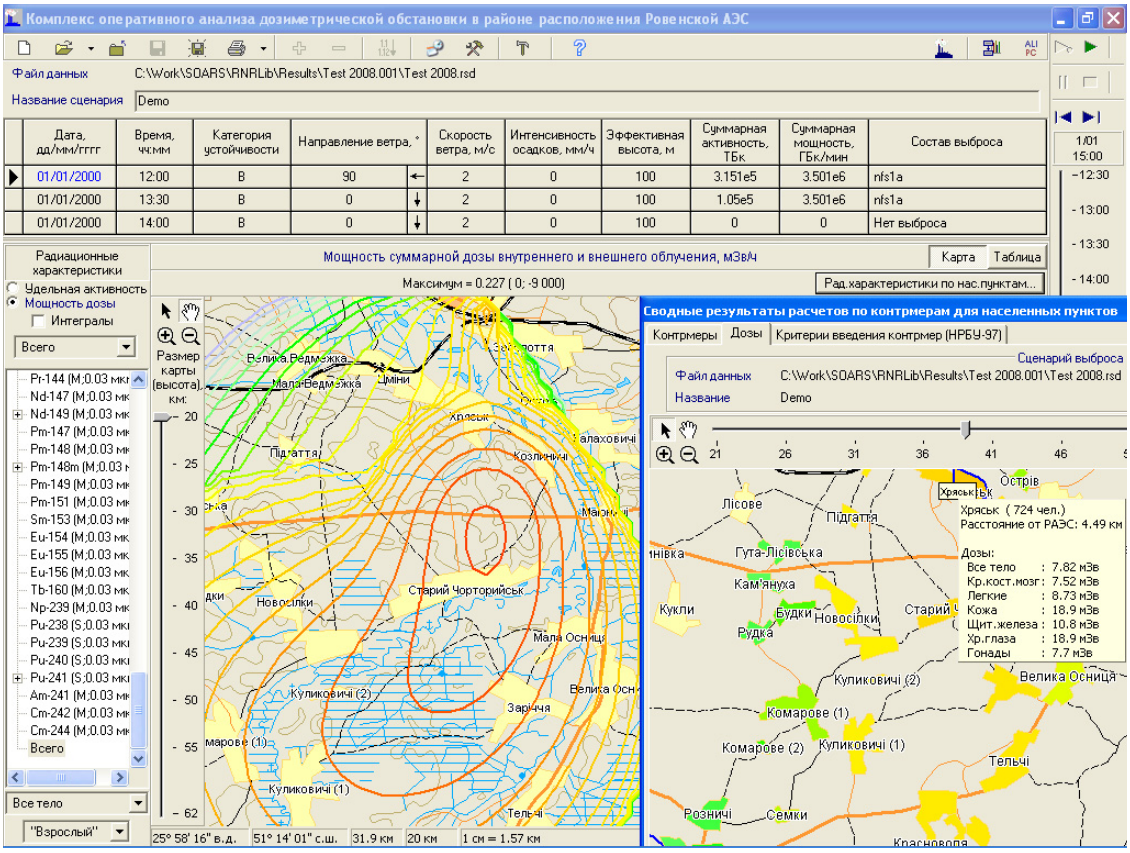

Fig. 1. The main window of SOARS and a fragment of *Summary results* window.

Atmospheric transport module

The module calculates the field of volume specific activities of radionuclides in the surface air and surface specific activities of radionuclides in the fallout to the ground surface. It is based on time-dependent model of atmospheric dispersion that is used to calculate the possible consequences of accidental releases at distances up to 30 km. In contrast to the standard Gaussian model of atmospheric dispersion (IAEA 1980), time-dependent atmospheric transport models can be applied in situations characterized by rapidly changing trend of release and in conditions with changing of meteorological fields during the time of pollutant transport (essentially, speed and direction of wind). Prolonged non-stationary release of radionuclides from the source is modelled by a sequence of discrete releases ("puffs") emitted from the source through some (sufficiently small) intervals. The detailed model applied in the SOARS for the calculation of the volume specific activity of radionuclide in the air is given in (Talerko 2005). This approach allows to increase the range of applicability of the model for the transport distances (within the supervised area of NPP) in comparison with the common Gaussian model, which (according to (IAEA 1980)) is recommended for use only at distances up to 10 km from the source. Discussion of features and limitations for the applicability of these models is given in (Talerko 2009).

The source data for calculations consist of release parameters and meteorological data (variable with time): release rate (activities of separate radionuclides), duration of release, release height, stability class, wind speed, wind direction, precipitation rate.

Module for calculation of external doses

The module calculates the effective dose and equivalent dose to organs (tissues), formed by external exposure from radionuclides in the plume and radionuclides in the fallout to the ground surface.

The module contains two models for calculation the external dose from the plume:

a) model of a semi-infinite spatially homogeneous source (IAEA 1980);

b) integrated model of the plume (Talerko 2005).

The homogeneous semi-infinite source model can lead to large errors in dose assessment from radioactive plume on conditions of high spatial gradients of radionuclide activity at small distances from the source of releases. Therefore integral model is used in substantially heterogeneous and rapidly changing fields of activity of radionuclides in the surface air. The model allows clarifying the dose assessment substantially, especially at distances comparable with the release height.

Calculation of effective dose of external exposure from fallout to the ground surface is based on the model of source, uniformly distributed on the infinite surface. Values of external dose rates per unit of surface specific activity of radionuclides on the ground surface are given according to (Eckerman, Leggett 1996).

Module for calculation of internal doses

The module calculates the effective dose and equivalent dose to organs (tissues), formed by internal exposure of radionuclides. The calculation is performed for each radionuclide in release composition and its progeny produced during radioactive decay.

Inhalation. The projected effective dose (for reference age τ), formed by inhalation intake of radionuclide i at time t, calculated as follows:

$$E_{\tau,i}^{inhal}(t)=\int_0^t \dot{E}_{\tau,i}^{inhal}(\theta)\,d\theta\,. \tag{1}$$

Internal dose rate due to inhalation intake ($\dot{E}_{\tau,i}^{inhal}(\theta)$, mSv·h^{-1}) is determined by the formula

$$\dot{E}_{\tau,i}^{inhal}(\theta) = A_{V,i}(x, y, 0, \theta)\dot{g}_{\tau,i}^{inhal}, \tag{2}$$

where $A_{V,i}(x, y, 0, \theta)$ is a volume specific activity of the radionuclide i in the surface air at the time θ (Bq·m^{-3}) for settlement with the coordinates (x, y); $\dot{g}_{\tau,i}^{inhal}$ is an effective internal dose rate due to inhalation for reference age τ per unit volume specific activity of radionuclide i in air, mSv·Bq^{-1}·m^{3}·h^{-1}.

The calculation of $\dot{g}_{\tau,i}^{inhal}$ is based on biokinetic and dosimetry models of the International Commission on Radiological Protection (ICRP). These values are pre-calculated using the IDSS software (Internal Dosimetry Support System), developed by the Ukrainian Radiation Protection Institute (Berkovski et al. 1998). The calculated coefficients are stored in the internal database, which contains data for more than 700 radionuclides (for six reference ages). IDSS demonstrated good correspondence of calculation results with the values of doses per unit intake published by ICRP. Calculation of equivalent dose to organs (tissues) is similar to the calculation of effective doses. For this purpose the corresponding values $\dot{g}_{\tau,i}^{inhal}$ are also stored in SOARS database.

Ingestion. Module for calculation of internal dose due to ingestion intake is a pilot module (under development and testing) which is not included in the SOARS, currently operated in RNPP. The projected effective dose, formed by ingestion of radionuclide i at time t, is calculated in this module as

$$E_{\tau,i}^{ingest}(t) = \sum_{\Theta} E_{\tau,i}^{ingest}(\Theta), \tag{3}$$

where $E_{\tau,i}^{ingest}(\Theta)$ is a projected effective dose for the reference age τ formed by ingestion of radionuclide i during the calendar date Θ.

The summation in the formula is performed on all calendar dates from the beginning of an emergency release to the date corresponding to time t. Effective internal dose for the calendar date Θ is determined by the formula

$$E_{\tau,i}^{ingest}(\Theta) = I_{\tau,i}^{ingest}(\Theta) \cdot e_{\tau,i}^{ingest}, \tag{4}$$

where $I_{\tau,i}^{ingest}(\Theta)$ is an ingestion intake of radionuclide i for calendar date Θ for the reference age τ, Bq; $e_{\tau,i}^{ingest}$ is an effective dose per unit of ingestion of radionuclide i for the reference age τ, Sv·Bq^{-1}.

An ECOSYS model (Muller, Prohl 1993) is the basis of assessment of ingestion intake, however typical Ukrainian food consumption levels are used in the module. A set of pre-calculated daily intakes (for the reference ages), formed by a fallout in every single day of the year, is used for dose calculation. The database of doses per unit intake created by IDSS software (as well as inhalational intake) is also used (Berkovski et al. 1998).

Module for countermeasures

Accordingly to Ukrainian legislation predicted and avertable doses to the population of settlements in supervised area are estimated (RSSU 1997). Predicted absorbed doses to organs (tissues) are calculated for 2 days after emergency release (for decision-making about emergency countermeasures). Avertable doses to whole body, thyroid and skin are calculated for 2 weeks after accident (for decision-making about urgent countermeasures). Avertable doses depend on efficiency and duration of one or several countermeasures (sheltering, evacuation, iodine prophylaxis, outdoor restriction). Scheme of calculation of avertable doses is described below as an example for effective doses.

For external exposure and internal exposure due to inhalation, the effective dose $E_{cm,\tau}$, avertable at time t (for the radionuclide i and for the reference age τ), is calculated as follows:

$$E^{p}_{cm,\tau,i}(t)=\int_0^t\left(1-\frac{1}{k^{p}_{cm,\tau,i}(\theta)}\right)\dot{E}^{p}_{\tau,i}(\theta)\,d\theta\,, \tag{5}$$

where $k^{p}_{cm,\tau,i}$ is a reduction function for the effective dose for reference age τ, radionuclide i, and pathway of dose formation p due to applying of countermeasures ($k^{p}_{cm,\tau,i}(\theta)\geq 1$ for each θ; $k^{p}_{cm,\tau,i}(\theta)\equiv 1$ if countermeasures are not applied or they are not effective); $\dot{E}^{p}_{\tau,i}(\theta)$ is an effective dose rate at time θ for radionuclide i and pathway p.

The effective dose due to ingestion for reference age τ, avertable during time t, is determined by the formula:

$$E^{ingest}_{cm,\tau,i}(t)=\sum_{\Theta}\left(1-\frac{1}{k^{ingest}_{cm,\tau,i}(\Theta)}\right)E^{ingest}_{\tau,i}(\Theta), \tag{6}$$

where $k^{ingest}_{cm,\tau,i}$ is a reduction factor for the effective dose from ingestion of radionuclide i during application of countermeasures in the calendar date Θ; $E^{ingest}_{\tau,i}(\Theta)$ is a projected effective dose from ingestion of radionuclide i during the calendar date Θ.

It is necessary to draw attention to 'behaviour mode' term, for which the following numerical characteristics are specified: for rural population – 0.29, for population of urban villages – 0.2, and for urban population – 0.13 (Likhtariov et al. 1996). The behaviour mode in several times (from 3.4 to 7.7) reduces the estimated value of external dose due to fallout to the ground surface.

For the calculation of avertable doses, SOARS uses charts of countermeasure application, which allows specifying the beginning, duration and efficiency (reduction factor for each pathway). As far as countermeasures can be applied with some delay, their application cannot avert a projected dose completely, but only reduce it (sometimes substantially). Therefore, some countermeasures for settlements may not be justified, despite significant accident doses.

Auxiliary modules

Auxiliary modules SOARS include: module with radionuclides data, reference information module, and geographic information module.

Module with radionuclides data provides detailed radionuclide data (radioactive decay constants, decay chains, spectral characteristics), necessary to perform the calculation of dosimetry situation.

Reference information module performs the displaying of data stored in the module with radionuclides data.

Geographic information module contains cartographic information about Rivne and Volyn regions within a radius of 50 km from RNPP and includes basic map layers (boundaries of the regions and districts, topography, vegetation, water bodies, settlements, roads and railways).

Application SOARS for impact assessments of emergency releases

The application of SOARS for the analysis of the impact of beyond design state accident (BDSA) on the power unit VVER-1000 on RNPP is considered below (NNEGC "Energoatom" 1999). Example demonstrates the possibility of SOARS for decision-making support on countermeasures. The release duration is limited to two hours (for simplifying the analysis), the release indicated in the Table 1.

Table 1. The composition of the release of radionuclides in the BDSA on VVER-1000.

Radionuclide		Release, TBq	
		0 – 1 hour	1 – 2 hrs
Inert radioactive gases	^{85m}Kr	1000	200
	^{87}Kr	2400	200
	^{88}Kr	3700	700
	^{133}Xe	19000	2000
	^{135}Xe	1100	200
Molecular iodine	^{131}I	810	40
	^{132}I	560	30
	^{133}I	560	–
	^{134}I	410	–
	^{135}I	370	40
Organic compounds of iodine (CH_3I)	^{131}I	96	14
	^{132}I	63	7
	^{133}I	67	7
	^{134}I	37	4
	^{135}I	44	4
Caesium	^{134}Cs	25	–
	^{137}Cs	13	–

Calculations were performed for the following meteorological characteristics: atmospheric stability class is E, wind speed is 2 $m \cdot s^{-1}$, wind direction is toward Kuznetsovsk. Calculations were performed for two effective heights of release: 40 m (outflow from the containment) and 120 m (release from the stack). The calculations did not take into account the ingestion intake of radionuclides with contaminated food. This component is essential for the formation of the dose, but it is more delayed in time than other pathways. This paper focuses on the first few hours after the release.

The results of the calculations (projected doses to population of settlements) are presented in Table 2. Since the criteria for urgent countermeasures are not achieved, and criteria for the introduction of urgent countermeasures are exceeded only lower bounds on the justification, specified for the doses to the skin, Table 2 shows only the values of doses to the whole body and thyroid. In the columns "Children" shows the values of doses are the highest among all reference ages, except for reference age "Adult". Mode behaviour in Table 2 is not taken into account.

Table 2. Projected dose exposure of various settlements in supervised area of RNPP (indicating the distance from the RNPP).

Release height	Organ (tissue)	Time	Kuznetsovsk (2,9 km)		Berezina (18,6 km)		Serkhiv (24,9 km)		Kolodii (7,1 km)	
			Children	Adults	Children	Adults	Children	Adults	Children	Adults
40 m	Whole body, mSv	Plume*	56	32	4.7	2.7	3.4	1.9	0.8	0.5
		2 days	65	41	5.3	3.3	3.8	2.3	1.0	0.6
		14 days	81	57	6.5	4.5	4.7	3.2	1.2	0.8
	Thyroid, mGy	Plume*	1000	510	84	43	61	31	15	7.5
		2 days	1010	520	85	43	62	32	15	7.6
		14 days	1030	530	86	45	63	33	15	7.9
120 m	Whole body, mSv	Plume*	37	21	5.8	3.3	4.2	2.4	0.9	0.5
		2 days	43	27	6.6	4.1	4.7	2.9	1.0	0.6
		14 days	54	38	8.3	5.8	5.9	4.1	1.3	0.9
	Thyroid, mGy	Plume*	670	340	110	54	77	39	16	8.2
		2 days	670	340	110	55	78	40	16	8.3
		14 days	680	350	110	57	79	41	17	8.6

* Doses which will be received during the plume passing above settlement are specified.

The main dose formation factor is the inhalation intake of iodine. Its contribution to the total effective dose (for the population of Kuznetsovsk) ranges from 80% (for release height 40 m) to 82% (120 m). Approximately 70% of these values formed by ^{131}I, and up to 8% formed by ^{133}I. The external exposure formed by radioactive inert gases is the following by importance. Its contribution ranges from 9% (for the release height 120 m) to 11% (40 m) of the total dose (most of which goes to ^{88}Kr). The contribution of other ways of the dose formation is considerably less: an external exposure from fallout to the ground surface gives about 4%, and inhalation intake of caesium radionuclides gives about 2%.

The above mentioned fractions are related to the duration of exposure during the plume passage. In considering the longer time intervals (2 and 14 days) for which countermeasures are applied, the growth of the fraction of external exposure from fallout will be observed. As shown in Table 2, the increase may reach 30% for 2 days and 80% for 14 days (for the effective dose). However, for such time intervals a mode of behaviour should be taken into account. Using of it will reduce the specified growth of doses at least in three times. On the other hand, the dose assessment may increase substantially, taking into account the ingestion intake.

Concerning doses to thyroid, it is obvious that they are almost entirely (~99%) are formed by inhalation intake of iodine radionuclides (~85% by ^{131}I, ~12% by ^{133}I) and almost did not increase after the passage of the plume release.

According to Table 2, levels of unconditional justification for outdoor restriction can be exceeded for the population of Kuznetsovsk (for both release heights).

The levels of unconditional justification of iodine prophylaxis may be exceeded for both release heights for children, and for the release height 40 m for adults (lower justification limits may be exceeded only for adults for the release height 120 m). Exceeding of levels for urgent countermeasures intentionally mentioned as possibilities, but it is not obligatory, because it is required to prove (for their justification) the exceeding of avertable (not projected) doses.

Depending of avertable doses from the operability of countermeasures introduction is described below. Since this dependence for the outdoor restriction is more obvious, it focuses on the iodine prophylaxis.

Dependence of the effectiveness of iodine prophylaxis on the time of stable iodine consumption after intake of radionuclide was adopted in accordance with (Prister et al. 2007). Results of calculation of avertable doses are given in Table 3.

Table 3. Dependence of avertable doses from the beginning of iodine prophylaxis.

Release height	Age	Organ/tissue	Projected dose	Avertable dose (depending on the time of the introduction of iodine prophylaxis after the release)				
				1 hour	2 hrs	3 hrs	4 hrs	5 hrs
40 m	Children	Whole body, mSv	56	47	45	37	27	24
		Thyroid, mGy	1000	940	880	740	530	480
	Adults	Whole body, mSv	32	24	23	19	14	12
		Thyroid, mGy	510	470	440	370	270	240
120 m	Children	Whole body, mSv	37	32	30	25	18	16
		Thyroid, mGy	670	630	590	490	350	320
	Adults	Whole body, mSv	21	16	15	13	9.1	8.1
		Thyroid, mGy	340	320	300	250	180	160

The results presented in Table 3, demonstrate a significant reduction of avertable doses (to thyroid) with increasing of delay of iodine prophylaxis introduction. Although the justification of iodine prophylaxis introduction (for a given release scenario) is remained even after 5 hrs after the accident, the fraction of dose that can be averted is reduced (compared to 1 h) almost two times (from 9 to 48%). The doses that will not be averted due to the late introduction of countermeasures, respectively, exceed half of the estimated doses to thyroid. The value of non-avertable effective dose (to the population) may even exceed the dose limit for the personal (category A). This example demonstrates the necessity for earliest possible introduction of iodine prophylaxis in the event of the considered BDSA.

Conclusions

SOARS is installed at RNPP (4 units) for the decision support in case of accidental releases of RNPP. SOARS was operated since 2003 and was commended by experts from the International Atomic Energy Agency during the OSART (Operational Safety Review Team) mission and by experts of the three missions of WANO (World Association of Nuclear Operators plants). The latest comprehensive emergency training, during which SOARS has been successfully used, were performed by NNEGC "Energoatom" in September 2008.

The further development and improvement of SOARS are planned:

1. Formation of the release scenario database at different accidents at NPP with VVER reactors.
2. Development and improvement of the atmospheric transport module to taking into account physical and geographical peculiarities of the supervised area of the NPP (complex relief, presence of large water bodies, and irregularities of the underlying surface).
3. Inclusion of ingestion intake of radionuclides to the module for dose calculation.
4. Automatic input of meteorological information for real-time of system operation.
5. Ability to assimilate the radiation monitoring data from the supervised zone of NPP to clarify the characteristics of radioactive contamination and the parameters of release sources (inverse problem of atmospheric dispersion and dosimetry).

Taking into account the prospects for above listed expanding of functionality, NNEGC "Energoatom" spreads SOARS for all 4 operating Ukrainian NPPs (15 units). Works on adaptation of SOARS for use at Khmelnitsky NPP (2 units) are carrying out now.

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Dose assessment for population in case of a beyound design basis accident at NPP

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Abstract

Radiation protection of the population in case of a reactor accident utilizes reference levels and generic criteria which are based on dose values. Therefore adequate provisions for effective and timely dose assessment for population in case of beyond design basis accident (BDBA) at NPP are important. Developing the background for such provisions is the objective of this study. In the result of our calculations and their analysis versus international recommendations it was found that after the BDBA at NPP with a PWR reactor there is no need for evacuation or providing sheltering for the public because the total effective dose will not reach currently recommended generic criteria (100 mSv in the first 7 days). However, emergency plans should be made to: a) recommend to the public to avoid eating potentially contaminated food or milk; b) perform an environmental monitoring and monitoring of food, water and fodder within 25 km from the NPP; c) provide food monitoring on the territory about 300 km round the NPP. The results of the study could be used for emergency planning purposes.

Introduction

One of the most important aspects of managing a nuclear emergency is the ability to promptly and adequately estimate the consequences of an accident. Because of the need for protective actions to be initiated promptly in order to be effective, nuclear accident assessment must take account of all information that is available to on-site and off-site organizations.

Radioecological modeling is used widely to assess the consequences of potential or ongoing releases at NPPs within the NPP country and beyond the national borders.

Models usually require information that might be available during an emergency. The data required are accident location, either an assessment of plant conditions or an estimated source term, and basic meteorological information.

All these models have been developed carefully according to the present state of science in the field of radioecology. Nevertheless, radioecological models are approximations of reality and hence deviations from real accident conditions may occur. The bases of all existing models is the principles of radioecological prognosis which take into account such parameters as source term, release path, meteorological data, breathing rate, the type of dwelling (e.g. shielding characteristics), dietary habits (e.g. special foodstuffs and amounts consumed), domestic habits (e.g. time spent indoors, frequency of personal washing, and laundering of clothes), etc.

Analysis of the consequences of beyond design basis accidents (BDBA) for the purposes of emergency preparedness and response are usually made for determination of sizes of emergency zones for threat category I and II facilities. Radiation protection of the population in case of a reactor accident utilizes reference levels and generic criteria which are based on dose values [1]. Therefore adequate provisions for effective and timely dose assessment for population in case of BDBA at NPP are important.

Material and methods

Radiological consequences for the beyond design basis accident at NPP with a PWR reactor (or WWER in case of a Russian design) were studied.

Beyond design basis accident is defined in the IAEA Safety Glossary as accident conditions more severe than as design basis accident (accident conditions against which a facility is designed according to established design criteria, and for which the damage to the fuel and the release of radioactive material are kept within authorized limits) [2].

Release out of primary system through the containment to the atmosphere was included in the considered scenario.

A gap release assumes that the core is damaged and all fuel pins have failed, releasing the gaseous fission products contained in the fuel pin gap. For each release route, the mechanisms that will substantially reduce the release are considered (e.g., containment sprays). The effectiveness of the reduction mechanism used is representative for a range of assumptions. In the case of this study it is a release from the containment. It assumes a release into the containment which leaks to the atmosphere. It is reduced it the release from the fuel passes through some process (e.g., sprays, filters) on the way to the atmosphere that will remove a large fraction of the iodine and particulate. It is assumed that an atmospheric release route is not reduced. The release is not reduced for large or wet releases because filters may clog and fail. Normally it is assumed that the release is not reduced if the filters are in containment because a large release may clog them. Hold up time (average time radioactive material released from core remains in containment before release) assumed to be zero.

The release rate will be very difficult if not impossible to determine early in an accident and yet it is the single most important factor in determining off-site consequences. Therefore it is important to bound the release rates by selecting the rate closest to best estimate and a reasonable worst case scenario. Most of the release from the core were chosen to be released within 2,5 hours.

Release from the containment assumed as a ground-level release, not isolated release location. In this case building wake effects need to be considered. If the wind speed was zero the release would be considered as a release from the definite height [4].

Parameters of the model used for calculations are presented in the Table 1.

Table 1. Plant conditions.

Parameter	Value
Power of reactor PWR-1000	3000 MW (th)
Level of core damage	10-50 % core melt (rapid release of volatile fission products)
Leak rate	0,02 %/hour
Release condition	Not reduced (no spray or pool, filters are assumed to blow out)
Hold up time	any
Release height	0 m
Containment volume	71040 m^3
Containment surface space	53250 m^2

Real meteorological data which typical for winter and summer seasons in Belarus accordingly were used for calculations (see Table 2).

All plant-specific and meteorological data have been selected for the purpose of the assessment of the worst case scenario of beyond design basis accident.

Table 2. Meteorological data for "winter" and "summer" scenarios of beyond design basis accident.

Parameter	Value	
	summer	winter
Wind direction	south-west	west, south-west
Surface wind speed	6,4 – 6,7 m/s	5,5 -11 m/s
Atmospheric pressure	993,7 GPa	1008,0 GPa
Air temperature	20 °C	-2,5 - -1,5 °C at night and in the morning 3,7-1,8 °C – in the day time
Cloudiness	100 %	0 %
Precipitation rate	No rain	1-4 mm/hour, snow 1-1.5 cm height
Mixing layer height	0,6 км	1,2 - 1,5 km at night 0,5 - 0,3 km in the day time and at evening
Stability class	D	F

Building shielding factor assumed to be zero, i.e. population spent 100 % of the time outdoors (conservative assessment).

Source term estimation in case of BDBA:

$$SourceTerm_i = FPI_i \cdot CRF_i \cdot \prod_{j=1}^{N} RDF_{(i.j)} \cdot EF_i \, , \qquad (1)$$

FPI_i – Isotope inventory;

CRF_i – Amount of isotope i released out of core/core inventory of isotope i;

RDF_i – Fraction of the isotope i activity available for release following reduction mechanism j;

EF – Fraction of activity available for release that is released [3].

Activity of radionuclides and doses for the public in case of the BDBA were calculated using The International Radiological Assessment System (InterRAS) software, which was developed for use by personnel who responds to a nuclear l emergencies. The InterRAS is a set of personal computer-based tools. InterRAS Version 1 is based on the U.S. NRC's RASCAL Version 2.1 code (NRC94) but was modified to allow assessment a greater range of accidents and to conform to the guidance in the IAEA Basic Safety Standard (IAEA96) [4, 5].

The “Source Term to Dose” (ST-DOSE) model was used. This model contains tools to estimate the distance at which urgent protective actions may be needed based on nuclear power plant conditions or release rates. The ST-Dose model will first generate a time-dependent “source term”, the release rate for each radionuclide from the facility as a function of time. The time-dependent release rate (the “source term”) than provides the input to an atmospheric dispersion and transport model.

The atmospheric dispersion and transport model estimates radionuclide concentrations to downwind, both in the air and in the ground deposition. The calculated concentrations than are used to estimate projected doses. The dose pathways are: cloudshine from the plume, inhalation from the plume, and groundshine from deposited radionuclides (assuming 4 days of exposure to groundshine).

Radionuclide concentrations downwind from the NPP and projected doses due to the accident were estimated at distances of 1, 2, 5, 25 and 50 km from the site.

Results

Results of calculation of total activity of the accidental release in case of BDBA at NPP with a PWR reactor are presented in the Table 3.

Table 3. Activity of radionuclides released to the atmosphere, Bq.

Nuclide	Activity, Bq	Nuclide	Activity, Bq	Nuclide	Activity, Bq
Kr-85	1,00E+13	Kr-85m	4,2E+14	Kr-87	8,4E+14
Kr-88	1,2E+15	Sr-89	3,9E+13	Sr-90	1,5E+12
Sr-91	4,60E+13	Y-91	3,30E+12	Mo-99	1,80E+13
Tc-99m	1,80E+13	Ru-103	1,20E+13	Ru-106	2,70E+12
Sb-127	1,2E+13	Sb-129	6,9E+13	Te-129m	1,1E+13
Te-131m	2,5E+13	Te-132	2,5E+14	I-131	4,1E+14
I-132	5,8E+14	I-133	8,3E+14	I-134	9,2E+14
I-135	7,3E+14	Xe-131m	1,7E+13	Xe-133	3,0E+15
Xe-133m	1,1E+14	Xe-135	5,8E+14	Xe-138	3,0E+15
Cs-134	2,6E+13	Cs-136	1,0E+13	Cs-137	1,70E+13
Ba-140	8,8E+13	La-140	4,40E+12	Ce-144	1,2E+13
Np-239	2,3E+14	Rb-88	1,2E+15	Rh-106	2,7E+12
Te-129	1,10E+13	Xe-135m	1,2E+14	Ba-137m	1,70E+13
Pr-144	1,2E+13	–	–	–	–

Total activity of the accidental release was $1{,}50 \times 10^{16}$ Bq for all the scenarios of BDBA.

The dose values obtained using the InterRAS in case of "winter" and "summer" scenario of beyond design basis accident are presented in tables 4-5.

Table 4. Maximum early doses for population in case of "winter" scenario of beyond design basis accident

Distance from site, km	Cloud shine dose, mSv	Ground shine dose, mSv	Effective inhalation dose, mSv
1	3,50	11,0	79,0
2	2,40	6,30	47,0
5	1,10	2,90	22,0
25	0,14	0,18	1,30
50	0,11	0,13	1,00

Table 5. Maximum early doses for population in case of "summer" scenario of beyond design basis accident

Distance from site, km	Cloud shine dose, mSv	Ground shine dose, mSv	Effective inhalation dose, mSv
1	2,10	5,40	40,0
2	0,97	2,30	17,0
5	0,44	0,66	5,00
25	0,05	0,05	0,37
50	0,01	0,02	0,14

The total effective dose values in the figures 1-4 were obtained by running InterRAS and interpreting the results of calculations for near and far zones of NPP in case of "winter" and "summer" scenario.

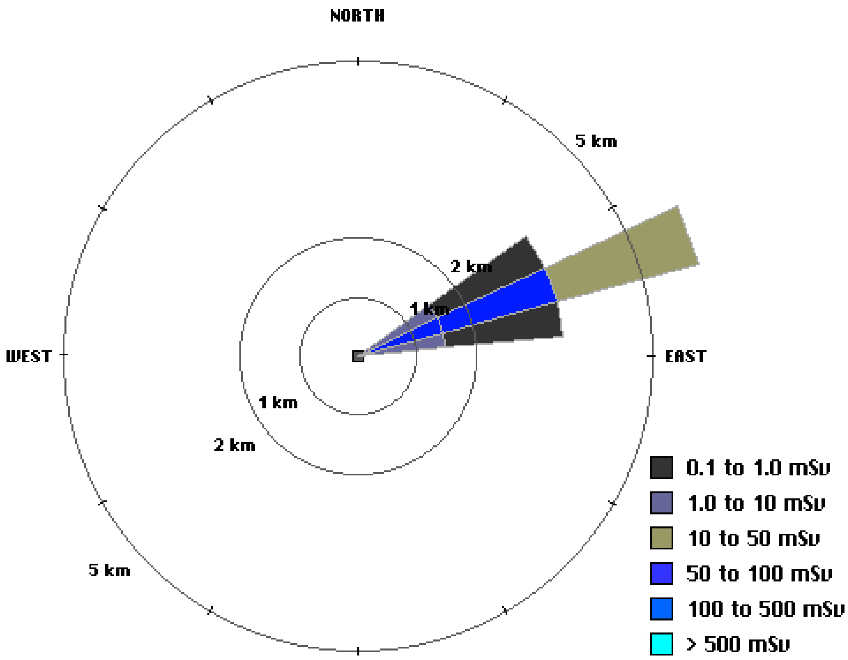

Fig. 1. Total effective dose in the near zone of NPP in case of "winter" scenario of BDBA, mSv.

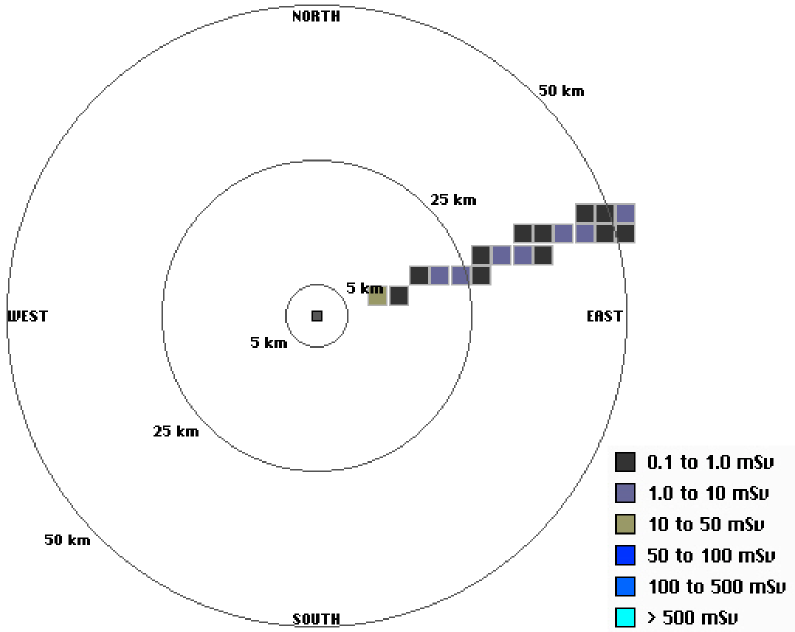

Fig. 2. Total effective dose in the far zone of NPP in case of "winter" scenario of BDBA, mSv.

In case of "winter" scenario the highest value of total effective dose for population living in the near zone was 94 mSv, bearing 70° (south-west) at a distance of 1 km from the site (see figure 1), the highest value for the population living in the far zone was 1 mSv, at a distance of 10,5 km, bearing 72°(see figure 2). The total effective dose at a distance of 50 km was 1,24 mSv.

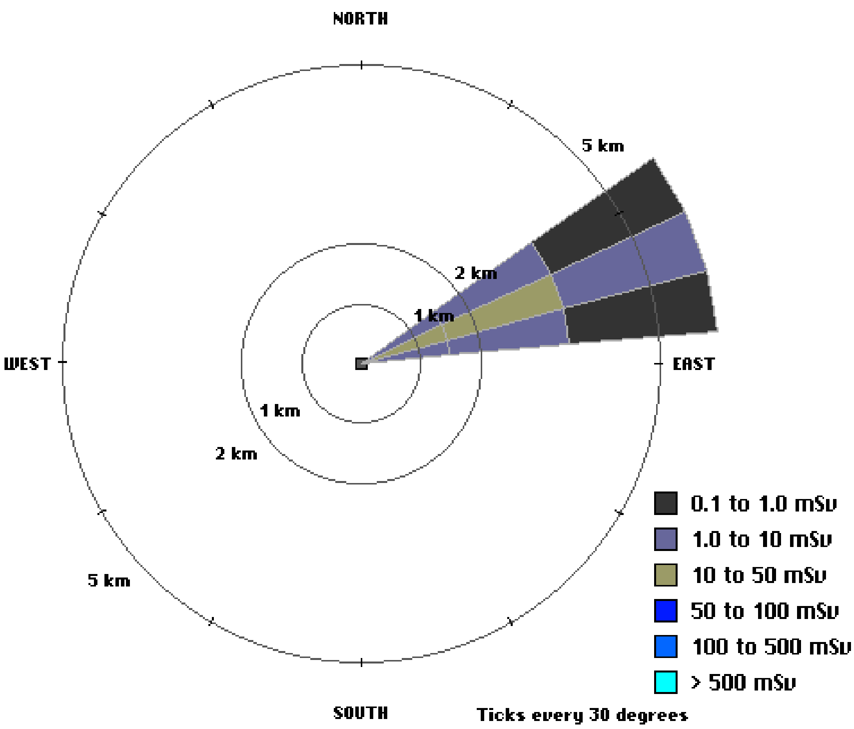

Fig. 3. Total effective dose in the near zone of NPP in case of "summer" scenario of BDBA, mSv.

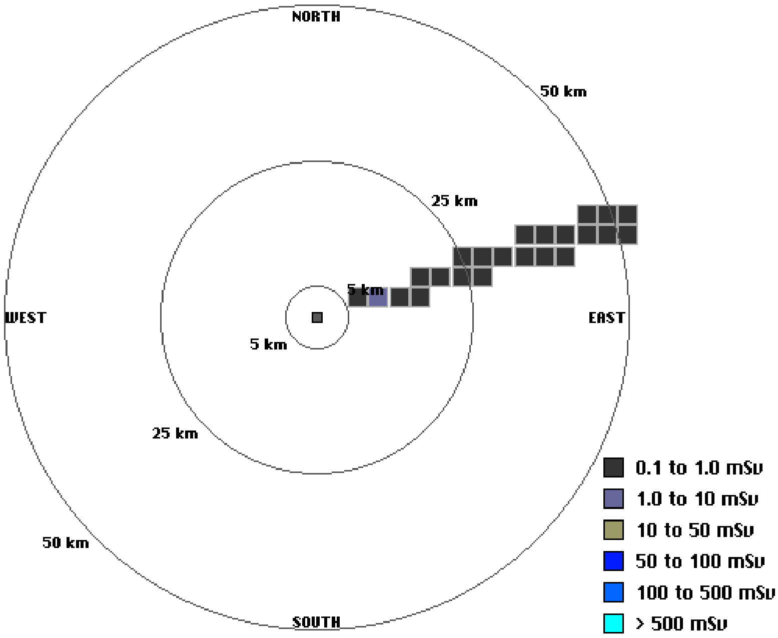

Fig. 4. Total effective dose in the far zone of NPP in case of "summer" scenario of BDBA, mSv.

In case of "summer" scenario the highest value of total effective dose for population living in the near zone was 47,5 mSv, bearing 70° (south-west) at a distance of 1 km from the site (see figure 3). The highest value for the population living in the far zone was 3 mSv, at a distance of 10,5 km at the same direction (see figure 4). At a distance of 50 km the total effective dose was 0,17 mSv.

Dose decreases with distance as shown in the formulas (1/R — for no rain and $1/R^2$ — with rain) [3]. It means that total effective dose decreases with increasing of the distance from the site (see figure 5). During Chernobyl accident this was a reasonable assumption if all the deposition is averaged, but is not valid for local contamination (hot spots).

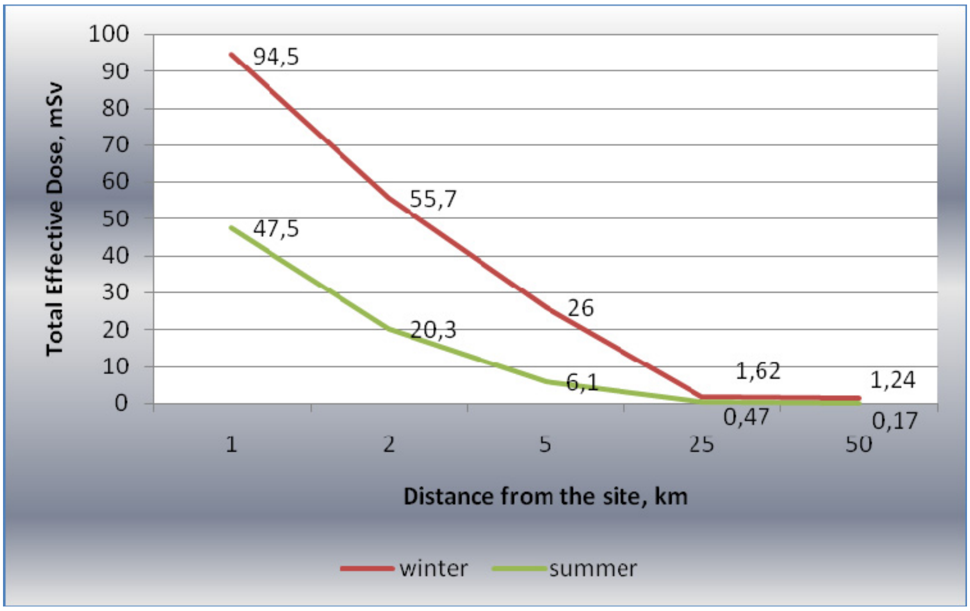

Fig. 5. Decreasing of total effective dose with increasing of the distance from the site.

Discussion

Class stability F ("winter" scenario) is the most unfavorable for radioactive release dispersion, as in this case the highest doses were observed (see Table 4). But dose reduction factors (e.g., building shielding factor) were not taken into account in calculations. In real situation doses for public in winter period would be much less that calculated numbers. Contrary to that, if the accident would happen in summer period, the doses for public could increase due to consumption of locally produced food which will appear to be contaminated.

The primary objectives for protection are to prevent the occurrence of deterministic effects in individuals by keeping doses below the relevant threshold and to ensure that all reasonable steps are taken to reduce the occurrence of stochastic effects in the population.

For facilities in threat category I (e.g., NPP) arrangements shall be made for effectively making and implementing decisions on urgent protective actions off the site [6]. These arrangements shall include the specification of off-site emergency zones for which arrangements shall be made for taking urgent protective action. These emergency zones shall be contiguous across national borders, where appropriate, and shall include:

(i) A precautionary action zone (3-5 km), for which arrangements shall be made with the goal of taking precautionary urgent protective action before a release of radioactive material occurs or shortly after a release of radioactive material begins, on the basis of conditions at the facility (such as the emergency classification) in order to reduce substantially the risk of severe deterministic health effects.

(ii) An urgent protective action planning zone (25 km), for which arrangements shall be made for urgent protective action to be taken promptly, in order to avert doses off the site in accordance with international standards.

(iii) Food restriction planning radius (300 km) – this is the area where preparations for effective implementation of protective actions to reduce the risk of stochastic health effects from the ingestion of locally grown food should be developed in advance.

In general, protective actions such as relocation, food restriction and agricultural countermeasures will be based on environmental monitoring and food sampling.

The sizes are shown in terms of a radius of a circle centered at the source of the potential release or criticality. However, the actual boundary of the zones should not be a circle but should be established to conform the geographical features such as roads, rivers or political boundaries [7].

Conclusions

In the result of our calculations and their analysis versus international recommendations it was found that after the BDBA at NPP with a PWR reactor there is no need for evacuation or providing sheltering for the public because the total effective dose will not achieve currently recommended generic criteria (100 mSv in the first 7 days) [8]. However, plans should be made to:

a) recommend to the public to avoid eating potentially contaminated food or milk;
b) perform an environmental monitoring and monitoring of food, water and fodder within 25 km from the NPP;

c) provide food monitoring on the territory about 300 km around the NPP.

The need for these actions may be than revised based on prevailing accidental conditions, environmental measurements and actual dose values promoted and/or received in an emergency. The results of the study could be used for emergency planning purposes.

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Prognosis of thyroid doses in case of an accident at a nuclear power plant

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Abstract

Despite all the precautions that are taken at the design stage and during the operation of nuclear power plants (NPP), there remains a possibility that failures of any of the systems or of the equipment may lead to a nuclear emergency with a release of radioactive materials into the environment that may require emergency response actions.

The purpose of this study was to make prognosis of thyroid doses in case of an accident at a NPP. In order to achieve this the following was done:

–assessment of thyroid doses caused by beyond design basis accident (BDBA) at NPP with water-water reactor (at different distances from the station);

–estimation of the length of the territories at which dose exposure may exceed the current international generic criteria for protective and other actions in case of the BDBA (PWR).

For modelling of the accident scenario geographic location and technical characteristics of the station and weather conditions were taken into account.

Simulation and evaluation of the potential nuclear emergency were performed using the software (InterRAS, RASCAL) which takes into account all the above-listed parameters.

The calculations also took into account the contribution of internal thyroid dose due to consumption of contaminated foodstuff into formation of the thyroid dose.

This study allowed estimating the thyroid doses, as well as the necessity of protective and other actions in case of the BDBA.

Results of prognosis using the international models have demonstrated that:

–there is a need in thyroid blocking for population at a distance up to 25 km from the station;

–the possibility of restrictions of consumption for potentially contaminated by iodine-131 food, milk and other products should be preplanned;

The results of modelling performed in this study form a basis for developing national arrangements for response to a BDBA at considered NPP.

Introduction

Today, the priority in nuclear energy development is improvement of the safety of existing nuclear power plants and development of the nuclear power reactors of new generation with advanced security features. High security level is achieved due to improvement of active and introducing passive protective and localizing systems, as well as due to the consequent implementation of the inherent safety concept[1]. Development of modern reactors with the feature of self-protectability can provide the resistance to equipment failures and mistakes of staff, limit radioactive consequences of the most severe accidents, exclude the need in evacuation of population.

However, despite the high security of a new type of nuclear power plants and low probability of occurrence of any possible failures, the probability still exists, and therefore there is no absolute guarantee that the radioactive release due to an accident will not happen (IAEA Safety Standards Series №GS-R-2, 2002.). Among the possible causes of the accident can be a degraded control of nuclear chain reaction in the reactor core, the formation of local criticality while refuelling, transportation and storage of nuclear fuel, degraded heat removing from the fuel elements, etc.

A nuclear accident, which accompanied by release of radioactive materials outside the station, can lead to environmental pollution and exposure of population.

The following categories of initiating events are considered within the safety justification process: design and beyond design basis accidents. Beyond design basis accident is an accident caused by not accounted for the design basis accident initiating events, or which is accompanied in comparison with design basis accident by additional failures of safety systems, the implementation of staff's mistaken decisions, which can lead to serious injuries or to core melt. Reducing of consequences is achieved by severe accident management and / or implementation of emergency plans to protect workers and the public.

As a result of the Chernobyl accident currently there is a significant increase of thyroid cancer incidences among inhabitants of Russia, Ukraine and Belarus (Kenigsberg J. et al 2004). Taking into account the possibility of such increase in case of an accident at a NPP and in order to make prognosis of possible consequences of severe accident at a nuclear installation the following was carried out:

– estimation of thyroid doses at different distances from the station;
– assessment of length of the territories on which the irradiation doses to population may exceed the current international generic criteria for protective actions and other response actions in emergency exposure situations;
– suggestion of measures to respond to such accidents.

Material and methods

Simulation and estimation of consequences of potential nuclear emergency is possible using the software which takes into consideration such significant parameters as geographical location and technical characteristics of NPP, weather conditions during the potential accident and accident plant conditions. InterRAS (The International

[1] Inherent safety concept – is a new level of quality at which the deviation of certain parameters of the NPP from the norm is adjusted automatically without operator or, in case of emergency situation, the reactor is blocked automatically.

Radiological Assessment System) is a set of personal computer-based tools developed for use by personnel who conduct an independent assessment of radiological accidents and protective actions and allows to carry out estimations of the release and doses to the public. Software models are used by specialists of leading organizations in radiation safety area – IAEA's and U.S. NRC's specialists (IAEA-TECDOC-955, 1997).

The model «ST-DOSE» (Source Term To Dose) was used for dose estimation. It allows calculating activities of released radionuclides, to estimate the distance at which the urgent protective actions may be needed and to assess integrated radiation doses which form due to accident release of radioactive material into the atmosphere. The model enables to estimate consequences of potential or actual release and requires only information that might be available during an emergency. The data required are accident location, either an assessment of plant conditions or an estimated source term, and basic meteorological information.

In this study a water-water energy reactor (WWER) accident with dry containment leakage was considered as beyond design basic accident. The scenario assumes releases from the core that are typical of cladding failure or core melt. At the beginning of the accident reactor power was 3200 MWt, 10-50% of core melted (rapid release of volatile fission products).

It was also assumed that the release passed by a dry route through the primary system into the containment atmosphere without passing through any systems (e.g., the suppression poll) that would remove iodine or particulate. Iodine or particulate airborne in the containment can be reduced by factors to account for the actions of sprays, filters or natural processes before release into the atmosphere. This removal is a function of the hold-up time. In this study the hold-up time was set equal to zero, it means that the release into the environment started immediately.

Leak rate of radioactive material is 0.02 % per hour.

Containment sprays and air discharge purification system are off. The release is ground level, the effect of building wake, which leads to the greater spreading of radionuclides around the station, are taken into account.

All above mentioned conditions were chosen like that to consider the worst scenario of the BDBA.

The duration of the radionuclides release into the environment due to the leakage through the containment was set equal to 2.5 hour.

Various real possible weather scenarios were considered to simulate transfer of radionuclides in atmosphere. The worst meteorological conditions, which lead to the highest doses to population, are presented in Tables 1-2. The data correspond to "winter" and "summer" seasons, respectively.

Table 1. Site meteorological data («winter» scenario of BDBA).

Parameter	Parameter value
Wind Direction	western turning into south-western
Wind Speed	5.5 – 11 m/s
Air Pressure	1008.0 hPa

Table 1. Continued.

Parameter	Parameter value
Air Temperature	-2.5 – -1.5 at night and in the morning 3.7 – 1.8 in the afternoon and in the evening
Cloudiness	0 %
Mixing layer height	1200 – 1500m at night 500 – 300m in the afternoon and in the evening
Stability Class	F
Intensity of Precipitation	from 1 to 4 mm/h
Snow cover height	from 0.01 to 0.15 m

Table 2. Site meteorological data («summer» scenario of BDBA).

Parameter	Parameter value
Wind Direction	south-western
Wind Speed	6.4 – 6.7 m/s
Air Pressure	993.7 hPa
Air Temperature	20 °C
Cloudiness	100 %
Mixing layer height	600 m
Stability Class	D
Intensity of Precipitation	None
Snow cover	None

Review of “summer” variant of weather conditions allows taking into account considerable contribution of internal dose caused by the consumption of contaminated foodstuff into total thyroid dose.

Thyroid dose consists of dose due to the inhalation of iodine-131 and dose due to the consumption of contaminated products, generally milk and leafy vegetables. The first constituent that is thyroid dose to adult person while he/she is performing an easy activity was calculated using the InterRAS model. Dose due to the alimentary intake of iodine-131 was calculated by hand to six age groups on the basis of iodine-131 concentration in food (milk, leafy vegetables).

Realise of iodine to the environment caused by the accident at nuclear installation can be estimated using the formula below:

$$Source\ Term_i = FPI_i \cdot CRF_i \cdot \prod_{j=1}^{N} RDF_{(i.j)} \cdot EF_i\ , \tag{1}$$

where: FPI_i – isotope inventory;

CRF_i – amount of isotope i released out of core/core inventory of isotope i;

$RDF_{(i,j)}$ – fraction of the isotope i activity available for release following reduction mechanism j;

EF_i – fraction of activity available for release that is released.

Results

Total release of radionuclides into the environment in case of the above described accident scenario regardless of weather conditions amounted to $1.5 \cdot 10^{16}$ Bq. Activities of emitted iodine isotopes are represented in Table 3.

Table 3. Activity of iodine isotopes released into the environment due to BDBA, Bq.

Radionuclide	Activity, Bq
I-131	4.10E+14
I-132	5.80E+14
I-133	8.30E+14
I-134	9.20E+14
I-135	7.30E+14
TOTAL	3.47E+15

Iodine emission is 23.1 % of the total radioactive material released into the environment as a result of nuclear accident. This value is in a good agreement with mean estimate of the iodine fraction usually released during accidents (IAEA-TECDOC-955, 1997).

Calculated thyroid doses caused by inhalation of iodine for "winter" scenario of BDBA are presented in Figures 1 – 2.

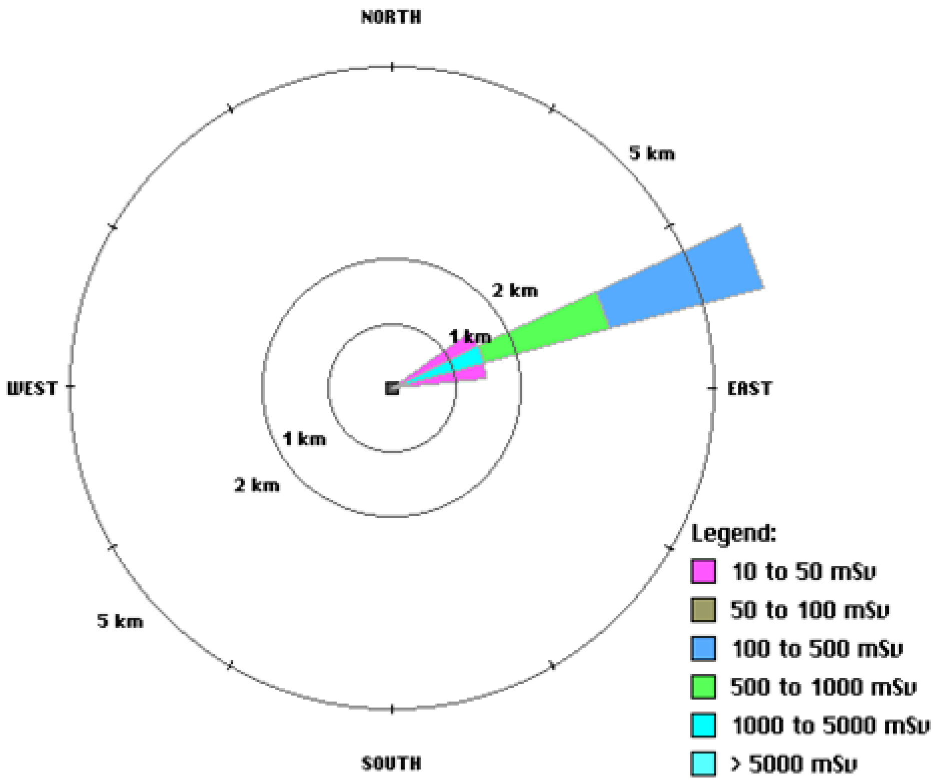

Fig. 1. Thyroid dose at a short distance from the NPP, mGy.

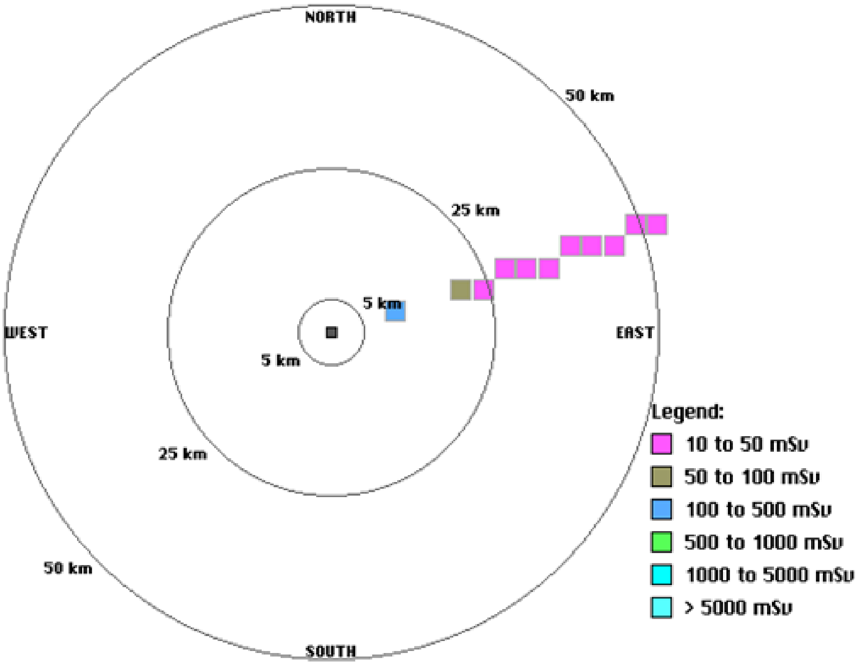

Fig. 2. Thyroid dose at a long distance from the NPP, mGy.

The maximum thyroid dose value for “winter” scenario is observed downwind 1 km away from the station and equals to $1.5 \cdot 10^3$ mGy. At a distance of 10.5 km in the same direction dose to the thyroid may reach $1.7 \cdot 10^2$ mGy.

Estimated doses to thyroid gland caused by inhalation of iodine for “summer” accident scenario are presented in Table 4.

Table 4. – Early doses at different distances from the NPP in case of «summer» scenario of BDBA.

Distance, km	Thyroid dose*, mGy
1	770
2	330
5	95
25	7.1
50	2.6
* Thyroid dose includes iodine only.	

In the “winter” version of the accident the weather corresponds to the atmospheric stability class F (moderately stable condition), which is most unfavourable for the dispersion of the radioactive release, and, consequently, leads to higher doses.

Internal exposure due to consumption of contaminated food contributes significantly to the total thyroid dose in the case of “summer” scenario. If the radioactive fallout occurred in the pasture period, the radionuclides rapidly incorporate into the trophic migration chains through consumption of contaminated pasture vegetation by animals, and then through the chain of pasture - animal - milk get into the human body and deposit at the thyroid gland.

Figures 3 – 4 show the calculated thyroid doses to six age groups caused by consumption of milk and leafy vegetables contaminated by iodine in the case of BDBA.

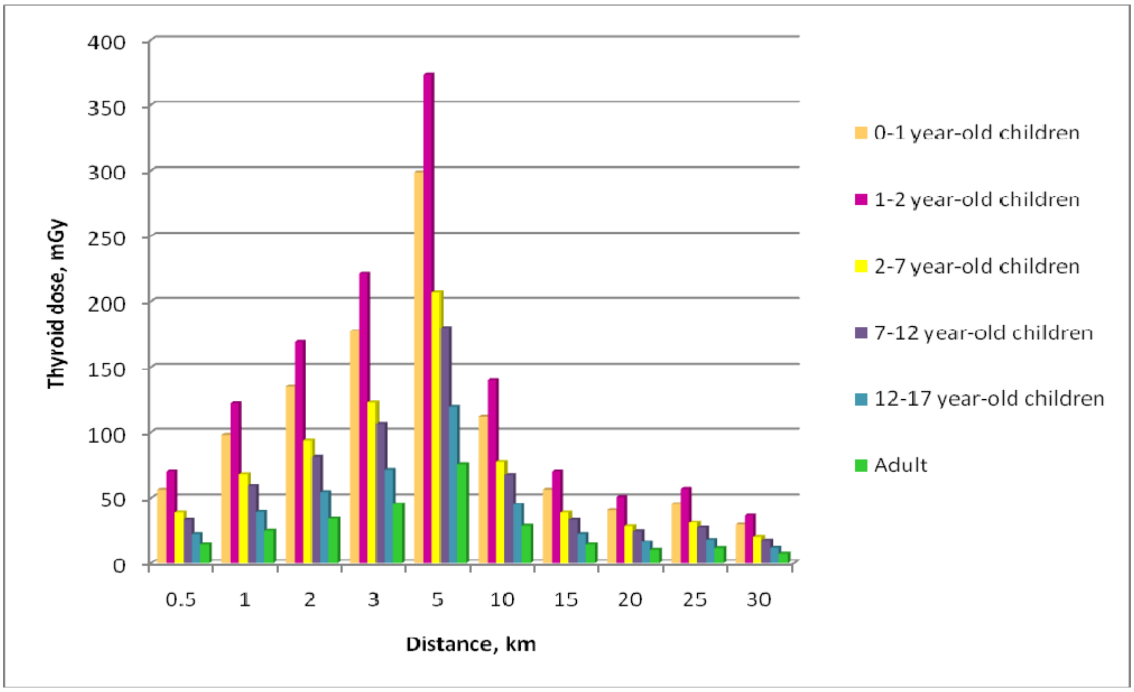

Fig.3. Thyroid dose to six age groups in 30 days after BDBA caused by their consumption of milk contaminated by iodine-131, mGy.

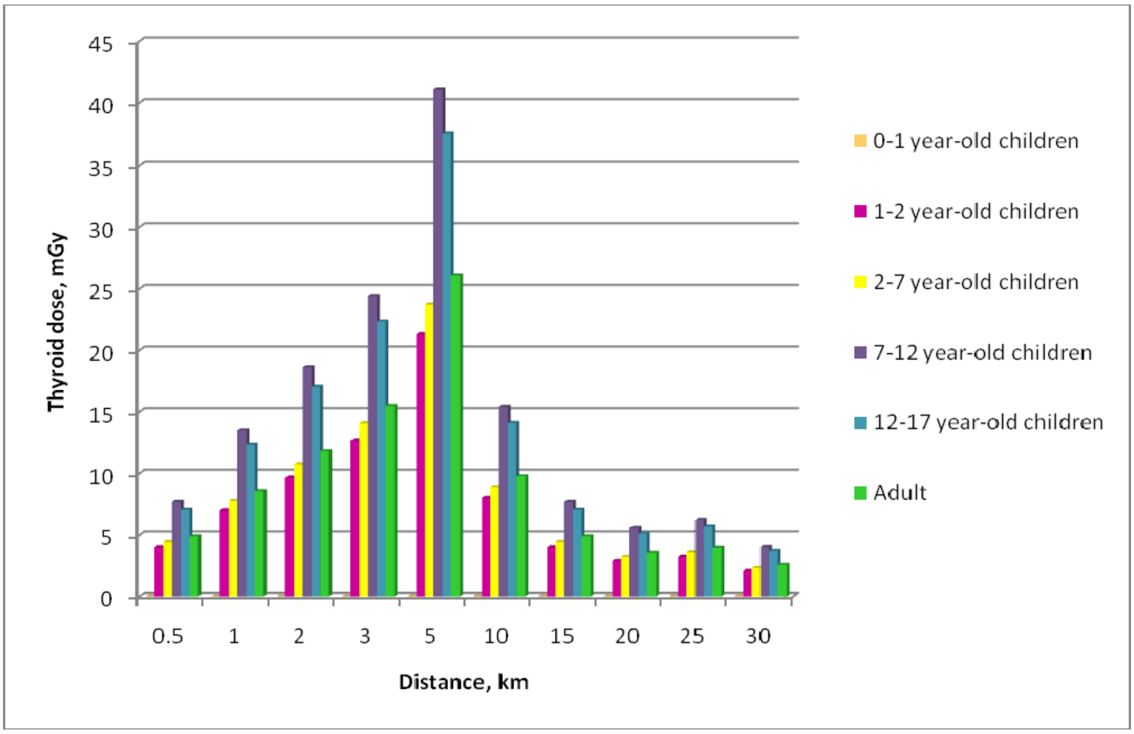

Fig.4. Thyroid dose to six age groups in 30 days after BDBA caused by their consumption of leafy vegetables contaminated by iodine-131, mGy.

The highest thyroid doses due to the intake of contaminated milk and leafy vegetables are observed at a distance of 5 km away from the station, which can be explained by the transition of radioactive cloud by air stream.

Differences in the consumption of considered products are the reason of that thyroid doses caused by the consumption of contaminated milk is much higher than the dose due to the consumption of leafy vegetables. Sex, age, and hence physiological differences lead to the fact that the largest dose because of consumption of contaminated by iodine-131 milk were observed in children that are not older than two years.

Conclusion

One of the most important aspects of managing a nuclear emergency is the ability to promptly and adequately estimate the consequences of an accident. This will help to make a quick justified decision about necessity of protective actions and other response actions and consequently will allow to avoid or to minimize different deterministic and stochastic health effects.

Thyroid dose estimation in case of BDBA showed that maximum doses due to the iodine inhalation are 1500 and 770 mGy under “winter” or “summer” accident conditions, respectively; radiation dose due to the consumption of milk contaminated by iodine-131 is 373 mGy, leafy vegetables – 41 mGy. Total exposure doses exceed generic criteria for protective actions and other response actions in emergency exposure situations – 50 mGy in the first 7 days after accident (IAEA-TECDOC-1432, 2004) – in both considered BDBA scenarios. The results demonstrated that there is a need in thyroid blocking for population at a distance up to 25 km from the station at the early phase of the accident. The possibility of restriction of consumption for potentially contaminated by iodine-131 food, milk and other products should be pre-planned as well. In addition the possibility of urgent environmental monitoring, monitoring of foodstuff and fodder at a distance of not less than 30 km from the station should be ensured, too.

The results of modelling performed in this study form a basis for developing national arrangements for response to a BDBA at considered NPP.

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Dose response of sugar and sweeteners for EPR retrospective dosimetry using sweets and chewing gum carried by victims at nuclear emergencies

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Abstract

After a nuclear accident it is important to reconstruct the individual doses for planning the medical treatment. Another task is to establish moderate or small dose levels, in order to inform people about expected radiation risk. Electron paramagnetic resonance (EPR) spectroscopy provides a mean to measure absorbed doses by quantification of free radicals induced in materials by ionizing radiation. Since people normally do not wear personal dosimeters, it is of interest to investigate radiation induced radicals in materials that are often carried by people with the intent to measure absorbed doses. The objective of this study is to investigate sugars and sweeteners for the possibilities of using sweets and chewing gum for retrospective dosimetry.

Sucrose, sorbitol and xylitol are investigated by means of EPR spectroscopy regarding the dosimetric properties; radical stability, dose response and dependence on environmental factors. The knowledge obtained is applied to one common type of chewing gum by determination of the dose response, obtained accuracy and lowest detectable dose for that specific material.

The sweeteners are sensitive to daylight and the EPR spectrum shows a complicated evolution with time. This is not a problem if the chewing gum is kept in dark and if the signal evolution with time is carefully mapped since the time between exposure and analysis is well known. The EPR spectrum of the irradiated chewing gum is mainly composed by the spectra from sorbitol and xylitol. By identifying the components of the dosimetric signal, the absorbed dose in the chewing gum can be determined and thus make an estimation of the effective dose to exposed individuals possible. With a new ultra sensitive EPR spectrometer, under installation, we expect the lowest detectable dose to be 50 mSv and the uncertainty in dose determinations less than 50 mSv.

Introduction

If a radiation accident occurs, it is of great importance to estimate the radiation dose to exposed individuals as soon as possible. If the absorbed doses are high (>1Gy) dose information may help optimize the medical treatment. At lower doses the information is important for risk assessment and epidemiological studies of long term effects of radiation.

Since people usually do not carry personal dosimeters there is a need to make retrospective dose assessments after the exposure. If several dose assessments are obtained from different locations at the site of exposure, a dose reconstruction map can be drawn. Surveillance cameras at the site of exposure can serve as a source of information on time and location of the individuals.

Electron Paramagnetic Resonance (EPR) is a well established method used for retrospective dosimetry. A recent review study (Trompier et al 2009) evaluates the materials that can be analyzed with EPR. It includes sugar, plastics, glass, cotton, wool, nails, hair, bones and tooth enamel. The most suitable materials for retrospective dose determinations are the ones best fulfilling the following requirements: High radiation induced signal specificity, low background signal, low UV-induced signal, low Minimum Detectable Dose (MDD), linearity at low doses, high signal stability and ubiquity. Sugar was reported well suited regarding radiation sensitivity and signal stability while the ubiquity was mentioned as a drawback.

Sugar has been used several times for retrospective dose reconstructions after real accidents (Hutt et al 1996; Nakajima 1994; Sagstuen et al 1983; Shiraishi 2002). The presence of sugar in confectionery and chewing gums nowadays are not as important, as sweeteners are widely used. A few studies on sweeteners have been performed (Hervé 2006; Hervé et al 2006). Using aspartame doses <1 Gy can be measured (Maghraby & Salama 2010). There is a need to further investigate irradiated sweetening in order to understand the complexity of the EPR spectra. With increased understanding, more accurate dose determinations can be performed. The aim of this work was to characterize and analyze the dose response signals in sweeteners for the possibilities of using chewing gum for retrospective dosimetry.

Material and methods

EPR-dosimetry is based on the quantification of radiation induced free radicals in a sample. This is done by obtaining the magnetization produced by the unpaired electrons of the radicals. When exposing the sample to microwaves while applying and sweeping a magnetic field over it, resonance will occur for the magnetic field values creating electron spin energy levels that match the energy of the microwaves (Weil 2007). Each kind of radical has its own magnetic properties. The radiation induced radicals in sucrose are well characterized (De Cooman et al 2008) but in sweeteners to a lesser degree. Radical studies have been performed on single crystals of xylitol and sorbitol irradiated and measured at low temperatures ≈ 4K (Budzinski et al 1980), where radiation induced radicals are very stable. At higher temperatures radicals are less stable and the EPR spectra changes with time and possibly also with exposure to UV-light.

Since different types of radicals often are saturated at different microwave power, power saturation measurements may help separate signals from different radical types in the EPR spectrum.

The following investigations were performed: Time dependence, Power saturation, Light sensitivity, Sample variation and Dose response. Each experiment followed the same scheme of sample preparation, irradiation, EPR measurement and data analysis. A standard method to perform each phase is described below. Deviations from the standard are then described in the sections of the specific experiments.

Sample preparation

Xylitol[1] and D-sorbitol[2] were purchased from Sigma-Aldrich and chewing gums "dental V6 + white fresh fruit[3]", "dental V6 + strong teeth spearmint[4]" and granulated sugar (sucrose) were bought in Swedish grocery stores. The sweetening of the chewing gum is mainly located in the coating while the inner substance is a gum base mixture. The coating was carefully separated from the rest of the chewing gum. A manual pellet press[5] was used for all sample materials, except sucrose, to form cylindrical tablets . No binding material was needed to form robust tablets. The tablets had masses of 200±5 mg which correspond to dimensions of 10 mm height and 5 mm diameter. Between irradiation and analyses the tablets were stored in light proof containers.

Irradiation

The irradiations were performed at room temperature with a Varian Clinac 600 C/D accelerator using a 6 MV photon beam. During irradiation the samples were placed in a PMMA phantom of height 17 cm and area of 20x20 cm^2 at depth 4 cm. Source to surface distance was 100 cm and the radiation field 10x10 cm^2. The absorbed doses were approximated as dose in water based on the gantry detectors recordings. The samples were irradiated at mean dose rate of 6 Gy/min.

EPR measurement

All measurements were performed with a BRUKER EleXsys E580 EPR spectrometer operating at X-band equipped with an ER 4102ST standard cavity. The tablets were placed in a quartz sample tube inserted in the cavity during measurements. A plastic pedestal containing a Mn^{2+}/MgO reference sample was placed inside the cavity to assure that the sample was centered in the cavity and that the position of the sample tube was kept constant between measurements.

Data analysis

Data analysis was performed in MatLab. The spectra were corrected for having base line slopes by subtraction of a linearly fitted base line. The Mn^{2+} line at g = 2.034 served as reference line as the spectra were normalized to its magnetic field value. For

[1] Xylitol Crystallin, minimum 99%, Sigma-Aldrich Sweden AB®
[2] D-Sorbitol, minimum 98%, Sigma-Aldrich Sweden AB®
[3] Dental V6 +white fresh fruit®, Cadbury Sweden AB
[4] Dental V6 +strong teeth spearmint®, Cadbury Sweden AB
[5] Manual pellet press®, Parr instrument company

the purpose of noise reduction the spectra were thereafter smoothed. The dose response signals were then determined from peak-to-peak signals of the spectrum.

Time dependence measurement

One pressed tablet of each Xylitol, sorbitol and V6[6] were irradiated to 50 Gy. The EPR measurements were performed with 80 sweeps for each tablet and were carried out 6 times from 24 hours after irradiation to 84 days after irradiation. Spectrometer settings were: 1 mW microwave power, 0.2 mT modulation amplitude, 100Hz modulation frequency, 1024 measurement points, 20 s sweep time, 10 ms time constant and the field sweep of width 10 mT was centered between the third and fourth line of the Mn^{2+} reference signal. The obtained EPR spectra were not normalized to the Mn^{2+} line intensity. The background was subtracted from all spectra.

Power saturation measurement

One pressed tablet of each xylitol, sorbitol and V6[6] were irradiated to 50 Gy. The EPR measurements were performed with 8 sweeps of 10 different microwave effects ramped exponentially from 0.2 mW to 100 mW. This was done 1, 16 and 84 days after irradiation. Spectrometer settings were: 0.2 mT modulation amplitude, 100Hz modulation frequency, 1024 measurement points, 20 s sweep time, 10 ms time constant and the field sweep of width 10 mT was centered between the third and fourth line of the Mn^{2+} reference signal.

Light exposure experiment

The samples sensitivity to light exposure was tested by within 15 minutes after a 10 Gy irradiation placing three tablets of each xylitol and sorbitol in a lightproof container while three other tablets of each material were placed exposed to daylight. All tablets were stored for 69 days and then measured with the spectrometer settings: Spectrometer settings were: 100 mW and 63 mW microwave power for sorbitol and xylitol respectively, 0.1 mT modulation amplitude, 100Hz modulation frequency, 1024 measurement points, 10 sweeps of 20 s, 1.28 ms time constant and the field sweep of width 10 mT was centered between the third and fourth line of the Mn^{2+} reference signal.

Measurement of variation between chewing gums

Seven V6 chewing gums[7] from different batches were investigated. Each was irradiated to 1.8 Gy and the coating was then pressed into a tablet. The tablet masses were 102±5 mg. EPR measurements were then performed three days after irradiation. Each tablet was measured once and one of them was chosen to be measured 7 times. The single measurements of the 7 tablets were then compared with the seven measurements of one tablet. Spectrometer settings were: 1 mW microwave power, 0.2 mT modulation amplitude, 100Hz modulation frequency, 1024 measurement points, 80 sweeps of 20 s, 10 ms time constant and the field sweep of width 10 mT was centered between the third and fourth line of the Mn^{2+} reference signal. The spectra were normalized to the

[6] Dental V6 +white fresh fruit
[7] Dental V6 +strong teeth spearmint

intensity of the Mn^{2+} line at g = 2.034. Smoothing of the spectra were also performed so that each point in the EPR-spectrum was an average of the nearest 10 points.

Dose response measurement of V6

V6 chewing gums[8] were irradiated at depth 8.5 cm in a PMMA phantom of surface area 20 cm * 20 cm. 20 cm of PMMA was also placed behind the samples to assure full backscatter. The Source Sample Distance (SSD) was 100 cm and the field size was 12 cm * 12 cm. For accurate dose determinations an NE 2571 ionization chamber traceable to the Swedish secondary standard laboratory was placed at the same depth and distance from field centre to simultaneously determine the absorbed dose in water. Two chewing gums received doses of 2.04 Gy, two 4.07 Gy, two 6.11 Gy and another two were irradiated to a dose unknown to the investigator. Two days after irradiation tablets were pressed of the chewing gums to masses of 100 ± 2mg. The tablets were read out in the spectrometer after one additional day with settings: 20 mW microwave power, 0.5 mT modulation amplitude, 100Hz modulation frequency, 1024 measurement points, 25 sweeps of 20 s, 10 ms time constant and the field sweep of width 10 mT was centered between the third and fourth line of the Mn^{2+} reference signal. A smoothing of N = 5 was performed on each spectra.

Dose response measurement of sucrose

After irradiation to doses from 1 to 14 Gy 400 mg of granulated sugar was filled into tubes of quartz and measured in a JEOL FR30EX Free radical monitor 4 days after irradiation. The settings used were: 4 mW microwave power, 2 mT modulation amplitude, 4096 measurement points, 4 sweeps of 4 s, and the field sweep of width 10 mT was centred between the third and fourth line of the Mn^{2+} reference signal.

Statistics and uncertainty analysis

To obtain the regression parameters A and B in the relation $f(x) = Ax + B$, where $f(x)$ is the EPR peak-to-peak signal and x is the absorbed dose, a least square fit was obtained from the measured peak-to-peak signals, $f(x_i)$. The least square fit was performed with the uncertainty of $f(x_i)$ given as

$\sigma_{f(x_i)} = \sqrt{\frac{1}{N-2}\sum_{i=1}^{N}(f(x_i) - B - Ax_i)^2}$. The uncertainty of the regression curve was calculated from $\sigma_{f(x)} = \sqrt{\left(\frac{df}{dA}\right)^2 \cdot \sigma_A{}^2 + \left(\frac{df}{dB}\right)^2 \sigma_B{}^2 + \left(\frac{df}{dx}\right)^2 \sigma_x{}^2 + 2 \cdot x \cdot cov(A,B)}$. The uncertainty of the dose σ_x was given by the ionization chamber measurement and σ_A, σ_B and $cov(A,B)$ were acquired from the least square fit function.

The dose of the unknown chewing gum was estimated by calculating the mean of the measured unknown dose peak-to-peak values and inserting it in the regression equation, $= \frac{f(x)-B}{A}$. The uncertainty of the unknown dose, σ_{x_u}, was then obtained by adding the equivalent uncertainties of the peak-to-peak value, $\frac{\sigma_{f(x_i=x_u)}}{A}$, and the regression uncertainty at the estimated dose, $\frac{\sigma_{f(x=x_u)}}{A}$, resulting in

[8] Dental V6 +strong teeth spearmint

$$\sigma_{x_u} = \sqrt{\left(\frac{\sigma_{f(x_i=x_u)}}{A}\right)^2 + \left(\frac{\sigma_{f(x=x_u)}}{A}\right)^2}.$$

Results and discussion

The spectra from the time dependence measurements are shown in figures 1 and 2. The spectra of V6 were found to be very similar to those of xylitol, which indicate that the content of xylitol in the chewing gum coating is large compared to the content of sorbitol. The xylitol spectrum was found to change considerably from 1 day after exposure to 84 days after exposure. At one day after exposure peaks, X_1 at 341.5 mT and X_2 at 343.6 are pronounced (see figure 1). The X_1 peak is decreasing after 1 day while the X_2 peak is slightly increasing with time up to 16 days, thereafter the peak is decreasing. A peak X_3 at 342.5 mT appears at 36 days and is increased till 84 days after irradiation. A negative peak X_4 at 345.1 mT was initially increasing until 16 days after exposure after which it decreased. This behavior is similar to the X_2 peak.

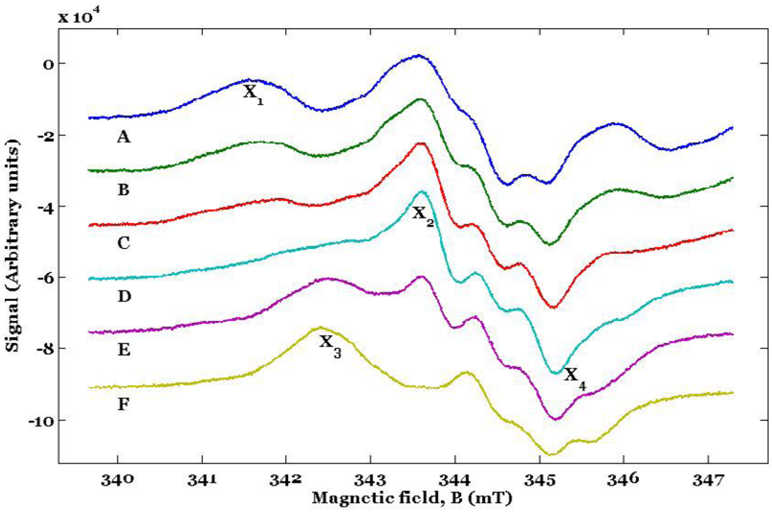

Figure 1. EPR spectra of xylitol irradiated to 50 Gy measured 'A' = 1, 'B' = 4, 'C' = 8, 'D' = 16, 'E' = 36, 'F' = 84 days after irradiation. Microwave power used was 1 mW. The peaks X_1 at 341.5 mT, X_2 at 343.6 mT, X_3 at 342.5 mT and X_4 at 345.1 mT were used for saturation measurements.

The sorbitol spectrum changed less than xylitol with time after irradiation (see figure 2). One peak, S1 at 342.6 mT was found rather stable through all measurements while S2 at 344.2 increased up to 36 days. A third negative peak S3 at 345.5 mT decreased gradually with time after exposure. These measurements indicate that S1 is the most stable signal and therefore best suited for dose response determinations.

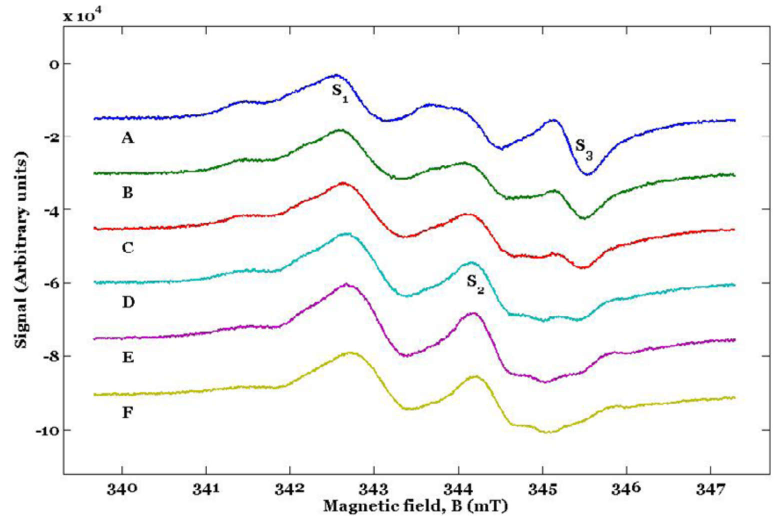

Figure 2. EPR spectra of sorbitol irradiated to 50 Gy measured 'A' = 1, 'B' = 4, 'C' = 8, 'D' = 16, 'E' = 36, 'F' = 84 days after irradiation. Microwave power used was 1 mW. The peaks S_1 at 342.6 mT, S_2 at 344.2 and S_3 at 345.5 mT were used for saturation measurements.

Power saturation measurement

The saturation curves were obtained from peaks in the EPR spectra at 10 different values of microwave power, see figure 3. The xylitol peaks were measured at magnetic field values X_1 at 341.5 mT, X_2 at 343.6 mT, X_3 at 342.5 and X_4 at 345.1 mT, see figure 1. The X_2 and X_4 peaks show similar behavior while the X_1 and X_3 peaks saturate at different power. The high magnitude of the X_2 and X_4 signals at high microwave powers suggest that these peaks might be most suitable for dose assessments.

The corresponding saturation curves for sorbitol showed that the signals of S_1 and S_3 were saturated at a microwave power of about 20 mW, while S_2 was saturated at about 3 mW.

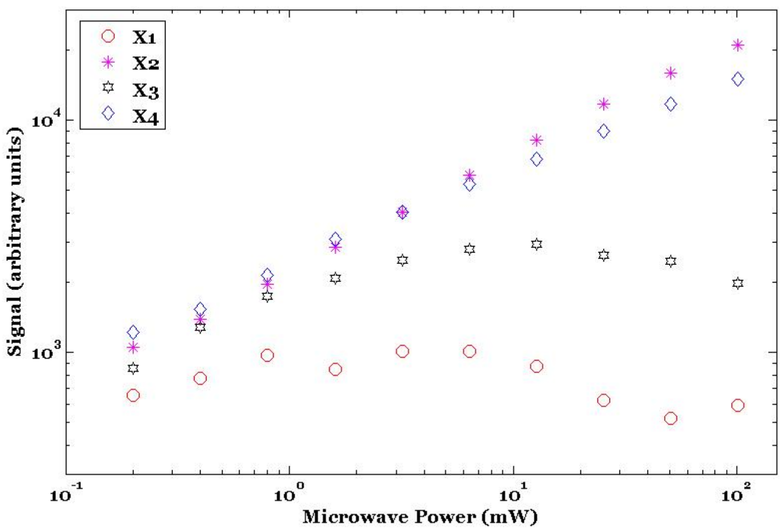

Figure 3. Saturation curves of xylitol at magnetic field values X_1 at 341.5 mT, X_2 at 342.5 mT, X_3 at 343.6 and X_4 at 345.1 mT.

Investigation of the variation between chewing gums

Peak to peak values were obtained from the EPR spectra at the magnetic field values of X_2 and X_4 and plotted for the background, single tablet measurement, one tablet measured 7 times and multi tablet measurement, 7 tablets measured one time. The peak-to-peak values of the single tablet show less scatter than those of the multi tablets. The scatter is about 3 times larger for the multi tablet measurement, which shows that the contribution of uncertainty from the tablet composition variation is large.

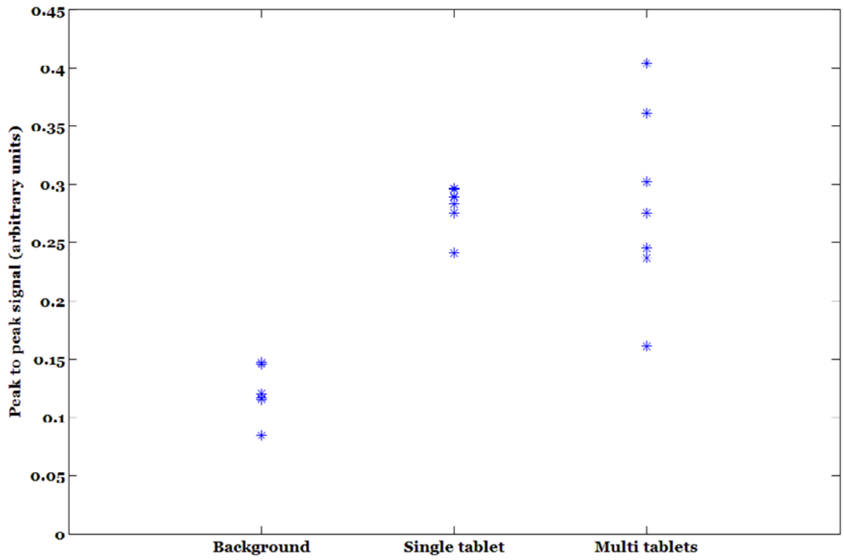

Figure 4. Peak-to-peak values obtained from EPR spectra of background and irradiated 1.8 Gy irradiated chewing gums. The background values correspond to two tablets measured 3 times each. The single tablet values correspond to one tablet measured 7 times and the multi tablets values correspond to seven tablets each measured once.

Dose response measurements

The dose response determinations of the chewing gum was performed for the peaks X_2 and X_4, see figure 1. The result is displayed in figure 5 showing the regression line with ± 1 Standard Deviation (SD) uncertainty indicated. To the right in figure 5 the dose determination of the unknown dose is inserted. The vertical error bar corresponds to one SD of the peak-to-peak signal of the chewing gum of unknown dose. Its corresponding uncertainty in dose is shown as the horizontal dotted line. This uncertainty was added to the uncertainty of the regression curve to obtain the combined uncertainty of the dose determination for the unknown dose. This dose was determined to be 3.76 ± 0.18 Gy (1 SD), which well agree with the given "unknown" dose of 3.56 Gy, indicated as a star in figure 5.

The sucrose dose response curve is showed in figure 6.

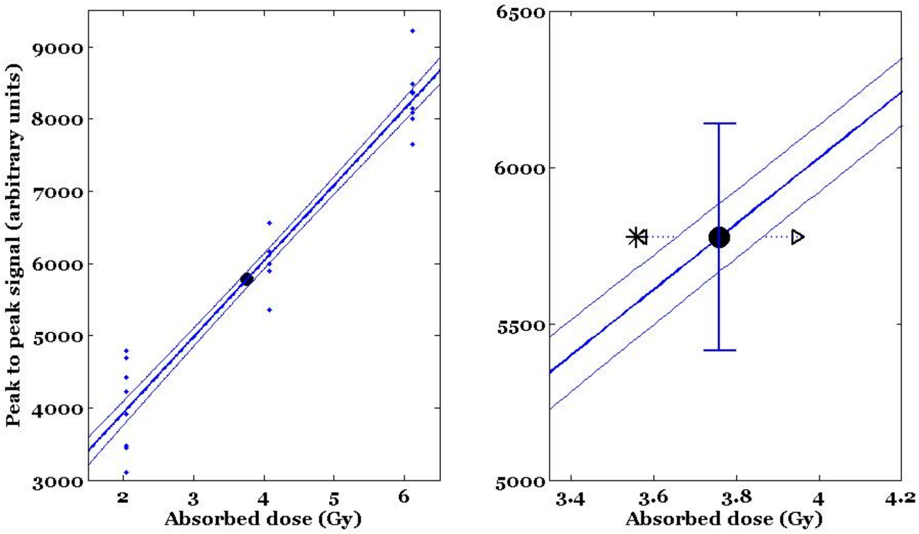

Figure 5. The figures show the dose response of V6 chewing gum. To the left the regression curve with the ± 1 SD uncertainty lines. The regression was obtained from two tablets irradiated with 2 Gy and measured 4 times, 2 tablets irradiated with 4 Gy measured 3 times and 2 tablets given 6 Gy measured 4 times. To the right the dose determination of the unknown dose is displayed. The vertical error bar corresponds to the uncertainty (± 1 SD) of the peak-to-peak signal points of the unknown dose chewing gum. Its corresponding uncertainty in dose is shown as the horizontal dotted line. The unknown dose was determined to be 3.76 ± 0.18 Gy (1 SD). The dose given was 3.56 Gy, indicated as a star.

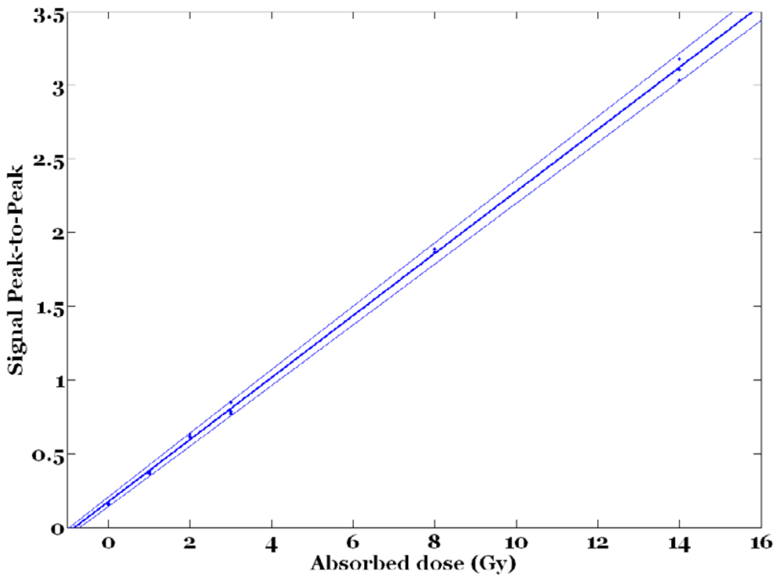

Figure 6. The dose response curve of sucrose. Signal peak-to-peak was measured in arbitrary units. The thinner lines indicate the limits of 2 standard deviations.

Conclusions

The EPR spectra of irradiated xylitol and sorbitol are analyzed regarding time evolution and microwave power. A chewing gum containing these sweeteners was analyzed regarding possibilities for retrospective dosimetry. An absorbed dose of 3.5 Gy given to a piece of chewing gum was possible to determine with an uncertainty of 5 %.

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Categorisation of sources: Is it only legal instrument for authorities, or also a practical tool for qualified experts and exposed workers?

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Abstract

The review of activity levels used in legislation is given and background discussed. We present basic presumptions, reference doses and scenarios used for calculation of *exemption level*, *clearance level*, *D-value*, A_1 *value*, A_2 *value* and level for *high-activity sources*. These levels are compared for some most common radionuclides.

Especially useful are A_1 *value* (for sealed sources), A_2 *value* (for unsealed sources) and *D-value* (for general sources). Practically in all incidents with sources below A_1 *value* or A_2 *value* it is highly improbable that the dose limits for professionals will be exceeded, if the persons involved are aware of source presence. If the persons are unaware of source presence, which could be the case when the source is uncontrolled, then the source with activity close to, or above *D-value* can be dangerous to all involved. The fact that the *D-value* is usually below or close to A_1 and A_2 *values* additionally supports the importance of source supervision and control.

It seems, as a rule, that in realistic circumstances and incidents with sources with activities below 1 GBq permanent injuries are not possible, and it is also very unlikely that dose limits for professionals would be reached. All sources with activities more than 1 TBq are very dangerous, with the potential to create permanent injuries, or even a person's death.

Introduction

During the licensing or registration process and also during the use of radioactive source all applicants, and later licensees or registrants, meet with some kind of the classification method when the source is categorised according to some predefined scheme. The purpose of this approach is to simplify legal decision process and to optimise requirements regarding required documentation and requests for safety measures and emergency preparedness.

Authorities use different categorisations for different purposes: the first and basic one is the decision whether some material should be considered radioactive source. For that purpose they use *exemption level*. And when some source should be released from regulatory control, the *clearance level* is used. For other categorisations *D-value* (for emergency preparedness requirements), A_1 and A_2 *values* (for transportation requirements) and *high-activity level* (for radiation safety and security requirements) are used.

For practically all radionuclides, all these levels are conveniently listed in relevant (but separate!) documents and available to all users. However, the real background of the values is not evident and clear to majority of people involved in authorisation process and use of sources. Therefore, we are going to review and describe scenarios that were used to develop and calculate the values that are used for categorisations and to compare them.

Since all values were calculated by experts and for realistic scenarios, they have also practical meaning and could be used as basis for quick and relevant estimates of potential exposure in different accidental situations.

Backgrounds of values used for categorisations, exemptions and clearance

The motivation for the introduction of categorisations, exemptions and clearance of sources originates in variety and quantity of practices where different radioactive sources with range of activities are used. The consequence is the quantity and complexity of requirements that should be implemented in legislation and supervision of sources. One of the basic principles of the system of radiation protection is optimisation, which also requires that the legal requirements and control should be commensurate with the potential magnitude and nature of the hazard. Similarly, the application of the requirements of legislation and practical radiation protection measures should be commensurate with the characteristics of the practice or source and with the magnitude and likelihood of the exposures (IAEA 1996).

Considering these requirements, the use of categorisations has positive influence on the following aspects of radiation safety:

- Regulatory measures: enables graded system for notification, registration, licensing and inspection, ensuring also optimisation of human and financial resources;
- Register of sources: provides ground for (optimised) decision which sources should be included and required level of details;
- Import/export controls: provides ground for (optimised) decision regarding the level of import/export controls and facilitates the communication between national authorities;
- Emergency preparedness and response: ensures that emergency preparedness and response to accidents are appropriate with the hazard posed by the source. It can also serve as a guide for prioritisation when control over multiple orphan sources must be regained;
- Security measures: enables graded basis for assisting in the choice of security measures;
- Communication with the public: provides basis for explaining relative hazards associated with events involving radioactive sources.

Use of exemption and clearance levels relies on similar foundations: besides establishing formal border between natural materials and radioactive materials (where exclusion does not apply), it also prevents unreasonable spending of resources and efforts on sources with insignificant radiation risk for workers or population.

Exemption levels

According to (IAEA 1996) and (EUC 1996), the basic criteria for the exemption of radioactive sources (and practices) are as follows:

a. The radiation risks to individuals caused by the exempted practice or source must be sufficiently low as to be of no regulatory concern;
b. The collective radiological impact of the exempted practice or source must be sufficiently low as not to warrant regulatory control under the prevailing circumstances; and
c. The exempted practices and sources must be inherently safe, with no appreciable likelihood of scenarios that could lead to a failure to meet the criteria in a. and b.

These criteria were elaborated in (IAEA 1988) and later in (EUC 1993) with conclusion, that source within practice may be exempted without further consideration if the following criteria are met in all feasible situations:

- The effective dose expected to be incurred by any member of the public due to the exempted practice or source is of the order of 10 μSv or less in a year, and
- Either the collective effective dose committed by one year of performance of the practice is no more than about 1 man·Sv or an assessment for the optimization of protection shows that exemption is the optimum option.

The value 10 μSv/year was chosen making an allowance for the possibility that more than one source can contribute to the exposure.

In the calculations for specific radionuclides effective dose value 10 μSv/year and equivalent dose 50 mSv/year to skin was used for normal working conditions and disposal, while for accidents, misuse and unexpected situations dose limit for members of public was used. Calculations were performed for 299 radionuclides, starting from six physical forms considered to cover existing range of use: gas/vapour, liquid/solution, dispersible solid (e.g. powder), non-dispersible solid, thin film/foil, and sealed source/capsule. All three exposure pathways were considered and total dose calculated summing across pathways. The most limiting value from all calculations was chosen for the exemption level. For activity concentrations, the source size of 1 m^3 was taken into account.

The rounded results of these calculations are summarised in tables with exemption levels (exempt activity concentrations and exempt activities of radionuclides) in (IAEA 1996) and (EUC 1996) and are used internationally.

Clearance levels

The EC Directive (EUC 1996) does not define clearance levels, except for provision that clearance levels shall follow the basic criteria used for exemption as stated in Annex I of the Directive and shall take into account any other technical guidance provided by the EU. The problem of clearance levels is more complex than the problem of exemption levels since it is not possible to satisfactory predict and describe all specific circumstances and destiny of radioactive material released from the control.

A specific approach is required e.g. for buildings, rubble and metals originating from dismantling nuclear installations. The European Commission has prepared and published special recommendations regarding these subjects (EUC 1988, EUC 1998, EUC 2000A and EUC 2000B). One of these documents, namely (EUC 1988), has been published at the end of the eighties, and was used as a basis for establishing basic criteria for exemptions, as we have already mentioned in the previous subchapter.

Ten years ago, EC introduced general clearance levels to cover any possible application (EUC 2000D, EUC 2001), without special restrictions on the origin and type of material to be cleared. These general clearance levels apply to any solid, dry material, also to natural radiation sources. Basic criteria for clearance are based on requirements of the EC Directive, as we have discussed previously in relation to exemption levels.

In scenarios used for calculation of general clearance levels, the following exposure pathways to a member of the public were considered: inhalation, ingestion, external gamma radiation and beta skin irradiation. In each case the most restrictive of the scenarios was adopted and the mass activity concentration resulting in dose 10 μSv/year was used to define the radionuclide specific clearance level. To clarify these calculations additionally, we will enlist some of used scenarios: inhalation of contaminated dust on the workplace during the whole year, inhalation of dust during a whole year by an infant, ingestion of material on the workplace (hand-to-mouth pathway, 20 g/year intake), a child playing on the soil with undiluted material (100 g/year intake), landfill worker who is working full time on the waste, person living in the house for which cleared building rubble has been used in construction, etc.

Ingestion via water pathways and vegetable consumption was also considered in clearance calculations. Rounded general clearance levels are listed in Table 1 in (EUC 2000D).

A_1 and A_2 values for transport

The risk associated with the transport of radioactive materials is specific and relates to manipulation and transfer of packages in conditions that are not controllable. Therefore, it is not possible to anticipate possible smaller or larger incidents, or even accidents. Even more, we can say that incidents and minor mishaps represent part of normal transport.

Type A packages are intended to provide economical transport for large numbers of small activity consignments, while at the same time achieving a high level of safety. They are tested for routine (incident free) transport conditions and normal transport conditions (with minor mishaps, like falling from vehicles, being dropped during manual handling, being exposed to the weather, being struck by a sharp object, or having other packages or cargo stacked on top). However, it is not possible to expect that the package will retain integrity in accident. In that case, we have to expect that integrity of built-in shielding or integrity of containment will be degraded. As a result, exposure of workers and/or public to the source is highly probable. The only possible solution for limiting the risk to workers or members of the public in that case is to limit the content (activity) of the package.

A_1 and A_2 *values* are used as limiting values for Type A package contents. These values were calculated with realistic scenarios for special form radioactive material (either an indispersible solid radioactive material or a sealed capsule containing radioactive material; integrity of both must be preserved under all applicable conditions, even an accident) and non-special form material. The Type A package contents limits are:

- A_1 – for special form material, and
- A2 – for non- special form material.

The contents limits are set so as to ensure that the radiological consequences of severe damage to a package are acceptable. Although the dose limits do not apply to potential exposures, it was decided (as described in (IAEA 2002)) that the dose limits for professionals for effective, committed and equivalent dose shall be used. The assumptions of potential exposure following the transport accident were as follows:

a. The effective or committed effective dose to a person exposed in the vicinity of a transport package should not exceed a reference dose of 50 mSv.
b. The dose or committed equivalent dose received by individual organs, including the skin, of a person involved in the accident should not exceed 0,5 Sv, or in the special case of the lens of the eye 0,15 Sv.
c. A person is unlikely to remain at 1 m from the damaged package for more than 30 min.

Scenarios used in calculations considered the following exposures and exposure pathways:

- external dose due to photons (0,5 h exposure at 1 m from the source),
- external dose due to beta emitters (0,5 h exposure at 1 m from the source),
- internal dose via inhalation (committed dose from 10-6 fraction of package content)
- skin contamination and ingestion doses (5 h exposure from beta contamination with 10-3 fraction of package content per m2 of skin)
- submersion dose due to gaseous isotopes (0,5 h exposure)

Results of calculations can be found in (IAEA 2002) together with derived A_1 and A_2 *values* for different radionuclides. These values are used internationally (IAEA 2005A) as a part of legislation for the transport of dangerous goods

D-values

The idea behind the introduction of *D-values* (D for "dangerous source") is to provide the tool for risk based ranking of different sealed and unsealed radioactive sources. Sources with high activity, if not managed safely and securely, can cause severe deterministic effects to individuals in a short period, whereas low activity sources are unlikely to cause exposures with harmful consequences. The *D-value* is that quantity of radioactive material, which, if uncontrolled, could result in the death of an exposed individual or a permanent injury that decreases that person's quality of life (IAEA 2006). In the calculation of *D-values*, the dose criteria listed in Table 1 were used (IAEA 2005B).

Table 1. Reference doses for D-values (from IAEA 2005B).

Tissue	Dose criteria
Bone marrow:	1 Gy in 2 days
Lung:	6 Gy in 2 days from low LET radiation 25 Gy in 1 year from high LET radiation
Thyroid	5 Gy in 2 days
Skin/tissue (contact)	25 Gy at depth of 2 cm for most parts of the body (e.g., from a source in a pocket), or 1 cm for the hand for a period of 10 hours
Bone marrow:	1 Gy in 100 hours for a source that is too big to be carried

In possible scenarios different realistic accident situations were considered and also dispersion situations that may be relevant to malevolent acts. It was found (IAEA 2006, IAEA 2003) that the following realistic scenarios for sealed source and dispersal of source are the most limiting:

a. An unshielded source being carried in the hand for one hour or in a pocket for 10 hours, or being in a room for days to weeks
b. Dispersal of a source, for example by a fire, explosion or human action, resulting in exposure due to inhalation, ingestion and/or skin contamination.

The results of calculations for each radionuclide were two values: D_1 for sealed source and D_2 for dispersal of source. Values are listed in Table 1 in (IAEA 2006) but for regulatory use, only one (lower!) value was designated as *D-value* for that particular radionuclide.

D-values for particular radionuclides serve for ranking the radioactive sources and practices in five risk categories, according to involved source activity (IAEA 2005B). In Table 2 risk categories of sources, corresponding A/D ratios and brief descriptions of associated risk are listed.

Table 2. Risk categories with corresponding A/D ratios and risk description for radioactive sources (from IAEA 2005B).

Category	Activity Ratio (A/D)	Risk Description
1	A/D ≥ 1000	Extremely dangerous to the person
2	1000 > A/D ≥ 10	Very dangerous to the person
3	10 > A/D ≥ 1	Dangerous to the person
4	1 > A/D ≥ 0.01	Unlikely to be dangerous to the person
5	0.01 > A/D ≥ Exempt. Lev./D	Most unlikely to be dangerous to the person

High-activity value

The legal term high-activity sealed source was introduced in the HASS Directive (EUC 2003) issued by European Commission as a result of a recommendation from the Working Group on Control of Sources, which was established by EURATOM Treaty Article 31 Group of experts. The aim was to improve the safety and security of sources, prevent orphan sources from arising and to harmonise requirements between EU countries without imposing unnecessary additional burden to holders of small sources.

The Directive introduced special requirements regarding authorisation, transfers, records, common requirements for holders, identification and marking, training and information, special arrangements for dealing with orphan sources, request for financial security and international cooperation and information exchange.

Technically, the high activity source is a sealed source that produces a dose exceeding 1 mSv/h at the distance of one meter. In the HASS Directive, the *high-activity values* for the most important radionuclides are listed, and for other (non-listed) the instruction to use 0.01 fraction of A_1 *value* from transport legislation (IAEA 2005A) is given.

Comparison of values

In Table 3 Exemption level, A_1, A_2, *D-value* and *high-activity level* are listed for some more common radionuclides. The data of Table 3 are also presented on Figure 1, but on the chart the order is changed to illustrate the relations between particular values.

Table 3. Different activity levels for some more common radionuclides.

Radionuclide	Exemption level (Bq)	A_1 (Bq)	A_2 (Bq)	D (Bq)	High-activity level (Bq)
Co-60	$1\times10^{+5}$	$4\times10^{+11}$	$4\times10^{+11}$	$3\times10^{+10}$	$4\times10^{+9}$
Ni-63	$1\times10^{+8}$	$4\times10^{+13}$	$3\times10^{+13}$	$6\times10^{+13}$	$4\times10^{+11}$
Sr-90 (Y-90)	$1\times10^{+4}$	$3\times10^{+11}$	$3\times10^{+11}$	$1\times10^{+12}$	$3\times10^{+9}$
Mo-99	$1\times10^{+6}$	$1\times10^{+12}$	$6\times10^{+11}$	$3\times10^{+11}$	$1\times10^{+10}$
Tc-99m	$1\times10^{+7}$	$1\times10^{+13}$	$4\times10^{+12}$	$7\times10^{+11}$	$1\times10^{+11}$
I-131	$1\times10^{+6}$	$3\times10^{+12}$	$7\times10^{+11}$	$2\times10^{+11}$	$3\times10^{+10}$
Cs-137	$1\times10^{+4}$	$2\times10^{+12}$	$6\times10^{+11}$	$1\times10^{+11}$	$2\times10^{+10}$
Ir-192	$1\times10^{+4}$	$1\times10^{+12}$	$6\times10^{+11}$	$8\times10^{+10}$	$1\times10^{+10}$
Ra-226	$1\times10^{+4}$	$2\times10^{+11}$	$3\times10^{+9}$	$4\times10^{+10}$	$2\times10^{+9}$
Am-241	$1\times10^{+4}$	$1\times10^{+13}$	$1\times10^{+9}$	$6\times10^{+10}$	$1\times10^{+11}$
Cf-252	$1\times10^{+4}$	$1\times10^{+11}$	$3\times10^{+9}$	$2\times10^{+10}$	$5\times10^{+8}$

As can be seen on Figure 1, the prevailing order is: *exemption level*, *high-activity level*, *D-value*, A_2 *value* and A_1 *value*. This is not completely true for pure beta sources (the highest is *D-value*, but very close to A_1 and A_2) and heavy radionuclides (where A_2 is low, due to their radiotoxicity,).

Except for heavy radionuclides, A_1 and A_2 *values* are quite close. We can also see that *D-value*, A_1 and A_2 values are close for all radionuclides, except for heavy radionuclides, again. For these, the *D-value* is approximately the geometrical mean of values A_1 and A_2.

For all listed radionuclides (except for Am-241) the *high-activity level* is lower than other levels. It seems that the value is really well chosen and could be used not only for sealed sources, but also more generally.

For majority of sources practically all levels are between 10^{10} Bq and 10^{12} Bq. For heavy radionuclides these values are between 10^{9} Bq and 10^{11} Bq. It seems that we can use, as a rule of thumb, that with any source below 10^{9} Bq we cannot expect permanent injury in realistic scenarios. We can also conclude that there is just a small probability that the dose limits for professionals would be achieved. On the other side, sources with an activity over 10^{12} Bq are always extremely dangerous for all radionuclides.

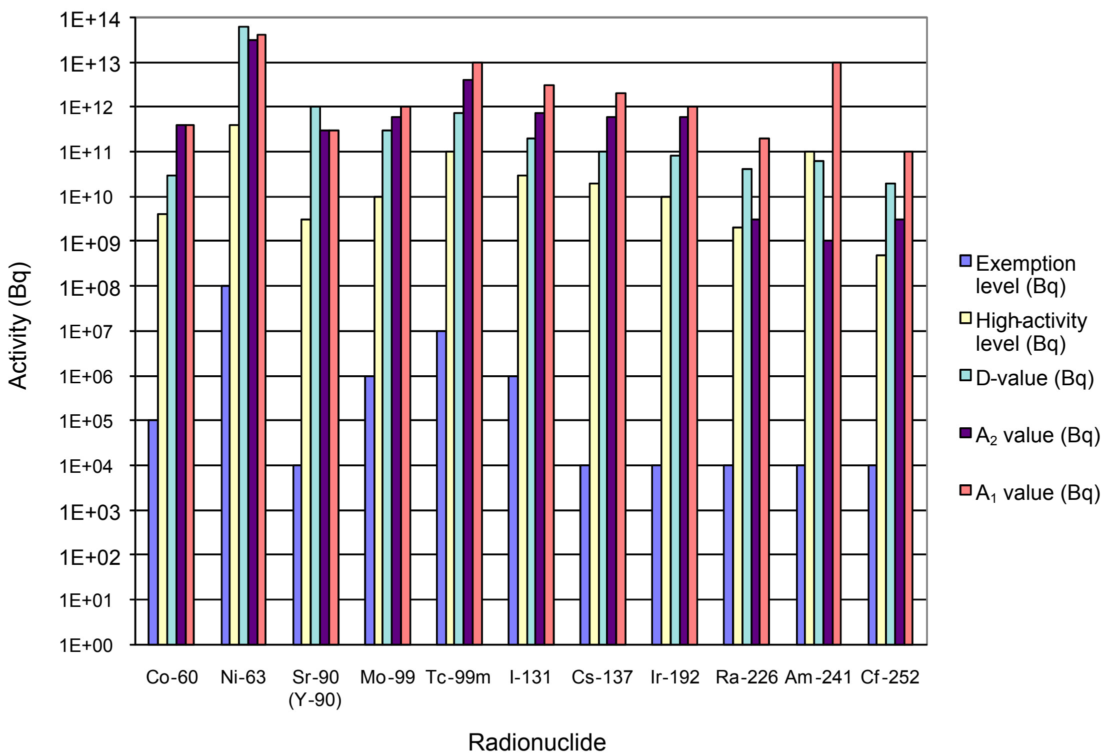

Figure 1. Activity values for some common radionuclides in prevailing order.

Conclusions

We have seen that different activity levels used in the legislation have also a practical meaning: the calculation was based on realistic foundations and scenarios. We can use them as an orientation regarding potential exposures and for rough assessment of dose in possible incidents or accidents.

It seems that especially useful are *A_1 value* (for sealed sources), *A_2 value* (for unsealed sources) and *D-value* (for general sources). We can say that when the persons involved in incident are aware of the source presence (like exposed workers or emergency personnel) and when the activities are below A_1 or A_2 value (as appropriate), it is highly improbable that dose limits for occupationally exposed workers will be exceeded. When the source is not under control and persons involved are unaware of it, and activities are close to or above *D-value*, then the source could be dangerous to involved persons.

The *high-activity value* could be used for simple and straightforward calculation of dose rate from the sealed source.

For the majority of examined radionuclides values are increasing in the following order: *high-activity level*, *D-value*, A_2 and A_1 *value*, while the e*xemption level* is at least four orders of magnitude below these values. The fact that *D-value* (when the source is considered dangerous) is usually below or close to A_1/A_2 *values* additionally supports the importance of supervision and control of sources.

Except the *exemption level*, the values of other levels for almost all common radionuclides are higher than 1 GBq. Therefore, it seems that in realistic circumstances and accidents it is very unlikely to expect permanent injury, or even achieving the dose limits for professionals for sources below 1 GBq. Sources with activity over 1 TBq are always very dangerous with the potential to create permanent injuries, or even a person's death.

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Emergency response to accidents involving radioactive material: Italian fire fighter experience

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Abstract

Since 1987 Italy has not been producing electricity through nuclear fission; nevertheless, due to the presence of four Nuclear Power Plants undergoing decommissioning and to the increasing use of radioactive sources in medical, industrial and research applications, potential accidents involving radioactive materials can still occur (and actually took place). Italian Fire Brigades, which are a national Organization, represent one of the main government structures involved in this kind of events, as far as the initial phase in concerned. Actually, Italian Fire Brigades teams act as first responders for any kind of calamity.

Accidents involving radioactive substances may be accompanied by hazards of different nature, and in these cases the wide range of tasks Italian fire fighters are used to face and their high level of training greatly simplify the approach to intervention.

During the last five years some small accidents took place in Italy involving radioactive materials such as orphan sources; in all cases the action of the first responders has proved of fundamental importance. Italian Fire Brigades have special trained teams equipped with suitable devices and dresses for monitoring and protection, and they are able to manage emergencies involving radioactive materials.

The aim of this article is to show the Italian response system provided by the Italian fire brigades from the very beginning of the accident until the post-emergency phase: some examples will be shown concerning recent events. Moreover the fire fighter's way of training, as far as radiation protection is concerned, will be presented.

The authors will illustrate also the co-operation between different organizations and services involved when a relevant accident occurs.

Introduction

A valid response to the emergency starts with an efficient organization in which the formation and maintenance of operational capabilities are constantly pursued goals.

Below, the position of C.N.VV.F. about it is shown and it will be shown how, through that organization, it has been possible to deal effectively with three types of radiological emergencies completely different from each other.

Material and methods

The organization of C.N.VV.F. in radiological emergencies

Control of radiation hazard is one of the institutional duties of C.N.VV.F. that, as a national technical organization, gets involved from the very beginning of the emergency. The response provided by C.N.VV.F. is based on a pyramidal structure in which at the bottom each Italian province is able to ensure first intervention just with limited equipment and training.

The intermediate level is represented by 22 special NR (Nuclear and Radiological) intervention teams, that are able to make more detailed assessments and measurements.

At the top of the pyramid there is the personnel of a specific laboratory in Rome with the most advanced equipment and training in Italy (Fig. 1).

Fig 1. Dislocation of the teams in Italy.

Basic team

103 teams, one for each provincial fire brigade headquarter, are able to respond within an hour from emergency. They may assess mainly the dynamics of the accident, getting useful technical information and performing instrumental measurements limited to gamma exposure (effective dose and effective dose rate) and to the presence of surface and air contamination (counts per second relative to the environmental level). The

instrumentation and preparation do not allow to quantify the level of risk but only to asses and monitor an anomalous presence of radiation. Self protection is the fundamental skill for these teams, through safely dressing and sampling.

Special NR team

22 teams are distributed throughout national territory, roughly one in each region. They are able to respond within 3-4 hours when alerted. The members of these teams are instructed in a theoretical-practical course three weeks long and then they are called annually for the retraining. In addition to the basic teams, the special teams use instruments to perform qualitative analysis on gamma-emitting radioisotopes and neutron dose measurements; moreover they sample environmental matrices, manage the containment and confinement of contaminated objects, and provide to the staff monitoring and decontamination.

Central NR Laboratory

A central laboratory is located in Rome and it is the apical response of C.N.VV.F. when a radiological emergency occurs. Personnel in the laboratory carries out dosimetric control activities of staff, qualitative and quantitative measurements of alpha, beta and gamma spectrometry, dose evaluation to personnel and operational planning to perform in accordance with Italian and international rules, together with other organizations relevant in post-emergency phase. The laboratory personnel manage the training of all the teams, the verification and calibration of instruments used by the teams and coordinate the staff in case of national or interprovincial emergencies.

Results

Although in Italy there are no more working nuclear power plants, increasing industrial, medical or research activity, use of radioisotopes is a source of significant risk.

The accidental scenarios that can be faced with the organizational structure of C.N.VV.F. include localized or diffuse emergencies in terms of released contamination, with restoration of normal conditions in the short or long term and emergencies requiring the employment of several special teams.

As an example three different situations that occurred in Italy over the past two years will be considered. These emergencies tested the system response confirming the effectiveness of the organization.

Discussion

The first case is the accidental melting of a source of ^{137}Cs in a brass refinery in Brescia.

The second concerns the presence of radioactive sources in the waste during the maxi emergency in Campania mainly in Naples.

The third arose by pellets with high concentration of ^{137}Cs coming from Eastern Europe that have been used for domestic stoves.

Accidental melting of a radioactive source

Too often sealed radioactive sources are hidden in metal scraps going to melting plants. If these sources are not detected they can be melted in the ovens and spread in the metallic ingots or dust.

In November 2008 a ^{137}Cs source was melted in a metal refinery working brass; ingots and bags of dust that were accidentally produced and sent to Germany for the recovery of residues of brass.

The alarm was triggered after several days, when the German authorities blocked some of these bags at the boundary because they were exceeding the threshold of radioactivity. They immediately informed the Italian authorities about it.

Fig 2. Brescia. Brass dust sampling for measurement of gamma spectrometry.

In that occasion all 3 levels of the fire brigades organization were involved. Actually after the first intervention performed by the basic team, the central Laboratory took over: its experts planned a thorough activity that went on for two months with a strong contribution of the special NR teams.

In summary, under the coordination of the central area three main activities were carried out:

- Boundaries and zoning of industrial area and access control
- Monitoring of remediation of contaminated plant components
- Measurements on samples of dust and ingots in order to authorize the release of the store and the resumption of the activity (Fig. 2)

This emergency was confined, since it involved just the area of the refinery and material contained within bags. Nevertheless, the duration of operations made it necessary to involve three special NR teams working alternated.

Emergency Campania radioactive waste

In June 2008 the region was the scene of a maxi emergency for the presence of large amount of waste in the streets and the lack of suitable locations to store it.

It was decided to ship the waste towards burning plants in Germany until a storage place was identified. As the previously shown emergency, German authorities detected some radioactive substances in the refuse. Then, a strict radiological monitoring of leaving waste was requested. This monitoring activity was managed by C.N.VV.F.

Fig 3. Naples. Measurements of dose rate on the field

In this case, with the central laboratory management, activity took about five months and involved many special NR teams coming from all over Italy as well as local basic teams that were working mainly to make measurements (Fig. 3).

Several radioactive sources were found and isolated, almost entirely of hospital origin.

Radioactive pellets emergency

In June 2009, widespread in north-central Italy, unusual concentration of ^{137}Cs were found in ashes coming from the combustion of certain types of pellets. The specific activity values were so high that, according to Italian law, ashes could have been included into restrictions for detention.

In order to contain the emergency some type of pellets supplies not yet sold were seized and a monitoring activity was scheduled. A collection of ashes in the houses of people who had requested help calling the emergency phone number was possible.

Although it was already clear that the concentrations did not show a serious radiation hazard, alarm and concern of citizens were such as to require a real response from the authorities.

In particular, using the widespread presence of local Fire Departments, tens of thousands of visits were conducted from the basic teams. When measurement's results were showing to pass a threshold of five times the value of the environmental background in total measures alpha , beta and gamma, samples of ashes were taken and related unburned pellets to evaluate later by gamma spectrometry.

The samples were brought at the nearest special NR teams building to carry out gamma spectrometry in order to estimate a presence of ^{137}Cs at least twice the environmental background.

Fig 4. Rome. Radioactive pellets samples.

Following this screening, positive samples of ashes and related unburned pellets were sent to the central NR laboratory for quantitative analysis (Fig. 4).

Sometimes, concentrations of ^{137}Cs up to 80 *kBq/kg* were found in the ashes while concentrations of about 300 *Bq/kg* were found in the unburned pellets.

Conclusions

Currently in Italy, as far as radiological emergencies is concerned, C.N.VV.F. is the organization most directly involved, certainly in the initial phase, due to widespread presence, but also in later stages. The C.N.VV.F. can provide technical experts who are able to plan operations in the medium and long term on the basis of assessments and measurements that could be conducted in complete autonomy from trained personnel for this purpose.

The current organization, together with relationships with technical personnel from other relevant authorities in radiation protection allows to maintain a sufficient team's level of training and to increase the professionalism of radio protection area.

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Analyses of causes and consequences of internal contamination incidents at NRG

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Abstract

In the last years several unforeseen events happened leading to internally contaminated workers and members of the public. Beside committed effective dose estimations incident investigations were performed into the direct and underlying causes. Using the Tripod method Basic Risk Factors were identified and used for prioritizing actions to prevent occurrence of these incidents in the future. Besides poor design characteristics, human failures could be traced back to organisational shortcomings as procedures, education and training and communication. An important finding was the personal perception of an internal contamination leading to concern and anxiety by those exposed. Justifying investment in terms of means and human capital according to the ALARA principle in relation to the gain in terms of dose reduction this fear perception is not taking into account. This suggest that a kind a “psychological” weighting factor must be introduced correcting for the enhanced ‘dose’ perception of internal contamination and in this way justifying extra effort preventing internal contamination.

Investigating microphone noise in mobile gamma-spectrometric HPGe measurements using accelometers

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Abstract

Microphonic noise caused by vibrations can be a problem in mobile gamma-spectrometric measurements using HPGe detectors. When driving on uneven roads moderate speeds might give raise to a considerable amount of vibrational stress. The resulting distorted spectra can be difficult to interpret, and potentially important spectral information may be lost. However, accelerometers can be used to detect and quantify the vibrations. These measurements can then be used to reduce the vibrational stress on the detector system.

To demonstrate the impact of noise from detector microphonics, vibrational data was gathered along with spectral measurements from an experiment with a car-borne mobile platform. The acquired accelerometer data could then be connected to the spectrum where distortions were observed. Speed and position on the road was also recorded and stored in the same record. This technique makes it possible to evaluate and compare different detector mounting configurations.

As this article shows, a sane mounting configuration will reduce the microphonic noise. This could be crucial for example in an orphan source emergency situation, where measurements must be gathered with speed and accuracy. Using accelerometers can be one way of achieving a reliable car-borne HPGe system by reducing the interference from microphonic noise.

Introduction

Vibrations causing small movements of the components in a Germanium detector's mounting can alter the capacitance between the field effect transistor (FET) gate and the detector bias supply. This may change the noise characteristics of the detector and in turn give raise to an electrical signal. Even a small change (10^{-7} pF) in the capacitance can cause an electrical signal equivalent of a few keV (Gilmore 2008). This mechanically generated noise is referred to as microphonic noise. Its main sources during normal operation conditions are environmental acoustic noise, vibrations of the surrounding equipment and events connected to liquid nitrogen such as turbulence and bubbling in the dewar affecting the cryostat (Morales et. al 1992).

Although this should be a common problem in mobile gamma spectrometry using HPGe detectors, where, by necessity, some vibrations occur, most of the literature found on the topic of microphonic noise deals with low background experiments (Baudis et al. 1998; Morales et al. 1992) or electrically cooled semiconductor systems (Upp et al. 2005). Ways of discriminating microphonic pulses include digital signal processing (Keyser et al. 2008) and statistical rejection techniques (Morales et al. 1992).

Possible explanations for the lack of applied studies are;

- NaI(Tl) detector systems are more frequently used in mobile gamma spectrometry
- The microphonic noise mainly affects low energy region (<100 keV)
- The ambient background is not controlled, hence disturbances might be overlooked

This article shows the importance of a vibration dampening mounting configuration when using a HPGe detector in mobile measurements and how accelerometers can be used to optimize the same configuration. Digital signal processing and statistical rejection techniques are out of the scope of this work since they do not deal with the source of the problem, but rather its consequences.

Material and methods

Spectrometry system

In the experiments a ruggedized, coaxial, P-type High-Purity Germanium (HPGe) detector[1], model no. GEM 100-S, with 123 % relative efficiency from Ortec[2] was mounted in a GMC van at a height of approximately 1 m above ground. The detectors cylinder axis was horizontally orientated with the end-cap facing the rear of the van. A digital, portable, Multi-Channel Analyser (MCA, Ortec Digidart), with the conversion gain set to 2048 channels was used. The MCA communicated with a laptop-PC over the USB (Universal Serial Bus) interface, sending a new pulse height distribution when requested. Measurements in the region below 40 keV using a P-type HPGe detector is normally pointless due to the thick dead-layer. But these noise pulses could potentially have effect on critical things higher up in the spectra, leading to peak-broadening.

A linear 3-axis accelerometer from ST Microelectronics (LIS3LV02DL) was mounted on the detectors aluminium cap to measure vibrations. The sensors can measure accelerations up to ± 6 g in three orthogonal directions, with a specified resolution of about 4 mg at 610 Hz. Acceleration directions are henceforth referred to as X, Y and Z, where -X is an acceleration in the driving direction, +Y to the van's right when driving and +Z down towards the ground.

To scrutinize the influence of the detector mounting configuration, two different sets of dampeners were used. Fist the ‘bare-bone’ configuration, where the detector was mounted directly on the floor of the car. Second the ‘dampener’ configuration, where the detector was mounted in a plastic tube, isolated with approximately 10 cm of polyester foam.

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1 s/n p41629A

2 Ortec, 801 S. Illinois Ave., Oak Ridge, TN, USA.

Experiment

By observing spectra while driving on an uneven road, conclusions about different mounting configurations and their effect on the microphonic noise level can be drawn. In this work, a straight stretch about 400 m long was driven repeatedly while sampling spectral and vibrational data. To estimate the radiation background, 22 stationary 60 s background spectra was first measured at ~ 20 m intervals along the road. The straight stretch was then driven at three different velocities; 10, 20 and 30 km h^{-1} to gradually increase the mechanical stress on the detector. Each speed was driven four times, giving a total of 12 measurement sets. Spectrometer, accelerometer and GPS data were collected at 1 Hz using in-house software written in C#. This procedure was repeated for both mounting configurations.

Statistical analysis

To separate the microphonic noise from the ambient radiation background a region of interest **R** was chosen, ranging from the cut-off energy at 10 keV to 100 keV. The distribution of the number of gross counts per second n_b within **R** will be approximately normal with mean m_0 and standard deviation $\sigma = \sqrt{m_0}$, if measured repeatedly. Thus an observation *x,* which is significantly greater than m_0, would be an indication that microphonic noise was detected in a spectrum.

Using the null hypothesis, H_0, "no microphonic noise detected" and the research hypothesis, H_1, "microphonic noise detected" the test statistic

$$u = \frac{\bar{x} - m_0}{D} \tag{1}$$

where $D = \frac{\sigma}{\sqrt{n}}$ was calculated for each set $x_1, x_2, \dots, x_n$ of measured spectra. Under H_0, u will be observations from N(0,1). The null hypothesis can thus be rejected on the 1-α level if $u \geq \lambda_\alpha$. The single sided critical regions for u are presented in Table 1.

Table 1. Significance levels for the test statistic *u*.

Significance	*	**	***
α	0.05	0.01	0.001
u	1.64	2.33	3.09

The sample size of the test group, n, was chosen to be 3 spectra to avoid the worst fluctuations and the level of significance, α, was set to 0.001. The limit for rejection of H_0, then becomes

$$u_l = \sqrt{3}\,\frac{\bar{x} - m_0}{\sigma} \geq 3.09 \tag{2}$$

Results and Discussion

Figs. 1-2 show the results from the hypothesis tests using the 'bare-bone' and 'dampener' configurations. The results summarized in Table 2 show that the number of rejected samples increases with increasing driving speed, which was expected. At the slowest speed (10 km h^{-1}), the statistical tests showed no evidence of microphonic noise being detected. By increasing the driving speed to 20 km h^{-1} about 19 % of the measurements showed a significant number of microphonic noise pulses in the low energy region, **R**, when using the 'bare-bone' configuration. The number of rejected samples increased to 79 % when driving at the fastest speed (30 km h^{-1}) using the same configuration. By mounting the detector in the 'dampener' configuration, these numbers sank drastically. The rejection fraction, also shown in Table 2, was 2 and 19 % for 20 and 30 km h^{-1} respectively. To summarize, the total rejection fraction, i.e. the total fraction of measurements showing an significant increase of the count rate in **R** due to microphonic noise were 20 and 4 % for the 'bare-bone' and 'dampener' configurations respectively.

Table 2. Summary of hypothesis tests.

Test group	Samples	Rejection of H_0	Rejection of H_0 (%)
bare-bone u_{10}	391	1	0.2
bare-bone u_{20}	213	40	18.8
bare-bone u_{30}	138	109	79.0
dampener u_{10}	392	0	0.0
dampener u_{20}	219	4	1.8
dampener u_{30}	146	28	19.2
bare-bone total	742	150	20.2
dampener total	757	32	4.2

Two samples with extreme test statistic values from each mounting configuration, thus containing significant levels of microphonic noise were analysed in detail. The samples, labelled A, B, C and D are marked in Figs. 1-2. Acceleration data for these four samples are presented in Figs. 3-4 and spectra in Figs. 6-7.

The accelerometer data in Figs. 3-4 reveals both vibrations and more shock-like events. The detector is subject to forces mainly in the Z (up, down) and Y (left, right) directions, while the data show more moderate vibrations along the detector's cylinder axis (X). An offset of about 1 g is present in the Z-direction due to gravity. The frequency of the vibrations seems to be generally higher in the samples A and B (bare-bone) than in C and D (dampener) while the amplitudes show a reversed relationship. This clearly demonstrates the complexity of the microphonic noise problem. By introducing a dampening material much of the microphonic noise disappears. As shown in Table 2, but some measurements with large (several g's), shock-like, accelerations still appear because of the uneven road and high velocities.

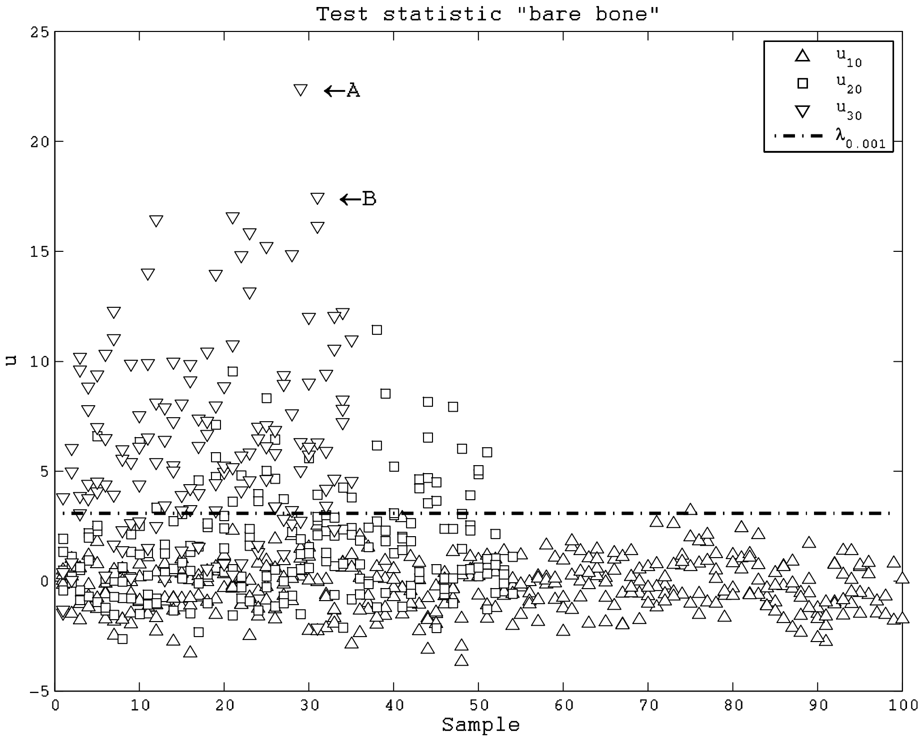

Fig. 1. Test statistic *u* for the 'bare-bone' configuration. The limit for rejection of H_0, λ_α, is marked with a dash-dotted line.

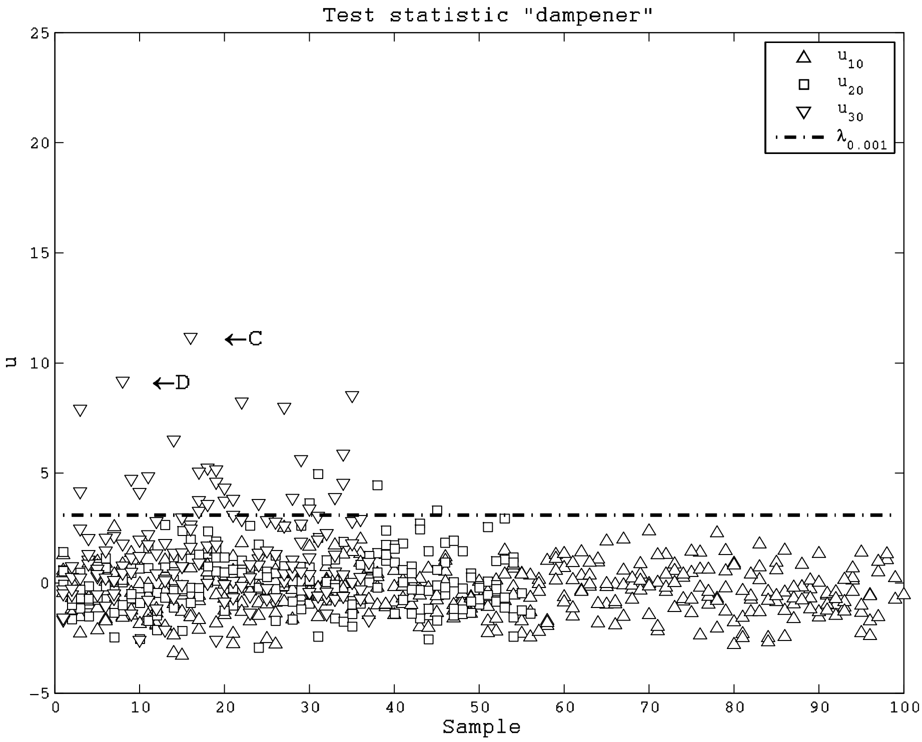

Fig. 2. Test statistic *u* for the 'dampener' configuration. The limit for rejection of H_0, λ_α, is marked with a dash-dotted line.

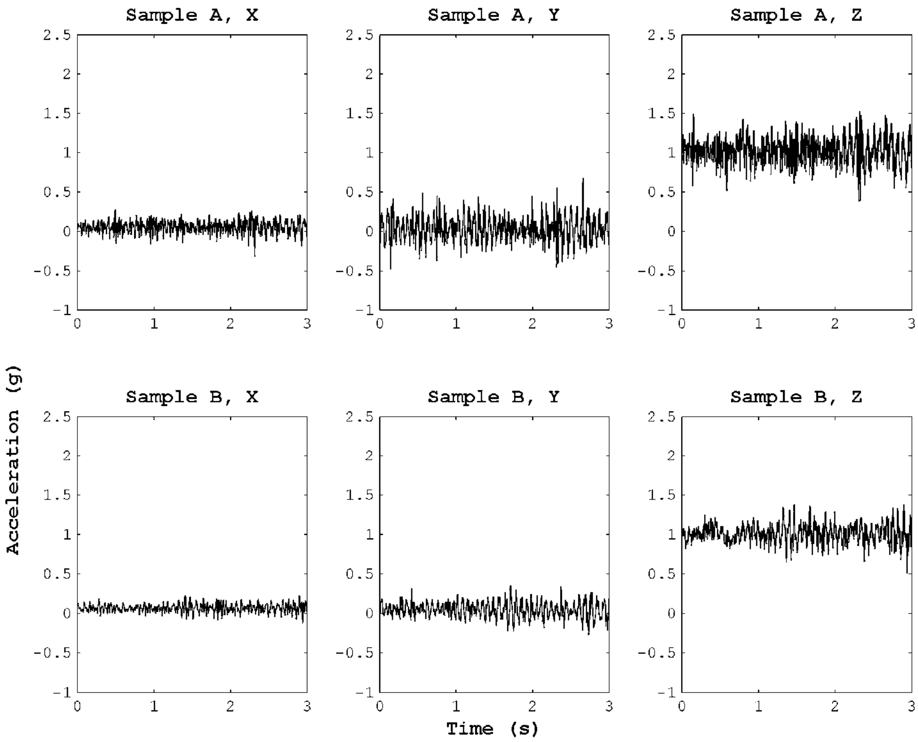

Fig. 3. Accelerometer data for 'bare-bone', samples A and B. No shock-like events are present, rather a steady ripple, especially in the Y and Z-directions.

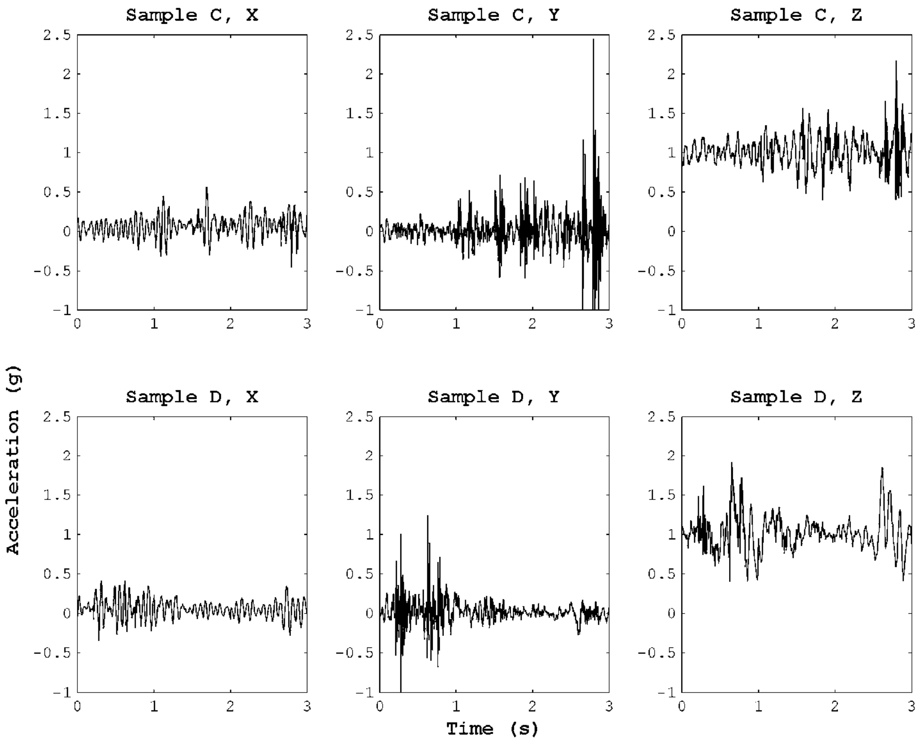

Fig. 4. Accelerometer data for 'dampener', samples C and D. Some shock-like events can be observed in the Y and Z-directions.

That a frequency shift occurs when introducing the dampener is confirmed in Fig. 5, where the single-sided Fast Fourier Transform (FFT) of all Z-direction acceleration measurements at 30 km h^{-1} is presented. All measurements for each of the two configurations were summed to show the different vibration characteristics. While frequencies in the range 10-50 Hz dominated in the 'bare-bone' measurements, the 'dampener' shows a peak below 25 Hz. When introducing the 'dampener' the total mass of the vibrating body also increases. This could be one factor that leads to a shift of vibration frequencies. The most obvious reason is of course the introduction of a soft dampening material - the polyester foam between the van's floor and the detector.

Spectral data for the four samples A-D are presented in Figs. 6-7. The microphonic noise mainly adds pulses in the low energy region below 20-30 keV. As much as 80 extra pulses per second due to microphonic noise were recorded (sample A). The 'bare-bone' samples A and B show more microphonic pulses than the 'dampener' (C and D), which also can be seen in the test-statistic, *u*, when comparing Figs. 1 and 2.

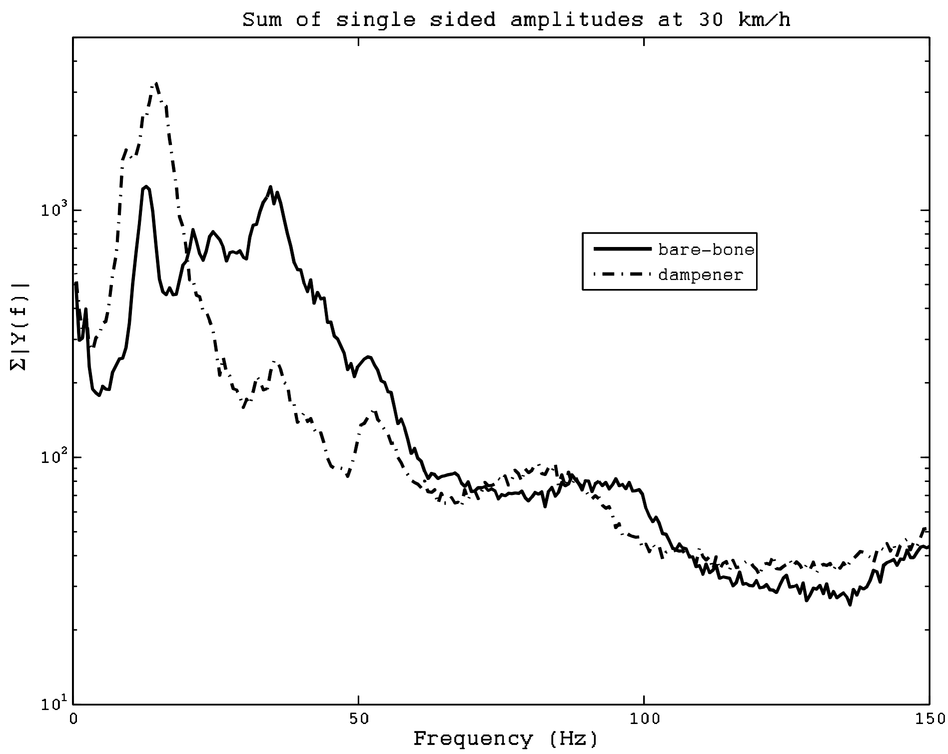

Fig. 5. Summed Single sided Fast Fourier Transform plot for all Z-direction acceleration measurements recorded at 30 km/h. A frequency shift occur when changing from the 'bare-bone' to the 'dampener' configuration.

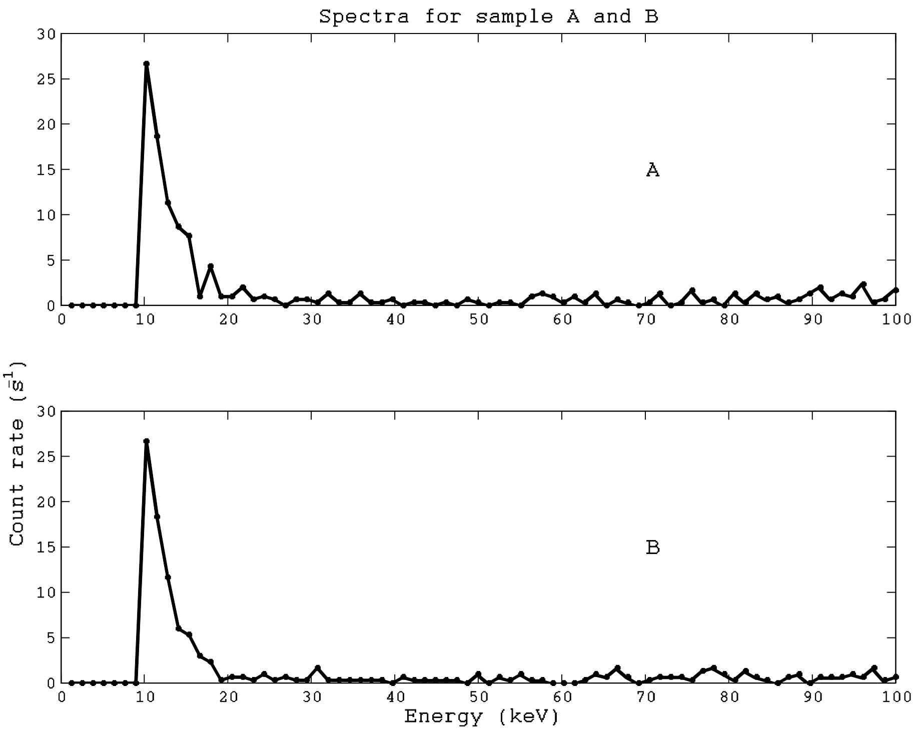

Fig. 6. Spectra for 'bare-bone' samples A and B. The main contribution in the low enery region (below 30 keV) is from microphonic noise.

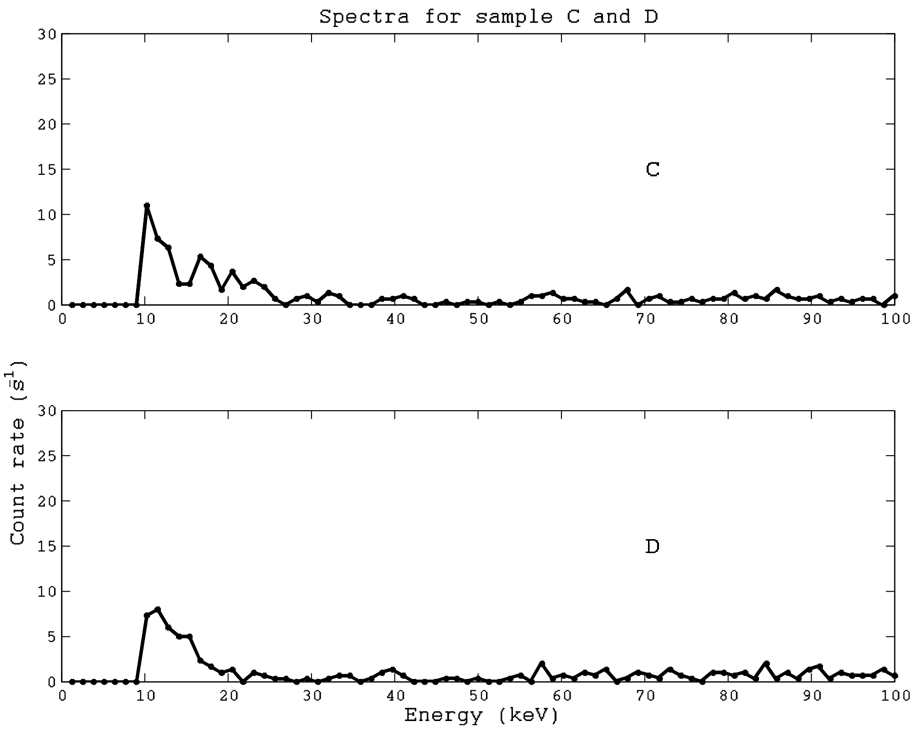

Fig. 7. Spectra for 'dampener' samples C and D. The main contribution in the low enery region (below 30 keV) is from microphonic noise.

Conclusions

As shown in this work microphonic noise occur when driving on uneven roads in moderate speeds. The microphonic noise increases as the speed of the vehicle, and hence the vibrations, increase. However, a few simple measures can reduce the microphonic noise considerably when using a car-borne HPGe detector system. A polyester foam dampener led to a reduction of the microphonic samples from 20 to 4 % of the total number of samples. Using a 3-axes accelerometer this work shows that the reduction was mainly due to a frequency shift in the vibrational forces acting upon the detector.

The results presented in this paper are hard to generalize, since they depend on a specific detector model, possibly even on an individual detector. However, it should be possible to conduct a more general experiment in a controlled laboratory environment using induced vibrations.

Acknowledgements

This work was supported by the Swedish Radiation Protection Agency (SSM). I thank Karl Östlund for his help with the construction of the dampener.

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Assessment of potential consequences of hypothetical radiological incidents in the seas of North-West Region of Russian Federation

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Abstract

In Northwest region of Russia there are the sites of storage and decommissioning of retired nuclear submarines, their nuclear reactors, spent nuclear fuel and radioactive waste. Many of those sites are not isolated and protected. There are also the atomic shipbuilding plants, naval objects. In the seas there is the traffic of atomic-powered vessels (submarines, ice-breakers). There is a hazard of release of radioactivity from those sites. This was the reason of implementing of the International Project “Improvement of the Murmansk region system of radiation monitoring and emergency response” (Project), managed by the European Bank for Reconstruction and Development. The Nuclear Safety Institute (NSI) of Russian Academy of Sciences was the main executor of the Project. Hypothetical scenarios and consequences of accidental radioactive contamination of coastal waters of North-Western region of Russia are discussed in this work. The scenarios assume the location of the hypothetical accidents to be in the White Sea and in the one of the Bays of the Barents Sea. The modeling of possible consequences of radiological incidents was implemented for water bodies that strongly differ in size (from 5 to 300 kilometers), tides, currents, and characteristic times of water exchange with Arctic Ocean. The modeling of migration of the radioactive substances was carried out with the use of computer model developed by the specialists of the NSI. This model is based on the well-known three-dimensional Princeton Ocean Model. The results of modeling of considered scenarios of accidents have shown that these hypothetical incidents do not constitute danger to the areas of industrial fishing in the Arctic Ocean as so as to the sea-shore coasts outside the water bodies were the hypothetical accident have taken place.

Management of the radiological situation regarding the wreck of the cruiser Murmansk

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Abstract

In December 1993, the decommissioned cruiser Murmansk was torn from its towing towards India for dismantlement and ran aground just outside the fishing village Sørvær on the coast of Finnmark in Northern Norway. The Norwegian Radiation Protection Authority (NRPA) received information about the wrecked vessel from the Norwegian Defence Command Headquarters. The coast guard carried out measurements and collected sea water samples. The initial concern was the possibility that there might be nuclear materials onboard the vessel, either in form of nuclear weaponry or spent nuclear fuel. NRPA made initial assessments and concluded that it was very unlikely that the vessel would contain radioactive substances that could pollute the surrounding area In June 2007 NRPA was notified by a private recycling firm that items taken from the wreck of the cruiser Murmansk showed sign of gamma radiation. The source was sent to a low level radioactive waste depository. Friday 1 August 2008 major Norwegian newspapers have lead stories on public concerns regarding radioactive substances at the ship. NRPA realized the need for a more thorough follow-up and decided to visit the wreck for new investigations. The investigations on the wreak was gamma dose rate measurements and collected samples of sea water and biota inside and outside the wreck. All samples was analysed and compared to expected values based on the Norwegian marine monitoring programme. The analysis did not show any radioactive contamination from the wreck. Also further measurements of concentrated sea water samples, alpha spectrometry of biota and sediments samples have not shown any trace of radioactivity. Handling of media was a main challenge through this event and it proved to be the most publicised news bulletin concerning NRPA in 2008. NRPA received very good public, political acceptance for handling the case and lead to a good cooperation between responsible authorities.

Overview of the observations on airborne and deposited radioactivity in Finland after the Chernobyl accident

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Abstract

The Chernobyl nuclear accident happened in the former Soviet Union on 26 April 1986. The air parcel trajectories originating from Chernobyl at the time of the accident show that the radioactive plume moved first northwestwards Over Lithuania the plume separated to two main paths. At lower altitudes (750-1000 m) the plume continued towards Sweden and Norway. At higher altitudes (1500-2500 m) the plume turned towards north. The plume arrived in South-Western Finland 27 April at 12 UTC for a release height of 2000 m. Then the plume went across the country north-eastwards and back to Soviet Union.

An aerosol beta activity monitor reacted to the Chernobyl-derived airborne radioactivity at Nurmijärvi, 40 km north-west of Helsinki, on the afternoon of 27 April. Unfortunately the alarm issued by the instruments did not cause any action due to a civil cervants' strike. The first alarm leading to a nation-wide alert occurred at Kajaani, north-eastern Finland, on the evening of 27 April 1986. A monitoring station of the Ministry of the Interior measured an increased exposure rate value of 0.1 mR/h (≈ 1 μSv/h) in connection with a rain shower. 29 April a rain area moved from the west coast of Finland in an easterly direction. The rain scavenged the activity to the ground causing notable increases in the external dose rate at several monitoring stations.

The monthly mean total beta activity concentration in April 1986 was the highest ever recorded, 1 Bq/m^3. The total beta activity concentration decreased five orders of magnitude at Nurmijärvi from 27-28 April to 31 May. In northern Finland the concentration level was clearly lower. In striking contrast to the 1960's nuclear weapons test fallout, the Chernobyl fallout was very unevenly distributed in Finland. The existence of "hot particles", highly radioactive agglomerates, in the Chernobyl plume was a specific feature during the early stages of emissions.

Introduction

The Chernobyl nuclear accident happened in the former Soviet Union, 130 km north of the city of Kiev, 26 April 1986. The reactor model, RBMK-1000, at Chernobyl was a graphite-moderated design using light water for cooling. The electrical output of the reactor model is 1000 MW. The accident destroyed one of the four reactors of the plant and released a huge amount of radionuclides into the atmosphere (International Atomic Energy Agency 1986).

The fourth unit of the power plant was scheduled for a regular maintenance shutdown 25 April 1986. In this connection a test programme was planned to be conducted. During the test a power surge in the reactor caused two subsequent explosions destroying the reactor and its surroundings. The graphite moderator caught fire and was burning for at least a week, thus prolonging the emissions of radioactivity into the atmosphere (International Atomic Energy Agency 1986). According to the Soviet estimates, all the radioactive noble gases of the core inventory were released during the accident as well as 10-20 % of the volatile nuclides and 2-6 % of plutonium isotopes and other refractory nuclides, e.g. ^{95}Zr and ^{144}Ce. A summary of the meteorological and radioactivity observations made in Finland is given in the following based on the paper of Paatero et al. (2010a).

Meteorological and hydrological situation

The energy released during the accident caused the radioactive plume to break through a ca. 400 metres thick inversion layer to the free troposphere. The air flow in the area around Chernobyl above the inversion layer was towards Poland and Lithuania, anti-cyclonally around the high pressure area in Russia (Fig. 1). Over Lithuania the plume separated to two main paths. At lower altitudes (750-1000 m) the plume continued towards Sweden and Norway. At higher altitudes (1500-2500 m) the plume turned towards north. The plume arrived in South-Western Finland 27 April at 12 UTC for a release height of 2000 m. Then the plume went across the country north-eastwards and back to Soviet Union. A frontal zone north of this route hindered the plume to reach northern Finland. The air stream brought, taking into consideration the season, exceptionally warm weather to southern Finland (Paatero et al. 2010a).

During 27 April, when the passage of the plume associated with the initial explosion occurred, there was no or very light rain in southern Finland. Slightly larger amount of precipitation occurred along the previously mentioned frontal zone from south-western Finland north-eastwards (Fig. 2a). During 28 April the weather was also quite dry in southern Finland except some local showers but during the next three days there was heavy precipitation along a zone from the coast of the Bothnian Bay south-eastwards towards the Kotka region (Fig. 2b). One third of the country was still covered by snow and even in southern Finland the agricultural activities hadn't started yet (Fig. 3)). Also the cattle was kept still indoors. This reduced the effects of the accident on various agricultural products during the early phase of the accident.

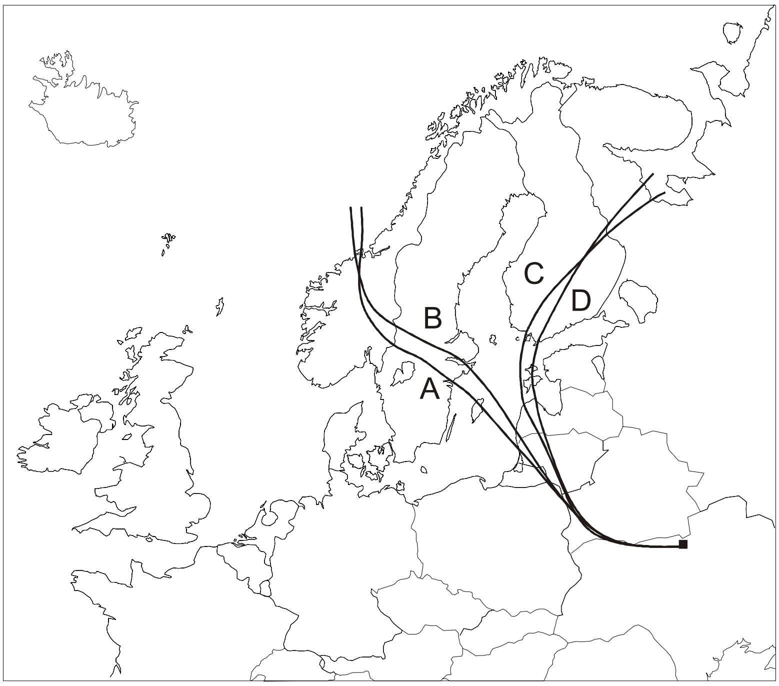

Fig. 1. Air mass trajectories from Chernobyl on 25 April 1986 at 21 UTC calculated with the TRADOS computer code (Redrawn from Valkama et al. 1995). Effective release heights: A = 750 m, B = 1000 m, C = 1500 m, and D = 2500 m.

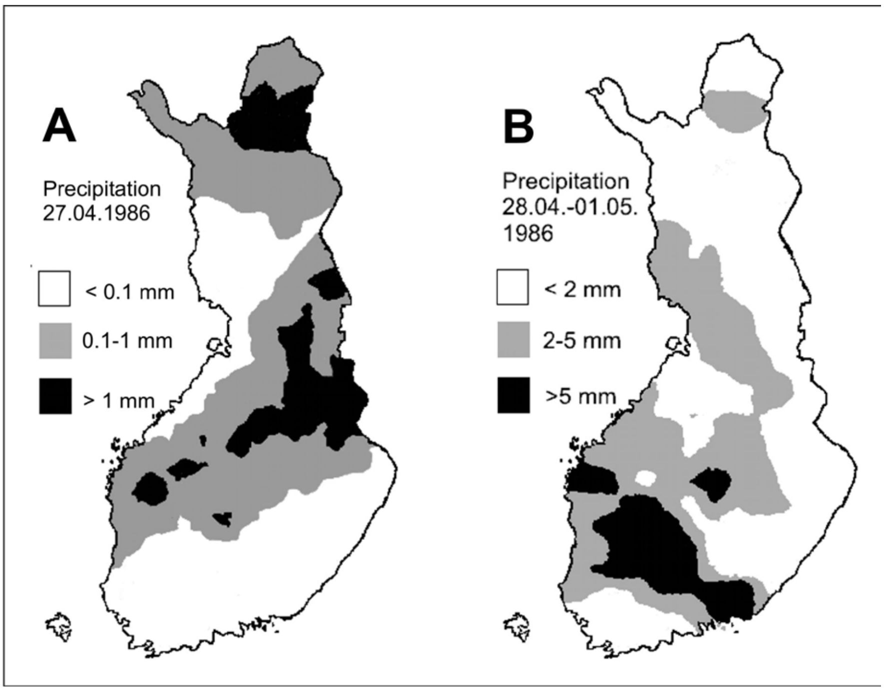

Fig. 2. Amount of precipitation (mm) in Finland, A) 27 April 1986, B) 28 April – 1 May 1986 (Data: Finnish Meteorological Institute).

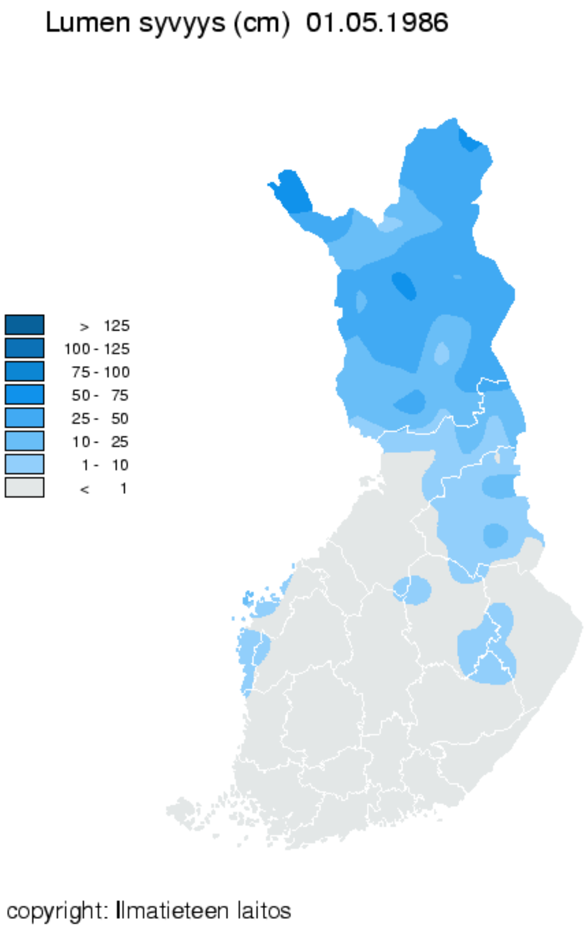

Fig. 3. Depth of snow cover (cm) in Finland 1 May 1986 (Data: Finnish Meteorological Institute).

Airborne radioactivity

An aerosol beta activity monitor (Paatero et al. 1994a) of the Finnish Meteorological Institute (FMI) reacted to the artificial radioactivity at Nurmijärvi but not in Helsinki on the afternoon of 27 April despite the only 40 km distance between the monitoring stations ((Fig. 4). This was probably due to the convection over inland Nurmijärvi while the lower troposphere was stratified in Helsinki due to the cold sea surface. Unfortunately the alarm at the Nurmijärvi monitoring station did not cause any action due to the civil cervants' strike. Most of the FMI's aerosol beta activity monitors in southern and central Finland detected artificial radioactivity on 28 April, especially in the afternoon owing to the increased vertical mixing of the troposphere.

The FMI has collected daily aerosol samples with a high-volume filter sampler at Nurmijärvi since 1962. The filter samples have been measured for total beta activity. The measurements are carried out five days after the end of sampling when the short-lived daughter nuclides of radon-222 have decayed to lead-210 (^{210}Pb) and the radon-220 progeny have decayed to stable lead (Mattsson et al. 1996). The measured activity consists of ^{210}Pb and possible artificial beta-emitting fission products. The monthly mean total beta activity concentration in April 1986 was the highest ever recorded, 1 Bq/m^3 (Paatero et al. 1994b). However, the time-integrated activity concentration was significantly higher in the early 1960s due to the Soviet and U.S. atmospheric nuclear tests. The total beta activity concentration decreased five orders of magnitude at Nurmijärvi from 27-28 April to 31 May (Fig. 5). Simultaneous data from Sodankylä showed that the daily activity concentrations were 1-4 orders of magnitude lower in northern Finland compared with Nurmijärvi during the first two weeks after the plume arrival.

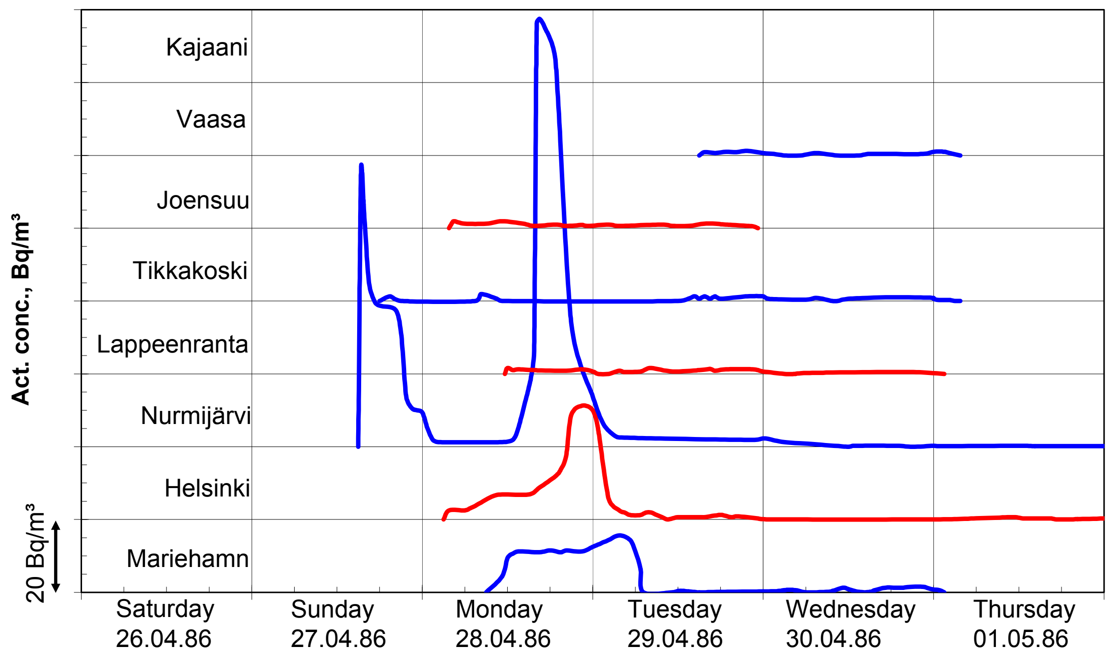

Fig. 4. Artificial aerosol beta activity in ground-level air (Bq/m^3) in Finland 26 April – 1 May 1986 (Redrawn from Mattsson and Hatakka 1986).

Between 28 April and 16 May 1986 the concentration of ^{137}Cs in ground-level air at Nurmijärvi decreased four orders of magnitude, starting from 10^4 mBq/m^3 (Finnish Centre for Radiation and Nuclear Safety 1986a). On 28 April the ground-level air at Nurmijärvi contained 32 µBq/m^3 of ^{239,240}Pu and 506 µBq/m^3 of ^{242}Cm (Jaakkola et al. 1986). For comparison, the annual mean ^{239,240}Pu concentration in the air in Helsinki varied between 7 and 26 µBq/m^3 in 1962-1964 due to the atmospheric nuclear tests (Jaakkola et al. 1979). Carbon-14 and tritium could be observed in the air in Helsinki only during the first three days after the arrival of the Chernobyl plume (Salonen 1987).

Highly radioactive agglomerates, so-called hot particles, were observed in Finland among other countries. These particles were either fragments of the nuclear fuel or particles formed by condensation of radionuclides and inactive structural materials of the reactor (Devell et al. 1986, Raunemaa et al. 1987, Lancsarics et al. 1988). The aerodynamic diameter of the Chernobyl-originated hot particles varied from one to several hundred micrometres; gravitational settling thus had to be taken into account when assessing their transport behaviour in the atmosphere (Pöllänen et al. 1997). Dozens of hot particles having an activity of over 50 Bq were detected from daily aerosol samples collected at Nurmijärvi, southern Finland between 27 and 30 April

(Fig. 6). The air volume of these samples was about 3500 m3. The number of particles with the activity ranging from 0.05 to 50 Bq in these filters varied between several hundreds and over ten thousand during these first days. Starting from 1 May only a few occasional hot particles were found (Mattsson and Hatakka 1986).

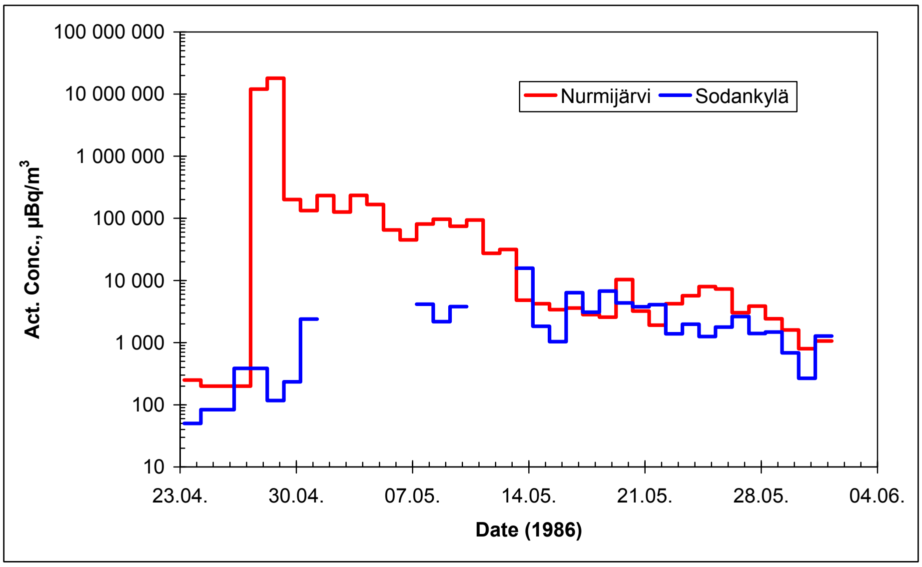

Fig. 5. Total beta activity concentration in ground-level air (µBq/m^3) at Nurmijärvi and Sodankylä 23 April – 31 May 1986 (Paatero et al. 1994b).

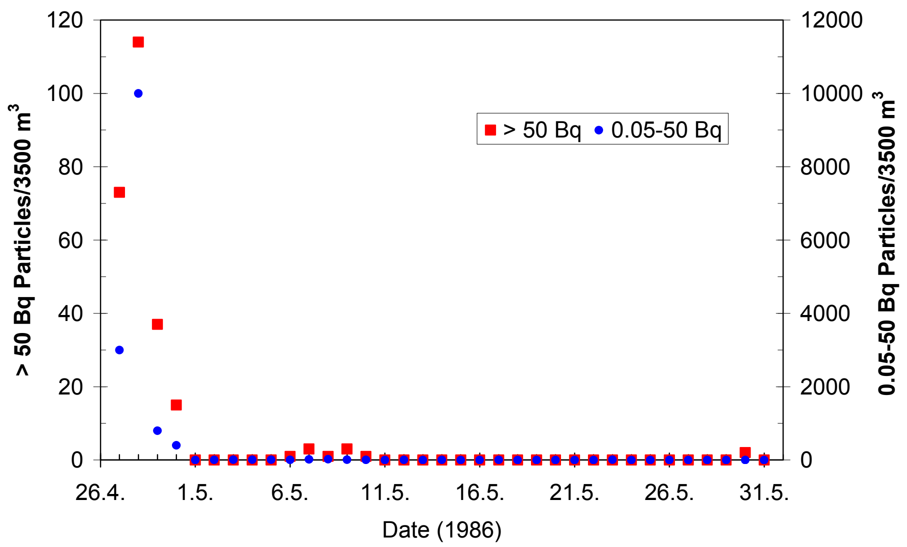

Fig. 6. Hot particles (> 50 Bq and 0.05-50 Bq) in the ground-level air at Nurmijärvi 27 April – 31 May 1986 (Mattsson and Hatakka 1986).

Deposited radioactivity

In striking contrast to the 1960's nuclear weapons test fallout, the Chernobyl fallout was very unevenly distributed in Sweden and Finland (Paatero et al. 2010b). In Finland the regional deposition pattern of different nuclides has been studied by in-situ gamma spectrometry and by collecting lichen, peat, soil and precipitation samples followed by laboratory analysis. Two main deposition patterns have been found. Refractory nuclides, such as ^{95}Zr and transuranium nuclides, were mainly deposited on a relatively narrow band from southwestern Finland towards north-east (Fig. 7. This pattern is associated with the passage of the air mass, that was over the Chernobyl region during the explosion of the reactor, and that contained debris of the reactor fuel (Paatero et al 2010a). The second pattern is associated with the volatile radionuclides, e.g. ^{134}Cs and ^{137}Cs. The deposition pattern of these nuclides is dominated by a heavy fallout area from western Finland towards south-east. The emissions of the volatile nuclides were more affected by the fire of the reactor ruins. The deposition pattern of these nuclides resembles the precipitation map of 28 April – 1 May in Figure 2b showing the importance of wet deposition mechanisms for removing these nuclides from the atmosphere.

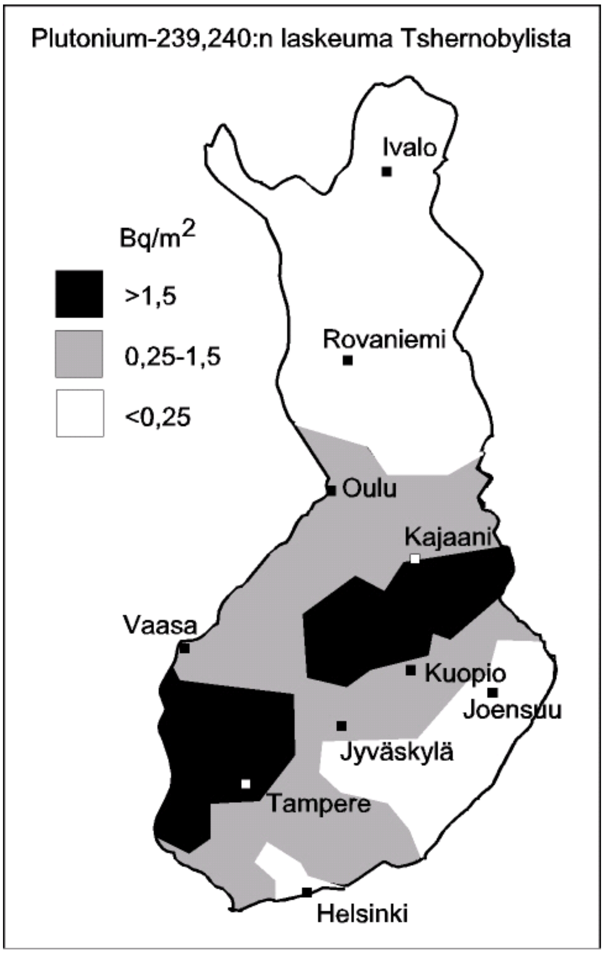

Fig. 7. Deposition of $^{239,240}Pu$ (Bq/m^2) in Finland after the Chernobyl accident (Paatero et al., 2006).

External radiation

The external dose rate was not significantly affected by the radioactivity in the ground-level air. The first alarm leading to a nation-wide alert occurred at Kajaani, north-eastern Finland, on the evening of 27 April 1986. A monitoring station of the Ministry of the Interior measured an increased exposure rate value of 0.1 mR/h (≈ 1 μSv/h) in connection with a rain shower scavenging the radionuclides to the ground (Finnish Centre for Radiation and Nuclear Safety 1986b). However, at the time the ground-level air there, as well as in most of Finland, was still practically clean from artificial activity excluding the above mentioned Nurmijärvi (Mattsson and Hatakka 1986).

Two days later, 29 April, a rain area moved from the west coast of Finland in an easterly direction. The rain scavenged the activity to the ground causing notable increases in the external dose rate at several monitoring stations. The recording of the external radiation in Uusikaupunki, on the west coast of Finland, is depicted in Fig. 8. The local rescue chief reported 2 May 1986 that the rain started just after the noon, at 12:30, of 29 April 1986 (Paatero et al. 2010a). Within three hours there was a 20-fold increase in the external dose rate. After its maximum, 3.7 μSv/h at 20:00 of 29 April 1986, the dose rate started to decrease with a half-time of 4.8 days. This suggests that much of the external radiation was due to ^{131}I (half-life 8.0 days) and ^{132}Te/^{132}I. The half-lives of ^{132}Te and ^{132}I are 3.2 days and 2.3 hours, respectively.

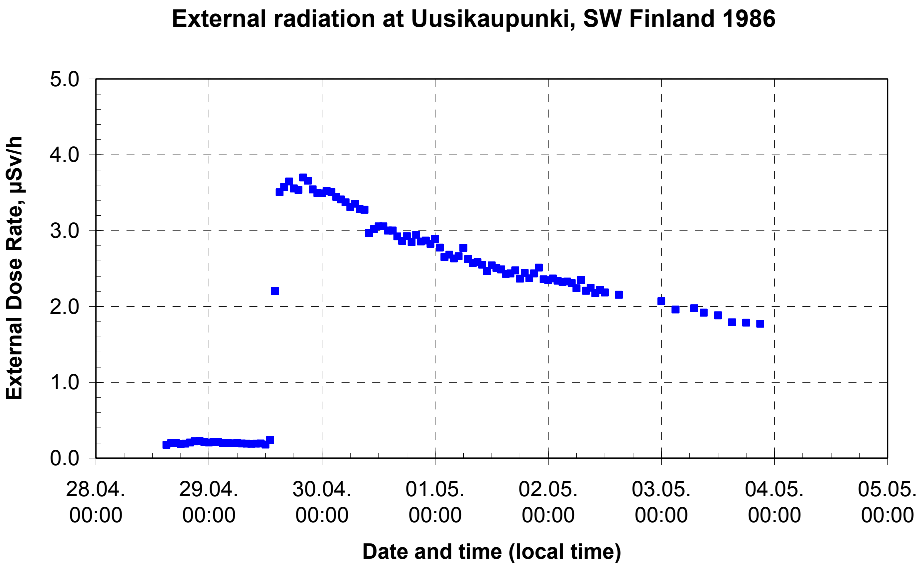

Fig. 8. External dose rate (μSv/h) in Uusikaupunki 28 April – 4 May 1986 (Data: Ministry of the Interior, Rescue Department).

Conclusions

The Chernobyl accident once again showed how a released radioactive or otherwise hazardous plume can, under certain meteorological conditions, rapidly move over long distances. A similar case was observed in December 1966 when radioactivity from a leaking underground nuclear test in Semipalatinsk, Kazakhstan, then USSR, was transported to Finland in three days (Kauranen et al. 1967).

In Finland medical consequences of the accident were mild and mostly psychological. No increase in birth defects or thyroid cancer occurrence has been observed in Finland (Harjulehto-Mervaala et al. 1992, Auvinen et al. 2001, Ikäheimonen 2006).

The Chernobyl accident revealed serious shortages in information dissemination between different agencies in Finland and abroad and to the mass media and the general public. After the accident a lot of resources in Finland were invested to improve the situation. These actions include both technical and organizational improvements. The authorities participating in the national radiation surveillance programme were able to obtain a variety of new radiation measurement equipment during the months and years following the accident. An important lesson was also that the proper functioning of the operational weather service, including weather observations, numerical weather prediction models and dispersion models utilising the previous two, has to be secured in all circumstances. Finnish authorities and university researchers managed, thanks to the experience gained since the late 1950's and the very rapid organisation of a coordinated research programme and funding by the Academy of Finland, to produce a huge amount of data in a very short time period after the accident covering a large variety of radiation protection aspects. Much of the information was gathered by research teams other than the actual radiation protection authorities. A possibility for such an ad hoc cooperation is important in a small country like Finland with limited human, economical and technological resources.

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Hot particles from atmospheric nuclear explosions

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Abstract

Hot particles were often observed with autoradiography in the 1960s and 1970s as a result of the atmospheric nuclear tests. These particles were formed by condensing material, both radioactive and inactive, from the exploding nuclear device, underlying soil etc. The Finnish Meteorological Institute (FMI) has monitored atmospheric radioactivity since the early 1960's at several monitoring stations in Finland by collecting aerosol particles onto paper or glass-fibre filters. After the measurement of total beta activity the collected filter samples have been archived. Since the mid 1960s some of the filters were also examined with autoradiography. Hot particles were frequently observed after atmospheric nuclear tests in Lop Nor, People's Republic of China. In this work the archived filters were analysed by digital autoradiography for screening the radioactive particles from them. Further analyses were done by scanning electron microscopy and energy dispersive X-ray spectrometer to study morphology, size and surface elemental composition of the localized radioactive particles. Based on electron microscopy results gamma and alpha measurements were performed. After this the samples were analysed by fission track technique to find out if fissile material is present in the particles. ICP-MS analyses were performed to determine plutonium and uranium content of the particles. Pu-239/Pu-240 ratios were determined with a sector mass spectrometer and U-235/U-238 ratios with a quadrupole mass spectrometer. The objective was to utilise modern instrumentation in the detection and analyses of hot particles in order to get new information on their physical and chemical characteristics. Contrary to particles found in e.g. lake or sea sediments or ice cores, the particles can be connected to individual nuclear tests. According to the preliminary results, uranium and plutonium were found in particles with a diameter of about 10 microns.

Biological dose reconstruction techniques operated by BfS

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Abstract

Biological dosimetry is an internationally established, independent method, employed in the area of radiation protection. Mainly utilised in addition to physical dosimetry, biodosimetry provides a chance for individual dose reconstruction and is often performed after an unclear or suspected radiation exposure in man. The field of application involves detailed individual dose assessments as well as fast, rough estimations in case of a large-scale radiological event. In this context, special cytogenetic markers such as chromosome aberrations and micronuclei analysed in blood lymphocytes turned out to be highly reliable and solid biological endpoints and are therefore widely used for biodosimetry purpose. The Federal Office for Radiation Protection (BfS) operates a laboratory which represents the national reference laboratory for biodosimetry in Germany. The focus lays on the dicentric analysis with dose effect curves for various radiation qualities, on FISH-analysis of stable translocations and on the analysis of micronuclei. The method of choice depends on the particular circumstances. Over the years, individual dose estimations in one or only a few persons have been a routine task. However, a new challenge has emerged in recent years in the form of a possible large-scale radiological event with the potential to involve a large number of exposed persons. In order to be prepared to act in an efficient manner in such an event, the BfS biodosimetry laboratory has improved its own capacity, in particular by increasing the throughput of analysed cells, e.g. by automation or by optimisation of the calculation mode in dependence of the required information. Nonetheless, the power of a single laboratory is only limited. To meet the requirements in case of a mass casualty, the BfS has established close working arrangements with several biodosimetry laboratories in Europe as well as on international level and is actively involved in national and global networking.

^{85}Kr in industrial krypton gas: origin, identification and dosimetry

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Abstract

Industrial krypton gas is produced by fractionated distillation of liquefied air, in which it has a natural abundance of about 1 ppm_V. The radioisotope ^{85}Kr is present in the atmosphere at a concentration of about 1.5 Bq/m^3, due to continuous emissions from nuclear installations. Pure krypton gas thus will contain 0.35 MBq/kg of ^{85}Kr, and gas cylinders will contain many MBq of the isotope, making it detectable by radiation monitors. A case of an explosion of such a cylinder inside a facory building with subsequent detection of increased radiation levels by the fire brigade, followed by evacuation of the building and decontamination of persons, is reported in the U.S. press.

We report on a case of transcontinental transport of krypton gas, triggering of a radiation alarm and subsequent in situ measurement by different radiometric techniques. Quantitative in-situ gamma spectroscopy identified the isotope ^{85}Kr and revealed activity levels consistent with its atmospheric concentration and with measured and calculated dose rate values.

Possible other cases of such „TEARM“ (technonogically enhanced artificial radioactive material) and possible techniques for detection and discrimination from attempts of nuclear smuggling are discussed.

Introduction

The radioisotope ^{85}Kr is a main fission product and contributes significantly to the releases of nuclear power installations. The main emissions occur during fuel reprocessing and are estimated as about 10^4 TBq/GW·yr (UNSCEAR 2000). In 2008, the main emitter worldwide was the fuel reprocessing plant at La Hague, France, having released about $1.55 \cdot 10^{17}$ Bq of this isotope (BMU 2009). The atmospheric distribution is homogeneous and currently the equilibrium between atmospheric releases and radioactive decay results in a concentration of ^{85}Kr in air of about 1.5 Bq/m^3, relatively constant with location and time.

The radiological properties of ^{85}Kr, apart from the well-known half-life of 10.756 yr and $ß^-$ decay mode, include a low-abundance (0.43%) gamma emission of 514 keV.

Stable krypton is an inert gas, which is present in the Earth's atmosphere at a concentration of about 1.1 ppm$_V$ (CRC 1984, Aoki 2005). It is used industrially in the production of halogen sealed beam headlights and high-efficiency dual-pane and triple pane windows. Pure krypton is obtained by fractionated distillation of liquefied air. Given the abovementioned concentration of ^{85}Kr in the atmosphere, a ^{85}Kr concentration in pure krypton of about 350 kBq/kg can be predicted.

The documentation for industrial krypton gas mentions the „slightly radioactive“ property (Air Liquide 2005a), in one case a value of 300 Bq/g is given for the activity concentration (Air Liquide 2005b).

Press records reveal an event involving the explosion of a krypton gas cylinder in a factory building located in Jacksonville, FL, USA, on January 31 2006 (New York Times 2006, First Coast News 2006). During rescue operations, a radiation alarm was triggered, and in consequence, about 40 persons were decontaminated. Apparently, the radioactive propertiy of pressurized krypton was neither known to the company using it nor to the first responders, nor was the uselessness of decontamination measures known when dealing with an inert gas.

Material and methods

Record of events

A cargo of industrial krypton in 159 steel cylinders, each containing about 30 kg of pressurized gas, was shipped from Russia to the U.S.A. by truck and ship. On its way it had activated a radiation alarm during a border crossing in East Europe, but was allowed to continue its itinerary. No records of the measurement had been taken, and none of the persons involved was aware of the inherent radioactive properties of the cargo. The authors' institution was asked by the shipping company to identify the source of the radiation and to determine the possible dose to workers during a planned manual repacking of the cylinders.

In situ measurements

The cargo was investigated using standard handheld instruments (beta/gamma contamination monitor and dose rate meter) in a first field trip, and by application of in-situ high-resolution gamma spectroscopy, using a hpGe detector of 10% relative efficiency, performed one day later. A photograph of the cargo with the dose rate meter in measurement position is shown in Fig. 1. A sketch of the measurement geometry for gamma spectrometry (with the detector in the same place) is displayed in Fig. 2.

Dose and dose rate calculations

External and internal doses and dose rates were calculated by different methods, in order to take into account radiation protection issues of the current situation and of the reported gas cylinder explosion.

Results

Handheld instruments

The measurements using handheld instruments revealed values about 50% above background (dose rate and count rate of the contamination monitor), at various positions on top of the cargo. Background subtraction resulted in a net mean dose rate of 0.058 µSv/h at the cargo surface. Shielding of natural background radiation from the ground by the cargo was assumed to increase this value.

Fig. 1. Cargo of steel cylinders containing pressurized krypton gas, with the dose rate meter in measurement position.

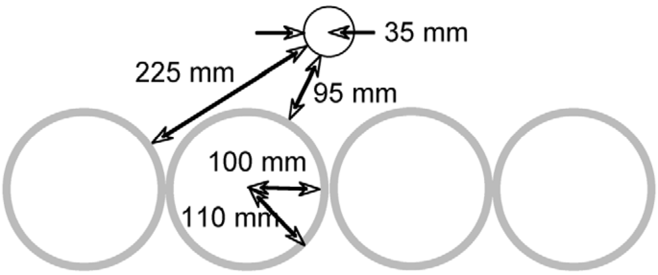

Fig. 2. Schematic representation of the geometry applied in the gamma spectrometry measurements.

Calculation of dose rate from activity concentration

After the first set of handheld meter measurements, an attempt was made to estimate the expected dose rate under the hypothesis of ^{85}Kr being responsible for the increased values. This was done using a handbook developed for photon shielding calculations (Foderaro 1976). The handbook did not provide the complicated geometry present in the measurement (similar to the gamma spectroscopy geometry shown in Fig. 2). Instead, a simplified planar geometry of 20 cm of pressurized krypton gas below 1 cm of steel was used. Including build-up effects, a dose rate of 0.074 µSv/h was obtained. The measured dose rate increase above the cargo was 0.058 µSv/h, plus an unquantified contribution from shielding of natural background. In view of this uncertainty, the result could be considered satisfactory.

In a second calculation, the Monte Carlo code EGS-Ray (Kleinschmidt 2001), which is based on the code EGS4 (Nelson 1985), was used with the same goal. Here, the geometry could be modelled more precisely, resembling the structure shown in Fig. 2. The obtained result was 0.77 µSv/h, in good agreement with the estimate based on tables.

Calculation of internal and external dose after a krypton gas release

Internal and submersion doses for the case of a release of krypton gas were obtained by applying tabulated dose conversion factors. The applied scenario was the complete release of the content of one gas cylinder (about 30 kg) and its homogeneous distribution within an air volume of 10,000 m^3, like it could be expected in a factory building, leading to an activity concentration of about 1000 Bq/m^3. The exposure time was taken as 10 minutes, and no ventilation was assumed (both values together were intended to be conservative for most situations). Dose conversion factors were obtained from ICRP Publication 72 (ICRP 1996) and German state regulations (BfS 2001). The former include doses from inhalation together with beta and gamma submersion, whilst the latter, also based on the ICRP model, allow for discrimination between the different dose contributions. Results are shown in Table 1 and reveal the dominant contribution of beta submersion. In general, doses are low, but the isotope concentration might trigger radiation alarms.

Table 1. Organ and effective doses resulting from a 10-minute exposure to the content of one krypton gas cylinder released accidentally into the air volume of a 10,000 m^3 building.

Exposure path	Conversion factor (Sv d/Bq m^3)	Data source	Dose type	Dose value (µSv)
Inhalation, beta and gamma submersion	2.2 10^{-11}	ICRP 1996	Effective	1.5 10^{-4}
Gamma submersion	1.4 10^{-13}	BfS 2001	Skin	9.6 10^{-7}
Gamma submersion	7.8 10^{-14}	BfS 2001	Effective	5.4 10^{-7}
Beta submersion	1.1 10^{-9}	BfS 2001	Skin	7.8 10^{-3}

In situ gamma spectrometry

Spectra recorded on the cargo surface (in the geometry shown in Fig. 2) and at 1 m lateral distance both revealed a prominent peak at 514 keV, proving the involvement of ^{85}Kr. The spectra also revealed the 511 keV e^+/e^- annihilation peak and several signals from natural isotopes (indicated in Fig. 3).

After performing a numerical efficiency calibration for the detector in measurement geometry by means of a commercial program (LabSOCS, Canberra Inc.), the obtained activity concentration in the cargo was 0.37 MBq/kg, in very good agreement with the expected value of 0.35 MBq/kg. Furthermore, the comparison of the spectra on top of the cargo and at the side of it revealed a decrease of the peaks resulting from natural background by about 40% or more. This finding supports the assumption of a decreased natural background dose rate above the cargo.

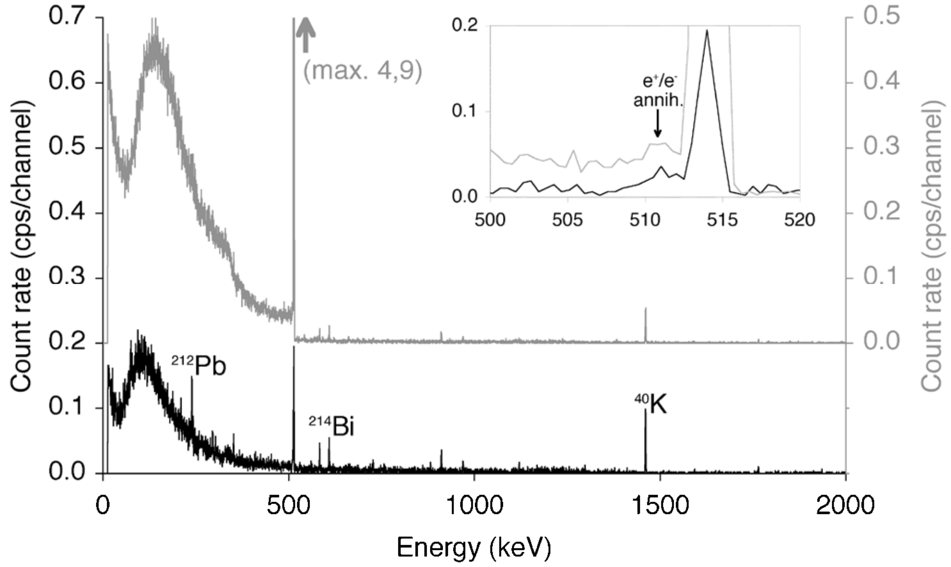

Fig. 3. In situ gamma spectra recorded on top (grey, right y scale) and at 1 m lateral distance (black, left y scale) of the gas cylinders. Spectra have been normalized for count rate. The insert displays the ^{85}Kr peak region, revealing also the 511 keV annihilation peak.

Discussion

By which methods could the isotope be identified unequivocally?

Knowing the concentration and the radioactive properties of ^{85}Kr in pressurized krypton gas, the measured dose rate is already a plausible indication for its presence. The increase of the reading of the (beta/gamma sensitive) contamination monitor by the same factor indicates the absence of external contamination by beta emitters. However, the expected dose rate is not readily available, e.g. in data sheets, and requires considerable knowledge and effort to be calculated. Judged from the present case, gamma spectroscopy appears to be the most straightforward way towards an unambiguous result. As in this case the characteristic gamma energy of 514 keV is extremely close to the 511 keV annihilation line, high resolution is required, requiring a semiconductor detector.

Which occupational doses could be expected?

Due to the low gamma emission probability in ^{85}Kr decay, despite the high activity involved (about 1.5 GBq in total), expected and measured external dose rates are very low.

Could an attempt of nuclear smuggling be excluded?

The overall consistency of the results revealed ^{85}Kr as the main radiation source. An additional source of radiation, which might have been hidden within the cargo, would have been detected in the gamma spectra in case of a gamma source of considerable activity. Pure alpha or low energy gamma emitters would not have been detected, but they would not have been found either by any other radiometric procedures.

Can similar cases be expected in the future?

The authors expect similar findings to appear with increasing frequency, due to the increase of the numbers of radiation monitors at borders, ports, scrap yards and waste repositories. The author's laboratory experience includes findings of ^{60}Co in stainless steel, depleted uranium in scrap metal and ^{131}I in hospital waste. The common property of these objects is their artificial origin and their concentration in specific materials due to human technical action. The authors therefore propose the usage of the term "TEARM" (for technologically enhanced artificial radioactive materials) in analogy to "TENORM" (for technologically enhanced naturally occurring radioactive materials). Labeling of such materials might help to identify them as (often weak) radiation sources and to discriminate them from potentially more hazardous material.

Conclusions

The measurements, calculations and considerations show that

– ^{85}Kr in industrial krypton gas is a weak source of radiation, often unknown to persons involved in its usage or transport;
– it might be advantageous to label materials like krypton gas as "slightly radioactive", indicating expected dose rate levels;
– high resolution in situ gamma spectroscopy appears to be the method of choice in the unambiguous identification of unknown radioactive materials.

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Airborne and satellite data acquisition of the contaminated landscape for the countermeasures in agriculture

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Abstract

In case of the fallout from large accident of a nuclear power reactor or other large-scale release of radioactivity into the environment, a large area to the extent of 10 000 – 100 000 km^2 could be contaminated. To find out as quickly as possible the extent, magnitude and radionuclide vector of the contamination is important task for agricultural countermeasures and foodstuff restrictions in early/middle phase of accident. In this paper are analyzed the modern data gathering methods of landscape contamination. They are based on combination of satellite technology and airborne gamma dose rate and/or gamma spectroscopy. The issue of optimal height of flight, plane velocity, "hot spot" detection and strategy of mapping process are discussed. This work was supported by the grant of the State Office for Nuclear Safety Czech Republic No. 1/2008 "The new methods for evaluation of landscape contamination after large accident of NPP".

Long-term development of incorporation dose at Korma County (Belarus) after the Chernobyl accident

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Abstract

Radiological long-term measurements were performed between 1998 and 2007 in Korma County, Belarus. The internal radiation exposure of the inhabitants in the village of Volincy was assessed by mobile in-vivo monitoring and has experienced a significant decrease from a high level. This was the result of the decreasing environmental contamination (top soil and crops) and specific advice offered to the population.

Presently the internal dose is only slightly enhanced over background values and of no special relevance to health. It is expected to decrease to less than 0.2 mSv/a in 2011 and to below 0.1 mSv/a in 2020.

Introduction

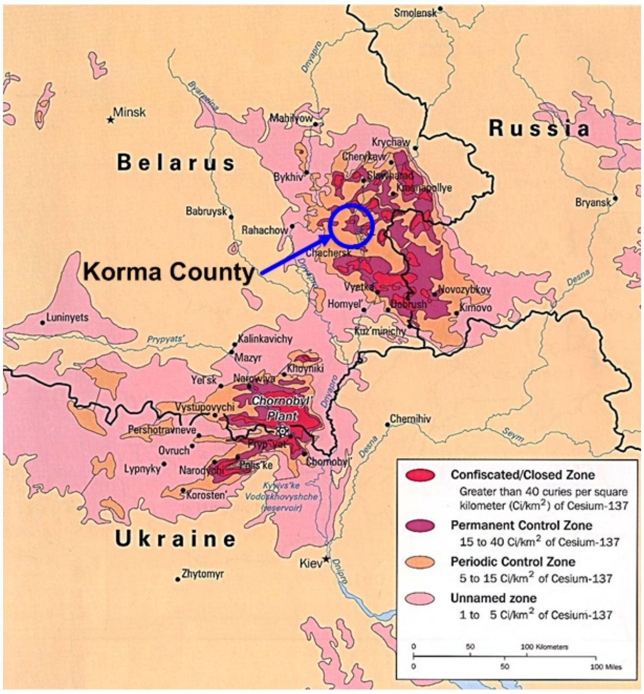

Fig. 1. Confiscated/closed and control zones after the Chernobyl accident; location of Korma County indicated (CIA 1996).

Due to the Chernobyl accident (Hill 2003) in 1986 large parts of Belarus (fig.1) suffered from radioactive fall-out. The German Chernobyl project (Hille et al. 1996] assessed the situation of the population during the years 1991-1993. Approx. 317.000 mobile in-vivo measurements had been performed in more than 240 settlements of Belarus, Russia and the Ukraine. Environmental measurements and some direct measurements of external dose added to the assessment of population dose.

In follow-up studies the long-term development of population dose has been investigated in highly contaminated regions. One project concentrated on the situation in Korma County (Dederichs et al.

2009). The measurements included among others mobile in-vivo monitoring of the population. We report on measurements performed between 1999 and 2007.

Korma County is situated approx. 70 km north of the city of Gomel/Belarus and about 200 km from the Chernobyl NNP (fig. 1). There are areas of comparatively low contamination, but also confiscated zones. The municipality of Volincy is quite remote, since it can be reached only by a road (fig.2) passing through the confiscated zone of Strumen. The municipality of Volincy is situated in the forest belt of Korma County. Though there is some agriculture it can be considered to have a more forestal character. On this account elevated body burdens had to be expected.

Fig. 2. Mobile in-vivo monitoring laboratory on the road to Volincy (Dederichs et al. 2009).

Material and methods

Mobile in-vivo monitoring

Van type mobile in-vivo monitoring laboratories were used to assess body burdens. The most recent version was based on a Mercedes Sprinter van. Fast enough to be moved from Germany to Belarus in a reasonable time and small enough for the small roads typically encountered at the remote areas it still is able to carry the weight of a heavily shielded incorporation monitor. The latter one is a self-designed whole body counter based on two large NaJ-detectors (40cm x 10cm x 10 cm). Data acquisition and analysis is PC-based. The calibration is weight dependent. A brick phantom (Kovtun et al. 2000) had been used to determine peak efficiencies.

Dose calculation

The calculation of dose is based on dose factors directly related to body burdens since a more or less chronical incorporation can be assumed. They were derived from age-dependent dose factors given in ICRP67 (ICRP 1994). For youths and children dose factors were extrapolated for each year of birth.

Results

At the municipality of Volincy mobile in-vivo measurements from autumn 1998 to autumn 2006 and at the municipality of Starograd from spring 2004 to spring 2005 are included in the present evaluation. Measurements were performed for a total of 499 individuals at Volincy and 274 individuals at Starograd. Measurements were sometimes done twice a year (spring, autumn), sometimes once a year (spring or autumn) and there were no measurements in 2003. Though many persons participated regularly in the measurements this has not been the case for everybody. A complete list of more than 2600 measurements including the results for each individual is given in the final report of the project (Dederichs et al. 2009).

In-vivo measurements at the municipality of Volincy

The municipality of Volincy consists of the three settlements Volincy, Kljapin and Klapinskaya-Buda. The conditions of living are slightly different in the three settlements. Measured body burdens (spring values) have been plotted in dependence of the year of birth of the person concerned for each of the settlements individually (Volincy fig.3, Klajpin fig.4, Kljapinskaya-Buda fig. 5).

Generally the highest body burden has been observed at Volincy itself. The bulk of body burdens strayed within a range of values which was more or less the same in each year.

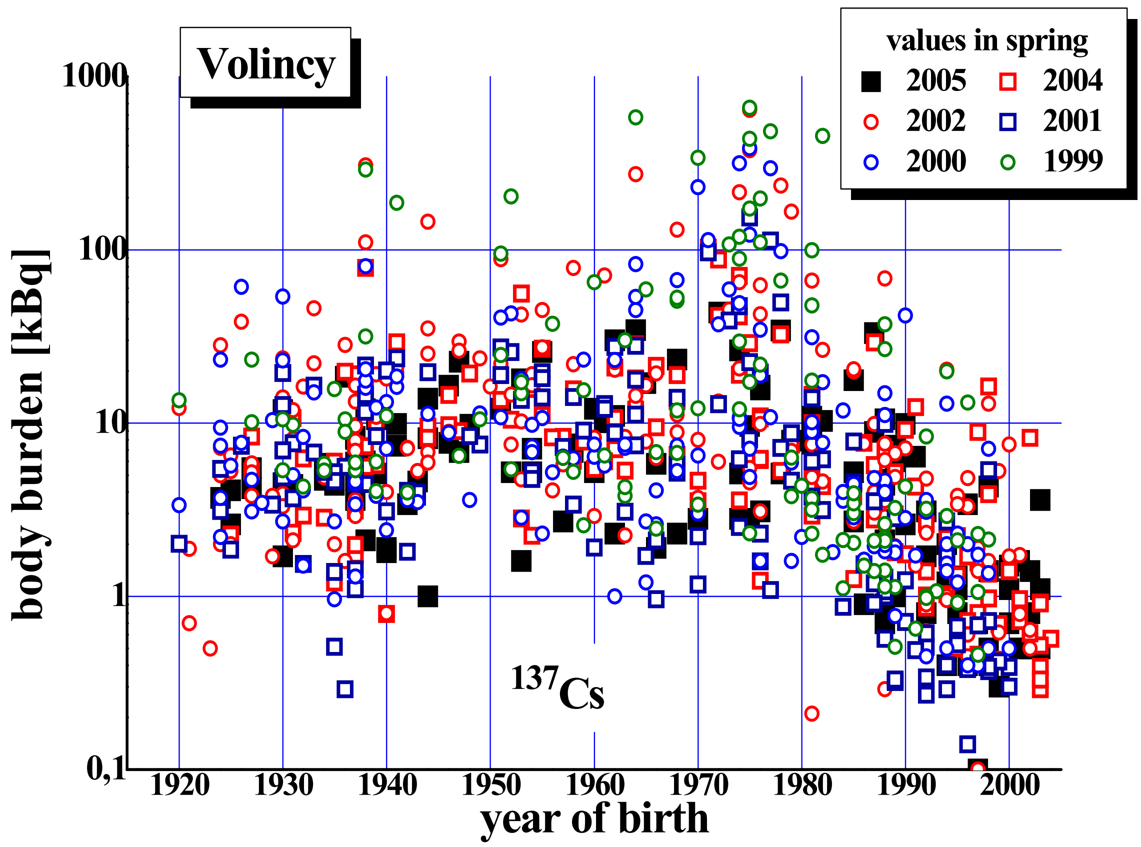

Fig. 3. Body burden (Cs-137) of the population at the locality Volincy (Dederichs et al. 2009).

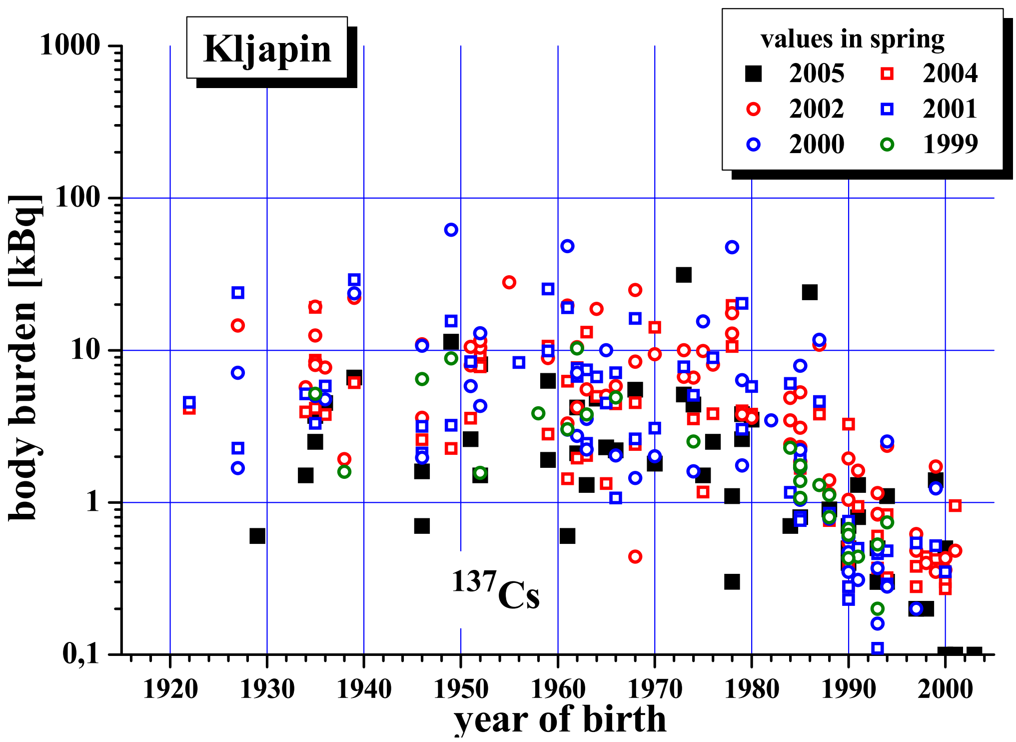

Fig. 4. Body burden (Cs-137) of the population at the locality Kljapin (Dederichs et al. 2009).

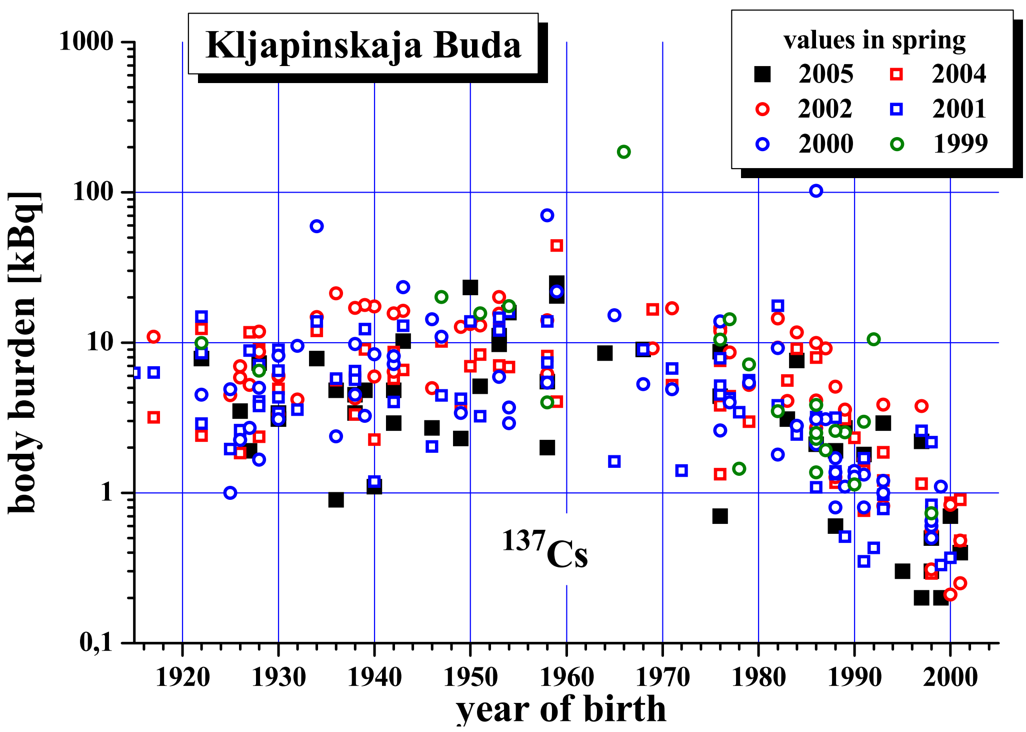

Fig. 5. Body burden (Cs-137) of the population at the locality Kljapinskaya-Buda (Dederichs et al. 2009).

Body burdens at Volincy were in the beginning generally 4 times higher than in the other two settlements and even in 2005 still two times higher. High values of several hundred Kilo Becquerel Cs-137 were occasionally observed and got less over the years. Such values were more common with people born between about 1970 and 1980 than with others. This effect of age was neither seen at Kljapin nor at Klapinskaya-Buda, the latter however not having much inhabitants born in those years. The population of Kljapin is characterized by a large fraction of employees of the local school.

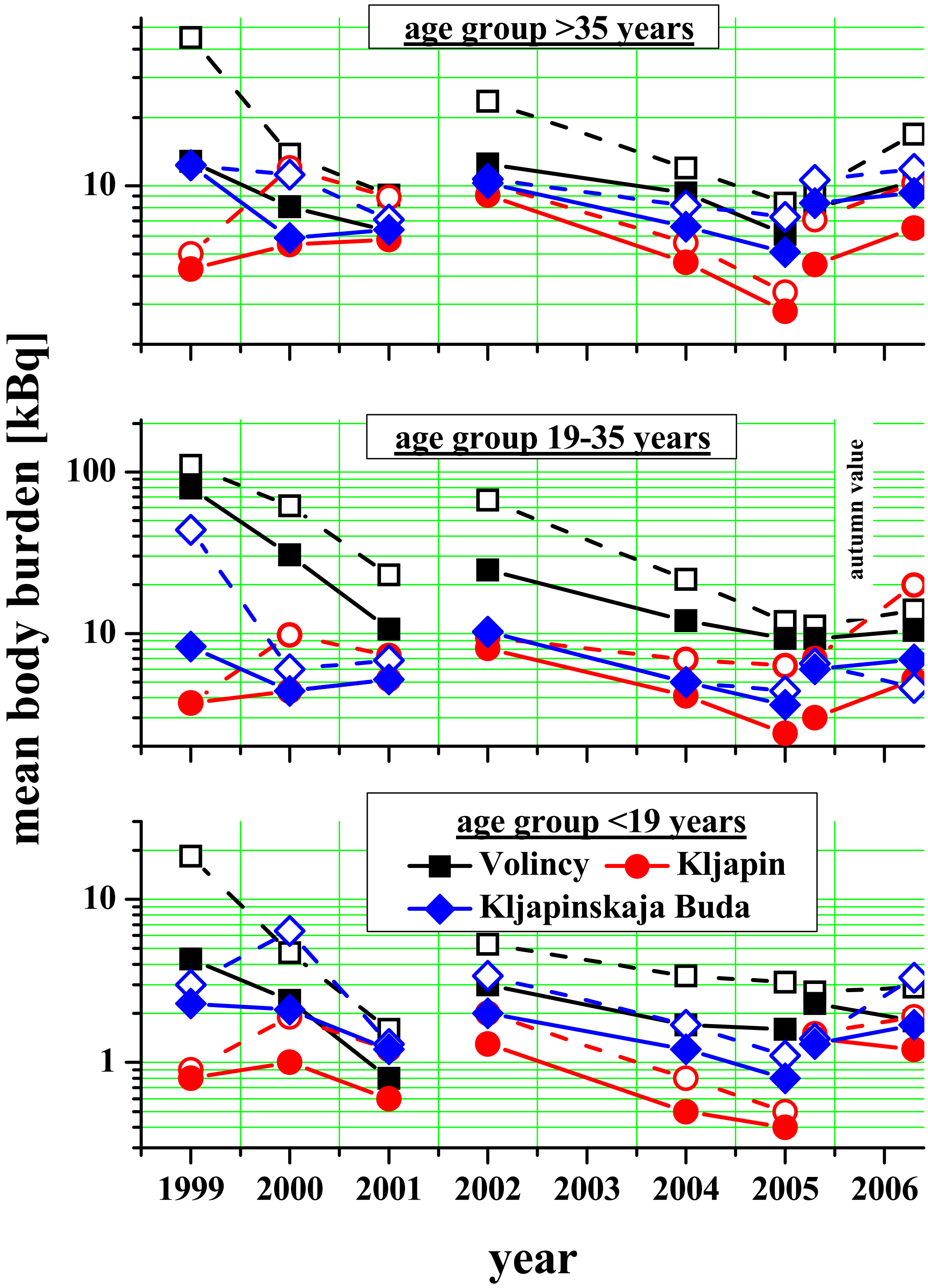

Fig. 6. Time-development of mean body burdens (Cs-137) at the municipality Volincy for different age groups and localities (Dederichs et al. 2009).

The arithmetic means of the body burdens have been evaluated for the municipality Volincy and each of its localities Volincy, Kljapin and Klapinskaya-Buda. Their development with time is shown in Fig. 6 for three different age groups:

- Adults > 35 a
- Young adults (19 a – 35 a)
- Children and other young people (<19 a).

Open symbols represent mean body burdens obtained by including all measurement results. Closed symbols represent mean body burdens calculated by excluding all outliers deviating from the mean more than three times the standard deviation obtained from the variance of the distribution of results.

In 1999 body burdens were quite high, especially in the locality Volincy. Over the time body burdens generally dropped significantly: 66% at Volincy itself and about 50 % in the other two localities. This can be compared to a decrease of Cs-activity in the environment of about 30%. It should be noted that people had been individually advised on possible means of dose reduction by the measuring team.

A first period of funding ended in spring 2001. Villagers did not expect an early continuation of the field missions and the campaign in spring 2002 somehow seemed to catch them by surprise. The discipline in following dose avoidance procedures had begun to decrease and not surprisingly the mean body burden increased significantly. However at continuation of regular field missions in 2004 mean body burdens had dropped down back to reasonable values.

To account for seasonal effects some measuring campaigns have been performed in autumns. In spring 2006 no assessment could be done since the municipality Volincy was not accessible by the mobile in-vivo monitoring laboratory due to high waters at the river Sotch. The mean body burdens for autumn 2005 and autumn 2006 are comparable. Only the age group of adults older than 35 years shows a slight enhancement, which is attributed to the consumption of mushrooms and the especially large number of mushrooms available in 2006.

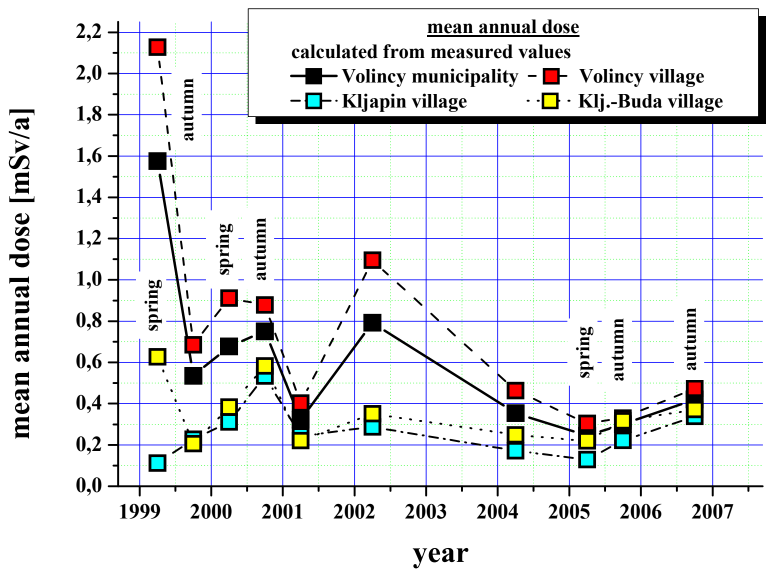

Fig. 7. The mean annual dose due to internal contamination (Cs-137) observed for different localities in Volincy municipality (Dederichs et al. 2009)

Internal doses have been calculated from measured body burdens individually. It has been assumed that for conversion to annual dose a seasonal factor of one can be applied. The mean annual dose is shown in fig. 7 for Volincy municipality as a whole as well as for the three localities separately. No distinction has been made between the different age-groups. Outliers have been included in the calculation of the mean. Though age-dependant dose factors have been applied time-development of the mean annual doses show the same tendencies already observed for the underlying body burdens: sharp drop from high to moderate values, increase in 2002, back to reasonable values in 2004 and stabilization of doses received in subsequent years. Especially in spring 1999 some rather high individual doses have been observed , in one case even exceeding the annual limit for professional radiation workers and in some other cases getting quite close to it.

In-vivo measurements at the municipality of Starograd

At Starograd and its different localities measurements were performed in spring 2004 and spring 2005. The people measured were mainly pupils, as can be seen from fig. 8 which is once more a plot of body burden against year of birth. Body burdens at Starograd are much lower than in the municipality of Volincy. The contamination of ground is comparatively low and the municipality's basic character is more agricultural. There is no necessity to plot the results obtained for the different localities of Starograd separately. The results correspond to internal doses of less than 0.1 mSv/a, with the exception of one single child, which only slightly exceeds this value.

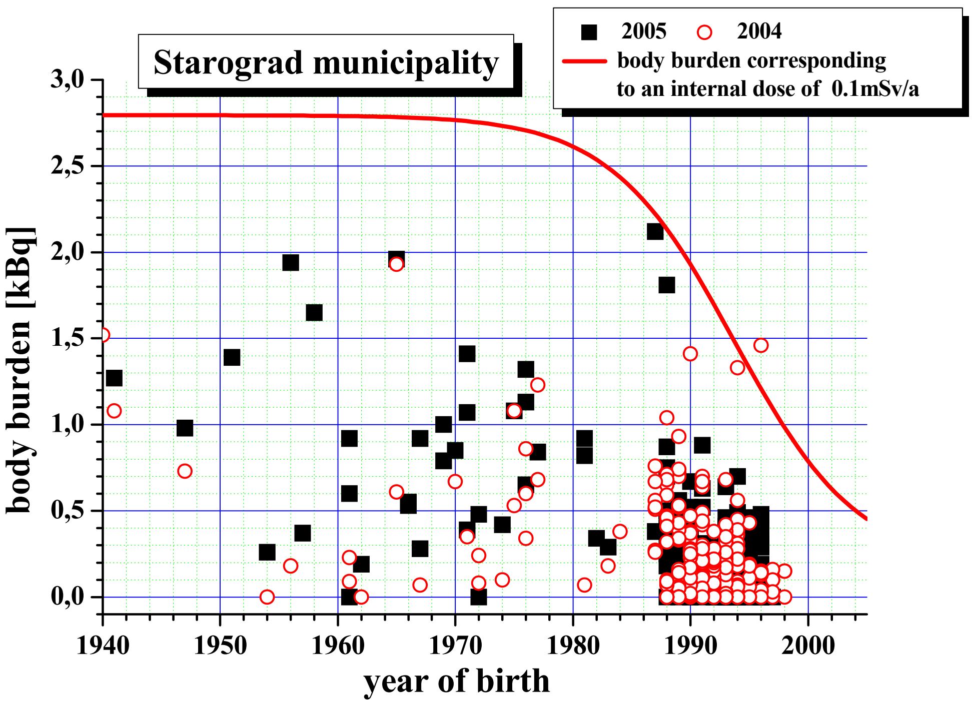

Fig. 8. Body burden (Cs-137) of the population at the municipality Starograd (Dederichs et al. 2009).

Discussion

In 1999 body burdens were quite high. Over the time body burdens dropped significantly. Besides the knowledge of the personal burden individual advice provided by the measuring team led to a changed attitude towards the consumption of contaminated food. Body burdens stabilized at more reasonable values. A slight increase in autumn 2006 is believed to be connected with the measuring campaign correlating with the end of the mushroom season and especially large harvest of mushrooms.

Comparison of different population groups at Volincy

The body burden of children and teens is quite low. Mean values of internal dose where below 0.2 mSv/a both at Kljapin and Klapinskaya-Buda after autumn 2000, at Volincy below 0.3 mSv/a. Children and teens usually receive meals at school where ‘clean’ foodstuff is used. It is obvious that additional meals taken at home do not create any considerable internal burden.

In the age group of 19-35 years old the situation is completely different. Whereas in the other two settlements the mean internal dose is comparable to other adults, at Volincy up to the 4-5 fold has been observed in some years. After thorough individual counselling mean values finally in 2005 and 2006 got more close to those of the older adults. It seems that members of this age group were especially careless in consuming forest products and contaminated meat.

Generally men seem to have about two times higher body burdens than women. About the cause can only be speculated (different composition of food, different fat/muscle ratio….).

Comparison of agricultural and forestal districts

The municipality Volincy is situated in a forestal area with elevated ground contamination, whereas the municipality Starograd is situated within an agricultural area which belongs to a part of Korma County where deposition of Cs-137 had been comparatively low.

Mean values of body burdens at Volincy and Starograd are shown in table 1 for the years 2004 and 2005. For the total population the incorporation in Volincy is about 20 times higher than in Starograd. This is more than the difference in deposition would count for.

It is known (Dederichs et al. 2009) that agricultural products and garden products contain considerable less activity than forest products. Forest products contribute essentially to body burdens observed.

The huge difference in body burdens is in agreement with an habit of consuming local food with no or only marginal exchange of products between different settlements of the county.

Table 1. Comparison of mean values of body burdens at the municipalities of Volincy and Starograd.

Population Group	Volincy 2004	Starograd 2004	Volincy 2005	Starograd 2005
Total	8.1 ± 7.0 kBq	0.38 ± 0.28 kBq	5,7 ± 4.9 kBq	0,28 ± 0.23 kBq
Adults	12,2 kBq (19-35 a) 8,5 kBq (>35 a)	0,84 kBq (> 18 a)	7,1 kBq (19-35 a) 6,4 kBq (>35 a)	0,52 kBq (> 18a)
Children and youths	2,4 ± 4,1 kBq	0,27 ± 0,18 kBq	1,6 ± 2,1 kBq	0,23 ± 0,19 kBq

Contribution of internal dose to population dose

The internal dose is only part of the population dose. Within the project also the contribution of the external dose has been evaluated. It could be shown that in the municipality of Volincy the major part of population dose is due to the external dose. Over time it seems to drop more slowly than the intermal dose (Fig. 9).

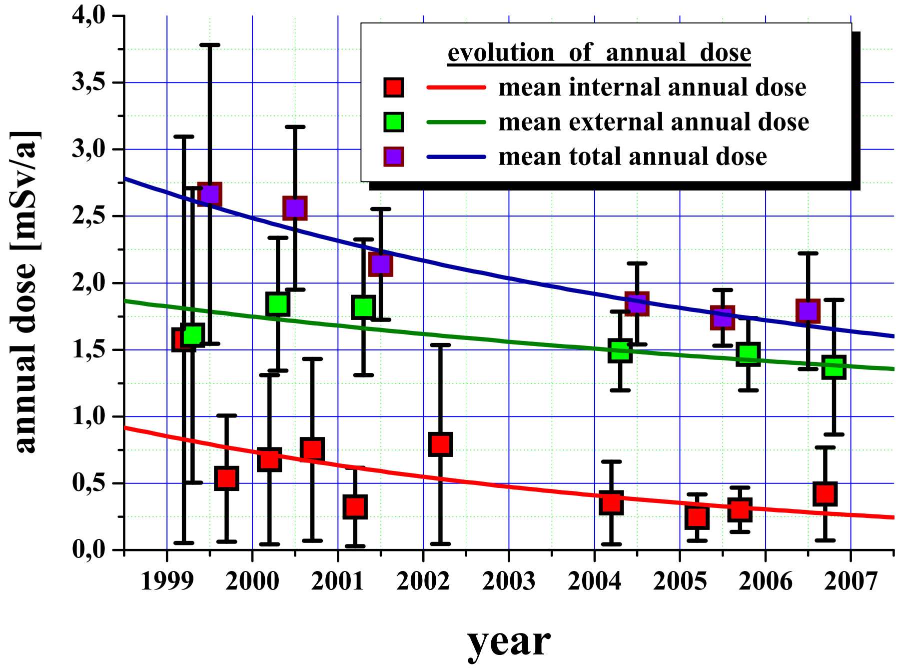

Fig. 9. Evaluation of annual dose at Volincy from 1999-2007 (Dederichs et al. 2009).

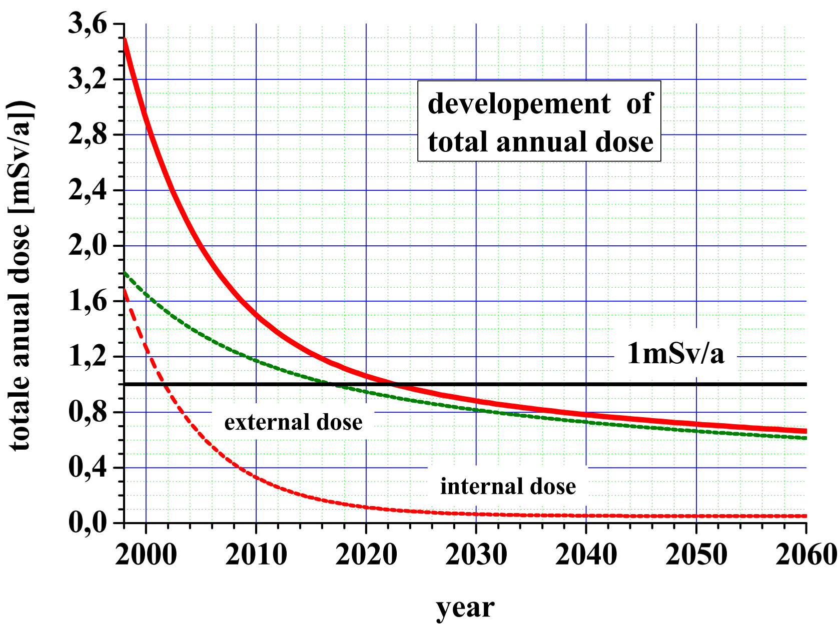

Fig. 10. The projected long-term development of population dose at Volincy (Dederichs 2009).

A model of the transfer of Cs-137 in the environment (Dederichs et al. 2004) has been applied to investigate the long-term development of population dose. The results are shown in fig.7. The mean annual internal dose (Cs-137) is likely to drop below 0.3 mSv in 2010. This value is comparable to the dose typically received from the natural body content of K-40.

In the years to come internal dose will gradually become negligible in relation to external dose, which already now governs population dose. It should be noted that the underlying estimation of external dose is rather conservative. Hence in the real world the population dose might drop below 1 mSv/a already before the year 2022 as forecasted by the model.

Conclusions

Generally the mean incorporation dose is about to drop below 0.3 mSv/a , a value comparable to the dose obtained by the body content of K-40. This was the result of the decreasing environmental contamination (top soil and crops) and specific advice offered to the population.

Presently the internal dose is only slightly enhanced over background values and of no special relevance to health. It is expected to decrease to less than 0.2 mSv/a in 2011 and to below 0.1 mSv/a in 2020.

Acknowledgements

The funding by the German Federal Office for Radiation Protection and the Walter-Gastreich-Trust are greatfully acknowledged. We are indebted to J. Höbig, W. Marquardt, H. Mertens and H. Preiß, who took part in the measurements.

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A real-time interaction environmental survey management system for nuclear and radiological emergencies

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Abstract

In emergency events, environment could be polluted very severely and need a variety of environmental surveys. The environmental survey workers may be exposed to radiation and radioactive pollution. There is a need to confirm these survey workers are safe when they are working for emergency response.

This study attempts to establish a management system with real-time interaction for environmental monitoring, which could analyze monitored environmental quality immediately, and manage the plans and areas of monitoring action online. The plans and areas of monitoring action could be adjusted interactively according to the managers' decision and monitoring results of environmental radioactivity. If the analyzed monitored results of the environmental quality exceeds a threshold standard, it can be an index of warning. Another purpose of this study is to provide a management system which could provide a safe area of monitoring actions with real-time interaction for environmental monitoring teams. The management has a data processing apparatus which could analyze the monitored data of the monitoring apparatus and send warning signal when it found that the monitored data exceeds threshold value. Then the monitoring teams and personnel could execute the work in safety.

Introduction

After nuclear power plant accidents at Three Mile Island and Chernobyl, Taiwan has intensified the efforts on the works of radiological emergency response and preparedness as other countries. Institute of Nuclear Energy Research (INER) belongs to Atomic Energy Council, Taiwan. One of INER's important mission is to continuously develop the tool for nuclear and radiological emergency response and research the effective strategy and police. However, the 911 event in 2001 in USA made us realize that the environmental survey for radiological emergency should be more active than conventional methods because of the increasing uncertainty on locations, environments, source terms, and occasions of radiological emergency events.

INER launched a project for establishing a mobile environmental survey system for Taiwan in 2003. At that time, Internet and geographic information system (GIS) have become the essential technologies of radiological emergency response that should be good at monitoring and data management for communicating and decision making [Jaromir Kolejka et al. 2002; NEA 2000]. We also noticed that GPRS (General Packet Radio Service), one of early stage wireless data internet services provided by GSM (Global System for Mobile communications) network, was in service in Taiwan and thought it can help submitting data real-time for environmental survey. Finally, we established a mobile survey system for the survey task of radiological and nuclear emergency in 2004. The mobile survey system was tested by encircling Taiwan Island mission. There was no monitoring data lost in the mission that shows very good network stability. After the successful mission, the system has been used in the radiological bomb emergency and nuclear emergency exercises in recent years.

There are more and more attentions paid on the safety and health of first responder. U.S. Department of Homeland Security (DHS) issued ”Planning Guidance for Protection and Recovery Following Radiological Dispersal Device (RDD) and Improvised Nuclear Device (IND) Incidents HS Planning Guidance provided a summary of radiation emergency worker guidelines (DHS 2008). Following that, “Planning Guidance for Response to Nuclear Detonation” was developed by U.S. Homeland Security Council (DSC 2009). This guidance emphasizes that Incident Commanders should make every effort to employ the “as low as reasonably achievable” (ALARA) principle when responding to an incident. Protocols for maintaining ALARA doses should include the following health physics and industrial hygiene practices.

The assessment error of evaluating contamination distribution for emergency response is usually very high comparing to real monitoring data. The error could lead first responder to a dangerous situation and exposure an unnecessary radiation dose. This study attempts to establish a management system with real-time interaction for environmental monitoring, which could analyze monitored environmental quality immediately, and manage the plans and areas of monitoring action online. The plans and areas of monitoring action could be adjusted interactively according to the managers’ decision and monitoring results of environmental radioactivity. If the analyzed monitored results of the environmental quality exceeds a threshold standard, it can be an index of warning. Another purpose of this study is to provide a management system which could provide a safe area of monitoring actions with real-time interaction for environmental monitoring teams. The management has a data processing apparatus which could analyze the monitored data of the monitoring apparatus and send warning signal when it found that the monitored data exceeds threshold data. Then the monitoring teams and personnel could execute the work in safety. We hope this study could give first responder more safety and health.

Material and methods

The primary working framework of the mobile survey system is shown in fig 1. The system consists of the mobile unit and the monitoring center. The mobile unit and the management center are connected by the wireless data service network (GPRS, 3G, HSDPA, et al.). The mobile unit was operated by a PDA initially, but it has been extended to could be operated by smart phone or notebook now. The monitoring center is operated by a server with GIS and SQL database. The coverage of wireless data network service is very good in Taiwan. The monitoring data collected from remote place could be displayed on the screen of the monitoring center in 10 seconds normally in the encircling Taiwan Island mission.

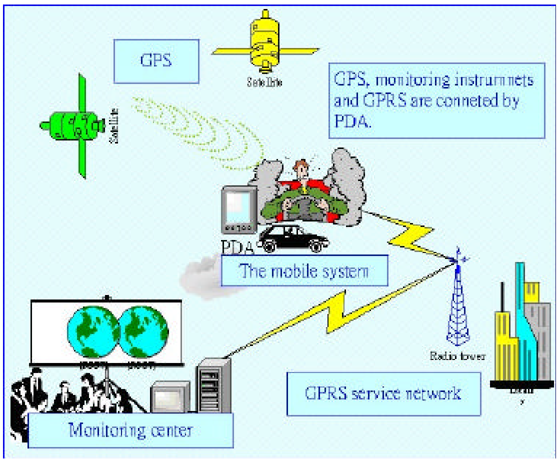

Fig. 1. The primary working framework of the mobile survey system.

The mobile unit could receive the survey mission map in a GIS electronic map format. Then the environmental survey workers execute environmental survey according to the map. The map also could include the GIS information of survey mission and dangerous areas and their expected survey data values. The warning signal of the mobile unit will be initiated when it gets into any dangerous area or the survey data value exceeds the preset limit of the specific areas. The preset limit of the specific areas can depend on the assessment results or regulation. The monitoring center can update the survey mission map according the new survey results and send the new mission map to the mobile unit through wireless data network immediately. The procedures could be explained by Fig. 2. The system could check the position data and the monitoring data automatically. It also could calculate the error between the monitoring data and the prior assessment if it found the monitoring data did not comply with the tolerance limit of the prior assessment. When the monitoring data exceed the preset limits, the monitoring center will send the warning message to the mobile unit and display is on its screen. The update of the mission map is determined by the survey manager.

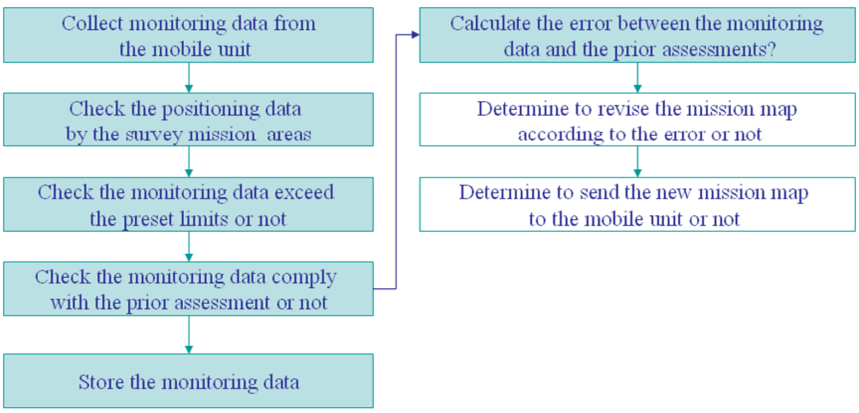

Fig. 2. The procedures for updating the mission map.

The mission map needs to be revised by the survey manager's professional judgements before sending to the mobile unit. The revision mainly depends on the error. The system provides a function to get the correction factor from the error to adjust the preset value of the survey mission map. The manager could determine the final value of the correction then the system will change its survey mission map automatically. The survey manager could examine and investigate the survey the survey mission map before pushing the bottom to send the new survey map to the mobile unit. Figure 3 is the screen of the mobile unit showing the survey mission map and its position.

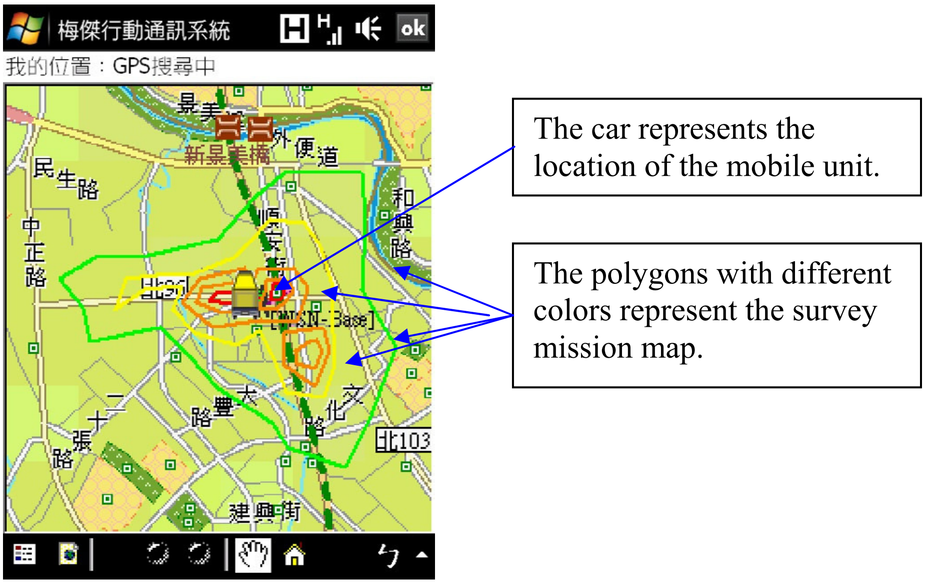

Fig. 3. The mobile unit and the survey mission map.

Discussion

The development of information and communication technologies has provided opportunity and stress. INER has established a mobile survey system that can really support environmental surveys, especially for nuclear/radiological emergency response. The benefits include:

- ✧ Visulization of showing the survey results;
- ✧ Imporving the ability of integrating survey data and relative information;
- ✧ Higher quality and quantity for collectind survey data;
- ✧ Inquiring survey data and making reports sooner;
- ✧ Interactions between the field survey workers and the survey manager;
- ✧ Better ability to reanalyze the data (fig. 4).

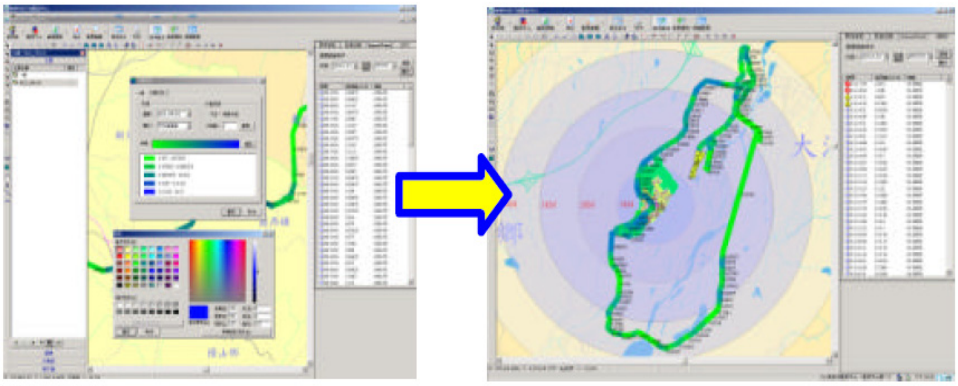

Fig. 4. Reanalyzing and visualizing survey results by classification with colours.

The protection of survey worker for emergency response has become a very important issue recently. However the environmental situation could change very quickly. A real-time interaction environmental survey system could be a very effective and helpful tool to the issue. That is a major purpose of developing our mobile survey system. The mobile units can get the survey mission map from the monitoring center before the survey workers go to the field of the nuclear/radiological events. The survey workers execute the mission according to the survey mission map. When the mobile unit arrive in the target area, the survey data will be sent back to monitoring center continuously. The mobile unit also could judge the position of the survey relative to the specific area of the survey mission map and send the relative message to the survey worker, including dangerous warning message. So the survey worker could be protected according to the initial assessment result. After collecting survey data, the survey manager could real-time revise the survey mission map through wireless network. Then the survey workers could be protected according to the real situation.

Conclusions

There are many ways that modern technologies could benefit environmental survey. INER has established successfully a mobile survey system by integrating modern electronic map, positioning and wireless data network that can really support environmental surveys, especially for nuclear/radiological emergency response. The protection of survey worker for emergency response has become a very important issue recently. A real-time interaction environmental survey system could be a very effective and helpful tool to the issue. The mobile survey system developed by INER could be a very effective and helpful tool to the issue.

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Detection of and response to nuclear security events

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Abstract
The paradigm of nuclear security continues to evolve. Whilst a body of work and a collective with relevant expertise is forming, the relationship and interconnectedness with the fields of nuclear safety and nuclear verification (safeguards) still needs day-to-day management to ensure States have in place an effective judicial, legislative, regulatory and policy framework which, when supported by coordinated implementation procedures, will assist in the prevention of malicious acts upon nuclear and/or associated facilities, as well as reducing the threats posed from the malicious use of lost, missing or stolen nuclear or other radioactive material to harm persons, property, society or the environment.

This is particularly the case in the detection of and response to malicious acts involving nuclear and other radioactive material out of regulatory control. Here the States' traditional response arrangements need to be enhanced and coordinated to ensure that threats posed by criminal or unauthorised acts involving such material, and any dispersal event occurring as a consequence of a successful malicious act, are adequately dealt with in terms of both the need to detect the presence of the nuclear or other radioactive material in the lead-up to a nuclear security event, as well as the proper collection and control of evidence to ensure successful prosecution of the offenders consistent with the relevant international legal instruments to which States may subscribe. This paper outlines these additional considerations.

Introduction
The IAEA maintains an Illicit Trafficking Database (ITDB) which contains confirmed reports of criminal or unauthorised acts involving nuclear or other radioactive material. The ITDB currently contains 1884 confirmed reports from the 110 States that participate.

While the circumstances of each ITDB case vary significantly, some credible reports indicate that criminal organizations have shown interest in nuclear material and facilities. The threats include criminals or terrorists acquiring and using nuclear material to build a rudimentary nuclear explosive device (in the worst case, an existing nuclear explosive device); using a radioactive source as an exposure device or achieving the dispersal of radioactivity by the construction of a radiological dispersal device; or through an act of sabotage at a nuclear facility or other installations or place where radioactive substances are used, stored or transported.

Since the consequences of the use of an improvised nuclear explosive device would be catastrophic, such events cannot be neglected although their probability may be much lower compared to the potential of an RDD or an act of sabotage. Intentional dispersal of radioactivity would, no doubt, have very significant psychological, health and economic consequences.

In responding to the threat the international community has developed a large range of international legal instruments, both binding and non-binding, which taken in their entirety outline the field of nuclear security best practice for States to consider in their ongoing efforts to combat nuclear terrorism. Although many of the international legal instruments were originally drafted to principally address non-proliferation or safety concerns they do, when combined with the newer counter-terrorism instruments originating principally after Sept 2001, contribute to the global framework that defines nuclear security. The primary instruments and documents include:

- Convention on the Physical Protection of Nuclear Material [CPPNM] and its Amendment of 2005 [1],
- International Convention for the Suppression of Acts of Nuclear Terrorism[2],
- United Nations Security Council Resolution 1540 [3]
- Code of Conduct on the Safety and Security of Radioactive Sources [4]

In the broad sense, these instruments deal with nuclear security issues, and in particular they are all related to malicious acts at the sub-national level (i.e. by non-state actors), involving nuclear or other radioactive materials. Combined these instruments require States to, inter alia:

- Criminalize, i.e. make punishable under national law certain offences, such as malicious acts and threats
- Make every effort to establish measures for preventing or protecting nuclear material and facilities and other radioactive material against such acts, and
- Take appropriate actions should such malicious acts occur. In this regard, preparations should include the recovery of material, its return and assistance and cooperation among States and the IAEA.

While recognising the sovereign right of States over matters of security, the IAEA has responded to requests to provide guidance that States can use to meet their committments and obligations with respect to these international legal instruments. The draft versions of this 'top-teir' guidance include:

- Nuclear Security Fundamentals [5]
- Nuclear Security Recommendations on Nuclear Material and Nuclear Facilities (INFCIRC/225/Rev.5) [6];
- Nuclear Security Recommendations on Radioactive Material and Associated Facilities [7]; and
- Nuclear Security Recommendations on Nuclear and Other Radioactive Materials out of Regulatory Control [8].

While the nuclear security series covers the entire spectrum of recommendations to a State in developing and sustaining a nuclear security regime, of particular interest in the inter-relationship between nuclear safety, nuclear verification and nuclear security is the detection and assessment of alarms and alerts and for a graded response to any criminal or intentional unauthorized acts involving nuclear or other radioactive material out of regulatory control. These ‘detection and response’ elements of a nuclear security regime cover the confirmation of a credible threat, assessment and interdiction of an attempted act and response to a nuclear security event and are contained principally in the latter of these documents.

Background

The draft Nuclear Security Recommendations on Nuclear and Other Radioactive Materials out of Regulatory Control document (the draft document) was prepared by the IAEA Secretariat in consultation with experts drawn from Member States, and has been the subject of a Technical Meeting in March 2010. The comments from the Technical Meeting were incorporated, as appropriate, into the draft document which is currently out for 120 day Member State review.

The summary of the more important contents of the draft document are given below, but may be subject to change prior to publication once Member State comments are received. In essence the draft document recommends that for the State to have an effective nuclear security regime for nuclear and other radioactive material out of regulatory control, they need to ensure the existence of two main elements, namely:

- A comprehensive and complete set of legislative provisions through adoption of criminal and administrative laws for providing relevant administrative and enforcement powers to the various competent authorities within the State, so that they can undertake their activities in an effective manner; and
- Provision of sufficient and sustained resources to the various competent authorities to enable them to carry out their assigned functions, including:
    - Measures to prevent a criminal or an intentional unauthorized act involving nuclear and other radioactive material out of regulatory control;
    - Detection, through an instrument alarm and/or an information alert, of the presence or indications of a criminal act or an unauthorized act with nuclear security implications involving nuclear or other radioactive material that is out of regulatory control and, in particular to:
        - develop a detection strategy;
        - establish detection systems; and
        - perform the initial assessment of the instrument alarms and information alerts promptly and assess the possibility of a nuclear security event.
    - Response to the nuclear security event, in particular to:
        - notify the competent authorities
        - assess the validity and potential consequence of the nuclear security event;
        - locate, identify, categorize and characterize nuclear or other radioactive material;

- secure such material and apply other response measures appropriate to the nuclear security event, such as neutralization of the device;
- recover, detain and/or seize and place such material under regulatory control;
- collect, preserve, store, transport and analyse evidence including the application of nuclear forensics measures related to a criminal act or an unauthorized act with nuclear security implications that involves such material; and
- apprehend and subsequently prosecute or extradite alleged offenders.

Legislative, Regulatory, Policy and Administrative Controls

The draft document recommends that, as part of an overall regime, the State should establish and maintain effective legislative and regulatory controls to govern nuclear security, including those that define any conduct which they consider to be a criminal act or an intentional unauthorized act involving nuclear and other radioactive material, and establish such offences as criminal offences under domestic law.

For effective and sustainable detection and response measures, it is important to rely on multidisciplinary infrastructures implemented by several independent competent authorities in the State, and to ensure proper cooperation, coordination, information exchange and integration of clearly defined activities and responsibilities within some form of coordinating body or mechanism

The following list of organisations could have some role in, or may be a competent authority for aspects of the nuclear security regime within a State. Consequently they should, if appropriate, be considered for involvement in this coordinating body or mechanism:

- Judicial bodies;
- Legislative authorities;
- Policy authorities;
- Intelligence services;
- Military forces;
- National threat assessment bodies;
- Law enforcement bodies;
- Border Forces;
- Customs authorities;
- Police;
- Regulatory bodies;
- Medical and/or health authorities;
- National nuclear energy agencies;
- Civil Defence; and
- Emergency Services.

The draft document recommends that the coordinating body or mechanism should inter alia:

- ensure the development of a comprehensive national detection strategy based on a multilayered defence in depth approach within available resources;
- ensure development of a national response plan for any nuclear security event in a graded approach commensurate with the threat and based on available resources;
- oversee the development and implementation of the national detection and response systems;
- re-evaluate and identify possible nuclear security gaps and resource needs and initiate proper corrective actions on a regular basis;
- ensure the establishment of contact points within the competent authorities as part of an overall coordination within the State,
- encourage the timely sharing of operational information among competent authorities within the State;
- ensure the establishment and maintenance of a reliable and comprehensive set of records for each nuclear security event, and encourage the exchange of information among competent authorities concerning any such event, using a common reporting and notification format; and
- ensure appropriate coordination and cooperation with relevant authorities in other States and international organizations.

The functions of the competent authorities as outlined in the draft document should include, inter alia:

- contributing to the development of the national detection strategy and response plan;
- developing, operating and maintaining the national detection systems, assessment procedures and the national response plan and providing the resources necessary for implementing and testing the associated activities;
- providing adequate training and information to all personnel involved in carrying out nuclear security detection and response measures;
- sustaining the detection and response capabilities and ensuring operational preparedness through sound management practices, addressing instrument maintenance, personnel training, exercises and process improvements; and
- cooperating with the coordinating body, other competent authorities and bilateral and multilateral counterparts as applicable, in part to ensure the effectiveness of their detection and response procedures and responsibilities.

The competent authorities would cooperate in the exchange of relevant information on the nuclear and other radioactive material that is authorised or 'under regulatory control', with a view to strengthening the capabilities of all concerned with nuclear security. Where appropriate, they should also cooperate with their counterparts in other States.

The draft document recommends that regulatory authorities should take appropriate actions when nuclear or other radioactive material is reported to be out of regulatory control, i.e., lost, missing or stolen. In particular, they should inform promptly the other competent authorities in the event of a suspected criminal or an intentional unauthorized act.

In addition the draft document recommends assigning priorities and designing the detection and response systems based on a national threat assessment, using a risk based approach in combination with an assessment of the:

- vulnerability to a criminal act, or an unauthorized act with nuclear security implications, both within and outside their borders;
- relative attractiveness of identified targets to a nuclear security threat;
- possible consequences of a criminal act or an unauthorized act with nuclear security implications, that involves the use of nuclear or other radioactive material; and
- possible evolution of the threat or vulnerabilities

Detection Measures

Detection of nuclear and other radioactive material that is out of regulatory control can be achieved via an instrument alarm or an information alert. The draft document recommends that the State should design and implement nuclear security systems based on such indicators, and ensure that the detection measures are supported by effective response measures

In order to prevent illegal transfer of nuclear or other radioactive material and detect the falsification of relevant documents, it is recommended that competent authorities have the power to adopt measures for authenticating documentation and package labelling for authorized shipments and for verifying the declared content of the authorized shipment of nuclear or other radioactive material by appropriate means

Using the national threat assessment, the draft document recommends that competent authorities should establish nuclear security systems for detection by instruments of nuclear and other radioactive material that are out of regulatory control. The detection systems should be based on a multilayered defence in depth approach and on the premise that such material could originate from both within or outside the State, and provide the necessary detection capability and capacity.

To assist in this regard, the draft document recommends that by while taking into account the prioritization of available resources, the competent authorities should develop an appropriate detection instrument deployment plan, considering the following:

- transportation routes inside the State's territory, at locations where likelihood of detection is maximized or in proximity to locations where nuclear or other radioactive material is produced, used, stored, consolidated or disposed;
- the existence of any strategic location;
- operational and detection performance specifications of the detection instruments, in accordance with national and international standards and technical guidelines;
- capabilities, constraints and limitations on detection instruments at both officially designated and non-designated air, land and water border crossings points;

- mobile and relocateable detection systems to provide flexibility and adjustments to evolving threat;
- detection requirements in support of law enforcement operations associated with information alerts; and
- detection of radiation at an event of national significance, such as a major public event or at a strategic location that is considered to be vulnerable to a malicious act using nuclear or other radioactive material.

Further the draft document recommends that the competent authorities should ensure that the following elements are included in the instrument deployment plan:

- initial installation, calibration, and acceptance testing of equipment; the setting up of a maintenance procedure, and the adequate training and qualification of users and technical support staff;
- systems and procedures for conducting a radiation survey or a radiation search for nuclear and other radioactive material out of regulatory control;
- defining threshold levels of an instrument alarm;
- establishing systems and procedures for performing initial alarm assessment and other secondary inspection actions such as localization, identification, categorization and characterization of nuclear and other radioactive material, including obtaining technical support from experts to assist in the assessment of an alarm that cannot be resolved on site; and
- provision and sustainment of supporting infrastructure to ensure effective detection, including personnel training, equipment maintenance, safe and secure disposition of discovered material and documented response procedures.

While the use of instruments is a major method of detecting potential malicious acts involving nuclear and other radioactive material, another major component of detection measures is the detection by the use of information. Here the draft document recommends that the State should continuously gather, store and analyse operational information with the goal of identifying any threat, suspicious activity or abnormality involving nuclear or other radioactive material that may indicate the intention to commit a malicious act within the State. The draft document also recommends cooperation with other States to provide and obtain information for better understanding of any threat, and the development of a policy on the dissemination of information to the news media with the aim of informing the public of lost, missing or stolen nuclear or other radioactive material so as to educate them in the risks associated with the material and to elicit information from the public about such material, taking care not to cause undue public concern.

As part of the information led detection measures the draft document recommends that procedures and protocols are implemented requiring health professionals, medical institutions and health authorities to immediately report the occurrence of any suspicious radiation injuries or illnesses to the relevant competent authorities, in accordance with domestic public health reporting policies. Such collection and analysis of information from medical surveillance as part of detection measures should, as appropriate, be reported and investigated by relevant competent authorities to determine the cause and consequence of the injury or illness.

It is further recommended that the competent authority with regulatory responsibility should require authorized persons to report immediately any non-compliance which they suspect could have nuclear security implications. Such a report would enable the competent authority to assess the event with the aim of preventing a consequent malicious act. Further the draft document recommends that any competent authority that receives a report that such material has been reported as lost, missing or stolen, promptly inform other relevant competent authorities.

Finally the draft document recommends that any instrument alarm or information alert should lead to the conduct of an initial assessment. The relevant competent authorities should implement procedures and protocols with the view to interdict and interrupt the potential criminal act or unauthorized act with nuclear security implications.

Response Measures

To ensure an adequate response to the detection of nuclear of other radioactive material out of regulatory control and that could be used for a criminal or intentional unauthorised act, the draft document recommends that, using legislative instruments as necessary, the State develop a comprehensive national response system for responding to such acts.

In particular the draft document recommends that the State should ensure that the responsibilities for implementing the various response measures are assigned to the relevant competent authorities, together with sufficient resources to effectively undertake these tasks.

The implementation of the response system of the State should be documented in a national response plan outlining the various response measures, and should be implemented coherently by the various competent authorities, ideally coordinated by the coordinating body. It is important that in responding to nuclear security events, the responsible competent authorities should complement and support the safety emergency response activities to mitigate and minimize the radiological consequences to human health and the environment at the international, federal, state and local levels. The coordination of competent authorities is vital for an effective response at the scene.

An important aspect of this response, as outlined in the draft document, is the recommendation that the State should adopt a graded approach to respond to the various possible nuclear security events and differing degrees of consequences. In order to determine the appropriate response and follow-on actions, the State should strive to develop its own national capability to quickly grade nuclear security events, based on health and safety concerns and on circumstantial factors and the involved nuclear or other radioactive material.

If the initial assessment described in the detection measures above are not conclusive, the draft document recommends that the relevant competent authorities ensure the establishment of procedures and protocols for final resolution of an instrument alarm which may result in the determination that a nuclear security event has occurred. The determination of a nuclear security event should lead to the activation of the national response plan by the relevant competent authority using the graded approach discussed above. For the assessment of information alerts, the competent authorities should obtain the necessary assistance from the assigned experts and the support organizations, in accordance with the established procedures and protocols.

The location of any nuclear security event should be managed as a potential crime scene. The draft document recommends that the competent authorities should ensure coordination among those involved in recovering control over the nuclear or other radioactive material, those concerned with safety and treating victims and those concerned with gathering evidence for possible subsequent investigation and prosecution.

It is important for the State to apply nuclear forensic techniques on any seized nuclear or other radioactive material in its designated laboratories for the purpose of identifying the source, history and the route of transfer, taking into account the preservation of evidence. Furthermore, traditional forensics should also be applied in designated laboratories for contaminated evidence, as necessary.

The draft document recommends that persons involved in the response should be suitably qualified and trained and should, as appropriate, be aware of the basic concepts of radiological crime scene management, evidence collection and radiation protection.

In order to manage the nuclear security event, the draft document recommends the establishment of a comprehensive national response plan (the Plan) in combination with, inter alia, the national radiological emergency plan. The Plan should serve as:

- A basis for establishing compatible operational tools needed for prompt and effective response; and
- A guide for the competent authorities who should ensure that all necessary preparedness and response tasks are given the appropriate resources and support.

The draft document recommends that the Plan:

- describes the process for various competent authorities to fulfil their obligations and responsibilities in response to nuclear security events, including steps to:
    - notify and activate all relevant competent authorities;
    - notify the relevant international organizations and potentially affected States;
    - coordinate various organizations and command and control units of a nuclear security event, including coordination of federal, state and local response organizations;
    - locate, identify and categorize nuclear and other radioactive material;
    - detain and/or seize, recover and control material or render harmless any threat or associated device;
    - collect, secure and analyse evidence;
    - isolate, classify, package and document, any nuclear or other radioactive material, for transport, carriage, storage or disposal and placement under proper regulatory control; and
    - initiate relevant investigations.
- contains an appropriate command structure with integrated command, control and communication systems to effectively respond to a nuclear security event, preferably with a single person or competent authority assigned to direct the response at the scene;

- has provisions for coordination among the competent authorities, including exchange of relevant information concerning their respective roles, responsibilities and procedures;
- describes the roles, responsibilities and procedures for the competent authorities for medical services, handling of hazardous material, radiation protection and safety and other technical support organizations and for nuclear and conventional forensic laboratories;
- arranges for informing the news media and public, as appropriate, in a coordinated, understandable and consistent manner;
- contains provisions for the transport of any seized or recovered nuclear or other radioactive material in accordance with the national transport safety and security regulations and requirements;
- identifies the standard operating procedures at the local level for nuclear security events. In addition, all local level response plans should be integrated into the Plan;
- takes into account, and coordinates with, the existing national radiological emergency plan and radiological emergency response procedures;
- incorporates the possibility of multiple and simultaneous nuclear security events. In addition, the Plan should incorporate the possibility of disruption of response infrastructure that would delay an effective response capability; and
- incorporates the mechanisms for requesting assistance, both domestically and internationally, when necessary, such as assistance for the recovery of nuclear and other radioactive material, rendering harmless the device and nuclear forensics.

The draft document further recommends that, upon detection of nuclear or other radioactive material out of regulatory control at a border crossing point, the State should work with the State of origin and other relevant States to return the material to regulatory control. The State should adopt a graded approach for such response that depends on the circumstances of the case and the nature of the material.

Nuclear forensics techniques to determine the source and route of transfer and to investigate loss of regulatory control are an important part of nuclear security. The draft document suggests that investigations may entail cooperation between or amongst States to identify the origin, history and the route of transfer of the nuclear or other radioactive material. Cooperation on nuclear forensics should be subject to the State’s domestic laws, regulations and policies and the draft document recommends that for States without sufficient nuclear forensics expertise and capabilities they enter into arrangements with other States or relevant regional or international institutions for the purpose of nuclear forensics analysis and interpretation.

Perhaps the biggest deterrent to the use of nuclear and other radioactive material for criminal and other intentional unauthorised acts involving nuclear and other radioactive material is the existence of comprehensive nuclear forensics libraries. The draft document recommends that a State should consider establishing nuclear forensics libraries for their inventory of nuclear and other radioactive material. These libraries should include databases of all material produced, used and stored in the State and, if applicable, supported by sample and literature archives. The State should be capable of responding to queries of other States regarding recovered nuclear or other radioactive material that may have been produced, used or stored on the State's territory.

Conclusions

Nuclear security is a distinct and growing body of knowledge relating to the use of nuclear and other radioactive material for criminal or intentional unauthorised acts. Aspects of safeguards and safety are fundamental to a successful nuclear security regime in a State, particularly related to the accountancy of nuclear material, and the register of radioactive sources. For nuclear and other radioactive material out of regulatory control, the Secretariat and Member States of the IAEA have drafted a guidance document to assist them in establishing an effective nuclear security regime.

This draft document is currently with Member States for 120 day review and copies of the draft document are available on the Agencies website at: http://www-ns.iaea.org/security/nuclear.security.series.htm, and comments are welcome from all Member States. It is anticipated that once comments are received and reviewed, another Technical Meeting will be held at the IAEA to ratify the final version of the document, before initiating the publication process.

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New threats and new challenges for radiological decision support

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Abstract

It is described how ongoing work will extend European standard decision support systems currently integrated in the nuclear power plant preparedness in many countries, to enable estimation of the radiological consequences of atmospheric dispersion of contaminants following a terror attack in a city. Factors relating to the contaminant release processes, dispersion, deposition and post deposition migration are discussed, and non-radiological issues are highlighted in relation to decision making.

Introduction

Over recent years the world has become increasingly aware that malevolent acts involving atmospheric dispersion of radioactive matter may occur and could severely affect large urban populations, both by leading to high radiation doses from dispersed radionuclides and by causing social disruption and fear. A comprehensive and reliable decision support system that can be operated in real-time is essential to ensure that the repercussions are addressed optimally and consistently from the very beginning and seen in the context of the actual health hazards. The dispersion could be carried out in different ways, involving, e.g., 'dirty bomb' devices, simple aerosol generators placed on a rooftop, or emission from an aeroplane. Depending on both the dispersion process and the initial contaminant matrix, aerosols with very different size spectra and physicochemical characteristics can be produced. The importance of aerosolisation processes, atmospheric dispersion in complex urban terrain, and post-deposition contaminant solubility, fragmentation and migration are all discussed in relation to the dose modelling needed to form reliable consequence prognoses. The paper reports on how these issues are being dealt with in an extension of existing European standard decision support systems to cover the consequences of terror attacks. Also non-radiological perspectives of radiological terror attacks are discussed.

Methods and results

A very wide range of radionuclides could at least in theory be envisaged for use in a radiological dispersion terror attack. In reality, however, the list of radionuclides that would be of primary concern would be likely to have only of the order of ten entries. The limiting factors include availability of existing sources with sufficient strength, problems in handling strong sources, initial physicochemical form of sources, types of radiation emitted, energies and photon/particle yield, and physical half-life of the contaminant(s). There seems to be almost consensus that this list would include ^{60}Co, ^{137}Cs, ^{90}Sr, ^{192}Ir, ^{226}Ra, ^{238}Pu, ^{241}Am, and ^{252}Cf (Andersson et al., 2009), but of course authorities and planners should also keep an open eye for the unexpected, as one of the main targets of all terrorism acts is to provoke a feeling of unpredictability, unpreparedness and uncertainty, which can in itself contain a considerable potential for anxiety, distrust and social disruption. In this section, factorial dependencies of radiological consequences, as well as non-radiological implications, of malicious atmospheric radiological dispersion are discussed.

Aerosolisation, dispersion and deposition

There has been a tendency to focus on the risk that terrorists might detonate a so-called ‘dirty bomb’ radioactivity dispersion device. It is considered likely that a ‘dirty bomb’ attack would have maximum societal impact if detonated in a highly populated area (Andersson et al., 2009; Sohier & Hardeman, 2006). Unless the explosion leads to evaporation of the contaminants, followed by formation of small condensation particles, most of the contaminant particles generated by this type of explosions would be large (some of it will be spread ballistically rather than being aerosolised), but it is likely that there would also be a significant release of particles in the size range of only a few microns (Andersson et al., 2008). These latter particles could disperse with the wind over a rather large city area. The initial physicochemical form (e.g., metal, powdered, solution, ceramic) of the applied source can in general greatly influence the aerosolisation process and thus the dispersibility in the environment. Radionuclide compounds that might plausibly be applied range from the virtually insoluble to readily soluble, e.g., depending on their previous application (Andersson et al., 2009). The extent to which phase transition will occur in the explosion process, thus potentially leading to formation of smaller and highly dispersible aerosols, also depends on how successfully the bomb is dimensioned. However, experimentation has shown that phase transition does not occur in connection with contaminants on ceramic form (Harper et al., 2007). Aerosolisation spectra based on data from experimentation are being incorporated in an ongoing extension of the European standard decision support systems (DSS) to predict the consequences of ‘dirty bombs’. It should be stressed that the aerosol spectra that would arise after a ‘dirty bomb’ explosion would in many cases be completely different from those expected at some distance from a large nuclear power plant accident, and parameters applied so far in European DSS (ARGOS, RODOS) can thus not be applied for ‘dirty bomb’ scenario calculations.

One parameter that greatly influences the size of the area over which the contaminants are spread in a ‘dirty bomb’ scenario is the plume rise. This also determines the significance of plume interaction with environmental structures (e.g., buildings, trees). The initial rise of the contaminated material after the blast will occur due to buoyancy and initial momentum. As the cloud rises, its movement will cause turbulent mixing with non-buoyant ambient air. Deceleration through decreasing buoyancy will reduce the boundary turbulence to that of the ambient air, by which point the 'initial' plume will have been formed. In the new feature of the European standard DSS dealing with 'dirty bombs', the parameterisation of plume rise relates to several independent blast studies (e.g., data from a US blast test series conducted in 1963 to investigate the effect of accidental conventional explosions spreading radioactive material from a nuclear device). The plume rise strongly depends on the amount and type of explosive applied.

For modelling the subsequent airborne dispersion of the contaminants over the environment, it is important to apply methods that adequately account for the mechanisms governing the flow and deposition in relation to the given dispersion altitude and scale. Here a considerable degree of simplification is traditionally applied in decision support models, which are designed to describe the long-range transport of contaminants from high-altitude releases following large nuclear power plant accidents. Since the plume here generally passes well over the various environmental obstacles, the influences on the plume propagation in inhabited areas can be reasonably simulated through the use of different overall roughness and deposition rate parameters compared with, e.g., rural areas (Päsler-Sauer, 2007; Mikkelsen et al., 1984; Mikkelsen et al., 1997). However, investigations made over recent years have shown such simplifications to be problematic, when the contaminant dispersion partially takes place at street level, as would be the case following a 'dirty bomb' explosion. This is particularly true for scenarios involving shifting wind directions (Astrup et al., 2005). Here, higher resolution models are required to address the issues of plume interaction with and flow along obstacles in the inhabited environment. Therefore, a new atmospheric dispersion module for inhabited areas, URD (Urban Release and Dispersion), based on Gaussian puffs and a calculation grid with highly resolved buildings, has been developed at Risø-DTU. This new implement is in some of its basic features inspired by the UDM code developed by the UK Defence Science and Technology Laboratory (Hall et al., 2002), and incorporates plume interaction with environmental obstacles in three different ways. The obstacles form barriers limiting the magnitude of horizontal eddies in the atmosphere, which in turn reduces the large-scale horizontal dispersion. At the same time, the interaction will increase the small-scale turbulence over a city, which will lead to greater small-scale dispersion. Finally, obstacles like downstream building walls constitute barriers that will to some extent delay the further dispersion of parts of the contaminants (Fackrell, 1984).

Obviously, deposition of aerosols on the different surfaces in an urban complex also depends strongly on particle size. Small liquid or vapour condensation particles can be formed, which will have a low deposition velocity to surfaces in the environment. Fragmentation particles will have a considerably higher deposition velocity, leading to a more concentrated contamination pattern over a somewhat smaller area. For instance, the dry deposition velocity to a lawn is typically higher by a factor of about 30 for 20 μm particles than for 2 μm particles (see, e.g., McMahon & Denison, 1979), and due to the different processes (Brownian diffusion, impaction, interception, gravitational settling, etc.) governing the dry deposition of particles with different sizes on environmental surfaces of different materials and orientation, also the distribution of the contamination on the various surfaces in the inhabited environment will be highly dependent on particle size. This issue is also addressed in ongoing DSS extension, through the inclusion of a series of data libraries describing the deposition of aerosols in intervals of the relevant size ranges for five different deposition 'modes': dry deposition, deposition in light rain, deposition in heavy rain, deposition in snow, and dry deposition to a snow-covered environment.

As mentioned above, also other types of malicious atmospheric dispersion of radioactive aerosols may occur, and aerosol formation processes may be simple. Small particles could for instance be formed over longer time by a nebulisation arrangement, perhaps on a rooftop in a city. Figure 1 shows a particle size spectrum measured with a Berner low pressure impactor in the vicinity of an aerosolisation arrangement, where an injected air stream generated particles by nebulisation of an indium acetyl-acetonate powder dispersed in alcohol. A simple medical inhalator nebuliser was used for this arrangement. These particles have a very low deposition velocity, and could thus travel far in the wind. It is foreseen that also estimation of the consequences of this type of scenarios will be enabled in the extended European DSS systems.

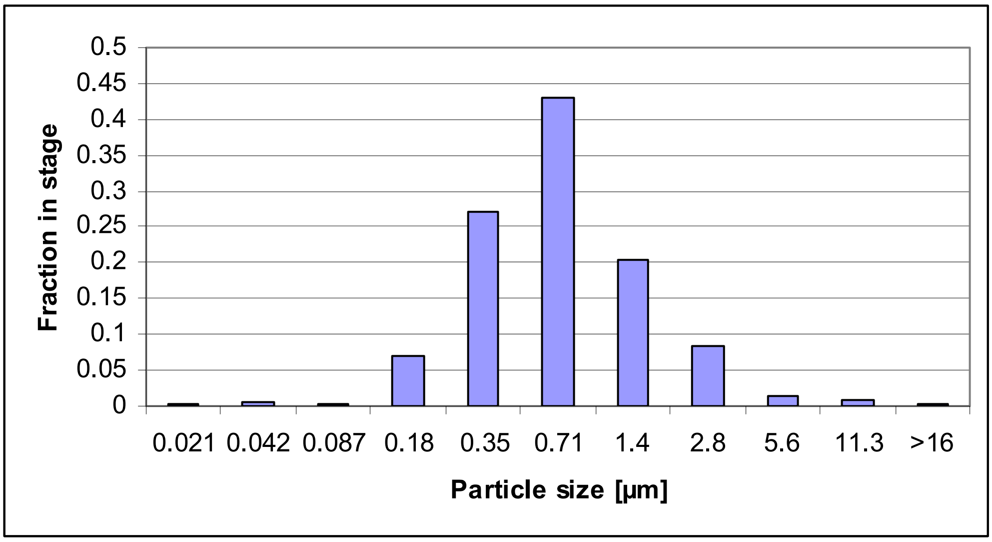

Fig. 1. An example of a size spectrum of aerosols generated by simple means through nebulisation.

Post deposition contaminant behaviour

With respect to post-deposition mobility, it is essential to distinguish between the different physicochemical forms of the deposited contaminants. For instance, aerosolisation of metals in a 'dirty bomb' has been reported to require phase transition, and the solubility and environmental behaviour of small particles created by condensation of evaporated contaminants would be expected to be very different from that of the larger particles generated by physical fragmentation of a virtually insoluble material. It is important to take into account that large particles are considerably easier to remove from surfaces in an inhabited environment, both by natural and forced processes, than are small particles and contaminants in solution. This can be illustrated by results obtained by hosing water at the same pressure on similar sandstone walls that had been contaminated by the Chernobyl accident, in Pripyat only about 3 km from the power plant, and in Vladimirovka, some 65 km away. In Pripyat, where much of the contamination was in the form of large and insoluble particles, the treatment removed some two-thirds of the caesium, but as far away as Vladimirovka, where the contaminants were primarily in the form of small, soluble condensation particles, only about one-fifth of the caesium could at the same time be removed (Roed & Andersson, 1996).

Kashparov et al. (2004) demonstrated that the dissolution in soil of deposited contaminant particles with high chemical stability could, depending on soil pH, be a process lasting over several years. This will delay the migration in soil of the contaminants initially present in a low solubility matrix. The current data in the European DSS describing the migration of contaminants in the urban complex is practically exclusively based on measurements of readily soluble ^{137}Cs from the Chernobyl accident (see Figure 2). The mechanisms that govern the fixation on most urban surfaces of radiocaesium on cationic form are highly element specific, and the values can therefore not be applied for scenarios where a different contaminant is of major importance.

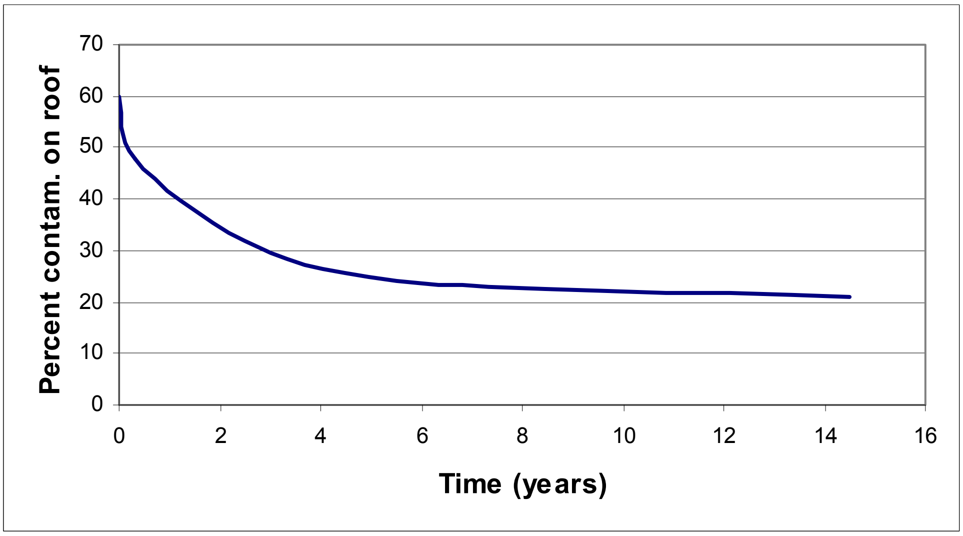

Fig. 2. Natural loss of Chernobyl radiocaesium on a clay roof through weathering over about 15 years (measurements made in Gävle city, Sweden; Andersson, 2009). The efficient long-term retention of a large part of the contamination is due to selective fixation of caesium cations in intact micaceous structures in the clay tiles.

It should also be noted that recent observations on contaminant particles from the Thule accident show evidence that natural spontaneous fragmentation of such low solubility particles can occur in, e.g., human body liquids. This means that large particles, which have high deposition velocities on human skin and in the human respiratory tract may be transformed into smaller particles that would have a longer natural clearance half-life. Thereby, the radiological consequences of the contamination would become more severe. This phenomenon requires further investigation.

Non-radiological concerns

Optimised decision making for intervention is an extremely complex process, which should take into consideration the full range of benefits and costs inflicted on the population as a whole as well as on population sub-groups. Not only radiological perspectives, but also a number of politically driven factors that can not be quantified on a generic scale will need to enter the decision matrix in the event of any contaminating incident. Such issues are often highly site and case specific, and can generally only to some extent be addressed in advance of a contaminating incident. Examples of factors that need to be included in a holistic justification/optimisation of intervention in the event of contamination of inhabited areas, but require quantification on the basis of political decisions, include the value of each unit of averted dose, the value of equity across the population, the value of public reassurance and psychological well-being, the value of maintaining societal functions, the value of lost income, the value of lost or damaged personal property, the societal value of preserving objects, and the value of avoiding environmental risks (Andersson, 2009). Involvement of citizens and stakeholders in the decision process for long-term restoration strategies is an important instrument in reaching generally acceptable and robust decisions that would not be prone to public resistance.

A number of tools have been created, which could be used to facilitate this process and help in balancing different types of factors against each other (Hämäläinen & Mustajoki, 2010; Belton & Stewart, 2002; Jackson et al., 1999; Zeevaert et al., 2001). The use of such systems in a participatory forum can, with the right facilitation and advisory support, form the basis for a case-specific ranking, which may seem reasonable and transparent to all involved, thus providing a useful platform for reaching agreement on the seemingly 'best' solution.. The weakness in the practical use of this approach lies in the valuing of weighting factors, and the summation over often many attributes to reach an estimate of the overall value of each countermeasure option may to some extent remove the focus from issues that demand undivided attention.

Specifically in relation to the psychological well-being of the affected population, it should be noted that both terror and accident victims may experience post traumatic stress disorder (PTSD) or acute stress disorder (ASD, occurring within a month of the impact). However, terror scenarios contain a number of psychological features that are different from those observed in connection with accidents. For instance trauma in connection with acts of terror can disrupt deeply held cultural assumptions about social values, and stressors of human design tend to produce specific senses of betrayal, blame and abandonment (O'Connor, 2009). A group that are at particular risk of developing lasting behavioural and emotional readjustment problems is the rescue workers, who very directly confront and witness the horrors of a terror impact. Therefore, contingency strategies should be developed carefully for this, and also other, population groups. Trauma triggers may include experience of life threatening danger or physical harm, bodily injuries, extreme violence and destruction, loss of communication, intense personal emotional demands, extreme fatigue, anticipation stress, and exposure to contamination (O'Connor, 2009). Specifically in connection with terror scenarios, 'information stress' can occur as people do not have a clue as to what is suddenly going on, and what the dangers might be, both in the very near future ('will more attacks follow soon?') and in terms of long-term consequences. Good communication strategies developed well in advance of an attack (including raising public awareness, since a well prepared public is less prone to panic) are essential in reducing the non-radiological impact. This is particularly important in connection with terror scenarios, due to their sudden and violent occurrence, which is not countered in other types of scenarios and leaves little or no time for any preparation (Danieli et al., 2005). Further discussions of non-radiological factors of concern can be found in Oughton & Forsberg (2009) and Andersson et al. (2009).

Conclusions

The above text illustrates how European decision support systems currently in operation for nuclear power plant preparedness will be extended to enable estimation of the radiological consequences of terror attacks involving atmospheric dispersion of radioactive matter. In relation to contaminant aerosolisation, dispersion and deposition, new source terms are being defined, together with release and deposition parameters that reflect the actual characteristics and dynamics of the contaminants originating from, e.g., ‘dirty bomb’ explosions. Also, a more refined urban aerosol dispersion model has been developed that takes into account the special features of dispersion at low altitude in a complex, inhabited terrain. It is demonstrated that also post-deposition migration parameters currently applied in the European decision support systems are inapplicable in describing the fate of contaminants from a radiological terror attack, and therefore new data libraries are being created for this purpose also. Finally, the implications of non-radiological factors for optimisation of intervention were discussed, and some factors relating to the psychological well-being of terror victims were highlighted.

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Nuclear inspections on container traffic in the port of Antwerp

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Abstract
Since 2007 Belgian Customs has been inspecting containerized shipments on the presence of nuclear smuggling with radiation portal monitors. Although no incidents of nuclear smuggling were found, several cases were found where non-natural radioactive materials were present in consumer goods.

Introduction
After the 9/11 events in the US an international programme against nuclear smuggling was started as part of the effort to tackle terrorism. The goal is to deter, detect and interdict the smuggling of special nuclear materials. One of the key aspects of this program is to equip border crossings with nuclear detection equipment.

In 2004 the Belgian government signed a memorandum of understanding with the US Department of Energy on the installation of detection equipment in the port of Antwerp. In February 2007 the inspections started in the port of Antwerp. In 2009 the seaport of Zeebrugge was also equipped. On a daily basis about 20 000 containers pass through the portal monitors in the Belgian ports causing about 150-200 alarms per day. Almost all alarms are due to the presence of NORM (naturally occurring radioactive materials).

One of the most important features of these inspections is to release containers with NORM or licensed radioactive materials in the shortest time possible. In order to minimise the effect of the inspections on the container flow, the inspections are performed in a standard three-phased approach.

The decision process is based on the Belgian regulations on radiation protection, international standards for nuclear inspections at borders, and experience from a research project with radiation portals in the port of Antwerp. [1-4] This decision process is part of a standard procedure, agreed on by Belgian Customs and the Belgian Federal Agency on Nuclear Control (FANC).

As a consequence Belgian Customs do not only look for the presence of nuclear smuggling in containers. In the decision process radiation protection concerns also play an important role, such as the safe transportation of radioactive substances (ADR class 7) and the incidental presence of radioactive material.

Material and methods

The Belgian Customs use a standard three-phased approach for nuclear inspections. In figure 1 this decision process is described in a general manner. For a more detailed standard of procedures please contact the authors.

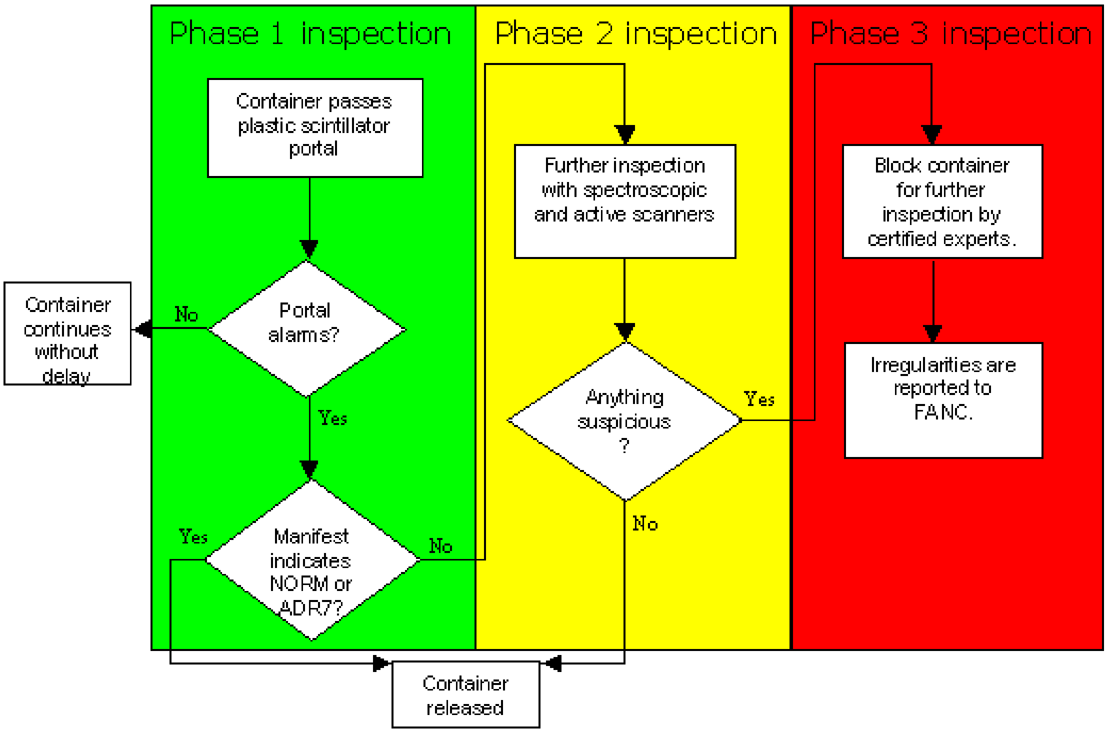

Fig. 1. Summary of the standard of procedures for nuclear inspections by Belgian Customs.

In each phase following procedures and equipment are used:

1. A primary inspection consists of a passage of the container through a radiation portal monitor equipped with both He-3 neutron detectors and plastic scintillator gamma detectors. If the portal detects an increase in radiation compared to the natural background, an alarm is raised. Containers that cause alarm are blocked and await a decision by a Custom Officer in charge. Decisions in this phase are based on the declared content of the container, which is compared to a database of known NORM materials. About 99% of the blocked containers are released in this phase because a known NORM substance is present inside. Note that licensed ADR 7 shipments are released without being blocked.
2. If the information regarding the content is not sufficient to release the container, a secondary inspection will be performed. This inspection consists of measurements of the gamma radiation present with spectroscopic equipment: (1) handheld Ge-detectors, (2) advanced spectroscopic portal monitors (ASP), or (3) car top systems. In most cases this is combined with an active scan of the container with X-ray scanners (3-6 MeV) in order to inspect the physical content of the

container. The X-ray scan image can identify smuggling scenarios such as lead shielding inside the container. In figure 2 the standard setup for secondary screening is depicted. If necessary these measurements can be performed with mobile spectroscopic systems, and a mobile active scanning unit.

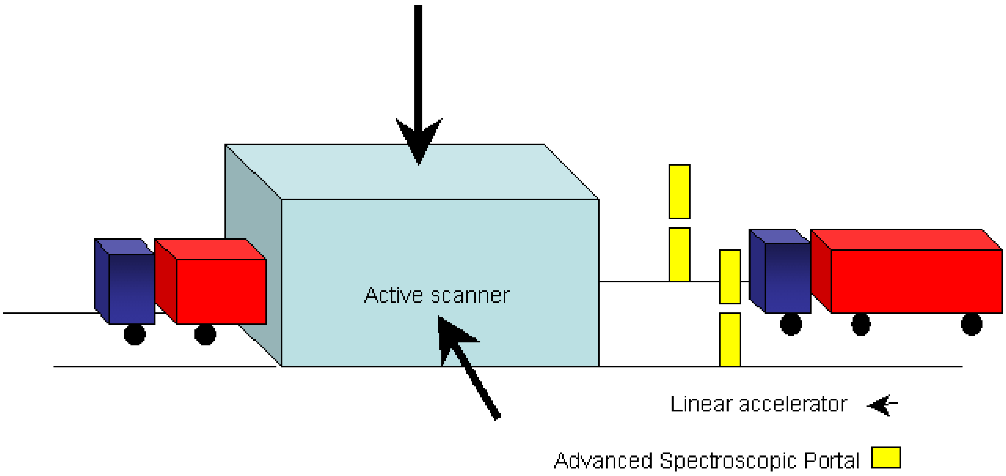

Fig. 2. Standard secondary setup in the port of Antwerp.

3. A tertiary inspection is performed in all cases were Customs cannot release the container based on the primary or the secondary inspection. In these cases a certified radiation expert or FANC itself takes over the inspection.

Results and discussion

The primary inspections with the portal monitors have a reasonably constant alarm rate, being about 1,4%. In the period 2007-2008 the installations were expanding in the port of Antwerp, resulting in growing numbers of occupancies. At present 43 portal monitors are installed in the port. In 2009 we experienced an effect of the global economic crisis resulting in a 7% decrease in occupancies.

Year	Number of Occupancies on primary portals	Number of alarms on primary portals	Alarm rate (%)
2007	1 447 386	20 501	1.42%
2008	2 868 650	40 991	1.43%
2009	2 659 259	36 785	1.38%

In most cases the primary inspection reveals the presence of NORM materials that are not subject to radiation protection regulations. The use of a NORM database has facilitated the recognition of NORM. At this point about 150 different classes of products, ores and consumer goods are present in this database.

The NORM database is used as a reference to release containers based on manifest information and alarm information. For each type of NORM an inspection limit is described. Above this limit Custom Officers will perform a secondary inspection and/or will demand more detailed information. Typically this information will consist of (1) the chemical composition of the product itself and (2) contact information of the manufacturer. The inspection limit is based on (1) statistical information taken from previous alarms on a certain type of NORM, (2) radiation protection regulations, and (3) expert judgement.

The number of secondary inspections has decreased over the years. Secondary inspections are performed on containers that cause suspicious alarms or alarms where the source of the radiation is not found after the primary inspection. At this point about one in 10 000 occupancies will result in a secondary inspection. The most important reason for this is the increased experience of Custom Officers.

Tertiary inspections revealed no cases of smuggling, but quite a large number of incidents involving contaminated materials and orphan sources. Typically one of the following scenarios arises:

1. Co-60 contamination of steel products due to the melting of a Co-60 source in a steel furnace.
2. In industrial by-products (example: concentrated metal ores) a contamination can be present, typically with Cs-137. This is due to an accidental processing of a radioactive source.
3. Cs-137 contamination of biological materials (including food) due to airborne Cs-137 after incidents (e.g. Chernobyl) or nuclear weapon tests.
4. TENORM (technically enhanced NORM) above exemption levels: e.g. radium paint on instruments, scaling on scrap…
5. Orphan sources in scrap.

Conclusions

Experience suggests that nuclear smuggling is not common. In the three years since the start of the inspections, no cases are reported in Belgium. On the other hand the installation of the equipment has proven to be a strong tool in protecting the public against the incidental occurrence of non-natural radioactive materials in consumer goods.

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Optical remote detection of alpha radiation

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Abstract
Alpha emitting radiation sources are typically hard to detect due to the short range of alpha particles in air. A remote detection of alpha radiation in air is possible by measuring the ionization-induced fluorescence of air molecules. The alpha-induced ultraviolet (UV) light is mainly emitted by molecular nitrogen and its fluorescence properties are well known. The benefit of this method is the long range of UV-photons in air. The main challenge of the optical method is to discriminate the weak fluorescence signal from the background lighting. The issue is addressed in the present paper by means of spectral filtering of the UV light. A portable demonstration device, utilizing spectral separation of fluorescence from the background lighting, is presented and the performance of the method is reported. Using specially selected room lighting, the device is able to detect a 1 kBq alpha emitter from the distance of 40 cm with one second integration time.

Introduction
Conventional alpha detectors require direct interaction with the particle which makes the localization of contamination a laborious task. Furthermore, alpha active nuclear materials pose a serious risk if they proliferate among rogue organizations. In this context, the research on novel alpha detection methods is well justified.

Previous studies have shown that remote detection of alpha radiation is possible by measuring the ionization-induced UV fluorescence of air molecules. It was shown that the UV fluorescence can be detected from a distance and even through a plexiglass of a clove box (Lamadie et al. 2005). Furthermore, the detection is possible even under a strong beta and gamma radiation background, because they do not induce as localized fluorescence as alpha radiation (Baschenko 2004). UV fluorescence is typically detected with photomultiplier tubes, but Lamadie et al. (2005) showed that it is also possible to use a CCD-camera and combine the fluorescence image with a normal photograph to gain position information. However, the camera detection requires very long integration times from several minutes to hours.

Fluorescence of air is mostly fluorescence of nitrogen, and molecular nitrogen has fluorescence peaks in the wavelength range between 300 nm to 430 nm. Table 1 shows the wavelength location of the most intense fluorescence peaks of nitrogen. The peaks are having a full-width of half-maximum of about 1 nm. The fluorescence properties of nitrogen have been intensively studied using electron excitation (Waldenmaier 2006), which might be the excitation mechanism also in the alpha radiation excited fluorescence. The secondary electrons are supposed to be responsible for the excitation of the fluorescence.

Table 1. Main fluorescence peak wavelengths of neutral (2P) and ionized (1N) nitrogen molecule. Peaks have full-width of half-maximum of about 1 nm. Two integers in parenthesis mark for vibration state in upper state and lower state, respectively.

Wavelength (nm) / transition
316 / 2P(1,0)
337 / 2P(0,0)
354 / 2P(1,2)
358 / 2P(0,1)
375 / 2P(1,3)
380 / 2P(0,2)
391 / 1N(0,0)

In this work, an optical detection method was developed with short integration time of about 1 second. The method is based on sensitive detection with photomultiplier tubes in photon counting mode, and spectral filtering of the fluorescence signal. The fluorescence peaks of nitrogen between 300 nm and 340 nm are used to record fluorescence signal, and the background reference is detected at 300 nm wavelength.

Material and methods

The optics for the demonstration device was designed with FRED Optical Engineering Software (Photon Engineering LLC). The software is based on optical ray tracing method to calculate the function and performance of the optics. A point like alpha emitter was modelled with the FRED software by assuming the alpha particles to travel 4.1 cm in air at maximum. During the alpha particle trajectory, it creates secondary low-energy electrons that are assumed to excite the fluorescence. Figure 1 (a) shows how the excitation sites are distributed around the point like alpha emitter. From these sites, the photons are emitted to random directions, as shown in fig. 1 (b). Thus, the point like alpha emitter forms a fluorescence emitting volume around itself. As the fluorescence source is not a point source, it causes some challenges to the optics and spectral wavelength separation.

The aim of the optical design is to collect as much of the emitted photons as possible. This yields to a large numerical aperture optics i.e. large collection angle for the optics. With the large collection angle and the volume emitter source, the spectral wavelength separation cannot be made by conventional spectrographs without losing the most of the photons. Thus, interference filters was selected for the demonstration device. The fluorescence is collected with a 40 nm bandpass filter (Semrock, Inc.) having the center wavelength at 320 nm. The effect of background lighting was detected by combining the fluorescence filter with a 15 nm bandpass filter having the center wavelength at 295 nm, which effectively limits the background detection to a narrow wavelength range of 299 nm – 303 nm.

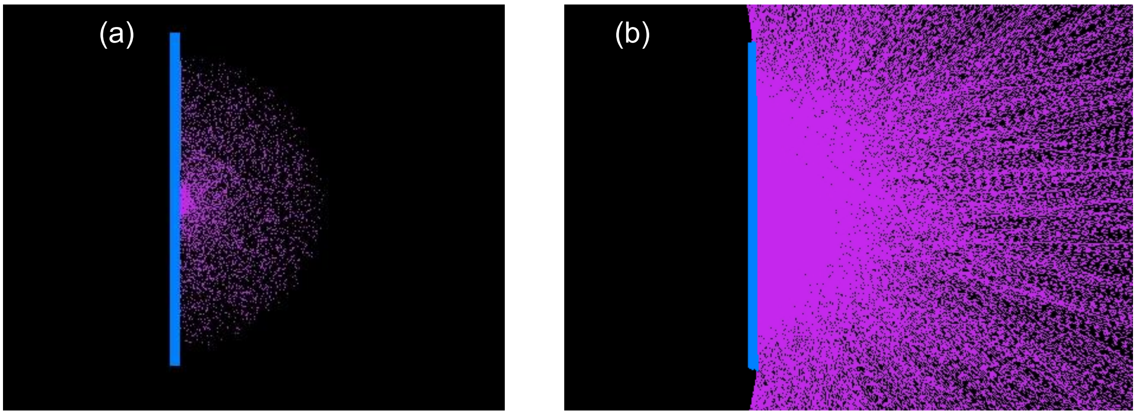

Fig. 1. Model of point like alpha emitter on surface acting as light source. (a) Alpha particles are inducing fluorescence photons in volume inside of the hemisphere having radius of 4.1 cm. (b) Fluorescence photons are emitted to random directions.

Figure 2 shows the optical design of the demonstration device without the interference filters. The design is optimized for the maximum collection efficiency for a detector having diameter of 15 mm. The total collection efficiency was calculated to be 0.12 % when the point like alpha emitter is located at 40 cm distance from the first lens having the diameter of 75 mm. The interference filters are placed in the both sides of the beam splitter in front of the detectors.

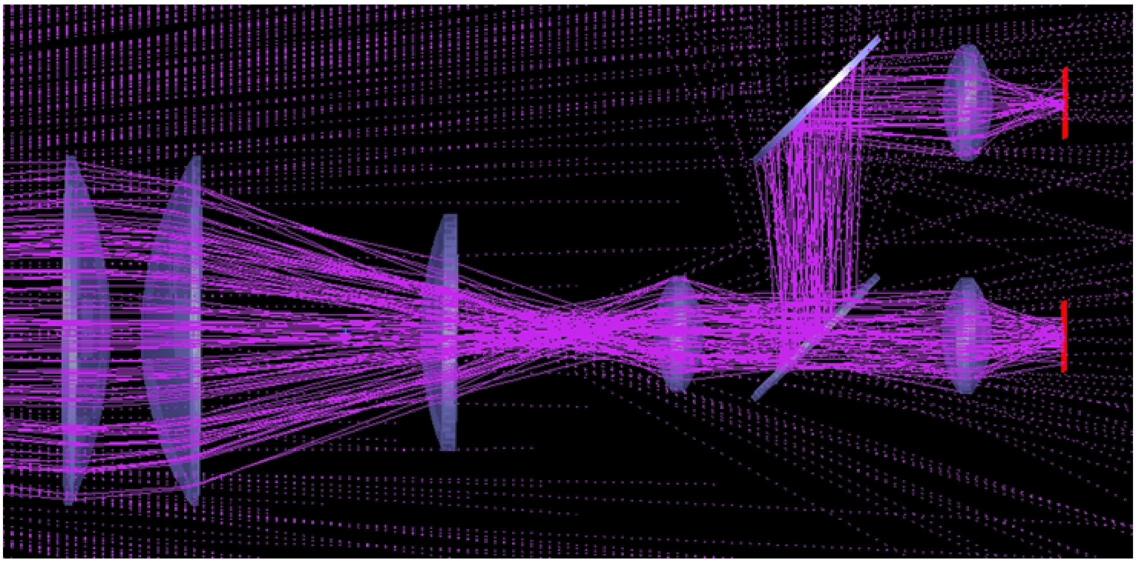

Fig. 2. Optical model of demonstration device. Vertical red lines at the right side are photomultiplier cathode surfaces. The lower channel detects the fluorescence signal at wavelength range of 300 nm – 340 nm, and the upper channel detects background lighting at 300 nm. The wavelength filters are not shown in the figure.

The optics is mounted in side of a tube, so that all the lenses are having the same optical axis. The tube is sealed in a way that no light will leak to the detector from the side. The only optical access to the detectors is through to the first lens. Channel multipliers MP 1982 (PerkinElmer Optoelectronics) were used as detectors. Figure 3 shows the demonstration device without the cover. The optics was packaged with detectors, electronics and a battery to a box having the total weight of 6 kg. The device can be used by free hands or on top of a tripod. The current detector readings are printed on a small display in the back of the device. The data can also be transferred out from the device in real time through a standard network connection.

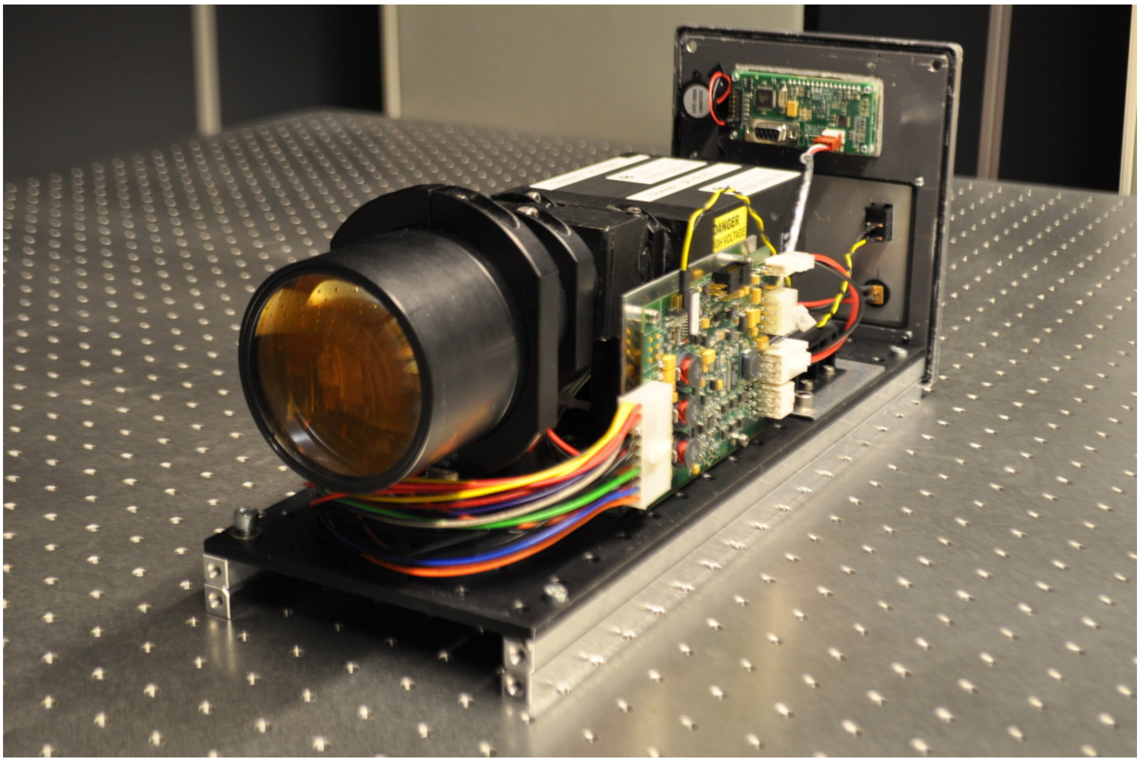

Fig. 3. Portable demonstration device for optical remote detection of alpha radiation. The cover of the device has been removed to show the optics (3" diameter), photon counting detectors, and electronics.

Results and discussion

The detection method was studied using a 10 kBq ^{241}Am source from an old smoke alarm device. The detectors counted about 3 cps dark counts in total darkness. In these conditions and without any spectral filters, the 10 kBq source yielded 150 cps signal to the detectors. The same experiment was repeated using N_2 purge as the same time. The measurement setup was covered with a box and the N_2 purge was replacing the air inside the box with nitrogen. As a result, the fluorescence signal increased to the value of 650 cps, which means more than 4 times enhancement in the signal. This is most likely due to the removal of oxygen that is effective quencher of the nitrogen fluorescence (Waldenmaier 2006). The removal of oxygen is not a relevant method in the most operational conditions, but it was valuable to notice that the fluorescence signal strength was enhanced with nitrogen purge as it can be predicted based on earlier studies with electron excitation (Waldenmaier 2006). The result of this little test is highlighted in the table 2.

Table 2. Fluorescence photon counts per second in normal atmospheric conditions and under nitrogen purge. The nitrogen purge was performed in a closed box to create a nitrogen atmosphere. Reduced quenching causes an increase in the fluorescence yield.

Fluorescence in air (cps)	Fluorescence under N_2 purge (cps)
150	650

The sensitivity of the demonstration device to the distance from the alpha emitter was studied. Figure 4 shows that the fluorescence signal is quite constant from 10 cm to 30 cm distance. The almost constant signal strength in this range is very good feature for operational work, as the operator does not need to tune the distance to the surface very accurately to get reliable readings. The fluorescence collection efficiency of the demonstration device drops rapidly as the distance gets longer than 30 cm, which is also predicted by the ray tracing model.

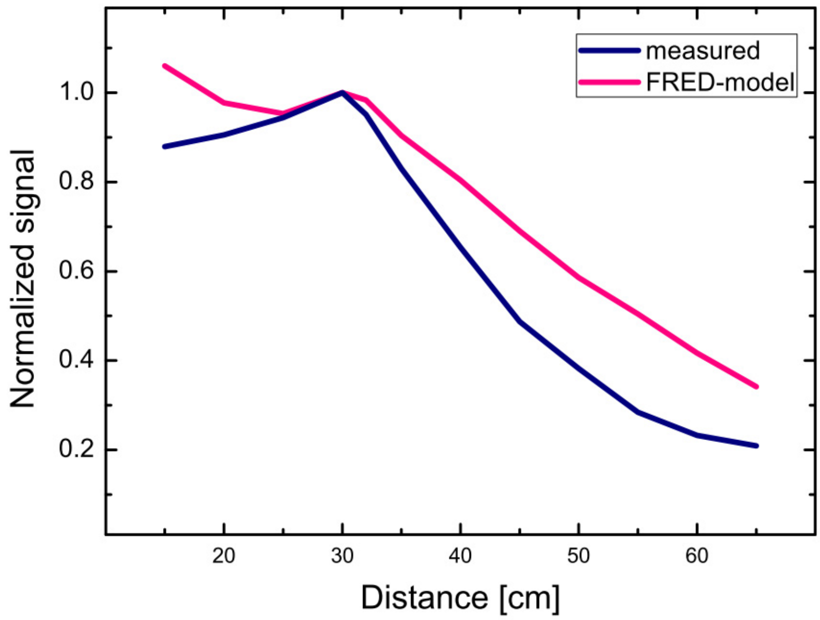

Fig. 4. Normalized fluorescence signal of demonstration device as function of distance from alpha emitter. The normalization is done at the distance of 30 cm from the alpha emitter to first lens surface of the device. The result of optical ray tracing model is shown as a reference.

The spatial sensitivity of the demonstrator device was studied by moving a point like alpha emitter transversally to the optical axis at the distance of 40 cm from the device. Figure 5 shows the measured curve as well as the one predicted by the ray tracing software. The agreement between the model and the measurement is very good, and it can be concluded that the ray tracing model is reasonably reliable. The spatial resolution of the detection is almost too good, as the 2 cm offset at the distance of 40 cm already halves the fluorescence signal. However, point like alpha emitters can be then localized very accurately.

The interference filters used in this work are having a transmittance of about 10^{-6} to 10^{-7} in the visible wavelengths. The visible light blocking efficiency of the demonstration device is enhanced by introducing three similar fluorescence filters in a row. The transmittance also drops at the bandpass wavelength, but as the transmittance of a single filter in bandpass is about 90 %, the effect is not pronounced. A small decrease in fluorescence signal is necessary to get rid of the visible background. Artificial lighting that does not produce UV light can then be used with such a filtering. The visible lighting is effectively filtered out, and the fluorescence is only marginally decreased.

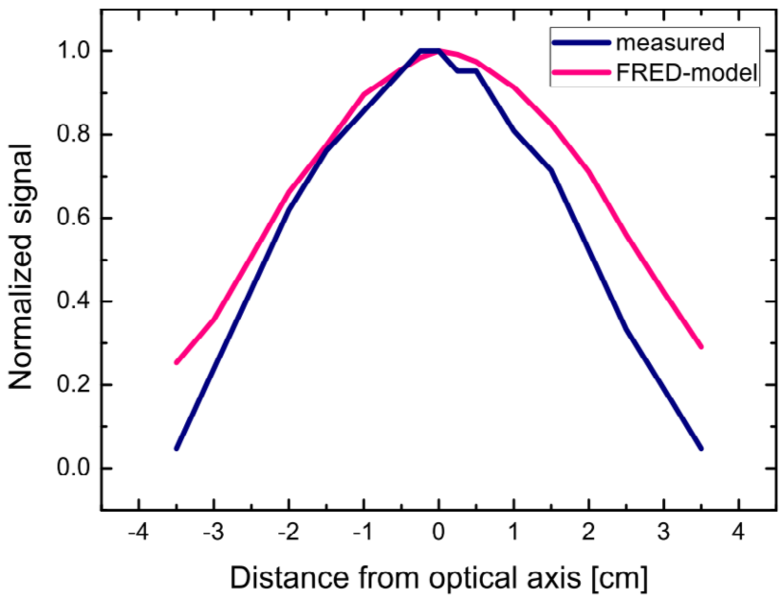

Fig. 5. Spatial distribution of normalized fluorescence signal of demonstration device as alpha emitter is moved away from the optical axis. The alpha emitter is assumed to be a point like source. The result of optical ray tracing model is shown as a reference.

Conclusions

A portable demonstration device with background compensation was constructed. The collected light consists of fluorescence signal and background light, which are divided with an optical beam splitter into two photomultiplier tubes. The both channels are filtered with interference filters to detect the fluorescence signal and the background light separately. The designed operating distance of the demonstration device was 40 cm from the point like alpha emitter. Using specially selected room lighting, the device was able to separate a 1 kBq alpha emitter from the background lighting with one second integration time. The new method looks promising for safety and security applications, where fast remote scanning of alpha radiation is required. However, the challenge of background lighting compensation needs always to be considered. The best practice is to work under artificial lighting conditions, where UV lighting can be avoided using LED-lighting or filtered lighting.

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Identpro/SIA, an identification algorithm for statistically "poor" spectra – Application to mobile or pass-by systems for real time discrimination of sources of interest

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Abstract

Source or contamination searches with mobile systems is generally limited by sudden changes in natural background leading either to setting detection thresholds at the maximum value of background fluctuations or, if set at the average level, of accepting some false positive identifications. This can be overcome by monitoring spectrum changes in real time that is then used in a process that separates the isotopes of interest from the background components or from other innocent alarms like medical isotopes.

Identpro/SIA uses an identification method that has been developed for processing statistically "poor" spectra. While laboratory measurements typically yield spectra with counts in the 100000 range, Identpro/SIA is able to process spectra with a few hundred counts from the source of interest in the presence of thousands of counts of background. Identpro/SIA uses a combined ROI / deconvolution iterative method. This method does not use a peak search technique and therefore is well adapted for low counts spectra with a having large statistical fluctuations.

It supports the full isotope library for homeland security needs as defined by the ANSI, IAEA, and IEC standards. Furthermore it has been optimized for SNM identification including NORM or Medical masking scenarios with large unbalanced ratios, much beyond current standard requirements.

One typical application is the identification and real time rejection of innocent alarms when pedestrians pass by a spectrometric portal. Other applications are source or contamination searches and mapping with car-borne or air-borne spectrometric devices.

Results of extensive testing by spectra injections, by actual source testing and from field feedback are presented.

Introduction

Classically, radionuclide identification is performed using long acquisition time to avoid the influence of statistical fluctuations. Acquisition time is obviously also related to the detector technology and size, the radiation level, the expected precision, etc.

More recently, applications such as homeland security or large area scanning for lost sources or contamination have required detection sensitivities down to a fraction of the background in time periods of a few seconds. Gross counting detection methods cannot meet in practice such a challenge due to fast changing background rates especially in indoor or in urban areas. An additional issue is the discrimination of medical radio-nuclides that are the most commonly encountered sources in unrestricted areas. Such cases are even more important when considering “masking” scenarios, i.e. intentionally hiding an illegal source, with similar spectrum characteristics, in an otherwise legal source of a medical isotope

This can be overcome in a process that is able to analyze “instant” spectra in real time, identify the radionuclide, and so discriminate the isotopes of interest from the background components or from other innocent alarms like medical isotopes.

Identpro was originally developed by Ray Gunnink in the early 2000s, and then in collaborative efforts with Mirion Technologies, continued to improve it. The C language version is named Identpro/SIA and has grown more than three times in size over early versions.

Identpro/SIA method

The IDENTPRO/SIA identification algorithm is designed and optimized for identification with poor statistics, and for detectors with poor or medium resolution (NaI, LaBr3, CZT). The algorithm first determines intensities by region of interest (ROI). Currently more than 70 ROI’s are used ranging from 20 to 2614 keV. If a net intensity is found in the complex 260-460 keV and 565-830 keV regions, the peak intensities are refined by a fitting process. After all the ROI intensities are determined, the algorithm relates these intensities to the isotopes using interference coefficients that have been stored in spreadsheet file.

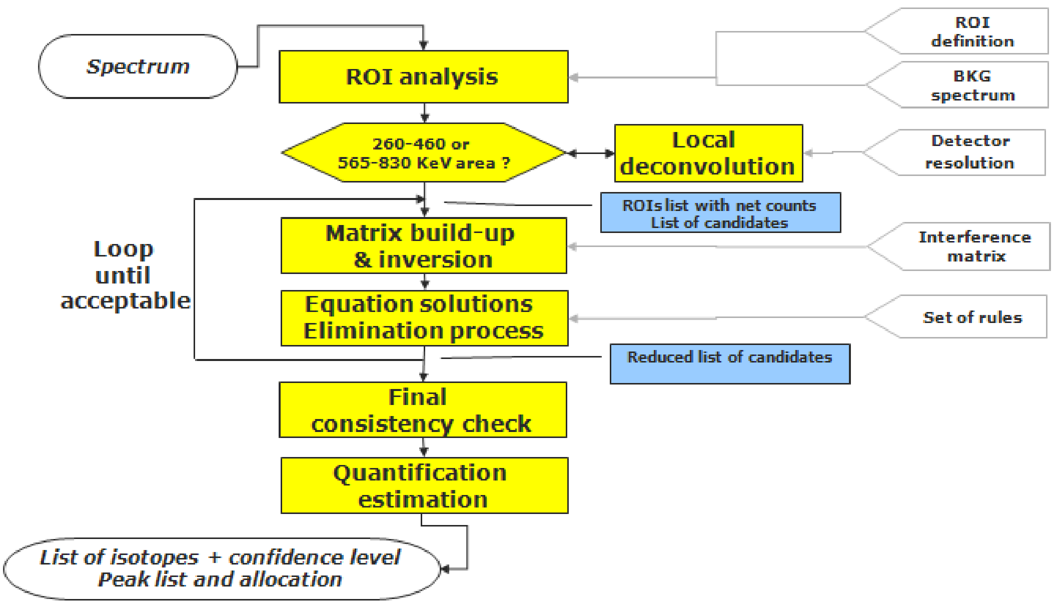


Fig 1. Flow chart of Identpro/SIA.

The next step is to solve the resulting set of linear equations by the method of least-squares, followed by rejecting isotopes yielding unacceptable results and starting a new loop with the reduced number of candidates. Finally the remaining isotopes are screened using additional criteria and decision logic tests

This method does not use a peak search technique and therefore is well adapted for low counts spectra with having large statistical fluctuations. While laboratory measurements typically yield spectra in the 100 000 counts range, Identpro/SIA is able to process spectra with a few 100 counts in the presence of 1000 counts of background

Testing the algorithm performances

Appropriate performance testing and evaluation of systems against approved standards in third party labs such as ORNL, PNNL, Austrian Research Center is mandatory, particularly for sources containing SNM. Despite the benefits we received from performance testing, one must recognize that the standard sources available in such laboratories cover only a limited number of scenarios. Furthermore the tests are go/nogo tests that do not evaluate the performance limits.

The most practical way to extensively test the identification algorithm is to use the so called injection method. Using a library of well defined spectra, i.e. with good statistics and with known intensities, one can weight, add and downscale to create sets of spectra corresponding to draws of various scenarios. It is critical that the process keep the proper statistical variation of counts within the channels, such as the Poisson law for low counts. The response of the algorithm to a set of downscaled spectra is then analyzed. The RASE program lead by the IAEA is an example of such a method.

We have used a color code to score the results of 100 draws:

Dark green: acceptable decision >95% of the cases
Light green: acceptable decision or acceptable decision +unknown >80% of the cases, wrong decision < 20% (remaining cases may be no decision)
Yellow: acceptable decision or acceptable decision +unknown >50%, wrong decision < 20% (remaining cases may be no decision)
White: acceptable decision or acceptable decision +unknown <50%, wrong decision < 20% (remaining cases may be no decision)
Orange: same as light green but added wrong isotope
Red: wrong decision substituting to right decision >20%

Table 1. Sample of the injection study output, scored with the color coded scale.

	50nSv/h		25nSv/h		15 nSv /h		10 nSv/h	
Pu93	Pu-239	98	Pu-239	92	Pu-239	53	Istat	82
	Pu-239 Unkwn	2	Istat	6	Istat	45	Pu-239	14
			Unkwn Pu-239	2	Pu-239 Ba-133	1	Pu-239 Ba-133	1
					Pu-239 Unkwn	1	Unkwn	2
							Am-241	1

We acquired hundreds of actual spectra from type testing, evaluation programs, and field feedback which allow us to downscale preferably real spectra. Real spectra have the benefit of showing unexpected cases such as unexpected secondary isotopes that are included in some sources. As an example, we observed Tl-200 and Tl-202 at low levels in medical Tl-201, high enough to be confused with masked Pu if not properly taken into account.

Single isotopes versus level and integration time

Identification of single isotopes can be challenging when one tries to go beyond the classical condition as set by current standards. For Pedestrian Portals used in dynamic mode, the reference conditions are 50 nSv/h and a passage duration of 2 to 3s.

We studied how the ID decision relates to the dose rates contribution and how it relates to the integration time.

As expected, high energy radio-nuclides such as Co60, K-40, U-238 and radio-nuclides having many peaks spread over the spectrum such as Ra-226 and daughters and Th-232 and daughters, are the most demanding cases. Isotopes with simple structure such as Cs-137, Co-57, Am-241 are easier to distinguish from background. One can note that Pu and enriched U are quite easily identified whereas DU is more difficult.

Table 2. Identification scoring for single isotopes of decreasing intensity.

	50nSv/h	25nSv/h	15 nSv/h	10 nSv/h	5 nSv/h
137Cs					
60Co					
241Am					
57Co					
67Ga					
75Se					
99mTc					
131I					
192Ir					
201Tl					
DU					
HEU					
Pu61					
Pu93					
U 20%					
U 3,1%					
U nat					
226Ra					
232Th					
40K					

Table 3. Identification scoring at 50nSv/h versus integration time.

	30s	15s	10s	5s	4s	3s	2s	1s
137Cs								
60Co								
241Am								
57Co								
67Ga								
75Se								
99mTc								
131I								
192Ir								
201Tl								
DU								
HEU								
Pu61								
Pu93								
U 20								
U 3,1%								
U nat								
226Ra								
232Th								
40K								

An example of detection and identification of very weak Pu is given below (Fig 2).

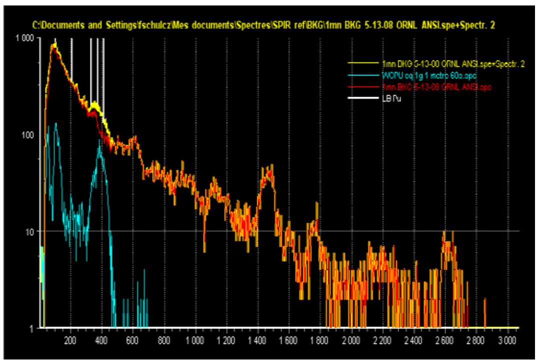


Fig 2a. Addition of low level LBPu (85cps) to Background (961cps), long spectra.

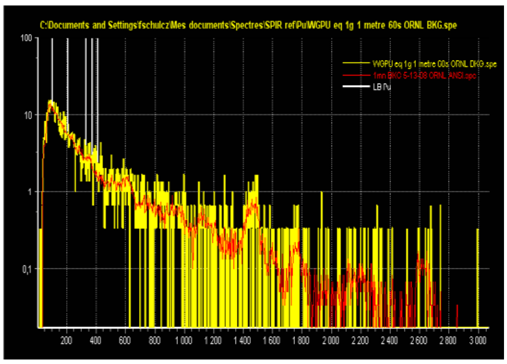
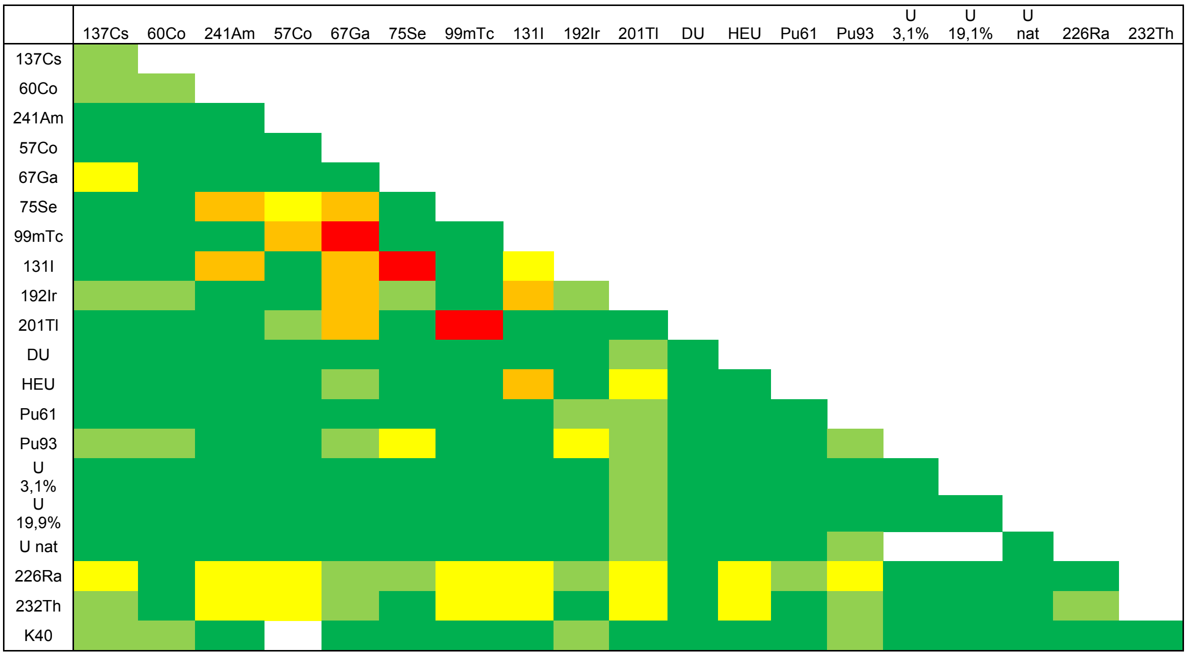


Fig 2b. Same with 3s duration. Although Pu is unnoticeable by eye, Pu warning flag is activated by Identpro/SIA.

Mixed isotopes results

We also evaluated the algorithm's response to isotopes mixed in 1 to 1 ratios. Table 3 shows that most of the cases are properly identified. The "Red" colored cases correspond to some mixed medical radio-nuclides which are unlikely due to the short half-life of medical radio-nuclides. The algorithm tends to make a safe decision when a second isotope added to a medical one is suspect because it may be an attempt to mask a threatening radio-nuclide.

Table 4. Identification results 3*3 NaI(Tl), 60s, 50nSv/h of each radio-nuclide plus background.



Masking scenarios

The most well known masking case is Tc-99m and HEU due to the proximity of the 186keV peak of U-235 and the 141 keV Tc-99m peak. With high levels of Tc-99m, even using sophisticated pile up rejection, a pulse pile-up tends to mask HEU in very unbalanced scenario (fig 4).

A very difficult combination is certainly I-131 and Pu (fig 3) especially when the intensities are largely unbalanced.

Other cases are less known, but also very difficult. E.g. Tl-201 has a secondary peak at 167 keV that can mask the 186 keV peak of U-235. Ga-67 has a 93keV peak and a 185 keV matching the X ray region and the 186 keV peak of U-235.

Even Cs-137 can mask HEU due to the backscattering peak and can also mask Pu, especially when it is shielded because the addition of a weak source of Pu only changes slightly the Compton edge distribution of Cs-137 spectrum.

We demonstrated the ability of the spectrometric pedestrian portal SPIR-Ident to identify masked SNMs in in-vivo medicals using short 2s spectra for all 16 scenarios involving mixtures of the most popular medicals (Tc-99m, I-131, Tl-201, Ga-67) and SNMs (DU, HEU, RGPu, WGPu) in ratios of 1:10. We also tested the scenarios using

1:20 ratios. These analyses still gave good results with the exception of I-131 and Pu and Ga-67 and HEU (Table 5).

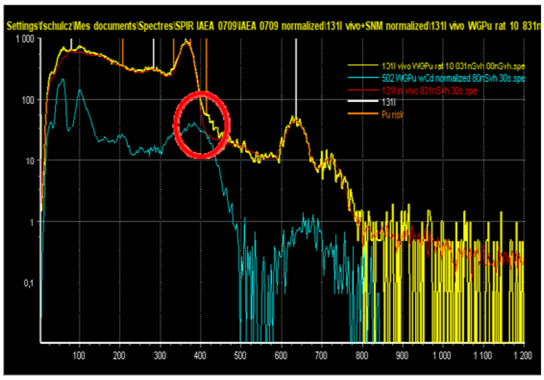


Fig 3. Addition of WGPu to in-vivo ^{131}I, ratio 1:10. Only a very slight bump can be seen on the right side of the 364 keV ^{131}I peak.

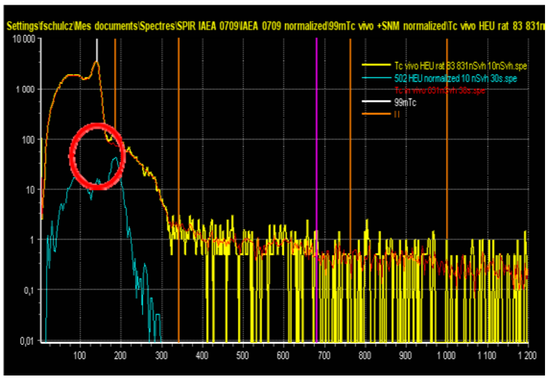


Fig 4. Addition of HEU to in-vivo ^{99m}Tc, ratio 1:10. Very low 186 keV peak hidden in ^{99m}Tc pile-up.

Table 5. Results of an injection study for SNMs masked by medicals in ratios 1:20 and 1:10 based on spectra acquired July 2009 at IAEA’s lab Seibersdorf.

Version 210809	Ratio	HEU			LEU			Pu61 0507			WGPu		
time 2s		Med	SNM	autre	Med	SNM	autre	Med	SNM	autre	Med	SNM	autre
99mTc in vivo	20 / 23	100	100		100	85 7 unid		100	99	Am, 4 Unid	100	100	
	10 / 12	100	100		100	100		100	100		100	100	
99mTc	20 / 23	100	100		100	84		100	65		100	100	
	10 / 12	100	100		100	100		100	99	3cs	100	100	
131I in vivo	20 / 23	100	100	8 unid	100	73 1 unid		100	6		100	37 12 unid	
	10 / 12	100	100	8 unid	100	100		100	100	16 unid	100	100	
	7 / 8	100	100	17 unid	100	100					100	100	
131I	20 / 23	100	100		100	42		100	0		100	12	
	10 / 12	100	100		100	100		100	100	6 unid	100	98	
	7 / 8				100	100		100	100	20 unid	100	100	
67Ga in vivo	20 / 23	100	77		100	31		100	40		100	100	
	10 / 12	100	100		100	97		100	100	3 Cscs	100	100	
67Ga	20 / 23	100	2		100	35		100	97	3 Cs	100	95	
	10 / 12	100	96		100	96		100	100		100	100	
201Tl in vivo	20 / 23	100	73		100	56		100	100	4 Cs	100	100	
	10 / 12	100	90		100	95		100	100	6 Cs	100	100	
201Tl	20 / 23	100	60		100	70		100	100		100	100	
	10 / 12	100	75		100	100		100	100	3 Cs	100	100	

Experimental data: spectrometric pedestrian portal for airport

At airports or seaports, where large crowds must be checked, alarms due to medicals radio-nuclides cause a laborious and time consuming secondary inspection procedure that is annoying to front line officers as well as to innocent passengers.

Results of practical testing and of injection studies (see above) now allow us to set a dose rate threshold for medical isotopes uunder which a spectrometric portal will detect masking scenarios with acceptable performance, avoiding the need of secondary screening, and over which a secondary screening is still necessary.

Therefore, the crucial question is if such a threshold will significantly reduce the need for secondary screening. In order to gain practical experience in this question, a SPIR IDENT portal has been installed at the Vienna International Airport in early April 2009 and has been in continuous operation there since

Fig 5. SPIR-Ident layout at Vienna's airport non Schengen exit.

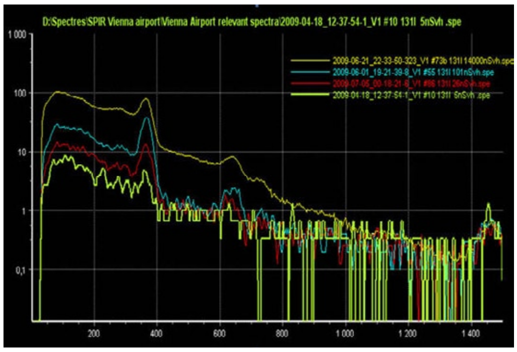


Fig 6. ^{131}I spectra 14µSv/h, 101nSv/h, 26nSv/h and 5 nSv/h maximum dose rate, captured at Vienna's airport exit.

Table 6. Results of 160 days operation with 170 medical alarms for about 1 million passengers.

all events for 160 days	all levels detected	all levels identified	max>15 nSv/h detected	max>15 nSv/h identified	max>25 nSv/h detected	max>25 nSv/h identified
201Tl	92	54	53	50	45	45
99mTc	35	30	15	15	12	12
131I	25	22	18	18	16	16
226Ra	7	3	4	2	0	0
125I?	5	0	0	0	0	0
111In	1	1	1	1	1	1
67Ga	1	1	1	1	1	1
unknown	1	1	0	0	0	0
possible Th	1	0	1	0	0	0
123I	1	1	0	0	0	0
Total	170	114	93	87	75	75
% identified		67%		94%		100%

Summarized results are the following:

- 170 cases during 160 days, one per 7000 passengers
- Tl-201 55% of the cases, Tc-99m 20% and I-131 15%
- Other isotopes are very rare but several cases of Radium
- Very low statistical alarm rate (2 per month)
- Many weak alarms 55% of the cases <25nSvh max,
- Few large alarms: 5% > 1µSv/h max

Identification capability of the SPIR-Ident spectrometric pedestrian portal has been demonstrated: 94% of the cases with peak level >15nSv/h maximum and 100% of the cases with peak level >25 nSv/h max are identified in real time when passing by. No incorrect identifications were observed and no false positives were reported despite cases with saturation

Experimental data: car-borne real time identification

Another application is source or contamination search and mapping with car-borne or airborne spectrometric devices. In the SPIR-Ident mobile implementation, a 2s spectrum is analyzed every 0.5s. Level information is continuously memorized along with the position given by a GPS. Levels are color coded on a map that can originate from Google Earth or from a preloaded map. When a detection occurs the spectra and the ID decisions are saved and a summary of the event is displayed as shown in Fig 7 and Fig 8.

Fig 7. Passing-by a legally transported source on the motorway.

Fig 8. Pedestrian and car driver with Tc-99m in the vicinity of a hospital.

The technology in SPir-Ident Mobile allows us to distinguish sudden background changes due to different road construction materials from actual alarms. Fig 7 shows a detection of a legally transported source at the moment the monitoring car passes by. The alarm duration was only 1 second but Cs-137 was identified within this duration. The monitoring car waited for the truck to get closer again to confirm the alarm which gave a second detection of about 20s. Fig 8 shows the detection and identification of persons injected with Tc-99m in the vicinity of hospital.

Poor statistical definition of actual spectra acquired during a car-borne survey can be seen in fig 9 and fig 10.

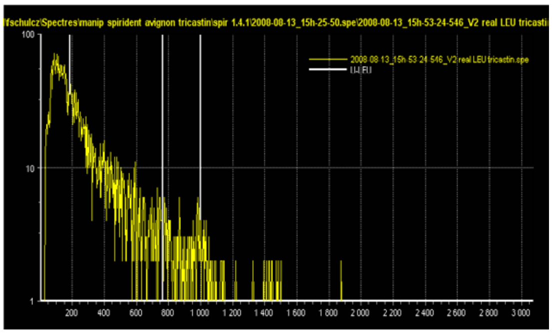


Fig 9. Uranium identification at background level when passing by an enrichment facility.

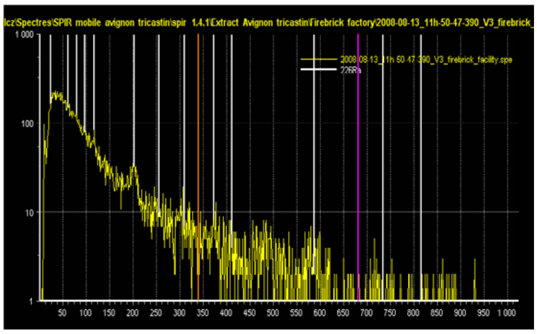


Fig 10. Ra-226 identification when passing-by a firebrick factory.

Conclusion

We gave an overview of Identpro/SIA as a valuable solution for identification of statistically “poor” spectra such as spectra processed by spectrometric portals in dynamic operation, spectra acquired in real time during mobile survey by handhelds, back-packs and larger mobile detection systems. So called “injection” studies have been widely used to improve ID algorithm and characterize the response to numerous scenarios, including demanding masking scenarios.

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[3] DUFTSCHMID, K., “ADVANCED DETECTION TECHNOLOGIES TO COMBAT NUCLEAR TERRORISM”, 54th Annual Meeting of the Health Physics Society Minneapolis, July 2009

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Developments in radiological-nuclear support to security through the Canadian CBRNE Research and Technology Initiative (CRTI)

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Abstract

The Chemical, Biological, Radiological-Nuclear, and Explosives (CBRNE) Research and Technology Initiative (CRTI) was created to fund projects in science and technology that will strengthen Canada's preparedness for, prevention of, and response to potential CBRNE threats to public safety and security (1). Canada's federal radiological community – collectively known within CRTI as the Radiological Nuclear Cluster – has benefitted enormously from the collaboration upon which CRTI insists in order to qualify for funding.

During the past few years, the RN Cluster has taken steps towards becoming more operational, through field exercises, providing reachback support for responders and, on occasion, forward-deploying scientists and technicians for major events. This foundation, coupled with creative application of technologies developed by partners, enabled Canada's radiological community to rapidly and efficiently develop a smart, scalable solution for radiation security at the Vancouver 2010 Olympic Games.

Introduction

The Chemical, Biological, Radiological-Nuclear, and Explosives (CBRNE) Research and Technology Initiative (CRTI) was created under the leadership of Defence Research and Development Canada (DRDC) in 2002. Its mandate is to fund projects in science and technology that will strengthen Canada's preparedness for, prevention of, and response to potential CBRNE threats to public safety and security. Twenty-one federal government departments and agencies have signed an agreement to work together, through CRTI and related programs at the DRDC Centre for Security Science, to help further the meaningful application of science to address problems and challenges in the CBRNE domain.

One of the greatest strengths of the CRTI program is its insistence on collaboration. To qualify for funding, project proponents must partner with scientists and end-users from outside their own organizations, including other government departments (federal, provincial or municipal), first-responder and first-receiver groups, academia, industry, and the international community. In addition, scientists and response personnel are invited to participate in science "clusters" – one for each

C,B,RN, and E discipline, and an additional one for forensics – to share knowledge, target gaps in prevention or response capabilities, and generally identify synergies and common interests to further smart and complementary use of resources.

Recently, the clusters have taken steps towards operationalizing their science in a very real way – not only by putting new technologies into the hands of traditional response personnel, but by more actively providing reachback support for responders and, on occasion, forward-deploying scientists and technicians for major events. This paper will briefly describe the evolution of field teams within CRTI's Radiological-Nuclear Cluster, and highlight how they were deployed for radiological surveillance and support to security for the Vancouver 2010 Olympic Games.

For reasons of operational security, a number of details have been withheld from this discussion.

Preparation

Almost 20 years before CRTI was created, Prime Minister Pierre Trudeau ordered the Department of Health and Welfare (now Health Canada) to design a framework that would allow federal RN expertise to be harnessed and directed towards RN emergency response, in a hurry, if required. The current version of this framework is the *Federal Nuclear Emergency Plan* (FNEP, (2), with an updated Appendix 5 (3) that describes primary and secondary responsibilities for all federal departments with a role to play in the response *following* a radiological or nuclear emergency. Most of the departments tasked under FNEP are now also members of the RN Cluster.

When the Vancouver 2010 Olympic Games appeared on the Canadian security horizon, it became clear that a new framework was needed for departments with a role to play before a full-blown RN emergency is realized – specifically, for pre-event surveillance and support to federal security personnel[1]. Documents such as the IAEA draft planning guidance for nuclear security at major events (4), as well as communication with radiation and nuclear security organizations from countries who had hosted similar events (including Finland, Greece, and the United States) formed the planning basis, and a concept of operations was developed jointly with the Royal Canadian Mounted Police National CBRNE Response Team [2](RCMP; responsible for CBRNE security within the Olympic domain) and federal radiation scientists. Planning was made significantly easier due to good relationships between all parties, developed through CRTI-funded projects and exercises, and by leveraging a mix of mature, relevant capabilities pre-existing within the Cluster.

Team-building through exercises, 2003–2010:

One of the very first projects ever funded through CRTI involved a series of four, increasingly challenging field exercises using actual radioactive materials. The first was held in 2003; teams of scientists and technicians from four departments were sent out into a field to locate, identify, quantify and retrieve a variety of sealed sources. They had little difficulty dealing with the sources, but the experience was eye-opening in

[1] Border security and support to municipal responders were handled separately and will not be discussed in this paper.

[2] The core of the National CBRNE Response Team is composed of RCMP, military assets and biological experts from the Public Health Agency of Canada; experts from additional domains were added for the Olympics.

terms of recognizing the logistical requirements for a field deployment, not to mention the need for all-hazards expertise and police officers when dealing with unknown agents and possible terrorists.

Science Teams

Over the years, steps were taken to address the short-comings and learn the lessons identified in the first and subsequent exercises. Roles were defined, equipment upgraded, and operating procedures drafted and tested. In 2007, a few months prior to the fourth and final exercise in the original series, a concept of operations document was written for the newly-named Federal Radiological Assessment Team (FRAT). FRAT was originally conceived as an *ad hoc,* multi-agency, multi-disciplinary group of RN experts who can be deployed to the site of an incident when specialized equipment and/or expertise are required for consequence management. FRAT is not a standing organization; rather, departments and agencies provide resources to FRAT on an as-required basis depending on the nature of the incident and other operational commitments. Contributing organizations include: Atomic Energy of Canada Limited, National Defence (Defence Research and Development Ottawa and Defence Nuclear Safety Division), Health Canada and Natural Resources Canada. Personnel from these organizations train together periodically, participate in the planning and execution of exercises and, lately, join forces to provide scientific support for major events. Any one of these can supply the Team Lead for a given operation and, typically, for exended deployments, personnel from different home organizations will rotate through the lead position.

Security Teams

The exercises, as well as additional training and simulation opportunities and joint CRTI projects, provided opportunities to interact with the responder community and, specifically, the National CBRNE Response Team (National Team). It would be fair to say that the scientific community was ready to embrace the policing community well before the police were ready for the scientists. However, as the exercise series progressed, participation changed from scientists alone, to scientists and police working separately on concurrent scenarios, to finally scientists and police working together. This mutual exposure was crucial to enabling the two groups to begin to understand each other and, subsequently, to find ways to make science truly work in a security context. By March 2008, FRAT had begun to expand its field capabilities from strictly post-incident response to include pre-incident surveillance and support to interdiction and intervention.

Deployment for V2010

In December 2008, the Government of Canada approved a request from the National CBNE Response Team for chemical and radiological scientific support prior to and during the Vancouver 2010 Olympic Games. Biological scientific support had been integrated with the National Team for years; the addition of chemical and radiological capabilities provided an unparalleled degree of technical reachback. The collective C,B, and RN deployed assets, along with a mobile forensic capability, became known as "Science Town." This paper focuses on the radiological assets; however, it should be understood that these are just one aspect of a broader initiative to complement security with operational science.

Specific support tasks fell into the following categories:

– Scientific reachback and advice
– Sample analysis
– Surveillance

Scientific Reachback and Advice

Throughout the deployment, the RN Science Advisor was the main liaison between the RCMP and the rest of the radiological science team. All information gathered by the scientists flowed through the Science Advisor to the RCMP; and all direction and information from the RCMP flowed through the Science Advisor to the rest of the radiological team.

The Science Advisor was required to have a strong background in radiation physics and health physics, good communication skills, experience working with RCMP, and a good knowledge of the expertise and skill sets that could be called upon from within the RN Cluster, if required. Four RN Science Advisors from three different organizations rotated through the six-week Olympic operation; each also served as the lead for the multi-departmental FRAT while deployed.

Sample Analysis

Samples and swipes were counted and characterized on-site by analysts using a variety of detectors in a vehicle-based platform, or "mobile laboratory." Mobile labs have been used by some cluster members for decades. In 2005, building on lessons learned from partners' earlier iterations, CRTI funded four identical mobile nuclear laboratories, strategically locating them with host organizations across Canada. The labs are very simple – they are effectively mobile bench-space with power and sufficient climate control to allow operation in the extremes of Canadian weather – making them extremely portable. In order to maintain operational readiness, they are routinely used by their hosts for decommissioning, field trials, and training and exercises. In the year prior to the Olympics, CRTI was able to fund a fifth mobile nuclear lab, piloting a new design which maintains mobility while adding workspace, networked computing, and some new features for improved detection and sample handling (especially for nuclear forensics). Future deployments will also include a trailer with mounted satellite and workspace for command and control.

During the Games, samples were collected by police and brought back to the scientists; under no circumstances were scientists to enter areas where there were unknown or significant non-radiological hazards. Samples would arrive at the forensic trailer first for basic triage, sample logging, and collection of traditional (non-CBRN) evidence. Scientific Advisors from all three disciplines and forensic experts would then discuss the situation and potential hazards in order to determine the safest and most expedient way to proceed with identification and characterization of CBRN agents. In addition to minimizing risk to scientists (by not forward deploying them) and limiting disruption to a potential crime scene, this approach also allows for centralization of equipment thereby minimizing the overall footprint of the deployed reachback.

Surveillance

RN surveillance for security purposes was something relatively new to both the scientists and the police in Canada; consequently, it took some time for the idea to gain acceptance from all stakeholders. In addition, the other elements of security planning with which RN surveillance had to mesh were extremely elastic; as a result, the strategy had to be flexible and adaptable, allowing for last-minute changes and adjustments even after installation. Because both buy-in and planning guidance were somewhat slow in coming, the team had less than a year to implement the strategy.

In addition to the short timeline, constraints were as follows:

- Discretion was paramount. The Olympics are considered a sporting event, not a security event; consequently, surveillance had to be discreet and minimally intrusive.
- Olympic venues were spread over a geographic area of more than 1000 km^2.
- Money and personnel were both very limited resources.
- A large amount of detection equipment was required for a very short time.

The solution centered on rapid detection and identification with reliable, well-characterized instruments and systems. It combined mobile and portable networked systems of spectroscopic detectors, which allowed a very few people to monitor very large areas, with less costly and sophisticated human-portable detectors, worn by police officers and other security personnel at strategic locations to add depth and redundancy. Finally, costs and procurements were justified by ensuring that almost all of the equipment was re-usable for future major events, and much would be re-deployed out to regional offices or used to replace aging inventory in environmental monitoring networks.

Discussion

The approach allowed the RN Cluster to play to its strengths, as well as to extract as much use from our limited resources as possible. Fortunately, the problem of monitoring a large area with few people is a common one in Canada and so it was possible to leverage existing capabilities – specifically those used for environmental monitoring, survey and mapping -- and re-apply them to security in a number of cases. By re-deploying or building on systems that were already extremely well understood and deploying the “normal” users to operate them, performance risk was kept to an absolute minimum, as was the need for specialized training.

In addition, using mobile and portable equipment ensured that detectors could be re-distributed throughout the Olympic domain if the threat assessment changed. Looking to the future, it is also now relatively easy to re-deploy this equipment for major events in other parts of the country.

Where possible, data was automatically captured and routed to a central location, where it could be accessed and analyzed locally or remotely. This allowed analysis duties to be shared between personnel deployed to Vancouver and those at home in Ontario, thereby ensuring that sufficient personnel were available to do the work while keeping the deployed footprint small, and minimizing wear-and-tear on personnel in the field. Again, this reduces performance risk and also helps ensure the well-being of the people in the field, increasing the likelihood that they'll be willing to participate again in future events.

It can be argued that relying on data transmission for a surveillance system is risky in that, if communications are lost, so is the ability to monitor. For the Games, redundant systems were in place, albeit less sophisticated ones, so that monitoring would continue uninterrupted even if a whole network was disabled. In addition, and on a strictly practical note, the risk was considered and accepted in light of the significantly greater situational awareness and significantly reduced personnel costs afforded by a distributed network of unmanned detectors and centralized data analysis.

Technology was a big part of what FRAT brought to the Vancouver Olympic Games. However, what the police wanted – more than detectors and data – was information from a reliable source. This what they always want, whether for a major event or a crisis in the middle of the night. For years, the only essential components for effective scientific support have been a telephone and a 24/7 phone number. The detectors and the data help during a large public event, but it is the context that an experienced RN subject matter expert can add to a detection (or suspected detection) that makes the information truly valuable. Fortunately, this context can be applied – with less precision, perhaps – even in situations where the data is collected by instruments and operators less familiar to the expert. Now that the Olympics are over, the challenge for FRAT will be to sustain the momentum built up over the past couple of years and find ways to ensure that the expertise remains accessible to those who need it.

Conclusions

The support and incentive provided by CRTI to develop capabilities in RN counter-terrorism encourages the Canadian federal radiological community to find ways to apply science and technology to security problems. This includes developing technologies and networks that can be adapted and exploited for multiple uses, both emergency and peace-time. Further, exercises, deployments, and collaborative work with Canadian police and security organizations provide opportunities to test and demonstrate the applicability of newly developed technologies for security purposes, and to reinforce the applicability of specialized expertise in helping responders deal with CBRN agents. Overall, CRTI has helped foster a climate in which scientists and responders are comfortable and well-positioned to combine their assets to defeat CBRN threats, as was evidenced by the successful deployment for the Vancouver 2010 Olympic Games.

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Radiological security measures at the United Nations Climate Change Conference in Copenhagen, 2009

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Abstract

From December 7 to 19, 2009 Denmark hosted the United Nations Climate Change Conference, COP15. For an event of such magnitude, security was a major concern. Danish Police in cooperation with UN decided to establish a dedicated CBRN (Chemical, Biological, Radiological, and Nuclear) security and emergency management project. The National Institute of Radiation Protection (NIRP) was expert advisor on the radiological security measures. These included access control and radiological surveillance at the conference site (Bella Center) and a mobile team that could perform screening tasks and emergency assistance in the Copenhagen area. Daily screenings of key locations in Copenhagen were made with carborne gamma spectrometry in cooperation with Danish Emergency Management Agency (DEMA).

The radiological security check at Bella Center was constructed to be discreet and with a minimal distraction of the traffic of delegates. The conference access control area was airport style with all together 25 lines with metal detector and x-ray machines for luggage scan. It was expected that approximately 15,000 delegates, journalists and NGO members would pass the security control over the course of the meeting.

For personal screening walk-by plastic scintillation detectors were installed. Detection of any gamma radiation above 60 keV would trigger an alarm. Police and UN-security personal were instructed to stop the traffic in the access control area in case of an alarm and to contact the radiation expert. NIRP radiation experts remotely monitored the detectors from a nearby office and could arrive at the access control area within a minute after an alarm. Their equipment included a mobile HR Ge-detector that could produce a high resolution gamma spectrum in minutes. Three incidents were recorded with persons who had received radiopharmaceuticals.

Introduction

Danish police had the main responsibility for security during the United Nations Climate Change Conference, COP15. The conference was held in Copenhagen from

December 7 to 19, 2009 and was the largest UN conference ever held outside the UN building in New York. UN had several specific requests to security including CBRN security and emergency preparedness measures. NIRP was therefore asked to plan and implement the radiological security action in accordance with the UN requirements.

NIRP is part of The Danish National Board of Health and is the Danish authority on radiation protection. This includes a 24-hour emergency service with one radiation expert on duty that can respond to any unintended incidences with radioactive materials. The radiation expert can also offer qualified assistance to police and other authorities in case of malicious acts involving radioactive or nuclear material.

However, the magnitude of the COP15 conference, required a corresponding reinforcement of the radiation security and emergency management. Internal education and training of in-house radiation protection officers started 8 month before COP15 and altogether 15 radiation protection officers took part in the event. This number does not include administrative personnel and lab technicians who were on-call duty during the cause of the event.

DEMA participated in the security actions with two vehicles equipped with carborne gamma spectrometry and with four HAZMAT (HAZardous MATerials) teams each trained to conduct sampling and measurements in case of a CBRN incident.

Material and methods

Altogether 3 different radiological security actions were arranged a) daily radiological screening of key localities, b) a mobile radiation expert team, c) access control at Bella Center, the conference venue.

Daily Radiological Screening

The purpose of the daily screenings on the streets of Copenhagen and in the vicinity of the conference centre was to observe any diversion from the normal radiological picture. The screening effort was concentrated around high security hotels and other places where VIP delegates were or could be present.

Several months in advance of the conference NIRP had contacted all licensed users of radioactive sources and companies transporting radioactive sources to ensure that NIRP was notified prior to the use of sources and transportation during the conference period. A ban on transport of high activity sources was issued. The screenings were performed by DEMA's carborne gamma spectrometry vehicles equipped with an Exploranium 4L NaI(Tl) detector (256 in^3, 16"x 4"x 4"). Measurements were made as 2-seconds real time. For neutron detection a NUCSafe Guardian PRST neutron detector was used.

Radiation Expert Team

The mobile radiation expert team was on 24 hour on-call duty and could be requested by the Police. It could respond to any situation involving radioactive substances or suspicion of use of radioactive substances in the Copenhagen area within an hour. The team was equipped with hand-held radiation instruments including FieldSpec Na(I) gamma spectrometers for nuclide identification. They would be able to advice Police and HAZMAT teams in setting up controlled areas and could handle and secure minor radioactive sources.

Radiological Surveillance and Access Control

The radiological access control at Bella Center was probably the greatest challenge since this had to be coordinated with several other security measures and involved close cooperation between Police, UN security personnel and a private security company. The radiological security measures had to be coordinated with x-ray luggage scans and metal detection of all conference delegates and their luggage. Access to the conference centre was open 24 hour a day and altogether 25 lines with luggage scanners and metal detectors were open during conference peak hours between 7 and 10 am. Up to 5,000 individuals were passing through the access control area within this time interval. Police and UN–security required that radiological security check was discreet and that any false alarms and incidents could be solved quickly and with minimal distraction of the traffic in the access control area. The main concern was false alarms from delegates who had been examined or treated with radiopharmaceuticals, as previously observed in international events where radiological security measures was performed (IAEA, 2009).

The problem was solved by installing three walk-by detectors placed between the first security check and the entrance to the accreditation area (Fig. 1 A). The walk-by detectors were Canberra Portia highly sensitive 40 x 60 x 5 cm^3 plastic detectors that were set to record any gamma radiation above 60 keV. The alarm level was set to trigger an acoustic and visual alarm if a source corresponding to 370 kBq Cs-137 passed the detector within 1 m. A more sensitive setting was possible, but impractical, since the alarm would be triggered by a radioactive source too far from the detector, and hence it would be more difficult to identify the source in a large crowd. Police and UN-security personal were instructed to stop the traffic in the access control area in case of an alarm and contact the radiation expert. NIRP radiation experts remotely monitored the detectors from a nearby office and could arrive at the access control area within a minute after an alarm. During peak hours a NIRP expert was present at the access control area together with other security personnel. NIRP experts were equipped with Thermo RadEye B20-ER GM detectors for dose rate measurement and Thermo GX 2'' Na(I) detectors for high sensitive gamma detection (Fig 1 B). All radiation experts wore Thermo EPD personal dosimeters. In case of an alarm from a walk-by detector, with many people present, NIRP experts could find the radioactive source with hand-held detectors.

The radioactive person or object was hereafter escorted to a visitation room for nuclide identification and activity. For radionuclide identification a Canberra Falcon 5000 portable cryo-cooled HR Ge-detector was used (Fig 1 B and C). The Falcon uses Falcon 5000 software as well as Genie2000 detector software. The software nuclide library was updated with all likely radionuclides used in industry and medicine.

Fig. 1 (next page), A. The Access control area with walk-by detectors to the right equipped with acoustic and visual alarms (see arrow), B. Mobile HR Ge-detector (Falcon 5000) and hand-held detectors used during the operation, C. Measurement of a person in front of the Ge-detector, and D. Surface contamination measurements in the plenum hall.


A
B
C
D
FORCE

Table 1. List of typical radionuclides used in radiopharmaceuticals. Data from NIRP.

Radionuclide	Type of Radiation	Energies (keV)	Yield (%)	Half Life	Average dose to patients (MBq)	Medical use
H-3	Beta	19	100	12.3 years	6	PET
C-11	Gamma Beta	511 960	200 100	20.4 minutes	420	PET
N-13	Gamma Beta	511 1199	200 100	9.97 minutes	670	PET
O-15	Gamma Beta	511 1732	200 100	2.04 minutes	997	PET
F-18	Gamma Beta	511 634	200 97	1.83 hours	382	PET
Cr-51	Gamma	320	10	27.7 days	4	Blood circulation
Co-57	Gamma	122	86	271.8 days	<1	Digestion
Fe-59	Beta Beta Gamma Gamma	273 466 1099 1292	46 53 56 44	44.5 days	<1	Several uses
Ga-67	Gamma Gamma Gamma	93 185 300	39 21 17	3.26 days	192	Several uses
Se-75	Gamma Gamma Gamma	136 265 280	59 59 25	119.8 days	<1	Digestion
Kr-81m	Gamma	276	4		5587	Lung
Sr-89	Gamma Beta	909 1492	<1 100	50.5 days		Therapy
Y-90	Beta Beta	546 2284	100 100	29.1 years		Therapy
Tc-99m	Gamma	141	89	6 hours	857	Alt
In-111	Gamma Gamma	171 245	90 94	2.8 days	224	Blood circulation Digestion
I-123	Gamma	159	83	13.2 hours	306	Nerve, endo-krine + div.
I-125	Gamma Gamma Gamma	27 31 35	114 26 7	60.1 days	2	Blood circulation
I-131	Gamma Beta	365 606	82 90	8 days	522	Blood circulation Therapy
Xe-133	Gamma Beta	81 346	37 99	5.2 days	2500	Nerve
Sm-153	Gamma Beta Beta Beta	103 634 703 807	28 35 44 21	1.95 days		Therapy
Lu-177	Gamma Beta	208 490	11	6.71 days		Therapy

Table 1 lists the most commonly used radionuclides in radiopharmaceuticals in Denmark. The Falcon 5000 software is designed to suggest radionuclides from the library as soon as a significant peak is found in the spectrum. The most recent background spectrum is automatically subtracted, when a radioactive source is measured. The radiation expert on duty measured a background spectrum each day. The background spectra contained K-40, U, Th and their typical decay series isotopes.

The radiological security measures in Bella Center also included background radiation measurements at specific localities in the centre and dose rate and surface contamination measurements at plenum halls and other key localities (Fig. 1 D). For surface contamination measurements we used Thermo RadEye AB100 detectors.

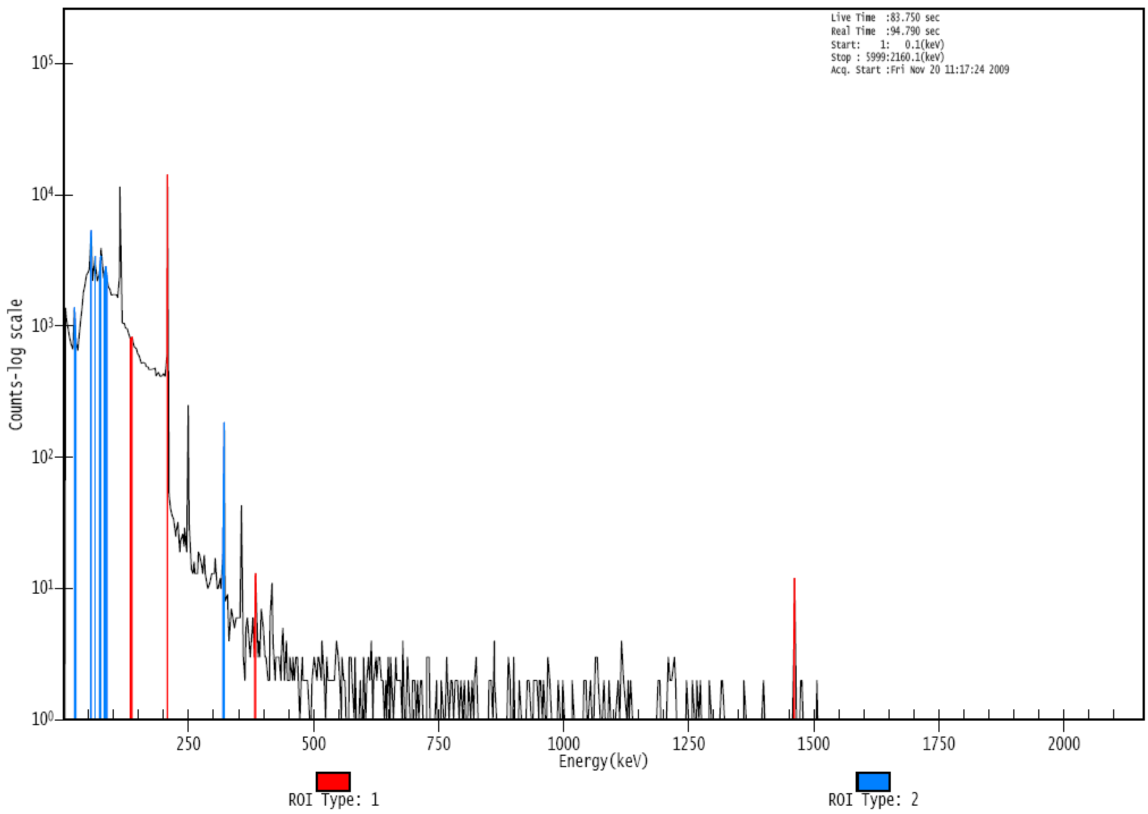

Fig. 2. High Resolution gammaspectrum taken 1 m from a patient who has been treated with radiophamaceutical containing Lu-177. A measuring time of little more that a minute is sufficient for precise radionuclide identification.

Radiological Incidents

During the entire operation 3 incidents were recorded, all at the access control area in Bella Center. All incidents were so called “innocent alarms” related to delegates or employers who had received radiopharmaceuticals. The radionuclides detected in the three cases were I-131, Tl-201 and Lu-177. In all cases, the delegate triggering the alarm of the walk-by detector was approached by a radiation expert with a handheld instrument, and the delegate was hereafter measured in the visitation room with the HR Ge-detector. Tl-201, a radionuclide used for heart diagnostics, was not on our original list of potential radiopharmaceutical, since it is not used in Denmark (Table 1). Identification was, however, fast from the Falcon 5000 nuclide library and since the delegate was aware that he had been examined, the measurement took only a few minutes.

All radioactive delegates were equipped with a certificate of the measurement with radionuclide and corresponding dose rate. They were kindly asked to report to the radiological expert on duty prior to every transit of the access control area in order to be guided past the walk-by detectors.

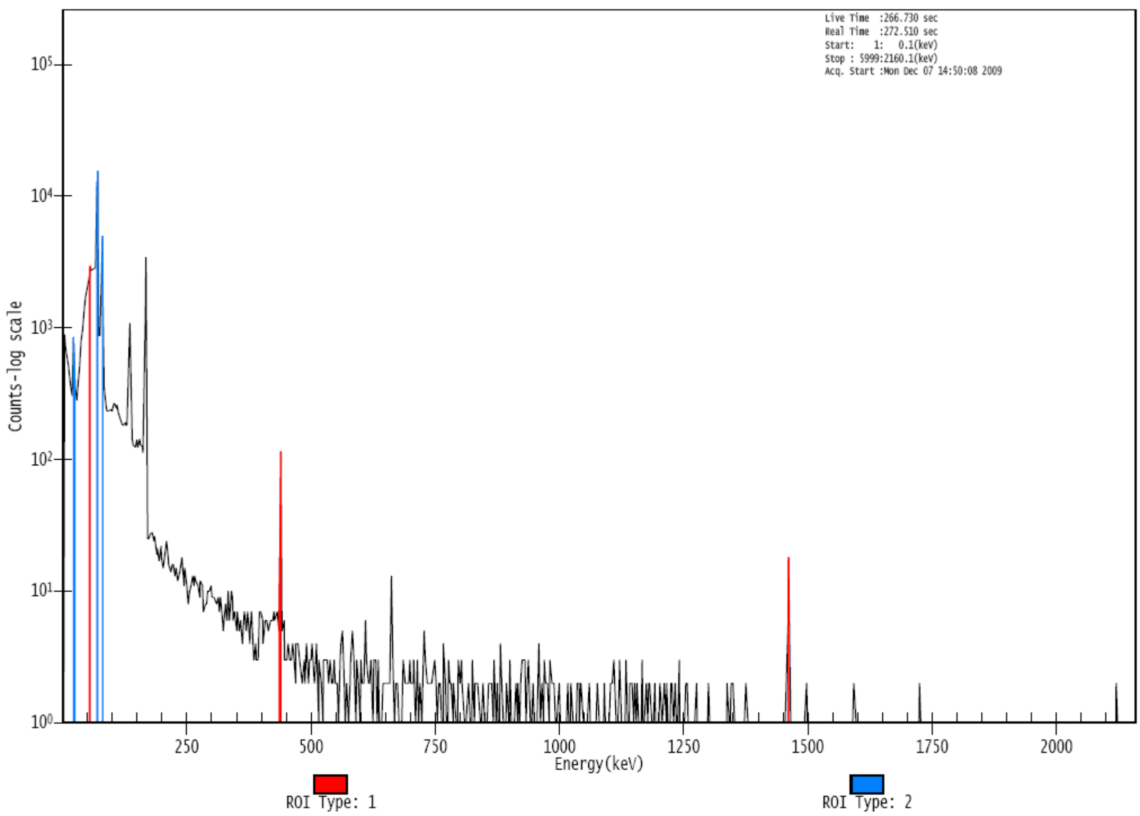

Fig. 3. High Resolution gamma spectrum taken 1 m from a patient who has been treated with Tl-201. The characteristic gamma energies from Tl-201 are at 69 and 167 keV, respectively. Note also the gamma energy from Tl-202 at 440 keV.

Gamma Spectra Recorded

Fig. 2 shows a typical gamma spectrum for a person who has received Lu-177. All gamma lines in the spectrum originate from Lu-177 (Table 1) or background U, Th and K isotopes. The background was, as previously mentioned, subtracted before the nuclide library match is performed.

The gamma spectrum for Tl-201 is shown in Fig. 3. It is well known that Tl-201 often contains traces of Tl-202, which can be seen in the spectrum at the characteristic gamma energy at 440 keV. In fact, Tl-202 is probably responsible for most of the dose rate from this particular patient since its half life is 12 days compared to 73 hours for Tl-201. The patient had received his examination a week before attending the conference and the dose rate was 0.2 μSv/h at 0.5 m from his torso at the time the gamma spectrum was obtained.

Conclusions and Lessons Learned

Events involving radioactive material intended for malicious purposes were not detected. This in itself is an important result.

We learned the following from the radiological security arrangements during COP15:

- The key element, and challenge, in the planning was the cooperation with other authorities concerned with security, which had not previously worked with radiation protection authorities.
- Training of radiation experts together with testing and preparation of equipment was of great value. It is important to test and practice any equipment prior to use during the operation.
- It is vital that instruments are purchased and tested in adequate time before an event, and that all radiation experts are familiar with the equipment used.
- It is important to recognise problems with innocent alarms and establish procedures for dealing with these. All innocent alarms recorded during COP15 came from radioisotopes for medical diagnosis or treatment. These cases were quickly recognised since a highly advances HR Ge-detector was used directly at the first “line of defence”.
- It is likewise important to reduce the possibility for innocent alarms by reducing transport of radioactive sources during the event and to require notification of use of sources in the public room.
- Considerations should be given to possible radionuclides in radio-pharmaceuticals. A nuclide library as complete as possible should be established and radiation experts should be trained accordingly.

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International action plan for strengthening the international preparedness and response system for nuclear and radiological emergencies

McClelland, Vince
UNITED STATES

Testing of a portal monitor to detect illicit trafficking of anthropogenic radioactivity in operational field use

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Abstract

Measurements were conducted at a cargo container checkpoint of the German Customs Office at the Hamburg Harbour. At this checkpoint a radiation portal monitor (RTM 910 of MIRION Technologies) has previously been installed. The RTM 910 is capable of neutron and gamma detection with a low energy-resolution. To detect the gamma radiation plastic scintillators are used. In this field experiment 30 to 40 cargo containers were controlled per hour. In this time approximately three alarms per hour were given. To examine what kind of radionuclides caused these alarms, it was necessary to analyse spectra of the radiation when such an alarm occurred. For the identification of the radioactive substances two other gamma detectors were used in addition to the RTM 910. One detector was a high purity germanium detector with an energy resolution of better than 2 keV. The other detector is based on sodium iodide and has an energy resolution of about 10 keV. Both were used to take spectra exactly at the time an alarm was given by the RTM 910 and to identify the alarm causing radionuclides by analysing the spectra with two different programmes. All the detected radionuclides are naturally occurring.

In addition measurements with the germanium detector were conducted in order to estimate radionuclides which cause the background.

Introduction

To detect illicit trafficking of radioactive material, radiation portal monitors are used at border crossings. The currently most widely used detector material is plastic scintillator. These plastic scintillators are sufficient for gross counting but they have disadvantages in nuclide identification due to their low energy resolution. Therefore their performance is compare to detectors with much higher resolution like e.g. sodium iodide (NaI) or high purity germanium (HPGe). This comparison helps to estimate how an appropriate use of such detections systems is possible under operational field conditions. Therefore spectra, recorded with such high resolution detectors, have been analysed regarding possible radionuclides. Furthermore measurements have been conducted to get an overview about naturally occurring radioactive materials (NORM) like e.g. potassium or the products of the uranium and thorium natural decay series which lead to an appreciable radiation background.

Material and methods

To get a first overview on the situation of cargo-control, measurements were conducted at the cargo container checkpoint of the German Customs Office at the Hamburg Harbour.

At this checkpoint a radiation portal monitor has been installed. This monitor, RTM 910 which was developed by MIRION Technologies, is capable for neutron- and gamma-detection with a low energy-resolution. To detect gamma radiation and to produce signals proportional to the intensity of the radiation, plastic scintillators are used. The signals are then collected by a central monitoring system (CeMoSys).

Thereafter the signals are analysed and compared to the radiation background. If the adjustable alarm limit is exceeded an alarm is given. The measured radiation is divided into six ranges of energy. This allows to discriminate between certain types of radionuclides (e.g. NORM or SNM: Special Nuclear Material).

The RTM 910 has two detectors (each detector 50 x 80 x 5 cm plastic scintillator) on both sides of an incoming street to the checkpoint. The distance between the two detector parts is six metres and the cargo containers are transported through this portal monitor with an average velocity of 30 km/h.

In the field experiment 30 to 40 cargo containers were controlled per hour. In this time approximately three alarms were given by the CeMoSys. To examine what kind of radionuclides caused these alarms, it was necessary to analyse spectra of the radiation when an alarm occurred.

For the identification of the radioactive substances two other detectors were used in addition to the RTM 910. These detectors were high-resolution spectrometric gamma detection systems.

One detector is called ReGeM and was produced by Canberra. This is a high purity germanium detector (coaxial germanium cristal, length: 59 mm, diameter: 60.5 mm) with an energy resolution of 0.3% at 662 keV. The other detector (Spir Ident, produced by Mirion Technologies) based on sodium iodide (40 x 10 x 5 cm) and has an energy resolution of 8% at 662 keV.

The idea was to take spectra exactly at the time an alarm was given by the RTM 910 and to discover the alarm causing radionuclides by analysing the spectra. To record the spectra simultaneously to RTM alarms it was helpful that the Spir Ident could be triggered on radiation above background level.

It was not possible to trigger the ReGeM on radiation increase in order to inspect the relevant containers. Therefore short time spectra (3- 4 seconds) had to be taken manually.

All measurements were conducted during a period of ten days. In this period approximately 320 cargo containers were inspected per day. During one hour, three to four alarms were given by the RTM 910, but due to problems with the central monitoring system only a small amount was recorded.

The second purpose was to determine the naturally occurring radionuclides which causes the radiation background. For that purpose 24 spectra had been taken with the ReGeM detector over 20 minutes each. The ReGeM detector was most suitable for this due to its high energy resolution.

Results

The results of the measurements with the Spir Ident are quite satisfying. It was possible to discover radionuclides in the spectra that were simultaneously recorded to RTM alarms. The identified radionuclides were potassium and products of the thorium and uranium decay chains. All these radionuclides are classified as NORM. The spectra were analysed with two different programmes; SMI (a product of Mirion Technologies) and Genie-2000 (a product of Canberra).

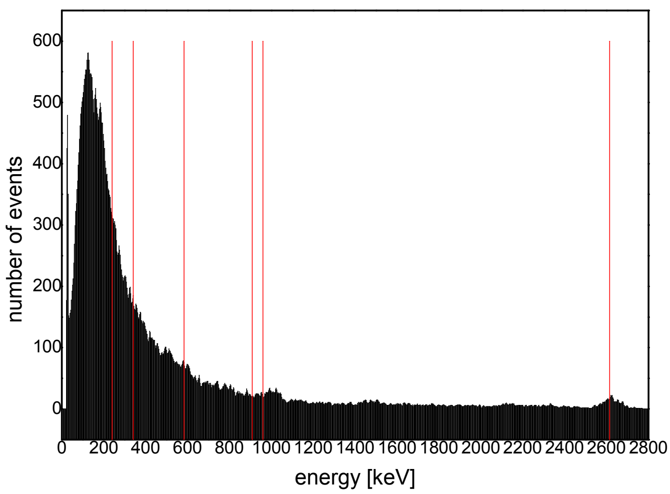

Fig. 1. Spectrum obtained with the Spir Ident due to an increase of background radiation and simultaneously recorded to an alarm given by the RTM 910. The positions of peaks related to the characteristic energetic lines of Th-228 are marked with red lines. Only at 2614 keV a peak was found, but it was sufficient to identify the presence of Th-228 with a probability of 60% by the key line of its daughter product Tl-208.

As written above all alarms can be explained with naturally occurring radioactive materials. The identification probability was 20 – 30 % in case of uranium (four times detected), 40 – 90 % in case of thorium (three times detected) and 40 – 70 % in case of potassium (ten times detected). The results of the SMI software were confirmed with the Genie 2000 Software.

The short time spectra recorded with the ReGeM detector were not usable since the measured rate of radiation was too small in order to get statistically significant information on possible radionuclides, even if several channels were aggregated.

The estimation of background causing radionuclides was analogous to the spectrum analysis before. Based on 24 spectra recorded with the ReGeM detector (each measurement time 20 minutes), the presence of several radionuclides has been determined. These radionuclides occur also naturally and were again nuclides like potassium or the products of primordial radionuclide decay chains, e.g. Bi-214 or Pb-14. Table 1 shows occurring naturally radionuclides as they were found in the recorded spectra.

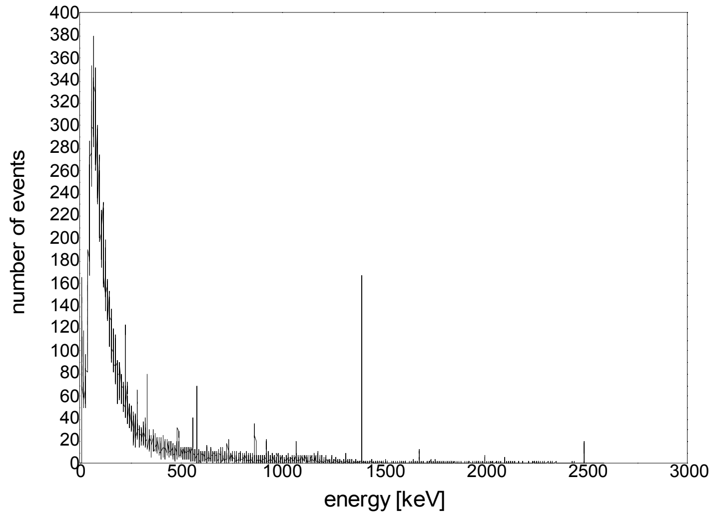

Fig. 2. Spectrum obtained with the Regem detector. The duration of measurement was 20 minutes.

Table 1. Radionuclides, discovered in background spectrum shown in figure 2 recorded with the ReGeM detector. As an example the count rate obtained from the spectrum shown in Fig. 2 is given for each emission line.

Radionuclide	Emission line [keV]	Count rate
K 40	1461	1030
Pb 214	295.224	88
Pb 214	351.932	291
Bi 214	609.312	253
Bi 214	768.356	49
Bi 214	1120.287	59
Bi 214	1238.110	35
Bi 214	1377.669	40
Bi 214	1764.494	71
Bi 214	2204.21	18
Pb 212	238.632	273
Tl 208	510.77	124
Tl 208	583.191	184
Tl 208	2614.533	155
Av 228	911.204	128
Ac 228	968.971	77

Discussion

All identified nuclides (10 x potassium, 4 x uranium, 3 x thorium), which are supposed to be the reasons for the alarms given by the RTM 910, occurred naturally. They are often contained in building materials but even they can be found in litter or K-40 fertilizer. Since such materials are often transported through the Hamburg Harbour their occurrence is not anomalous.

Conclusions

The Spir Ident detector is suitable for container inspections; every RTM910 alarm was also simultaneously accompanied by a detection of a higher radiation background by the Spir Ident and instantaneous nuclide identification with SMI.

The ReGeM detector was not suitable for container control because of the low count rate that did not allow for any detection. The detection efficiency of a germanium detector is too low for short time measurements. However the ReGeM detector was suitable to estimate natural radionuclides that occur in the measurement area permanently. Therefore 24 spectra had been analysed. It was possibly to identify the typically occurring radionuclides that are part of natural decay chains of uranium or thorium. These spectra can further be used for consistency checks of e.g. radiation background estimations.

Acknowledgements:

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Detection of radiation sources and assessment of measurement signals for nuclear security

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Abstract
Nuclear security is an important item at major public events and political meetings. The organizers and the law enforcement have to guarantee that no radioactive material is hidden at the venues. Portal monitoring is effective in controlling flow of people and goods. Alone, however, it is not sufficient for radiological security. Mobile measurements have to be carried out during pre-event search and during the event itself. Radionuclide detection capability is needed inside and outside the venues, and also at the major traffic nodal points. Mobile measurements are technically demanding.

The Radiation and Nuclear Safety Authority (STUK) started a project, known as VASIKKA, for in-field measurements. With EU-support another project, SNITCH, was launched for data management. Together they are intended to detect criminal use of radionuclides at the major public events and political meetings. End-users of the system are law enforcement agencies.

VASIKKA and SNITCH secure critical venues through integrated radionuclide detection capability, based on mobile monitoring, efficient data communication (in-field sensors) and data management, including automated data exchange from database to database via Internet. SNITCH transfers the alarms and the key analysis results immediately to experts or to duty officer. This gives timely response, based on assessment of the key findings, for the law enforcement against unauthorized or criminal acts related to nuclear or other radioactive materials.

Radiation measurement system VASIKKA
A backpack size measurement system called VASIKKA has been designed to be used by the field teams. The system consists of a gamma-ray spectrometer, a neutron count rate detector and a rugged data collection and management computer. $LaBr_3$ scintillation detector is used in the gamma spectrometer. The spectrometer has good energy resolution (< 3 % @ 662 keV) and performs well in different environments.

The measurement system uses a short acquisition time to enable mobile measurements with good spatial resolution even with relatively high speeds. A robust summation algorithm is the basis of the peak detection. VASIKKA has three modes of operation, fast, medium and slow:

- SRCH (4 s)
- MON1 (40 s)
- MON2 (400 s)

The user does not have to select any measuring mode; the monitoring modes run in the background, and provide alarms when the triggering level has been exceeded. The alarm is based on the concept of peak significance (Table 1).

Table 1. Confidence on nuclide identification in VASIKKA software. The initial categorization is based on peak information. In addition, nuclide-specific discard rules are implemented to avoid false identification. For definition of the peak significance *S*, see text.

Confidence level	Number of peaks	Rule	Comment
1	1	$1 \leq S \leq 1.5$	Small peak (H_0)
2	1	$1.5 < S \leq 2$	Small but clear peak (H_0)
3	1	$S > 2$	Unequivocal peak (H_0) - no other information on nuclide ID. Typically ^{137}Cs @ 661 keV
4	2	$1 \leq S_{1,2} \leq 1.5$	Two small peaks (H_0)
5	2	$1.5 < S_1 \leq 2$ $1 \leq S_2 \leq 2$	Two small peaks, at least one of them clear (H_0)
6	2	$S_1 > 2$ $S_2 \geq 1$	Two peaks, at least one of them unequivocal (H_0)
7	≥ 3	$S_1 > 2$ $S_2 > 2$ $S_3 \geq 1$	Three or more peaks, two of them unequivocal (H_0)
8	2	$S_1 > 2$ $S_2 > 2$ $0.7 < A_1/A_2 < 1.3$	The ratio of the measured peak areas is within 30 % as predicted by the decay data and the efficiency curve. No low energy line is considered.
9	≥ 3	$S_1 > 2$ $S_2 > 2$ $S_3 > 2$ $0.7 < A_i/A_j < 1.3$	The ratios of the measured peak areas of three most significant peaks are within 30 % as predicted by the decay data and the efficiency curve. No low energy line is considered.
10		Human review	

The peak significance (S) is defined as the normalized ratio of the peak area and its standard deviation. S is calculated in such away that it receives a value of 1 at the false positive risk level of 1:10^6 ; this means that the normalization factor is the abscissa $k_\alpha = 4.75$ of the Gaussian distribution.

All measurement and analysis results are stored in the local LINSSI database running in the data management computer.

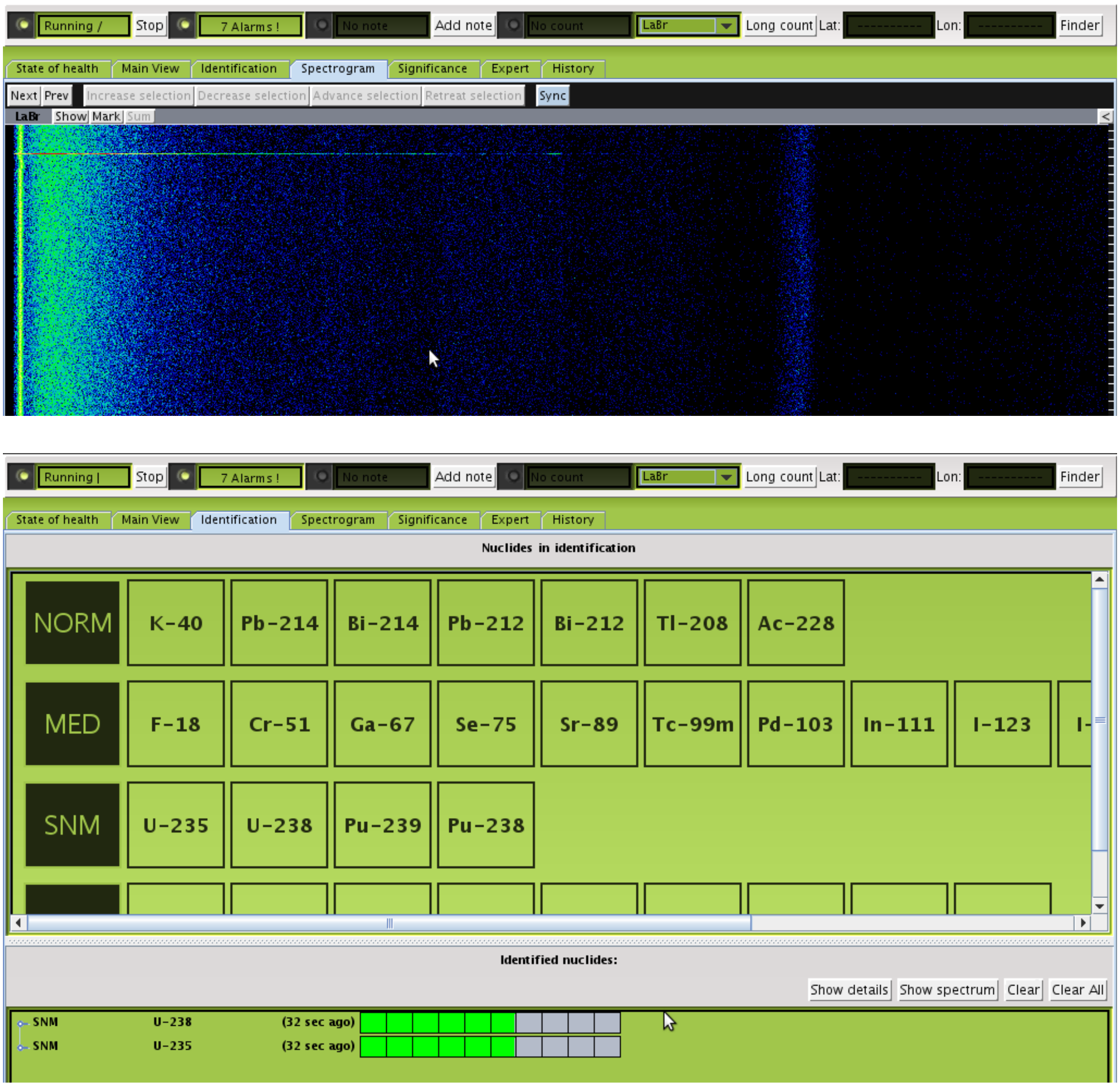

Figure 1. Screenshots from the VASIKKA user interface.

VASIKKA implements an automated connection to the SNITCH remote data management and analysis system. Several automated data transfer modes can be used to send the data to the reachback facility (e.g. all measurements, measurements with signals above the preset alarm limits, manual long measurements only) depending on the band width of the data transfer link. If the on-line data transfer is not possible, the measurements can be saved to a memory stick and sent manually to the reachback facility.

SNITCH – a data management system

An important element of integrated radionuclide detection capability is mobile monitoring. SNITCH provides efficient data communication and management, including automated data exchange from database to database via Internet. SNITCH transfers the alarms and the key analysis results immediately to Command and Control for alarm handling. This gives timely response capability for law enforcement against unauthorized or criminal acts related to nuclear or other radioactive materials. SNITCH also provides means for reachback functionality to expert organizations. Measurement results can be sent manually or automatically to an expert organization for detailed analysis and the analysis results can be sent back to the field team.

Main usage model of the SNITCH system can be described as "multiple users – single expert" model. The expert in the model can be a person or the automated analysis pipeline inside the SNITCH system. The automated pipeline can be regarded as an expert because the SNITCH system server may implement more powerful analysis tools than the simple mobile devices can.

SNITCH provides a user-friendly way to exchange measurement data and analysis results. Depending on the level of integration of the measurement device to the SNITCH system, the data exchange can even be fully automatic and no user actions are needed. However, data can be sent to the SNITCH system manually and the results can be retrieved manually also.

SNITCH is built around the LINSSI database version 2.2 and thus the native data exchange method is the exchange of LINSSI markup language files (lml-files). In addition to lml-files, four commonly used open file formats are supported: ANSI 42.42 (n42-files), IEC 61455 (iec-files), CTBTO IMS 2.0 pulse height data (phd-files) and IAEA ASCII file format recommendation for gamma spectrometers (spe-files). However, only lml-files contain analysis results.

Inside the SNITCH system the processing of the data are broken down into small sub processes that each work in "first in first out" (FIFO) fashion. This means that they will process the tasks they receive in the order that they received the tasks.

Data exchange interfaces in the SNITCH system

Currently two data exchange (upload) interfaces have been implemented: email message interface and web browser interface. Email messages are versatile enough to support SNITCH data exchange. All supported file formats are basic ASCII text files; therefore, they can be sent as message body text, but file attachments are also supported. Multiple files can be sent in single message as file attachments, where as only one file can be sent in the message body. Also the additional information files can be sent as email attachments, if other data file formats than lml-files are sent. The subject field of the email message is used for the processing instructions. Currently only the database and analysis methods are selected based on the keywords in the subject field. Sending of email messages can also be automated with scripting languages and mail sending software that has a command line interface.

Web browser interface is mainly used for manual data insertion. Web browser file upload is a well known and simple way to upload data. Web browser is also a great way to collect additional information needed when uploading files other than lml-files. Additional data can also be checked and validated before the uploaded data files are inserted to the data processing. The additional data are also saved to the server into the user profile, so that user does not have to input the same data several times. Web browser interface has intrinsic encryption feature through the https protocol. This eliminates the possibility to capture the data in transit. The web browser interface is a safe way to send classified data.

Web services interface is used for the fully automated machine to machine data transfer. Several services can be provided. For example, different service will be provided for each of the supported file formats, because they all have different additional data needs and they must be inserted into different preprocessing FIFOs. Web services is implemented with the SOAP protocol that can be transmitted over the https protocol. Thus, the inherent encryption features of the https can again be used to prevent in transit data capture and simple username and password can be used for the user authentication.

Data processing in the SNITCH system

Data processing in the SNITCH system has been broken down into small sub processes. Each of the sub processes process the data in FIFO manner. However, this does not ensure that the data flowing through the multiple FIFOs is processed in FIFO manner as a whole. Data entering the data processing first might not come out first if some branching/splitting occurs. Also all sub process steps process data simultaneously, so several incoming measurement data messages are processed simultaneously, if needed.

Data processing has the following generic sub processes

1. file format conversion to lml
2. upload lml-file to database
3. notify upload
4. run automated analysis
5. send email messages

Some of the sub processes are used several times in the processing of the incoming data. Generic flow of data is show in Figure 2. The experts can do manual analysis to the incoming measurement data or they can review the results produced by the automated analysis sub processes and add their own comments to the final results.

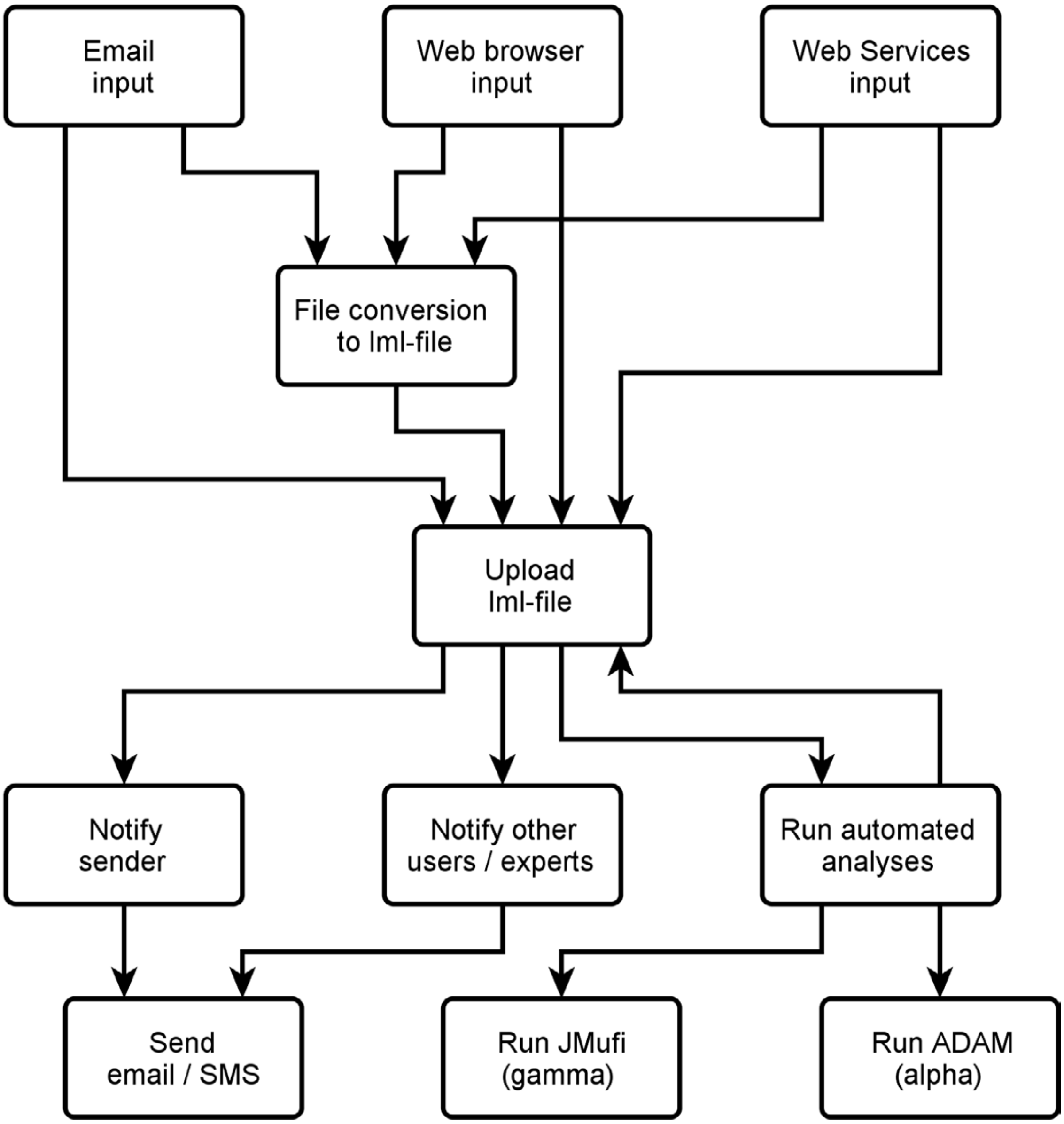

Figure 2. Generic flow of data in the SNITCH system. SNITCH can launch different analyses processes; two of them are shown, JMufi for gamma spectrometry and ADAM for alpha spectrometry.

SNITCH training package

A crucial factor in the successful deployment of the SNITCH reach back capability is the ability of the field team to make the relevant radiation measurements and to retrieve the expert analysis results and recommendations from the SNITCH system. The law enforcement officers in the field team need some additional training to operate the radiation measurement system and the SNITCH data management system. To address this issue, a training package has been composed. The training package will give the law enforcement officers the basic knowledge needed to make the radiation measurements safely and properly, so that the radiation expert in the remote location can trust the validity of the measurement data received through the SNITCH system. The training package comprises the following subjects:

1. SNITCH data management system;
2. Basic training on radiation, radioactive material and radiation protection;
3. Radiation measurement and surveillance;
4. Operational awareness in RN-situations;
5. Security screening of places, people and goods; and
6. seven different scenarios involving radioactive material.

The first three items are designed to give the basic knowledge on the SNITCH system, radiation safety and radiation measurement techniques. Operational awareness in RN-situations module will teach how to collect and process information other than radiation measurement information to support the evaluation of the situation. The Security screening module provides useful radiation measurement tactics that give reliable results when places, peoples and goods need to be screened with limited resources and timetable. In this case the reliability means that radioactive materials are found or absence of them can be proved with adequate certainty. Scenarios in the sixth module include criminal usage of RN-material with varying threat levels and amounts of material actually used.

VASIKKA - SNITCH interplay

Data flow between the field team using VASIKKA measurement system and the reachback expert facility using the SNITCH data management system is shown in Figure 3. The process is

1. Field team makes a spectrometric measurement of the suspect material or area.
2. Measurement is automatically analysed by VASIKKA software. In a simple case the measurement is analyzed correctly and the field team can jump directly to step 7.
3. However, in a complex case the measurement cannot be fully analyzed in the field and the measurement data (with the results of the automated analysis) are sent automatically to the remote expert facility.
4. Automated data handling will store the data to the LINSSI database in the SNITCH server, makes the predefined automated analyses and alerts the expert either via email or SMS. Alert to the expert can be an optional step if more powerful automated analysis tools are needed.
5. Expert on duty can look into the results of the automated analyses and make additional analyses manually. Expert on duty can insert his remarks and recommendations to the final analysis results.
6. Final analysis results with the remarks and recommendations of the expert on duty are transferred automatically back to the VASIKKA system used by the field team.
7. Field team takes appropriate actions based on the received final analysis results and/or the recommendations of the expert on duty.

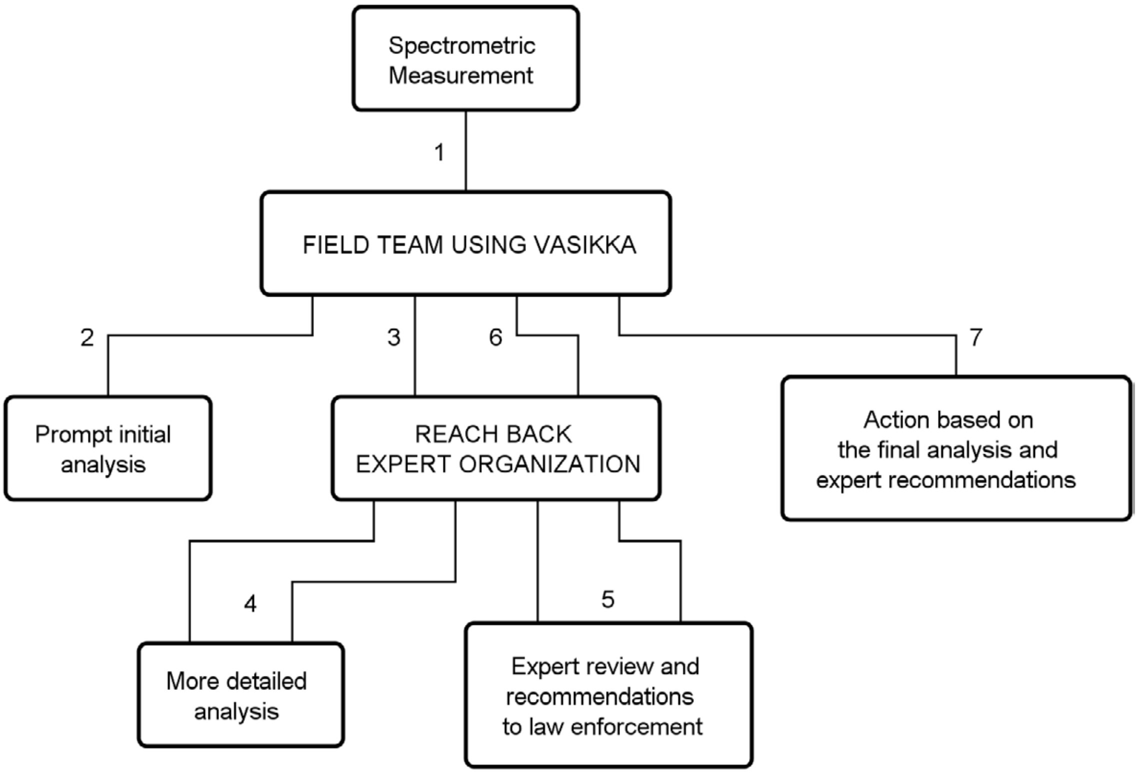

Figure 3. Integrated measurement and remote analysis concept.

Manual use of the SNTICH system is also possible. In step 3 the data can be sent via email or web page and the feedback (steps 6 and 7) can also be received via email or a web page.

Conclusions

The automated VASIKKA – SNITCH mobile measurement and remote data management system is a valuable tool in a radiation incident management. It increases the reliability of the correct interpretation of the measurement made in the field. The system has already been tested in real operational conditions by a Finnish counter terrorist unit, and the performance has been satisfactory.

Acknowlegements

SNITCH system has been developed with financial support from the European Commission - Directorate-General Justice, Freedom and Security under the Prevention of and Fight Against Crime Programme 2007.

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Indoor positioning for nuclear security

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Abstract
Mobile measurements are needed to search for nuclear material out of regulatory control at major public events and political meetings. The mobile measurement teams may have to screen hotels, living quarters and other venues before the event and during the event. Measurement and positioning information are crucial for planning the mission and for reporting the findings.

Traditionally the global satellite navigation system is used for positioning outdoors. Positioning indoors is difficult. Various indoor positioning technologies are available on the market. However, they often need a specific infrastructure installed in the building which limits their use.

Radiation and Nuclear Safety Authority (STUK) has performed indoor positioning tests with Ekahau's wireless network based system. The experiences with the field tests were encouraging. Also a novel indoor positioning system has been developed by VTI Technologies. In this system navigation is based on the measurement of the length and direction of every step of a person. It uses a chest-worn speed and distance measurement module, originally developed for the sports market, together with an instrument-grade gyroscope and a magnetometer.

Introduction
Searching for nuclear material is of utmost importance in security arrangements at major public events or political meetings. The mobile measurement teams go through hotels and event venues and search for radioactive material. STUK has developed a portable device, VASIKKA, for mobile measurements. The measurement system is based on spectrometers that write measurements and analysis results to the LINSSI database (a database for gamma-ray spectrometry). Simultaneously the navigation system writes the positioning data to the same database. The measurements and position data are linked together with time stamps. The idea of separating the radiological measurement system from the navigation system makes it possible to use various kinds of positioning systems for indoor and outdoor applications.

The spectra and measurement analysis results together with the team locations are transmitted to the command and control centre for further analysis. The results of the measurements are shown on screen, including the movement of the teams on the online digital map. In case radioactive material is found, an alarm is generated and, the mapping system indicates the place on the map. The command and control centre uses the information and gives instructions to the field teams.

Indoor positioning

Traditionally a global satellite navigation system (e.g. GPS) is used for positioning. It works well outdoors, but positioning inside a building is challenging. Navigation is difficult or impossible in areas where satellite reception is limited. Indoors navigation could be carried out manually by pointing out the current position on the digital layout of the building. In this case there is always a risk for a human error. The possibility of loosing track increases when the person doing the measurement is not familiar with the building to be searched.

There are various indoor positioning techniques available. They are usually meant for tracking goods in a warehouse or tracking people at a hospital. They need a specific infrastructure installed in the building or a time-consuming model of the building before they can be used. For authority work, the navigation system should be easy to set up and easy to use in buildings where there are no pre-installed techniques available.

Ekahau's Real-Time Location System

Ekahau is a Finnish company specialized in indoor positioning systems. One of their products, the Real-Time Location System (RTLS), has been tested together with VASIKKA in mobile measurement field tests. The Ekahau RTLS is a Wi-Fi-based positioning solution. It uses existing wireless local area network infrastructure. The Ekahau RTLS consists of the Ekahau Positioning Engine (EPE) server software, the Ekahau Location Survey (ELS) and battery powered Wi-Fi tags. Ekahau RTLS uses existing Wi-Fi standard access points as the reference devices for tag location.

Before Ekahau Wi-FI positioning is operational in a new environment, a positioning model of the building has to be made. It requires a layout of the building in digital format. The site survey is made by walking through pre-defined tracks and by registering the available wireless network access points. When the model is completed it is uploaded to the EPE server.

The Ekahau RTLS was used in field test in the STUK head office and in the Itäkeskus shopping centre. In the tests VASIKKA was used for radiological measurements and a separate Java program, IPARM (Indoor Positioning Aware Radiation Measurement), as a navigation system (Garlacz 2009). IPARM retrieves position data from the Ekahau server and writes it to the LINSSI database. The average accuracy of the Ekahau RTLS is one to three metres.

The initial test results on the Ekahau system were promising (Syrjälä 2009). The first field test was carried out in the STUK head office (Fig. 1). A Cs-137 calibration source was used in the test. The measurement team walked along the corridors and passed the table where the source was placed. The system clearly detected the source and created an alarm. The alarm is shown as a pink spot on the map. The test proved that the accuracy of the positioning is good enough to find even low-active radiation sources.

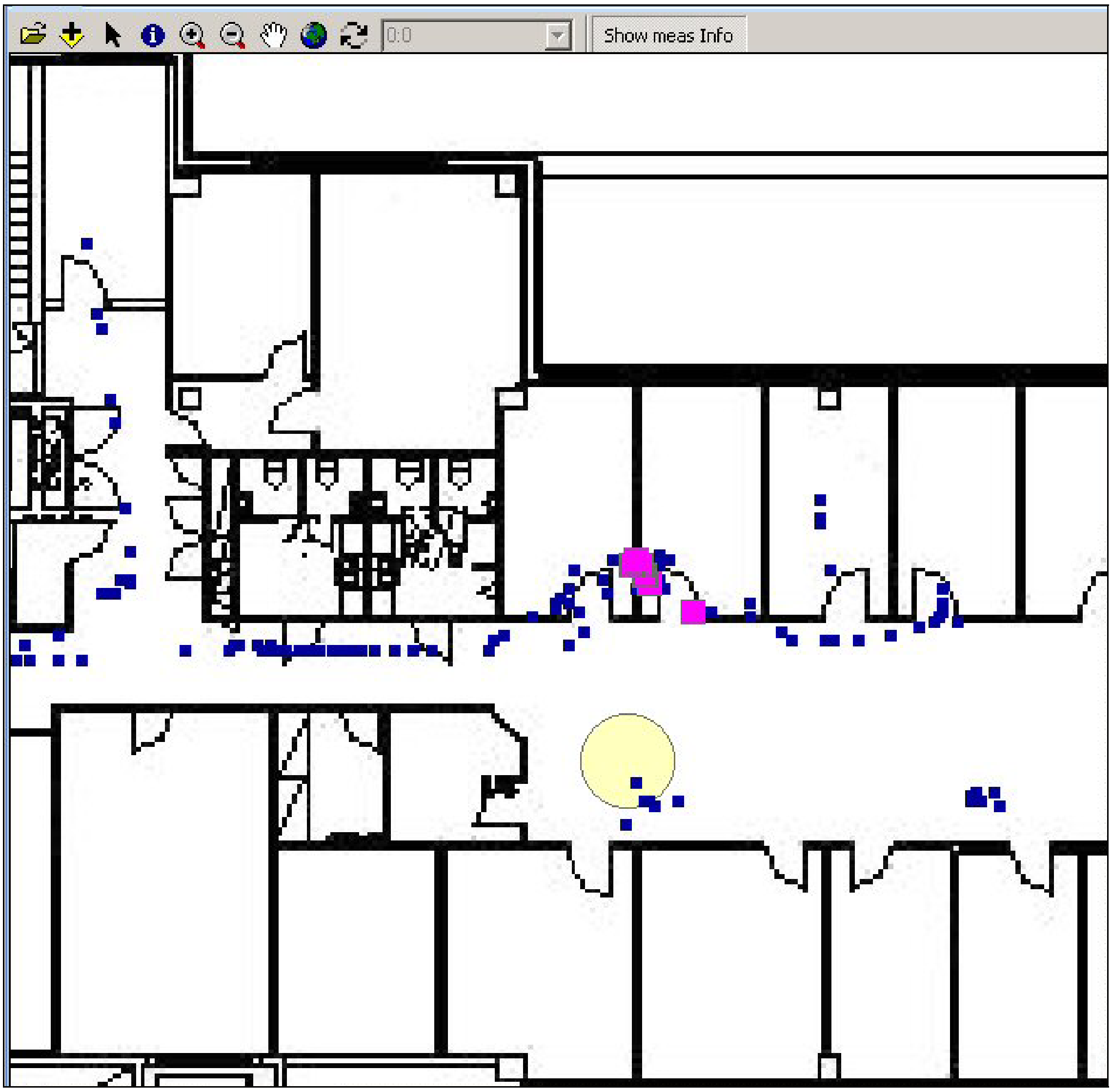

Fig. 1. Ekahau RTSL used together with VASIKKA in field test. In the test Cs-137 calibration source was placed on a table in the corridor (yellow circle).

The Ekahau positioning system has also been tested in the Itäkeskus shopping centre (Garlacz 2009). The first test was carried out in June 2009 and the second test period was in February 2010. The positioning system worked well, but the complex environment made the building of the model more problematic. The constructions in the shopping centre caused errors in the connections between the Wi-Fi tags and the positioning server. It was also noticed that the wireless network access points of the shops were not in the fixed places they were supposed to be. Some of the access points had been moved to a different place and, therefore, the model had to be rebuilt.

With the wireless network technique, building the model and defining the wireless network access points can be time-consuming. If the wireless network environment is alternating, the model may have to be rebuilt every time the model is used for navigation to ensure best positioning accuracy. Therefore, it is suitable for stable environments where there is a fixed wireless network.

Step navigation by VTI Technologies

A novel indoor positioning system has been developed by VTI Technologies. In this system the navigation is based on the measurement of the length and direction of a person's every step. The step length is measured using a MEMS (Micro Electro Mechanical System) accelerometer and a novel and robust algorithm converting acceleration to horizontal speed. The direction of each step is measured with a MEMS angular rate sensor. The initial direction could be taken, for example, from some well-defined event or using radio navigation (e.g. GPS). If an electronic map of the premises is available, active map matching and filtering can be used to get initial position and direction or improve overall accuracy.

Results from the step navigation prototype test below indicate an accuracy of a couple of meters during a 5-minute walk with a total of 2520º turns.

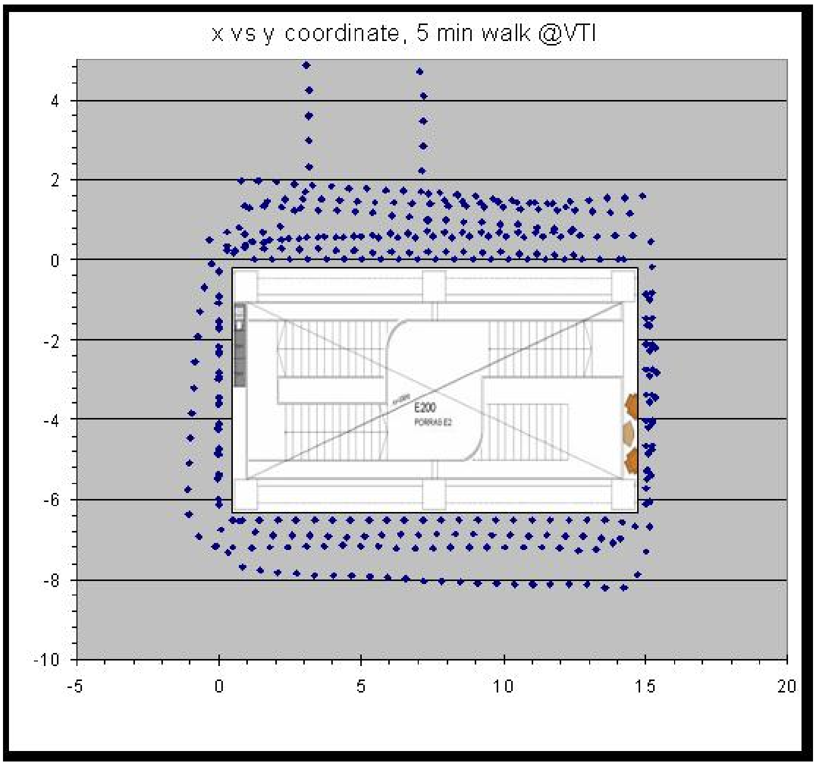

Fig. 2. Results from a short step navigation session at VTI premises using the combination of an accelerometer and an instrument grade gyro.

There is ongoing work to improve the accuracy and robustness of the step navigation system as well as to widen its usability to other motion, like e.g. crawling.

Conclusions

Although there are some indoor positioning systems available they are not technically ready for in-field use by mobile measurement and search teams.

In the wireless network based system, building the model is time-consuming and the model has to be updated every time there are changes in the wireless network. It is not suitable for ad hoc navigation. It can be used in stable environments such as hotels or conference facilities with fixed wireless network.

The step navigation system is easy to build up but it is still under development. The idea seems good but it has not yet been tested in a real time environment.

The radiological measurement should work the same way indoors and outdoors whereas only the navigation system may vary. The online mapping system should be able to detect whether the position is indoors or outdoors. It is envisaged that in the future the user does not have to know when the system is using satellite or indoor navigation system for positioning but the software behind the system performs all the necessary actions.

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Control of nuclear materials and related radiation safety

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Abstract

Nuclear materials are special fissionable materials and source materials, namely uranium, plutonium, thorium, etc. The definition of nuclear material is very often related to implementation of nuclear safeguards. Handling nuclear materials is strongly regulated by national legislations, e.g. laws regulating nuclear safety or physical protection, as well as by international agreements e.g. EURATOM Treaty from 1957, Non – Proliferation Treaty from 1970, Convention on the Physical Protection of Nuclear Material from 1979. The material can be found at different premises, namely laboratories, storages, workshops etc. When handling such material not only radiation risk should be taken into account but also other risks. A comprehensive analysis of all risks should be done and it is vital that harmonised safety measures are applied. The regulations of nuclear material are changing from the first regulations in the middle of the last century, when such material was called "a product" in the Manhattan project. Especially after the Second World War the nuclear material became a subject of very intensive investigations and as a consequence the development of regulations followed. A special attention to nuclear materials was intensified after 11 September 2001 regarding serious threats. The article gives an overview of the development of regulations related to nuclear materials taking in parallel the development of radiation safety standards. It focuses on specifics of different national and international definitions of nuclear materials as well as on a use of other terms related to nuclear materials, e.g. special nuclear materials, source material, byproduct material. An overview of relation between present standards regarding nuclear safeguards and radiation safety requirements (ICRP 103, EU BSS, IAEA BSS) is given, taking into account a control of the material from its cradle to grave, e.g. finally to its disposal as radioactive waste.

Direct Alpha Analysis for Forensic Samples (DAAFS): Techniques, applications, and results

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Abstract

The goals of the DAAFS project are to develop new methods of sample acquisition and analysis for in-situ measurement of alpha radiation. In-situ alpha tests of DAAFS have been performed using either a well-characterised swipe technique with a fluoropore membrane filter, or a vacuum sampling system and optional surface impactor. The unique components of the DAAFS system include: 1) a variety of specialized sampling techniques, 2) a Monte Carlo spectral acquisition simulation software called Advanced Alphaspectrometric SImulation (AASI) that simulates the alpha spectra by using the specific geometry and sample collection parameters, 3) an Advanced Deconvolution of Alpha Multiplets (ADAM) software package for fitting actual sample spectra, and 4) an advanced MySQL database management/telemetry subsystem called LINux System for Spectral Information (LINSSI) with alarming and notification via email and SMS.

In addition to the nuclear safeguards and security design goals, other potential uses for DAAFS are: thoron progeny identification and possibly quantification of alpha emitters in drinking water, urine analysis, decommissioning and contamination measurements in nuclear reactors and laboratories.

Introduction

Currently, analysis of alpha emitters in field samples in either a nuclear security or safeguards context requires time consuming full radioisotope chemical separation to obtain a high level of specificity. Additionally, the ability to perform high-throughput alpha analysis capability became apparent during the Litvinenko ^{210}Po (a pure alpha emitter) incident in the UK. The DAAFS system was designed to function in a mobile laboratory as a platform for rapid direct measurement of samples. The system will be fully tested through field and lab exercises by the end of the project in 2011.

Conventional alpha spectroscopy is not well suited to nuclear security applications when a timely response is desired. Traditional techniques employed in alpha spectroscopy often require radioisotope chemical separation methods – a time consuming process when rapid response time is essential. The requirement for full laboratory support to perform conventional techniques is neither amenable nor practical in field deployment, on-site analysis, or other common situations where malevolent use of alpha radiation has occurred. In the event of a Radioactive Dispersion Device (RDD) incident, rapid alpha assessment would be an ideal capability to have to detect radioactive materials that are easier to identify and quantify using alpha detection such as ^{241}Am.

The direct spectral measurement methodology of DAAFS is very different from conventional alpha spectroscopy techniques in a number of key areas. The primary difference between the DAAFS approach and conventional techniques are the acquisition and handling of samples. Samples for analysis are collected in a highly controlled and repeatable fashion to minimise sample variations. Minimising sample variability provides consistent input to the ADAM spectral deconvolution (Toivonen et al. 2009) and AASI (Siiskonen and Pöllänen 2005) simulation software. Samples are not subjected to any radiochemical treatments, making the DAAFS process a non-destructive analysis technique, resulting in relatively rapid (on the order of minutes to hours) assessment. The non-destructive nature of DAAFS is ideal as evidence is intact for criminal prosecution. Additionally, the ability to perform other radio analyses on the same sample can be very valuable as it minimises the number of samples required along with the necessary sampling equipment. Both of these are important features for incident response. Training first responders in common sample collection techniques minimises the training and knowledge level required of non-expert personnel.

The equipment for performing direct measurement of samples was designed to be field-portable. The measurement equipment used for DAAFS is shown in Figure 1. The system is compact and is easily portable in a mobile laboratory. The materials used to perform sampling vary depending on application. Figure 1 shows the largest piece of sampling equipment for DAAFS - the Lilliput vacuum sampler. All the other sampling equipment is more portable, requiring only a single person to operate and perform all procedures.

Figure 1. DAAFS spectral acquisition equipment (left) and sampling of clothing from pre-Olympic Exercise with Lilliput vacuum and nozzle (right).

DAAFS was designed for rapid assessment, and as such, it emphasizes qualitative observations over quantitative. Risks to the public and responders are known as efficiently as possible, when a positive result occurs, the quantitative results can be requested via conventional radiochemistry techniques performed by a laboratory. The

Litvinenko incident, where a pure alpha emitter, ^{210}Po, was used as a poison is a prime example where application of DAAFS technology would have been more useful than conventional techniques to identify the hazard, rapidly define and assess the crime scene, and indicate areas needing decontamination.

During field use, a single non-expert can perform the sampling and spectra acquisition with a brief training session, with minor advice coming from off-site if necessary. Once the spectrum is acquired, the DAAFS system is designed so that all sample and measurement associated data can be sent via e-mail to offsite experts for analysis. This “reachback” capability is made possible through the use of an open source database using the LINSSI database design (Aarnio et al. 2008), and computer hardware capable of wireless WLAN access.

Material and methods

DAAFS sample collection types can be broadly described as being a member of one of the following categories: swipe, air filter, and liquids. Each category has a specific methodology for sample collection to obtain optimal results in a direct alpha measurement system.

Acquiring a high quality swipe sample requires a media with the key feature of not allowing sample material to penetrate the matrix. Rather than using conventional glass fibre airfilters, a fluoropore membrane filter is used (Pöllänen and Siiskonen. Fluoropore membranes are an ideal choice as the media resists sample penetration and the resulting alpha particle energy absorption. This means superior peak resolution and allows an accurate match to the energy calibration. Swipe samples are most successful when used to sample hard surfaces.

Air filter samples are acquired using the same type of media as the swipe above, but using a vacuum either with a nozzle or without. For field use, a portable battery powered vacuum sampler called Lilliput (Senya Ltd.) is used. The vacuum provides an intake flow rate up to 12 m^3/h, and uses standard round air filters. The filters are mounted and laminated for insertion into the nozzle apparatus or direct attachment to Lilliput for aerosol sampling. A sample is collected for approximately 15 min when used in airborne collection mode. The optional nozzle is used to collect samples from textiles and other soft surfaces. Samples collected using Lilliput are acquired in triplicate for analysis using other desired techniques.

Once the sample is acquired it is measured in a standard commercial alpha spectrometer. The analyst has two novel tools at their disposal. The first tool is a spectral simulation package called AASI. The program simulates the results of the spectral acquisition process, when relevant physical parameters associated with the sample and measurement is provided. AASI allows an expert to examine the impact of parameters associated with sampling (for example particle size distribution), measurement (detection geometry), and calculation of detector efficiency and coincidence factors. More importantly, it allows an expert to determine and examine the validity of their assumptions about the sample collection and measurement process. The second and most important component is the advanced spectral fitting package, ADAM. The program performs the fitting using a Gaussian peak model with two-component exponential low-energy tail, and is capable of performing calibrations and accepts coincidence correction factors. ADAM is capable of operating in either batch or non-interactive mode, and includes a comprehensive alpha emission library based on ENSDF data.

Results

The DAAFS systems has undergone preliminary testing on many different samples and participated in one field exercise. In preparation for the 2010 Olympics in Vancouver, Canada, the DAAFS system was deployed for a simulated Olympic terrorism scenario. During the exercise, a suspect was apprehended, who had an unknown substance that was thought to be radioactive in origin on his clothes. The DAAFS system was activated to acquire a sample and perform an analysis. Lilliput acquired a sample off the suspect's clothes, which was then counted at a sample-detector distance (SDD) of 13 mm. After only 10 minutes the analyst was able to conclude that a radioactive substance was present, and that the identity of the substance was ^{241}Am. Simulations performed in AASI using 100000 decays of Am seem to reasonably approximate the observed spectra (Fig. 2).

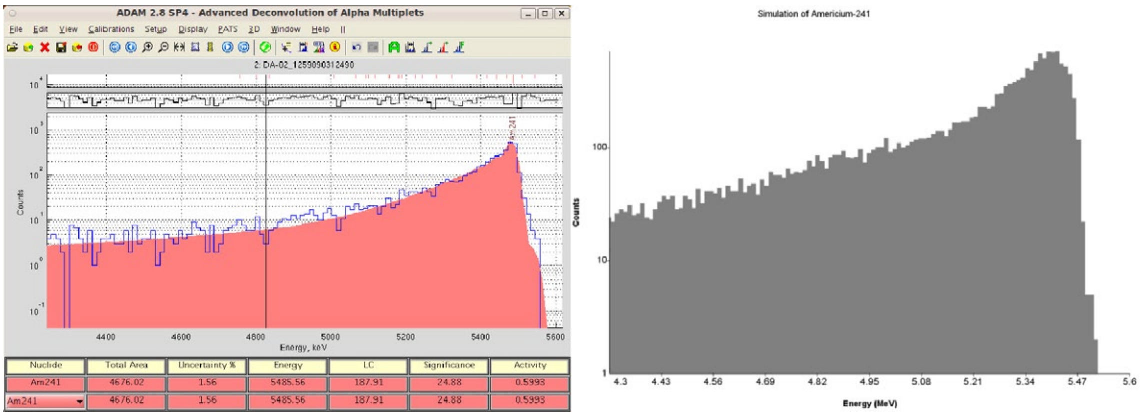

Figure 2. ADAM (left) analysis and AASI (right) simulation of Lilliput vacuum-sampled clothing from pre-Olympic Exercise showing ^{241}Am being identified.

Preliminary examinations of liquid samples were performed using an IAEA drinking water intercomparison sample that appears to have been spiked with ^{230}Th (blue). In the resulting spectra, acquired after 24 hours of direct measurement at a SDD of 13 mm, the possible presence of a ^{232}Th impurity (red) can be seen in Figure 3. The presence of the impurity is not conclusive, but is reasonable hypothesis if the thorium was separated from spent nuclear fuel.

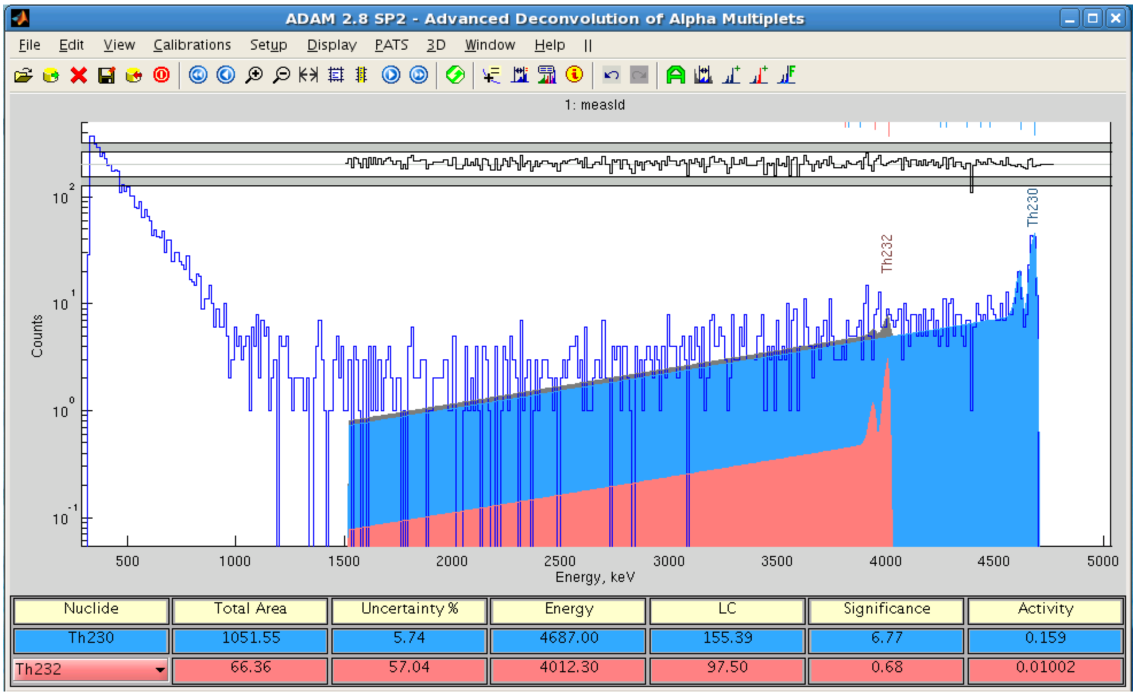

Figure 3. Analysis of an evaporated IAEA drinking water sample on planchet showing main peak Th-230 and presence of possible impurity Th-232.

The final example is an aerosol sample from a basement with high radon levels. The Lilliput vacuum sampler was used in an aerosol collection model of operation, or without using the sampling nozzle. The progeny of radon (^{212}Po – purple, ^{214}Po – yellow, ^{212}Bi – blue, ^{218}Po – red) are shown immediately after sampling with a spectral acquisition time of roughly 2 hours.

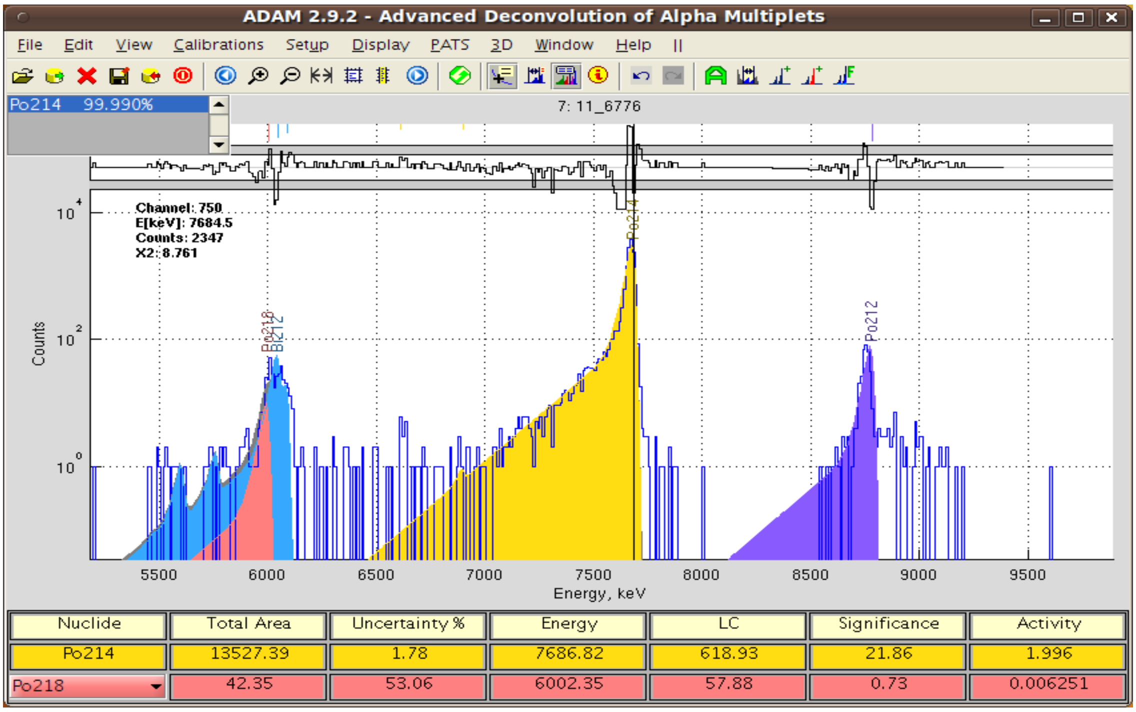

Figure 4. Analysis of aerosol sampler from a basement showing radon progeny after 2 hours acquisition.

Discussion and conclusion

DAAFS has been used on a wide variety of samples and radioisotopes to identify alpha-particle emitting isotopes. DAAFS has demonstrated success for nuclear security applications requiring qualitative assessment under specific conditions where a “thin” sample can be produced. Successful field trials illustrated the portability and off-site real-time expert support available to users of the DAAFS system. Further testing on different radioactive materials and the use of new sampling techniques such as ion specific sampling materials (3M Empore^{TM}) will be tested during the final year of the project (2010-2011). Planned trials for the DAAFS system include: depleted uranium, fresh and spent nuclear fuel, further liquid trials, and decommissioning feasibility studies.

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Explosion tests using radioactive substances

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Abstract
The results of several field tests in which the short life-time radioactive matter (RaS) was released by explosion in the free or indoor environment are presented. The primary goal of these tests was to verify the detection methods applicable for the obtaining of relevant data set – time and space distribution of the dose rate, surface and volume activities which can be used for modelling analyses of short radioactive substances dispersion by explosion.

Introduction
Potential disuse chemical, biological or radioactive substances for committing a terrorist attack have brought new dimensions into possible scenarios of terrorist attacks. Specific place belongs to so – called "dirty bomb"[1-6], i.e. radioactive substances (RaS) dispersed using a conventional explosive or any other mechanism/system.

Model analyses focused on evaluating possible consequences of a terrorist attack using a RaS are performed in a number of countries [2-6]. The National Radiation Protection Institute (SURO) in Prague in the frame of the State Office for Nuclear Safety (SUJB) grant[1] realized set of tests in which a radioactive substance (RaS) was dispersed by explosion. The main goals of these tests were:

- to verify in real conditions set methods appropriate for evaluation dispersed radioactive substance - assessment distribution dose rate, surface and volume activities, aerosols mass concentration, etc.,
- to obtain relevant set of data usable for testing and development of the mathematical codes/models fit for testing and evaluation of a propagation these substances on a short distance,
- to verify possibility realize tests and in more complex geometry (indoor environment, terrain obstacles, etc.).

The presentation is focused on the description, comparison and evaluation of the two tests carried-out in free areas (tests No. 1 and 2) and two tests in free area with

[1] Project SUJB No. 8/2008 "Methods and measures to reduce the occurrence and liquidation of consequence of terrorist abused of radioactive matters is handled by the SURO in Prague with by an extensive team of contract-bound research workers and institutions.

artificial obstacles (tests No 1P and 2P). Two tests in the small house and in the bus are shortly mentioned.

Tests characteristics

For the tests a combined controlled explosion system (industrial explosive) with the RaS dispersion in the selected space angle was chosen. As radioactive substances Tc-99m activities ≤ 1 GBq was used. Radionuclide was diluted in water coloured with potash and put (Fig. 1) in a spheric glass bottle (volume of 6 ml).

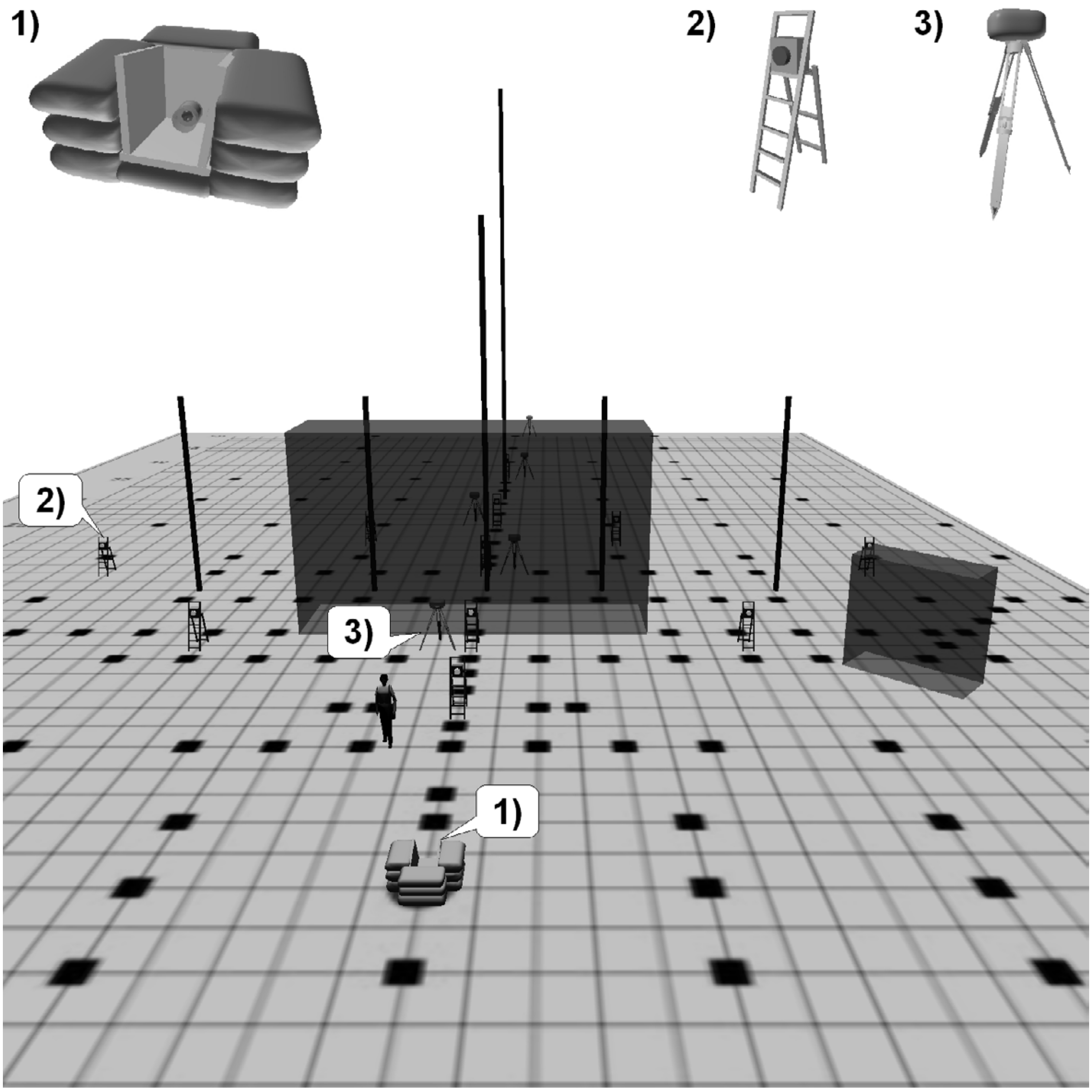

Figure 1. Location of filters, detectors and obstacles on the polygon – test 2P (table 1); the configuration of an explosive – a glass sphere with RaS is demonstrated in figure on left side above (1); No. 2 labels a localisation of aerosols collectors, and 3- DustTraks in the figure.

The survey performed and in this paper presented tests is given in the Table 1.

Tests were carried out on testing polygon of the National Institute for Nuclear, Chemical and Biological Safety (SUJCHBO) Příbram – Kamenná, based on the permit given by the SUJB and an appropriate Mining office board.

Material and methods

Optical and infrared (ThermaCAM P65 infrared imaging system) technology was used to record the time development of the scanned scene before and after a controlled explosion.

Measurement of selected meteorological values was carried out using a mobile automatic weather station with the following being measured – wind speed and direction, air pressure, temperature and humidity.

The measurement of dose rates was carried out by portable devices (GR135 miniSpec, Exploranium and NB 3201, Tesla).

Table 1. Characteristics of the tests.

Test No.	Date	Explosion time	Activity (MBq)/date, time (h)	
1	15.05.2008	11:30	1058	15.05.08, 10:10
1P	21.10.2008	12.34	900	21.10.08, 11:00
2	05.05.2009	12:22	1222	05.05.09, 12:22
2P	14.07.2009	12:42	1088	14.07.09, 11:00

The surface activities were detected using paper collection filters located densely both directly on the polygon area of app. 50x40 m^2 and in selected places on vertically placed columns (height $z \leq 12$ m). To evaluate the time distribution of the RaS dispersion, some of the filters were changed after the explosion in selected times. Up to 550 filters were located and measured in one test to evaluate the surface activities (Fig. 1). Filters activities were measured by HpGe semiconductor gamma spectrometry at 9 spectrometric chains in special shielding.

Radionuclide's volume activities (activity concentrations) in air were determined in several selected (up to 10) sites using aerosols sampling devices (SENYA - JL-150, HUNTER, DWARF 100) including a cascade impactors to determine the aerosol's size distribution.

By means of the DustTrak - DT model 8520 (TSI) laser nephelometers, a distribution of a mass concentration of an atmospheric aerosol in time with a very short (1 s) integration time was monitored - the particle masses within the size range (0.24-10) µm were effectively determined.

The more details of the tests experimental arrangements are described in [7, 8].

Results, discussion

The meteorological conditions characteristics for tests carried-out in free area with and without obstacles are summarized in Table 2.

Table 2. Summary of meteorological conditions in tests performed in free area with and without obstacles.

Test No.	1	1P	2	2P
Date	15.5.2008	21.10. 2008	5.5.2009	14.7. 2009
Explosion time	11:30	12:44	12:22	12:42
Temperature [°C]	22.2 - 22.3	14,8 - 15,05	9.7 - 10.4	23.6 - 27.9
Relative air humidity [%]	41 - 47	68 - 71	42 - 56	51-71
Condensation point [°C]	8.4 - 10.6	9,3 - 9,8	0 - 1.5	15.8 - 18.6
Wind speed [km/h]	1.2 - 6.6	0 - 3,24	1.4 - 16.2	0 - 6.5
Gusty wind speed [km/h]	0	0 - 3,24	3.2 - 25.9	0 - 17.6
Wind direction	S-SSW	SW-ENE	SW - NNE	SSE-NNW
Air pressure [hPa]	1009.1 - 1009.2	1012.9 - 1013.2	1021.2 - 1022.9	1012.5-1013.4

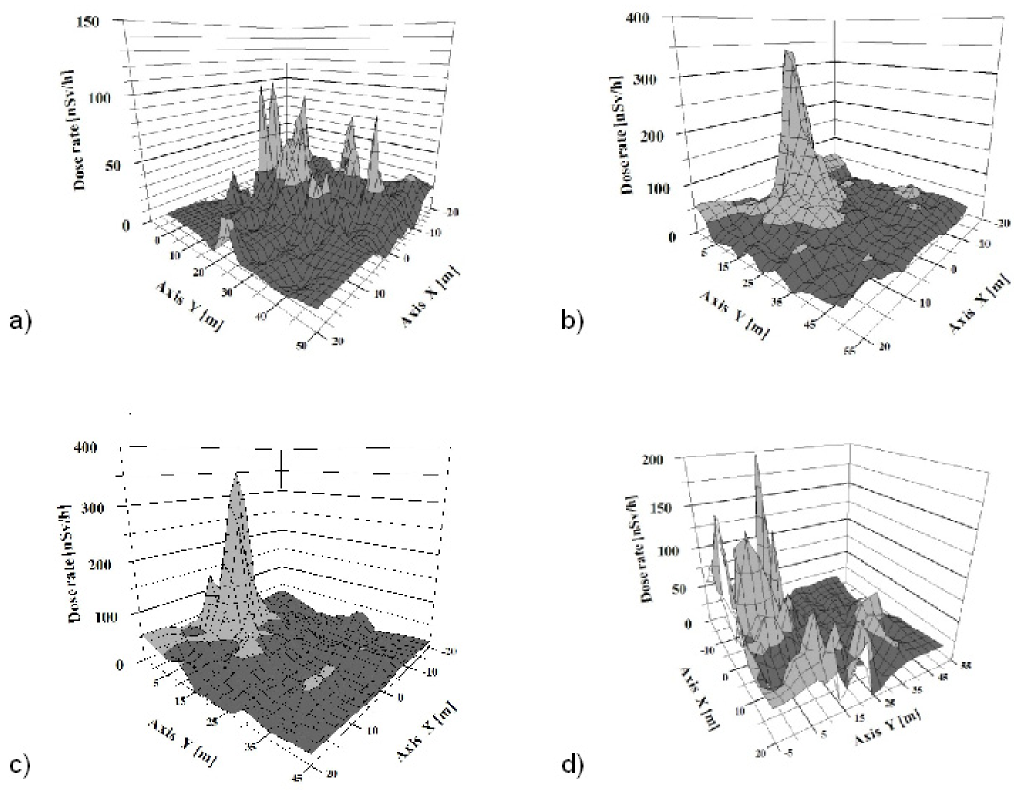

Figure 2a – 2d: Dose rate distribution [nGy/h] - direction of RaS spread – from left to right - axis y; axis x – perpendicularly to the spread axis y; a - test 1, b - test 2, c – test 1P, d – test 2P. [9].

Note:
The black area in Figs. 2a-2d represents territory where the dose rates are lower than 20 nGv/h (after subtraction of the background value in given locality), i.e. an area of the high uncertainties.

Results of the *dose rates* measurements for these 4 tests are presented on Fig. 2a – 3d. The nature of the dose rate distributions in individual tests strongly depended on meteorological conditions, e.g. stable conditions in the 1st test, high temperature and very low wind velocity without gusty evoked lower values of dose rates with less decrease in y – axis direction in the comparison to test 2.

A comparison of the ***surface activities*** distribution on the ground for 4 analysed tests is demonstrated in Fig. 3a – 3d.

Small amount of the RaS dispersed - tenths as far as percentages only was deposited on the test polygon (app. 50 x 40 m^2) in the performed tests - in the 1st test 0.8%, in the 2nd test 2.8% and in the 1P and 2P tests 0.25% and 0.18 % respectively. Differences were given different weather conditions (wind speed and direction, gusts wind, air humidity, temperature - inversion – see tab. 2) influencing the deposition at otherwise practically the same arrangements of the tests. Nevertheless, generally low values of the surface activities correspond of findings (see below), that prevalent amount of the RaS released is transferred outside polygon in the first minutes after explosion.

During 1P and 2P tests practically immediately after explosion, a wind direction gently shift from direction of the RaS propagation (y-axis). Due to this shifts maximum values of surface activities for the given distance on the y axis shifted on the right (on x-axis) in the test 1P (Figs. 3c) and to left in the test 2P (Figs. 3d,).

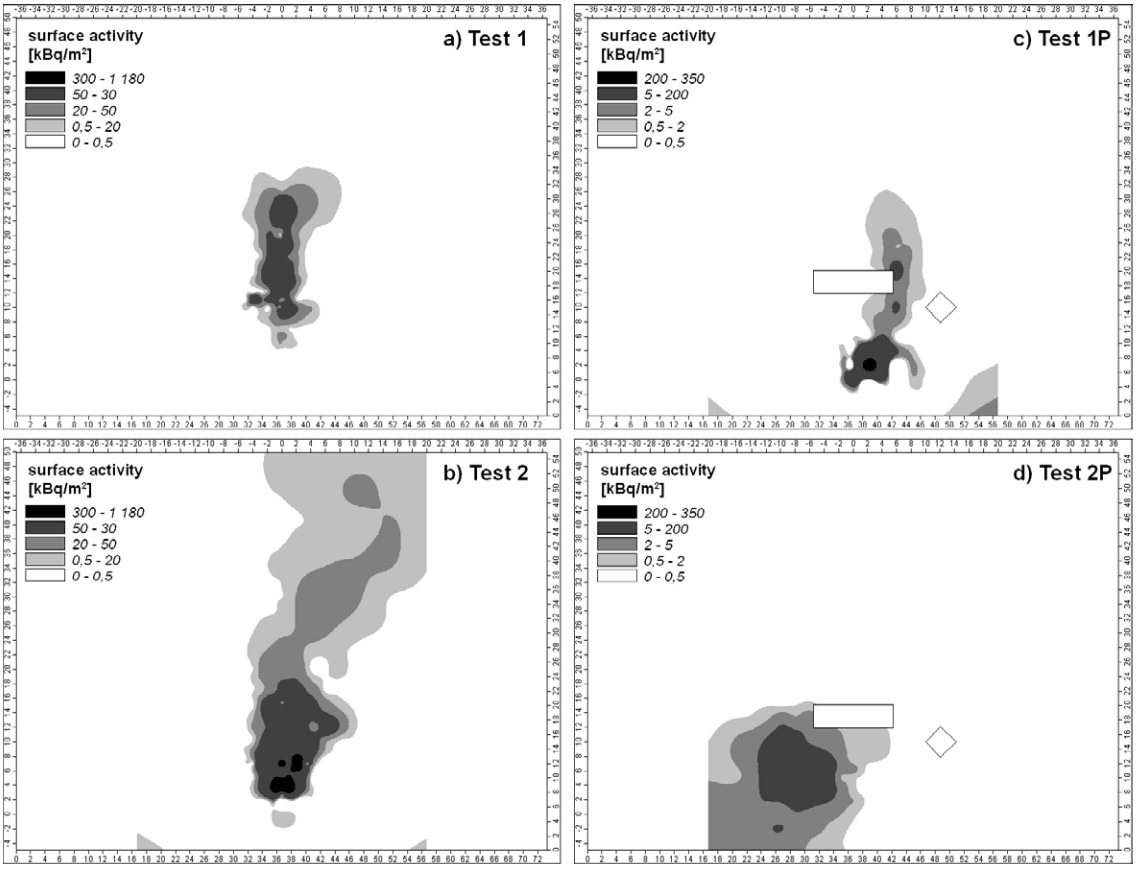

Figure 3a – 3d. Interpolation (Multilevel B-Spline) [15] distributions of surface activities [Bq/m^2] on polygon area - direction of RaS spread (axis y) – bottom-up.

Up to a distance of y < 20 m, the *horizontal surface activity* along the x axis ≤ ±5 m decreased by more than two orders of magnitudes, which indicates that the angle delimited for the RaS dispersion is relatively narrow. At distances of y ≥ 20 m, the value of the maximum surface activity for given y was shifted depending on instantaneous meteorological conditions in direction of the x-axis by more than ± 5 m. More detailed information on surface activities distribution during these tests is given in [8].

Fig. 4a shows time distributions of the surface activity average rate ($Bq/m^2/min$) for the exposure time of a given horizontally situated filter (test 2). Along the axis of the RaS propagation the surface activity in further sampling procedures was lower by two orders of magnitude compared to the first sampling. At higher distances perpendicularly to the propagation axis (x = ± 8 m), the decrease in the activity with time was much slower, but activities in the first sampling were lower by orders of magnitude.

This finding is also demonstrated by measurements of volume activities fig. 4b, as well by measurements of the aerosols mass concentration by laser nephelometers DustTraks [8,10].

The RaS dispersion in *vertical direction* was more uniform than along x – axis (again due to directionality of the RaS release). No marked and reproducible maxima in z – axis direction were manifested; in several cases, at a maximal height (z < 12 m), the activities were comparable with or higher than subsurface activities measured on the ground. Time distribution of vertical measured surface activities was similar as for horizontal measured activities.

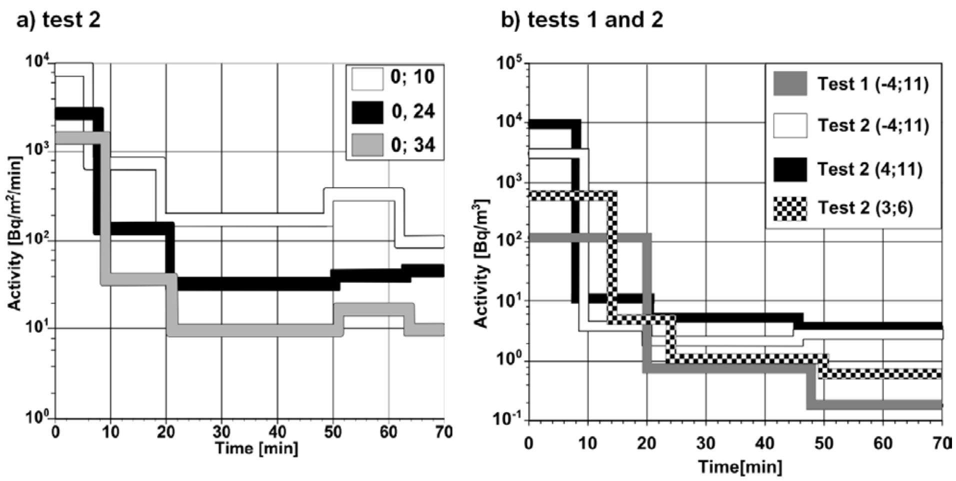

Figure 4. Time distribution of the surface activities rate - 6a (test 2) and the volume activities – 6b - (tests 1,2).

Notice: numbers in the Figs represent position [x,y] m of the collecting filters or position of the aerosol collectors.

Table 3 presents the total activities (the sum of the **volume activities** of all particle sizes), distribution (in %) of ***volume activities*** in dependence of aerosol aerodynamic diameter (AD) for test 1, 2 and 2P. High values of volume activities measured in first minutes after explosion are an explanation of the low total surface activities on collecting filters covered area – the bulk of released RaS leaves the monitored polygon in the first minutes after the explosion, even if meteorological conditions are very stable. The distributions were slightly bimodal with the boundary around 0.4 µm and so the activity median aerodynamic diameters (AMAD) and their geometric standard deviations (GSD) were estimated without taking into account the activities connected with the finest particles (AD < 0.39 µm). AMADs and GSDs are presented in the table 3 too. In the test No 1 collecting substrates from certain stages were combined and counted together and the AMADs and GSDs were not possible to estimate. The cascade impactor in the test 2P in the distances (0, 18) m and (0, 25) m were placed behind the obstacle.

The aerosol by size corresponds to the industrial aerosol (a large proportion of particles with AD > 1.3 µm) with a relatively high GSD value.

Table 3: Distribution of the volume activities in dependence of the aerosol aerodynamic diameter (AD) - tests 1,2, 2P.

Test No	1			2			2P		
Coordinates (x,y) [m]	(0,11)	(0,25)	(0,35)	(0,11)	(0,25)	(0,35)	(0,11)	(0,18)	(0,25)
Sampling time [min]	117	117	129	85	85	84	69	48	47
Total activity [Bq/m^3]	627	102	1,7	811	404	249	19 300	64	62
AD [µm]	Activity [%]								
> 10.2	10	4	18	13	7	11	37	20	21
1.3 - 10.2	47	20	47	63	58	65	37	39	39
0.39 - 1.3	15	6	15	13	9	14	10	9	17
< 0.39	28	69	21	10	26	10	16	32	23
	AMAD and GSD for aerosols with AD > 0.39 µm								
AMAD [µm]	-	-	-	2.5	1.7	2.8	8.4	5.4	4.2
GSD	-	-	-	3.3	4.8	3.8	4.3	3.5	4.2

Notices
The cascade impactor in the test 2P in the distance (0,18) m was placed behind the obstacle.

Tests in enclosed places

Two pilot tests in enclosed places were also carried out in 2008:
- Jul 22, 2008 – test in a single-storey building - “Small house“;
- Sep 30, 2008 – test in a bus - “Bus“.

The **Small house** test was carried out in a single-storey building with two rooms (industrial explosive, RaS - Tc-99m 302 MBq activity in glass sphere of 6 ml -similarly as in the free-area tests). Omnidirectional RaS dispersion took place in this case.

An identical method as in free-air tests used for the detection of surface activity. Filters were located (app. 300 filters were located and measured) in both rooms on floors, walls and ceiling, in a staggered arrangement in a distance of app. 1 m from each other (with an exactly defined position). Apart from filters and devices for aerosol taking, 6 phantoms simulating presence of persons in the building were located in the “Small house” – see Fig. 5a.

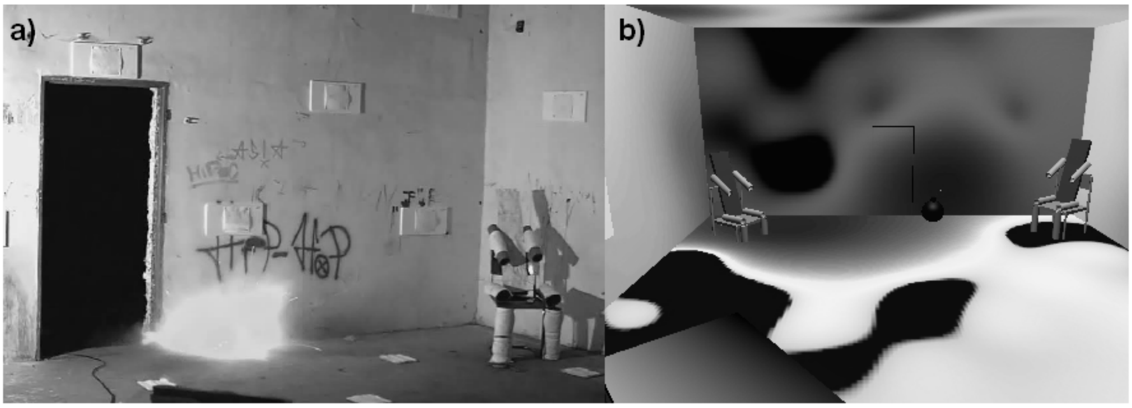

Figure 5. a) An explosion in which Ras was dispersed in “Small house” (location of collecting filters and phantoms can also be seen in the Figure); b) surface activity distributions after RaS dispersion in "Small house“ test.

The volume activities were measured by 4 sampling devices, two of which located in each room. The interpolated surface activity distribution in large room is demonstrated in Fig. 5. Table 4 presents the distribution of the volume activity in dependence of the aerosol aerodynamic diameter (AD).

Table 4. Distribution of the volume activity in dependence of the aerosol aerodynamic diameter (AD) - tests “Small house”.

AD interval [μm]	Volume activity [Bq/m3]	Volume activity [%]
> 8.85	664	50
1.13 - 8.85	391	30
0.34 - 1.13	114	9
< 0.34	149	11
Total	1318	100

Another test in enclosed places was carried out in the “Karosa” type **bus**. An identical explosive as in the “Small house“ test was used - collecting filters were located in an aisle between seats in the first and second row from the driver. 150 MBq of ^{99m}Tc solution was dispersed. The detection technique was the same – nearly 400 filters were placed on the floor, seats, walls and ceiling of the bus and 8 phantoms were put on seats. The cascade impactor was also used to measure the distribution of atmospheric aerosols. The impactor was placed on the final, back seat in the bus's direct axis with the entry towards the place of RaS dispersion. The dose rate and surface activity distribution is demonstrate in fig. 6.

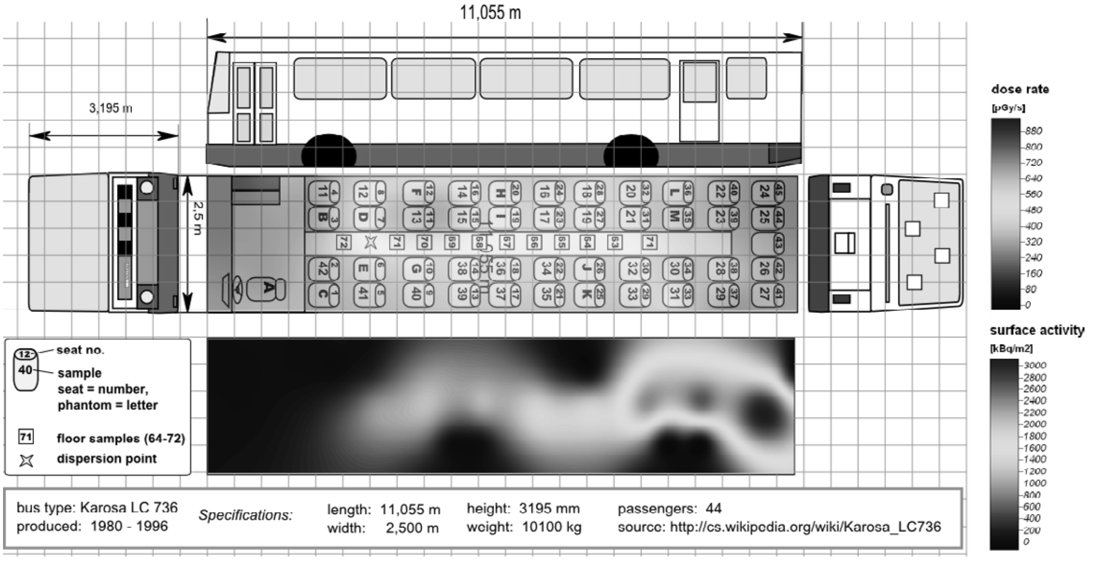

Figure 6a,b. Dose rates (a) and surface activity (b) distributions after RaS dispersion in "Bus“ test (location of collecting filters can also be seen in the Figure).

Regarding small distance between place of RaS dispersion and an entrance of the impactor in tests „small house" and „bus", majority of released RaS was captured on the great particles transferred by blast wave in the short time after explosion.

Conclusions

Presented comparison results demonstrates that the obtained set of data – dose rates, surface and volume activities, mass concentrations - is applicable for evaluation of the RaS propagation on short distances. It is easy to understand that absolute values of particular quantities will always be strongly dependent on the actual meteorological situation at the time of the RaS release under otherwise identical conditions.

The measurements of the surface and volume activities demonstrated the prevalent amounts of the released RaS are transferred across monitored area within very short time. For the vertical distributions (up to 12 m above the ground level) of surface activities the similar results were obtained. These findings confirmed as well measurements using DustTraks [8, 10], which showed, that a spread of aerosols driven by the explosion pressure wave up to a distance of 50 m during less than 1 sec, after that the distribution of surface and volume activities corresponds the propagation by the convection, depends on instantaneous meteorological conditions.

Preliminary evaluation made using different codes [11- 13] demonstrates that the obtained data/results may be used for purposes of model calculations and upgrade and development of existing models (programme IAEA EMRAS II [14].

Acknowledgements

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Genomic-based biodosimetry monitoring analysis method

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Abstract
Large-scale radiologic emergency due to terrorism or large-scale accidents could result in potential radiation exposure of hundreds to thousands of people. The present guidelines for biological evaluation after such an event are still scarce. Therefore, there is a need for biomarkers of retrospective biodosimetry after radiation exposure and assessment human health risk. The frequency of chromosome translocations in individuals exposed to low to medium doses of whole-body irradiation served up till now these goals. Measurement of chromosome translocations in peripheral blood lymphocytes is presently the golden standard to quantify the effects of ionizing radiation and has been used for workers exposed to low or chronic doses. The major problem associated with the assessment of chromosomal aberrations is that it requires a considerable amount of time and labor for aberration scoring and that the sensitivity is rather low.

In this study, we used whole genome microarray expression profiling to identify genes with the potential to predict radiation dose across an exposure range relevant for dose discrimination and medical decision making in a radiologic emergency. Human peripheral blood from 10 healthy donors was irradiated ex vivo using mainly 0.1 Gy X-rays as low dose and 1 Gy as a high dose to be compared to non irradiated control cells, and global gene expression was measured at 8h after exposure identified previously as optimal time for gene expression. Data analysis revealed a clear cut in terms of the pathways modulated with the two doses. At the high dose (1 Gy) we observed mainly the induction of p53 responsive genes (MDM2, DDB2, XPC, EDA2R, SESN1, CCNG1) and pathways associated with cell death processes. In contrast, at the lower dose (0.1 Gy), we observed modulation of another set of pathways mainly associated with mitochondrial oxidative stress and chromatin remodelling.

In conclusion, our analysis method allowed us to discriminate between low and high dose exposures based on the modulated pathways, in contrast to other studies attempting to identify sets of marker genes expressed in a dose-dependent way, which might be biased by inter-individual variation.

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State-of-the-art activity measurement and radionuclide metrology in radiation protection

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Abstract
Numerous issues in radiation protection require appropriate activity measurements. Although today many methods are established to assess radiation exposure to man by physical models and mathematical simulations, there is also a strong requirement in technologically adequate, fair accurate, traceable and reliable activity measurements techniques and radionuclide analytics.

To ensure effectively traceable measurement methods for all medically and technologically used radionuclides up to end-user applications, a network of primary and secondary national metrological activity standards are provided by national metrology institutes under the coordinative umbrella of the Bureau International des Poids et Mesures (BIPM) in Paris. The international progress of activity metrology and measurement is scientifically supported by the International Committee for Radionuclide Metrology (ICRM).

In this paper, an overview of state-of-the-art activity measurement methods for all categories of radiation protection is presented. Special focus is given on current developments in emerging medical application, radioecology, NORM and radon measurements, and uncertainty assessment. Recent improvements and practical implementation on the metrological quantification of radionuclide activity – from fundamental metrology to end-user applications – are covered. The scientific support of ICRM in the field of radionuclide metrology is illustrated. Eventually, international, European and national standards and scientific and technological co-operations and networks in radionuclide metrology and analytics are discussed with regard to upcoming necessities determined by recent ICRP recommendations and IAEA & EU Basic Safety Standard development.

Introduction

To step into radionuclide metrology and activity measurement, one has to consider a common understandable and mostly agreed definition of the quantity activity. A widely agreed common scientific definition can be found in IEC 60050:2003:

> Activity A: quotient, for an amount N of radionuclide in a particular energy state at a given time, of $\langle \mathrm{d}N \rangle$ by dt, where $\langle \mathrm{d}N \rangle$ is the expectation value of the number of spontaneous nuclear transitions from this energy state in the time interval of duration dt.

$$A = -\frac{\langle \mathrm{d}N \rangle}{\mathrm{d}t} = \lambda \cdot N \qquad (1)$$

where λ is the decay constant of the radionuclide.

This definition is amended by the simpler and more operational definition in ISO 921:1997, entry No 23:

> Activity A: number of spontaneous nuclear disintegrations occurring in a given quantity of material during a suitably small interval of time divided by that interval of time.

In any case the unit of the quantity activity is defined in the International System of Units SI (BIPM, 2006) to s^{-1} with the specific name *becquerel* (Bq).

In many applications of radiation protection measurements derived activity quantities like activity concentration or surface contamination with their specific units are established eg. $Bq{\cdot}kg^{-1}$, $Bq{\cdot}m^{-3}$, and $Bq{\cdot}m^{-2}$.

Metrological activity measurement methods

For metrological purpose adequate standardisation measurement methods for representing the unit *becquerel* for specific radionuclides are given in the *Generic grouping of nuclides* (CCRI(II), 2008). The state-of-the-art radiation detectors, measurement geometries, radiation types and counting modes, currently used for metrological radionuclide activity standardisation, are shown in Tab. 1 and Tab. 2. The international key comparisons for primary standard activity measurement is supported by the *International Reference System* SIR of the *Bureau International des Poids et Mesures* BIPM in Paris (SIR, 2010).

The currently achievable measurement uncertainties in activity primary standardisation methods depend on the applied method and the radionuclide of interest. The expanded relative uncertainties (coverage factor $k = 2$ according to JCGM, 2008) of state-of-the-art activity primary standardisation methods range from 0,4 % to 5,0 %.

The international coordination of primary standard activity measurement capabilities supports the calibration services for activity measurement instruments of national metrological institutes and designated laboratories. Radiation protection is only one of various applications of activity measurement (eg. nuclear medicine diagnostics, brachytherapie).

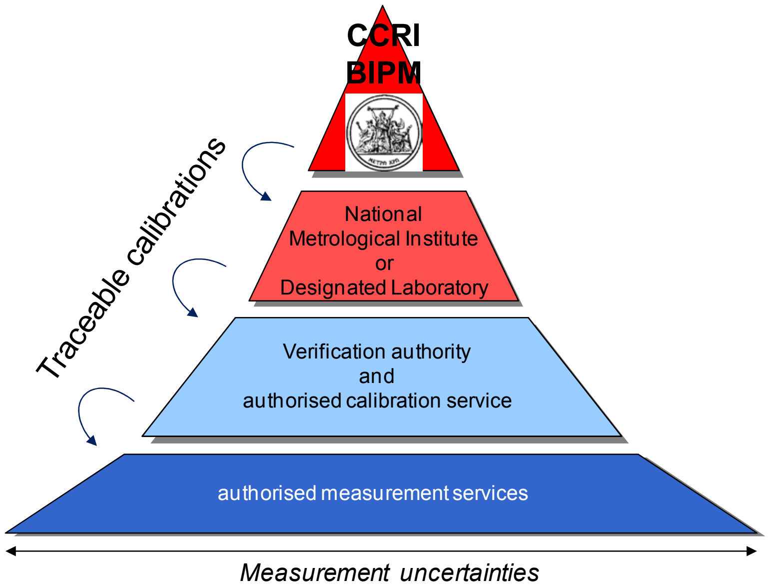

Figure 1. Traceability of measurements from CCRI - Consultative Committee for Ionising Radiation at BIPM - Bureau International des Poids et Mesures to authorised measurement services.

Comprehensive overviews on currently used methods for metrological radionuclide activity standardisation are given in Simpson & Judge (2007) and Pommé (2007).

Table 1. Radiation detectors and measurement geometries used in metrological activity standardisation measurement methods (source: Ratel, 2010).

Geometries	Radiation detectors
4π	Proportional counter
Defined solid angle	Pressurized proportional counter
2π	Liquid scintillation counting
Undefined solid angle	NaI(Tl)
	Ge(HP)
	Ge(Li)
	Si(Li)
	CsI(Tl)
	Ionization chamber
	Grid ionization chamber
	Bolometer
	Calorimeter
	PIPS detector

Table 2. Radiation types and counting modes used in metrological activity standardisation measurement methods (source: Ratel, 2010).

Radiation type	Counting mode
Positron	Efficiency tracing
Beta particle	Internal gas counting
Auger electron	CIEMAT/NIST
Conversion electron	Sum counting
Mixed electrons	Coincidence
Bremsstrahlung	Anti-coincidence
Gamma rays	Coincidence counting with efficiency tracing
X - rays	Anti-coincidence counting with efficiency tracing
Photons (x + γ)	Triple-to-double coincidence ratio counting
Photons + electrons	Selective sampling
Alpha - particle	High efficiency
Mixture of various radiations	Digital coincidence counting

The applicable primary standardisation measurement method depends on the decay scheme of the radionuclide of interest. In many cases high-geometry methods with solid angles of the sensitive detection volume around the radioactive source of almost 4π are chosen. For large-area sources 2π surface emission methods are used. Specific measurement methods used in high-geometry activity standardisation are 4π and 2π proportional counting including internal gas counting for gaseous samples, 4π gamma-ray counting, 4π beta + 4π gamma sum counting, 4π windowless sandwich spectrometer (CsI(Tl)) and liquid scintillation counting. Primary standardisation by LSC uses CIEMAT-NIST or triple-to double coincidence rate method. Some high-geometry methods are also high-efficiency methods. In this case, low and medium activity sources are suitable for satisfactory measurement uncertainties.

For alpha emitters defined solid angle methods taking into account the geometry factor Ω/4π are applicable. This method requires delicate sample preparation to avoid undefined self-absorption inside the source. The background rate for alpha detectors (PIPS) are comparable low around $1 \cdot 10^{-3}$ s^{-1}.

Coincidence counting is another applied method for activity primary standardisation. In this branch sum peak coincidence method, 4π beta-gamma coincidence and photon-photon coincidence counting has been implemented.

Examples for state-of-the art primary standardisation methods commonly used for metrological radionuclide activity determination are given in table 3. In this table the currently achievable expanded relative measurement uncertainties (coverage factor k = 2) are shown. These values are the metrological top level figures for traceable calibration of activity measurement instruments and methods.

Table 3. Examples of top-level primary standardisation methods for radionuclide metrology.

Geometry	Detector 1	Radiation 1	Detector 2	Radiation 2	Mode	Uncertainty range (k=2)
4π	liquid scintillation counter	beta particle or alpha particle or Auger electron or gamma ray or X-ray	---	---	CIEMAT/ NIST or triple-to-double coincidence ratio counting	0,4 to 3,5
4π	pressurised proportional counter	(beta particle or alpha particle) and gamma ray	---	---	high efficiency	1,0 to 4,0
4π	NaJ(Tl)	gamma ray	---	---	high efficiency	0,4 to 4,0
definded solid angle	PIPS detector	alpha particle	---	---	---	1,0 to 4,0
4π	proportional counter or pressurised proportional counter or liquid scintillation counter	beta particle or alpha particle	NaJ(Tl) or Ge(HP)	gamma ray	coincidence or anticoinciden ce	0,4 to 2,0

Activity measurement methods in operational radiation protection

Because radiation protection is mainly a health safety issue, activity measurements has to be carried out on a high quality level. This requirement leads to traceable calibration of activity measurement instruments. This means that any activity measurement done for radiation protection purposes have to been carried out by measurement instruments which are traceably calibrated at primary or secondary standards. In practice this technical claim is ensured by institutional authorisation or accreditation or verification of the used measurement instrument. The achievable measurement uncertainties of operational activity measurement methods are higher than metrological standardisation measurement methods due to uncertainty propagation in the calibration chain.

Different radiation protection branches (Tab. 4) require specific measurement methods.

Table 4. Radiation protection branches using activity measurement methods.

Radiation protection branches
Radiation protection of workers - operational RP
Individual monitoring – internal dosimetry
Environmental monitoring
NORM — Naturally occurring radioactive materials
Radon
Nuclear and radiological emergencies and incidents
Nuclear security and malevolent use of radioactive sources
Illicit trafficking
Radioactive waste management
Decommissioning of nuclear facilities
Education, training and research

In the following sections some general aspects of main relevant activity measurement method used in operational radiation protection are discussed.

Low-level gamma-ray spectrometry for environmental monitoring

Gamma-ray spectrometry for monitoring environmental material like water, air, soil, and food, is one of the most common radiometric method applied in radiation protection. Large volume high-purity germanium semiconductor detectors in special low-level shielding facilities support activity concentration measurement in different samples in the activity range from some mBq to some MBq, or expressed as activity concentration in the range from 10^{-5} to 10^{5} Bq·kg^{-1}.

Generally the Ge(HP) spectrometers are calibrated with standard reference activity sources and/or via Monte Carlo simulation methods. Depending on the efficiency of the detector / volume and geometry of the germanium crystal, sample volume and geometry, measuring time, and the decay scheme of the radionuclide of interest, achievable relative measurement uncertainties (k = 2) for this method are typically in the range of 4 % to 20 %. In most applications the achieved uncertainty fits the purpose because the measurement is done for environmental contamination assessment and not for individual monitoring.

One main problem of this method and restriction in measurement accuracy is the representative collection of samples. If enough measurement capacity is available, typically 15 to 30 samples of the monitored material are necessary to ensure similar uncertainty contributions as for the measurement itself.

Radon gas and radon progeny activity measurement

Dose exposition due to inhalation of radon progenies is of interest in radiation protection both at NORM workplaces and at home. Due to the simpler measurement of radon gas in many cases only radon gas activity measurements are legally regulated and carried out. To assess the lung and effective dose a mean equilibrium factor of typically 0,4 between radon gas and radon progenies indoor is applied. State-of-the-art conventional radon gas and radon progeny detection systems are working in the activity range from some Bq·m^{-3} to some 10^5 Bq·m^{-3}. Calibration of radon gas activity measurement instruments and radon detectors are done in radon chambers traceable to secondary or primary radon gas activity standards.

Again for these methods the representative collection of monitored air respectively the sampling position has large influence on the total uncertainty and individual dose determination. Whereas typical expanded relative uncertainties for measured Rn-222 activity concentrations in indoor air are in the range from 7 % to 20 %, the deviation between different sampling positions in the same room could be up to 30 % (eg. due to relative distance of the instrument / detector from fresh air and radon inflow ventilation flows inside the room). This fact leads to the necessity of careful detector / instrument positioning in the observed room.

Surface contamination activity measurement

This activity measurement method is widely used in all application branches of radionuclides eg. nuclear medicine, research laboratories or at NORM industrial workplaces. Purpose of this method is the safeguarding of radioactive contamination of workplaces to avoid unwanted exposition of staff members. Instruments for this purpose are hand-held and hand-foot-clothing monitors equipped with large-area proportional counters or plastic scintillation detectors.

Depending on the measured radionuclide, the covered activity-per-surface-area range goes from some 10^{-2} Bq·cm^{-2} to some 10^4 Bq·cm^{-2}. Required expanded relative uncertainty vales (k = 2) are in the range from 15 % to 40 % depending on the specific application and the radionuclide of interest. In many countries these types of instruments are calibrated with large-area surface standard activity sources. In Austria the traceability of these instruments are assured via legal verification (‘*Eichung*’) each second year.

Liquid scintillation counting in radiation protection

For many radionuclides – especially such with very low beta particle energy emission like H-3 or C-14 and all type of alpha particle emitting radionuclides – liquid scintillation counting technology is a simply accomplishable and up-to-date activity measurement method. For example in activity concentrations of alpha emitters in water samples could be carried out with this method with satisfactory accuracy, linearity and reproducibility in the range from 10^{-5} Bq·kg^{-1} to 10^5 Bq·kg^{-1}. For some radionuclides like uranium isotopes, radiochemical sample preparation is required before counting.

For simple activity measurement tasks like measuring the Rn-222 activity concentration in water, minimum expanded relative measurement uncertainties (k = 2) of 5 % could be achieved in very careful laboratory circumstances. In many cases the typical achievable range of the expanded relative uncertainty including sample preparation is from 10 % to 40%. Also these figures are useful for most applications in radiation protection like environmental and operational monitoring.

In Tab. 5 the currently most commonly used activity measurement detection systems and instruments are listed together with the available international IEC and ISO and European CENELEC (EN) standards. The Technical Committees for radiation protection measurements instruments and methods are at IEC[1]: SC 45B ‘Radiation Protection Instrumentation’, at ISO[2]: TC 85/SC2 ‘Nuclear Energy / radiation protection’ and at CENELEC[3]: TC 45B ‘Radiation protection instrumentation’. In all these standards the application conditions and technical requirements of the specific instrument types are included. At CEN[4] no specific technical committee on ionising radiation protection measurement is established so far.

Table 5. Activity detection instruments for radiation protection applications.

Detector type	Traceability reference source	International & European standards
Portable *α, β, α/β* counter	Area source	EN 60325, (prEN 62363)
Hand, foot and clothing *β/γ* monitor	Area source	IEC 60325, IEC 61560, ISO 7503-1, 2, 3, EN 61098
Floor monitor	Area source	EN 60325, (prEN 62363)
Laundry monitor	Area source	IEC 61256
Well-type *γ* counter	Point source / volume source	IEC 61562, ISO 11929-1, 2
γ spectrometer	Point source / volume source	IEC 61563, IEC 61275, IEC 62002, ISO 11929-3
α spectrometer	Area source	
Liquid scintillation counter	Liquid source	
Whole body counter	Volume source	IEC 61582, EN 61582
Gate monitor	Area source	IEC 60325, IEC 61560, ISO 7503-1, 2, 3, EN 62022
Transit goods monitor	Area source	IEC 61098, IEC 61137, ISO 6980-1, 2, 3, ISO 8769-1, 2, (FprEN 62244)
β/γ water monitor	Liquid source	IEC 60861, IEC 61311, EN 60861
Noble gas monitor	Radioactive gas source	IEC 60761-3, IEC 61524, EN 60761-1, EN 60761-3
Tritium monitor	Liquid / gas source	IEC 60761-4, 5, 6, EN 60761-1, EN 60761-5
Iodine monitor	Gas source	IEC 60761-4, 5, 6, EN 60761-1, EN 60761-4
Dust monitor	Filter source	IEC 60761-1, 2, EN 60761-2
Radon monitor Passive radon detectors	Radon emanation source, radon chamber	IEC 61263, IEC 61577-1, 2, 3, IEC 61578

[1] www.iec.ch
[2] www.iso.org
[3] www.cenelec.eu
[4] www.cen.eu

The International Committee for Radionuclide Metrology - ICRM

The International Committee for Radionuclide Metrology[5] is an association of radionuclide metrology laboratories whose membership is composed of delegates of these laboratories together with other scientists actively engaged in the study and applications of radioactivity. It explicitly aims at being an international forum for the dissemination of information on techniques, applications and data in the field of radionuclide metrology and measurement.

The Bylaws were adopted at the General Meeting in Gaithersburg, Maryland (USA) in 1997. The ICRM as a formal organization grew from efficient contacts among radionuclide metrologists who participated in the First International Summer School on Radionuclide Metrology in Hercig Novi, Yugoslavia in summer 1972.

ICRM activities are largely the responsibility of its working groups. Each group is guided by a coordinator who acts as a net worker for ideas, joint projects and communications and may organize conferences and workshops. There are now six working groups in the following thematic fields:

- Radionuclide Metrology Techniques
- Life Sciences
- Alpha-Particle Spectrometry
- Gamma-Ray Spectrometry
- Liquid Scintillation Techniques
- Low-Level Measurement Techniques
- Non-Neutron Nuclear Data

The (biennially) recently held 17th International Conference on Radionuclide Metrology and its Applications," had been hold in Bratislava in September 2009[6]. The conference papers will be published shortly in *Applied Radiation and Isotopes*. The next conference will be hold in September / October 2011 in Tsukuba, Japan, organized by the National Metrology Institute of Japan, Advanced Industrial Science and Technology (NMIJ/AIST).

The recent conference of the Low-Level Measurement Techniques ICRM working group - the 5th International Conference on Radionuclide Metrology – Low-Level Radioactivity Measurement Techniques ICRM-LLRMT'08 - was held in Braunschweig, Germany, in September 2008 (Arnold, Jerome, Hult, 2009).

All ICRM conferences and workshops are announced on the ICRM homepage. The participation in ICRM conferences and workshops is open to all scientists and persons who are interested in the topics.

Conclusions and perspectives

The metrological basis of radionuclide activity measurement has been well developed on an international high quality level since the last five decades under the scientifical and technological umbrella of the *Bureau International des Poids et Mesures* BIPM in the frame of the Consultative Committee for Ionising Radiation CCRI(II). More than 60 metrological institutes and laboratories operate high-level primary and secondary

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[5] http://physics.nist.gov/Divisions/Div846/ICRM/
[6] http://www.icrm2009.sk/

standards for metrological radionuclide activity measurement. The International Reference System SIR at the BIPM coordinates key comparisons for all radionuclides of interest in radiation protection. The international infrastructure in radionuclide metrology supports the high-quality activity measurement instrumentation in radiation protection. In most Europen countries the traceability of activity measurement instruments used in operational radiation protection is ensured by traceable calibration or verification. The increasing application of emerging radionuclides in medicine, industry and science generates permanent developments in activity measurement instrumentation. The necessary methodical developments are challenging tasks both for radionuclide metrologists and radiation protection experts. In Europe, the research and development network in metrology is scientifically and financially supported by the European Metrology Research Programme EMRP[7]. Within this co-ordinated research programme joint research projects in all branches of metrology and also in activity measurements are carried out to increase and ensure the necessary high-quality scientific and technological basis of radionuclide activity measurement instrumentation.

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[7] http://www.emrponline.eu/

Location and identification of radioactive materials on sea using airborne gamma-ray spectrometry

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Abstract
Considering the nuclear security, one of the threats is that a nuclear device or radiological explosive device is transported to the harbour over the sea. A ship can carry rather large radiological device and if it is detonated close to densely populated areas or economically or strategically important locations the consequences may be severe. Possible source term can also be a marine nuclear reactor release or underwater detonation of nuclear weapon. In these cases the detection of source term is based on identification of fission and activation products that are spread and diluted to a large volume of sea water. The mapping is based on same methodology that is applied to the detection of radioactive fall-out on the ground. As the water masses are constantly moving, the radioactive materials are also changing the location and this requires constant monitoring activity for long period of time. As part of security measures, these scenarios can be exercised by using airborne gamma ray measurement devices on the sea. Measurement over sea is actually very sensitive method as the background from natural radiation is close to zero and the detection capability is significantly better over the sea than over the land. The experience gained through these exercises show that also high-speed jet plane can be used in the identification tasks as we need only few seconds measurement time over the source. The detection system and search logic are illustrated in the presentation.

Introduction
Typical problem in the mobile on-site gamma-ray measurement is to be able to track the radioactive source location with high efficiency and still be able to characterize the source by identifying and quantifying the radioactive substances and their activity. In fact, location, detection, identification and quantification are competing with each other; good location accuracy requires high measurement resolution (i.e. short measurement time for individual gamma-ray spectra) and this reduces the possibility to detect, identify and quantify the radioactive source. Multiple level measurement

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strategy can be used to ensure the detection and location power together with good quantification and identification capability.

This type of measurement capability is needed in case a large radiation source or it's activity is released in the environment and there arises need to locate and assess the risks related on source strength and type. This kind of situation is possible for example due to:

- Release of radiation due to use or testing of nuclear weapon
- Accidental release of radiation from nuclear reactor, medical isotope production or other kind of facility handling radioactive materials.
- Release of radiation due to malevolent use of radiation
- Radiation due lost or forgotten radioactive source

This paper is presenting the results of an exercise that was performed by Finnish Defence Forces by using a mobile radiation source (Cs-137, 3 GBq) in a ship on the Baltic Sea. This exercise was set up to study the detection of mobile source and possible radioactive contamination in the water due to release of radioactive material. The measurements were performed using a jet-plane with both medium volume NaI detector (6x4") and HPGe detector (70% relative efficiency). Typical cruising speed was 370-400 km/h and altitude above the water was 100 meters. Each NaI spectrum was collected with 1 second integration time and HPGe spectra with 2 seconds integration time. This method was developed to ensure quick response measurement in case of nuclear or radiation emergency [1].

Results and discussion

In real time measurements, it is important to show the results immediately for the analyst performing the measurement. This requires that a graphical display with capability to visualize the radiation levels. A spectrogram display as shown in the figure 1 has to be able to show the energy and location of the source so that the operational staff can react on finding. This is the spectrogram of NaI detector. This feature has been built in the gamma spectrum analysis software UniSampo [2].

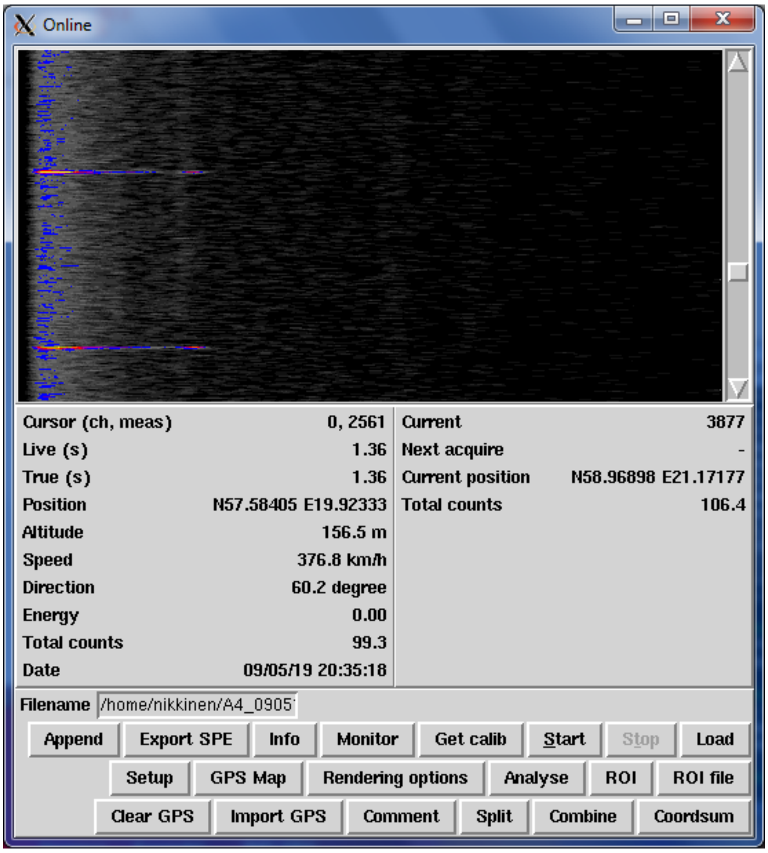

Fig. 1. Spectrogram display during the exercise (each horizontal pixel line represent one single spectrum, more intensive colour the pulses), high activity source is visible only in a few spectra. This view provides actual real-time detection and identification capability already during the flight.

Accurate post-processing is required to extract further information on source location and characteristics. It is possible also to lower the detection threshold using the post-processing. This is possible by combining the measurement points together and comparing the smoothed average with surrounding measurement points. In the measurements computer software GMLINT [3] has been used. The pulse rate of the background is used to calculate the critical limit in accordance with Currie [4]. As a result, it is possible to make a geographically referenced detection capability map as seen in figure 2 [5]. Possible source locations are highlighted in the processing window.

Number of corrections can be applied to the results:

– Distance between source and the detector (three dimensionally)
– Attenuation of the gamma rays in the air (gamma energy dependent)
– Attenuation of gamma rays in the body of the aircraft including angular response

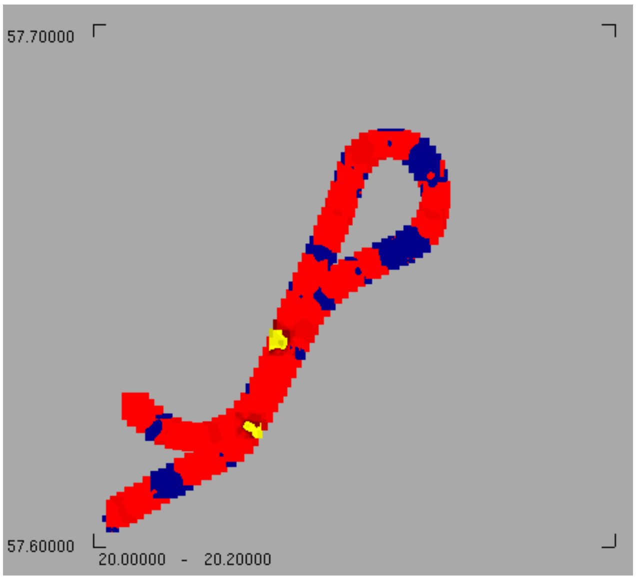

Fig. 2. Display of GMLINT software during the post-processing of data. Source location, uncertainty and critical limit can be displayed during the processing. This provides the end-user possibility to assess source locations and strengths. The yellow dot is showing the maximum source strength. As the ship was moving during the exercise the source was detected in two different locations.

More accurate information is available if HPGe detector information is combined with the analysis. In these kinds of measurements the background pulse-rate is relatively low and due to better energy resolution the pulses are concentrated on significantly more narrow energy window. As a result, the identification of the source is easier. Figure 3 shows one flight line and behaviour of pulse-rate for Cs-137 when by-passing the source during the exercise.

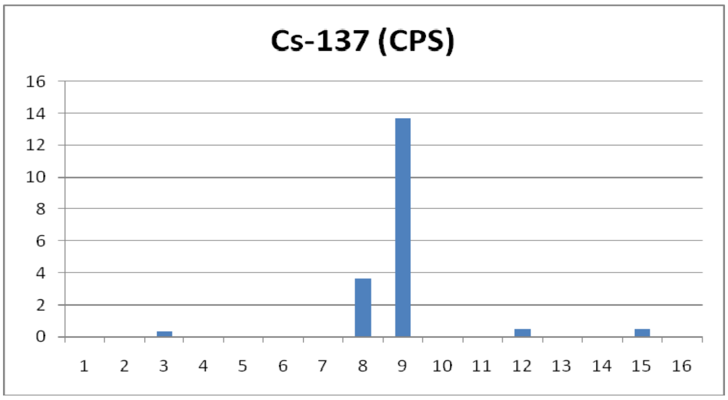

Fig. 3. Histogram display of HPGe detections, the maximum value is clearly detectable. Further analysis can be performed by calculating the weighted average to estimate the exact source location if when the gamma rays emitted are detected in more than one spectrum.

One additional question was where in the boat the source was located. As the size of the boat was rather large (218 meters long ship), it would be desirable to locate the source accurately during the over-flight. As the ship was moving, it was decided to bypass the ship by flying along the axis of the ship.

It is proposed to include also minimum and maximum source strength estimates to the automatic procedure. Maximum is the source strength if the measurement target is optimally located in the middle of integration time and the aircraft is flying exactly on the top of the source as shown in figure 4. Minimum would be if the source location maximum is in the end or beginning of integration time and the distance of the source is in the middle of measurement flight lines.

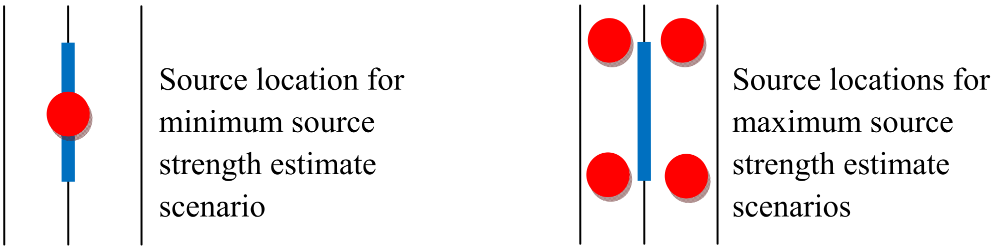

Fig. 4. Minimum and maximum estimates of the source strength. On the left side the source is right on the flight line (narrow line) and in the middle of integration line (thick line), this provides the minimum value for source strength. On the left side the possible source locations are in the end or begin of integration time and between the flight lines. This provides the maximum estimate for the source strength. All the source strength approximations should be between these two estimates.

Conclusions

As a conclusion the tools that are used for the radiation source detection and source location were working very well during the measurements. The flight team was able to locate the radioactive source carrying ship, to characterize the source type (open Cs-137 source) estimate the source activity. In this kind of measurement the uncertainties on source strength is large but the measurement team was able to give approximate source strength information in relatively good accuracy. It is also possible to calculate minimum and maximum source strength relative to source distance to the flight line.

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Quantification of NaI(Tl) whole body counter spectra using the EGSNrc Monte Carlo System

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Abstract

A generic EGSNRc-Monte Carlo model was developed for a whole body counter (WBC) with four NaI(Tl) detectors in a stretcher geometry - which is the setup both at Karlsruhe Institute of Technology (KIT) and at University Hospital of Cologne (UKK) – and the bottle phantoms previously used for their calibration. The main purpose of this work was to determine whether numerical calibrations can replace the ones based on measured spectra of known activity concentrations. Methods: Three main methods were developed and applied for the evaluation of measured spectra:

a) Simulation of nuclide decay and evaluation of total gross counts in a nuclide-specific region-of-interest (ROI), including the contribution of scattered photons

b) Simulation of nuclide decay and using these calibration spectra for a "full spectrum fit" to measured ones

c) Simulations of artificial "single energy" decays and deriving efficiency factors from a set of such simulations.

All methods were able to successfully replace the measured calibration spectra as proven at both institutions in a number of intercomparison exercises. A full set of "single energy" simulations offers a quick and flexible way to calculate efficiency factors for any nuclide without the need for further simulations or measurements. Moreover, Monte Carlo simulations offer the possibility to study the response of the counting system without the need of handling radioactive materials or relying on the properties of a real nuclide.

Introduction

Gamma spectrometry in whole body counters is used to detect and quantify radionuclides in man. This can be done as routine measure of incorporation monitoring or as means to assess doses after incidents. Further uses of whole body counting are measurements of the retention of pharmaceuticals labelled with radionuclides for medical diagnosis (ICRU 2003). All nuclides emitting gamma rays of sufficiently high energies (>100keV) with a significant emission probability (>3%) can be measured by in-vivo counting. Examples are fission or activation products such as ^{133}Ba, ^{137}Cs and

^{60}Co which are used in many nuclear applications or ^{99m}Tc, ^{201}Tl and ^{131}I from the field of nuclear medicine as well as naturally occurring ^{40}K.

Whole Body counters use an array of detectors placed around the person to be measured inside a shielding chamber. The person to be measured can either stand in front of the detectors, sit on a chair or lay on a bed. During the measurement the detectors may stay fixed or move around/along the patient (scanning mode). Detectors used for in-vivo counting are either scintillators such as NaI(Tl) or CsI(Tl) or semiconductor detectors like high purity germanium crystals (ICRU 2003). The activities of the identified nuclides can be calculated from the count rates in defined regions of the measured spectrum (ROI) using efficiency factors. The latter are classically determined from measurements of known activities in physical phantoms representing the human body. The phantoms used need to represent the properties of human body regarding scattering and absorption of radiation (ICRU 2003). Typically bottle phantoms filled with radionuclides diluted in water (e.g. BOMAB) or brick phantoms (e.g. IGOR) are used for calibration of whole body counters. Alternatively Monte-Carlo Methods can be used to simulate radiation transport during the measurement process. Such simulations can then be used to derive the efficiency factors. A Monte Carlo Code using the EGSNrc system (Kawrakow and Rogers 2000) was developed and applied to both installations. Several methods for the evaluation of simulated and measured calibration spectra have been tested and applied to both set ups. The results of this study will be presented in this paper.

Material and methods

The whole-body counters at Cologne-UKK and Karlsruhe-KIT

Both institutions, Karlsruhe Institute of Technology (KIT) and University Hospital of Cologne (UKK), operate whole body counters with a similar set up, referred to as "stretcher geometry". The person lays on a bed which is surrounded by four fixed NaI(Tl)-scintillation crystals facing the person. Figure 1 shows the KIT whole body counter during the measurement of a bottle phantom. The nuclides are identified and their activity is determined from the sum of the four spectra of the measurement. The position of the detectors is optimized such that the count rates in the sum of the four spectra are as independent from the distribution of the source inside the person measured as possible. Properties of the two installations are summarized in table 1.

Fig. 1. The KIT whole body counter. A bottle phantom representing a 70kg person is placed on the bed. Only the upper two detectors can be seen in this figure.

Table 1. Properties of the two whole body counters used in this study.

	KIT	UKK
Chamber shielding	15/25cm Steel + 1.5mm Lead	16cm Steel + 3mm Lead
Detectors Postions Orientation	20.3 x 10.2 cm (8 x 4''), Fa. Bicron 2 above, 2 below the bed tilted around person's body axis	12.7 x 10.2 cm (5x4''), Fa. Scionix 2 above, 2 below the bed Parallell to bed
Channels in spectrum	256	1024
Energy Range	0 – 2500keV	0- 2000keV
Routine Counting Time	300 s	450 s

The Bottle Phantoms

The bottle phantom consists of a set of 2-1 and 1-l water-filled Cautex bottles in which a known amount of the radionuclide is diluted. A number of human shape geometries ranging from 10 kg/70 cm to 100 kg/190 cm can be arranged with different numbers of bricks/bottles. The concentration of the radionuclide in each bottle/brick was the same, thus a homogenous distribution of the nuclide within the body is assumed. In man, this is found only for some nuclides, e.g. ^{40}K or isotopes of cesium.

The EGSNrc usercode "Flaschenphantom"

A usercode for the EGSNrc Monte Carlo system has been developed (Breustedt and Eschner 2010). The EGNSrc system consists of several FORTRAN subroutines (called egscode) that provide the mechanisms for the radiation transport calculation. The geometry of the problem to be simulated and the tallying (extraction of information) has to be programmed by the user and is then coupled with the egscode by a defined set of subroutines (Kawrakow and Rogers 2000). The usercode "Flaschenphantom" provides the general geometry of the whole body counter chamber, the bed and the four detectors. A set of "bottles" (simulated as columns of water, without the bottles themselves) can be placed at the stretcher. All information needed by the usercode can be specified via a plain-text input file. Thus the code is flexible and can easily be adapted to other whole-body counters of the stretcher type. A set of input files describing the series of bottle phantoms (10kg -100kg) in the UKK and KIT set ups (including information on detector resolution and spectrum output) has been developed. Via the inputfile the user has to enter the geometry to be simulated, the nuclide which is assumed to be homogenously distributed in the bottles and the number of decays to be simulated. For some nuclides all information required (photon energies and emission probability, normalized beta spectrum if available) has been collected in text files, which can be addressed by the user. In order to study the response of the whole body counters as a function of photon energy, some artificial "one line" nuclides with a hundred percent emission probability were defined. After finishing the simulations the code will provide 5 output files (plain text format) which contain the spectra of the four detectors and the sum of the four spectra. Other information about the simulation is provided in a separate plain text file. All spectra are normalized to counts per decay simulated. This unit is equivalent to counts per second per Becquerel to which measured spectra can be normalized. The resolution of the detectors is taken into account by a convolution of the simulated energy spectra with a Gaussian kernel of width according to photon energy. The parameters required for this Gaussian broadening can be determined from an observed FWHM vs. energy relation. Thus a direct comparison of measured and simulated spectra is possible, if the widths of the channels in the simulated spectra are chosen accordingly.

Methods for Evaluation of spectra

Three methods to assess the activities from the spectra have been applied ins this study. For each of the nuclides observed in the spectrum a dedicated region of interest ROI (usually centred around the full energy peak of the main emission line) was defined. Generally the count rate in the spectrum (or the parts thereof, ROI) is dependent on the activity of the nuclide measured. The proportionality factor eff(ROI $\leftarrow$ nuclide) is called efficiency.

$$Cps\ in\ ROI = eff\,(ROI \leftarrow nuclide) \cdot Acitivity$$

The classical approach in gamma spectrometry takes only the full energy counts into account by using the net peak area which is determined by subtracting a scatter background below the peak. In in-vivo counting applications the spectra measured have very low count rates. This imposes problems on the determination of the scatter background below the peak. We tried to avoid these by using all counts in a given ROI for the determination of the activity. The scatter contribution to this gross count rate is taken into account by calculating efficiency factors of the form eff($ROI_i \leftarrow nuclide_j$) which then describe the probability of observing a count in the ROI for nuclide i after one decay of nuclide j. For each pair of nuclide and ROI this factor needs to be determined from calibration spectra which contain only counts of this nuclide. This requires a calibration measurement or simulation for each nuclide to be assessed. For whole body counting only a few nuclides are expected thus the overall effort to calibrate the counter still remains feasible. If more than one nuclide is present the efficiency becomes a matrix and the vector of activities A_j can be calculated from the observed vector of count rates C_i in the ROIs by inverting this matrix.

$$\begin{pmatrix} Bq\,nuclide 1 \\ \vdots \\ Bq\,nuclide\,n \end{pmatrix} = \begin{pmatrix} eff\,(ROI\,1 \leftarrow nuclide\,1) & \cdots & eff\,(ROI\,1 \leftarrow nuclide\,n) \\ \vdots & \ddots & \vdots \\ eff\,(ROI\,n \leftarrow nuclide\,1) & \cdots & eff\,(ROI\,n \leftarrow nuclide\,n) \end{pmatrix}^{-1} \cdot \begin{pmatrix} cps\,in ROI\,1 \\ \vdots \\ cps\,in ROI\,n \end{pmatrix}$$

A shortfall of this matrix approach is that the assessed activities become dependent on each other due to the system of linear equations used to determine the activities. Nevertheless we used two methods for the Monte Carlo determination the elements of the efficiency matrix. One method (later referred to as “direct simulation”) is a complete simulation of the nuclide decay. The elements eff($ROI_i \leftarrow nuclide_j$) can then be calculated straightforward by integrating the simulated spectrum in the given ROI i. The second method (later referred to as “reference simulation” is based on simulations of artificial nuclides which emit only one energy with a 100 % probability per decay. Such nuclides were defined for 20, 40, . . . 1980 keV, 2000 keV. An example for the derivation of efficiency factors eff($ROI_i \leftarrow nuclide_j$) for real nuclides from the set of "reference simulations" is given in the results section.

A second approach for the determination of activities from the measured spectra (later referred to as "fitting method") is also based on calibration spectra obtained with one nuclide at a time. Here the whole range of the spectrum, normalised to cps/Bq, is used to determine the activity. A weighted sum of the calibration spectra for the nuclides identified is built. A least squares fit to the measured spectrum (normalized to cps) is used to determine the weighting factors A_i which then are the activities. A prerequisite for the fitting of reference spectra to the measured spectrum is that all use the same energy calibration. This is ensured by the quality assurance procedures used in both labs, e.g. in the daily QA at KIT a point source is measured and the amplification factors are adjusted to fit the measured peaks in defined channels. Deviations between the different geometries are apparent in the scatter part of the spectrum at low energies. Thus it is useful to introduce a lower energy limit for the optimisation. We usually use only counts above 100 keV in the fitting procedure. In this method the assessed activities are also dependent on each other due to the common fitting.

Results

Several simulations in different geometries and for different nuclides were run. The photons were followed down to 1 keV, electrons down to 10 keV in these simulations. Usually 10 million decays were simulated, providing reasonably good counting statistics in the spectra. Depending on the nuclides' decay schemes the simulation runs took only a few hours on a modern computer. Simulated and measured spectra are in a good agreement. The direct comparison of the simulation and measurement of a 70 kg bottle phantom filled with ^{131}I in the UKK counter is shown in figure 2.

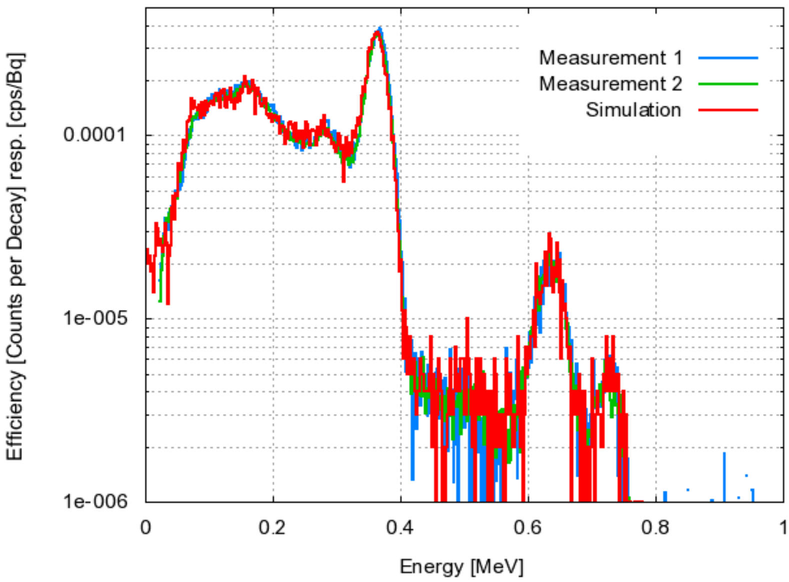

Fig. 2. Comparison of measured (blue, green) and simulated (red) spectrum of a 70 kg phantom filled with ^{131}I in the UKK whole body counter.

The only nuclide, where we observe deviations of measured and simulated spectra is ^{40}K. Here the spectra are in a good agreement above 250keV, below this energy the simulations underpredict the count rate in the spectra. The same effect could also be observed with another simulation code, MCNP5 (Brown 2002). Investigations on the reasons for these deviations are still in progress. Interestingly simulations with electron emissions only showed that the shape of the missing counts is that of a "Bremsstrahlung" spectrum for the electrons emitted by ^{40}K. If the electron emission in the simulations is artificially enhanced by a factor[1] of ~4.4, the simulated 40K spectra are in good agreement with the measured ones (Breustedt and Eschner 2010).

Efficiency factors eff(ROIi ← nuclidej) were calculated in the different geometries (10 kg – 100 kg) for several combinations of ROI and nuclides. The efficiency factors determined by the two methods "direct simulation" and "reference simulation" are in a good agreement with each other and with the ones derived from a series of measurements. For calculating the efficiency(ROI_k ← nuclide) from the set of "reference simulations" all of the simulated spectra are integrated in the same ROI and the results are assigned to the simulated energy. These values eff($ROI_k \leftarrow E_\gamma$) give the probability to observe one count in the ROI_k after the emission of one photon of E_γ in the phantom. An example of the resulting curve of 'efficiency in $ROI_{I\text{-}131}$ vs. emitted energy' is shown in Figure 3. $ROI_{I\text{-}131}$ is chosen from 316 – 411 keV centered around the ^{131}I line at 364 keV. The curves have a characteristic shape. For emitted energies below the ROI boundary, the efficiency values are zero. For emitted energies within the ROI high efficiency values are observed due to the photopeak being in the ROI. For higher energies only scatter events, which give lower efficiencies, are observed in the ROI.

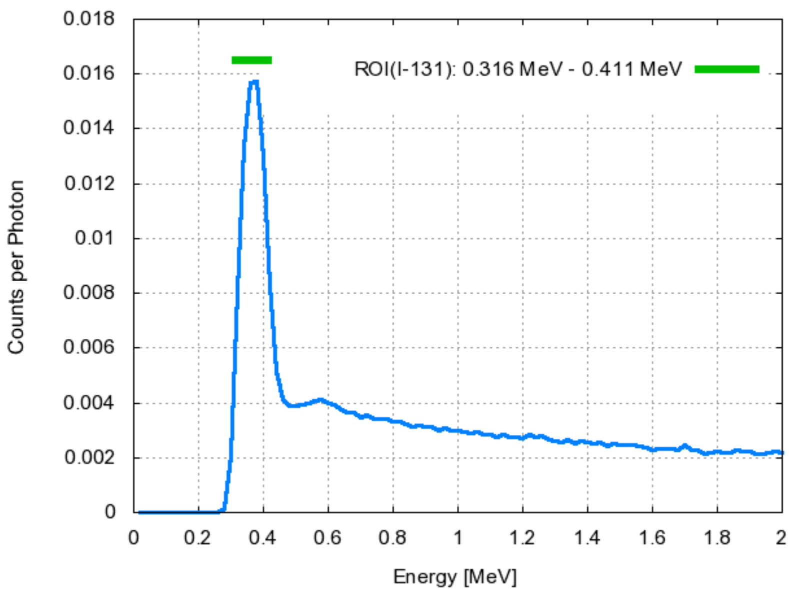

Fig. 3. Efficiency vs. energy curves (KIT counter) for ROI $_{I\text{-}131}$ centred around the peak at 364 keV. Here the ROI remains fixed and the efficiency for registering counts in this region of the spectrum after emission of a photon of a given energy is shown.

[1] Interestingly this factor is the same for both Monte Carlo systems EGSNrc and MCNP5.

Table 2. Calculation of efficiency from "Reference Simulations". Here the calculation of eff(ROI_{I-131} ← ^{58}Co), which gives the probability to observe one Count in ROI_{I-131} (from 316 – 411 keV) after the decay of one ^{58}Co nucleus in the phantom.

Energy [keV]	Emission Probability [%]	eff(ROI_{I-131} ← E_γ) [Counts per Photon]	contribution to eff(ROI_{I-131} ← ^{58}Co)
511	29,80	0.003876	0.001155
811	00,69	0.003329	0.003296
863	99,00	0.003131	0.000022
1674	00,52	0.002265	0.000012
		eff(ROI_{I-131} ← ^{58}Co) = 0.004485 **[cps/Bq]**	

Using the decay data of a nuclide the efficiency for the contribution of this nuclide to the count rate in a chosen ROI can then be calculated from such 'efficiency in ROI vs. emitted energy' curves. For each photon emitted by the nuclide its contribution to the efficiency is taken from the curve and summed up, weighted by the emission probabilities of the photons. The calculation of the nuclide specific efficiency factors from the curve shown in Figure 3 is shown in Table 2. Here eff(ROII-131 ← 58Co), which gives the probability to observe one count in ROII-131 (which again is centred around 364 keV, the main emission line of 131I) after one decay of an 58Co nucleus is calculated. Once a set of "reference simulations" is available for a given geometry, this method can be applied to derive efficiency factors for every nuclide. A contribution of Bremsstrahlung generated by emitted electrons is not taken into account. This may for some nuclides (e.g. 40K) lead to deviations. Nevertheless the efficiency factors eff(m, ROI ← nuclide) calculated with this method are consistent with the ones derived from direct simulations of the nuclide decay or from real measurements as can be seen in figure 3 which shows as example an plot of the "efficiency vs. mass" curve for 18F in the ROI which is centred around its main emission line at 511 keV. The mass dependency was parameterized by fitting a function of the type

$$eff\,(m, ROI \leftarrow nuclide) = a + b \cdot m^{-1} + c \cdot m$$

with free parameters a, b and c.

In the fitting procedure the main problem in the application of the simulated spectra was the underestimation of Bremsstrahlung counts in the spectrum of 40K. Thus the lower threshold needed to be raised as high as 250 keV in order to achieve good results for simulated spectra. If the 40K spectrum was replaced by a measured one, the lower threshold could by set down to 80 keV without any problems. Nevertheless the activities calculated by the application of simulated and measured spectra in the fitting method are in good agreement. The results were comparable to the ones calculated via ROI techniques.

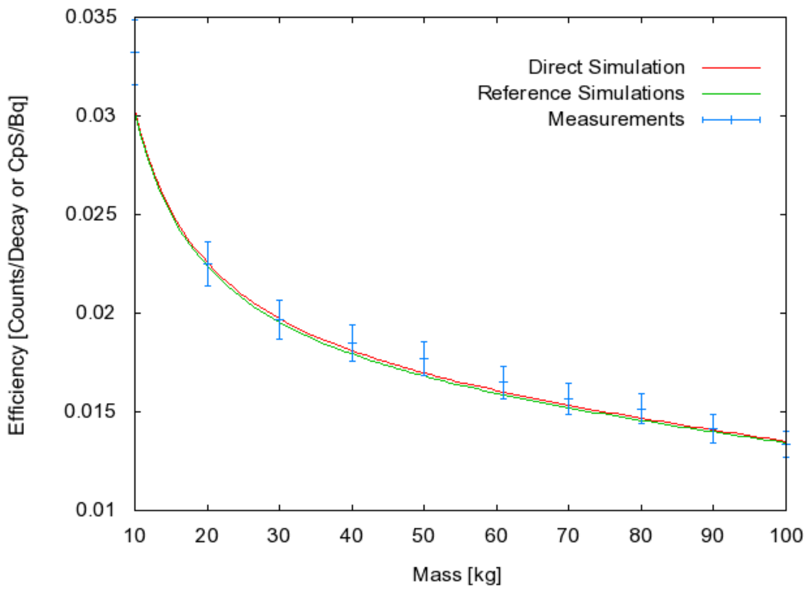

Fig. 4. Efficiency vs. mass curves for measurement of ^{18}F in a ROI centered around 511 keV. The lines show fits to efficiencies calculated from simulations (red: direct simulations; green: sets of reference simulations). Points show efficiencies derived from measured spectra.

Application example: Spectra from Intercomparison Exercises

Intercomparison exercises are one tool for quality assurance of in vivo measurement systems. The three methods described above were applied for the quantification of spectra during several intercomparison exercises. In these exercises, which are offered each year by the German federal office for radiation protection (Bundesamt für Strahlenschutz, BfS) (König et al. 2004), a brick phantom filled with activity (unknown to the participants) is measured by each participating institution. UKK and KIT participated regularly in the exercises of the last 10 years, which used the nuclides ^{133}Ba, ^{137}Cs, ^{60}Co and ^{40}K. All of the methods were able to quantify the reference activities within ±15%. As one example, the results for the KIT counter in the recent 2009 exercise are shown in table 3.

Table 3. Results of the 2009 intercomparison exercise at KIT. The activities calculated with the three approaches and the reference values provided by BfS are shown. The deviation from the reference values is given in brackets.

Nuclide	Reference Value [Bq]	Assessed Activity [Bq]		
		Direct Simulations of nuclide decay		Reference Simulations
		Fitting Method	ROI Method	ROI Method
K-40	2807	2605 (-7%)	2596 (-8%)	2631 (-6%)
Co-60	1176	1098 (-7%)	1045 (-11%)	1056 (-10%)
Cs-137	4332	3910 (-10%)	3895 (-10%)	3881 (-10%)
Ba-133	2564	2711 (+6%)	2403 (-6%)	2176 (-15%)

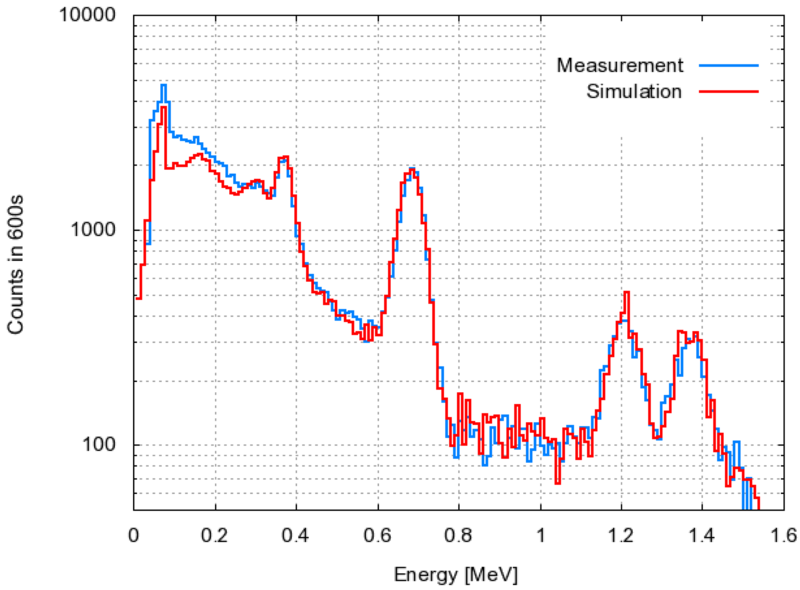

Fig. 5. Application of "fitting method" to spectrum of intercomparison exercise 2003 at KIT. The deviations below 250keV are due to Bremsstrahlung which is underestimated in the simulations. For the nuclide ^{40}K approximately half of the counts below 250 keV are due to Bremsstahlung.

Discussion and conclusions

None of the three methods we applied uses any algorithm to determine net peak areas. Only the gross counts in a Region of Interest need to be determined by an easy sum of counts in the according channels. This is an advantage especially in low count rate spectra as encountered in whole body applications, where net peak area algorithms may be hard to apply. The activities calculated are dependent on the other ones derived from the same measurement. This clearly is an disadvantage of the three methods and may lead to misinterpretations of the spectra. However, a large disagreement with activities derived from net peak areas algorithms has not been observed by us.

All Monte Carlo based methods were able to successfully replace the measured calibration spectra as proven at both institutions in a number of intercomparison exercises. The Monte Carlo Simulations offer the possibility to calibrate a whole body counter without the need of measuring activities. Thus single-nuclide spectra for the fitting method or the determination for efficiencies eff(ROI ← nuclide) become available for any nuclide needed. One problem observed by us is the improper simulation of Bremsstrahlung resulting from emitted electrons, which leads to an underestimation of low energy (< 250 keV) counts in the simulated spectra. This is observed for the nuclide ^{40}K which is always present in whole body counting. Studies to resolve this problem are currently done at KIT.

A full set of "single energy" simulations offers a quick and flexible way to calculate efficiency factors for any nuclide without the need for further simulations or measurements. Moreover, Monte Carlo simulations offer the possibility to study and evaluate the response of the counting system without the need of handling radioactive materials or relying on the properties of a real nuclide. A full calibration of the in-vivo counting system for using one of the three methods presented here may require a large amount of calibration spectra. But using Monte Carlo Simulations these spectra can be calculated without large effort in a rather short time.

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New opportunities of radiation portal monitors with plastic detectors

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I. Introduction

The prevention of unauthorized transportation of nuclear and radioactive materials through borders of the states demands the creation of a high quality system of the radiation control. As it is shown by the international experience, radiation portal monitors intended for the first stage of the control, i.e. the detection of a radioactive source in an object which crosses a control zone, create the basis of such a system. In highly sensitive gamma-neutron radiation monitors, the detection of gamma sources is provided by plastic detectors. Despite of a rather low cost of such detectors, the creation of systems of the border (customs) radiation control is rather expensive as the total number of radiation monitors is very high. It is enough to mention the Sheremetjevo-2 airport in Moscow and the Rotterdam sea port [1, 2].

A tendency has recently appeared to use the monitors based on essentially new detectors, providing not only detection of sources, but also their identification (the so-called spectroscopic radiation portal monitors). However, these are so expensive that can hardly replace traditional monitors with plastic detectors. So, it is very important to search for ways to increase their efficiency.

In the present paper the authors consider a possibility to increase the efficiency of traditional monitors with the use of additional lead collimators installed on both sides of each plastic detector.

The purpose of this research was not the elaboration of certain design recommendations of radiation portal monitors, but the analysis of the general relationships of the collimator influence on the basic monitor characteristics.

II. Results and discussion

II.1. Decrease of the measured natural gamma background

As known, radiation portal monitors with plastic detectors for gamma source detection use the traditional algorithm connecting a threshold of detection (minimum detectable signal **S**) with the background value **B**

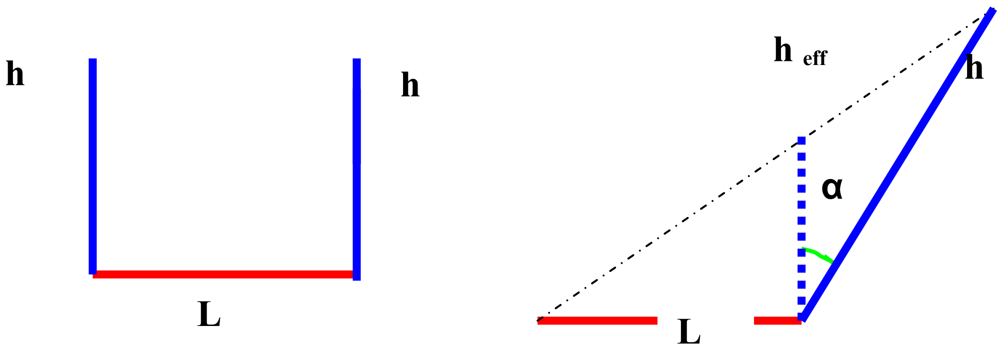
$$S = B + n \cdot \mathrm{sqrt}(B), \qquad (1)$$

where: **B** is value of the measured background;

sqrt (B) is the root-mean-square deviation of the background (so-called "sigma"),
n is the set number of "sigma" which defines both the false alarm rate, and the minimum detectable signal.

From the equation (1) it is clear, that a decrease of the background value measured by the given monitor should lead to a decrease of the minimum detectable signal other things being equal.

One of quite obvious technical decisions leading to a decrease of the measured gamma background is the use of a collimator installed on each plastic detector of the portal monitor. We have examined two versions of the lead collimator installation on the detector: collimator is perpendicular to the detector surface, and collimator is declined from the vertical by some angle (Fig. 1).



a) b)

Fig. 1. Versions of collimator installation on the detector (top view):
a) collimator of the length h is perpendicular to the detector (L is the detector width);
b) collimator of the length h is declined from the vertical by an angle α (h_{eff} is the effective length of collimator).

Experiments were carried out with a VM-250AG vehicle portal monitor [3]. A distance between pillars was 6 m, collimators were installed on all four plastic detectors, width of the detector was 152 mm, and collimator thickness was 10 mm.

As a result of simple geometrical considerations it is possible to obtain the following dependence:

$$B/B_0 = 1 - [\mathrm{arctg}\ (K \cdot h/L)]\ /90 \qquad (2)$$

where: B_0 is the natural background value without collimators;
B is the natural background value with collimators;
h is the collimator length, L is the detector width;
K is the numerical coefficient; its value can be found by calculations and experiments.

The equation (2) describes a version when the collimator is perpendicular to the detector surface (Fig. 1a). When the collimator is declined by an angle **α** from the vertical (Fig. 1b), it is necessary to use its effective length $\mathbf{h_{eff}}$ instead of the actual collimator length $\mathbf{h}$:

$$h_{eff} = (L \cdot h \cdot \cos\alpha)/(l + h \cdot \sin\alpha) \qquad (3)$$

Fig. 2 presents the calculated (by the equation (2)) and experimental data of relative background values when collimators of different lengths are installed perpendicular to the detector surface. As seen, the use of the collimator does significantly reduce the background value and, therefore, should result in a decrease of the minimum detectable activity (mass) of a radioactive source.

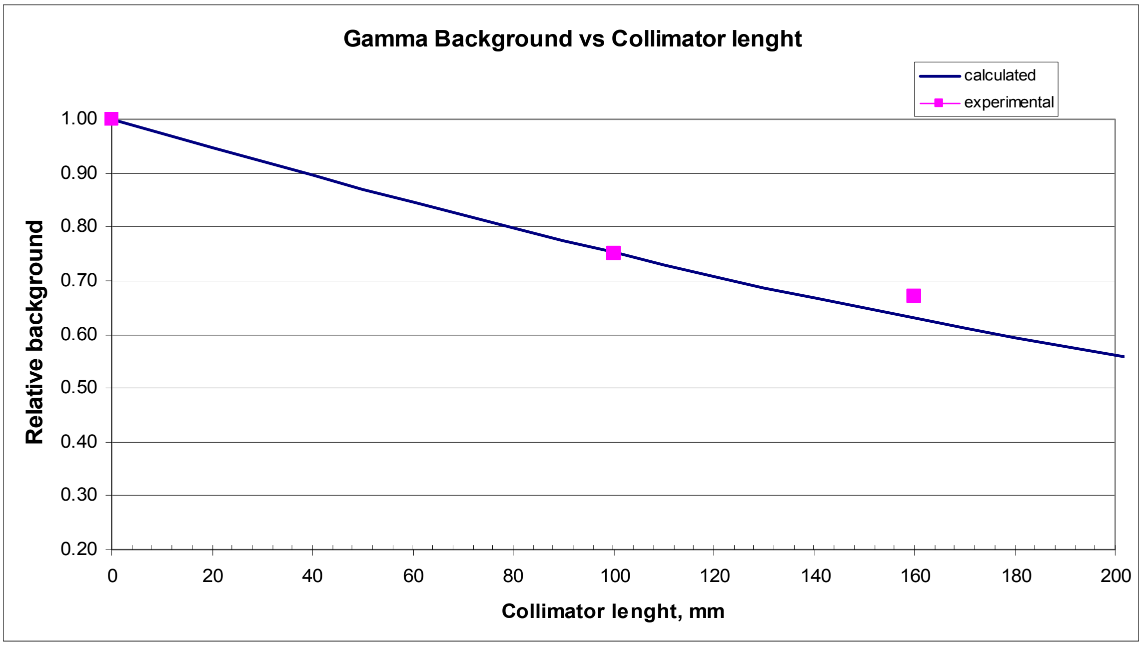


Fig. 2. Dependence of relative gamma background on collimator length (L=152 mm, K=0.625).

As known, when a vehicle is moving through the radiation portal monitor, the so-called "background suppression" effect, i.e. a decrease of recorded gamma background is observed. This effect is the cause of the fact that strong enough gamma sources which the monitor should detect easily become "invisible" [4]. Thus, a decrease of this effect also should promote the detection of weaker sources.

We have experimentally investigated the collimator influence on the "background suppression" effect by placing a truck between pillars of the portal monitor and measuring the gamma background values (Table 1). As seen from the data presented, the use of the collimator results in decreasing the " background suppression" effect both in absolute values (e.g. 90 cps instead of 160 cps for 160 mm collimator length), and in "sigma" values (2.9 instead of 4.3).

Table 1. Collimator influence on the background suppression caused by a truck inside of the radiation portal monitor.

Collimator length (alfa=0)	0 mm	100 mm	160 mm
Initial Background	1390 cps	1038 cps	933 cps
Backgr with truck	1230 cps	916 cps	843 cps
Backgr. suppression	-160 cps	-122 cps	-90 cps
Backgr. suppression	-4.3 sigma	-3.8 sigma	-2.9 sigma

II.2. Detection of the gamma sources.

The installation of a collimator results in two opposite effects. On the one hand, as it was stated above, there is a decrease of the natural gamma background measured by the monitor. On the other hand, the "field of view" of the detector is decreased and, as a consequence, the total time of a source exposition decreases too. Both of these effects depend directly on a ratio of the collimator and detector sizes. The efficiency of the collimator use is dependent on which of the effects is higher.

Let's consider the movement of a source with a speed of 2.2 km/s, in a distance of 3 m from the detector surface (a distance between the monitor pillars is 6 m). Fig.3 presents the experimental and calculated dependences of relative signal values on time without collimator and with collimator of 100 mm length. The Figure shows that the collimator installation causes a significant narrowing of the signal profile and, therefore, a decrease of the effective time of source exposition. If the collimator length is increased, the signal profile is even more narrowed.

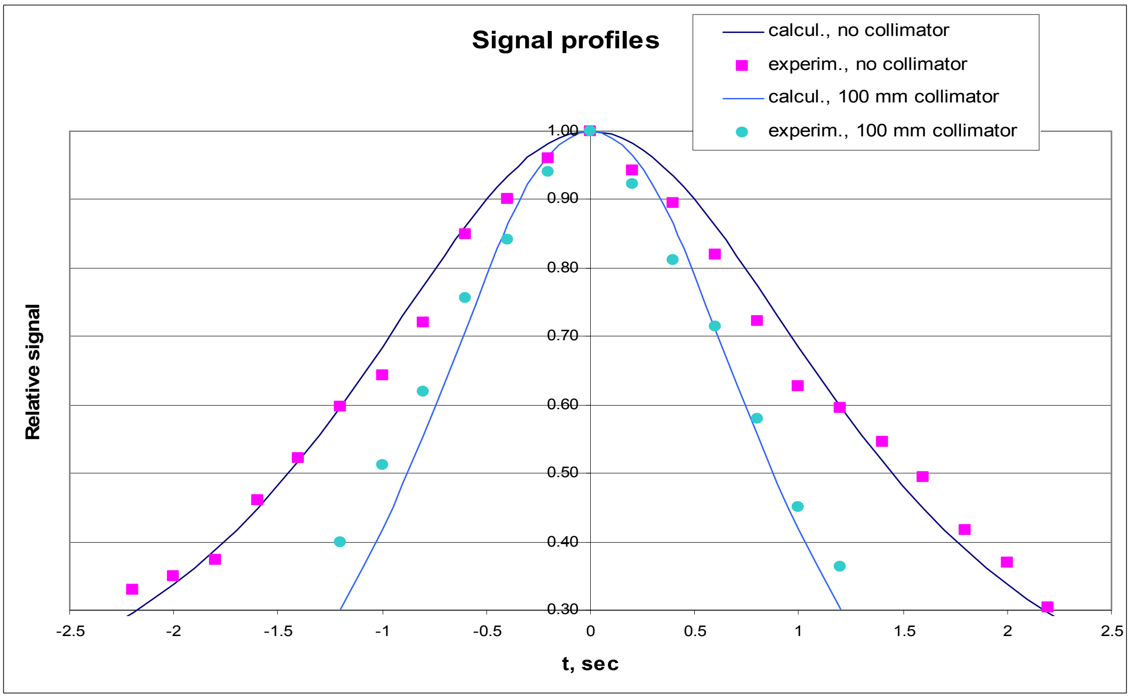


Fig. 3. Dependence of the signal profile on the collimator length.

It is possible to increase the source exposition time by changing an angle of the collimator declination (Fig.1), however, in this case the background also increases.

We estimated the results of mutual influence of these factors for collimators of different lengths installed under different angles Alfa to the vertical. The ratio of the maximum signal to the background "sigma" value was chosen as the studied parameter. The dependences obtained are presented in Fig. 4. As seen, the obvious maxima in the range of 20 degrees are observed for collimators of all the studied lengths.

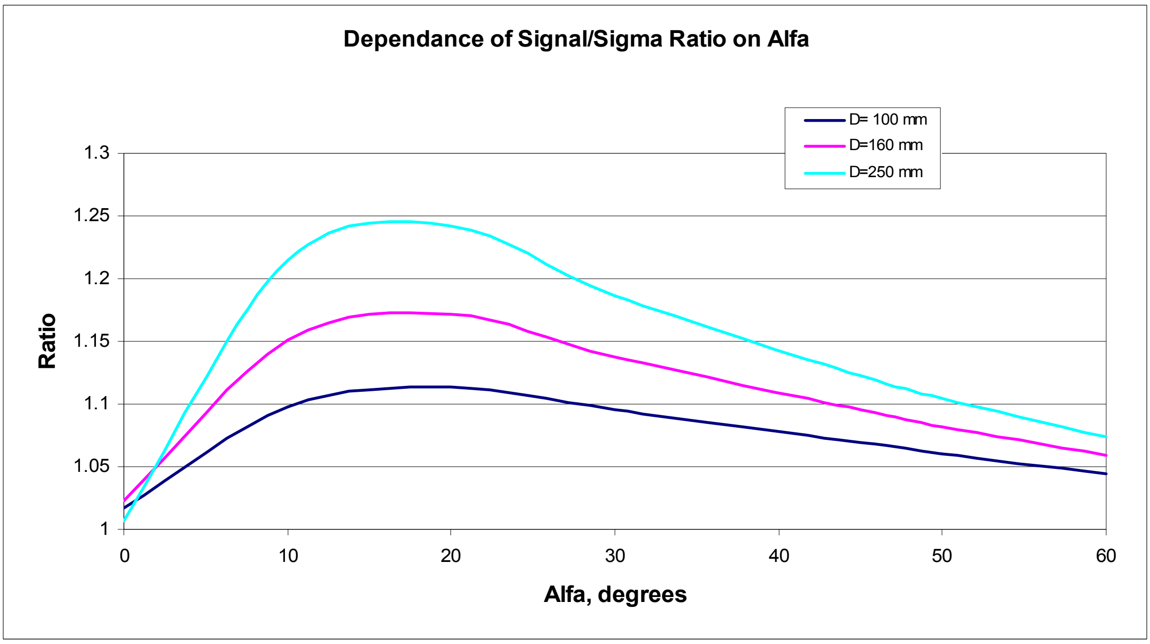


Fig. 4. Dependences of the signal/background ratio on the collimator declination angle Alfa for collimators of different lengths.

An increase in the signal/background ratio due to the collimator installation under an optimum angle results in an increase of the source detection probability. Dependences presented in Fig.5 testify that the collimator installation perpendicular to the detector surface (angle **α**=0) does not results in an increase of the source detection probability. However, the collimator installation at the optimum angle enables the monitor to detect a source (the detection probability is more than 90 %) which is practically not detectable without collimator (the detection probability is less than 60 %). Our calculations have shown that the collimator installation at the optimum angle results in a decrease of the minimum detectable signal by approximately 20 % (at the given detection probability).

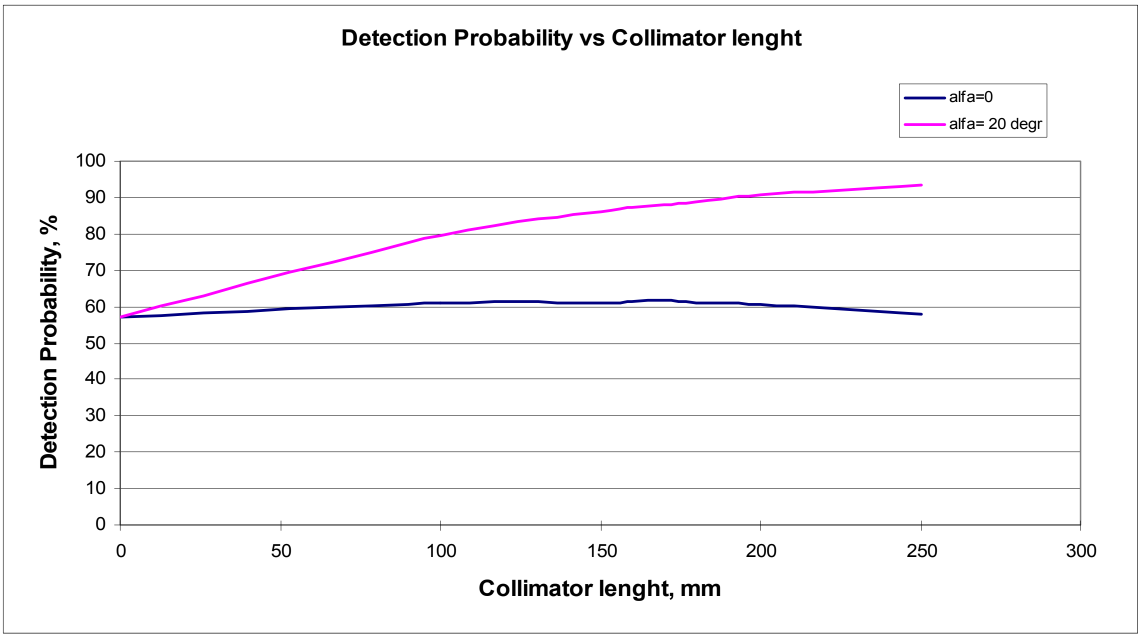


Fig. 5. Dependences of the source detection probability on the collimator length at different angles of the collimator declination to the vertical.

It is necessary to emphasize that the false alarm probability is the same for all the above cases as the alarm threshold (number of "sigma", equation (1)) was not changed in these calculations.

Conclusions

1. Installation of additional lead collimators on the plastic detectors of radiation portal monitors results in a decrease of the value of the measured gamma background. A decrease can reach 50% at collimator length of 250 mm.
2. Installation of the collimators causes as well a decrease of the detector "field of view" and, as a consequence, a narrowing of a signal profile; therefore, both the time of the source exposition and the probability of its detection decrease.
3. Installation of the collimators at some angle to the vertical results in an increase of the exposition time, however, the measured gamma background increases too. Calculations have shown that the maximum of the signal/background ratio for collimators of different lengths is observed at an angle of declination of 20 degrees.
4. Installation of the collimators at the optimum angle of 20 degrees provides an increase of the detection probability of the given source (or a decrease of the minimum detectable signal at the given detection probability) at the same false alarm probability.
5. We have paid the main attention to the collimator influence on **vehicle** portal monitors where the "field of view" of plastic detectors and the signal profile are of special importance. For **the pedestrian** portal monitors, a decrease of the measured gamma background can be more critical.
6. Additional advantages could be obtained with use of collimators of other shapes, for example, plate collimators [5].

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Measurements and simulations of fission neutron spectra at the MEDAPP beam at FRM II and subsequent developments

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Abstract

The neutron spectrum of the fission neutron beam of the MEDAPP facility at FRM II was simulated with MCNPX and transmission calculations. The simulations were verified by multiple foil activation using threshold reactions. The measured spectra were unfolded and equalisation calculations performed. The full flux of the beam was determined by three methods: a specially developed gold-waterbath-method; by the calculation on the basis of dose measurements in a water phantom; and by the combination of the results obtained by activation foils.

It was shown that the geometric and spectral shape of the beam are widely independent. Together with the measurement of the gamma spectrum of the beam, this paves the way for a dose-planning system for tumour therapy at FRM II and further developments like the construction of a new collimator to enhance geometrical beam shaping. An overview of these developments is given.

Introduction

The MEDAPP facility is located at the Forschungsneutronenquelle Heinz Maier-Leibnitz (FRM II) (Wagner 2008). See fig. 1 for an overview of the facility. It possesses a neutron beam consisting of mainly non-moderated fission neutrons. They originate from a pair of uranium plates containing 540 g U, 93% enriched in ^{235}U, with an effective area of 15x15 cm², fed by thermal neutrons from the reactor core. The plates are positioned in contact to the entrance of the beam tube SR10. At the exit of SR10, thermal neutrons are captured by a boron filter. The beam quality (spectrum and neutron-to-gamma ratio) can be adjusted using filters of lead and polyethylene. Geometrical shaping is achieved by means of a multi-leaf collimator (MLC). For the tumour therapy, only one well-defined beam quality is used. Besides, numerous other biological and technical applications are possible.

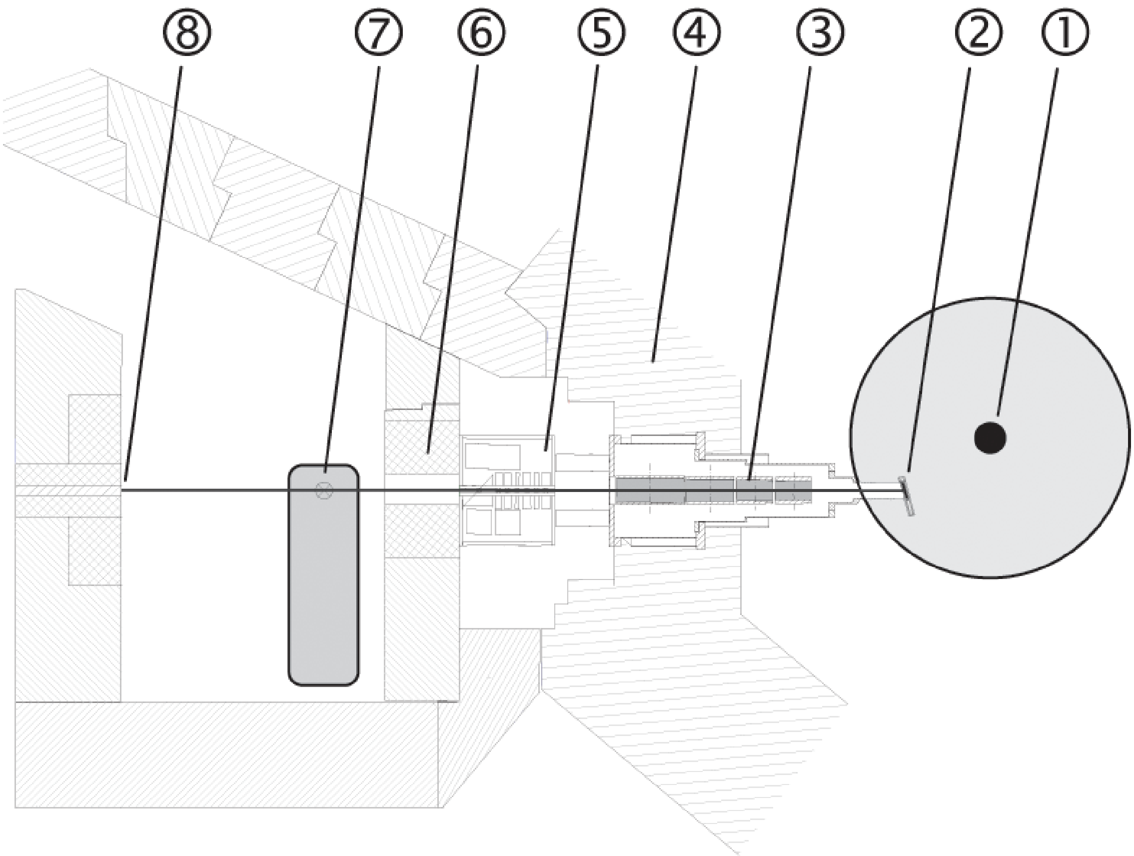

Fig 1. The MEDAPP facility. 1: Reactor core of FRM II inside a heavy water tank; 2: Converter plates; 3: Beam tube with shutters; 4: Biological shield (reactor pool wall); 5: Filter bench; 6: MLC; 7: Patient's position (SSD = 592cm); 8: Beam dump.

By now, the dose distribution is determined by measurements in a water phantom and results are transferred to the actual application at the patient. This delivers a sufficient accuracy for the irradiation of tumours situated near to the surface, although in the age of fast computers and modern simulation codes, a more precise-to-plan application is desirable. To develop such a dose planning system, five essential parameters of the beam must be known: Neutron spectrum and flux, gamma spectrum and flux and the correlations between the geometric position and those four items.

In a series of simulations and measurements, all parameters and correlations were determined. This paper mainly deals with the identification of the neutron spectrum, the neutron flux and the geometric correlations. In addition, an overview on the measurement of the gamma spectrum and a discussion on current developments based on these findings is given.

Neutron beam quality

Simulations

Prior to the measurements, simulations were carried out to be able to plan measurements more precisely and to lay the foundation for the unfolding of the spectra and the equalisation calculations. MCNP4C was used initially, later calculations were conducted using MCNPX, version 2.6.0. There was already a detailed model of the reactor core available which was used to calculate the first part of the simulation, from the reactor core to the converter and further on to the entrance of the beam tube. The spectrum at this point was analysed and it was shown that it fully matches a reactor spectrum consisting of a Maxwellian distribution of thermal neutrons, an 1/E-distribution for intermediate energies, and the Watt spectrum for fission neutrons.

From thereon, the simulation was continued using RSSA files which were modified as described in (Breitkreutz 2007) using the program "ModRSSA" which was developed especially for this purpose. Using the technique of cell flagging, it could be shown that due to the collimation and a SSD (source skin distance) of nearly 6 m, the contribution of scattered particles to the resulting spectrum at the position of the patient is insignificant. It was therefore possible to calculate much more detailed spectra by means of straightforward transmission calculations applied to the spectrum at the entrance of the beam tube.

Due to the limited statistics that were achievable using RSSA files, it was not possible to extract the geometric-spectral correlations mentioned above in detail. However, there was no indication that such a correlation exists in an extent relevant for tumour therapy. As a consequence of this, the problem of the large source area and its projection onto the target could be solved by a straight-forward geometric approach, realised in a program called "ausleuchtung". This program calculates the geometric shape of the beam, i.e. the total neutron flux dependent on the geometric position in the treatment room, by summing up all visible source points (silhouettes / shadow image approach).

Measurements

The measurements consisted of three parts, namely the verification of the spectrum using threshold activation foils, the determination of the total neutron flux by a bath method, and the verification of the shadow image assumption used in "ausleuchtung", i.e. the determination of the beam profile after the MLC. (Breitkreutz 2008)

To measure the fission neutron spectrum, a total of 19 reactions with threshold energies ranging from 0.17 MeV, ^{103}Rh (n,n') ^{103m}Rh, to 12.40 MeV, ^{58}Ni (n,2n) ^{57}Ni, were used, with most reactions starting below 3.5 MeV. This high number of different reactions was possible due to the high intensity of the beam and barely no disturbing underground from thermal neutrons. After activation, the resulting radioactive isotopes were identified using a HPGe detector.

The total neutron flux density was determined using three different techniques: It was calculated from the activation foils with supplemental (n,γ)-reactions in combination with equalisation calculations below. Second, as the neutron spectrum was known, the fast flux could be calculated from depth dose measurements using the twin chamber method in a 200-l-water phantom (PTW-Freiburg). Third, the total flux was measured in the same water phantom in which gold foils were arranged in equidistant positions throughout its volume. This technique is similar to the well-known $MnSO_4$ bath method, but avoids handling large amounts of radioactive fluids and allows well to distinguish the contributions from thermal and epithermal neutrons. The horizontal beam was guided in the middle of the bath via an empty closed aluminium tube. The efflux from the bath was also determined using gold foils on its surfaces. This set-up is shown in fig. 2. The activation was again determined using a HPGe.

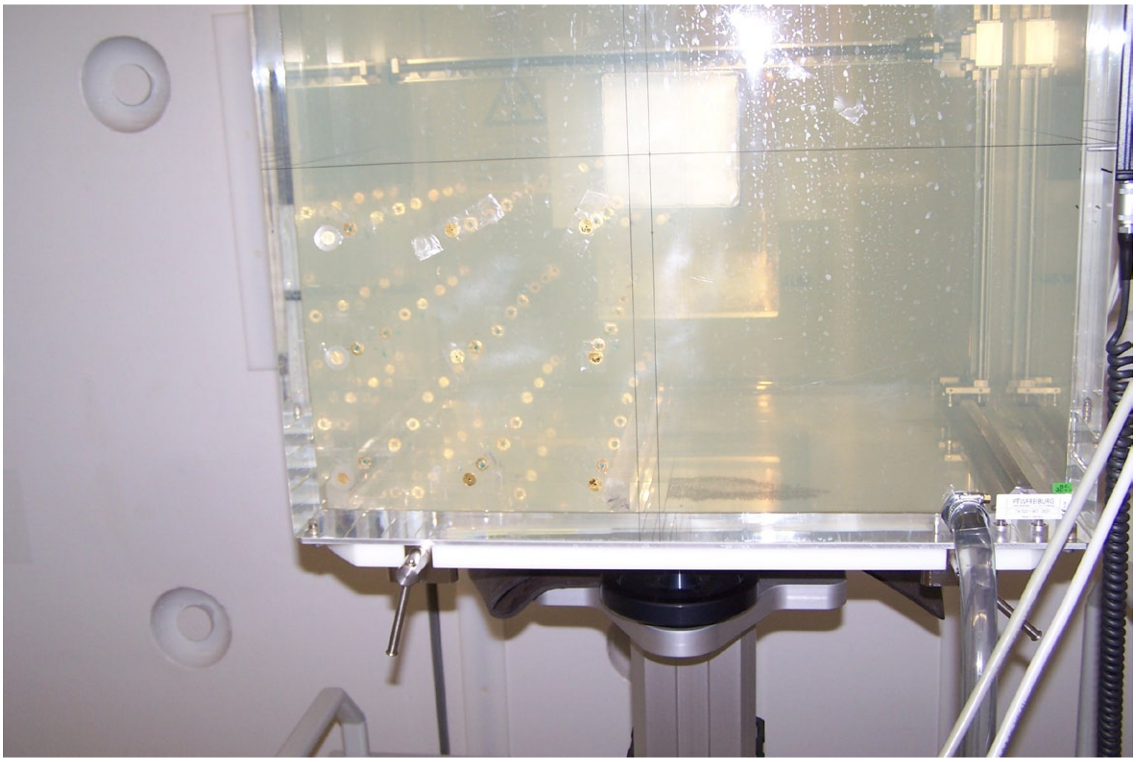

Fig 2. Measurement of the total flux by use of equally distributed gold foils in a big water bath. For symmetry reasons, it was sufficient to arrange about 150 foils in one quadrant around the beam.

To determine the correctness of the shadow image approach, a 1D flux profile was recorded using indium wires. The reaction ^{115}In (n,n') ^{115m}In has a threshold energy of 0.35 MeV. Of course, spectral changes can not be captured completely in this way, but it can be assumed that a significant spectral change goes hand in hand with a change of the flux above the threshold energy. Therefore, if the beam profile simulation and the 1D measurement agree well, it can be concluded that the spectral-geometric correlation is insignificant above this threshold. MCNPX calculations support this assumption.

As an additional verification, spectra at various positions in various depths inside a PE phantom were measured and simulated. It could be shown that the fission spectrum is hardened with increasing depth, though not significantly within the first five centimetres, and a thermal neutron flux in the same order of magnitude as the fast neutron flux builds up.

Equalisation calculations

Due to the low information density contained in the activation foil measurements and the high uncertainty in some of the cross sections for the fast reactions, it is hardly possible to unfold the spectrum from these measurements without any prior knowledge. Therefore spectra from transmission calculations and MCNPX simulations were used as start spectra. It turned out that the classic unfolding codes like MSANDB tended to converge to solutions that could not be explained by the materials in and the geometry of the beam. In fact, this behaviour could be suppressed when some measured reactions were excluded, but this approach is not easy to justify and, hence, was not followed up. Instead, equalisation calculations using MSITER were conducted. These calculations roughly follow the procedure required by DIN 1319-4, even though the (co-) variances of cross section uncertainties are not taken into account.

Neutron results

All three flux measurements described above delivered a total neutron flux of 3.2(1)·10^8 n/(s cm²). The simulated neutron spectrum was essentially confirmed; only with slight adjustments above 5 MeV and in the intermediate range between thermal and epithermal neutrons, where. the MCNPX simulation showed large statistical errors.

The mean energy of the spectrum was determined to be 1.9(1) MeV, slightly lower than the 2.0 MeV of a pure fission spectrum. The final spectrum is shown in fig. 3.

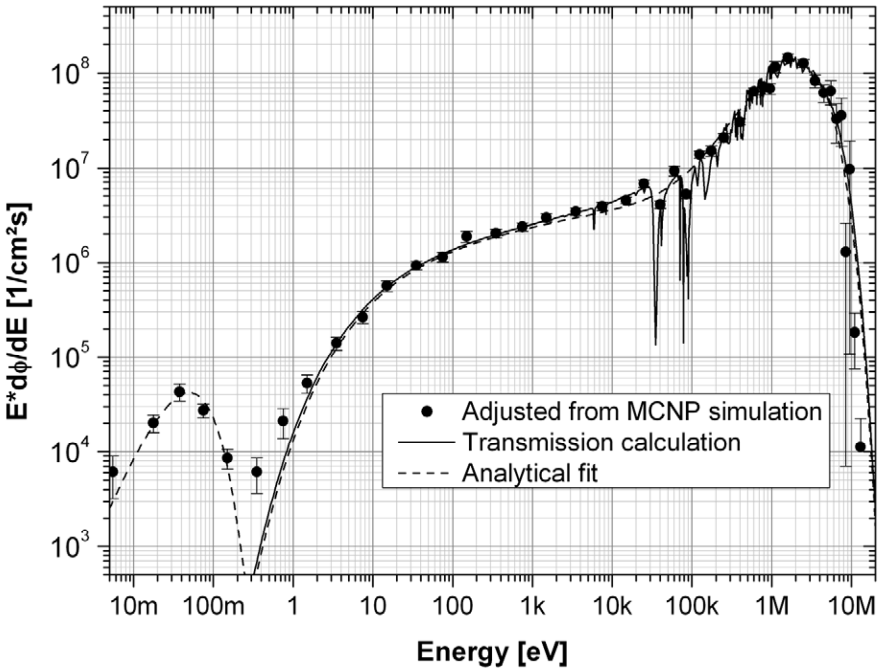

Fig 3. Neutron spectrum of MEDAPP at the patient's position. The small thermal peak results from scattering in the MLC which is not included in the transmission calculation. Due to the low resolution of the MCNP simulation (43 energy groups), the resonances from lead (filters) and aluminium (covers) clearly show up only in the transmission calculation.

Gamma beam quality

Simulation

A simulation of the gamma part of the beam was not fully possible until support for delayed gamma emission was added in MCNPX 2.6. Before, only the prompt gammas emitted during fission as well as on capture events in neutron-absorbing structure materials and the related Compton part could be simulated. With MCNPX 2.6, the delayed gammas emitted by fission products and activations due to neutron capture reactions were taken into account by means of an approach with 25 energy groups. Essentially the same models as for the older neutronic simulations were used for the simulation.

However, in the case of gamma radiation it was hardly possible to achieve satisfactory statistics by the original simulation approach using RSSA files at the beam tube entrance. To overcome this deficiency, the program "autospect" was developed. It is similar to "ModRSSA" but features some kind of peak detection routine as the complex gamma spectrum can not be fitted with a closed expression as easily as the neutron spectrum. "Autospect" starts to acquire the particles stored in the RSSA file with a high number of energy groups. It then detects peaks in this group structure by the unsteadiness of the statistic error and starts condensing the remaining energy groups until a specified level of statistics is reached. In this way, an inhomogeneous group structure is produced with sharp peaks and wider ranges in-between, all with an acceptable statistic error. With the spectrum gathered by "autospect", the simulation was continued at the entrance of the beam tube and brought further on to the patient's position. Fig. 4 shows the spectrum as determined by "autospect" and a transmission calculation to the patients position. Some important peaks like the 1779 keV ^{28}Al decay peak are missing as they are hidden in the coarse delayed gamma group structure mentioned above.

The measurements described below were simulated apart. For this, a MCNPX model of the detector was created.

Measurements

The measurement of the gamma spectrum proved to be a far higher cradle than the neutron spectrum. Due to the inevitable fission neutrons and the high intensity of the beam, the spectrum could not be measured directly in the therapy beam with a gamma detector (Jungwirth 2009).

Two measurements were conducted to determine the gamma spectrum: First, a measurement on the beam axis at very low reactor power (approximately 50 kW instead of 20 MW) with a 5" NaI using additional attenuators. At full reactor power, only scattered photons were measured under a defined angle apart of the direct beam using the 5" NaI detector and, for better resolution at lower energies, a HPGe detector. A special collimator was constructed for the latter measurement.

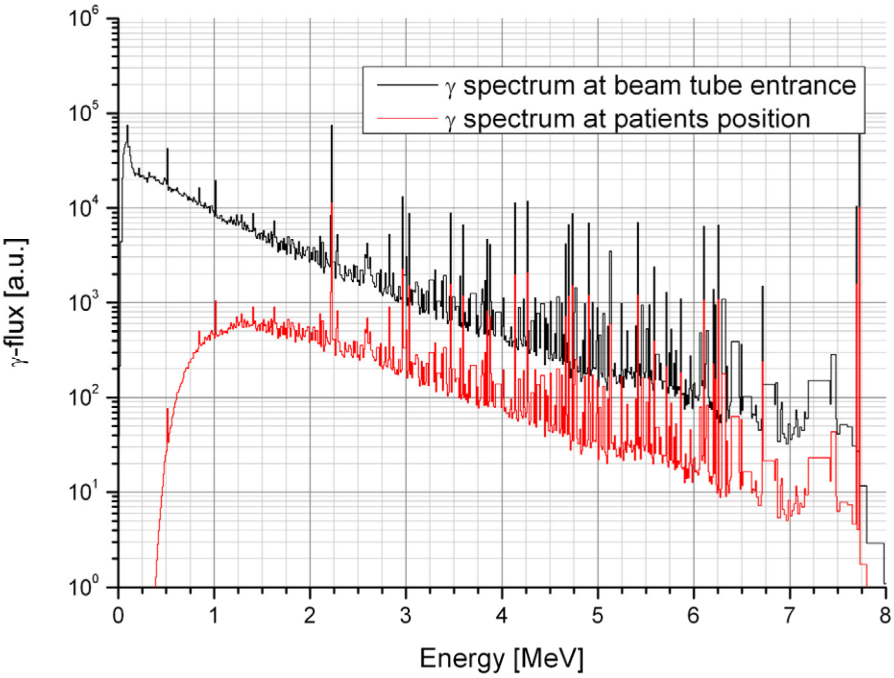

Fig 4. Gamma-spectrum determined by "autospect" and spectrum at the patients position calculated by transmission calculation.

Fig. 5 shows a measured gamma spectrum on the beam axis with partially closed shutters, i.e. with about 20 cm iron in the beam. In this case, all major peaks from fig. 4 can easily be identified, except decay peaks that were not included as line emission but within the 25 energy group structure in the simulation.

The total gamma flux was determined similarly to the neutron flux determination by evaluation of a depth dose curve in a water phantom in comparison to a simulated one. Of course, in this case a good knowledge of the spectrum is required, too.

Results

The simulated detector showed a drastic underestimation of the Compton spectrum. Peak efficiencies were ok (±4%). As a result of this, all measured spectra showed an increasing deviation in the high energy range starting from about 3.5 MeV up to the measured maximum of 10 MeV (or, in other words, an underestimation of the low energy part). Moreover, the depth dose curve showed deviations in depths larger than 5 cm, also pointing towards an overestimation of the high energy part in the simulations.

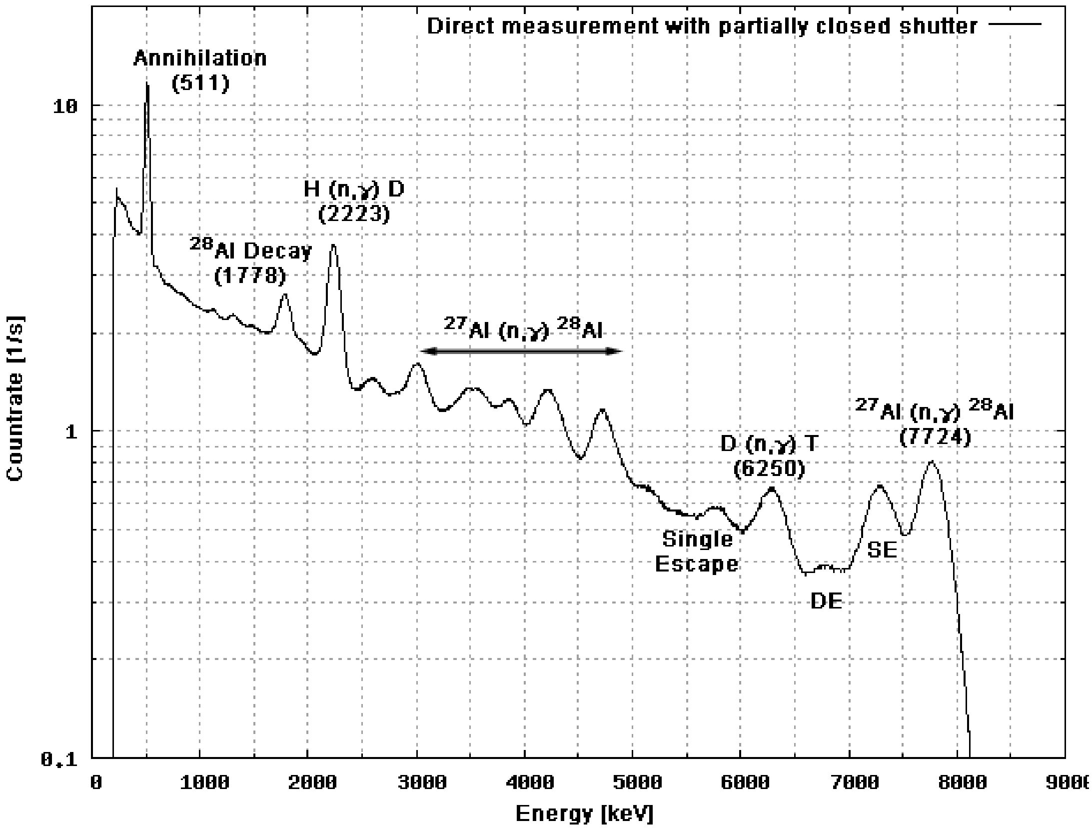

Fig 5. Measured gamma spectrum directly in the beam with partially closed shutters.

The mean energy of the gamma part of the beam is about 2.4 MeV, the total gamma flux 3.7(1) · 10^8 γ/s cm². As explained before, the accuracy of the total flux assignation depends on the correctness of the assumed spectral shape which is not the case here to the full extent. Therefore the total flux was determined from the first three depth dose points only. The uncertainty of the number given above is therefore by trend higher than quoted here. We are currently investigating the reasons for the deviations in the spectrum.

The spectral-geometric correlation has not been measured yet. However, results from simulations again suggest that there is no significant correlation.

Further Developments

A new MLC

As the current design of the beam collimator is not optimal regarding the large penumbra around the main beam, a new collimator is currently being constructed with improved field shaping and thereby increased radiation protection for the healthy tissue of the patient.

Based on the verified shadow image approach, a program "ausleuchtung" (see sec. 2.1) was developed which can be used to simulate the geometric shape of the beam depending on the collimator setting within seconds. With its help, the construction of the new collimator was optimised towards steep beam intensity edges for small fields as they are predominantly needed for irradiations of the head and neck (Schenk 2009).

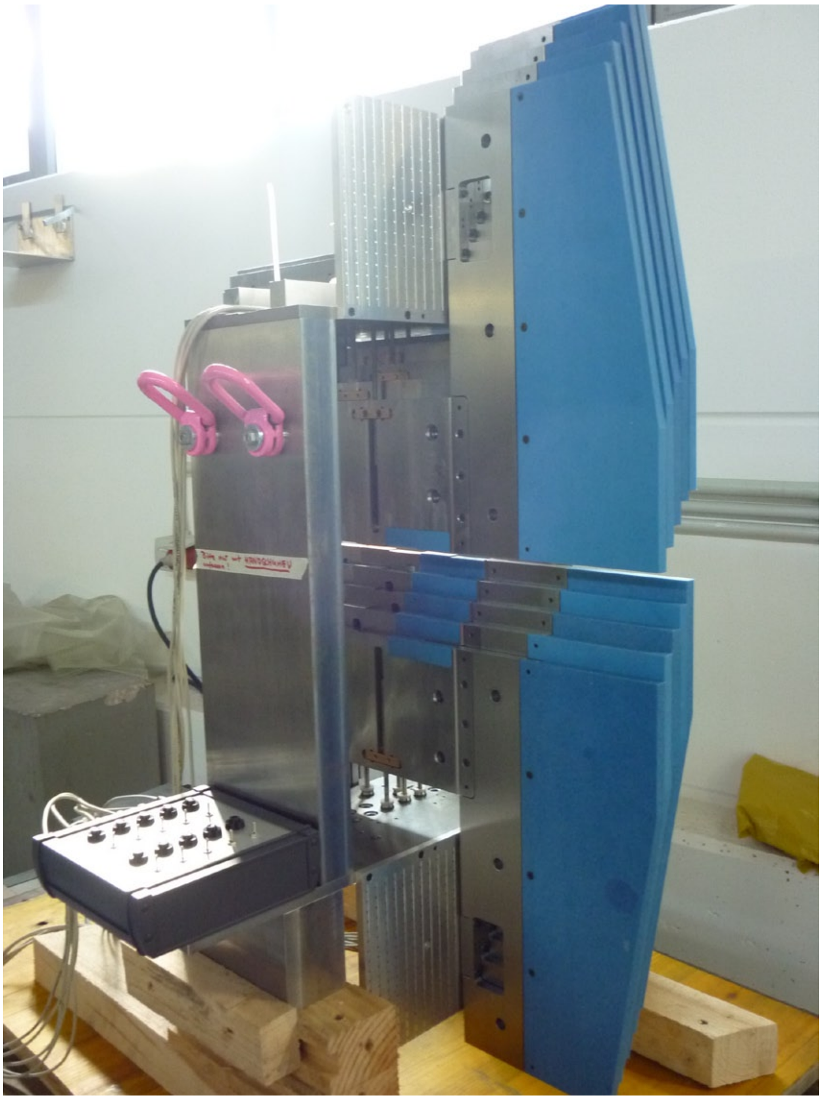

Fig 6. Prototype of a new collimator for MEDAPP.

Using the measured neutron and gamma spectra, the shielding and self-shielding against activation of the collimator is being optimised, yielding an increased radiation protection for the staff of MEDAPP. The collimator is electronically driven so that the collimator can be closed automatically and the medical staff is not exposed to the activated metal parts of the MLC.

Towards a dose planning system

For the development of a dose planning system, it is essential to know all the beam parameters mentioned in the introduction. As soon as the remaining discrepancies with the gamma spectrum are solved, all these parameters are known. In fact, spectral uncertainties play a minor role in the medical treatment because the photon component of the beam contributes only about 10 % to the total RBE-dose.

By now, based on the approach of "ausleuchtung" and MCNPX, a program was developed which utilises all the findings described above to simulate 3D depth dose distributions in a water phantom in a reasonable amount of time (some hours). The errors which were introduced to the depth dose curve by all the measures like denying all spectral-geometric correlations and the adjustments made by neglecting scattered particles in the beam tube are assumed to be negligible compared to other uncertainties like the exact elemental composition of the patient's tissue or cross section uncertainties for fast reactions.

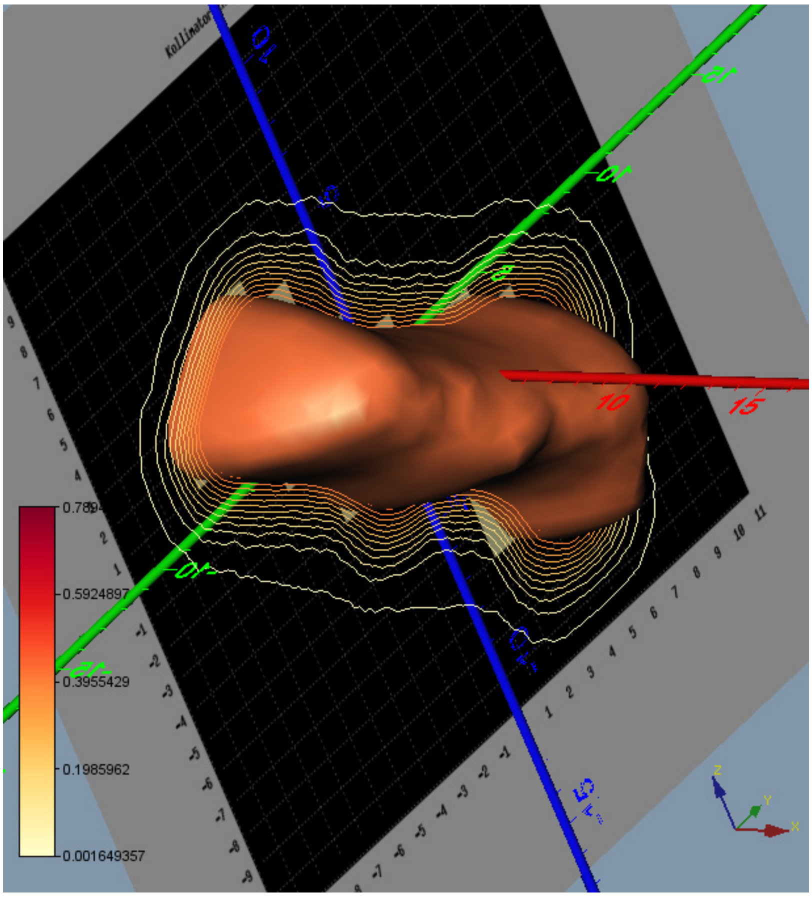

Fig 7. Example of a water phantom dose plan for MEDAPP: 2D and 3D iso-doses for a certain MLC setting (background image). The beam is entering along the red x-axis from behind through the MLC image.

Conclusions

The measurement of the neutron and the gamma spectrum and the determination of the spectral-geometric correlations have laid the foundation for the development of a dose planning system and a number of other programs. The radiation protection of the patient will be brought a good step forward when the dose planning system is eventually realised. The enhancements in radiation protection by the construction of a new collimator, which is optimised towards field shaping, again is mainly for the benefit of the patient. The staff profits from the enhanced activation self-shielding of the new collimator and more precisely calculable activation after long-term irradiations when the facility is used for technical applications.

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Feasibility study of a low cost wireless ionizing radiation sensor network

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Abstract

The Radiation Protection Department at the Technical University Eindhoven has developed the operational need for an ionizing radiation monitoring instrument mapping real-time radiation emanated by industrial processes into the environment and workplace. Available instruments are energy consumptive, static and labor intensive in data collecting so this feasibility study will investigate an alternative. The aim of this feasibility study is to design a device collecting dosimetric data in a more economical and ergonomically manner. The end product is a modular, energy efficient wireless radiation sensor which constitutes automatically a flexible, secure and failsafe network with multiple modules. A module consists of a ZigBee protocol based transceiver capable of forming several types of network topologies and a MDA300CA sensor board interconnected with a custom-made ionizing radiation sensor. The radiation sensor is constructed using a widely used PIN-photodiode sensitive to photon energies ranging from 35 keV till 1173 keV. The received data streams are easily viewed and interpreted using an open-source PostgreSQL database. The wireless radiation sensor network is a real-time monitoring instrument which can be implemented in existing organizations and expanded on in several ways using commercially available components.

Introduction

For radiation protection of workers and environment a source oriented approach is indispensable. For such an approach to be effective, operational area monitoring is very helpful. Traditionally, area monitoring is done with handheld survey monitors or by fixed area monitors. The latter monitors are sometimes equipped with local or remote alarm indications. If source conditions in a workplace are relatively stable, periodic manual surveys are usually quite adequate. However, in situations like ours of strongly fluctuating source strengths or variable locations, real time monitoring is often desired.

Eindhoven University of Technology hosts a 30 MeV proton cyclotron, mainly used for large scale radionuclide production and radionuclide processing facilities. In all stages of production and processing there has been a growing need to monitor radiation levels in more detail in workplaces. Until now this has been done by a wired network of area monitors (wall mounted GM tubes) and by many small scale active and passive dosemeters. We found the wired network to be inflexible and expensive; small standalone dosemeters fit our needs much better, but require of lot of manual work and manual data processing.

Therefore, we strive for an area monitoring system that is networked and flexible. The main goal of this study is to investigate the feasibility of a cost-effective modular area monitoring system, consisting of wireless sensors.

Material and methods

The system we designed has modules with ZigBee protocol based transceivers from Crossbow Technologies, capable of forming several types of network topologies and a MDA300CA data acquisition board with a custom made radiation detector. The detector was constructed using a widely used PIN-photodiode, capable of detecting photon energies ranging from 35 keV to 1332 keV.

Some properties like sensitivity, linearity and efficiency of the PIN-photodiode were investigated by using different calibration sources. Especially efficiency at low photonic energies was expected to be increased so filtration might be a crucial aspect (*C-R. Chen et al., 1993, R.H. Olser et al., 1991*). Calibration has been done with secondary standard calibrated Cs-137 sources with different activities (*P.W. Cattaneo, 1991*).

ZigBee

The ZigBee Alliance uses an energy efficient protocol IEEE 802.15.4 for wireless transmitting data (*P. Baronti et al.*, *2007*) and distinguishes three different devices. The *ZigBee coordinator (ZC)* is the most capable device. This coordinator forms the root of the network tree and might bridge to other networks. There is only one ZigBee coordinator in each network since it is the device that started the network originally. It is able to store information about the network, including acting as the Trust Centre & Repository for security keys. The *ZigBee Router (ZR)* runs application functions and is acting as an intermediate router, passing on data from other devices. The *ZigBee End Device (ZED)* contains just enough functionality to talk to the parent node (either the coordinator or a router); it cannot relay data from other devices. This relationship allows the node to be asleep a significant amount of time, combined with efficient use of memory, therefore giving long battery life.

In our experiments the integration of a ZigBee transceiver module, an experimental board (MDA300CA), the sensor and electronics and electromagnetically shielded wiring operates as a ZR or ZED, all powered by two LR6 (AA) batteries. The signals coming from the detector are sent to an internal counter in the microprocessor. This value is then stored, time-stamped and combined with data from other sensors and finally transmitted to the nearest module using the 2.4 GHz open band so the data collected by the sensory node in this network is eventually transmitted to the base node. There is no interference between other protocols, like 802.11a/b/g (WiFi), using the 2.4 GHz open band frequency. A ZigBee transceiver combined with an USB-connected programming board (MIB400) acts as a ZC.

The sensor

The sensor used is a daylight filtered PIN-photodiode (OSRAM BPW34FS) with a designed sensitivity range of 780 nm and 1100 nm. This semiconductor is commonly used for measuring infrared photons or X-rays but is also quite suitable for detecting photons in a range between 35 keV and 1173 keV.

A PIN-photodiode consists of three layers where the intrinsic (I) high resistance layer is important for detecting photon energies higher than 10 eV (*C-R. Chen et al., 1993*). The intrinsic layer reduces the leakage current between the P-N-junctions and also acts as an active volume for high energy photons and secondary electrons originated in the active volume or surrounding materials. Undiscriminated and unfiltered, the efficiency of a PIN-photodiode steeply increases at 10 keV and shows a decline until 10 MeV as seen in figure 1 (no filter).

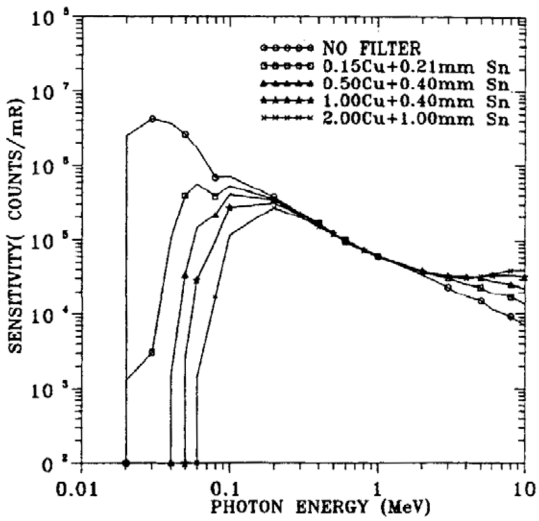

Fig. 1. Calculated sensitivities (in counts/mR) for several filter thicknesses (C-R. Chen et al., 1993).

We have limited our measurements between 35 keV and 2 MeV. Sensitivity and efficiency has been investigated by using several calibrated radioactive sources; Am-241, Ba-133, Cs-137, Mn-54, Co-60 and Na-22.

The electronic design

The used design is based on a PIN-photodiode (OSRAM BPW34FS) (*B. Denmark, 2003*). Major concern is noise reduction, therefore at the end of the amplification chain a discriminator cuts off the signal at low energies causing effects like reducing sensitivity as seen in figure 2.

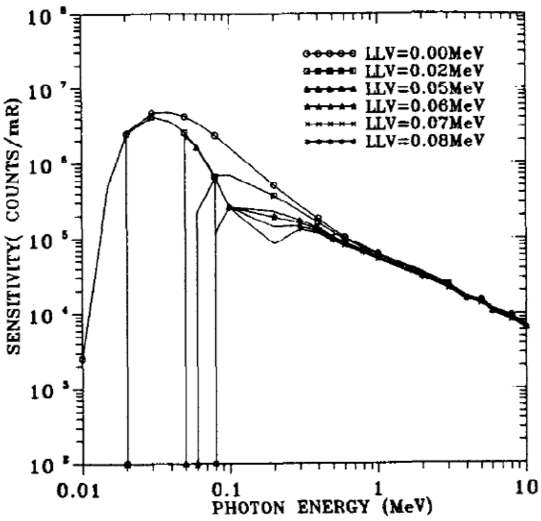

Fig. 2. Sensitivity curves with different settings of discrimination level (*C-R. Chen et al., 1993*).

The PIN-photodiode has a spectral range of sensitivity between 780 and 1100 nm. Exclusion of electromagnetic interference and daylight is achieved by a metal casing. The PIN-photodiode is embedded in the print so its angle sensitiveness is maximized.

In beneficially deploying this network power consumption of the sensor electronics had to be reduced. The energy consumption of the detector was about 80-90% of the total energy consumption of a node. The detector electronics consumes about 7 mA in active mode, for the greater part due to amplification electronics (4x MAX4475). The amplification electronics can be set in standby mode using an external digital pulse. Using this standby mode the power consumption of the detector electronics is about 0,001 mA.

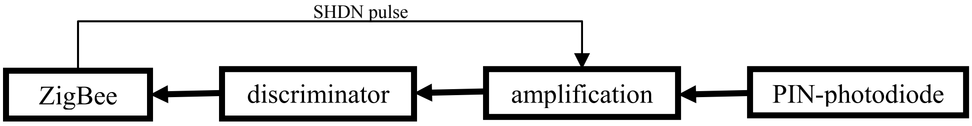

Fig. 3. Schematic view of the detector electronics.

The voltage ripple created in the PIN photodiode enters a four stage amplification line as seen in figure 3. After final amplification the pulse height is cut off with a pulse height discriminator so the pulse resembles a square wave. These square waves are used as input for the counter in the transceivers' microprocessor (ATMEL ATmega1281). To reduce power consumption the shutdown mode of the amplification chips (4x MAX4475) are routed to the transceiver module. This enables development of future measuring algorithms.

For valid representation of the collected data it is necessary to calibrate the detector. The calibration has been done with secondary standard calibrated Cs-137 sources with different activities. Using an Am-241 source (490 MBq) and the calibration installation of the Radiation Protection Service of the University of Technology Eindhoven the linearity of the detector has been investigated by verifying the inverse square law. The measured data are stored in a provided open source SQL database (PostGreSQL) for further analysis.

Results

The discrimination level setting has a strong influence on the detector efficiency for photon energies up to several hundreds keV as seen in figure 4. There is an optimum between noise reduction and the low energy cut-off level. This reduces sensitivity and flattens the efficiency curve. Therefore sensor filtration was not required like in figure 1. Our prototype is capable of measuring photon energies from 35 keV, using I-125 seeds, up to 1173 keV, using a Co-60 source.

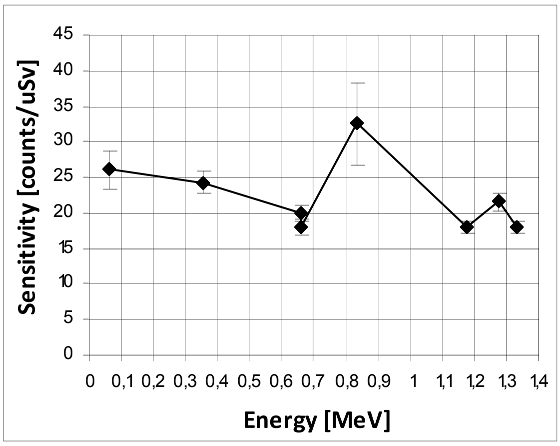

Fig. 4. Sensitivity of BPW34FS_no_filter using a set of calibration sources.

The measured sensitivity of the detector is about 20-25 counts per microSv.

The discriminator voltage is Vcc dependent. Battery voltage should not be below 2.3 V to prevent signal loss.

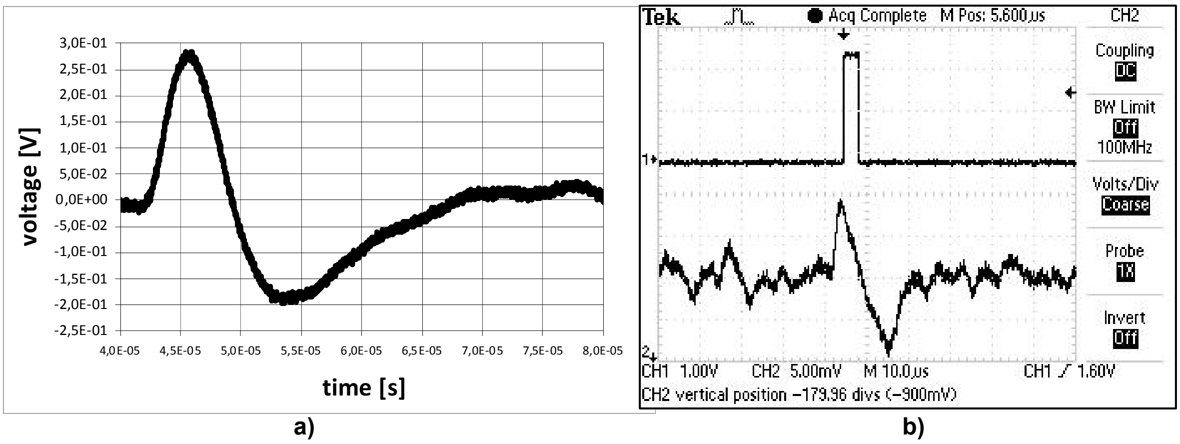

Fig. 5. a) Non-discriminated pulse created using a Cs-137 source and b) X-ray photon pulse converted into a square wave.

The pulse width varies between 5 and 25 microseconds and increases with higher photon energies as seen in figures 5a en 5b. The internal counter is capable of detecting these short pulses so the transceiver can send every second a time-stamped data packet with the amount of counted pulses.

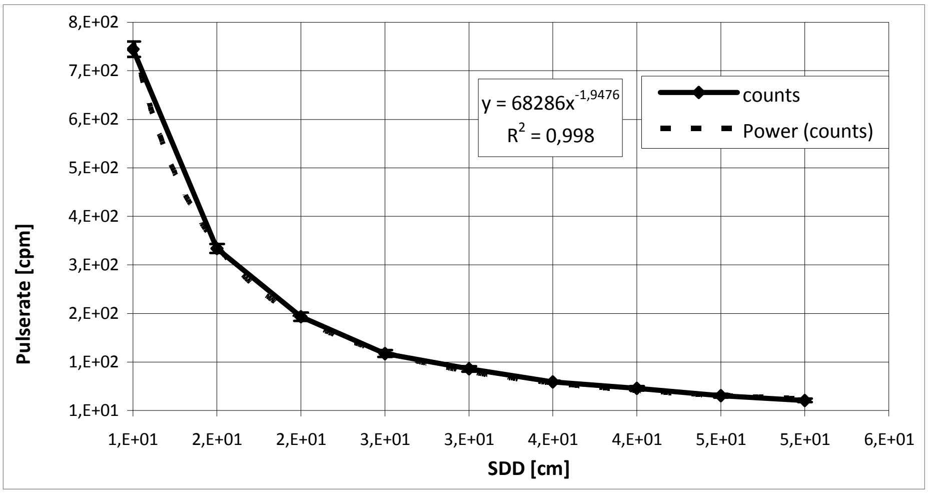

Fig. 7. Measured linearity of BPW34FS_no_filter using Am-241.

The linearity of the detector node has been tested using Am-241 (490 MBq). The pulserate measured in counts per minute follows the inverse square law indicated by figure 7.

Discussion

The first prototype with one gateway, one repeating node and one detector node was developed and tested successfully. The concept has good prospects for use in small to medium sized installations. Nevertheless, a complete wireless radiation sensor network has not been tested yet. In future detector nodes the node programs have to be modified so the frequency of emitting packets is reduced to a data packet every minute. This will also reduce power consumption.

Lowering the temperature of the PIN-photodiode by applying a Peltier element on the back of the PIN-photodiode could reduce noise by a factor 10 (*Z. Bian et al., 1985*).

For further sensitivity maximization a scintillation crystal can be mounted on the sensitive sides of the diode (*E. Fioretto et al., 2000*) or several PIN-photodiodes connected in series acting as one detector (*A. Sertap Kavasoglu et al., 2008*). The detector has not been tested in high frequency pulsed radiation fields.

The cost of a complete network system depends on the mapping detail degree, the size of the location, the signal attenuation between nodes, all in all, the amount of nodes. The cost of a detector node depends on further development of the hardware, like miniaturization of detector design, sensitization of the sensor and integration of detector and transceiver module in one design, and further development of software like data processing alarm algorithms, node programming and a comprehensible end-user software interface.

Conclusions

The development of a low cost reliable radiation detector is feasible. This study showed good sensitivity, efficiency and linearity of the detector. The wireless radiation sensor network is a real-time area monitoring instrument which can be implemented in existing organizations and workplaces. The wireless radiation sensor network can be expanded in several ways using commercially available components. Using a flexible wireless sensor platform creates a foundation for future applications determining activities in facilities for mapping radiation fields.

Acknowledgements

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Results of model calculations and algorithm development related to a 3D silicon detector dosimetric telescope

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Abstract

One of the many risks of long-duration space flights is the excessive exposure to cosmic radiation. The dose equivalent in orbit may be two orders of magnitude higher than that under the shield of Earth's atmosphere. In order to determine the dose equivalent on board different spacecrafts, the development of a 3D silicon detector telescope got underway in the Hungarian Academy of Sciences KFKI Atomic Energy Research Institute several years ago.

In the course of my doctoral research I was taking part in the R&D activities related to the instrument, such as designing the measurement system, developing the electronic system, reconciling the scientific, the electronic and the mechanical considerations of the development, working out the algorithms of the on-board software as well as coordinating, controlling and managing the development (Hirn 2009a). The present paper addresses two important issues of my work: the method of determining in almost real time the time intervals of the South Atlantic Anomaly (SAA) crossings of the International Space Station (ISS) and the calculations performed in relation with the anisotropy of the cosmic radiation field in low Earth orbit in order to compare the performances of 1D and 3D telescopes on board ISS. To switch between the SAA and non-SAA spectra I developed an algorithm, which has been tested with model time spectra generated from time spectra measured by the DOSTEL 1D telescope. I calculated the parameter of the algorithm such that the probability of false switching was less than 10^{-3}. With model calculations I also analyzed the effects of the angular dependence of the geomagnetic cut-off, the shielding of the Earth and the East-West asymmetry of the trapped particles on the measured deposited energy spectra. The results have shown that in case of the untrapped radiation the differences in the spectra could be attributed only to the shielding effect of the Earth. As for the trapped radiation a more pronounced directionality was obtained.

Introduction

Due to the lack of shielding provided by the atmosphere of our planet, the dose rates measured on the International Space Station (ISS, at an altitude of ~340 km) are about two orders of magnitude higher than those from cosmic origin measured on the Earth's surface. Since dose equivalent, which characterizes the stochastic biological effects of the radiation, was defined in terms of a LET (linear energy transfer)-dependent quality factor, determining the LET spectrum and the quality factor of the cosmic radiation is necessary. For this reason, the development of a 3D silicon detector telescope (TriTel) started in the KFKI Atomic Energy Research Institute (AEKI) several years ago. The instrument comprises three mutually orthogonal, fully depleted PIPS detector pairs that are connected as AND gate in coincidence and has almost uniform sensitivity in 4π. A big advantage of TriTel, either as a stand-alone instrument or when operated together with the Pille thermoluminescent dosimeter system (Feher et al. 1981), is the capability of determining not only the absorbed dose but the LET spectrum and the average quality factor of the cosmic radiation as well (Pazmandi et al. 2006). From the absorbed dose and the quality factor the dose equivalent can be also determined.

Anisotropy of the cosmic radiation in low Earth orbit

The galactic cosmic radiation (GCR) in the near-Earth free space is approximately isotropic. However, because of the shielding effect of the Earth's magnetic field and the Earth itself, there is a lower limit on the energy of primary cosmic ray particles to enter given points in low Earth orbit from different directions (geomagnetic cut-off, Lemaître, Vallarta 1933). The solar component of the cosmic radiation also shows significant anisotropy. Its intensity on board ISS is usually negligible compared to GCR. However, its contribution to the radiation environment may be significant during solar particle events. My work did not address the anisotropy of the solar component.

Even at lower altitudes, due to a shift (~500 km) and tilt (~11°) of the geomagnetic axis compared to the Earth's rotational axis, the magnetic field is not symmetrical in relation to the rotational axis. Therefore in the region between South Africa and South America the inner radiation belt protrudes into lower altitudes (2-300 km). This designated region is called the South Atlantic Anomaly (Stassinopoulos, Staffer, 2007). The International Space Station passes the SAA in two time windows a day and in 2-3 consecutive orbits in a time window. The time elapsed between two time windows is approximately 8 and 16 hours respectively. When travelling through the SAA, astronauts are exposed to trapped radiation of the inner Van Allen belt, which shows a pronounced directionality. Although only about 5% of the mission time on board ISS is spent in the SAA, the astronauts may collect more than 50% of their total dose during this short time period (Apathy et al. 2007).

In case of ISS, which has usually an airplane-like attitude, i.e. the attitude is stabilized in the local vertical reference frame (ESA 2006), the anisotropies in the radiation field, such as the effect of the Earth's shadow, the angular dependence of the geomagnetic transmission factors and the East–West asymmetry in the SAA, might not be ignored in dosimetric measurements, unlike in case of Space Shuttle flights where, due to the changing orientation of the spacecraft, anisotropies usually tend to be averaged out (Badhwar et al. 1999). 1D telescopes used for dosimetry have strong directional sensitivity and therefore they might either under- or overestimate the dose equivalent. The application of 3D telescopes with three, mutually orthogonal axes improves significantly the measuring precision of the instrument.

A number of papers addressed the models of the anisotropic radiation environment on board ISS. The results of the calculations presented in (Wilson et al.

2006) have shown that the addition of the angular dependence of both the trapped protons and GCR transmission factors considerably increases the usefulness of the basic cosmic radiation models. Time resolved measurements performed with Liulin type silicon detector instruments on board ISS were used to evaluate the adequacy of the environmental revised models. The comparison has shown that further corrections are needed (Wilson et al. 2007). Nealy et al. presented validation of their revised environmental models in which they compared the results of the calculations with that of the measurements carried out with several passive and active detectors on board ISS. Albeit the agreement was found to be fairly good, they suggested further improvement of the models (Nealy et al. 2007).

Material and methods

The SAA switching algorithm

After a signal coming from a detector has been processed and digitized, the content of the appropriate channel in the active deposited energy spectrum is incremented. In order to collect the SAA and non-SAA spectra separately, it is necessary to switch automatically between them, i.e. to make the SAA spectra active when the ISS entered the SAA and the non-SAA spectra active when it leaves the anomaly. The algorithm for performing this procedure was developed based on the results of the measurements carried out with DOSTEL in different shielding configurations (Burmeister et al. 2000). The effectiveness of the method was tested with model time spectra calculated from the time spectra obtained by DOSTEL after taking into account the differences in the telescope geometry. Depending on the parameter of the algorithm and the expected number of counts, the switching between the SAA and non-SAA spectra might take place due to the statistics of the measurement. In order to determine the probability of this "false switching", statistical uncertainty was added to the calculated spectra by means of a pseudo-random number generator (Hirn 2009b).

Modelling the anisotropy of the GCR at low Earth orbit

In my calculations it is assumed that the 3D telescope is installed outside the wall and shielded by the space station from the nadir direction (Fig. 1). The x-axis of the telescope points along the orbital velocity vector, the z-axis toward the nadir direction, and the y-axis completes the right-handed coordinate system. Calculations are performed for ISS orientation with zero pitch, roll and yaw angles. Free-space GCR spectra for protons and alpha particles were obtained with the FLUX module of the CREME96 model (heavier ions were not included in the calculations).

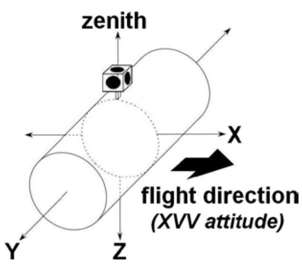

Fig. 1. Orientation and position of the telescopes.

The GTRN module of CREME96 can be used for evaluating geomagnetic shielding but only in a vertical cut-off approximation (Tylka et al. 1997). In order to determine the directional distribution of the geomagnetic cut-off rigidity, the expression developed by Stoermer for characterizing the interaction of a charged particle with a dipole magnetic field was used (Stoermer 1937)

$$R = \frac{C_D \cos^4 \lambda_M}{r_D^{\,2}\left(1+\sqrt{1-\cos^3 \lambda_M \sin\zeta \sin\psi}\right)^2} \qquad \text{Eq. 1.}$$

where R is the directional cut-off rigidity in GV, λ_M is the magnetic latitude, r_D is the distance from the magnetic dipole centre in Earth radii, ζ is the zenith angle, ψ is the azimuth angle measured clockwise from the magnetic north and C_D is a constant which is directly proportional to the dipole moment, and has a value of 59.6 GV for the 2000 IGRF (International Geomagnetic Reference Field) dipole. Geographical coordinates of the ISS as a function of the time were provided by the coordinate generator package included in ESA's Space Environment Information System (SPENVIS 2003; Heynderickx et al. 1996). The magnetic coordinates λ_M and r_D were obtained after coordinate transformations. At the altitude of the ISS the magnetic field of the Earth can be well approximated by the eccentric dipole model. The parameters of the coordinate transformations i.e. the colatitude (θ_N) and the azimuth (φ_N) of the point of intersection of the geomagnetic axis with the Earth's surface in the northern hemisphere in the centred dipole model as well as the translation in the x, y and z directions (η^*, ζ^* and ξ^*, respectively) are shown in Table 1.

Table 1. Parameters of the coordinate transformation.

Parameter	Value	Reference field model
θ_N	79.54°	IGRF 2000
φ_N	288.43°	IGRF 2000
η^*	–0.06308 R_{Earth}	IGRF 2000
ζ^*	0.04713 R_{Earth}	IGRF 2000
ξ^*	0.03149 R_{Earth}	IGRF 2000

Stoermer's expression does not account for the presence of the solid Earth. Since the details of how exactly the Earth's umbral shadow changes with altitude is still unknown (Smart et al. 2000, 2006), for the sake of simplicity, the Earth was considered as a solid sphere. At the altitude of the ISS the solid angle occulted by the Earth is 1.4π. The cut-off distributions as a function of the particle's energy and the angle of incidence were calculated in the reference frame fixed to the given telescope axis. The angle of incidence was discretized with a discretization step of 5°. With knowledge of the free-space GCR spectrum and the cut-off distribution, the LEO GCR spectrum was determined. To estimate the deposited energy spectra the FLUKA Monte Carlo transport code developed by the Italian National Institute for Nuclear Physics (INFN) and the European Organisation for Nuclear Research (CERN) was used (Ferrari et al. 2005; Battistoni et al. 2007). The geometry was simplified to two 300-micrometer-thick silicon detectors at a distance of 8.9 mm with an active area of 222 mm^2 and an aluminium slab representing an effective shielding. Concerning the aluminium slab two different thicknesses were chosen:

– 0.5 mm of aluminium in front of the detectors (mechanical constraints);
– 74 mm (20 g/cm^2) of Al in the direction of the ISS.

Particles were generated from behind the slab. The deposited energy spectra were calculated with the DETECT card in coincidence mode. The minimum total energy to be scored in one event in the measuring detector region, as well as the coincidence threshold was 70 keV. Below this value the contribution of charged particles to the coincidence spectra is negligible (Hirn et al. 2007).

Modelling the anisotropy in the SAA

In the Earth's magnetic field charged particles are moving in spiral paths around the geomagnetic field lines bouncing back and forth between the mirror points. The SAA is also close to a mirror point where the pitch angle with respect to the magnetic field vector approaches and recedes from 90°. The proton flux is therefore maximized in the plane normal to the local magnetic field. Protons arriving from the west have trajectories with gyration about a point located above the reference observational point and hence they encounter less residual atmosphere than protons arriving from the east. This results in an asymmetry, where at a given point the flux of protons coming from the west is higher than the flux of protons coming from the east (Badavi et al. 2006). Although, there exist models (Watts et al. 1989; Badhwar, Konradi 1989) that are capable of predicting the directional flux of the charged particles, most of the cosmic radiation models that have been used for describing the cosmic radiation field are omnidirectional, including the "de-facto" standard AP8 and AE8 trapped proton and electron models.

Directionality of the proton flux then can be expressed as follows

$$\frac{J(\theta,\psi)}{J_{4\pi}} = F_N \exp\left[\frac{-(\pi/2-\theta)^2}{2\sigma_\theta^2}\right] \exp\left[\frac{r_g \cos I \cos\psi}{d_s}\right] \qquad \text{Eq. 2.}$$

where J is the vector flux, $J_{4\pi}$ is the omnidirectional proton flux provided by the AP-8 trapped proton models implemented in SPENVIS, θ and ψ are the pitch and azimuth angles respectively, d_s is the ionospheric scale height, I is the magnetic dip angle, r_g is the proton gyro-radius in km, σ_θ is the standard deviation of the pitch angle and F_N is a normalization factor (Kern 1994). To get a rough estimation of the directionality of trapped particles, calculations were performed with the positional version of the 'radiation sources and effects' package of SPENVIS. In the calculations, one single SAA transit was considered. The orbit was tailored to pass through near the centre of the SAA (centred near 29.4° S, 45.6° W in 2010).

Results

Performance of the SAA switching algorithm on model time spectra

Since the number of counts measured with TriTel depends on several factors (e.g. altitude, solar activity, shielding configuration), it might considerably change during a mission or from one mission to another. Therefore the switching algorithm developed is based not only on the absolute value of the number of counts but on its relative change as well. The switching between the SAA and the non-SAA spectra takes place in real time. Passage through the anomaly is indicated by an SAA flag, its value outside SAA is 0 (the measured counts contribute to the non-SAA spectra). The number of counts in consecutive 60-second-long time intervals is registered in the time spectra. In case the relative change in the sum of the time channels of the x-, y- and z-axis exceeds a predefined δ value, the SAA flag changes to 1 and the number of counts starts to be registered in the SAA spectra. TriTel switches back to the non-SAA spectra (value of the SAA flag is 0) only if the number of counts decreases below the value registered at the time of the previous switch (Hirn 2009b).

From the calculations the value of δ at which the probability of false switching is less than 10^{-3} was found to be 0.34 and 0.25 for the spectra generated from the measurements with DOSTEL during the STS-84 mission and in the Matroshka-1 experiment, respectively. The difference in the two values can be attributed to the fact that during the STS-84 mission DOSTEL was inside the Space Shuttle, while in the Matroshka-1 experiment it was located outside the wall of the ISS. Therefore, in the former case, due to the smaller number of counts and so the higher standard deviation, the relative difference between the numbers of counts in two neighbouring time channels might be more significant. Since the number of SAA peaks in the model time spectra has shown weak statistics, no detailed statistical analysis has been performed on the maximum value of δ (δ has a nearly uniform distribution between 0.3 and 0.8).

In the view of the above I proposed $\delta = 0.35$. Fig. 2 shows the performance of the algorithm applied for the model time spectra generated from the STS-84 measurements of DOSTEL for this case.

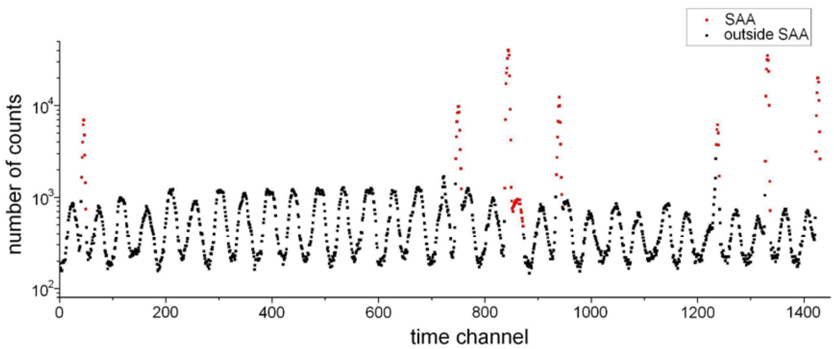

Fig. 2. A sample from the results of the SAA-switching algorithm applied to the model time spectra.

Effects of the anisotropy of untrapped GCR particles on the model calculations

The cut-off distributions for 0°, 20°, 40°, and 60° angles of incidence are shown in Fig. 3 in the case of a telescope axis pointing toward the geographical zenith direction. Below 10 GV, the four curves are practically identical. The contribution to this low cut-off region comes from segments of the orbit when the ISS is passing at higher latitudes where the change in the geomagnetic cut-off due to the change in the zenith and azimuth angles is not so significant. However, the differences in the curves above 10 GV originate from the zenith dependence of the cut-off at latitudes that are close to the geomagnetic equator.

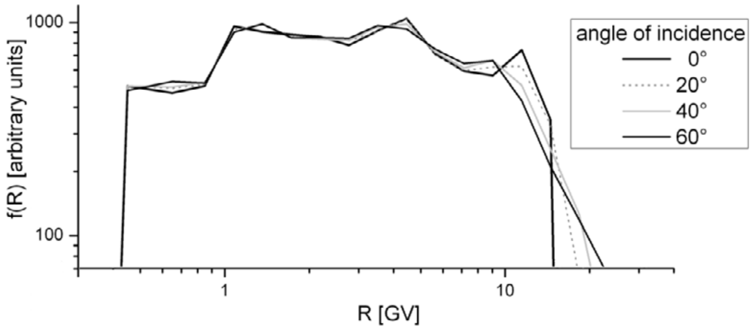

Fig. 3. Cut-off distribution for different angles of incidence for a telescope pointing towards the geographical zenith direction.

As for the total deposited energy spectra, the spectra in the x- and y-directions were found to be the same due to symmetry reasons and, in spite of the planar geometry of the detectors, no difference could be seen in the zenith direction either. Because of the planar geometry of the silicon detectors, the absorbed dose measured can be significantly higher for non-relativistic particles incident perpendicularly to the axis of the detector than for particles with normal incidence to the effective surface of the detector. Since the nadir, from which direction the radiation is considerably shielded by the Earth, is perpendicular to the axes of the x- and y-telescopes, the contribution of these higher energy deposits is lower in the total deposited energy spectra in the x- and y-directions. However, the lower cut-off values in the zenith direction have the opposite effect. In order to interpret the results in more details and determine the significance of the effects described above, further calculations are needed. The effects of the anisotropy of the galactic cosmic rays in LEO arise in the coincidence spectra and so in the calculated LET spectra, as well (Fig. 4).

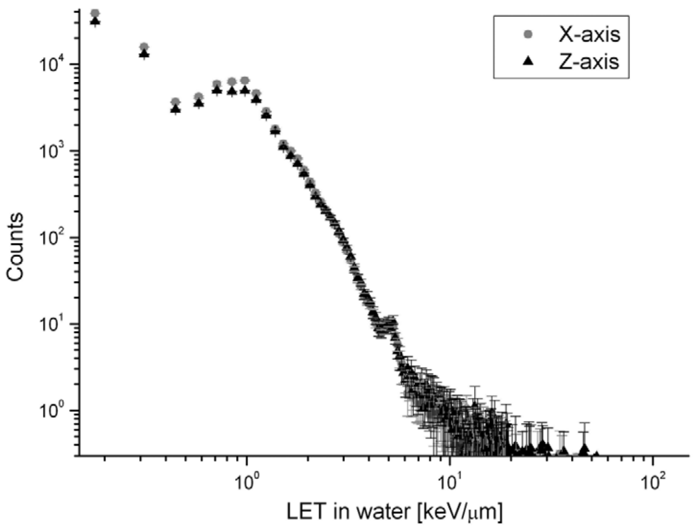

Fig. 4. LET spectra of protons and alpha particles determined from the coincidence spectra in the X and Z directions (the Y coincidence spectrum was found to be identical with the X spectrum; therefore it is not indicated in figure).

Contrary to the total deposited energy spectra, the number of counts in the LET spectra in the x- and y-directions is significantly higher than in the z-direction. The fact, that in case of the z-axis, 50% of the field of view is occulted by the Earth, while this ratio is only 30% in case of the x- and y-axis, might result in this effect (because the altitude of the spacecraft is 340 km, a considerable part of the particles coming from the –z directions can give a contribution to the coincidence spectra of the x- and y-axis).

Since only protons and alpha particles were considered in the calculations and the uncertainties in the number of counts related to the energy deposit of particles with a LET value higher than 10 keV/μm in water was high, no significant difference could be observed in the calculated quality factors. In order to study the differences in the quality factors, ions heavier than alpha-particles shall be included in the calculations as well (Hirn 2010).

Effects of the anisotropy of trapped particles in the SAA on the model calculations

The integral fluxes of trapped protons that can enter the sensitive volume of the detector are plotted in Fig. 5 as a function of the magnetic pitch angle. The fluxes were determined in seven distinct positions in the section of the orbit of the ISS crossing the SAA. Inside the SAA, protons arrive from directions within 10–15° of the plane perpendicular to the geomagnetic field line. Even if we take into account that the geographical and geomagnetic axis do not coincide, the velocity vectors of the trapped protons lie outside the field of view of the z-telescope. Therefore, these particles will contribute only to the coincidence spectra in the x- and y-directions but not the z-direction (Hirn 2010). A more quantitative description of the directionality of the trapped radiation can be performed by analyzing Eq. 2.

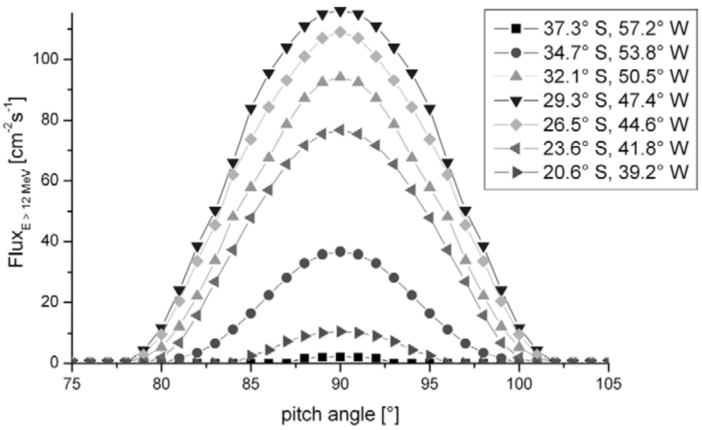

Fig. 5. The integral flux of trapped protons as a function of the pitch angle inside the SAA.

Discussion

The optimal value of the δ parameter of the SAA switching algorithm for a given experiment can be determined only after the first measurement data has been evaluated. However, the fine-tuning of the parameter is necessary. In the knowledge of the time spectra the effects of the time-shifts and the false switching might be corrected during the on-ground evaluation of spectra.

The calculations presented in the paper show that the effects of the directionality of the geomagnetic cut-off are negligible compared to that of the Earth's shadow when integrated over the orbit. In order to demonstrate the effects of the anisotropy of the untrapped radiation in the measured dose equivalents and quality factors, it will be necessary to take into consideration heavier ions as well. The trapped protons inside the SAA evidence an even more pronounced anisotropy. The velocity vectors of the trapped protons lie outside the field of view of a telescope pointing toward the zenith direction. However, with a 3D telescope, the LET spectrum of the trapped radiation can be studied as well. The author wishes to acknowledge the services provided by SPENVIS and CREME96 hereby.

Conclusions

In this paper only the anisotropy of the radiation field outside the ISS was addressed. Inside the space station, due to the complexity of the shielding distribution, simulating the deposited energy spectra is even more complicated; hence, 3D measurements on board the space station should be performed. In the light of the above, three different versions of the TriTel 3D telescope are being developed at AEKI. One version, in the framework of the European SURE project, will perform measurements on-board the Columbus module of the ISS, one in cooperation with the Institute of Biomedical Problems on-board the Russian segment of the ISS and finally another version will measure the LET spectra and the dose on board the ESEO satellite in the near future.

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Sequential separation of alpha and beta emitters from natural samples by mixed solvent anion exchange and their subsequent determination

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Abstract

Determination of alpha and beta radionuclides in environmental samples, food and drinking water is important from the aspect of human health and environmental protection. Low level activities determination of pure alpha and beta emitters in natural samples, prior to detection and quantitative determination, requires isolation of radionuclide from the sample and simultaneous separation from interfering elements. The extraction chromatography is mostly used method in last decade. However, due to chemical and mechanical stability and low price, ion exchangers are a good alternative to expensive extraction chromatographic resins because they don't lose their capacity with multiple usage, whereas resins do. It is very important, for the ion exchanger to be applicable in the separation, that its capacity is not a limiting factor (ion exchange must not be a dominant mechanism of bounding). This work will show that anion exchangers in nitric form, in combination with alcohol solution of nitric acid, can be used for separation of all kinds of cations. The effect of dielectric properties of solutions on the bounding strength of particular cations to anion exchangers in nitric form, its effect on exchanger selectivity conversion for Ca, Sr, Y, Pb, Th, U, Am and Pu from HNO_3 solutions in methanol, ethanol and acetone, as well as the separation possibility of various cations, based on changes in solution properties, will be presented. The effect of change in dielectric constant on the bounding strength of certain cations to the exchanger will be shown in temperature range from -50 to 20°C, and how it can be used in cation separation. For quantitative determination of certain isotopes, already existing methods have been modified (LSC, Cherenkov counting, alpha and gamma spectrometry, ICP MS). Finally, simple and rapid methods, which are used in daily Laboratory work, have been developed for isolation and quantitative determination of $^{89,90}Sr$, ^{210}Pb, and other isotopes.

Optically stimulated luminescence in salt tested against thermally stimulated luminescence in LiF and ambient survey measurements in a ^{137}Cs contaminated village in Belarus

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Abstract

Ordinary household salt (NaCl) have previously been investigated in the laboratory and has exhibited several promising properties for dosimetry, e.g. a linear dose response down to a few mGy and a low detection limit. In an attempt to test NaCl as a dosemeter outside the laboratory, we here report the first results from the use of NaCl as a dosemeter under normal environmental conditions, both indoor and outdoor. For this purpose, special dosemeter kits, each containing two chips of LiF and 2×10 mg of NaCl were designed. During the summers of 2008 and 2009 the dosemeter kits were positioned pair wise inside and outside the outer walls of 17 houses located in a village which was highly contaminated by ^{137}Cs from Chernobyl, in the Gomel region (Belarus). After 2.5 – 3 months the dosemeters were collected and read out using thermoluminescence (TL) and optically stimulated luminescence (OSL), respectively. The estimated dose rates from the two dosemeter systems were then compared for each kit and in addition, the dose rate at positioning and collection of the dosemeters was measured using a hand-held ambient dose survey meter (GR-110) based on a 5.0 x 3.8 x 3.8 cm^3 NaI(Tl)-detector.

The radiation level in the village was found to be highly inhomogeneous, even within the gardens around the houses. On average, the radiation level around the houses was 3 – 8 times higher compared to a normal background dose rate (0.10 µSv h^{-1}). Despite the relatively low signals, a strong correlation was observed between the results of the salt dosemeters and the portable NaI(Tl)-detector (r^2 = 0.89), but the correlation between NaCl and LiF was weaker (r^2 = 0.64). The repeated study in 2009 confirmed the potential for ordinary cheap household salt as a tool for retrospective dose estimates.

Introduction

In order to estimate radiation absorbed doses to the general public after an event that involves unwanted exposure of ionising radiation, several retrospective methods have been suggested, see e.g. (ICRU, 2002; Alexander et al. 2007). The Medical Radiation Physics group in Malmö has especially focused on optically stimulated luminescence (OSL) of household salt (NaCl) as a tool for retrospective dosimetry. The dosimetric properties of different brands of household salt have been investigated in the laboratory (Bernhardsson et al. 2009; Christiansson et al. 2008a; Christiansson et al. 2008b) and several promising properties for dosimetry have been found. The next step is now to test salt as a dosemeter *in situ*, under normal environmental conditions and during extended periods of time. This has been done in an environment that has been affected by a major surface contamination of anthropogenic radionuclides, mimicking a situation in which the dosemeters are aimed to be used in the future, after e.g. radiological and nuclear accidents as well as after antagonistic use of radiation and radioactive substances.

The Vetka district in the Gomel region (Belarus) was found to be an area that fulfils these conditions where one of the villages, Svetilovichi, was one of the most contaminated settlements outside the 30 km zone after the Chernobyl catastrophe. Parts of this district were decontaminated in 1986 – 1987, but not the village of Svetilovichi. The people living in this village were advised to relocate to areas with less contamination, but most of the inhabitants decided to stay instead of abandon their homes and village. However, in 1991 a more systematic decontamination was carried out in the village (roads and outside public buildings, i.e. kindergartens and the hospital). Thereafter, in 1999, and as part of the IAEA regional project RER/9/059 (Andersson K.G. et al. 2001), Svetilovichi was selected as a village for practicing and improving the existing techniques of decontamination. New equipment was bought and experts on decontamination were training the personnel at the federal state company Polesye (Gomel) who is specialized in cleaning contaminated buildings. Up till 2006, about 50 houses and gardens have been decontaminated according to the program and today, all of the most heavily contaminated houses have been decontaminated.

During the summers of 2008 and 2009, measurements of the gamma radiation level in the village of Svetilovichi were carried out by the Medical Physics group in Malmö (Lund University, Sweden) in cooperation with the Chernobyl Committee (Minsk, Belarus) and the federal state company Polesye (Gomel, Belarus). One of the aims of the project was to investigate household salt as an optically stimulated dosimeter (OSLD) when used in a contaminated environment.

Material and methods

The village of Svetilovichi is situated in eastern Belarus (Latitude: 52°47'43.76"N; Longitude: 31°19"11.88"E), between Gomel (Gomel region, Belarus) and Novozybkov (Bryansk region, Russia). Due to heavy rainout of the passing plume of the Chernobyl release, the Gomel region received an average ^{137}Cs surface deposition of 154 kBq m^{-2} [Drozdovitch V. et al. 2007]. Since 1990s, individual effective dose estimates of the internal and external exposure have been carried out in a number of the nearby villages, on the Russian side of the border, see e.g. (Thornberg C. et al. 2001; Thornberg C. et al. 2005; Bernhardsson C. et al. 2008).

Assemblying and positioning the dosemeters

To investigate the radiation level in some of the decontaminated and non-decontaminated houses in Svetilovichi, between 5 and 14 dosemeter kits were placed inside- and outside each of the houses described in Table 1. Each dosemeter kit contained two chips of LiF (TLD100, Harshaw) and two sections (about 10 mg each) of NaCl (Falksalt fint bergsalt, Hansson and Möhring, Halmstad, Sweden), a regular seasoning salt in Sweden. The TL-chips and the salt were placed between two 4 mm thick sheets of PMMA, thus forming a dosemeter kit with two different luminescent materials. Before the dosemeter kits were put together, during the night before the distribution, the salt was exposed to sunlight (bleached) in order to empty the stored dose information. To avoid further bleaching of the luminescence signal, all dosemeter kits were covered with light-tight black tape. Another precaution was to place the dosemeter kits in plastic bags filled with silica gel, to protect the dosimetric material from rain and moist during the varying outdoor weather conditions.

Table 1. Description of the houses and the extent of countermeasures carried out at each dwelling. Fifty-six dosemeters were used at houses 1-7 in 2008 and 69 dosemeters at houses 8-17 in 2009. The thickness of the outer walls on the wooden houses was about 0.2 – 0.3 m and 0.5 m on the brick houses.

House	Walls/Roof	Decontaminated[a]
1.	Wood/Aluminum sheet	Yes
2.	Wood/Eternit	No
3.	Wood/Eternit	Yes
4.	Bricks/Aluminum sheet	Partly[b]
5.	Wood/Eternit	Yes
6.	Bricks/Eternit	No
7.	Bricks/Eternit	No
8.	Wood/Eternit	No
9.	Wood/Eternit	No
10.	Wood/Eternit	No
11.	Wood/Eternit	Yes
12.	Wood/Eternit	Yes
13.	Wood/Eternit	No
14.	Wood/Eternit	No
15.	Wood/Eternit	Yes
16.	Wood/Aluminum sheet	No
17.	Wood/Eternit	No

[a] Decontamination of a building includes; cleaning of roofs and walls using high-pressure water as well as removal of the top soil layer (10 cm depth) from the house wall and 2 m out and replacing it with sand. Highly contaminated roofs were removed and replaced with new ones.
[b] Walls on houses made from bricks were not cleaned.

The investigated houses were all documented by Polesye and had a known history from the owners. In this way, only houses built after 1986 and where no major changes of the constructions (after 1986) had been carried out, were selected for the measurements. The dosemeter kits were fixed firmly on both the inside- and outside of the walls of the houses, in opposite positions on the wall where it was possible and about 1 m above the floor level. Rooms that were most frequently used e.g. living room, bedroom and kitchen were prioritised for the indoor measurements. The ambient dose rate at the position of the dosemeters was registered with a handheld 3.8×3.8×5 cm^3 NaI(Tl)-detector (GR-110, Exploranium, Canada). These measurements were repeated when the dosemeters were collected approximately 2.5 months later and a mean value of the two measurements was used as a reference value to the dosemeter readings. To compare the absorbed doses to the dosemeter kits with the readings of the GR-110 instrument, a general calibration coefficient (free in air using a ^{137}Cs source; SMM, Department of Emergency Preparedness and Environmental Monitoring, certificate No:06-08S01) of 1.15 nSv/h cps^{-1} has been used.

Read-out procedure

The dosemeters were distributed, at 56 positions in 2008 and 69 positions in 2009, during the last week of May and were re-collected in the middle of August. They were immediately transported back to Sweden and shortly after that, the LiF chips were read-out in a TL–reader (Toledo, Vinten Instruments Ltd., England) at Sahlgrenska University Hospital in Göteborg. To assess the signal in the salt, it was optically stimulated by blue light (λ = 470±30 nm) at the Medical Radiation Physics department in Malmö, using a TL/OSL-DA-15 reader (Risø National Laboratory, Roskilde, Denmark). The net OSL signal was defined as the registered luminescence during the first 2 s subtracted by the average luminescence after 10-12 s, 20-22 s and 31-33 s of stimulation. In order to convert the net OSL signal to an absorbed dose a pre determined calibration coefficient ($c_{specific}$ = 313 counts mg^{-1} mGy^{-1}) for this particular brand of salt was used (Bernhardsson et al. 2009). It should be noted that $c_{specific}$ is determined for a specific brand of salt, from a single jar, and then generalised to hold for all salt jars of the same brand, thereby assuming that the salt in all jars are identical. To account for changes between individual dosemeters and salts taken from different jars, there are other means of calibration that can be used. One approach is to use the single-aliquot-regenerative-dose (SAR) protocol (Murray et al. 1998) where successively increasing doses, in the range of the dose to be determined, are given to the salt after the read out. This approach was however impossible to carry out at the time of the readouts in 2008 and 2009 and hence the protocol and calibration coefficients described by Bernhardsson et al. (2009) have been used.

A correction for the additional dose originating from the exposure during transportation from Svetilovichi and storage before read-out has been carried out. To estimate the magnitude of this dose, ambient survey meters were used at various positions along the travel route from Belarus and the final destinations (the laboratories in Malmö and Göteborg, Sweden, respectively). From these measurements an average background dose rate was calculated and subtracted from the total dose accumulated during the probing time.

Results and discussion

The average dose rate during transportation and storage was determined to 0.10 µSv h^{-1}. The total dose accumulated during transit and storage varied individually and has hence been subtracted from the total dose registered by each dosemeter. Unfortunately, the dose to the TLDs used in 2009 was not possible to retrieve due to problems with the TLD-reader and therefore, the OSLD to TLD comparison was not achievable that year.

In Fig. 1 is the absorbed dose as determined by NaCl and LiF plotted as a function of the corresponding dose obtained from the handheld NaI(Tl)-detector at the position of the luminescent dosemeters. In spite of the low detector signals there is a fairly good correlation between the dosemeters and the radiation protection instrument. However, the OSLDs exhibit on average, a 15% higher absorbed dose than the TLDs. This might indicate a too small signal background subtraction of the NaCl batches, or a too high signal background subtraction of the LiF-chips. It could also be a consequence of sensitivity changes in the LiF-chips during the measuring period. Another explanation could be that anomalous fading of the signal is more rapid in the specific LiF-chips used, compared to NaCl, at least during the first 2 – 3 months.

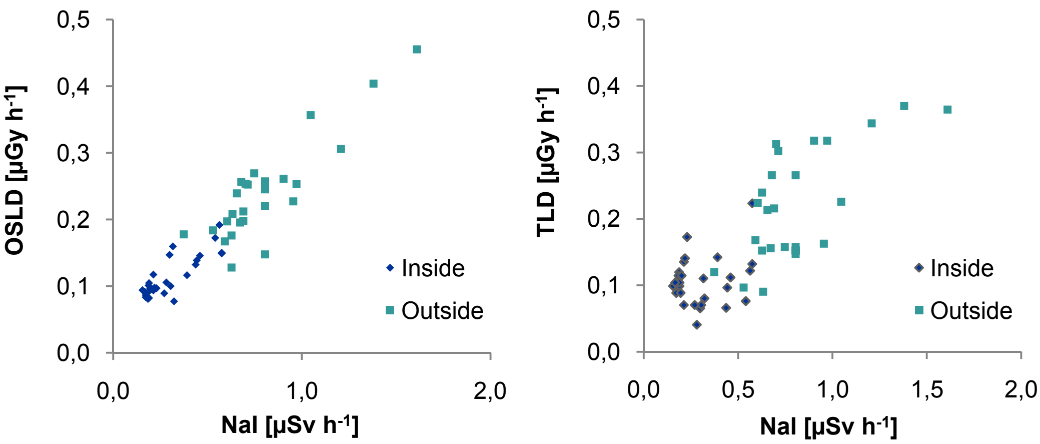

Figure 1. Absorbed dose in OSLD and TLD as a function of the dose registered by GR-110 (average of measurement at outset and pickup of the kits). Included in the figures are measurements from 56 dosemeter kits (positions) in 7 houses, during the summer of 2008. Different colours have been used to distinguish between dosemeters positioned inside and outside the buildings.

Generally, the dose rates indoor were rather low, comparable to the normal background radiation level in many European countries. Outside the houses the average dose rate was 3 – 8 times higher compared to the indoor values. The pair-wise correlation between the results of the 3 measurements in terms of Pearsons's r^2 was higher for the outdoor measurements compared to those inside. Nevertheless, there was a moderately or a strong correlation between all of the investigated dosemeters (Table 2). The correlation was strongest between the OSLDs and GR-110, which may partly reflect the similar responses between these two detectors for the photon energies studied. It is however apparent that some of the specific LiF chips used in this study exhibited a rather low sensitivity and this in turn may have adversity influenced the detection limit (0.01 mGy, as specified by the manufacturer).

Table 2. Covariance matrix (r^2-values), for the three dosemeters tested in the survey in 2008.

	OSLD	TLD	GR-110
OSLD	1	0.64	0.89
TLD	-	1	0.65
GR-110	-	-	1

The dose rates measured with the OSLDs, plotted as a function of the correspondingly dose rates of GR-110, in the years 2008 and 2009, is shown in Fig. 2. There is a noticeable difference in the linear relationship between the two years, with:

$$OSLD_{2008} = 0.24 \cdot NaI + 0.04 \qquad \text{Eq. 1}$$

$$OSLD_{2009} = 0.40 \cdot NaI + 0.10 \qquad \text{Eq. 2}$$

Even though it was not the same houses that were investigated, the dose response in the salt should not be affected by this factor. The ranges in the registered doses are similar in both years but the slope and intercept of the lines are significantly different (Eq. 1 and Eq. 2).

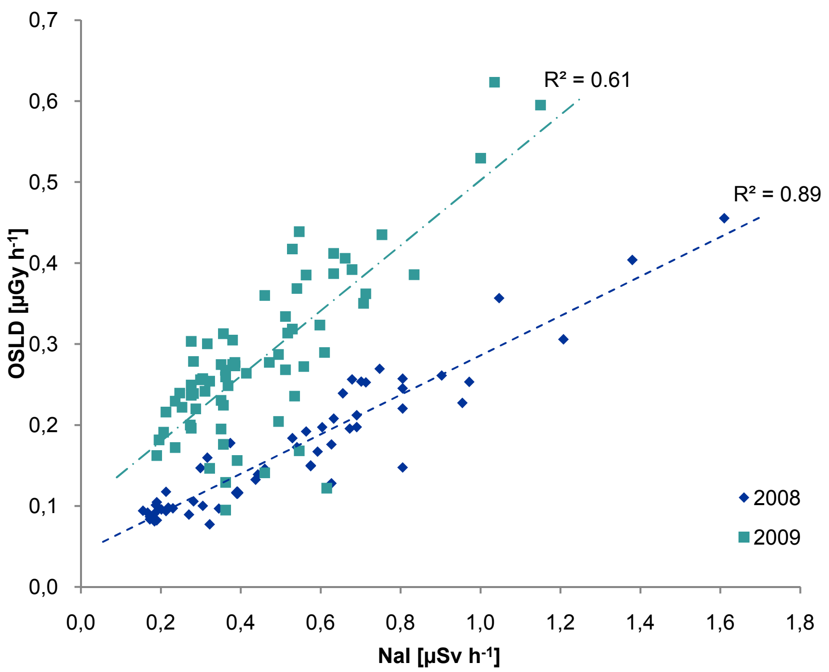

Figure 2. Absorbed dose rate as measure by the salt dosemeters in 2008 and 2009 as a function of the dose rate as measured with GR-110 at the position of the dosemeters.

The sensitivity is about twice as high in the salt used in 2009 compared to 2008, despite the fact that they are from the same brand of salt. The reason for this diversity is that the particular salts used were taken from different jars. The jar of salt used the first year was packed in December 2006 and the salt used in 2009 was packed in March

2009. Hence, the salts were probably extracted from different salt layers within the same, or different, mines and possibly mixed with salt from other deposits before it was packed and distributed. Changes in these fractions will accordingly change the properties of the salt as a dosemeter.

Another observation is that the salt used in 2009 appears to have a slightly weaker correlation to the GR-110 reading, compared to the salt used in 2008. One factor that influences this is the distribution of grain sizes within a particular package of salt. Large grains will generally give less luminescence per absorbed dose compared to small grains, at a given weight. Since the salt was not sieved (fractionated due to grain size), this might be one of the major factors for the discrepancy found in the OSLD measurements in 2008 and 2009 caused by using two different jars of salt.

Conclusions

Even though the study was carried out with pre-manufactured dosemeter kits, the salt demonstrates a promising potential as a tool for retrospective dose reconstructions. The salt in the dosemeter kits exhibits a good or a strong correlation to measurements carried out using a radiation protection instrument and TLDs. The repeated survey increases the confidence in salt as an OSLD, despite the fact that the measuring conditions were slightly different during the two different years.

Small changes to the experimental set-up by changing a few factors can further increase the usefulness of salt as a dosemeter. One of these is to sieve the salt before the read-out, thereby making the salt more homogenous and thus, minimising the risk of analysing inefficiently used grains. Another one is to use individual calibrations of each dosemeter, e.g. by the use of the SAR-protocol. One benefit with this approach is that changes in the sensitivity between two different jars of salt, from the same brand, will be incorporated in the calibration. Another benefit is that almost every salt on the worlds market becomes available as a potential dosemeter.

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Visualization of hot particles in lung of radionuclide associated with emergency preparedness by means of clinical gamma cameras

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Abstract

In a situation when radionuclides accidentally or deliberately are dispersed in the environment and individuals have been internally contaminated there is an urgent need for localization of the radionuclide in affected individuals. The aim of this study is to examine the possibilities to visualize hot particles of radionuclides with gamma energies close to and beyond the upper level of the pulse height analysers (PHA) of clinical gamma camera systems. The aim is also to examine the possibilities to visualize hot particles of pure β-emitters with a clinical gamma camera. An anthropomorphic phantom was used to mimic uptakes in the lung region of three different radionuclides associated with emergency preparedness ^{60}Co ^{137}Cs and ^{90}Sr/^{90}Y (0.8, 0.16 and 2.2 MBq respectively). The radionuclides were located in or outside the right lobe of the lung insert. The conclusion of this study is that point sources, mimicking a hot particle of a given radionuclide can clearly be visualized even in cases when the primary gamma photon energy exceeds the upper limit of the PHA. This study indicates also that point sources of pure β-emitters can be visualized. Hence gamma camera systems can be useful for rapid assessment of inhaled hot particles in connection with radiological and nuclear accident.

Introduction

There are a number of radionuclides of special concern within the emergency preparedness; for instance ^{60}Co, ^{137}Cs and ^{90}Sr/^{90}Y, which are frequently used in hospitals, in the industry and at research laboratories. In addition to these sources, nuclear power plants generate airborne aerosols and surface contamination of ^{60}Co as a result of neutron activation of stable cobalt in construction materials, which can accidently be ingested or inhaled by workers (D. W. Whillans, W. J. Chase and W. H. Wolodarsky 2007). Another source of radionuclides in nuclear power plants is fission products of ^{235}U such as ^{137}Cs and ^{90}Sr/^{90}Y. These radionuclides are generally not dispersed to the surroundings except in cases of severe nuclear reactor accidents, such as the Chernobyl accident in 1986. Other potential exposure pathways to humans are transportation accidents or releases in connection with terrorist activities where radionuclides are involved (ICRP 96, 2003). In previous mention scenarios there is a need for methods to rapidly screen suspected intakes of e.g. ^{60}Co and ^{137}Cs in individuals by means of in-vivo measurements. Potential resources that are useful in case of nuclear and radiological emergencies must be explored. Equipment and personnel connected to radiological work at general hospitals are one such resource. Clinical gamma camera imaging systems, which are available in many hospitals, would be a valuable addition to the capacity of emergency measurements.

Wallström et al., (1998) examined the possibility to use a gamma camera for visualizing of ^{137}Cs in man by using bottle phantoms. The major part of the work was performed with a gamma camera consisting of a NaI(Tl)-detector with a crystal thickness of 1.2 cm. The limitations with many gamma camera systems are that some radionuclides associated with emergency preparedness scenarios emit gamma photons with energies beyond the upper level of the energy window of the PHA. Therefore, imaging of organ uptakes of the high energy photon emitters must be deduced by detection of scattered radiation. A previous study using a Siemens MultiSPECT 2 with various acquisition settings for imaging of ^{46}Sc and ^{60}Co in the lungs and abdomen were carried out at our department (Hansson et al., 2008).

The aim of this work is to evaluate the possibility to use clinical gamma camera systems for visualization of the presence of radionuclides associated with emergency preparedness in the lungs, also in cases where the primary photon energies exceed the normal operative range of the PHA. In addition to gamma emitters associated with emergency preparedness (e.g. ^{60}Co and ^{137}Cs) we aim to evaluate the possibility to visualise presence of pure beta-emitters (^{90}Sr/^{90}Y) by the detection of bremsstrahlung.

Material and methods

Gamma camera systems and imaging of high energy gamma quanta

Two gamma camera imaging systems have been used in this study, the principal measurements being carried out on a Siemens MultiSPECT 2 (Siemens, Erlangen, Germany), The MultiSPECT 2 is a dual-head imaging system with NaI(Tl)-detectors having a field-of-view of 53.3x38.7 cm^2 and a crystal thickness of 0.95 cm. Complementary pilot measurements have been carried out on a Symbia SPECT-CT (Siemens, Erlangen, Germany) which is a SPECT-system equipped with an low dose CT. The point source of the radionuclide was placed on the right lung lobe close to the heart compartment or inside the same lung compartment of the phantom (Fig 1). The first alternative was used when the physical size of the holder of the point source did not allow it to pass in to the inside of the lung compartment. When exploring the MultiSPECT 2 capability to visualise lung uptakes of point sources, the entire energy window, 45-535 keV, of the PHA was used. High energy collimator was used in connection with the MultiSPECT 2 measurements. The matrix size of the SPECT image was 128x128 and filtered back projection reconstruction algorithm was used. For the Symbia SPECT-CT an energy window of 100-300 keV was used, together with a medium energy collimator. The matrix size of the image was 64x64 and here an iterative reconstruction algorithm was used.

Phantom characteristics

An anthropomorphic phantom (Alderson-phantom, RSD 2008) specifically designed for calibration of imaging systems within nuclear medicine has been used in this study. The phantom is composed of different materials with attenuating properties comparable to human tissues. Phantom inserts shaped as human internal organs, such as lungs, liver and heart, can be filled with various radioactive solutions of known activity to simulate different organ uptakes. By supplying additional human tissue equivalent material on the anterior of the phantom torso, three different body sizes can be simulated. The body sizes are; *i.*) average male 74 kg, (personal communication Helen Kelly, RDS) *ii.*) large male more than 74 kg and *iii.*) large female more than 74 kg. Only the "average male" phantom was used in this study.


Fig 1. Left frame: The Alderson phantom positioned on the patient couch of the MultiSPECT 2 gamma camera. Right frame: The lung insert of the Alderson phantom with a point source on the outside.

Radionuclides

Three key radionuclides associated with the emergency preparedness scenarios as mentioned before were used in this study. The photon energies emitted by these radionuclides was 661.6 keV (^{137}Cs) and 1132 keV and 1332.5 keV, respectively (^{60}Co). The possibility to visualize uptakes of pure β-emitters in the lung has also been investigated using a commonly occurring β-emitters ^{90}Sr/^{90}Y ($E_{\beta,Average}$=0.196 MeV / $E_{\beta,Average}$=0.934 MeV). For the ^{90}Sr/^{90}Y the radiation detected by the gamma camera consists of the bremsstrahlung originating from the interaction between the high-energy β-particles within the tissue material. In Table 2 the investigated radionuclides and some of their properties are summarised.

Table 2. Strength of the point sources used simulating hot particles. ^{90}Sr was in equilibrium with the daughter ^{90}Y that emits the *β*-p.

Radionuclide	$T_{1/2}$ [year]	Main energy [keV]	γ-yield [%]	Activity [kBq]
^{60}Co	5.3	1173/1332	100/100	800
^{137}Cs	30	662	85	160
^{90}Sr /^{90}Y	29/0,0073	196/934 (average β-energy)	100/100	2200

Results and discussion

Visualisation of lung uptakes for different radionuclides

Fig 1-3 show images (matrix 256*256) obtained with the MultiSPECT 2 visualise ^{60}Co, ^{137}Cs and $^{90}Sr/^{90}Y$ in the Alderson phantom. An increased intensity at the position of the sources is clearly seen in all images compared to the background. The obvious disadvantage with planar gamma camera imaging in this case is the lack of correlation of the point source and the anatomy in the image. It could be a help to use so called landmarks in form of ^{57}Co sources of low activity placed on well defined points on the thorax of the internal contaminated person in the measuring situation. In fig 1 the image representing a hot particle in the lung containing ^{60}Co appears as a cloud with no distinct delimitation. Such an image may be of limited help in an emergency situation, but with another levelling of the SPECT image the area of the cloud can be reduced and become somewhat more guiding in terms of localisation of the particle. The general problem is that the high gamma energy (1173 keV/1332 keV) emitted from ^{60}Co is more than twice that of the ULD at 535 keV of the MultiSPECT 2.

Fig 4-6 show images (matrix 64*64) obtained with the Symbia SPECT-CT visualising hot particles of ^{60}Co, ^{137}Cs and $^{90}Sr/^{90}Y$, respectively, in the Alderson phantom. The obvious advantage with fused SPECT-CT imaging compared to planar gamma camera image is the possibility to correlate the point source to a specific point in the anatomy. The correlation between the point source in the anatomy is not only given in one dimension but in three dimensions. N.B. that the image layers of the SPECT and CT are morphed at the coronal plane of the phantom.

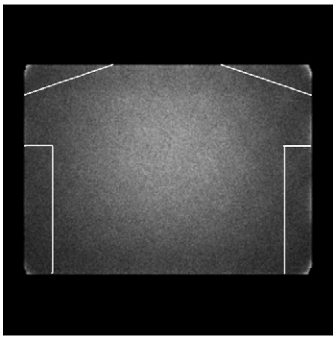

Fig 1. ^{60}Co, 800 kBq, point source positioned on the right lung lobe.

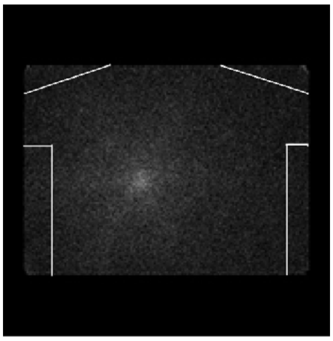

Fig 2. ^{137}Cs, 160 kBq, point source positioned on the right lung lobe.

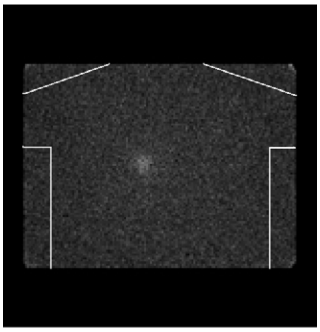

Fig 3. $^{90}Sr/^{90}Y$, 2300 kBq, point source positioned in the right lung lobe.

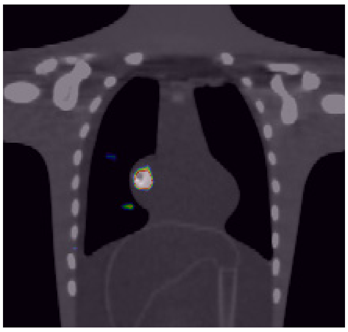

Fig 4. ^{60}Co, 800 kBq, point source positioned on the right lung lobe.

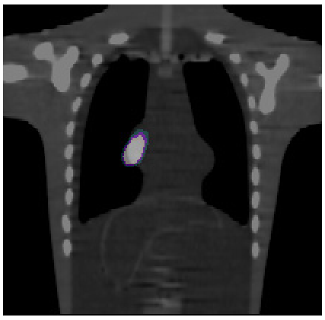

Fig 5. ^{137}Cs, 160 kBq, point source positioned on the right lung.

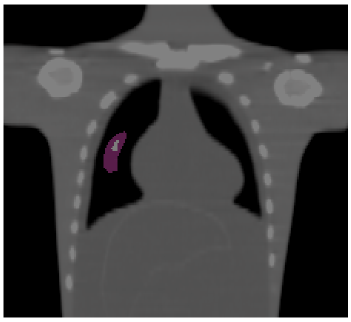

Fig 6. ^{90}Sr/^{90}Y, 2200 kBq, point source positioned in the right lung.

Conclusions

The conclusion of these pilot studies can be summarised as follows;

- Lung uptakes of a point source with gamma energy high above (^{60}Co) the ULD appear as a cloud with no clear delimitation by planar gamma camera and as a point with distinct delimitation by SPECT-CT imaging system.
- Lung uptakes of a point source with gamma energy close to or right above (^{137}Cs) the ULD can be visualized as a point with distinct delimitation both by planar gamma camera and by SPECT-CT imaging system.
- Lung uptakes of a point source of a pure beta-emitter (^{90}Sr/^{90}Y) can clearly be visualized as a point with distinct delimitation both by planar gamma camera and by SPECT-CT imaging system. In fact, the bremsstrahlung of ^{90}Sr/^{90}Y is more clearly delineates the hot particle in the planar image than do the scattered ^{60}Co radiation.

Our study thus indicates that clinical gamma cameras and SPECT-CT can be a useful instrument for visualisation of human lung contents of gamma emitting hot particle normally associated with emergency preparedness scenarios, and for pure β-emitting radionuclides as well. Future work is to examine which settings of the SPECT-CT are optimum for visualisation of high photon energy emitting sources in the inhalation tracts, and the detection limit in terms of activity for which the source can be visualised in an image at given acquisition times.

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Whole-body counters for measurement of internal contamination in Finland

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Abstract

STUK - Radiation and Nuclear Safety Authority - has two whole body counters for measurement of internal contamination. Both counters use high purity germanium detectors. The stationary system is installed inside a 50-ton iron room. The new mobile unit is built on a truck chassis. Both counters are used to assess the internal exposure of radiation workers and that of the Finnish public in general. Up to now, information of the location of the internal contamination has not been available due to measurement geometry in the mobile unit.

For this reason, a project was started to design and build a new counter in order to obtain location information and to improve detection efficiency. Monte Carlo simulations exploiting MCNPX were used together with voxel phantoms to guide the design process. The mobile unit can also be mobilised in emergency situations and it provides a fast and reliable method for both screening and dose assessment purposes. The need for assessing internal radiation doses in emergency situations is evident and has been demonstrated after the accidents e.g. in Brasil and Ukraine. This paper describes a design for the new mobile counter and presents calculated predictions of detection limits and efficiency.

Introduction

Assessment of internal radiation doses can be done using results from direct measurement of people or indirectly by excreta measurements. Estimations can also be made using air concentration data or activity concentrations in food stuffs combined with consumption data. By experience from occupational contamination cases we have noticed that air concentration data tends to give too low an estimate of the intake as well as estimates done from dietary data. In principle, the dietary data gives estimate for the whole population. However, individual differences are significant, depending on how much each individual uses wild-caught fish or wild berries or mushrooms. Therefore, the need for reliable measurements is obvious. The aim of measurements is to determine the intake of radioactive substances. The internal radiation dose is then assessed using metabolic and dosimetric models. *In-vivo* measurements are used to assess the internal exposure of radiation workers as well as the exposure of the public. In cases with high internal contamination, the purpose of measurements is to help in deciding if medical treatment or other types of measurement for more exact dosimetry is needed. In situations with prolonged exposure repeated measurements are recommended. In emergency situations direct measurements should be done as soon as possible after an alert to give support for decision making and to reassure the general public.

Equipment

STUK has obtained a new mobile whole-body counter for measurement of internal contamination (Figure 1). The unit has been built on standard Volvo FE 42R truck. The carrossery is insulated and equipped with air conditioning, along with electric and diesel heaters which makes all year usage comfortable. The monitoring unit uses 230 V AC. This external power is backed up by a UPS system consisting of a powerful sine wave inverter, a sophisticated battery charger, a high speed AC transfer switch and AC distribution in a single light weight and compact enclosure. A battery pack (24 V/400 Ah) has been added in order to maintain measuring devices 24 hours in case of loss of the external mains. This battery pack is also charged by the alternator of the truck when the engine is running.

Figure 1. A new mobile whole-body counter.

The whole-body monitor inside the carossery consists of two HPGe detectors and Dspec Pro digital electronics from Ortec (Ortec 2005). The present geometry used is a modified chair with a shadow-shield made of lead to reduce background in detectors (Figure 2). The detector set-up consists of a coaxial p-type HPGe-detector with a 90% efficiency and a GAMMA-X detector which is a coaxial n-type HPGe having an efficiency of 80%. The former detector is placed in the middle of the chair for whole-body measurements and the latter is placed closer to the upper body, providing the possibility to detect iodine accumulated in the thyroid, for example. The GAMMA-X has an ultra thin entrance window made of beryllium, providing good efficiency also for low energy γ -rays. The detectors are surrounded by a 5 cm thick lead shield. The typical time used in a routine measurement is 1000 s. Background is determined using the background phantom consisting of 14 pieces of 5 kg sugar bags. If needed the measurement distance can be adjusted by moving the detectors and the lead shield up or down.

Figure 2. The present measurement set-up in the mobile unit. The modified chair is made of lead. The white shadow shield is also made of lead from "waist" down. Upper part is made of steel covering the detectors and supporting them.

The stationary whole-body counter is shown in Figure 3 below. This chamber is built of old steel in order to avoid background radiation (often tiny amounts of ^{60}Co) from freshly rolled sheets. This indoor unit, usually called Ironroom, uses scanning bed geometry, detectors being positioned in a fixed location surrounding the person. The detectors presently in use are two HPGe (both 80 % crystals) above the person and one 28% crystal under the bed. In addition, two NaI detectors are installed above and two below the bed. The analysis program provides also the total count rate versus location graph. However, as the detectors are not collimated, only a modest spatial resolution can be achieved.

Figure 3. Inner view of the Ironroom. The indoor whole body counter uses HPGe and NaI detectors, some of which are visible above the person being measured.

The background phantom used in Ironroom is made of plastic containers filled with sugar. The containers include pieces for thorax, arms, legs, head and neck. The neck part can be replaced with a thyroid phantom.

Method

The efficiency calibration for both units is performed using the adult St. Petersburg whole-body phantom (Kovtun 2000) with ^{60}Co, ^{137}Cs, ^{40}K and ^{152}Eu rods (energy range 122-1460 keV). Figure 4 shows the efficiency calibration for a mobile unit. Both calculated and measured curves are shown. In addition, three St. Petersburg thyroid phantoms with ^{133}Ba capsules are available: adult, teenager (14 years old) and child (6 years old). Barium is used to imitate ^{131}I as its gamma energies are in the same energy range. The body burden of adult persons is determined roughly from knees to nose in the mobile unit. The Ironroom measures whole person. The whole-body phantoms from 12 kg to 110 kg corresponding ages from two-year old to adult were used to obtain the correction to the efficiency due to the size of the measured person. The correction factor ranges from 1.4 (12 kg) via 0.9 (90 kg) to 1 (110 kg).

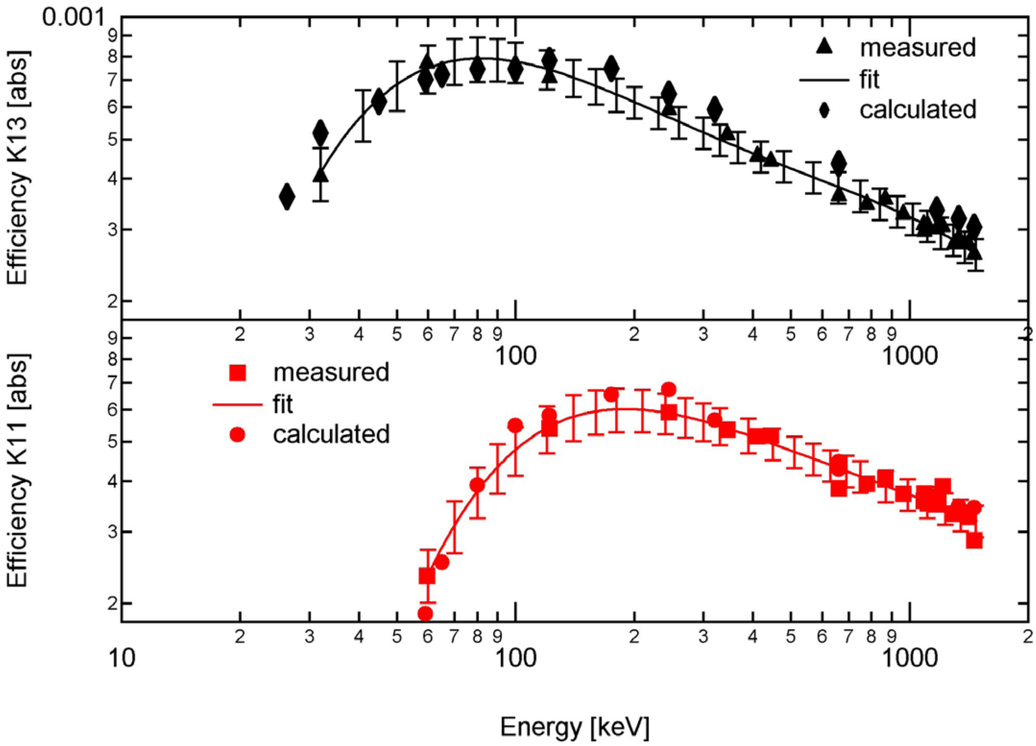

Figure 4. Efficiency curves for the p-type HPGe (K11, lower) and n-type GAMMA-X HPGe (K13, upper) detectors.

The uncertainty on the absolute γ-efficiency is 10 % for γ-rays >200 keV and goes up to 15 % towards lower energies. The final uncertainty on the activity measured will be determined by adding quadratically the statistical uncertainty of the identified γ-peak and that of the efficiency. For the most of the cases, uncertainties on the branching ratios and half-lives of the nuclei are so small that they can be neglected. The minimum detectable activities (MDAs) (Moilanen 2007) for the most commonly detected artificial nuclei, ^{137}Cs and ^{60}Co have been determined to be of the order of 100 Bq. MDAs are roughly factor of two smaller in Ironroom. The minimum detectable activities for the most of the common radionuclides found in nuclear power plants, industry or radiopharmacy are sufficient from the point of view of radiation protection. As the MDA depends on the background level it should be determined when the background changes. In emergency situations the MDAs can be higher due to the higher background from the environment. However, as the MDA increases as a function of square root of background rate, the equipment will be able to handle it. *In-vivo* monitoring can also be used to follow prolonged exposure e.g. after a nuclear or radiological accident. Presently, the efficiency calibration assumes that the radionuclides are homogenously distributed in human body.

Mathematical calibrations and new set-up

STUK has launched the project in order to upgrade the measuring units. The main goal of the project is to obtain better treatment for the radionuclides that are not homogenously distributed in the body, for instance freshly inhaled radionuclides in the emergency situations. Other goals are to improve ergonomics for the person being measured and to gain more information about the location of the contamination in the body in the cases where intake path is not known. In addition, lower detection limits for some specific nuclei (like ^{241}Am, ^{123}I) which emit only low energy (<200 keV) gamma rays will be necessary to achieve. To begin with, Monte Carlo simulations will be used with voxel phantoms. The code selected is MCNPX (Monte Carlo N-Particle eXtended, Pelowitz 2005).

The simulations were started by reproducing the calibration for the present set-up where good measurements are available (Figure 4). For a new set-up, different possibilities were considered and ordinary chair geometry was chosen for a further study (Figure 5).

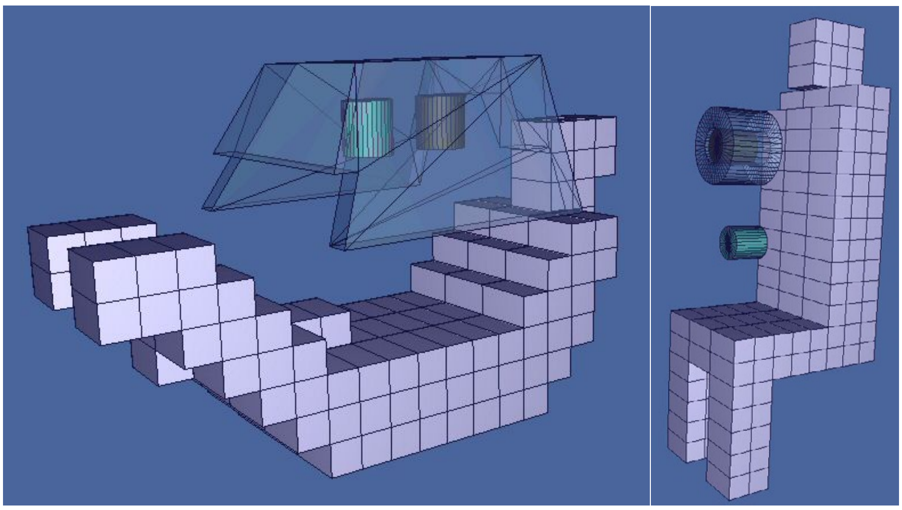

Figure 5. Models for MCNPX simulations. The present mobile unit set-up is on left and the new set-up is shown on the right. Transparent part is the lead made background shield. In the new set-up, shield of the lower detector is removed from the picture.

The background shield will be made of lead (not shown in Figure 5). There will be 5 cm of lead at the back, top and bottom and front side is open for a measurement. Cylindrical collimators around detector crystals can be moved along the axis, allowing fine tuning of the spatial resolution.

Figure 6 below presents calculated detection efficiency for a present and a new set-up in mobile unit. The overall efficiency with the new set-up will be better by a factor of 1.7 as the detectors can be closer. However, the detection efficiency will improve only for the region below the detectors. This can be compensated by using the sum spectrum for species like ^{137}Cs, which is known to be evenly distributed. For non-even distributions localisation of the activity can be done by comparing the measured two spectra. As can be seen in Figure 6, spatial resolution will be just enough to determine whether the activity is in lungs only or whether it has already been transferred further, for instance to liver as ^{60}Co would do.

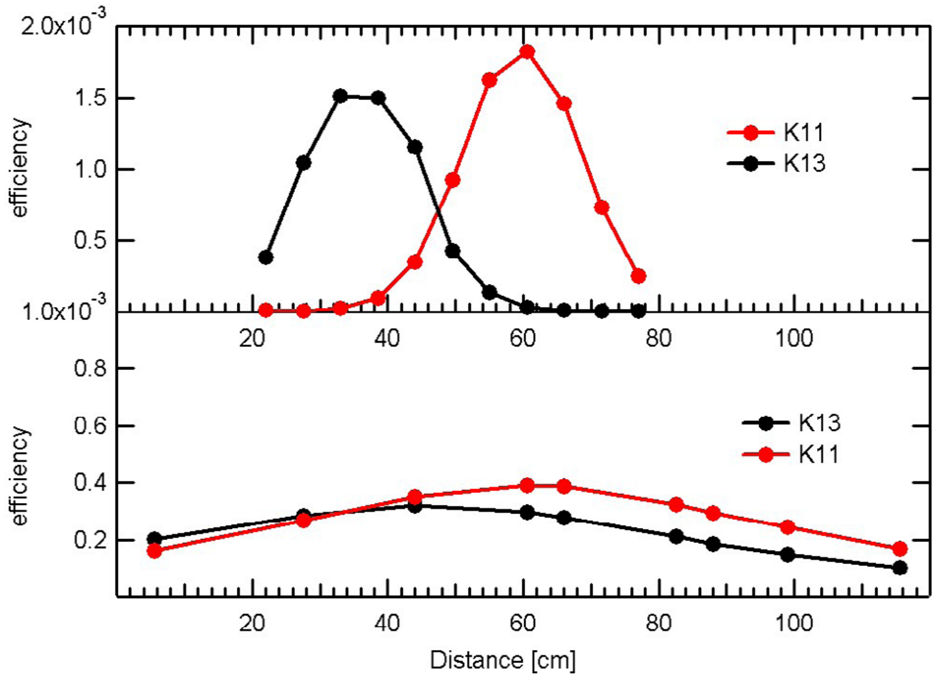

Figure 6. MCNPX simulation results for a polyetylene brick phantom. Each point has been calculated by placing 30 cm long cylinder of ^{137}Cs in different locations perpendicular to longitudinal axis of the phantom. The top curve displays the new set-up and the lower is for the present one.

Conclusions

In-vivo measurements are used to assess the internal exposure of radiation workers as well as the exposure of the public. The whole-body counters in STUK fulfil the requirements defined by the dose registration limits of radiation workers. In emergency situations it will be necessary to perform also direct measurements on people for reassurance of the public even if such measurements would not be necessary from a strictly radiation protection point of view. The new measurement unit will provide fast and reliable method for both screening and dose assessment purposes. The gamma ray detection efficiency for the whole energy range will be greatly increased. A significant improvement will be the new feature of localisation of the internal contamination. In addition, it will give more precise information about possible ^{131}I accumulation in the thyroid. The new set-up presented above will be studied further in STUK. Calibrations with well defined calibration phantoms need to be performed before commissioning.

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Determination of chest wall thickness of anthropometric voxel models

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Abstract

Chest wall thickness is a major quantity for the calibration of lung counting detector systems. A method for the determination of chest wall thickness of anthropometric voxel models is introduced and applied. The computer-aided method was compared successfully with mechanical measurements and found valid within small error ranges. This method allows the generation of three-dimensional, colour-coded profiles for visualisation of tissue thicknesses overlying an organ of interest. Furthermore Monte Carlo simulations of lung counting scenarios involving two germanium detectors were performed to show the performance of the method with a selection of voxel models.

Introduction

Chest wall thickness – an important counting efficiency calibration parameter for *in vivo* measurements

The detector efficiency for *in vivo* measurements of low energy photon emitters in the thorax (e.g. lung or liver) depends strongly on chest wall thickness (CWT). Hence, CWT is an important calibration parameter for counting efficiency. Systematic errors for the determination of counting efficiency can be reduced if the accuracy of the estimation of CWT can be improved.

The first challenge is a clear definition of CWT. For lung counting, the distance from the anterior skin surface of the thorax to the pleura (thin skin surrounding the lung) can be taken as a definition for CWT. However, this distance is not constant throughout the chest, but CWT can be averaged in the region that is covered by the detector. Consequently, position and size of detectors are important parameters for this definition.

Sumerling and Quant (1982) suggested a standard geometry for placing two scintillation detectors, one on the left, and one on the right side of the anterior chest. The centre of the detector was positioned in the intercostal space of the second and third rib. They performed ultrasound measurements on each side in the area of a circle with the detector's diameter to determine CWT.

Physical phantoms are used for calibrating counting efficiency. For example, the LLNL Torso Phantom (Griffith *et al.* 1978) has a set of overlays that can additionally be put onto the chest for the determination of counting efficiency. This way, counting efficiency can be described as a function of CWT.

Chest wall thickness of anthropometric voxel models

Anthropometric voxel models are increasingly used to calculate counting efficiencies for *in vivo* measurement systems numerically with radiation transport simulations based on Monte Carlo methods. An MCNPX (Hendricks *et al.* 2005) based Monte Carlo simulation environment for the partial body counter at KIT has been developed and validated (Hegenbart *et al.* 2009). Simulations can help to investigate counting efficiencies of radionuclides incorporated in arbitrary organs that might not be available in the laboratory's inventory. Thus, sensitive parameters can be identified which is helpful to develop strategies to reduce systematic errors of classical counting efficiency calibration.

Using voxel models can in principle help to represent an individual better than an ordinary physical calibration phantom, especially, if the model is customisable. Several novel techniques have been developed and allow generating customised anthropometric voxel models. The generation of customised voxel models for better representation of individuals should be based on biometric data of the individual. Hence, there is a demand to develop methods for measuring biometric data of voxel models in virtual reality. Those methods need to be comparable to conventional methods in reality. A previous example for this was the measurement of chest circumferences and cup sizes of a series of female voxel models (Hegenbart *et al.* 2008).

Since CWT is a sensitive biometric parameter for counting efficiency, it is meaningful to develop a method to determine the CWT of anthropometric voxel models.

Material and methods

Determination of CWT of voxel models

The determination of the CWT of anthropometric voxel models is based on measuring and averaging distances of body surface (skin) voxels to voxels of the target organ, i.e. the lung in case of lung counting.

The method is inspired by the determination of CWT with ultrasound, as for example presented by Sumerling and Quant (1982). The surface voxels are chosen according to a given detector diameter and its position and alignment onto the chest. The voxels lie in a cylindrical projection of the diameter of the detector along its axis onto the body surface. Starting from the surface voxels, the shortest distance – often perpendicular to the surface – to the nearest voxel of the organ of interest is determined. Depending on detector diameter and voxel resolution, hundreds to thousands of surface voxels are selected. Single distances x_i for each voxel are stored in a list.

The arithmetic mean can be calculated with the distances x_i from the list. Moreover, after defining a linear attenuation coefficient – suitable for the concerned tissue and photon energy – the *effective* CWT (Dean 1973, Kramer and Hauck 1997) can be calculated.

$$CWT_{eff} = -\frac{1}{\mu}\ln\left[\frac{1}{n}\sum_{i=1}^{n} e^{-\mu x_i}\right]$$

CWT_{eff}: effective chest wall thickness (mm)

n: number of list entries

x_i: distance i (mm)

μ: linear attenuation coefficient (mm^{-1})

The effective CWT takes the exponential attenuation law into account. Short distances are given higher weight, since photons can pass through them more likely. The higher the photon energy considered for the attenuation coefficient, the closer the effective CWT gets to the arithmetic mean. For photon energies starting from 60 keV, the effective CWT can be approximated with the arithmetic mean (Kramer 1999).

The introduced method for determination of the CWT of voxel models is integrated in the in-house developed software Voxel2MCNP (Hegenbart 2009). This software is able to generate and visualise *in vivo* measurement scenarios and furthermore to prepare and evaluate MCNP(X) Monte Carlo simulations with voxel models. The viewer of the software allows the generation of three-dimensional, colour-coded illustrations of the local distribution of the tissue thickness, i.e. a CWT-profile of a voxel model. This information can be used for optimisation of detector position and helps judging if a voxel model keeps a reasonable anatomy after customisation.

Validation of the method

The *in vivo* Measurement Laboratory (IVM) at KIT has a copy of the LLNL Torso Phantom (Griffith *et al.* 1978). The thicknesses were measured mechanically with a calliper within 14 cm diameter from the centre of the three concentric markings on the chest cover and on each overlay defining the detector position for lung and liver measurements. The physical phantom was compared with the voxel model of KIT's LLNL Torso Phantom, which was generated from computed tomography (CT) images performed at the Vincentius Kliniken of Karlsruhe by Prof. Dr. J. Lehmann and his team. The CT images were segmented and successfully used to validate Monte Carlo simulation of an *in vivo* measurement scenario (Hegenbart *et al.* 2009).

The mechanically measured data were compared to the corresponding CWT data obtained from the computer method from the voxel model. Furthermore, a comparison with effective CWT values from literature (Taylor 1997) was performed.

Monte Carlo simulations

Monte Carlo simulations of lung counting scenarios have been performed to investigate the relationship of voxel model's CWT and the corresponding counting efficiency. The voxel models used in this investigation were the

- MEET Man (Sachse *et al.* 1996) and Voxelman (Zubal *et al.* 1994) with organ density and material data from ICRP 89 (ICRP 2002),
- in-house developed voxel models of the LLNL-Phantom without and with four overlays with measured organ densities and materials specified by the manufacturer,

- LLNL-Phantom of the EURADOS Intercomparison on Monte Carlo modelling (Broggio 2009),
- ICRP Adult Male (ICRP 2009), and
- Godwin (Zankl *et al.* 2005) and Frank (Petoussi-Henss 2002) of HMGU.

With these voxel models *in vivo* counting scenarios were generated involving ^{241}Am-photon-emissions (Schötzig and Schrader 2000) homogeneously distributed in all lung voxels. The voxel models lie in supine position on a model of the examination table of KIT's partial body counter. In this scenario two detectors were placed on the chest using the validated model of the *in vivo* Measurement Laboratory's Canberra XtRa HPGe detector with a diameter of 7.5 cm (Marzocchi *et al.* 2009). The detector was inclined by 25° in order to be parallel to the skin surface of the chest with about 1 to 2 mm distance. The detector axis was centred on the third rib, 7 cm away from the centre of the sternum. This setup was symmetrical for one detector on the left and on the right side.

The CWTs (arithmetic means) of these voxel models have been determined with the above introduced method considering the detectors' positions and diameters. A CWT profile was generated for all phantoms.

MCNPX 2.6 (Hendricks *et al.* 2008) was used to simulate 10 million photon histories for each voxel model. The voxel model syntax for the MCNPX-input-file was generated with Voxel2MCNP according to recommendations by Taranenko *et al.* (2005). The code was run in MCNPX's mode p (photon transport only). The electron transport mode was skipped, because no significant differences were found in test calculations. Thus, it was possible to save computer time. The cross-section library PLIB=04p for photons and ELIB=03e for electrons were used.

MCNPX's F8 tallies were assigned to the active HPGe-crystal volumes. The counting efficiency was calculated in the energy range from 57.5 keV to 61.5 keV, which is about +/-1.5 FWHM. The net counting efficiency was determined subtracting the background with the trapezoidal rule from the full absorption peak of the 59.5 keV gamma line of ^{241}Am. Results from both detectors were summed up.

Results and discussion

Validation of the method

Table 1 shows a comparison of mechanically measured CWT values and calculated values from the corresponding voxel models. Both columns show the effective CWT with μ/ρ=0.93 cm^{2}g^{-1} for polyurethane tissue equivalent material with a muscle/adipose ratio of 50/50 for the overlays and μ/ρ=1.13 cm^{2}g^{-1} for 100% muscle for the chest cover, respectively, both at 17 keV (NIST 2009). The comparison of effective CWT values was preferred. The first reason for this was, to have a more strict comparison, which takes variances stronger into account. The second reason was, to make a comparison with literature values.

The values are in agreement with a maximum deviation of 1.0 mm. The values for the chest cover show the largest deviations. This is can be explained by the comparably large variances of the single measurements. 2σ-errors were estimated for the calliper measuring (0.1 mm) and for position inaccuracies (0.95 mm) on the voxel model. The method for determination of effective CWT of voxel models can be taken as valid, considering an error of roughly 1 mm.

Most measured and calculated values lie in the range of values reported in the literature. Some phantom values are smaller than those reported by Taylor (1997). The investigated LLNL phantom at KIT is a third-generation, commercial version and might not be comparable with the second generation investigated by Taylor (1997). Manufacturer's specifications are unknown.

Table 1. Effective CWT of the mechanical measurements in the three concentric markings (liver, left and right lung) of the LLNL phantom of the phantom compared with the calculated values of the corresponding voxel model (Gün 2010). The obtained values are faced to literature values (average and range) from five laboratories, which owned the second generation of the LLNL phantom (Taylor 1997).

Configuration (concentric marking)	CWT_{eff} (mm) of phantom	CWT_{eff} (mm) of voxel model	Absolute deviation (mm)	Literature average values (range) (mm)
Chest cover (left lung)	15.4	16.2	0.8	16.4 (15.6-17.2)
Overlay B1 (left lung)	6.1	6.1	<0.1	6.6 (5.4-8.9)
Overlay B2 (left lung)	12.1	12.5	0.4	12.6 (11.2-14.5)
Overlay B3 (left lung)	17.2	17.3	0.1	17.6 (16.4-19.0
Overlay B4 (left lung)	24.0	24.2	0.2	24.7 (24.2-26.1)
Chest cover (right lung)	14.0	15.0	1.0	14.6 (13.3-15.0)
Overlay B1 (right lung)	6.3	6.5	0.2	7.0 (6.0-9.0)
Overlay B2 (right lung)	12.3	12.1	0.2	12.7 (10.8-13.8)
Overlay B3 (right lung)	17.0	17.4	0.4	17.9 (17.4-19.6)
Overlay B4 (right lung)	24.8	25.3	0.5	25.6 (24.8-27.2)
Chest cover (liver)	13.6	13.2	0.4	14.1 (12.9-15.2)
Overlay B1 (liver)	6.2	6.2	<0.1	6.8 (4.4-9.9)
Overlay B2 (liver)	11.2	11.1	0.1	11.8 (9.5-14.4)
Overlay B3 (liver)	16.4	16.3	0.1	18.1 (16.4-20.6)
Overlay B4 (liver)	24.5	24.4	0.1	24.7 (23.7-27.3)

CWT profiles

Figure 1 shows colour-coded CWT profiles (skin-pleura distance) of a selection of three voxel models. The profile of KIT's LLNL voxel model shows strong asymmetric shading due to the position of the heart, which covers partly the left lung lobe. This uncovers some limits of the LLNL phantom's anatomy in terms of realism. The other two models exhibit just a slight asymmetry. The ICRP Adult Male's chest wall is coloured partly in dark red, because he is less obese compared to the MEETMan. The red zones are suitable for placing detectors in order to get high count rates. Notable are local CWT minima near the second rib and at the lower end of the lung lobes. Such coloured profiles can be generated for every organ (e.g. the liver).

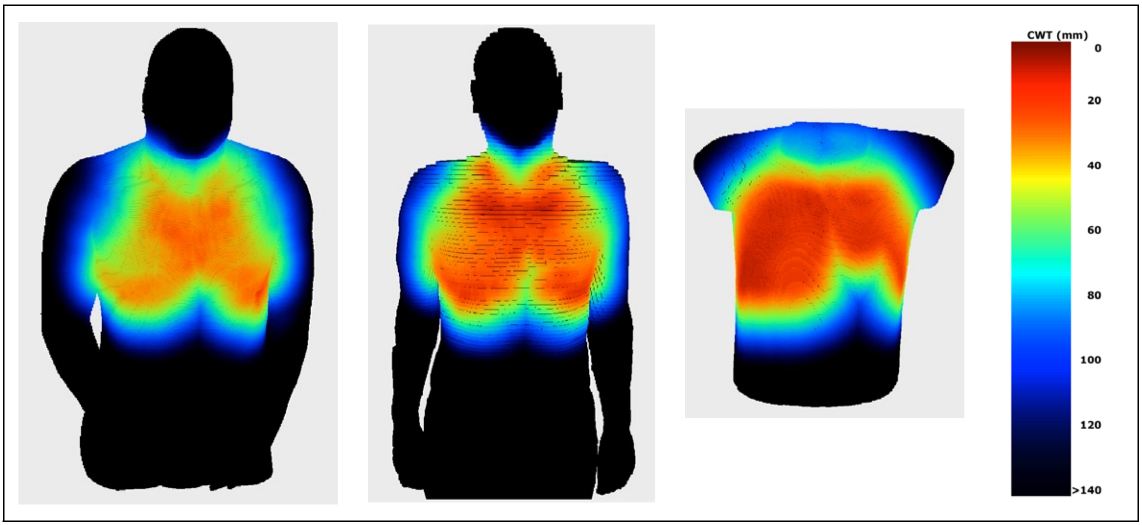

Fig. 1. CWT profiles of the MEETMan (left), the ICRP Adult Male (middle) and KIT's voxel model of the LLNL Phantom without overlay (right). Deep red colour indicates a small CWT (see legend, right). Larger CWT are indicated with the transition to shades of green. Dark-blue and black areas are not meaningful, since the distance to the lungs is too high and angles are not perpendicular to the skin surface anymore.

Monte Carlo simulations

Figure 2 shows the summary of the results. The left and the right detector's counting efficiencies for a ^{241}Am-lung-counting scenario were summed and plotted over the determined CWT (arithmetic mean of left and right) for each voxel model. Similar to routine *in vivo* measurements, the KIT's five LLNL voxel models were used to determine an exponential calibration curve. As a first approximation, the exponential calibration curve is suitable for most voxel models, except the ICRP Adult Male and Godwin. Their counting efficiencies are considerably higher. Yet unknown, the responsible parameters for the observed higher efficiencies need further investigation. The EURADOS LLNL voxel models P0 and P4 lie on the curve as expected. Since air cavities under the chest cover of these models were not considered, CWTs are higher and hence, counting efficiencies are lower compared to the corresponding KIT LLNL models. Two resolutions of the MEETMan were also investigated and compared. The plot shows that CWT and efficiency differ only slightly within the error bars.

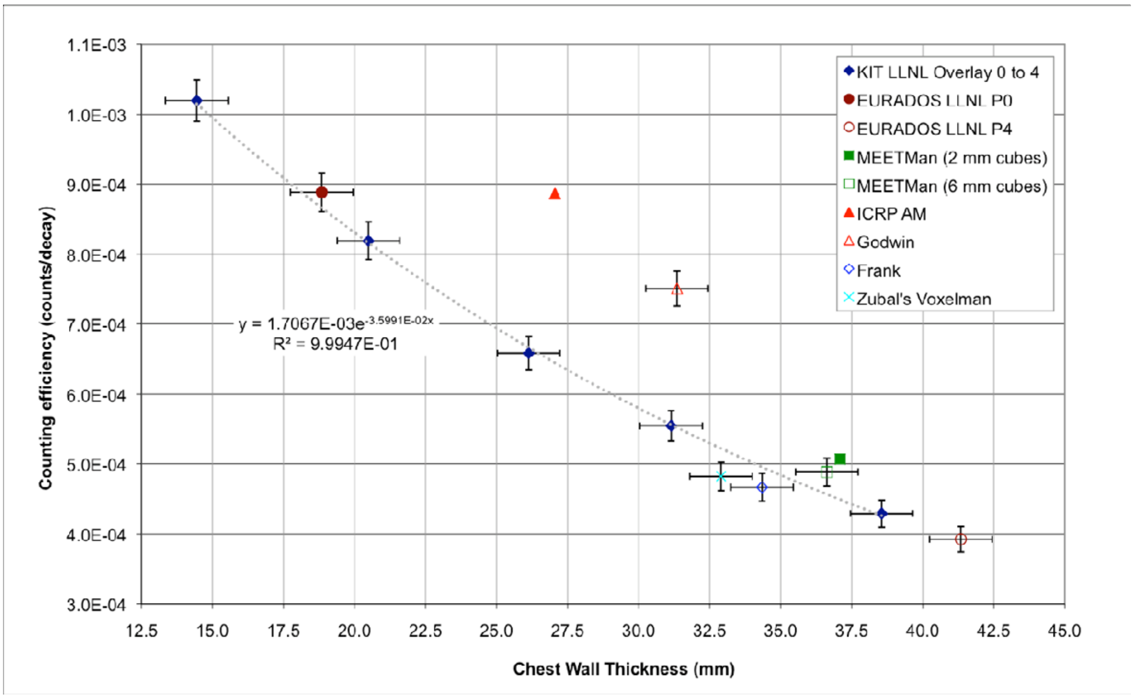

Fig. 2. Lung counting efficiencies calculated with MCNPX plotted against the determined CWT of various voxel models. An exponential fit (dotted gray line) was performed with Microsoft's Excel with data from the five KIT LLNL models (solid diamonds). The formula and the R2-coefficient are given in the plot. The horizontal and vertical error bars indicate the error (2σ) of the CWT and the net counting efficiency in the ROI (2σ).

Conclusions

The authors presented and validated a method for determination of CWT of anthropometric voxel model with an error of about 1 mm.

This method can also be used to generate colour-coded, three-dimensional CWT-profiles. The profiles are helpful for verifying the realism of anatomy of voxel models, i.e. models of man-made phantoms or customised/adapted models. Furthermore, it gives useful information for detector positioning.

An exponential calibration curve depending on CWT was generated from KIT's LLNL voxel models for a typical ^{241}Am-lung counting setup with germanium detectors. As a first approximation, the curve can be used for counting efficiency calculation for most voxel models considered in this investigation. The ICRP Adult Male and Godwin show that there are other important parameters that play a role in the determination of lung counting efficiency. Thinkable is another important quantity in the assessment of thoracic deposits of nuclides emitting low energy photons: the composition of the tissue in the anterior chest wall, i.e. the fraction of muscle, adipose and bone tissue. Next steps will involve the development of algorithms, which determine such fractions in voxel models.

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Measurements and Monte Carlo computation of the ^{241}Am counting efficiency pattern along the case #0102 USTUR leg phantom

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Abstract

Some Actinides like Americium are bone-seekers, and according to their biokinetics are retained in the skeleton after an intake. Because of the radiosensitivity of red-marrow and endosteum cells, the assessment of the bone activity is an important matter to take into account. *In vivo* counting of bone seekers is usually performed at the skull or at the knee, since the attenuation of radiation is here limited.

The USTUR (U.S. Transuranium and Uranium Registries) disposes of a unique physical phantom containing the left half of the skeleton of a donor significantly contaminated by ^{241}Am. The left leg phantom, containing bones from hip to foot, embedded in muscle equivalent plastic, has an activity of 1026 Bq. This phantom was measured in three European *in vivo* facilities and one Canadian laboratory; Monte Carlo (MC) simulations were also performed to assess the reliability of numerical calibration based on voxel phantoms.

Using different Germanium detectors the same efficiency pattern was found by participants, when counting positions were comparable. A sharp increase of counting efficiency is found at the patella level, about 10 times higher than at other locations, at this level a 5 cm displacement of detectors can result in 50% change in efficiency. Measurements on each side of the leg show a significantly different efficiency pattern.

A voxel phantom was build after acquisition of 234 CT scan images and used to simulate the measurements. Despite the gross feature of the efficiency pattern is reproduced by MC simulations, discrepancies larger than 50% have been found for some measurements. Several factors can explain these discrepancies: difficulties in building the voxel model and in reproducing the measurement conditions, uncertainty about the total activity and the activity distribution in bones.

Based on this study recommendations will be given to ensure reliable measurements, the efficiency obtained in this study will also be compared with other relevant data.

Advanced Monitorin In Mixed Radiation Area – AMIRA

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Advanced Monitoring In Mixed Radiation Area – AMIRA

AMIRA is the result of a research project in cooperation with the Radiobiological Institute of the University of Munich, a workgroup of the University of Saarland and Saphymo GmbH. An active, universal applicable dosimeter for radiation protection has been developed providing precise neutron and photon measurements within mixed unidentified radiation fields. AMIRA uses a miniaturized tissue equivalent proportional counter tube (TEPC), simulating a microscopic volume of 1 µm diameter. Using a microscopic detector offers the advantage that for each event of energy deposition not only the proportion of dose but also the lineal energy, the micro dosimetric analogon of the LET, can be determined. This allows assigning to each individual event the correct quality factor for the determination of the equivalent dose. Thus, a separate indication of neutrons and photons is not really required for radiation protection applications although it is principally possible.

The energy response of most conventional instruments is often varying by a factor of 10. Measuring results of PTB Braunschweig – Germany and IRSN Cadarache – France prove an extremely high energy response within the entire measuring range.

The latest version of the measurement device is based on reduced size and weight, disposes of an economized MCA with more than four decades dynamic measurement range and suppresses microphonic effects. The software saves the LET spectrum, calculates and saves the total dose according to ICRP60 as well as the neutron/photon dose by LET separation. Due to the small, compact construction it could be achieved for the first time to make use of the TEPC technology for applications everywhere where there is occupational exposition. Examples can be found with aircrew dosimetry or in the complete spectrum of the nuclear fuel cycle for the determination of the personal dose and dose rate.

Introduction

The exact measuring of the exposure rate gains more and more importance due to growing activities of human beings within radiation exposed areas (i.e. aviation, astronautics, medicine, nuclear techniques, etc.). People are mostly confronted with mixed radiation exposure composed of charged particles, neutron and photon rays. Therefore, at workplaces being subject to radiation exposure dosimeters for measurement of radiation, dose rate and environmental surveillance are applied.

Common dosimeters using semiconductor detectors cannot determine the correct proportion of neutrons and photons of the equivalent dose, although especially the neutron exposition is known to have a quite high injury potential to humans. This results from the great variety of interacting processes and their distinctive dependence from material composition and neutron energy. Further, the high and energy depending RBW (relative biological effectivity) requires the discrimination of neutrons and photons with most of the detector conceptions.

This fact as well as the growing interest in active electronic dosimeters providing lower detection limits as passive equipment, more flexibility in use and real-time monitoring features (e.g. alarm function) generated the motivation for the development of a new personal dosimeter. While official dosimetry is still exclusively applying passive detectors the in-plant use of electronic dosimeters for monitoring tasks is increasing. It is expected that the passive dosimeters will be replaced by electronic units in the medium term.

Along with the described AMIRA development the correct measurement of neutron radiation using a personal dosimeter based on the Tissue Equivalent Proportional Counter (TEPC) technology should be realized. Neutrons and photons within mixed unknown radiation fields should be accurately detected using an active handy measuring device.

Material and methods

The most important challenges with respect to this project were the reduction of mass and size of a sufficiently sensitive TEPC, the improvement of the gas booster's long-term stability as well as the elimination or reduction of the sensitivity against acoustic noise and shock (microphony) which is inherent to all TEPCs. Additionally, a miniaturization of all required electronic components and the reduction of power consumption should help to optimize the use of the new instrument as personal dose meter.

Apart of the detector the measuring system also includes a high-voltage supply, a charge sensitive pre-amplifier, a 4 decade multi channel analyzer, a display, PC-interface and power supply (rechargeable batteries). All components are designed for minimum power consumption.

Based on a tissue equivalent proportional counter (TEPC) with a simple geometry and modular design a plastic (A150) is used which due to its interaction with neutrons simulates the characteristics respectively the features of a biological tissue with a microscopic volume of 1µm diameter. The Radiobiological Institute of the University of Munich (SBI) already proved the basic technical feasibility of such detectors.

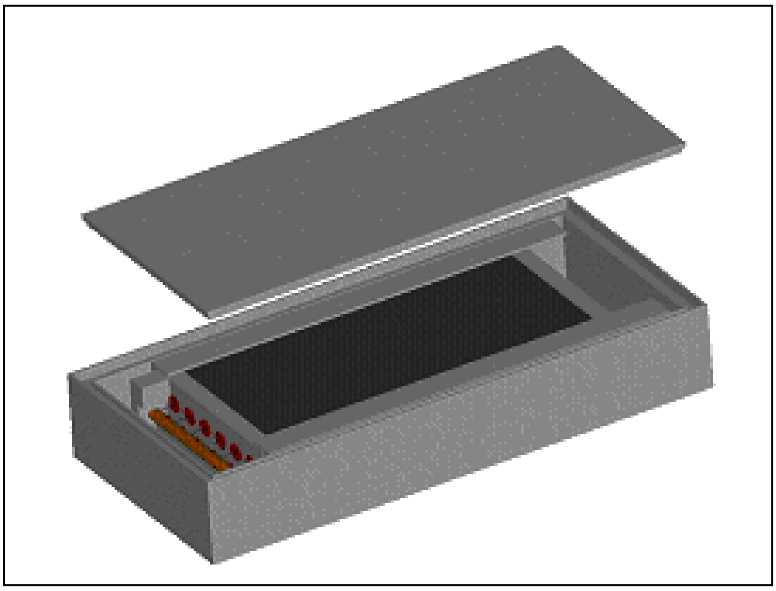

Fig. 1. Miniaturized multi-element TEPC (Tissue Equivalent Proportional Counter).

Within applications in the radiation protection area tissue equivalent micro dosimetric proportional counters offer the great advantage that they are simulating the energy deposition within the tissue - that means the so called lineal energy - the micro dosimetric analogon to the LET (= linear energy transfer) is determined for each individual event of energy deposition. This measuring method is therefore able to allocate a quality factor to each given radiation contribution. It therefore allows giving evidence about the biological radiation effect within mixed radiation fields (determination of equivalent dose).

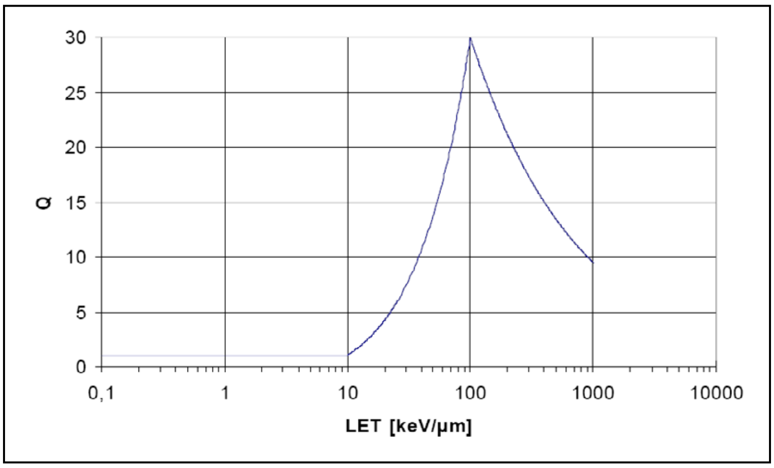

Fig. 2. Quality factor Q as function of LET L [1].

Equivalent dose and dose rate are calculated in a five second interval and displayed. Determination of absorbed dose contributions of neutron and photon radiation is not required as the different radiation components are automatically evaluated according to the respective LET. Analogous to each measured impulse a weighting factor Q according to ICRP 60 is assigned to allow to indicate the result using the measuring categories $H_p(10)$ respectively $H^*(10)$. Fig. 3 shows the LET spectrum of Cs-137.

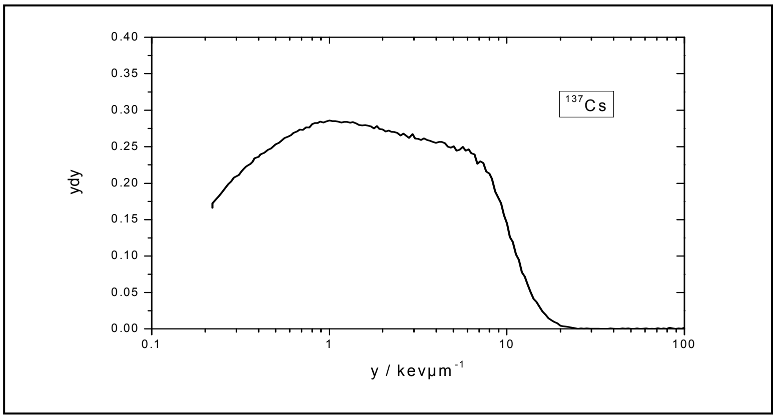

Fig. 3. LET spectrum for Cs-137.

An alarm function with simply settable dose rate thresholds is available. The accumulated event spectrum can be downloaded to a PC and can then be displayed and analysed in detail.

Fig. 4 shows a block diagram of the miniaturized electronics. The signal received from the TEPC reaches the pre-amplifier as a negative slope. This charge sensitive component is acting as trans-impedance amplifier and is adjusted to an optimal signal-noise-ratio. Subsequently, the signal is sampled by a fast high resolution ADC and transferred to a micro processor which registers the pulses using digital filter algorithms.

For dose meter configuration and data download the instrument provides an integrated RS-232 interface. The software running on PC also allows displaying the spectra of lineal energies – almost in real-time.

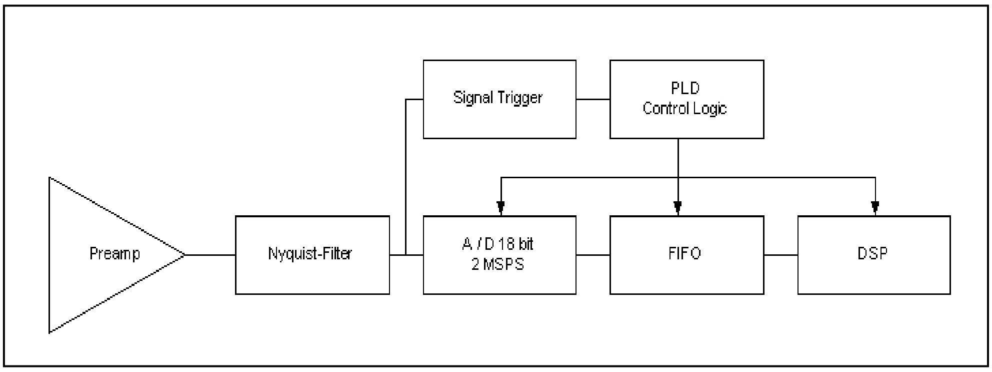

Fig. 5. Block diagram of the signal unit.

Results

Since the energy of neutron fields spreads over several decades the measuring technique is faced with a big challenge. The energy dependency of most of the commercially available measuring equipment is varying up to a factor 10 about this range. Fig. 5 is demonstrating that the new TEPC-design concept features an extremely high energy response across the entire field.

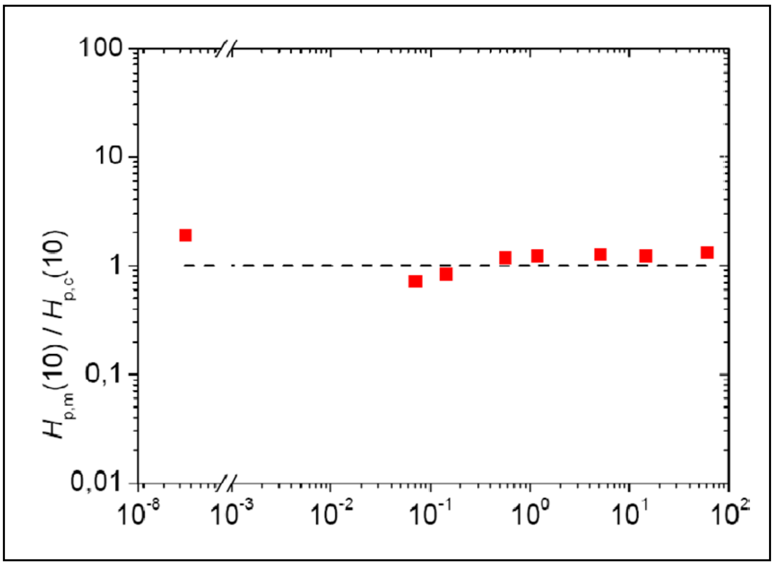

Fig. 6. Energy response. Source: H. Roos (IRSN-measurements).

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Simple method to determine ^{234}U and ^{238}U in water using alpha spectrometry

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Abstract
A simple method was developed to determine the activity concentration of ^{234}U and ^{238}U in water using alpha spectrometry. The method is based on rapid evaporation of water, mounting the residue into a vacuum chamber of an alpha spectrometer, alpha particle counting and subsequent spectrum analysis using a novel analysis program known as Adam. The method was compared to liquid scintillation spectrometry and alpha spectrometry with radiochemical sample processing. Consistent results were obtained.

Introduction
In a nuclear or radiation emergency or in nuclear security there is a need for rapid and simple off-line and on-line analysis methods to detect radionuclide threat agents in water. Efficient tools are necessary especially when screening the samples with low and medium activity concentrations. Simplified methods have recently been developed for example for liquid samples (Semkow et al., 2009), swipe samples (Ihantola, 2009) and air filters (Pöllänen and Siiskonen, 2006). Nevertheless, there still room to develop the methods.

Radiochemical sample processing is often prerequisite for successful measurements in alpha spectrometry. However, the processing may be too tedious and time-consuming especially in the case when the analysis results must be obtained rapidly. In addition, operations in the field pose special requirements although the processing is typically performed in a fixed laboratory.

In the simple and rapid off-line method investigated here, water samples were evaporated and the residues were counted as such using alpha spectrometry, i.e. without further radiochemical sample processing. A novel spectrum analysis tool known as Adam (Advanced deconvolution of alpha multiplets) was used to unfold the alpha spectra. The results were compared to those obtained by scintillation spectrometry and alpha spectrometry with radiochemical sample processing. The main focus was put on the congruity of the results obtained using different methods.

Material and methods

Simplified method for alpha spectrometry

Eleven drilled well water samples were selected for the study. An aliquote of 15 ml was evaporated in Teflon container to stainless disc under IR lamp. After evaporation the alpha particles were counted in an alpha spectrometer (AlphaAnalyst, Canberra) for one week (Fig. 1). The samples were measured from shelf one, where the distance from the semiconductor (PIPS, Canberra) detector is 2 mm. The area of the detector was 450 mm^2.

After evaporation, the walls of the Teflon container were rinsed using 1M HCl acid to determine the influence of possible adsorption. The activity concentration of rinsing acid was determined using liquid scintillation spectrometry (Quantulus 1220, Perkin Elmer), and the final results were corrected accordingly.

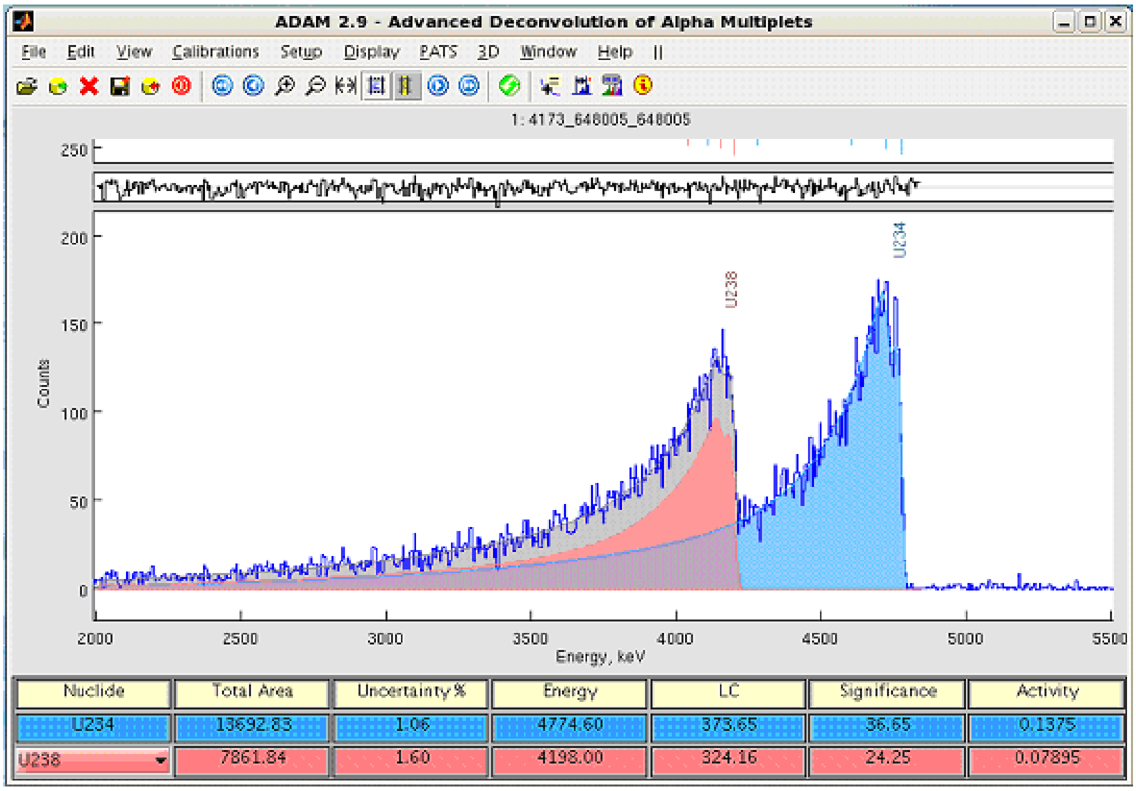

Fig. 1. Screen snapshot of an alpha spectrum from evaporated residue (sample ID 4173 in Figs. 2 and 3). The families of the peaks of ^{238}U and ^{234}U were fitted using Adam. All peaks are assumed to have the same shape. ^{234}U was not taken into account in the fitting. Reduced residual is shown at the top.

Gross alpha analysis

Sum activity concentrations were determined by liquid scintillation method (Salonen 1993, Salonen and Hukkanen 1997). The measurements were performed to screen the level of occurrence of the long-lived alpha-particle emitting (^{238}U and ^{234}U, ^{226}Ra and ^{210}Po) radionuclides in water. The samples (38 mL) were prepared by evaporating water into dryness with a freeze-dryer in a Teflon-coated polyethylene vial. The residues were dissolved in small amount of HCl acid and then scintillation cocktail Ultima Gold AB was added. The sample was counted one month after the sample preparation. During that time ^{226}Ra attains equilibrium with ^{222}Rn and its short-lived daughters. Gross alpha and beta are determined with a low-background liquid scintillation spectrometer (Quantulus 1220, Perkin Elmer).

Radiochemical uranium determination

Radiochemical processing of the samples was performed to determine the activity concentrations and ratios of uranium isotopes (Sill and Williams 1981, Sill 1987). The water samples were concentrated by applying iron scavenging. The precipitate was dissolved in concentrated HCl. Uranium was separated from other radionuclides by ion exchange method (using Dowex 1x8, 50/100 mesh). Uranium was co-precipitating with CeF3 for the alpha counting (AlfaAnalyst, Canberra). Minimum detectable activity was 2.0 mBq for one litre water sample and 180 hour counting. ^{232}U was used as a chemical yield tracer.

Spectrum unfolding using Adam

Adam is a spectrum analysis tool for alpha spectrometry (Toivonen et al., 2009). The peaks of a nuclide are treated as a family of individual peaks with energies and yields obtained from a nuclide library. Because of the omission of radioelement separation there may be a number of peaks of different radionuclides present in the spectra.

In the fitting the peaks of each radionuclide are treated as a group which enables efficient spectrum unfolding even in the case of complex alpha spectra. The shape of the peaks of the nuclides used in the analysis was same. Convolution of a Gaussian distribution with low-energy side double exponential tail is used in the fitting. The energy region considered in the analyses was 2–8 MeV. In the measurements the number of the channels was 1024 (5.9 keV per channel).

Uncertainty estimation

The activity uncertainty of the ^{232}U tracer used in the radiochemical analyses was 5%. The uncertainty for the geometrical detection efficicency used in the simple method was estimated to be 10%. The uncertainty (Δf) of the $^{234}U/^{238}U$ ratio between the simple method and the radiochemical determination was calculated according to the following formula

$$\Delta f = \sqrt{\left(\frac{\Delta x}{y}\right)^2 + \left(\frac{x \times \Delta y}{y^2}\right)^2} \times f\,, \qquad (1)$$

where Δx is the uncertainty of $^{234}U/^{238}U$ ratio in the simple method, Δy is the uncertainty of $^{234}U/^{238}U$ ratio in the radiochemical determination and f is the abovementioned isotopic ratio between the simple method and the radiochemical determination.

Results

The correlation between the activity concentrations obtained by the simple method to those determined by liquid scintillation method and radiochemical analyses was good (Fig. 2). The calculated correlation coefficients between the rapid method and gross alpha determination and radiochemical analyses were 0.98 and 0.99, respectively. For the method developed here the adsorption onto the walls of the evaporation container was on the average 8% (range 5–14%).

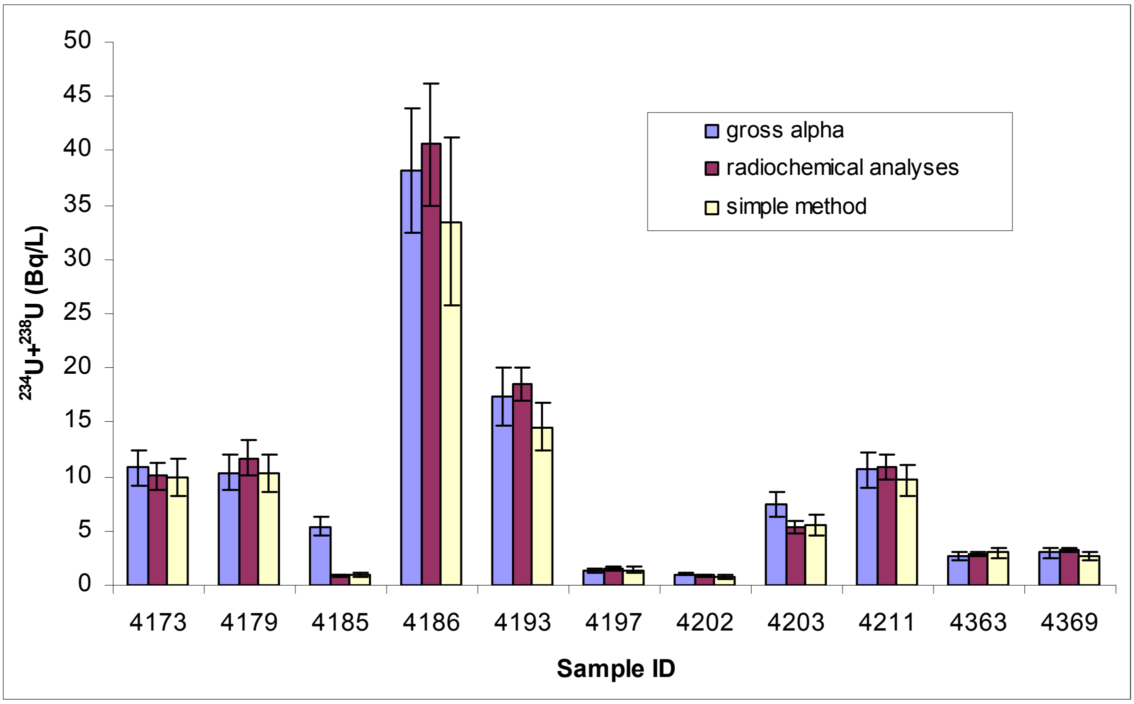

Fig. 2. Total activity concentration of uranium (^{234}U + ^{238}U) obtained from gross alpha determination, radiochemical sample processing and the rapid evaporation method developed here. Numbers in the x-axis refer to the sample identification number.

U isotope ratios obtained by the rapid method developed here and those obtained by radiochemical analyses were close to each other (Fig. 3). The difference was 10% on the average.

The presence on ^{226}Ra (main peak alpha energy 4784 keV is only 10 keV larger than that of ^{234}U) must be taken into account to unfold the spectra correctly. Among the 11 samples considered here there were three samples that contained notable amounts of ^{226}Ra (Fig. 4). Radium contribution was accounted for in the present analysis.

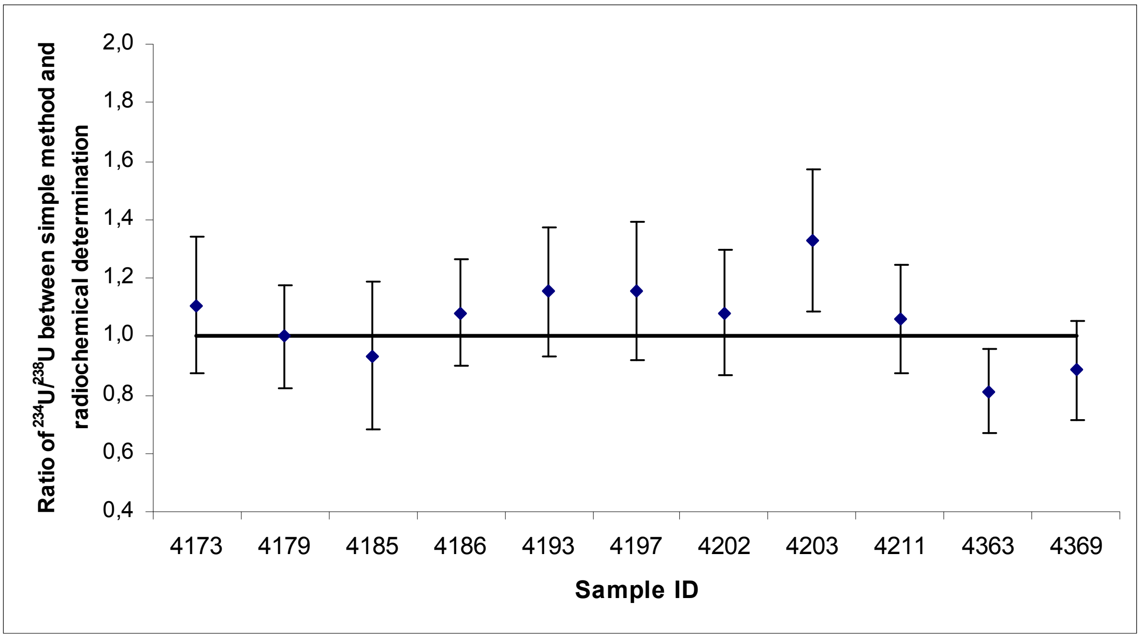

Fig. 3. ^{234}U/^{238}U activity ratio for different samples determined by alpha spectrometry with radiochemical sample processing relative to that of the simple analysis method developed here. Error bars refer to 2σ uncertainty and the numbers in the x-axis refer to the sample identification code.

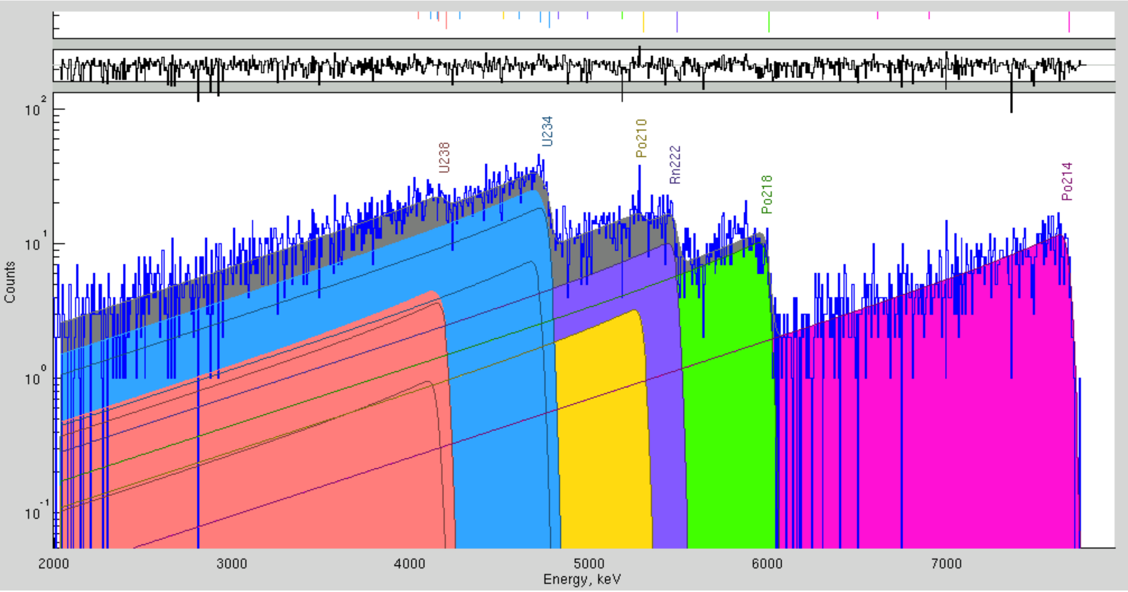

Fig. 4. Alpha particle energy spectrum from the sample 4185 (see Figs. 3 and 4) containing notable amounts of radon and its progeny (^{218}Po, ^{214}Po and ^{210}Po). Here (in this figure), the contribution of ^{226}Ra is not taken into account in the fitting.

Discussion and conclusions

Simplified sample manipulation with sophisticated analysis methods is advantageous when obtaining the results rapidly is of importance. However, straightforward water evaporation cannot entirely replace the radiochemical sample processing. This is because peak broadening in the alpha spectrum, caused by the absorption of the alpha particles in the evaporation residues, makes spectrum analysis challenging and, thus, influences the identification of the radionuclides. Too thick residues may even prohibit the analysis and since no radioelement separation is performed for the source, unequivocal nuclide identification may sometimes be impossible.

Despite of the drawbacks mentioned above the method developed here has many advantages: The sample manipulation is simpler, more rapid and inexpensive compared with radiochemical processing. If the samples could be counted without major sample manipulation (or using only minor processing) it gives the possibility to redirect resources in a laboratory towards sophisticated data analysis. This does not require expensive investment for the laboratory facilities. In addition, the information from several alpha-particle emitting radionuclides can be obtained in only one measurement.

In the present paper it was shown that simple and rapid water sample processing gave comparable results to those obtained with radiochemical processing and gross alpha analysis. The method gave information on all alpha-particle emitting radionuclides present in a water sample. Although the spectrum analysis software Adam is currently under evaluation it worked well and was capable to unfold complex alpha spectra reliably. Using the method presented here the sample preparation is simple and availability of the results is fast.

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The progress in Minimizing Activity and Dose with Enhanced Image Quality by Radiopharmaceutical Administration (MADEIRA) project

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Abstract

The introduction of new and more beneficial diagnostic procedures combining CT and PET or SPECT is related to a rapid growth of medical exposures. The European collaborative project MADEIRA (Minimizing Activity and Dose with Enhanced Image Quality by Radiopharmaceutical Administration) within the 7th EURATOM FP aims to improve 3D nuclear medical imaging in terms of increase of spatial and temporal resolution and also of reduction of the radiation exposure. The project is organized in 5 work packages, which are identified as: WP1 assessment of clinical data, WP2 PET magnifier probe development, WP3 physics-based image processing; WP4 biokinetic and dosimetric modelling; WP5 training and dissemination activities. In particular in WP1 quantitative biokinetic data were collected in patients after administration of 18F-choline and those data were used to develop detailed compartmental models and to evaluate the internal dose to the patient. Moreover, the biokinetic models were used as a basis to define more efficient protocols for data collection (WP4). The collected tomographic images were used to test different reconstruction algorithms (WP3) including some newly developed ones for which a patent is pending. In addition a special phantom for checking image quality in nuclear medicine imaging was designed and also for this a patent is pending. WP2 deals with the construction of a PET insert probe. The different modules of the probe have been developed and their performances monitored, and the probe prototype is going to be tested in clinical conditions with the use of the developed reconstruction methods. Finally, training and dissemination activities include the organization of two training courses (Milano, November 2008; Malmö November 2009), and the Malmö Conference on Medical Imaging (June 2009) as well as the tutoring of graduate and doctoral students and young post-doctoral researchers.

^{177g}Lu produced with high specific activity by deuteron irradiation for metabolic radiotherapy

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Abstract

Nowadays metabolic radiotherapy is in constant progress and there is a range of new radionuclides (RNs) and radiopharmaceutical compounds on the market. Although the production of RNs is improved, often it is not possible to achieve high AS, even in NCA conditions, and sometimes long-lived radionuclidic impurities are co-produced.

^{177g}Lu (t1/2 = 6.734 d, β-emission 100 %, $E_{\beta- \, max}$ = 489.3 keV and E_{γ}= 208.4 keV) is one of the therapeutic γ emitting RNs, which is starting to find several applications in nuclear medicine, especially for metabolic radiotherapy of cancer and radioimmuno-therapy, thanks to its favorable decay characteristics (half-life of the order of few days, relative low energy of negatrons and gamma emission suitable for detection). Presently, the production of ^{177}Lu is carried out mainly in thermal nuclear reactor either in carried added (CA) form by direct neutron capture reaction ^{176}Lu(n,γ)^{177}Lu on enriched ^{176}Lu target, or by neutron capture reaction on enriched ^{176}Yb target, followed by negatron decay. This second method produces a high specific activity (AS) no-carrier-added (NCA) radionuclide, since ^{177}Yb decays to the ground state ^{177g}Lu only.

We started to study an alternative method for the production of ^{177}Lu by the deuteron activation of natural or enriched Yb targets. In order to optimize the irradiation conditions, the thin target yields were experimentally determinated in the energy range from the threshold up to 19 MeV. The thick target yields were measured in order to compare with the production yield by neutron activation route. The contamination from the long-lived metastable level was evaluated too. Moreover a suitable radiochemical separation of Lu radioisotopes from Yb target and from the contaminants was pointed out.

High-grade radiochemical analyses act as a basis for good assessment in radiation protection

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Abstract
High-level risk assessment is mostly based on results of qualitative and accurate radioactivity measurements. Accreditation is one of the most effective tools to maintain high quality in the laboratory. This means, for example, proficiency of the personnel, documented working procedures, validated methods, participating in intercomparisons, maintenance of measuring equipment, workable laboratory information management system and continuous improvement of the working practices in the laboratory. By maintaining quality system and accredited determination methods STUK assures high-grade quality in radiochemical analyses that are used as basic data of doses in risk assessment in radiation protection work.

Introduction
High-level risk assessment is mostly based on results of qualitative and accurate radioactivity measurements. In future, the need for knowledge of low doses increases and raises requirements for research infrastructure and quality management in laboratory. It means documented working practices, traceability of the results, method validation, active participating in intercomparisons, good professional skills of personnel, stability of measurement equipment, proper data management, uncertainty estimations, plan of action in case of errors and continuous improvement of working practices. It also creates the need for accredited determination methods.

At STUK, development of a uniform and modern quality system started in 1997. Accreditation of laboratory processes was awarded by FINAS (The Finnish Accreditation Service) in 1999 and was renewed twice in 2003 and 2007 according to the European Standard EN ISO/IEC 17025. The accreditation field is “Test of radiation safety and related environmental sampling”. Fields of testing in radiochemical analyses are both artificial and natural radionuclides. Other fields of testing are gammaspectrometric analyses, direct measurements of people, airborne ^{222}Rn concentrations, chromosome analyses, sampling for control of radioactivity and method for local laboratories (NaI(Tl) detector).

STUK’s radiochemical analyses are carried out at two different locations, in the main office in Helsinki, and in the regional laboratory in Rovaniemi, in northern Finland. The facilities comprise a total of 50 laboratory rooms for pre-treatment of samples, radiochemical and biological treatment and analysis of samples as well as measuring rooms.

Radiochemical analyses are made from foodstuffs, all kinds of environmental samples, swipe and urine samples. ^{3}H and ^{222}Rn are determined without radiochemical separation using direct measurement by liquid scintillation spectrometry. Ion chromatographic extraction is used in ^{89,90}Sr and ^{210}Pb separation and ion exchange in ^{239,240}Pu, ^{238}Pu, ^{241}Am, ^{234}U, ^{235}U, and ^{238}U separation. ^{210}Po is determined via spontaneous deposition onto silver disk and measured using alpha spectrometry. Low-background proportional counter is used with ^{89,90}Sr measurements, liquid scintillation spectrometry with ^{90}Sr and ^{210}Pb measurements and alpha spectrometry with measurement of actinides. The radiochemical analyses are validated and their quality is maintained by participating regularly in intercomparison exercises and professional tests. All data from samples and measurements are stored either in LIMS or LINSSI databases in order to maintain traceability of the results.

By maintaining quality system and accredited determination methods STUK assures high-grade quality in radiochemical analyses that are used as basic data of doses in risk assessment in radiation protection work.

Analyses methods

Radiochemical determination demands nearly always digestion of the sample and elemental separation. Elemental separation is done because of the interference of other elements. Radiochemical analyses are made from all samples that can be digested into liquid form. Sample types analysed are environmental samples (e.g. sediments, soil, mushroom, various berries, lichen, moss, water), drinking water and food samples, excretion samples (urine), swipe samples and industrial products. Summary for the detection limits for radionuclides ^{90}Sr, ^{239,240}Pu, ^{3}H, ^{222}Rn, ^{226}Ra, ^{228}Ra, ^{234,235,238}U, ^{210}Pb, ^{210}Po, gross alpha and gross beta are given in Table 1.

Artificial radionuclides

Strontium (^{89,90}Sr)

Strontium separation from interfering nuclides is performed with extraction chromatographic resin, Sr resin (Triskem International) and measured either with low-background proportional counter (Risø or LB770 Berthold) or liquid scintillation counter (Quantulus). Proportional counter is used if both ^{89}Sr and ^{90}Sr are to be measured, liquid scintillation counter is used if only ^{90}Sr is to be measured. In proportional counter method the strontium is precipitated after Sr separation as a carbonate onto filter paper, and ^{89}Sr and ^{90}Sr together with its ingrowing daughter nuclide ^{90}Y are measured as soon as possible. After a minimum of 18 days, the ingrowth daughter nuclide ^{90}Y is separated together with stable Y from Sr and measured few times in a couple days. ^{90}Sr is calculated from the ^{90}Y measurements and ^{89}Sr from the first measurement by subtracting the ^{90}Sr contribution. In liquid scintillation method the ^{90}Sr is measured together with its ingrowth daughter ^{90}Y after a minimum of 18 days from the Sr separation (Fig. 1 (B)). The chemical yield is determined with inactive Sr and Y carriers. The efficiency in proportional counting is typically 40-50% and in liquid scintillation counting ~200%.

Plutonium ($^{239,240}Pu$)

The routine method for separation of plutonium is based on ion exchange with Dowex 1x4 anion exchange resin. Radioactive tracer ^{242}Pu is added and the sample is brought into solution either by wet ashing or microwave digestion. The sample is introduced into ion exchange column (Dowex 1x4, 50-100 mesh, in NO_3^- -form) in 8 M HNO_3, washed with 8 M HNO_3 and concentrated HCl, and the Pu fraction is eluted with conc. HCl + 1 M NH_4I solution. The plutonium is electrodeposited on stainless steel disc and counted with an alpha spectrometer. The chemical yield is determined with the radioactive tracer ^{242}Pu. The counting efficiency is approximately 35-40%.

Tritium (^3H)

Tritium measurements are made from water, urine and swipe samples. Water and urine samples are distilled for purification. Aliquote from a distilled sample is taken and UltimaGold uLLT scintillation cocktail is added to the 20-ml plastic scintillation vials. In Figure 1, the tritium spectrum of a water sample is presented. Swipe samples are measured with no pre-treatment. Pori-Vuoltee-groundwater is used for a background sample. Counting efficiency is determined measuring also a ^3H standard sample in every sample sequence. Activity concentration of tritium is determined with a low-background liquid scintillation spectrometer 1220 Quantulus.

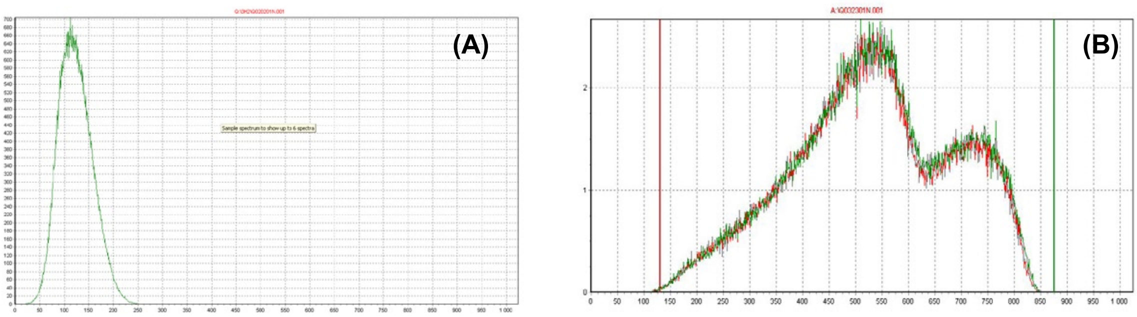

Fig. 1. Tritium spectrum of water sample (A). Strontium-90 spectra together with its ingrowth daughter ^{90}Y measured with Quantulus 1220 liquid scintillation spectrometer (B).

Natural radionuclides

Gross alpha and beta

The gross alpha and beta measurements are performed mainly in order to screen for the level of occurrence of the long-lived alpha-emitting (^{238}U and ^{234}U, ^{226}Ra and ^{210}Po) and beta-emitting (^{40}K, ^{210}Pb and ^{228}Ra) radionuclides in water. The sample (38ml) is prepared by evaporating water into dryness with a freeze-dryer in a teflon-coated polyethylene vial. The residue is dissolved in small amount of HCl acid and then scintillation cocktail Ultima Gold AB is added. The sample is counted one month after the sample preparation. During that time ^{226}Ra attains equilibrium with ^{222}Rn and its short-lived daughters. Gross alpha and beta are determined with a low-background liquid scintillation spectrometer 1220 Quantulus. An example of the gross alpha spectra of a drilled well water sample is given in Figure 2(A).

Radon (^{222}Rn)

Radon is measured in a homogeneous solution with a liquid scintillation spectrometer 1414 Guardian. The sample is prepared by adding 10 ml of water into a glass vial (equipped with a cap containing an aluminium foil) pre-filled with liquid scintillation cocktail Ultima Gold XR. The concentration of ^{222}Rn is calculated from the alpha spectrum in the window which covers the most part of the alpha peaks. The alpha counting efficiency of radon in the selected alpha window is 268%±3%.

Radium (226Ra) and Radium (228Ra)

^{226}Ra activity is calculated from the gross alpha spectrum on the basis of the counts measured in window set on the area of the ^{214}Po peak. This will give quite accurate results to ^{226}Ra because no other natural or artificial radionuclide has alpha emissions in the same energy range >7.5 MeV. The counting efficiency of ^{214}Po (and thus of ^{226}Ra) in the selected window is 86±3%. ^{228}Ra is determined via its daughter nuclide ^{228}Ac using gamma spectrometric measurement.

Uranium (^{238}U, ^{234}U and ^{235}U) and Thorium (^{232}Th, ^{230}Th and ^{228}Th)

Uranium and thorium are analysed by a radiochemical method. It allows the accurate determination of each uranium and thorium isotope separately. The water sample is concentrated by applying iron scavenging and the iron precipitate is dissolved in concentrated HCl. The other environmental samples are digested by either wet ashing or microwave digestion. Uranium and thorium are then separated from other radionuclides by ion exchange method (by using Dowex 1x8, 50/100 mesh). Uranium and thorium are coprecipitated with CeF_3 for the alpha measurement. The sample is counted with AlfaAnalyst spectrometer (Canberra). ^{232}U is used as a chemical yield tracer in the uranium analysis and ^{229}Th in the thorium analysis.

Lead (^{210}Pb) and polonium (^{210}Po)

^{210}Pb and ^{210}Po concentrations were determined using spontaneous deposition of ^{210}Po on silver disk and alpha spectrometric measurement (AlphaAnalyst) of the ^{210}Po activity (Fig. 2 (B)). The solution, remaining from the ^{210}Po deposition, is stored at least for 200 days to allow the in-growth of ^{210}Po which is a daughter product of ^{210}Pb. The second ^{210}Po deposition is then carried out and its ^{210}Po activity is counted. The final ^{210}Po result is calculated using the results from these two depositions. The ^{210}Pb result is calculated from the second ^{210}Po deposition. ^{209}Po and ^{208}Po are used as chemical yield tracers. Lead can also be determined using extraction chromatography. ^{210}Pb is separated from other radionuclides by Triskem´s Sr-resin. The sample is counted after a 30-day period during which ^{210}Bi attains nearly 100% equilibrium with ^{210}Pb. The result is then calculated from ^{210}Bi and ^{210}Pb together. Inactive lead is used for yield determination.

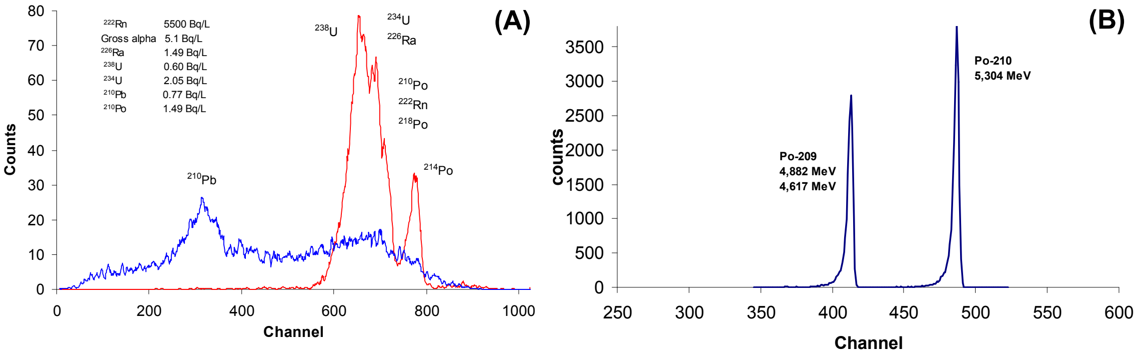

Fig. 2. The alpha and beta spectra of the Quantulus from a drinking water sample with different radionuclide contents (A). Alpha spectrum of ^{209}Po and ^{210}Po isotopes measured with Alpha Analyst (Canberra). ^{209}Po is used as tracer (B).

Table 1. The minimum detection limits for radionuclides ^{90}Sr, ^{239,240}Pu, ^{3}H, ^{222}Rn, ^{226}Ra, ^{228}Ra, ^{234,235,238}U, ^{210}Pb, ^{210}Po, gross alpha and gross beta at 95% confidence level. The amount of samples used in analyses depends on material. In this table MDA is calculated for the sample amount given in the table.

Nuclide	Equipment used in the measurement	Examples from amount of sample used in analysis (L, kg)	Counting time (h)	MDA (Bq/sample)
^{90}Sr	Quantulus 1220, proportional counter	30 L	15	0.005
^{239,240}Pu	AlphaAnalyst (Canberra)	100 L	60	0.0005
^{3}H	Quantulus 1220	0.008 L	10	0.008
^{222}Rn	Guardian 1414	0.01 L	1	0.0017
^{226}Ra	Guardian 1414, Quantulus 1220	0.04 L	3	0.00004
^{228}Ra	HPGe detector (Ortec, Canberra)	2 L	16	0.14
^{238}U	AlphaAnalyst (Canberra)	1 L, kg	180	0.002
^{235}U	AlphaAnalyst (Canberra)	1 L, kg	180	0.002
^{234}U	AlphaAnalyst (Canberra)	1 L, kg	180	0.002
^{210}Po	AlphaAnalyst (Canberra)	1 L, kg	180	0.002
^{210}Pb	AlphaAnalyst (Canberra)	1 L, kg	180	0.002
^{210}Pb	Guardian 1414, Quantulus 1220	1 L, kg	3	0.02
Gross alpha	Guardian 1414, Quantulus 1220	0.04 L	3	0.00012
Gross beta	Quantulus 1220	0.04 L	3	0.008

Quality control in radiochemical laboratory

Good laboratory practices carried out at STUK in analytical work for radiochemical and gammaspectrometric analyses are described in detail in publication Ikäheimonen et. al 2006. A short summary of these practices is described here. Concerning radioactivity determinations there are only few international standards available. Therefore, almost always the methods used are laboratory-developed methods.

Proficiency of personnel and working practices

Proficiency of the personnel must be on a level where everybody knows what he is doing and why. Skilful laboratory personnel know what things can affect the result. Working with environmental samples with low radioactivity levels implies that the analysts must be qualified and aware of good laboratory practices. Especially, attention needs to be paid on contamination. This means that glassware and equipment are classified for different purposes according to the activity concentration of the sample. Also separate rooms are organized for pre-treatment of the sample, analyses and measuring equipment. In a laboratory with good working practices all instructions are found in written form in laboratory manuals.

Measurement rooms

STUK has four laboratory rooms for low-background alpha and beta measurements. The rooms are constructed of a special concrete containing an extremely low amount of natural radioactive elements and are equipped with a special air ventilation system. Measuring equipment is separated according to the measured radionuclide and activity concentration of the sample. Monitoring of temperature and humidity in the measurement rooms is arranged and all data is stored for further evaluation. The measurement rooms are equipped with uninterruptible power supplies (UPS) and outside backup generator to assure continuous power supply.

Equipment stability

STUK has a large number of high-quality measurement equipment used for radiochemical analyses. The equipment are three alpha spectrometer systems from Canberra (AlphaAnalyst), three Guardian 1414 liquid scintillation spectrometers, three Quantulus 1220 liquid scintillation spectrometers and three proportional counters (Fig. 3).

Confidence in the performance of equipment is maintained by regular calibrations and service, routine checks and control charts. Additionally, for good flow of information regular instrument meetings, four times per year, are arranged. In the meetings, all changes in operations of the equipment are documented. Equipment is operated only by authorized personnel. Up-to-date instructions on the use and maintenance of equipment are found in equipment manuals. All significant information from software and item of equipment is documented.

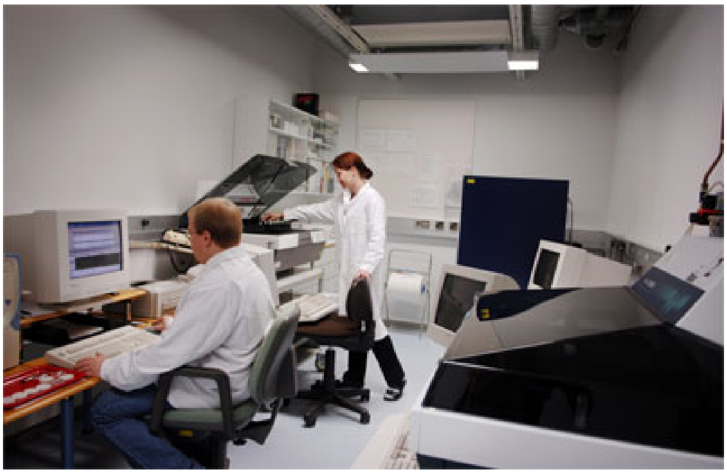

Fig. 3. Activity concentrations of tritium, Rn-222, C-14 and Sr-90 are determined using Guardian 1414 liquid scintillation spectrometer.

Regular calibrations

Standard 17025 requires that the calibration programmes are established for the equipment having significant effect on the results. At STUK the equipment used for radioactivity measurements are calibrated regularly. For all the analysis methods the interval for calibration is determined. For example, ^{222}Rn determination is calibrated between 3 to 5 years and ^{89,90}Sr between 5 to 7 years. Calibration procedure and the results are documented as closed circuit technical documents (TecDocs). With methods where internal standard is used as tracer, no regular calibrations of equipment are needed e.g. Pu. In these cases the activity of the tracer used is controlled regularly. The results are documented as a TecDoc in order to attain traceability of the results.

Control measurements

The laboratory needs to maintain confidence on its calibration status. This is done by making control measurements using blank samples, background and reference samples and internal measurements of the equipment (alpha spectrometer) indicating that calibrations are valid during activity measurement. All measurements are done regularly and the results are documented in x-cards (Fig. 4).

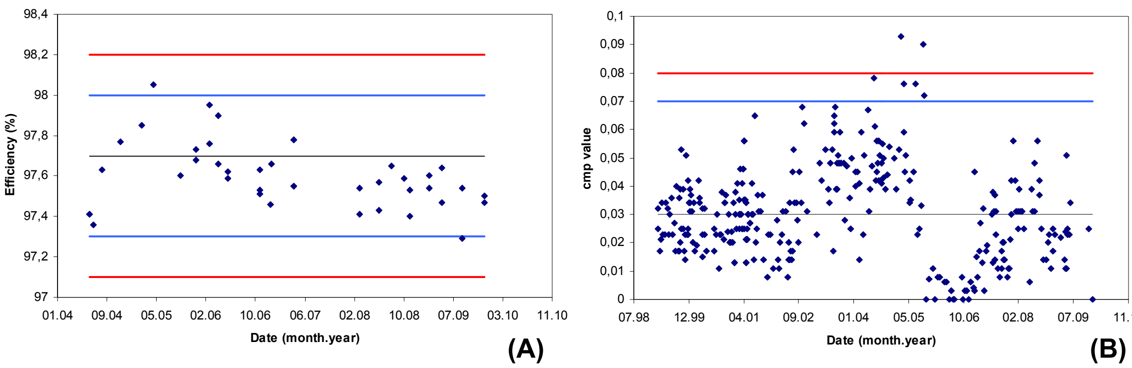

Fig. 4. Regular C-14 measurements of the reference sample (A) and background measurements from gross alpha determination (B) using Quantulus 1220 liquid scintillation spectrometer. Blue line presents the alarm level and red line the action level.

Regular maintenances

Confidence in the performance of equipment is maintained by regular service which is done by authorized personnel. Typically, measurement equipment is maintained once a year.

Intercomparisons

The laboratory takes regularly part in internal and international intercomparison exercises or proficiency tests. These are good tools to assure high quality of analyses and also excellent tools for analysis development. STUK participates in 5 to 10 international intercomparisons per year (Fig. 5). Additionally, internal intercomparisons are arranged to ensure convergent results and personnel competence.

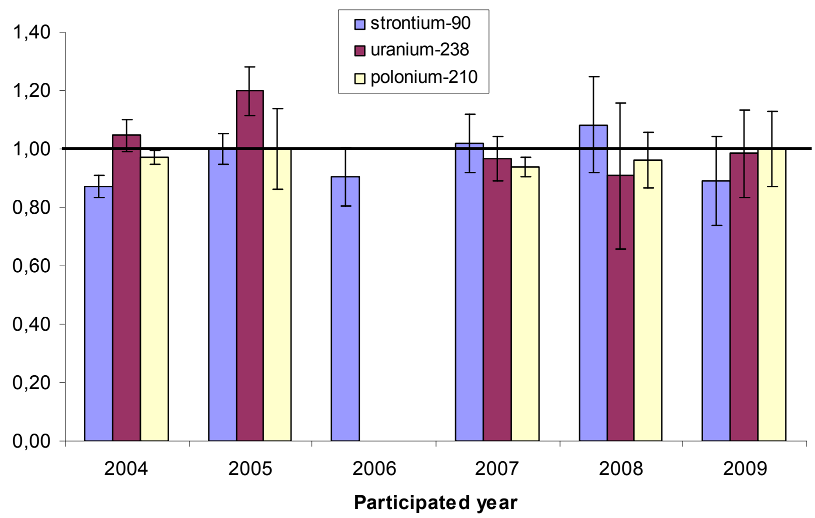

Fig. 5. Intercomparison results for ^{90}Sr and ^{238}U in 2004 – 2009. Analyzed samples were urine, lake water, sea water, drinking water, spinach and milk power.

Laboratory information management system

A uniform Laboratory Information Management Systems, LIMS and LINSSI, are used for data management, follow-up samples and chemicals, and analysis flow control. The system is password-protected and access is given only for authorized persons. The system helps the management of laboratory work, data management, traceability of the results, reporting and quality assurance in the laboratory. In the future, the same system will manage also the equipment used with radioactivity determinations.

Development of laboratory work and analyses methods

The most important issues in accredited laboratory are continuous improvement of laboratory work and analyses methods as well as the whole quality management system. New analyses methods are first tested, then validated and taken in use if they are found eligible for the analysis. Validation procedure is planned and all work tasks are documented as closed circuit technical documents (TecDocs). The laboratory work is enhanced by self-assessments, external and internal audits, various customer surveys and regular meetings with customers.

Conclusions

Radiochemical analyses are demanding analyses and demand proficiency of the whole personnel, well documented working procedures and properly controlled equipment. Validation of a new analysis method is a long procedure and requires a careful plan before the actual work can begin. In routine work, attention is needed on regular control measurements and contamination in order to maintain high quality of the analyses and measurement equipment. Workable laboratory information management systems provide an effective tool for evaluation of control measurements.

Accreditation is an excellent way to bring clarity in the routine work and regular external assessors improve the continuous maintenance of laboratory work. Laboratory work can be continuously improved by self-assessments, internal audits and customer surveys or regular meetings with customers.

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Condition of liver, and iron and plutonium content

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Abstract
Plutonium and iron share a common metabolism in transportation and accumulation in human body. This was also supported by this study, where the accumulation of plutonium and iron in liver was investigated. The $^{239,240}Pu$ and iron contents in liver correlated positively (Rs = 0.378) in healthy livers, but not in livers having pathological changes. In contrast to iron content , the plutonium content of fatty degenerated or cirrhotic livers was significantly lower than that in normal livers. This difference suggests that plutonium and iron are not similarly accumulated or excreted in fatty degenerated and cirrhotic livers. The plutonium contents in lung and bone of these persons did not differ from the average values of the same age group, which may suggest that plutonium is excreted from the liver cells mainly by feces, instead of having been recirculated.

$^{240}Pu/^{239}Pu$ ratio in peat, lichen and air filter samples contaminated by global nuclear test fallout, the Chernobyl accident and other nuclear events

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Abstract

The $^{238}Pu/^{239+240}Pu$ activity ratio by alpha spectrometry is a useful method for separating the influences from nuclear weapons testing and the Chernobyl fallout. However, $^{238}Pu/^{239+240}Pu$ activity ratios in global nuclear test fallout and weapons-grade Pu are nowadays so similar, that it is difficult to reliably distinguish the two Pu sources by using this activity ratio. Besides, the activity concentration of ^{238}Pu is usually relatively low compared to ^{239}Pu and ^{240}Pu in environmental samples, leading to high uncertainties in $^{238}Pu/^{239+240}Pu$ activity ratio. As a complementary method for alpha spectrometry, ICP-MS provides the tool for identifying the origin of the Pu from global fallout or weapons-grade Pu, since the $^{240}Pu/^{239}Pu$ mass ratio is clearly different in Pu from nuclear weapons testing and weapons-grade Pu.

$^{238}Pu/^{239+240}Pu$ and $^{241}Pu/^{239+240}Pu$ activity ratios have been determined for peat and lichen samples collected in Finland immediately after the Chernobyl accident in 1986, air filter samples collected in Sodankylä in 1963 and air filter samples collected in Astana, Kazakhstan, in 2002. The Pu activity ratios indicated contamination from global nuclear test fallout in air filters from Sodankylä, and in peat and lichen samples Pu has been deposited from global fallout and the Chernobyl accident. $^{238}Pu/^{239+240}Pu$ activity ratio in air filters from Astana corresponded with the ratio in nuclear fuel of high burnup. In this study, a more exact estimation about the origin of Pu in these different types of samples from Finland and Kazakhstan was sought by determining their $^{240}Pu/^{239}Pu$ mass ratios.

The membrane filters used for alpha counting were decomposed and the resulting solutions containing Pu were purified with extraction chromatography. Mass concentration of ^{239}Pu and ^{240}Pu in the samples was determined by quadrupole ICP-MS. Results from $^{240}Pu/^{239}Pu$ mass ratio determinations for the investigated samples will be presented in detail in the poster.

Sensitivity of the standard and Fpg-modified comet assay for the estimation of DNA damage in peripheral blood lymphocytes after exposure to gamma rays

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Abstract

The comet assay is a rapid and sensitive technique for measuring DNA damage. This assay detects single and double stranded breaks, sites of incomplete repair, alkali labile sites, DNA-DNA and DNA-protein cross-links. In addition, particular enzymes such as formamidopyrimidine glycosilase (Fpg) can be used for detection of oxidative damage at the level of DNA molecule by cleavage of 8-oxodG, FaPyGua, FaPyAde and other ring-opened purines caused by reactive oxygen species (ROS). The aim of this study was to test the sensitivity of both standard alkaline and Fpg-modified comet assay and to detect the type of DNA damage caused by gamma rays. In that manner, human lymphocytes were exposed to gamma radiation doses of 0.1 Gy and 4 Gy *in vitro*. With the standard assay increase in DNA damage was noticed for both exposure doses but it was significant only at higher dose of 4 Gy whereas at lower dose there were no statistically significant increase in neither of the standard comet assay parameters. Fpg-modified protocol showed significant increase in all the parameters measured for both exposure doses indicating that the modified version is capable of detecting wider scale of DNA damage induced by gamma radiation. In addition, with modified protocol it is possible to detect ROS mediated DNA damage, thus significant increase in modified comet parameters in comparison to the standard ones suggests that gamma radiation did induce oxidative damage in DNA molecule. Correlation between different protocols of the comet assay suggests that Fpg-modified version is more sensitive to gamma radiation by virtue of measuring oxidative DNA damage in addition to the basal DNA strand breaks. Results obtained lead to the conclusion that gamma rays affects DNA molecule by ROS that are the most frequent product of gamma radiation. Additionally, human lymphocytes proved to be sensitive to ionizing radiation depending on the radiation dose and are suitable biomarkers for this type of research.

Introduction

Ionizing radiation is nowadays omnipresent in human lives because of the increasing development of technology, industry and medicine. People live and work near nuclear facilities and places of testing nuclear weapon and also use nuclear (gamma) radiation in medical purposes. In each case, human organism bears the strong genotoxic effect making the structure of the DNA molecule unstable and causing the genesis of many changes in it (Mettler and Voelz 2002). The damage of the genetic material caused by ionizing radiation is one of the best precondition indicators for development of malign diseases such as breast, gall-bladder or thyroid cancer (Cooke and Evans 2006). Gamma radiation affects the DNA structure directly, causing strand breaks, or indirectly causing cleavage of the water molecules and damaging the DNA molecule by reactive oxygen species (ROS) (Ito et al. 2007). These changes can be examined in human lymphocytes using different cytogenetic techniques.

The comet assay is a rapid, simple, visual and sensitive technique for measuring and analyzing DNA damage at the single cell level (Fairbairn et al. 1995, Faust et al. 2003, Møller 2006, Olive 1999, Singh 1988). This technique can be performed at the level of individual cells and requires a small number of cells per sample. Single cells can be used *in vivo* and *in vitro* as well as in biomonitoring of population exposed to radiation or chemical mutagens (Faust et al. 2003, Garaj-Vrhovac et al. 2002). The comet assay detects single and double stranded breaks at the level of DNA molecule, sites of incomplete repair, alkali labile sites, DNA-DNA and DNA-protein cross-links. Besides, the comet assay can be used for detection of the level of the DNA fragmentation in apoptosis (Collins 2004, Husseini et al. 2005, Olive 1999, Piperakis et al. 1999). Particular enzymes such as formamidopyrimidine glycosilase can be used for detection of oxidative damage at the level of the DNA molecule caused by ROS (Collins et al. 1993, Speit et al. 2003).

The aim of this study was to detect the type of DNA damage caused by ionizing radiation. For that purpose human peripheral blood lymphocytes were exposed to gamma radiation doses of 0.1 Gy and 4 Gy *in vitro.* Considering that, two forms of the comet assay, standard alkaline and Fpg-modified were used.

Material and methods

Blood sampling

The study was performed on peripheral blood samples obtained from a healthy female non-smoking donor (age 24 years). The donor was not exposed to ionizing radiation, vaccinated or used medicals for a year before blood sampling. Whole venous blood was collected under sterile conditions in heparinised vacutainer tubes (Becton Dickinson, Franklin Lakes, NJ, USA) containing lithium heparin as anticoagulant. After collection, blood was divided into a large number of samples. All experiments were conducted on peripheral blood lymphocytes cultivated at 37ºC in an atmosphere of 5% CO_2 in air.

Exposure conditions

The whole blood samples were irradiated with gamma radiation on the ice. As a source of radiation Gammacell 220 (Ruđer Bošković Institute, Zagreb, Croatia) was used. Vacutainers (volume 5 cm^3) containing blood samples were exposed to radiation doses defined as doses in the water, but irradiation was performed in the air. Samples were irradiated with radiation doses of 0.1 Gy and 4 Gy that is equal to radiation periods of 32.3 s and 23 min and 9 s at the temperature of 21°C. Significance of the absorbed dose was 3% (Miljanic et al. 1994). To get homogenate samples, they were stirred after irradiation, cooled to 4°C, transported to the laboratory on ice and processed as quickly as possible.

Alkaline comet assay

The comet assay was carried out under alkaline conditions, basically as described by Singh et al. (1988). Fully frosted slides were covered with 1% normal melting point (NMP) agarose (Sigma, St Louis, Mo, USA). After solidification, the gel was scraped off the slide. The slides were then coated with 0.6% NMP agarose. When this layer had solidified a second layer containing the whole blood sample mixed with 0.5% low melting point (LMP) agarose (Sigma) was placed on the slides. After 10 min of solidification on ice, the slides were covered with 0.5% LMP agarose. The slides were then immersed for 1 h in ice-cold freshly prepared lysis solution [2.5 M NaCl, 100 mM disodium EDTA, 10 mM Tris–HCl, 1% sodium sarcosinate (Sigma), pH 10] with 1% Triton X-100 (Sigma) and 10% dimethyl sulfoxide (Kemika, Zagreb, Croatia) added fresh to lyse cells and allow DNA unfolding. The slides were then placed on a horizontal gel electrophoresis tank, facing the anode. The unit was filled with fresh electrophoresis buffer (300 mM NaOH, 1 mM disodium EDTA, pH 13.0) and the slides were placed in this alkaline buffer for 20 min to allow DNA unwinding and expression of alkali-labile sites. Denaturation and electrophoresis were performed at 4°C under dim light. Electrophoresis was carried out for 20 min at 25 V (300 mA). After electrophoresis the slides were rinsed gently three times with neutralization buffer (0.4 M Tris–HCl, pH 7.5) to remove excess alkali and detergents. Each slide was stained with ethidium bromide (20 µg/ml) and covered with a coverslip. The slides were then stored in sealed boxes at 4°C until analysis.

Fpg-modified comet assay

The analysis of oxidized purines was performed using an Fpg FLARE™ assay kit (Trevigen Inc, Gaithersburg, Maryland, USA) with some modification (Comet Assay interest group website: http://cometassay.com/). Within the kit the manufacturer provided all the reagents used. Fully-frosted microscopic slides were prepared. Each slide was covered with 1% normal melting point (NMP) agarose (Sigma). After solidification, the gel was scraped off the slide. The slides were then coated with 0.6% NMP agarose. A low melting point (LMP) agarose was melted and stabilized in a water bath at 37 °C. For each sample and control, 5 µl of cell homogenate was mixed with 100 µl of LMP agarose (provided with the FLARE™ assay kit) and placed on the slides. After 10 min of solidification on ice, the slides were covered with 0.5% LMP agarose. The slides were then immersed in a pre-chilled lysis solution (provided with the FLARE™ assay kit) and kept in a refrigerator at 2 °C for 60 min. Followed the immersion in the FLARE™ buffer, three times for 15 min. After lysis, the slides were treated with 100 µl of Fpg enzyme (1:500 in REC dilution buffer). The enzyme was diluted right before use. Control slides were treated with 100 µl of REC dilution buffer only. The slides were placed horizontally in a humidity chamber at 37 °C for 30 min.

All slides were then immersed in an alkali solution (0.3 M NaOH, 1 mM Na_2EDTA; pH 12.1) for 40 min. Followed electrophoresis in a pre-chilled alkali solution (0.3 M NaOH, 1 mM Na_2EDTA; pH 12.1) at 1 V/cm for 20 min. After electrophoresis, the slides were rinsed gently three times with neutralization buffer (0.4 M Tris–HCl, pH 7.5) to remove excess alkali and detergents. Each slide was stained with ethidium bromide (20 µg/ml) and covered with a coverslip. Slides were stored at 4°C in sealed boxes until analysis.

Comet capture and analysis

A total of one hundred randomly captured comets from each slide were examined at 250x magnification using an epifluorescence microscope (Zeiss, Oberkochen, Germany) connected through a black and white camera to an image analysis system (Comet Assay II; Perceptive Instruments Ltd., Haverhill, Suffolk, UK). The analysis did not include the edges and damaged parts of the gel as well as debris, superimposed comets, and comets without distinct head (“clouds”, “hedgehogs”, or “ghost cells”). To quantify DNA damage, the following comet parameters were evaluated: tail length, tail intensity (percentage of DNA in tail), and tail moment. Differences in the tail length, tail intensity and tail moment between samples obtained with standard alkaline comet assay and Fpg-modified comet assay were considered as oxidative DNA damage in a single cell.

Statistical analysis

The various parameters measured in the exposed and control groups were evaluated using Statistica 5.0 package (StaSoft, Tulsa, Okla, USA). Each sample was characterized for the extent of DNA damage by considering the mean ± SE (standard error of the mean), median and range of the comet parameters. In order to normalize the distribution and to equalize the variances, a logarithmic transformation of data was applied. Multiple comparisons between groups were done by means of ANOVA on log-transformed data. Post-hoc analysis of differences was done by Scheffé test. The level of statistical significance was set at $P<0.05$.

Results

Results of the standard alkaline comet assay are presented in the Figure 2A. These results showed statistically significant increase of the mean values for all three parameters of the standard comet assay ($P<0.05$) in the sample irradiated with the dose of 4 Gy in contrast to control sample and sample irradiated with the dose of 0.1 Gy. Sample irradiated with the dose of 0.1 Gy showed slightly increased values for all parameters measured but there were no significant differences in compare to the corresponding control. Results of the Fpg-modified comet assay are presented in the Figure 2B. These results showed statistically significant increase in all three parameters of modified comet assay ($P<0.05$) for both irradiation doses in compare to the control sample. Generally, the mean values of all three Fpg-modified comet assay parameters were significantly higher than of the standard comet assay. These findings suggest that the Fpg-modified comet assay is more sensitive to gamma irradiation than the standard alkaline comet assay.

Discussion

Our goal was to test the sensitivity of the standard alkaline and Fpg-modified version of comet assay after exposure to different doses of gamma radiation. This type of radiation is a potent carcinogen mainly due to it potential as oxidative-induced damage agent. It produces variety of primary lesions in DNA such as single and double strand breaks, DNA-DNA and DNA-protein cross-links, alkali-labile sites and damage to purine and pyrimidine bases as well as oxidize bases and abasic sites (Cadet et al. 1999, Đurinec et al. 2006, Garaj-Vrhovac and Zeljezic 2004, Sudprasert et al. 2006).

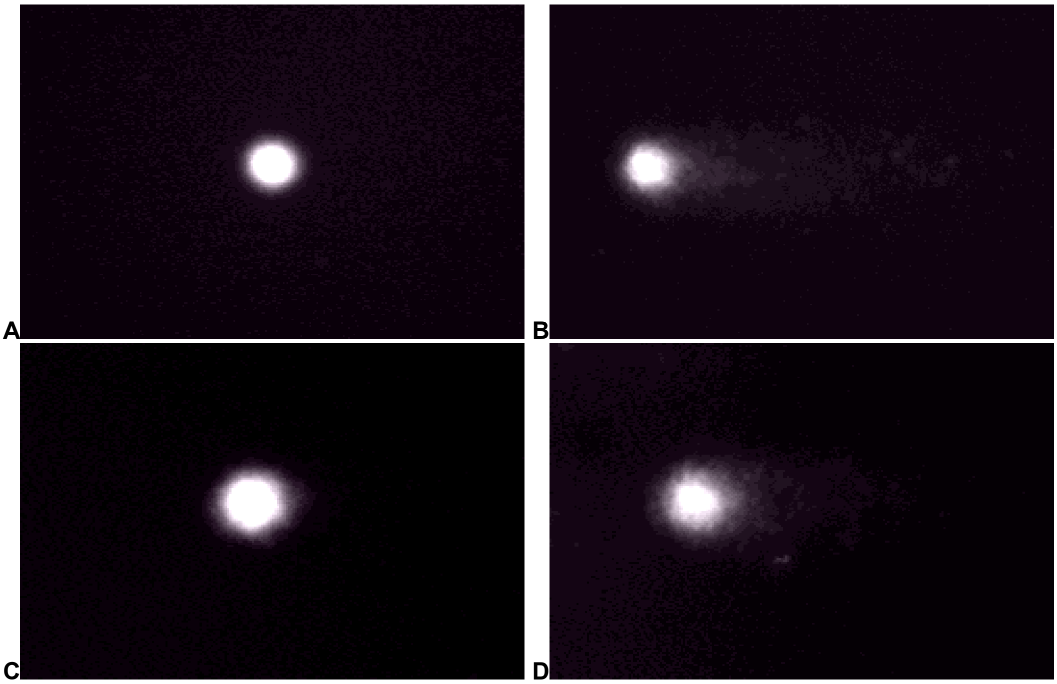

Fig. 1. Comet assay microphotographs represent undamaged lymphocyte from the un-exposed sample (A). Image (B) represents damaged lymphocyte that has comet appearance. Fpg-modified comet assay microphotographs represent undamaged lymphocyte from the un-exposed sample (C). Image (D) represents damaged lymphocyte that has comet appearance. Cells were stained with ethidium bromide. Cells were photographed under the fluorescent microscope using a 60x objective equipped with a 515 nm to 560 nm excitation filters and a 590 nm barrier filter.

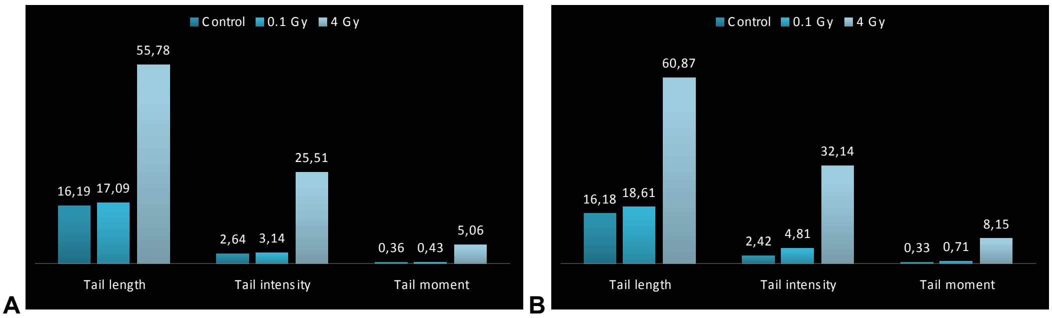

Fig. 2. Parameters of the alkaline comet assay (A) and Fpg-modified comet assay (B) (tail length, tail intensity and tail moment), in human peripheral blood lymphocytes exposed to 0.1 Gy and 4 Gy gamma irradiation.

Both protocols used in this study showed genotoxic effect of gamma rays on DNA molecule of human peripheral blood lymphocytes *in vitro*. With the standard alkaline comet assay increase in DNA damage was noticed in both exposure doses but it was significant only at higher dose of 4 Gy whereas at lower dose there were no statistically significant increase in neither of the standard comet assay parameters. Usage of the Fpg-modified protocol showed significant increase in all the parameters measured at both exposure doses indicating that the modified version of this assay is capable to detect wider scale of DNA damage induced by gamma irradiation. In addition, with modified protocol it is possible to detect ROS mediated DNA damage, thus significant increase in modified comet parameters in comparison to the standard one suggests that gamma radiation did induce oxidative damage in DNA molecule *in vitro*.

Some of the previous studies also showed that the Fpg-modified version of the comet assay is more sensitive for detection of DNA damage than the standard alkaline one (Domijan et al. 2006, Fracasso et al. 2006, Husseini et al. 2005, Garaj-Vrhovac et al. 2002, Rössler et al. 2006). According to that fact, scientists revealed that using the standard alkaline comet assay it is possible to detect damages at radiation doses from 5 cGy to 10 cGy and in some adapted experimental conditions (*e.g.* addition of the Fpg enzyme) it is possible to detect damages even at radiation doses of 0.6 cGy (Dusinska and Collins 1996, Malyapa et al. 1998, Verbeek et al. 2007). The additional reason why the Fpg-modified comet assay is more sensitive than the standard alkaline one is that the Fpg enzyme helps to detect oxidative damage of the DNA molecule by cleavage of 8-oxodG, FaPyGua, FaPyAde and other ring-opened purines (Karakaya et al. 1997).

Conclusions

In this study correlation between different protocols of the comet assay was made suggesting that Fpg-modified version is more sensitive to gamma radiation by virtue of measuring oxidative DNA damage in addition to basal DNA strand breaks. Results obtained lead to the conclusion that gamma radiation affects the DNA molecule by ROS that are most frequent product of the gamma radiation effect. Human peripheral blood lymphocytes proved to be sensitive to ionizing radiation depending on the radiation dose and are suitable biomarkers for this type of research.

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Gamma spectrometric sample measurements at STUK laboratories

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Abstract

In this paper, the facilities and applications for gamma spectrometric sample measurements at the Finnish Radiation and Nuclear Safety Authority (STUK) will be presented. The Department of Research and Environmental Surveillance has 15 HPGe spectrometers operated in specially constructed counting rooms. The software for the spectrum analysis and detector calibration are the key components of the accredited method. Integrated Laboratory Information Management System (LIMS) databases offer tools to manage the large amount of data involved in spectrum analyses.

Applications of STUK's gamma spectrometry include surveillance of environmental radioactivity in Finland, radioecological studies of both natural and artificial radionuclides and contracted services to industry and trade and for various organisations and institutes. STUK's laboratory is also one of the Radionuclide Laboratories certified by the Comprehensive Nuclear Test Ban Treaty Organisation (CTBTO). One of the tasks of the Department is to maintain measurement standards for radioactivity determinations.

Introduction

Finnish Radiation and Nuclear Safety Authority (STUK) has more than 40 years of experience in gamma spectrometric analysis of environmental samples.

Since 1999, the method of gamma spectrometric sample measurements has been accredited according to the Standard EN ISO/IEC 17025. The method of analysis of environmental samples, biological samples and foodstuffs is two-graded: 1) measurements of ^{137}Cs, ^{134}Cs, ^{131}I, ^{40}K and natural decay series of U and Th, 2) advanced analysis of all radionuclides emitting gamma-rays in energy range of 30 keV-2700 keV. The accreditation has been renewed in 2003 and 2007.

More than 4 000 samples are analysed annually in the Department of Research and Environmental Surveillance, and more than 500 additional quality control measurements are performed.

Laboratories and equipment

The sample measurements are carried out in three separate counting rooms; two of them are located in Helsinki and one in Rovaniemi, in the premises of the Regional

Laboratory of Northern Finland. The Helsinki counting rooms are constructed from selected concrete and mortar materials where the abundance of natural radionuclides is low. In addition, the rooms are equipped with special air conditioning in order to decrease background radiation due to indoor radon, and also to prevent contamination of the laboratory during an incidence of fallout. Continuous monitoring of temperature, humidity, air pressure and ambient radon is arranged in order to evaluate changes in detector background and in operation of the measurement equipment. The measurement rooms are equipped with uninterruptible power supplies (UPS) and a reserve power generator (Fig 1.).

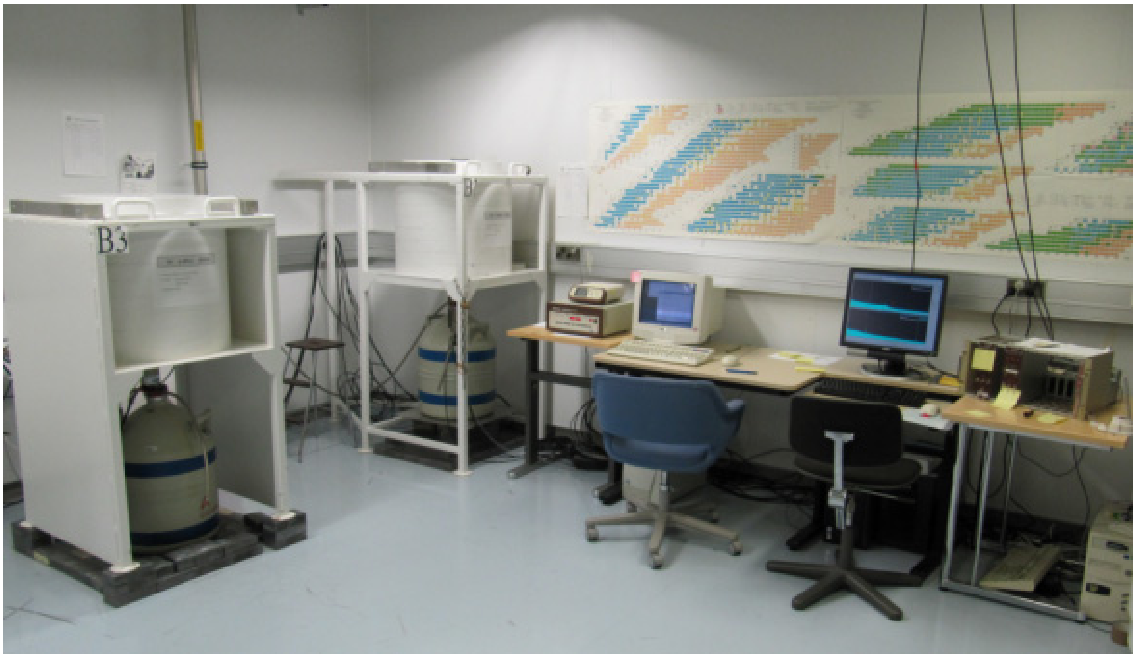

Fig. 1. One of the gamma spectrometry laboratories in the Research and Environmental Surveillance department. HPGe detectors are installed in heavy lead shields.

Currently there are altogether 14 low-background, high-resolution spectrometers at STUK laboratories for sample measurements (Table 1). In addition, two high-resolution spectrometers are employed for environmental monitoring sample measurements at sampling stations. The p-type HPGe detectors are of coaxial design with vertical cryostats. The relative efficiencies of the detectors range from 33% to 90% and energy resolution from 1.6 keV to 2.1 keV at 1.33 MeV. The detectors are installed in 12-14 cm thick lead background shields which are lined on the inside with cadmium and copper. The electronics of the spectrometers are either conventional NIM-modules or all digital stand-alone units. Measurement control and spectrum analysis are performed at a local microcomputer or on any network computer with the proper software installed. Lowest reachable detection limit is a few mBq/sample.

The laboratories in Helsinki use liquid nitrogen for cooling of spectrometers. The supply of nitrogen is arranged with vacuum insulated pipelines from a storage tank outside the building. All the detectors at Rovaniemi laboratory are electrically cooled.

Table 1. HPGe spectrometers for sample measurements at STUK laboratories.

Detector code	Manufacturer and model	Relative efficiency %	Energy resolution keV at 1.33 MeV	Purchase year
G1	ORTEC GEM-35185	33	1.76	1985
F5	ORTEC GEM-35190	38	1.70	1986
F6	ORTEC GEM-35190	39	1.75	1986
V2	ORTEC GEM-18190	18	1.90	1988
P1	ORTEC GEM-30185	30	1.69	1989
F9	ORTEC GEM-90205	87	1.87	1994
N2	ORTEC GEM-80190	78	1.80	1995
P3	ORTEC GEM-30190	36	1.74	1996
B1	Canberra BE5030	50	2.00	1999
P4	ORTEC GEM-35200	38	1.85	2001
G2	ORTEC GEM 50-S	70	1.85	2004
B4	Canberra BE5030	50	1.90	2005
B3	Canberra BE5030	50	1.88	2005
B5	Canberra BE5030	50	1.80	2007
F1	ORTEC GEM-8530	50	1.74	2008

For sample measurements there are three standard measurement geometries at the laboratory. Two of them are simple cylindrical plastic beakers with diameters of 42 mm and 74 mm, correspondingly. The height of beakers is 26 mm, and the volumes are 35 and 105 ml, correspondingly. The third beaker in use is a standard 0.5-l Marinelli beaker (Fig. 2.).

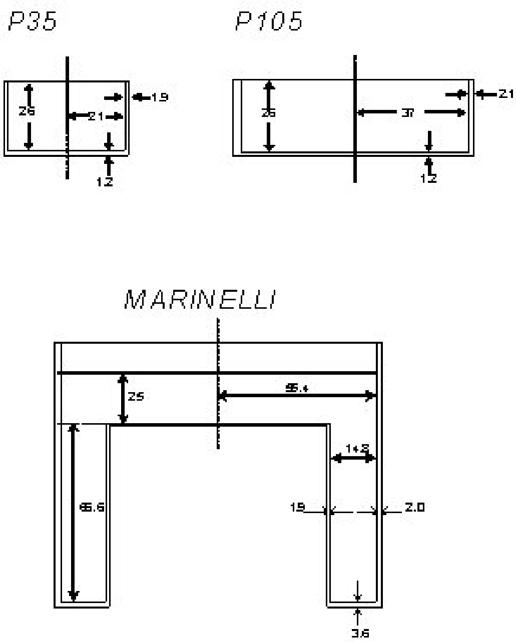

Fig. 2. Three standard measurement geometries for sample measurements at STUK.

All the beakers are measured right on top of the detector end-cap. The efficiency calibrations for other measuring geometries, densities or sample matrices can be obtained using a semi-empirical computer method developed to transfer full-energy peak efficiency to other geometries for the same detector (Ugletveit et al. 1989, Aaltonen et al. 1994).

The experimental efficiency calibrations of the detectors have been obtained with both separate single-line nuclides in water solution and multinuclide standard sources. The following nuclides have been used: ^{40}K, ^{51}Cr, ^{57}Co, ^{60}Co, ^{65}Zn, ^{88}Y, ^{109}Cd, ^{113}Sn, ^{137}Cs, ^{203}Hg, ^{210}Pb and ^{241}Am.

Analysis software and databases

In the early 1980s STUK developed a spectrum analysis programme especially for the analysis of low-level environmental sample measurements. The original GAMMA-83 code was written in Fortran language and run in a mainframe computer (Sinkko 1981, Sinkko and Aaltonen 1985). In the late 1980s, the code was ported to microcomputers and MS-DOS operating system. A MS-Windows version of the programme was written in 1997. This version introduced several new features, e.g. interactive user interface, evaluation of detection limits for selected nuclides, various optional output files.

GAMMA-99 programme searches peaks, evaluates their area and, after subtracting the background, calculates activity concentrations for identified nuclides. The varying heights and densities of the samples and the effect of true coincidence summing are taken into account in calculating the results. The program can be operated either in interactive or automatic mode. The updated master nuclide library has at present 139 nuclides and 1027 gamma lines, including decay data for calculation of coincidence summing corrections.

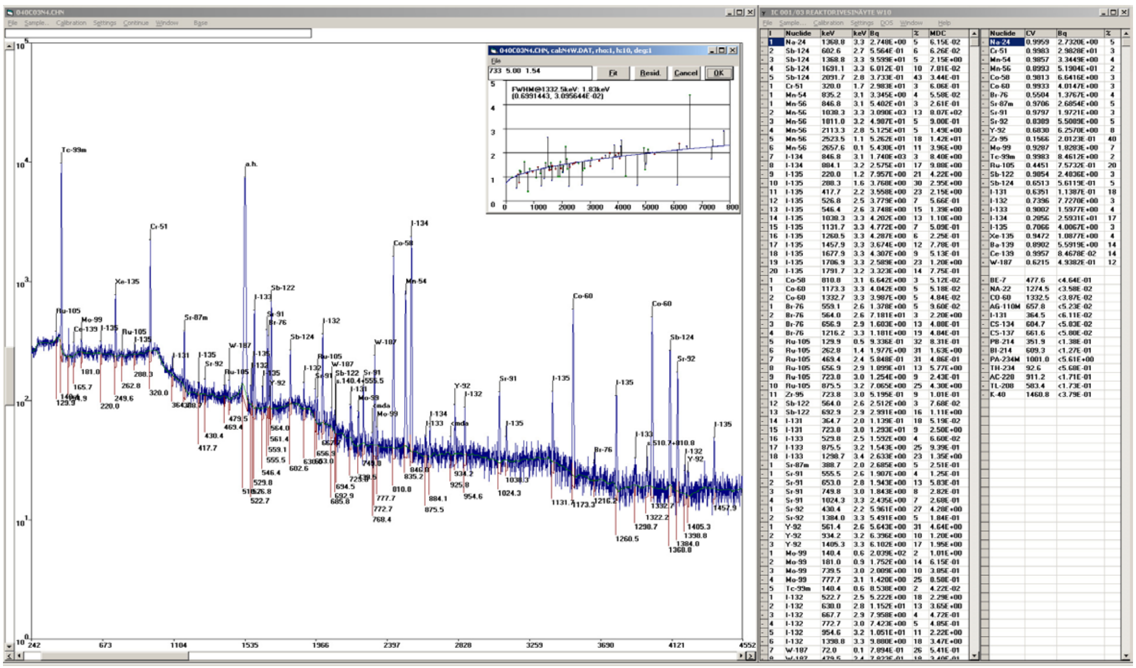

Fig. 3. GAMMA-99 user interface; spectrum, result and energy resolution views.

Recently, the laboratory has implemented and validated a new analysing software package, Unisampo/Shaman. UniSampo is the newest member of the Sampo family of gamma-ray spectrum analysis programs. UniSampo carries out the first part of the analysis, i.e. determination of the peak energies and intensities with the corresponding uncertainty estimates (Aarnio et al. 2001) .Peak analysis results from UniSampo are the input for Shaman programme which is a rule-based expert system for qualitative and quantitative radionuclide identification. Shaman utilizes a reference library with 3 648 radionuclides and 80 062 gamma lines extracted from the ENSDF and NUDAT databases. The programme takes into account e.g. true coincidence summing and is capable of identifying also sum peaks, escape peaks, and X-ray peaks (Aarnio et al. 1995, Ala-Heikkilä et al. 1995).

In 2004, uniform Laboratory Information Management System (LIMS) software was taken into use at the analytical laboratories at STUK. All analytical data, including basic information on sampling and samples, raw measuring data and results, are stored in the database.

In 2010 especially for gamma spectrometry designed database, LINSSI, will be integrated into the LIMS system. LINSSI (Linux System for Spectral Information) is a MySQL database that covers the whole production chain from sample preparation to final analysis results. Also full control of calibrations and their histories is supported, providing complete traceability of the results. As a database system specifically designed for gamma spectroscopy, LINSSI allows for more comprehensive recording of various quantities related to a spectral analysis than LIMS. LINSSI is being developed in collaboration with STUK, Helsinki University of Technology and Health Canada (Aarnio 2006).

Quality assurance

Confidence in the performance of equipment is maintained by regular calibration and servicing, routine checks and control charts. The most important control charts, background, energy resolution and detection efficiency, are based on regular QC measurements. Computerised techniques record the performance characteristics of spectrometers and software. These include peak widths and positions, which are evaluated during the spectrum analysis of every sample measurement, thus decreasing the frequency of regular QC measurements. The obtained values are recorded in a log file.

Corresponding warning limits and action limits are set, taking into account the different characteristics of spectrometers. The acceptance criteria in the charts are normally set as warning limits of 2σ and an action limit of 3σ above and below the long-term average data. However, with low-level measurements the differing behaviour of various background components requires careful consideration. While count rates of the peaks due to ^{40}K and cosmic background can be considered as stable, the count rates from daughter nuclides of ambient radon might vary either periodically or irregularly. ^{210}Pb in detector material and shielding is a special source of background, since it decays at a constant rate and can even be calculated for the future. Each of these components might require separate update frequency, also taking into account the contribution of the background to the total uncertainty of a particular result.

The dominant uncertainty contributions in gamma-ray spectrometry of environmental samples are usually those of count rate, full-energy peak efficiency and source-detector geometry. In special circumstances other sources of uncertainty will have a major contribution. Examples of these are the same or higher background count rate of a nuclide compared to sample activity, low energy gamma-rays from high density samples, and cascade-summing correction of the poorly-defined decay scheme of a rarely analysed nuclide (Ikäheimonen et al. 2006).

Successful participation in proficiency tests and inter-laboratory comparisons is highlighted as the best way of proving good quality. Since it is not possible to have exercises for all the gamma-emitting nuclides, it is important to show quality of the analyses of a few nuclides representing a variety of situations, especially including the wide range of gamma-ray energies.

To test and demonstrate proficiency also for short-lived nuclides, STUK co-organizes intercomparisons of reactor water analysis. The nuclear power plant coolant water samples are taken and prepared by NPPs and delivered promptly to the participating laboratories. Since all the participants get their sample within a few hours, it is possible to analyse and compare results of nuclides with half-lives shorter than one hour (Klemola 2008).

Applications

Applications of gamma spectrometry at STUK include surveillance of environmental radioactivity in Finland, radioecological studies of both natural and artificial radionuclides, contracted services for industry and trade and for various organisations and institutes. These applications provide a wide variety of sample types to be analysed including air, deposition, water, milk, foodstuffs, biota, sediments, swipe samples, building materials and various export products.

Since 2003, STUK has been one of the radionuclide laboratories certified by the CTBTO (Comprehensive Nuclear Test Ban Treaty Organisation) to support the network of radionuclide air sampling stations. These laboratories perform additional analysis of a suspect air filter samples to verify presence or absence of fission or activation products. Radionuclide laboratories also analyze samples from stations as a part of network quality control program. The main requirements for the CTBTO certification are quality system, documented procedures for sample analysis and good quality of analytical results. Certified laboratories are required to participate in proficiency test exercises organised by the CTBTO.

One of the tasks of the gamma spectrometry laboratory is to maintain measurement standards which ensure the traceability and accuracy of activity measurements of gamma-emitting radionuclides. Measurement standards are accurate spectrometers with traceable calibration, validated methods, or radiation sources. STUK is a member in a network for ionising radiation standard laboratories and participates in the activities of International Committee for Radionuclide Metrology (ICRM) and its Gamma-Ray Spectrometry Working Group.

Gamma spectrometry gives also support to radiation and nuclear safety authorities at STUK performing analyses for dosimetry, metrology, inspections, NPP radiation safety, safeguards and quality assurance. Gamma spectrometry is the most efficient method also for emergency preparedness, which presents a special challenge for the maintenance and development of software and equipment and also personnel competence for rapid response.

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When is *in situ* gamma spectrometry motivated?

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Abstract

In situ gamma-ray spectrometry has since the introduction of portable germanium detectors been a widely used method for the assessment of radionuclide ground deposition activity levels. It is, however, a method that is most often associated with fairly large and poorly known uncertainties. A large part of the combined uncertainty originates from the source characterization, e.g. soil density and activity depth profile. In order to reduce this uncertainty soil sampling and subsequent laboratory analysis is often needed. The more samples collected for this characterization, the lower uncertainty in the *in situ* measurement result. However, if a large enough number of soil samples are collected, the uncertainty in the ground deposition activity measure directly obtainable from soil sample measurements will surpass the uncertainty of the *in situ* measurement. This will then render the *in situ* measurement superfluous. In this work we have considered *in situ* gamma ray spectrometry from a decision-maker-point-of-view; at which uncertainty level is *in situ* gamma ray spectrometry fit-for-purpose, and at which uncertainty level are laboratory measurements of soil samples preferable.

Review of modern application of gamma-beta spectrometer – radiometer MKGB-01 "RADEK"

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Review of a modern application of spectrometer- radiometer of gamma-beta-radiation MKGB-01 "RADEK"

In the year 2001 STC "RADEK" Ltd. company produced spectrometer- radiometer of gamma-beta-radiation MKGB-01 "RADEK". Since then the device is widely used by various scientific and educational institutions, certification authority, research laboratories and organizations of nuclear fuel cycles. Apart its common use as a measuring instrument of activities of γ-radiating radionuclides in samples of ground, vegetation, water, foodstuff, materials of construction and other substances, it was also applied as a basis for the construction of unique spectrometric units.

Thus, the Railway Service Institute uses MKGB-01 as a measurement instrument of radionuclide contamination of railway acres and roads. On basis of MKGB-01 the underwater gamma-radiation spectrometer was developed for Geologic Institute, allowing the radioactivity measurement of bottom silt with the moving of detector along the bottom. The same spectrometer was also produced for the Federal Security Guard Service of Russian Federation for installation on boats patrolling water area.

The special place in radioactivity measurement of radionuclides belongs to spectrometry of body radiation. So the SEG-10P was developed as a modification of MKGB-01. It is completely implemented in the form of armchair and allows to measure the activity of incorporate γ- emitting radionuclides. The apparatus is used by nuclear power plants and medical centres.

In 2006 in the Russian Centre of Emergency and Radiation Medicine the expert spectrometric unit for human radiation measurement was constructed and put into operation. It is completely composed of analyzers of MKGB-01 spectrometer (14 pieces) and different types of detectors (scintillation and semiconductor detector). The unit is presented by steel chamber with 4x2x2 meters dimensions edged by lead, cadmium and copper. The chamber is provided by scanning operation.

Due to STUK Finland has about 50 devices of such kind.

Introduction

Nuclear-physical analysis methods of samples are widely used in various fields of science and technology. It provides an opportunity to resolve the tasks in such spheres as medicine, ecology, geologic intelligence, atomic industry, etc. To study the radionuclide composition of samples, among others, gamma- and beta spectrometric methods of analysis are used.

In the beginning of 2001 the company STC “RADEK” manufactured a spectrometer-radiometer of gamma and beta radiation MKGB-01 “RADEK”. Since that moment this device is widely used in different scientific, educational organizations, centers of certification, research laboratories and enterprises of NFC (Nuclear Fuel Cycle). Besides the standard use of this device by way of mean of measuring of γ and β radiating radionuclides activities in samples of ground, vegetation, water, foodstuffs, constructing materials and other substances, a unique spectrometric complexes are also constructed on its base.

Review

Thus, the institute of railway transport (VNIIZhT, Russia, Moscow), uses MKGB-01 by way of mean of measuring of radionuclide contamination of railroad track right of way and of automobile roads[1]. This complex is called “Main gamma-spectrometer of high sensivity (MGSHS)” and is placed in laboratory rail car of VNIIZhT. High sensitivity of gamma-spectrometer is determinated by a large volume of scintillator.

Constructively the spectrometer MGSHS consists of two DSU (detection and selection unit with 2 detectors each); unit for summing signals; four analog-digital converters (ADC) MD-129; three high-voltage power supplies; low-voltage power supply and PC. Functionally and programly the complex allows to use a signal from GPS - receiver, to turn on the alarm signal (bleeper, light, etc.), means of objective control, etc. Each DSU has 2 detectors in it on the base of crystals NaI(Tl) with size 100x200 and a PMT-49 (Photomultiplier Tubes) for registration of external gamma-radiation, and a stabilization unit based on beta-gamma coincidence (detector on the basis of scintillating plastic and PMT-85 for stabilization, source of ionizing radiation on the base of ^{60}Co with activity less than the minimum significant).

Conducted in 2007 the induction and performance tests on the known contaminated territories showed, that during the correlation of graphs of ^{137}Cs density contamination measurements, performed with an interval of 17 years, paying respect to the decay and migration of radionuclide correction, the discrepancy doesn’t exceed 20%. At this moment the spectrometer is used in laboratory rail car, belonging to VNIIZhT [1].

Also an underwater gamma spectrometer on the basis of MKGB-01 is developed for geological institute, which allows to measure the radioactivity level of bottomset beds through the aid of the detector moving along the bottom. The spectrometer consists of a detector towed on bottom and of an onboard analytical complex. The full consist of the complex includes: a sealed shock-proof sleeve with an obstacles bypass system “Eel”, containing a scintillation gamma-radiation detector with crystal CsJ(Tl) with size 80×80 mm; an analytical unit, containing an analog-digital converter board (ADC); power supplies and amplifiers (Fig.1); control PC with appropriate software; armored towing - connecting cable [2].

To stabilize and to control the amplification, a reference source of ^{113}Sn mounted in the detector sleeve was used, which allowed to change operatively the amplifying coefficient depending on the involving facts. The spectrometer is calibrated in the measurement units of specific activities on the volumetric, saturated for gamma-radiation state standard norms. Estimated minimum detected activity with the exposure time of 30 seconds is as follows for ^{137}Cs - 4 Bq/kg, ^{226}Ra - 10 Bq/kg, ^{232}Th - 7 Bq/kg, ^{40}K - 90 Bq/kg. The working depth of spectrometer sleeve immersion - 500 meters, and by towing with speed 3–4 knots, using 600 meters of a tow cable-hemp – to 200 meters. By towing the underwater gamma-spectrometer the exposition set of each spectrum was about 30 seconds. The towing speed – 3 knots. Thus, in average each single spectrum corresponds to 50 meters of profile.

Fig. 1.

An example of underwater use of spectrometer MKGB-01 in geological purposes is the charting of ferromanganese nodules fields (FMK) in the gulf of Finland of Baltic sea. It’s well known that FMK of marginal and inland seas sorb intensively the ^{226}Ra and are largely enriched with it. At the same time the accumulation of ^{137}Cs by nodules is insignificantly to the aleuropelite precipitations, which are intensively accumulating ^{137}Cs in zones, related to the impact of the Chernobyl disaster. These provisions give a theoretical basis of FMK fields charting by means of underwater gamma-spectrometry. Conducted in the gulf of Finland the experimental-methodological works have confirmed the assumptions made above. It can be stated a fairly reliable charting method of FMK fields on the ratio ^{226}Ra/^{137}Cs. Besides it the underwater gamma-spectrometry directly on the activity of ^{137}Cs, and also on the ratio ^{137}Cs/^{40}K and ^{226}Ra, ^{232}Th/^{40}K in many cases allows to separate and to chart the fields of aleuropelite sediments, of sands and boulder-pebble material.

Another successful example of underwater gamma-spectrometer use in geoenvironmental purposes was its application for radiogeochemical situation study in bottom sediments on the mass burial areas of PDUO (potentially dangerous underwater objects), representing the container landfills, containing radioactive waste, or single large objects, containing large quantities of radioactive substances. The researches of four burial plots showed, that by one of the profiles, passing close to two sunken ships with radioactive substances and containers landfill, an extended anomalous zone with ^{137}Cs activity up 1500 Bq/kg was detected. In a container burial area with radioactive waste, a local anomaly with length about 100 meters with ^{137}Cs activity in the maximum of 80 Bq/kg was also detected by another profile. Besides, in different sectors, the ^{137}Cs anomaly with activity from 40 to 60 Bq/kg were allocated by several profiles. It should be noted, that all the identified anomalies are located in the distribution area of aleuropelite precipitations, which, as known, sorb intensively ^{137}Cs. Cesium background activity for the entire area was vastly negligible and ranged from values below the MDA (minimum detectable activity) to 7 Bq/kg.

Thus, the towed underwater spectrometric complex MKGB-01 proved to be quite suitable for solving problems of ^{137}Cs areal distribution study in bottom sediments and of local pollution identification in bottom sediments, associated with specific objects [2].

The same spectrometer was created by order of Federal Security Guard Service of the Russian Federation (FSGS RF) for installation on boats, patrolling the water areas.

A special place in the measurement of radionuclide activity is occupied by spectrometry of human radiation. In this field a spectrometer of human radiation SEG-10P is developped, it's a modification of MKGB-01. It is designed as a chair with console module on the anvil, aimed at the geometric center of the chair (Fig.2).

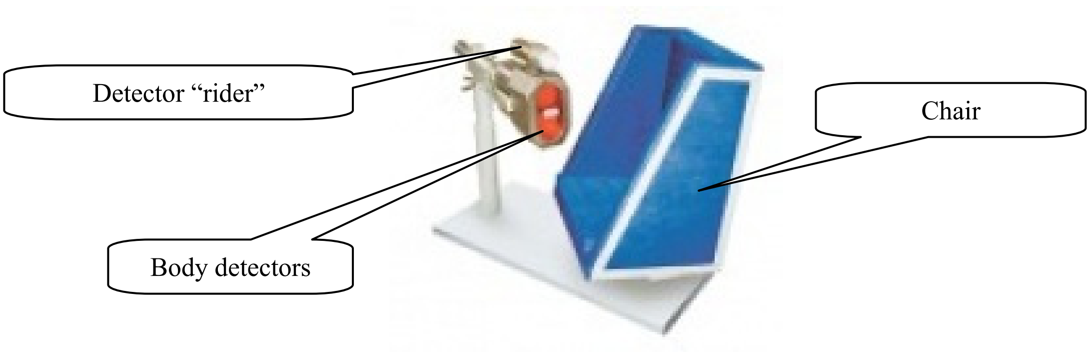

Fig. 2.

The module consists of two body detectors and so-called detector "rider", and allows to evaluate the activity of incorporated γ- radiating radionuclides in human body, lungs and thyroid. The spectrometer calibration is made by means of unified phantoms UF-02T (body) and FSZ-04T (thyroid). The phantoms are made of rectangular polyethylene blocks with bulk 1 kg with openings for placing the rod sources of radioactive material. The spectrometer is widely used on nuclear power plants and in medical centers.

In 2006 in the center of emergency and radiation medicine in Saint-Petersburg an expert low-background spectrometric complex of high sensitivity "SICh-E" for measuring human radiation was manufactured and was put into operation. In fact it's a totality of low-background spectrometers of human radiation and consists of 28 different spectrometric tracts (scintillation and semiconducting), based on electronic modules of spectrometer-radiometer MKGB-01 (14 two-channel analyzers MD-198, Fig.6). The low background of spectrometer is provided by its construction, which represents a steel chamber with size 4x2x2 meters, plated from the inside by lead, cadmium and copper (Fig.3,4,5).

Fig. 3. Chamber of spectrometer of human radiation inside view.

Fig. 4. Detectors view inside the chamber without couch.

Fig. 5. Detectors view inside the chamber with couch and with phantom.

Fig. 6. View of 14 analyzers MD-198 from the front side in the control cabinet.

The complex is intended for expert surveys of the content of incorporated radionuclides of persons, exposed to the radiation in various accidents and incidents, of nuclear technology cycle factory staff during the current and periodic monitoring of internal irradiation, and also of people, living in the radioactively contaminated areas [3]. The works performed on the spectrometer:

– identification of radionuclide composition and measurement in the linear longitudinal scanning mode of low activity levels of gamma - radiating radionuclides throughout the human body in minimum dependence from the character of their distribution in organism and from variations of examined body anthropometric parameters (the contingent of examined people – adults and children from 5 y.o.);
– identification of radionuclide composition and gamma – radiating radionuclides activity measurement in particular human organs: lungs, thyroid, liver, reins, etc.;
– measurement of ^{90}Sr radionuclide content in bones by the bremsstrahlung spectrum as in skeleton entirely, and in separate skeleton parts (in frontal bone, shin bone, etc.);
– measurement of content in the lightweight transuranic radionuclides with low photon radiation energy: plutonium, americium, etc., and also of gamma-radiating radionuclides content with energy less than 300 keV;
– research of radionuclides metabolism in human body in the estimation of parameters;
– research of homeostasis (on potassium metabolism by measuring the ^{40}K radionuclide activity in human body).

The widespread use of MKGB-01 is not limited only by RF. Thanks to STUK (Radiation and Nuclear Safety Authority) Finland has about 50 such devices. Since the mid 1980's in local laboratories of Finland from 6 to 20 counters Mini-Assay were used for evaluation the specific and radioactive substances volumetric activity in foodstuffs and of radon in potable water. The counters showed the total intensity of gamma radiation in samples, and when they began to fail, STUK started to develop a modernization plan of radiometric control equipment. Thus, a new equipment was purchased – gamma-spectrometer MKGB-01, which is used for radionuclide evaluation in samples. The spectrometer consists of a detector Nal(Tl) with size 63×63, located in a lead protection with thickness 60 mm, computational electronics and of a gamma radiation spectrums processing program. Two measurement geometries are used – Marinelly container with volume 1l for liquids and cylindrical cuvette with volume 320 ml for solid foodstuffs. The minimum detectable specific activities of ^{137}Cs with measuring time - 1000 sec. in laboratory - 50 and 30 Bq/kg for cuvette with volume 320 ml and Marinelly container, respectively. The spectrometer is calibrated for samples density from 0,2 to 2,0 g/cm^3 [4].

Referring to the scientific-research sphere of application of spectrometer MKGB-01, it's necessary to mention its use in the maquette of device, which realizes the method of KX-gamma coincidences, created in the ionizing radiation measurements department of D.I. Mendeleev Institute for Metrology (VNIIM). The activity evaluation results, obtained on this maquette, enable to recommend it for inclusion in the primary standard of RF [5].

Another example of MKGB-01 use in the research activity can be the Nephrology Department of Saint-Petersburg Medical University, where spectrometer MKGB-01 is used as an equipment for reins function study. For this, a so-called parameter GFV (glomerular filtration velocity) is evaluated. A patient is injected by a certain amount of radioactive medication (DTPA ^{99m}Tc) with known (measured in syringe on the spectrometer) activity, then after some time the blood of the patient is sampled, centrifuged and the activity in blood plasma is measured, some time later the blood is sampled, centrifuged again and the activity is measured again too. Then, according to the given algorithm, a so-called clearance is evaluated, with the help of which the reins function is studied.

Thereby, the gamma-beta spectrometer MKGB-01 have found a wide application in various applied and scientific problems, and, therefore, the realization of this project can be considered as successful.

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Radiation monitoring network with spectrometric capabilities: implementation of $LaBr_3$ spectrometers to the Finnish network

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Abstract

Dose rate monitoring networks equipped with Geiger-Müller or proportional counters can provide an alarm with sensitivity to approximately 0.1 µSv/h increase over the ambient background. For airborne activity the 0.1 µSv/h increase translates to activity concentrations in the kBq/m^3 range. Better sensitivity and nuclide identification can be achieved by gamma spectrometers with sufficient energy resolution.

$LaBr_3$ scintillators provide a good energy resolution of about 2.5 % at 662 keV enabling the separation of I-131 (364.9 keV) and Pb-214 (351.9 keV) lines. The detectors are large enough to yield an adequate sensitivity, making them suitable for dose rate monitoring network applications. Major drawbacks of $LaBr_3$ detectors are the radioactive impurities in the material and the gain variation as a function of temperature. A fitting algorithm based on the La-138 and K-40 multiplet at 1430-1470 keV was developed to stabilize the detector. With fully automatic software controlling the $LaBr_3$ spectrometer during data acquisition the system gain is stable and provides environmental spectra of excellent quality. The same fitting algorithm is also applied to the analysis of the whole spectrum.

Spectrometers have recently been installed around the Finnish nuclear power plants. They send measurements to a central database at STUK headquarters over a secure wireless TETRA network at ten minute intervals. The spectra are automatically analyzed and checked for the presence of small signatures of man-made activity. Airborne or fallout nuclide activity concentrations and the dose rate are calculated from the spectra. The detection limit for I-131 is roughly 40 Bq/m^3. In comparison, an activity concentration of 1700 Bq/m^3, yielding an increase of 0.1 µSv/h in the dose rate, would be needed to create an alarm in the Geiger-Müller monitors.

In summary, the $LaBr_3$ spectrometer system now in operational use in Finland, is an excellent enhancement to the country-wide radiation monitoring network providing nuclide identification with high sensitivity.

Introduction

The Finnish ambient dose rate monitoring network consists of 255 monitoring stations. The network covers the whole country, with several stations around the Finnish nuclear power plants in Olkiluoto and Loviisa. Ambient dose rate at the station is monitored with three independent Geiger-Müller counters, with two identical counters for low dose rate and one counter for high dose rate. Instrumentation also includes a rain sensor. The measurement data from the stations is sent in ten minute intervals to the Radiation and Nuclear Safety Authority (STUK) headquarters to form a real-time picture of the dose rates. An increase in dose rate over a predefined limit, verified by the three independent Geiger-Müller counters at the station, generates an alarm at the regional Emergency Response Centres in addition to alerting STUK's duty officer.

The Geiger-Müller counters of the monitoring network can discern about 0.02 µSv/h change in the ambient dose rate. With the natural background in Finland varying between 0.05 µSv/h - 0.30 µSv/h, the alarm limit has been set to 0.4 µSv/h. In addition to a fixed alarm limit, a station-specific comparison of the latest dose rate value to a smoothed background over the last 7 d measurements is used to detect anomalous cases of concern. The alarm level is set to a 0.1 µSv/h increase over the average. For I-131, an air concentration of 1700 Bq/m^3 is needed to raise the dose rate by 0.1 µSv/h. Countermeasures, however, are needed already at the concentration levels of 1000 Bq/m^3 of I-131. Another drawback is the fact that Geiger-Müller counters do not provide identification information on the possible radionuclides responsible for an increased dose rate. An illustrative case is a weather-related increase in the radon progeny concentration of the surface air leading to elevated dose rate. Discerning these cases with natural origin reliably and promptly from events that might require further action can be a challenge to network operators, if spectroscopic measurements are not available.

The Finnish dose rate monitoring network was updated in 2005-2007. The prospects of adding spectrometric instrumentation to achieve better detection limits and nuclide identification was envisaged, but the NaI scintillators considered at the time did not provide sufficient energy resolution for e.g. resolving the I-131 gamma line at 364.9 keV photon energy from the Pb-214 radon progeny gamma line at 351.9 keV. Despite the lack of suitable spectrometers the monitoring station structure was designed to be flexible, with several interface options for adding new instrumentation.

Recently, a new scintillator material $LaBr_3$, has emerged as a viable alternative. The Cerium-doped $LaBr_3$ presents a scintillation photon output higher than NaI(Tl), (63,000 photons/MeV) at a wavelength suited for a photocathode. In addition, the crystals have an excellent energy resolution and are available at sizes suitable for gamma spectrometric measurements (Dorenbos et al. 2004, van Loef et al. 2001, Pani et al. 2007). The energy resolution of $LaBr_3$ scintillators is 2.5-2.9 % at 662 keV, whereas the resolution of NaI scintillators is 6-7 %. The operational lifetime of a $LaBr_3$ scintillator coupled with a suitable photomultiplier tube and related signal processing electronics is estimated to be several years.

$LaBr_3$ scintillators also exhibit problematic characteristics that need to be considered when designing a measurement system. The $LaBr_3$ material contains radioactive La-138 and Ac-227 impurities, which contribute to the background complicating the analysis of the spectrum. La-138 in particular has a disturbing impact on the detection limits due to the increase of spectral baseline. In addition, the gain variation of the detector and the coupled photomultiplier tube may be of several percentages depending on the temperature and the detector orientation in terrestrial magnetic field.

Despite these complications, the overall performance of $LaBr_3$ scintillators is good enough for continuous environmental monitoring applications. In this paper, we show how the problems related to $LaBr_3$ material and gain instability have been overcome. We discuss how these detectors have been added to a geographically sparse monitoring network and show that automatic analysis procedures can be applied to process the environmental spectra from $LaBr_3$ scintillators.

Material and methods

Measurement setup and data processing

The $LaBr_3$ scintillators installed to the Finnish dose rate monitoring network utilize Saint-Gobain BrilLanCe 380 crystals with a height and diameter of 1.5 inches. The detector, including Hamatsu photomultiplier tube (R6231-100) is connected to a Princeton Gamma-Tech Instruments MCA2500 multi-channel analyser (MCA). The MCA2500 is connected via an ethernet interface, allowing cable lenghts up to 100m without a separate signal amplifier. This is an advantage for the typical monitoring station installation, where the central unit is located indoors and the sensors stand outside at some distance from nearby buildings.

The detector and the MCA are housed inside a weather resistant and insulated casing. The casing insulation has been tested in the expected outside temperature range of -30 C° … +30 C° to keep the detector and the MCA unit inside a manufacturer specified operational temperature range of 0 °C ... +50 °C, while preventing the condensation of humidity on the electronic components. The aluminium casing has a wall thickness of 1.5 mm at the scintillator end, not preventing the detection of low energy gamma radiation.

The monitoring station central unit is running on an embedded Linux computer. The Intel PXA255 computer has no moving parts ensuring long life time, with the open Linux software architecture offering software vendor independence and flexibility, including monitoring station software updates with user scripts and installation of additional programs written with C/C++ programming language. The control unit has several interfaces for external equipment and sensors, such as the ethernet connection used to interface with the $LaBr_3$ detector's MCA.

Individual stations communicate via a TETRA-based, closed government radio network called VIRVE (Fig 1). The VIRVE network is mainly used by law enforcement authorities and rescue services to relay voice communications, but the TETRA network also supports TCP/IP protocol and text messaging, with both services utilized by the radiation monitoring network stations to relay measurement data. The setup allows for two-way communication enabling remote management and software updates.

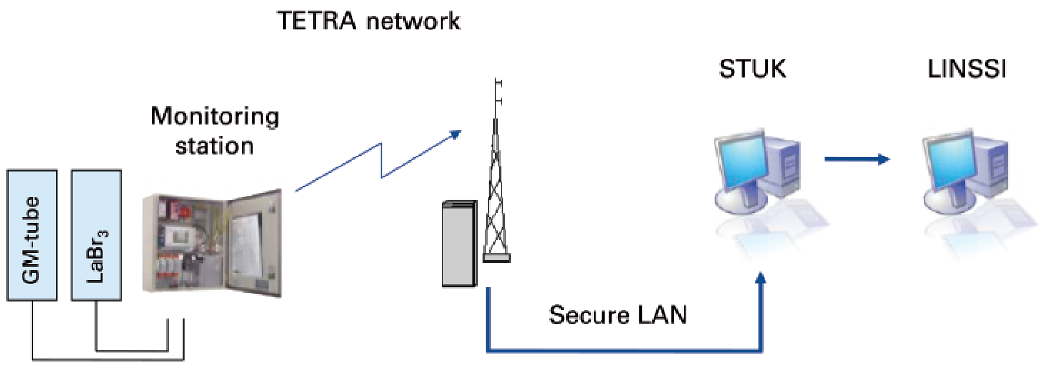

Fig. 1. Architecture of the Finnish radiation monitoring network.

The $LaBr_3$ detectors are operated in the same data acquisition mode as the Geiger-Müller counters, with a spectrum reading interval of ten minutes. The MCA conversion gain is set to 2048 channels, with the energy range extending up to 3200 keV. After a spectrum is read, it is processed at the monitoring station by a specialized multiplet fitting algorithm *mufi*. The purpose of the *mufi* algorithm is to stabilize the energy calibration against gain fluctuations induced by changes in external temperature. The program analyses the position of the La-138/K-40 multiplet at the photon energy region of 1430 keV - 1470 keV. If the multiplet peak positions have moved from their nominal values, *mufi* returns the channel shift, which is converted to a change in the amplifier gain. A new gain setting is then sent to the MCA unit to keep the energy calibration stable.

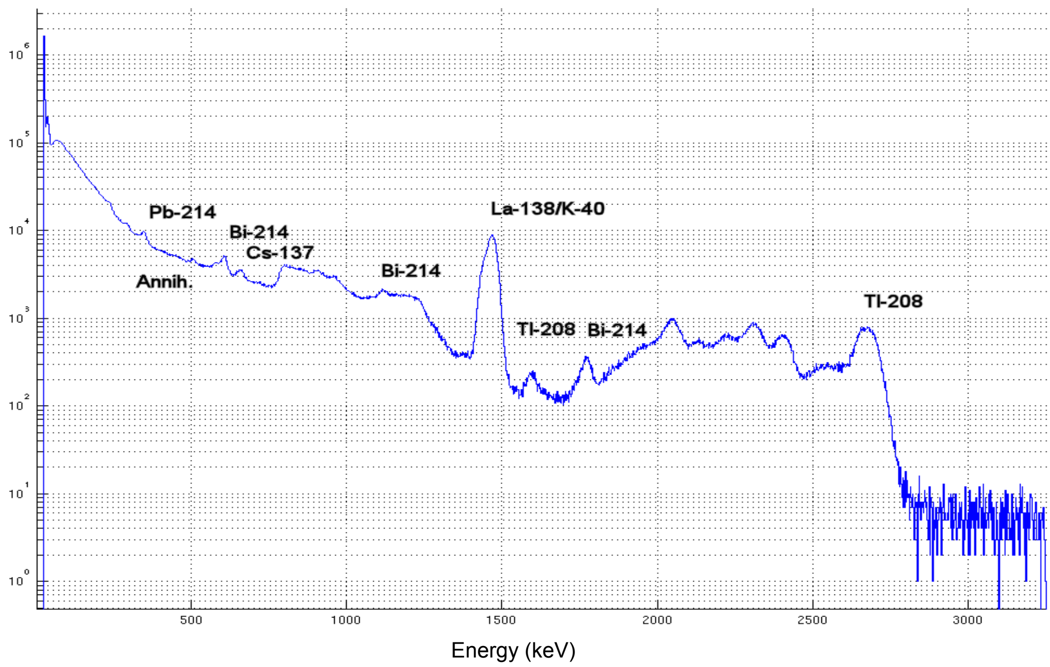

Fig. 2. A spectrum of the background radiation measured with the $LaBr_3$ detector. The La-138/K-40 multiplet is used to stabilize the energy calibration against gain fluctuations induced by changes in temperature. Measurement time was 24 h.

The spectral data and the *mufi* analysis results are then send from the monitoring station to a central server at STUK and stored to a gamma spectrometry database *Linssi* (Aarnio et al. 2006). A summation spectrum with a length of 60 min is formed from the individual 10 min spectra. These are automatically analyzed by a second multiplet fitting algorithm *jmufi*, which analyses predefined spectral regions for the presence of interesting gamma peaks. The *jmufi* software provides nuclide identification and count rates, detection limits and error estimates for the various quantities. In addition to nuclide identification, dose rates are calculated from the 10 min spectra to compare with the the Geiger-Müller counter values. The *jmufi* program and the dose rate calculation method are discussed in detail in Toivonen, 2008. A spectrum of the background radiation integrated over a 24 h measuring period is shown in fig. 2. Several radon progeny related peaks are apparent in the spectrum, with the La-138/K40 multiplet peak being the most prominent feature.

As of spring 2010, 19 monitoring stations have been equipped with $LaBr_3$ detectors. Most of these spectrometric stations are situated around the Finnish nuclear power plant sites in Olkiluoto and Loviisa (Fig 3.). Installations will continue with few additional $LaBr_3$ spectrometers during the year 2010.

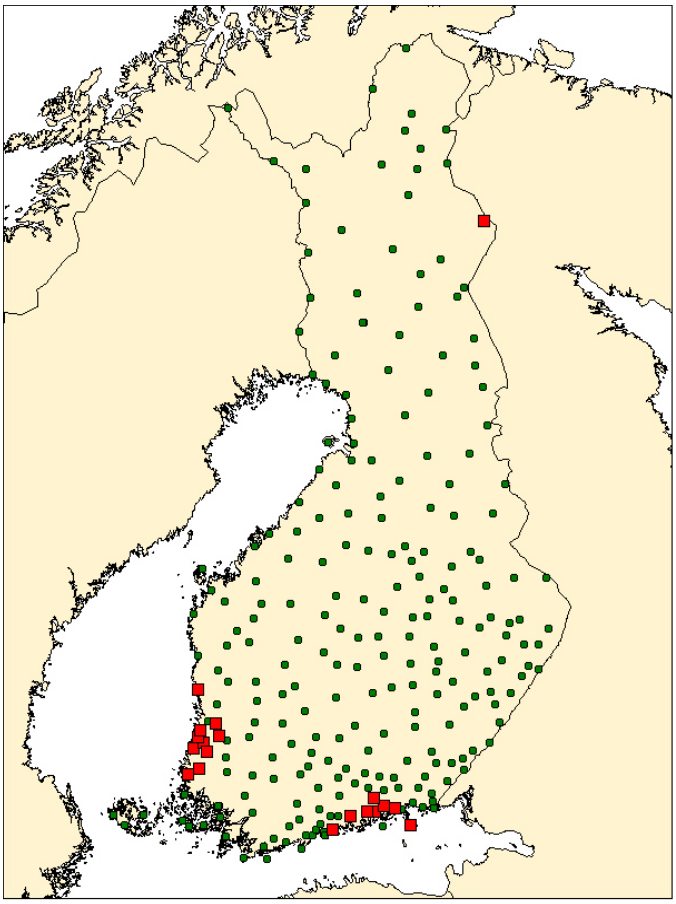

Fig. 3. Ambient dose rate monitoring network. Stations with $LaBr_3$ detectors are shown with red squares.

Conversion of detector count rates to activity concentrations

The nuclide specific count rates recorded at the spectrometers need to be converted to more general quantities. For a dose rate monitoring network, two distinct cases are of interest. In the first case, the radioactive nuclides are airborne and the nuclide specific air concentrations (Bq/m^3) need be derived from the respective count rates. In the second case, the radioactive nuclides are deposited on the ground and the interesting quantity is the fallout (Bq/m^2). The efficiency curves relating the nuclide-specific count rates to the different concentrations are also needed to estimate the detection limits (contamination level) which would generate the alarm. To this end, we have developed a semi-empirical method to derive the peak efficiency calibrations for the two different cases outlined above. We have derived equations for the efficiency curves of extended source geometries, utilizing a point source peak efficiency curve at a fixed source-detector distance.

The point source peak efficiencies can be easily measured with a selection of well calibrated sources at some finite source-detector distance *r*. From this data, the point source efficiency at an arbitrary distance is calculated. For airborne radioactivity one needs to consider the $1/r^2$ counting geometry dependance and the absorption of the radiation in air. The angular dependence can be neglected in the present case as the $LaBr_3$ crystals with equal height and diameter have only a small directional sensitivity. Finally, the point efficiency is integrated over the involved space to give the peak efficiency curve for the volume source. A similar approach is used to derive the peak efficiency curve for the fallout. A more detailed discussion of the method is given by Toivonen et al. 2009.

We obtained the experimental point source efficiency data at five different photon energies with a detector-source distance of 5 m. The experimental data set was supplemented by Monte Carlo calculations at different photon energies to yield a point efficiency curve over the whole energy region of interest. Fig. 4 shows the efficiency curve for the volume concentration, together with a scaled point source efficiency curve. With the calculated efficiency curves the nuclide specific count rates can be converted to corresponding air concentrations and fallout.

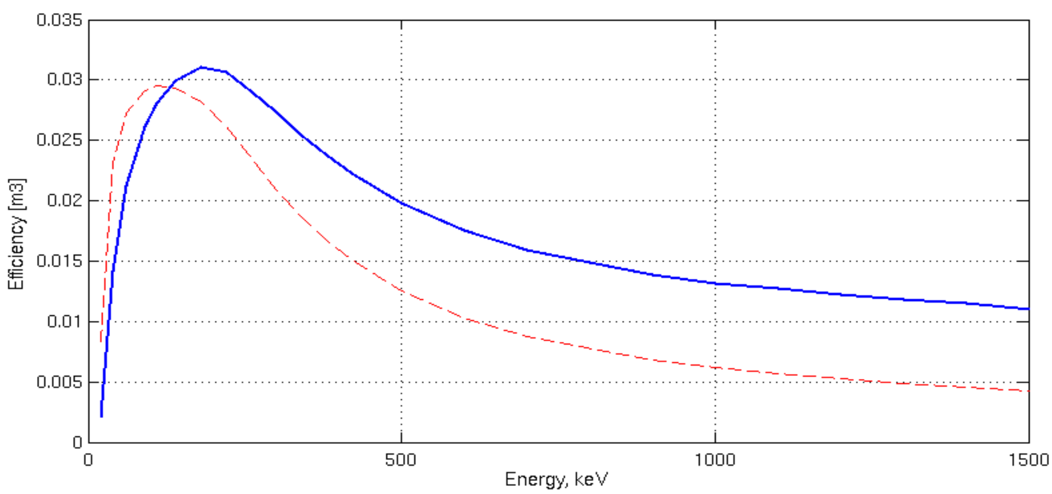

Fig. 4. Efficiency curve for the air concentration. The dashed line is provided for the reference; it is the shape of the point source efficiency at 5 m, multiplied by a factor of 10,000

Results

After deriving the efficiency curves for the relevant source geometries we proceed to discuss the detection limits of nuclides and compare the situation with a monitoring station equipped solely with Geiger-Müller counters. One of the most important benchmark nuclides is I-131 due its relevance in nuclear power plant accident scenarios, where significant amounts can be released into the atmosphere. Then the dose delivered to the thyroid gland by the inhaled or ingested iodine is of high concern.

Using *jmufi*, the main gamma line of I-131 at 364.9 keV photon energy is well separated from the Pb-214 radon progeny gamma line at 351.9 keV in the $LaBr_3$ spectra. As discussed before, an activity concentration of 1700 Bq/m^3 of I-131 in the air would increase the dose rate by 0.1 μSv/h, triggering an alarm at the Geiger-Müller counters of the Finnish monitoring network. For $LaBr_3$ detector, with typical background count rates and a 10 min measurement, the decision limit (Lc) for I-131 at risk level of 10^{-6} (type I error with $k\alpha = 4.75$) yields to approximately 30 Bq/m^3 using the derived peak efficiency shown in fig. 4. The MDA (Ld) is 40 Bq/m^3 at risk level of 5 % (type II error), which is a factor of 40 smaller compared to the alarm level of Geiger-Müller counter.

For Cs-137, in a 10 min measurement, the decision limit is approximately 26 Bq/m^3 and MDA (Ld) is 35 Bq/m^3.

Table 1 summarizes the detection limits of some nuclides in a 10 min measurement with the $LaBr_3$ detector. For a comparison, the activity concentration increasing the dose rate by 0.1 μSv/h is given. For many of the nuclides the $LaBr_3$ detection limits are better by a factor of 30-100.

Table 1. The $LaBr_3$ spectrometer detection limits for airborne nuclides. For a comparison, the alarm limit (increase of 0.1 μSv/h in the dose rate) for the Geiger-Müller counter is also given.

Nuclide	Gamma energy (keV)	LaBr3 Detection limit (Bq/m^3)	GM-tube alarm limit (Bq/m^3)
Am-241	59.5	200	33000
Co-60	1332	20	250
Cs-137	662	30	1100
I-123	159	50	4000

Discussion

The value of nuclide specific information in addition to dose rate is well demonstrated by an event in Olkiluoto area in November, 2009. At a late evening of 22nd of November starting 23:00 UTC, the dose rate at some of the monitoring stations near the Olkiluoto nuclear power plant site started to rise (Fig.4). The values did not at any time during the event cross the alarm limits. However, the change was noticed at a very early stage and an investigative action was taken. Although the rain sensors and weather data of the monitoring stations indicated a rain shower passing over the area, suggesting radon progenies related event, it was not possible to confirm this conclusively from the dose rate data only. The dose rate reached the peak values at 00:10 UTC on the 23rd of November, with a dose rate of 0.20 μSv/h measured at Eurajoki monitoring station.

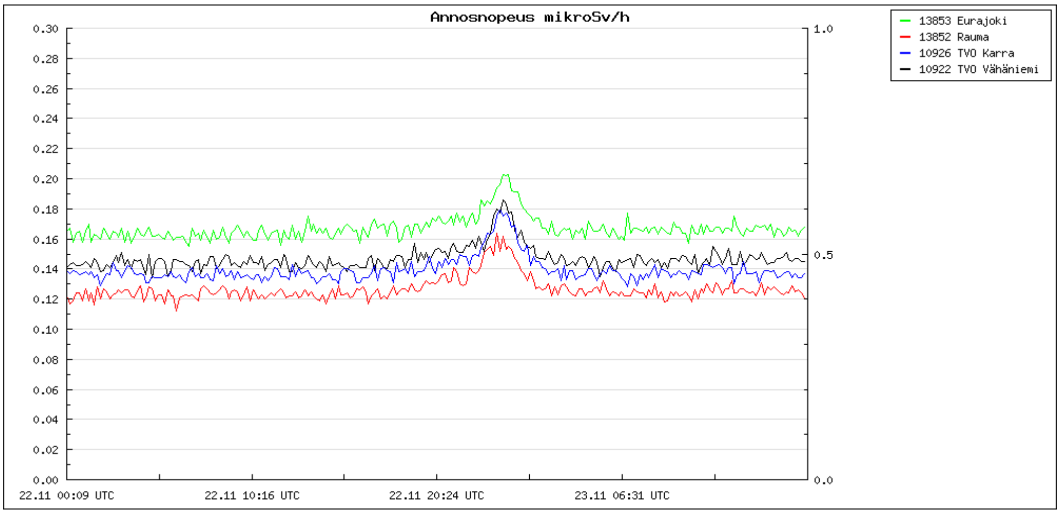

Fig. 4. Dose rates from four monitoring stations around Olkiluoto between 12:00 UTC 22.11.2009 - 12:00 UTC 23.11.2009.

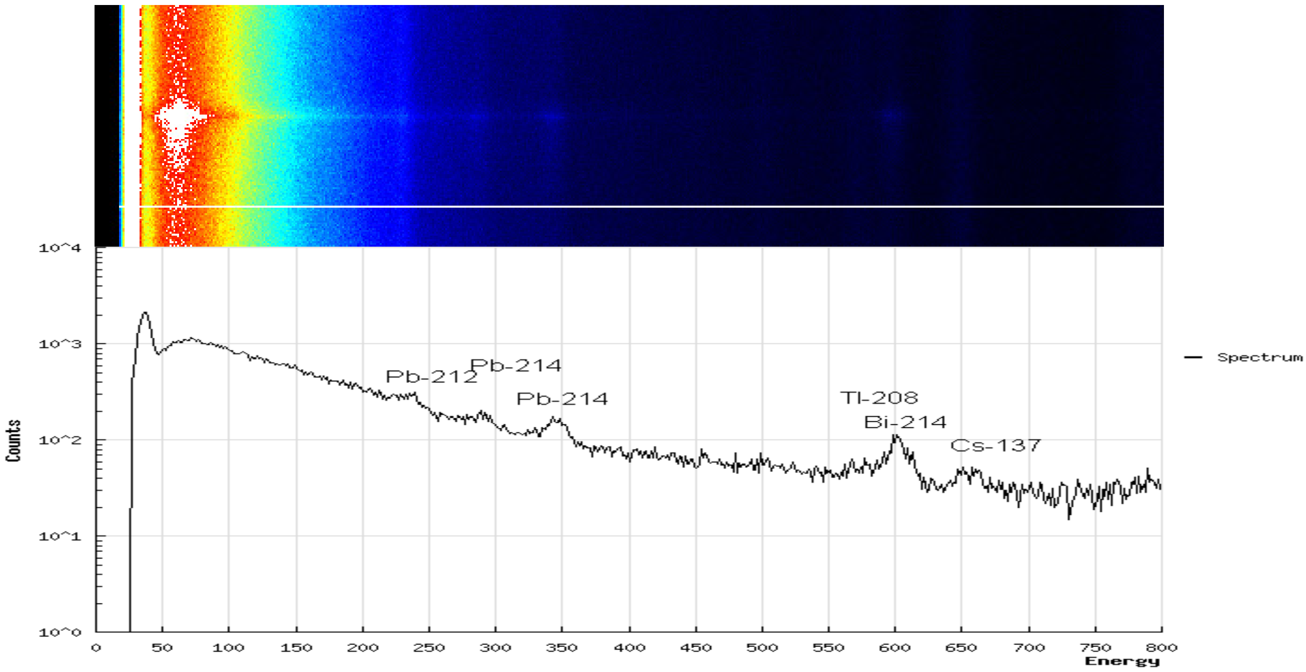

Fig. 5. Waterfall plot of the $LaBr_3$ spectra from Eurajoki monitoring station between 12:00 UTC 22.11.2009 - 12:00 UTC 23.11.2009. The energy scale (x-axis) is from 10 keV to 700 keV, while the y-axis is time. Increase in the intensity of several radon progeny related peaks is observed around midnight. A spectrum recorded at 23:40 UTC shows that the prominent peaks are related to radon progenies.

At that time, the spectra of the $LaBr_3$ detector installed at the Eurajoki station were investigated, and increase in radon progenies was immediately confirmed to be the culprit for the dose rate rise. A waterfall plot of the spectra between 12:00 UTC 22.11.2009 - 12:00 UTC 23.11.2009, shown in fig. 5, reveal several radon progeny related gamma peaks gaining in intensity starting at 22.30 UTC. A single spectrum measured at 23:40 UTC shows that the most prominent peaks can be associated with Pb-212, Pb214 and Tl-208. The waterfall plot clearly shows how the rise and decay in the intensity of these peaks closely follows the dose rate curve. Even though the event did not generate an alarm in the dose rate monitoring network, as the radon-related events almost never because the typical dose increase is below 0.1 µSv/h, the value of $LaBr_3$ scintillators is clearly demonstrated. With the spectroscopic information at hand, the network operator was quickly able to verify the origin of the dose rate increase and confirm the non-alarming nature of the event.

Conclusions

The Finnish dose rate monitoring network has been equipped with 19 $LaBr_3$ spectrometers. These spectrometers have been installed to monitoring stations near the nuclear power plants. This fully automated $LaBr_3$ spectrometer subnetwork provides environmental spectra of excellent quality. Nuclide-specific activity concentrations can be derived from the measured gamma peak count rates. The detection limits for many nuclides are better by a factor of 30-100 compared with the standard Geiger-Müller counters of the monitoring network.

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Sample screening to locate active particles with position-sensitive alpha detector

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Abstract

In order to speed up the analysis of collected samples, new sample screening techniques are required both in nuclear safeguards and nuclear forensics. One promising approach is using a position-sensitive alpha detector. In this work, screening is performed with a Double-Sided Silicon Strip Detector (DSSSD) supported with a novel analysis algorithm. The method is shown to accurately locate the most active particles from a sample containing multiple micrometer-sized particles consisting of weapons-grade plutonium. In addition to the particle locations, the DSSSD also gives the alpha spectrum that can be used to perform a preliminary isotope analysis. Compiling isotope and location information allows further assay methods to be focused only on the spots containing the key isotopes.

Introduction

Analysis of samples containing potentially radioactive particles is a complicated, multi-phased process. Typically, the samples are first screened in order to select the most probable spots. Thereafter, more time-consuming destructive assay methods, such as mass spectrometry, can be used to investigate the selected spots in detail. In increasing the throughput of the analysis without losing any interesting particles, the accuracy of the screening method has a crucial role. In addition to speed and accuracy, it is often essential that the method used is non-destructive to ensure that the sample stays intact for possible subsequent analyses.

Traditional gamma spectrometry is often very inefficient for detecting small amounts of heavy alpha-active isotopes. The gamma yields of such isotopes are typically orders of magnitude smaller than the alpha yields, or the emitted photons have too low energies to be measured. These properties also apply to many isotopes of plutonium and uranium that are of particular interest in nuclear safeguards and nuclear forensics. Thus, to ensure that these isotopes are not ignored during the screening, the screening method cannot only rely on gamma spectrometry but the emitted alphas must be detected as well.

In this work, a new screening method for particles emitting alpha radiation is presented. The particles are located with a position-sensitive Double-Sided Silicon Strip

Detector (DSSSD) and novel software. The performance of the method is investigated by analysing a sample containing multiple micrometer-sized particles from a nuclear bomb. The results are also compared with autoradiography performed to the same sample.

Measurement

The sample analyzed in this work contained pieces of an actinide particle on a filter. The original particle, containing ^{241}Am and ^{239,240}Pu, was isolated from a soil sample collected from Thule, Greenland (Roos et al., 2010) near the location where a US bomber carrying thermonuclear weapons crashed in 1968 (see Eriksson 2002 and references therein). Following chemical treatments with nitric acid and other reagents broke the particle down into several smaller particles. These smaller particles were collected on a filter paper with a pore size of 0.45 µm. All particles in the sample were too small to be visible with the naked eye. To image the sample, it was measured for a few days with the same autoradiography method as in Pöllänen et al. 2004. The autoradiography image obtained is shown in Fig. 1. As can be seen from the image, there are several particles in the sample, two of them having a significantly higher activity than the others.

The measurement of the sample was performed with the alpha-gamma coincidence setup of Particle And Non-Destructive Analysis (PANDA) testbed (Turunen et. al 2010). This setup contains a broad-energy HPGe gamma detector and a position-sensitive alpha detector connected to an event-mode data acquisition system. However, in this work, only the data from the alpha detector were utilized. The alpha detector in PANDA is a Double-Sided Silicon Strip Detector, consisting of 32 active strips in x and y directions. The width of each stripe is 2 mm, and thus, the stripes generate a 64×64-mm² grid consisting of pixels with a size of 2×2 mm². The typical energy resolution of this type of detectors for alphas from ^{241}Am is 55 keV (Micron 2010).

For the measurement, the sample was set inside the measurement chamber of PANDA, and the pressure inside the chamber was pumped to less than 10^{-5} mbar. The surface of the sample matrix was parallel to the alpha detector, and the distance from the detector to the sample was 3.5 ± 0.5 mm. The sample was located approximately at the centre line of the DSSSD. The sample was measured for one week, and the number of accepted alpha hits used in the analysis was about 33 000.

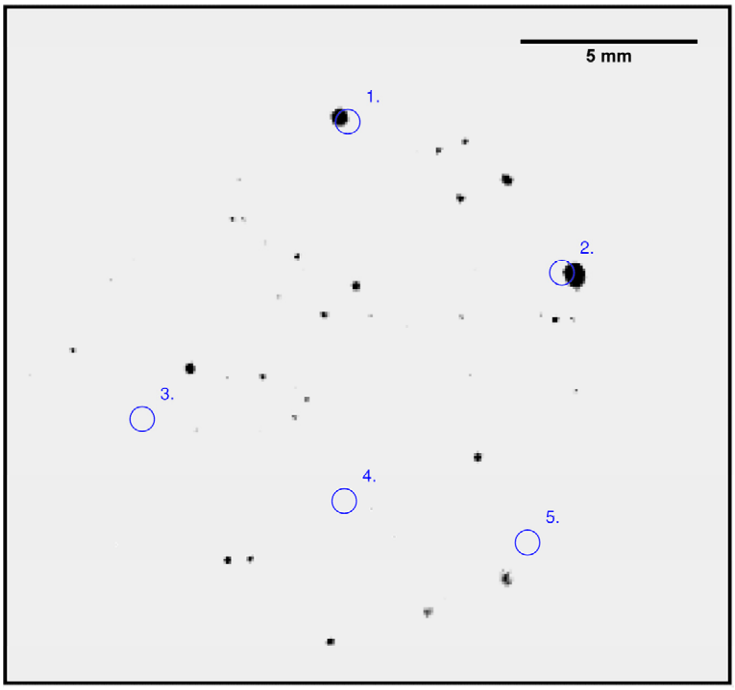

Fig. 1. Autoradiography image of the sample. The blue circles refer to the particle locations obtained with automated analysis (see Table 1) .

Methods

Let us assume that the DSSSD detects every alpha particle hitting the front surface of the detector. In this case, the absolute detection efficiency of each pixel (ε_{abs}) only depends on the solid angle of the pixel (Ω) seen from the source location:

$$\varepsilon_{\text{abs}} = \Omega/(4\pi).$$

For a rectangular pixel, Ω can be analytically calculated with the following equation

$$\begin{aligned}
\Omega = {} & \arctan\left(\frac{(x_2-x_p)(y_2-y_p)}{z_p\left[(x_2-x_p)^2+(y_2-y_p)^2+z_p^2\right]^{1/2}}\right) \\
& -\arctan\left(\frac{(x_1-x_p)(y_2-y_p)}{z_p\left[(x_1-x_p)^2+(y_2-y_p)^2+z_p^2\right]^{1/2}}\right) \\
& -\arctan\left(\frac{(x_2-x_p)(y_1-y_p)}{z_p\left[(x_2-x_p)^2+(y_1-y_p)^2+z_p^2\right]^{1/2}}\right) \\
& +\arctan\left(\frac{(x_1-x_p)(y_1-y_p)}{z_p\left[(x_1-x_p)^2+(y_1-y_p)^2+z_p^2\right]^{1/2}}\right)
\end{aligned}$$

where (x_p, y_p, z_p) is the location of the source and $(x_1, y_1, 0)$ $(x_2, y_2, 0)$ locations of the opposite corners of the pixel (Gotoh et al. 1971). If the location of the source and number of alphas emitted (C) are known, an estimate for the number of alphas detected with one pixel can be calculated as

$$c = C\Omega(x_p, y_p, z_p)/(4\pi).$$

The location and activity of a particle can be determined by finding the parameters that maximize the probability of the detected counts to be produced by the model. Due to the small pixel size of the DSSSD and the low activity of the sample, the number of counts on many pixels is small. Thus, the number of counts does not follow Gaussian statistics, and only properties of the Poisson distribution can be used.

Results

Fig. 2 shows the measured hitmap (counts per pixel). As can be seen from the figure, without further analysis, only one particle can be clearly detected from the raw data. However, by analysing the spectrum with the software designed, the detected hitmap can be explained with five point sources. The parameters of the particles that explain best the shape of the hitmap are shown in Table 1. Fig. 3 shows a hitmap that was simulated by using a set of particle positions and activities obtained from the fit. Comparison of Fig. 2 and Fig. 3 reveals that the simulated hitmap resembles the measured one.

The locations of the fitted particles are also shown in Fig. 1 together with the autoradiography image of the sample. This figure reveals that the automated analysis of the DSSSD data successfully located two particles. The other particles found are not real but describe the shapes in the hitmap generated by multiple small particles close to each other.

Since DSSSD also has a relatively good energy resolution, the measurement produces not only the alpha hitmap but also the alpha spectrum of each pixel. Fig. 4 shows an alpha spectrum obtained by summing the spectra from the pixels near the active particles in the sample. This spectrum already reveals that the particles in the sample probably contain ^{241}Am and ^{239}Pu or ^{240}Pu. If more detailed information of the isotopes is required, the spectrum could be deconvoluted with Adam (see Vesterbacka et al. 2010).

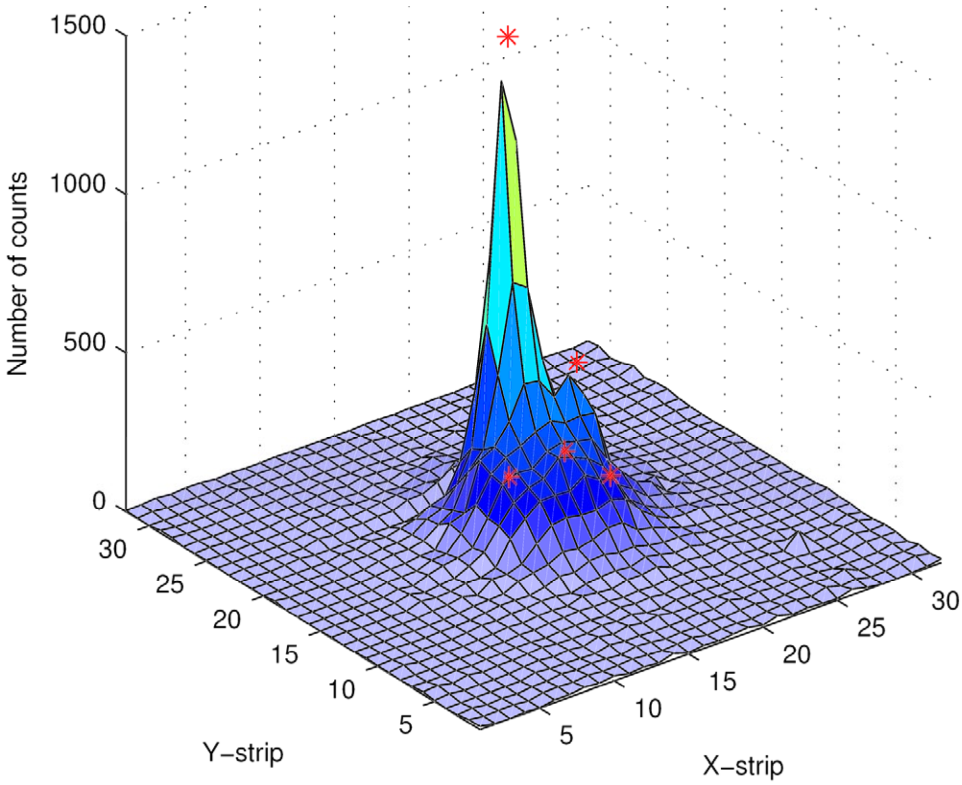

Fig. 2. Measured hitmap and locations of the fitted particles.

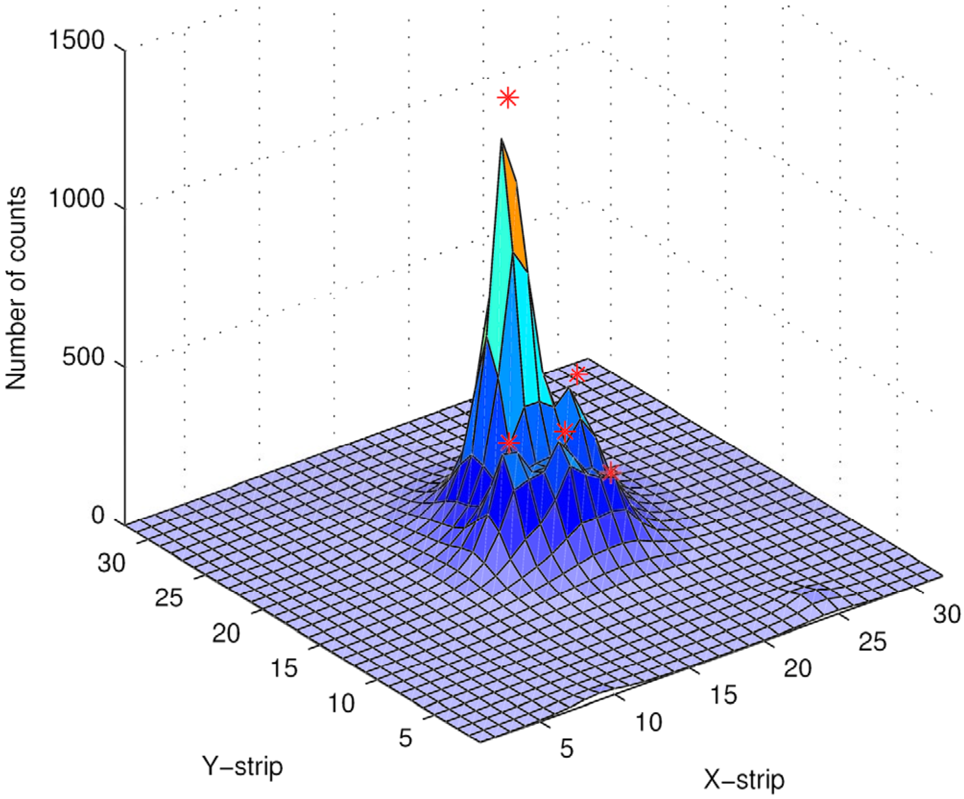

Fig. 3. Simulated hitmap and locations of the fitted particles.

Table 1. Parameters for the fitted particles. The origin of the coordinate system is located at the centre of the detector.

Particle ID	Number of alphas, C	X-coordinate, x_p (mm)	Y-coordinate, y_p (mm)
1	39884	6.8	1.7
2	10151	2.4	7.6
3	9970	-1.3	-4.4
4	9094	-3.8	1.3
5	5086	-5.1	6.4

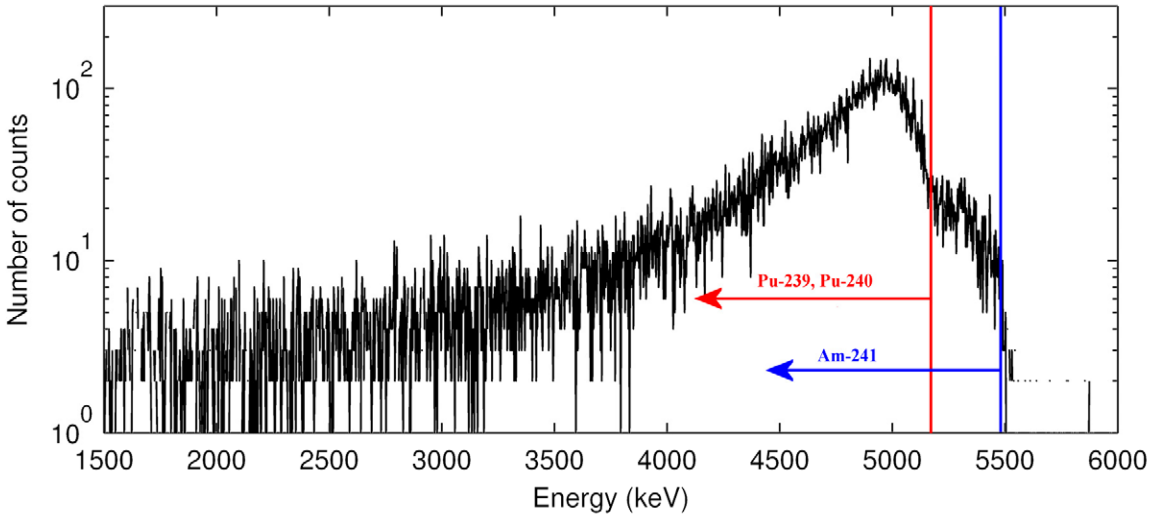

Fig. 4. Summed alpha spectrum from the pixels near the active particles.

Discussion and conclusions

Our research clearly showed that a Double-Sided Silicon Strip Detector can be used to locate alpha-active particles with very low activity. From the analysed sample, the method was able find one particle having a significantly higher activity than the other particles in the same sample. In addition, it was clearly shown that this one particle cannot explain all counts detected but the sample must contain multiple particles. The locations of the other particles given by the analysis software are not as precise but can still be used to determine the most interesting spots for more comprehensive analysis.

Compared to traditional autoradiography, DSSSD has several advantages for sample screening. First, DSSSD does not only give the locations of the particles but also accurate estimates for their activities. Second, the particle locations from the measured data are obtained fully automatically. Third, even though the pixels of the DSSSD are typically very large compared to the pixels in autoradiography images, preliminary tests have shown that with a dedicated location algorithm, the particles can easily be located with an accuracy of less than the pixel size (here 2×2 mm²).

Fourth, unlike in autoradiography, DSSSD also measures the alpha spectrum that can be used to find out the isotopes of the particles. Comparing the alpha spectra of different pixels, even differences in the isotope composition between the particles can be detected. To determine the isotope compositions of the particles, the data from the gamma detector in PANDA can be utilized as well. For example, generating an alpha hitmap only of those counts in coincidence with 59.5 keV gammas from ^{241}Am may be very helpful in locating particles containing plutonium.

On the other hand, using DSSSD to locate particles has also some limitations compared to autoradiography. The greatest disadvantage of DSSSD is the relatively large pixel size that makes it impossible to distinguish particles close to each other. In addition, since DSSSD can only be operated in a high vacuum and each strip of the detector must be read individually, the measurement system is fairly complex. However, there are already similar detectors on the market with a pixel size as small as 400×400 μm² (Micron 2010). Therefore, the locations could be determined with significantly higher accuracy than what is presented in this work.

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Position-sensitive measurement system for non-destructive analysis

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Abstract

The Finnish Radiation and Nuclear Safety Authority, STUK, is conducting research on non-destructive sample analysis. One of the main tools used for this research is called PANDA (Particles And Non-Destructive Analysis).

Rather than just being a device tailored for a specific use, PANDA is more like a development platform. It has two vacuum chambers; the first is for loading and changing the samples and the second for the measurements. The currently operational measurement set-up consists of an HPGe detector to detect gamma radiation and a Double-Sided Silicon Strip Detector (DSSSD) to detect alpha particles. These detectors are placed face to face in PANDA's measurement position 1 (MP1). Samples are transported from the loading chamber and placed in the middle of these detectors using a linear feedthrough. The distance between the HPGe and DSSSD is about 8 mm. This setup allows simultaneous alpha-particle screening of large-area samples (around 40 cm^2), position-sensitive alpha-gamma and gamma-alpha coincidence studies and various half-life investigations. For example, alpha-gamma coincidence technique provides nearly background-free gamma spectra of alpha-decaying nuclides.

After the sample is screened in MP1, it can be transported to the measurement position 2 (MP2) where the interesting parts of the sample can be further investigated. At first, MP2 will host a single small-area (10 mm^2) silicon drift detector that has superior energy resolution as compared to the detectors of MP1. The main focus of MP2 will be in detecting conversion electrons, alpha particles and X-rays.

The data of MP1 are recorded in event mode, i.e. all events are timestamped. The resulting binary event files are then converted to xml-format and transferred to a database. Various analysis algorithms are currently being developed to facilitate the analysis process. The performance of PANDA is presented through the measurement and analysis of various environmental samples.

Introduction

A novel device for non-destructive sample analysis has been designed and built [1, 2]. This device is called PANDA (Particles And Non-Destructive Analysis). PANDA broadens measurements and analysis prospects of radioactive samples by applying techniques and instruments typical of basic research. These are, for example, double-sided silicon strip detectors, event mode data acquisition and software-based coincidences.

PANDA

PANDA is not only a measurement setup but rather a development platform to study different methods to measure and analyze radioactive samples. Two vacuum chambers are installed side by side on an aluminium pedestal (Fig. 1). The chambers are connected to each other with a gate valve. One chamber is for loading the samples and the other for making the measurements. The samples are transported from the loading chamber to the measurement chamber with a linear feedthrough.

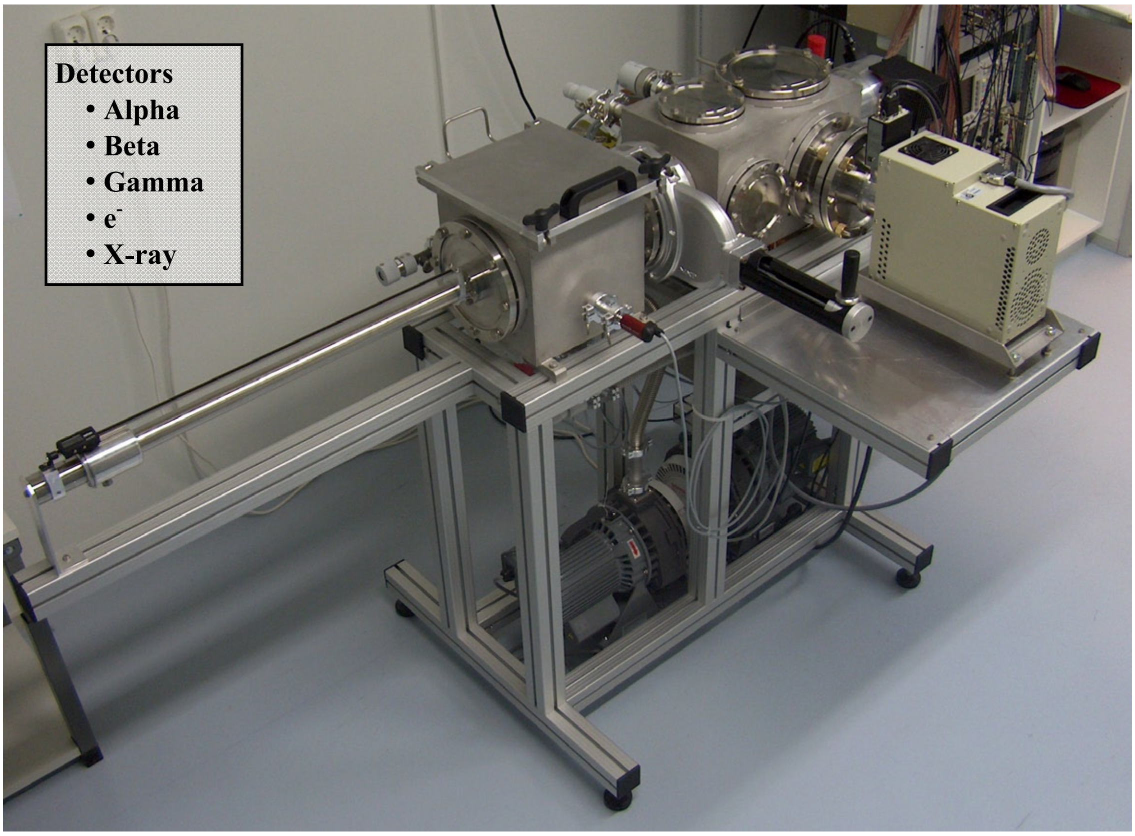

Fig. 1. The PANDA measurement setup.

The measurement chamber has two measurement positions. Measurement position 1 (MP1) is primarily meant for screening of large-area samples such as air filters. It hosts a High Purity Germanium (HPGe) detector and a Double-Sided Silicon Strip Detector (DSSSD) facing each other. The HPGe is used to detect photons and the DSSSD is intended for alpha particles. The samples are transported with the linear feedthrough between the two detectors, see Fig. 2. The source-to-detector distance can be adjusted from a few millimetres to a few centimetres to fit different sample types.

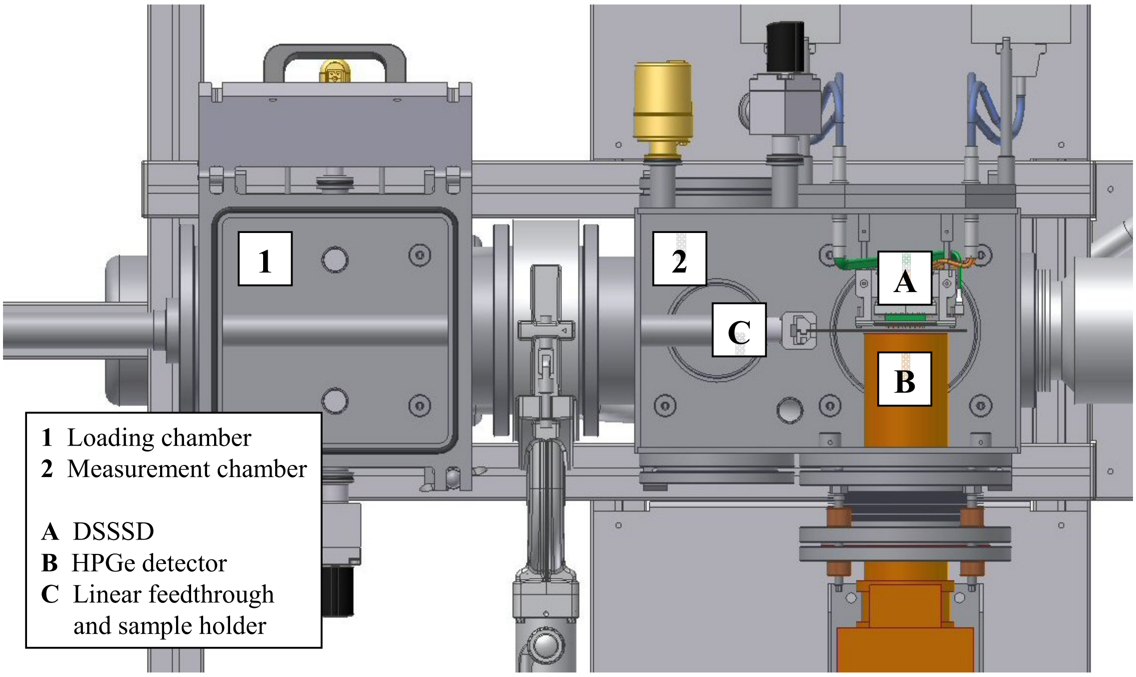

Fig. 2. Top view of PANDA's vacuum chambers.

The position sensitive DSSSD is used to locate interesting radioactive particles. It has 32 front and 32 back strips. Each strip is 2 mm wide and 64 mm long, so the active area of the DSSSD is 64 x 64 mm^2. It can be considered as 1024 individual pixels with a pixel size of 2 x 2 mm^2. After the location and partial analysis of the particles in MP1, the further analyses can be continued in measurement position 2 (MP2). It is devoted to the studies of individual radioactive particles and chemically processed samples. The first version of MP2 has a small silicon drift detector that can be used to detect conversion electrons, alpha particles and X-rays. This detector has an excellent energy resolution, about four times better for low energy photons as compared to the HPGe of MP1. Later on, MP2 will be expanded to include two or more detectors.

PANDA records data in event mode and the events are timestamped. This provides a lot of flexibility and additional possibilities to the data analysis. It enables, for example, to create alpha-gated gamma spectra that are nearly free of background [1]. Analysis of such spectra is simplified, since all gamma peaks are created by the alpha-decaying nuclides. In the near future similar gating can be done to beta-emitting radionuclides as well. This will be done by adding a third detector to MP1. The data are stored to a database and various analysis algorithms are being developed to further improve and intensify the analysis sequence.

Air filter measurement with PANDA

For performance evaluation, PANDA is tested with different kinds of samples. For initial tests, a measurement with an air sample was carried out (Fig. 3). The following results are taken from [2]. The air sample was collected to a Fluoropore filter with a small commercial sampler. The collection time was 2.2 h and the total air flow through the filter was 12.7 m^3. The measurement with PANDA was started 0.5 h after the sample collection ended. The total acquisition time was 3 d 5 h.

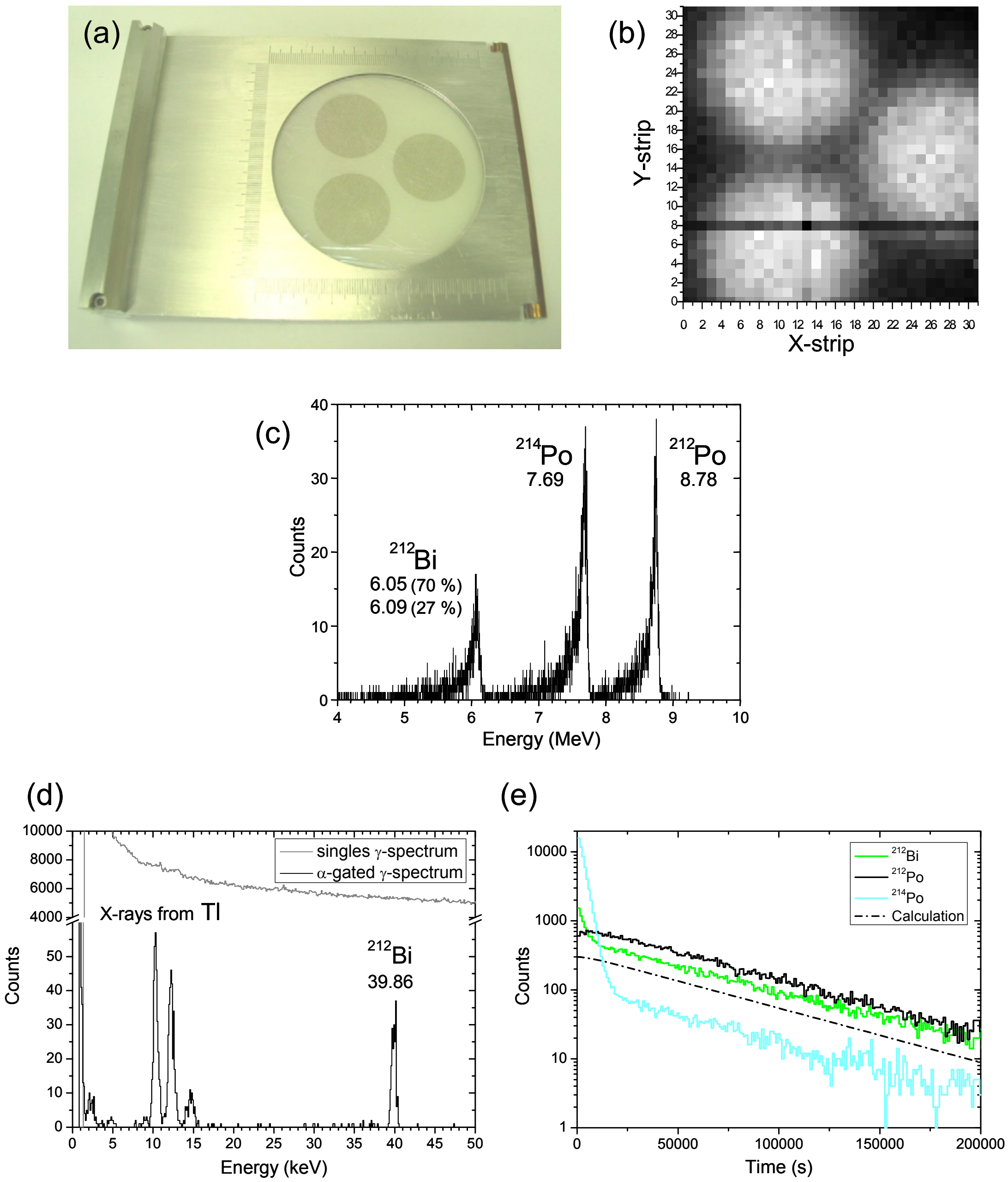

Fig. 3. An air sample measurement made with PANDA. (a) Air filter installed to a sample holder. (b) XY-hitmap of the DSSSD. (c) Alpha spectrum from a single strip. The alpha peaks are from radon daughters ^{212}Bi, ^{214}Po and ^{212}Po. (d) Conventional gamma spectrum and an alpha-gated gamma spectrum made using data from ^{212}Bi alphas. (e) Time behaviour of ^{212}Bi, ^{212}Po and ^{214}Po activity and a theoretical curve showing the shape of the ^{212}Bi and ^{212}Po. Figures 3b-e are taken from [2].

The air filter installed to a sample holder is presented in Fig. 3a. There is a 0.5 µm Mylar foil in front of the filter to protect the DSSSD from contamination. Figure 3b shows an alpha XY-hitmap recorded by the DSSSD. The highest alpha particle count rate is in the brightest pixels. An alpha spectrum of a single strip is shown in Fig. 3c. Three significant alpha peaks belonging to radon daughters ^{212}Bi, ^{214}Po and ^{212}Po are clearly visible.

Figure 3d presents both the singles and the ^{212}Bi alpha-gated gamma spectrum. The alpha gate drastically reduces the background. In the alpha-gated gamma spectrum the 39.9 keV gamma peak related to ^{212}Bi alpha decay and Tl X-rays are easily visible.

With the timestamp data it is possible to study the time behaviour of interesting nuclides in the samples. The time behaviour of ^{212}Bi, ^{212}Po and ^{214}Po activities are shown in Fig. 3e. The graphs are made by gating the timestamp data with the three energy regions where the alpha peaks are located, see Fig. 3c. In the present case, the tailing of the two peaks with the highest energy corrupt the data from the peaks below them. The shape of the time behaviour of ^{212}Po agrees well with the calculated curve since it is not affected by other nuclides. The shape of the calculated curve comes from ^{212}Pb which is the grandmother nuclide of ^{212}Po. Note that since ^{212}Pb is a beta decaying nuclide it is therefore not directly visible in the recorded alpha spectrum. The count rate in the ^{214}Po region is initially dominated by genuine ^{214}Po counts (effective half-life 26.8 min). Later, tailing effects from the ^{212}Po peak (effective half-life 10.64 h) takes over. The ^{212}Bi count rate initially shows contributions from the ^{214}Po peak tails which quickly decay off and show the genuine ^{212}Bi count rate (with a very little fraction attributable to ^{212}Po tailing counts). The flat cap of the ^{212}Po count rate, however, shows the delay in reaching the equilibrium between ^{212}Pb -> ^{212}Bi -> ^{212}Po.

Impactor sample measurements with PANDA

The second sample used here to demonstrate PANDA's abilities is a sample from air monitoring of Thorium with a Berner low-pressure impactor. Figure 4 presents three different measurements of the same sample. In Fig. 4a there is an image of the sample made using autoradiography technique. As seen, the sample consists of 26 active spots that form a spiral shape (diameter ~ 23-28 mm, distance between consecutive spots ~ 3 mm). The same number of spots in the same formation is also clearly visible to the naked eye just by looking at the sample's surface.

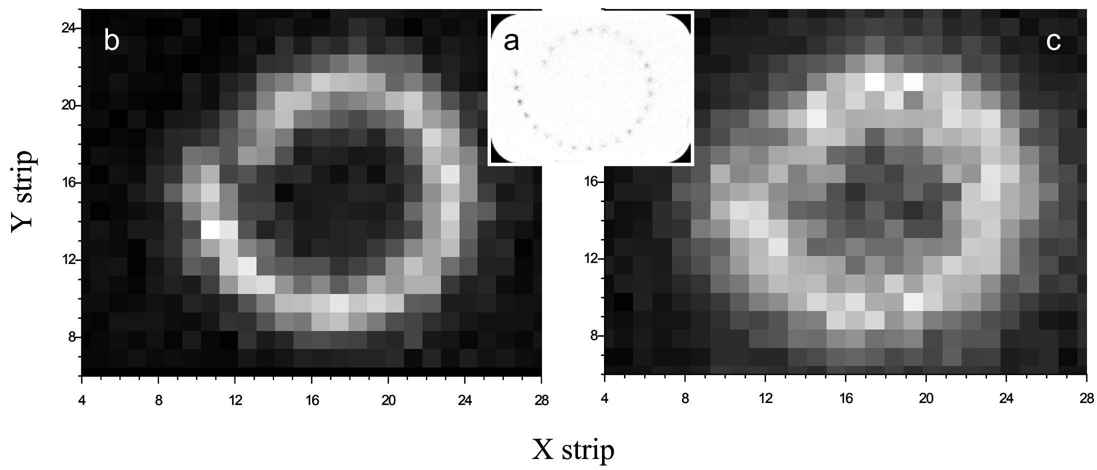

Fig. 4. Impactor sample. (a) Measured with autoradiography technique. (b) XY-hitmap of the DSSSD when sdd is 3 mm. (c) XY-hitmap of the DSSSD when sdd is 4 mm.

Figures 4b and 4c show two alpha particle hitmaps measured using PANDA's DSSSD. The source-to-detector distances (sdd) used were 3 mm and 4 mm, respectively. These hitmaps show only a portion of the DSSSD that is close to the size of the imaging plate shown in Fig. 4a.

As can be seen from the two hitmaps in Fig. 4, the quality of the image obtained with the DSSSD is strongly depended on the source-to-detector distance. There are limitations to how close to the detectors the samples can be installed in PANDA and, at the moment, 3 mm is close to the minimum. Also, since the pixel size (2 mm * 2 mm) in the DSSSD is quite large, it is very difficult to distinguish particles located only a few millimetres apart. When comparing Fig. 4b and 4c, it is clear that even one millimetre in the source-to-detector distance can make a great difference. More about finding and locating particles are discussed in Ihantola et al. [3].

Discussion

With the alpha-gamma coincidence technique, it is possible to reduce the background to nearly zero and improve the detection limits greatly. For example, PANDA is able to detect and locate a particle containing 10^{-9} g of $^{239,240}Pu$ in a few minutes [1]. Notice that the interesting alpha region for Pu and U is between 4 to 6 MeV and, as shown, the contribution of the radon daughters to the alpha spectra is mainly at energies above 6 MeV.

Using the DSSSD in a close geometry instead of a single large area silicon detector reduces the evenly distributed, naturally occurring alpha background significantly. This is because it is possible to use only the pixels right above an interesting spot of the sample in the analysis, i.e., the background recorded by these pixels is only a fraction of the background recorded by the entire detector [1].

As demonstrated, PANDA is a measurement setup that can, for example, greatly improve the analysis of air samples in Certified Laboratories. However, the sampling and subsequent sample preparation must be non-destructive, similar to the sample treatment at Cinderella-type air sampling stations [4].

Acknowledgements

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Application of the MKGB-01 spectrometer-radiometer in the KX-gamma coincidences setup at the D. I. Mendeleyev Institute for Metrology

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Abstract

The prototype of setup that implements of KX-gamma coincidences technique developed at the ionizing radiation department of the D.I. Mendeleyev Institute for Metrology is described. The basic element of setup is MKGB-01 spectrometer-radiometer (STC "Radek"). MKGB-01 units - NaI(Tl) crystals of 100 µm x 40 mm and 80 x 80 mm are used for photon radiation detection. The standard set of “CAMAC” units is used as a secondary electronic part. Original design multi-channel counting system is used for counting rate measurements. The main metrological characteristics of the setup model were determined: a long-term stability, background, dead time, resolution time.

Using the described setup the electron-capture radionuclides ^{54}Mn, ^{57}Co, ^{65}Zn, ^{88}Y, ^{139}Ce activity was measured. The combined uncertainty (K=2) of activity was estimated in the range 0,5-1,0 %. Using KX-KX coincidences technique ^{125}I activity was measured with combined uncertainty of 1,0 % (K=2). The results obtained are in good coincidence with international comparisons results.

The main result of investigations is a conception of a new KX-gamma coincidence setup based of the MKGB-01 serial spectrometer-radiometer. A number of requirements for the secondary electronic units have been declared, a device for source positioning has been designed and an achievable uncertainty of the KX-gamma coincidence method has been evaluated. The supplementary investigation for determination of possibility to use the MKGB-01 spectrometer-radiometer for realization in beta-gamma coincidence technique.

Introduction

The photon sources based on ^{54}Mn, ^{57}Co, ^{65}Zn, ^{88}Y, ^{139}Ce and ^{125}I radionuclides are used for spectrometers calibration, applied in radio-ecological measurements. Moreover, ^{125}I based sources are applying in nuclear medicine. These radionuclide disintegrates via electron capture (1) which is accompanied by characteristic radiation from the K, L shells, etc., and the excited state of daughter atom without delay becomes stable with gamma radiation emission (accompanied by the conversion characteristic radiation). The positron emission can also be present.

The main problem is the activity measurement of radionuclides in covered sources, when it is impossible to use beta-gamma coincidence methods (2). X-ray and γ-radiation coincidence method is practically the only method to measure radionuclides activity directly in the covered source.

In our opinion, the most appropriate, is the coincidences of KX and γ-radiation. The detector with thin crystal and Be entrance window was used for KX-radiation, and detector with large crystal and Al entrance window for γ-radiation. The crystals design provides registration mainly only one type of radiation in each detector, and the entrance windows construction excludes registration of KX-KX and KX-LX coincidences. The ^{125}I activity was measured by KX-KX coincidences method using two thin crystals NaI(Tl) (3).

Materials and methods

STC «Radek» produces spectrometer-radiometer MKGB-01, equiped by several detecting units of different type. In the ionizing radiation department at the D. I. Mendeleyev Institute for Metrology a prototype of setup for realisation of KX–γ и KX-KX coincidences method based on this commercially available device has been constructed.

The detecting units based on NaI(Tl) crystals 100 mkm*40 mm and 80*80 mm respectively were used for registration of KX and γ-radiation. For realization of KX-KX coincidences method two NaI(Tl) crystals, 40 mm diameter and 100 mkm thickness were used. Background radiation was shield by using 10 mm thickness lead screen. The prototype block-diagram is shown on Fig.1.

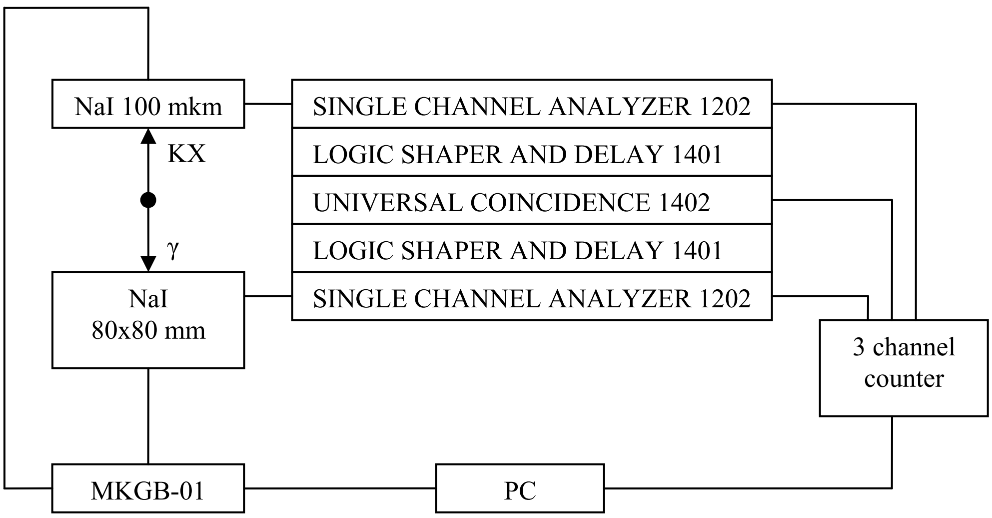

Fig.1. Prototype of setup block-diagram.

The control of spectrometer operating modes (PMT high voltage, preamplifier gain factors and etc.) was carried out using the MKGB-01 firmware.
The registration system is created using set of CAMAC module and consist of single-channel analyzer, pulse shaper and coincidence unit. X-rays, gamma-rays and coincidence counting rates are measured using original design multichannel counting system.

Before making the measurements the main characteristics of the measurement system were determined. These characteristics are following:

Warming time is the time interval, after which the device readings satisfy the requirement:

$$\left|N_i - \bar{N}\right| \leq 3 \cdot S(\bar{N})$$

Where N_i is the device reading of the i-th measurement; $\bar{N}$ is the mean value of the device readings calculated as a result of m measurements performed two hours after the device has been turned on; $S(\bar{N})$ is the standard deviation of the mean value calculated by equation:

$$S(\bar{N}) = \sqrt{\frac{\sum (Ni - \bar{N})}{\sum (m-1)}}$$

Reproducibility after turning off is the relative deviation of the device reading after it has been turned off and turned on again at the unchanging (invariable, constant, permanent) irradiation field:

$$R = \frac{\left|N_i - N_j\right|}{N_i} \cdot 100\%\,,$$

where N_i and N_j are the device readings before and after turning off.

Instability during the working day is the maximum deviation of detector readings from the mean value during the working day:

$$D = \frac{\max\left|N_i - \bar{N}\right|}{\bar{N}} \cdot 100\%$$

Temperature dependence of sensitivity is the variation of device readings when the environment temperature is changing :

$$k(t) = \frac{N(t) - N(t_o)}{t - t_o} \cdot 100\%\,,$$

where $N(t)$ is the device reading when the environment temperature is t°C; $N(t_o)$ is the device reading when the environment temperature is t_o°C.

"Dead"time of the devices was determined using the "two sources technique":

- gamma or X-ray source was placed at a fixed point to provide the device counting rate of about 10^4 s^{-1}, the counting rate N_1 was measured;
- device and source remained immovable, the other source was placed in a position providing to make the counting rate $N_{12} \sim 2N_1$, then the counting rate N_{12} was measured;
- counting rate N_2 with the first source moved out was measured.

The "dead" time value τ was calculated by equation :

$$\tau = \frac{1}{N_{12}}\{1 - \sqrt{\frac{(N_{12} - N_1)(N_{12} - N2)}{N_1 N_2}}\}$$

The following results were obtained for the described prototype.

For X-ray channel:

- warming time is less than 30 min;
- reproducibility after turning off does not exceed 0.1 %;
- instability during the working day does not exceed 0.1%;
- temperature dependence of the sensitivity is 0.05%/°C;
- "dead" time $\tau = 1.21 \pm 0.01$ μs for first KX-detector
- "dead" time $\tau = 1.44 \pm 0.02$ μs for second KX-detector

For gamma-ray channel:

- warming time is less than 30 min;
- reproducibility after turning off does not exceed 0.1 %;
- instability during the working day does not exceed 0.1%;
- temperature dependence of the sensitivity is 0.05%/°C;
- "dead" time $\tau = 2.42 \pm 0.02$ μs.

The coincidence system resolving time τ_p was measured using a pulse generator and is equal 0,514, 1,004 and 2,018 μs for 1402 unit modes «½», «1» and «2» respectively. These τ_p values provided the random coincidence contribution less than 10% of coincidence counting rate.

The final result - activity value was calculated according to Campion (4) and Brian (5) equations.

Measurement modes selection.

The setup operating modes were the follow:

- The source-detector distance in any cases should minimize of summary events probability.
- The KX-channel energy «window» is configured to register the KX-radiation only. In γ-channel the integral operating mode with lower threshold higher than KX-radiation energy is used. It is accepted, that the KX-radiation registration efficiency by γ-detector is close to zero.
- The γ-channel lower threshold for ^{54}Mn and ^{88}Y activity measuring is adjusted above 255 keV to avoid coincidence registration with back scattered radiation.
- The γ-channel lower threshold for ^{65}Zn activity measuring is adjusted above 511 keV to avoid coincidence registration with annihilation radiation.
- The γ-channel lower threshold for ^{57}Co and ^{139}Ce activity measuring is adjusted in energy 80 keV and 100 keV respectively.
- For measuring ^{125}I activity in both KX-channels, the integral operating mode with lower threshold higher than the radiation emission peak (≥15 keV) is used.

Derivation of measurement equations

In general the measurement equations system for electron-capture radionuclide accompanied by internal conversion of γ–transition (^{125}I, for example), taking into account the summary events in each channel, will be follows:

$$\begin{cases} N_1 = N_0 \cdot (\varepsilon_{\gamma 1} I_\gamma + \varepsilon_{kx1}(I_{ec} + I_{ic}) - \varepsilon_{\gamma 1} I_\gamma \varepsilon_{kx1} I_{ec} - \varepsilon_{kx1}^2 I_{ec} I_{ic}) \\ N_2 = N_0 \cdot (\varepsilon_{\gamma 2} I_\gamma + \varepsilon_{kx2}(I_{ec} + I_{ic}) - \varepsilon_{\gamma 2} I_\gamma \varepsilon_{kx2} I_{ec} - \varepsilon_{kx2}^2 I_{ec} I_{ic}) \\ N_c = N_0 \cdot (\varepsilon_{\gamma 1} I_\gamma \varepsilon_{kx2} I_{ec} + \varepsilon_{\gamma 2} I_\gamma \varepsilon_{kx1} I_{ec} + 2\varepsilon_{kx1} I_{ec} \varepsilon_{kx2} I_{ic}) \end{cases} \qquad (1)$$

Where: $N_1; N_2; N_c$ - pulse counting rates in γ-channel, KX channel and in coincidence channel, $\varepsilon_{\gamma 1}; \varepsilon_{\gamma 2}; \varepsilon_{kx1}; \varepsilon_{kx2}$ - efficiencies of γ and KX-detector to γ and KX-radiation; I_γ - emission probability of γ-radiation, $I_{ec} = P_k \omega_k$ - emission probability of KX-radiation due to electron capture, $I_{ic} = (\alpha_k \omega_k)/(1+\alpha_t)$ - emission probability of KX-radiation due to internal conversion. The coincidences with LX – radiation are not considered for reasons described above.

In conditions of summary events low probability ($\varepsilon_{\gamma 1}; \varepsilon_{\gamma 2}; \varepsilon_{kx1}; \varepsilon_{kx2}$<5%) the equations (1) are simplified to (2).

$$\begin{cases} N_1 = N_0 \cdot (\varepsilon_{\gamma 1} I_\gamma + \varepsilon_{kx1}(I_{ec} + I_{ic})) \\ N_2 = N_0 \cdot (\varepsilon_{\gamma 2} I_\gamma + \varepsilon_{kx2}(I_{ec} + I_{ic})) \\ N_c = N_0 \cdot (\varepsilon_{\gamma 1} I_\gamma \varepsilon_{kx2} I_{ec} + \varepsilon_{\gamma 2} I_\gamma \varepsilon_{kx1} I_{ec} + 2\varepsilon_{kx1} I_{ec} \varepsilon_{kx2} I_{ic}) \end{cases} \qquad (2)$$

The contribution of γ-radiation to KX-detector $N_2^\gamma = N_0 \cdot \varepsilon_{\gamma 2} I_\gamma$ is defined using aluminium filters (1mm for ^{54}Mn, ^{57}Co and ^{65}Zn and 5mm for ^{88}Y) and copper filter (1 mm for ^{139}Ce).

Modifying the equations (2) and computing I_γ, I_{ec} and I_{ic} from the data (1), we obtain measurements equation for each radionuclide. The numerical value of corrections standard uncertainty is indicated in brackets.

For ^{54}Mn

$$\frac{N_1 \cdot (N_2 - N_2^\gamma)}{N_c} = N_0 \qquad (3)$$

For ^{65}Zn 1115 keV γ-transition only is taken into account.

$$\frac{N_1 \cdot (N_2 - N_2^\gamma)}{N_c} = N_0 \cdot \frac{(a \cdot I_{ec}^a + b \cdot I_{ec}^b)}{I_{ec}^a} = N_0 \cdot 0.989(9) \qquad (4)$$

Where: $a; b$ - probability of electron capture to the 2 and 0 level for ^{65}Zn,

$I_{ec}^a; I_{ec}^b$ - emission probability of KX-radiation due to of electron capture to the 2 and 0 level.

For ^{139}Ce

$$\frac{N_1 \cdot (N_2 - N_2^\gamma)}{N_c} = N_0 \cdot \frac{(I_{ec} + I_{ic})}{I_{ec}} = N_0 \cdot 1.239(5) \qquad (5)$$

For ^{88}Y 898 keV and 1836 keV γ-transitions are taken into account.

$$\frac{N_1 \cdot (N_2 - N_2^{\gamma})}{N_c} = N_0 \cdot \frac{(a \cdot I_{\gamma}^{898} + X \cdot I_{\gamma}^{1836}) \cdot (a \cdot I_{ec}^{a} + b \cdot I_{ec}^{b})}{(a \cdot I_{ec}^{a} \cdot (I_{\gamma}^{898} + X \cdot I_{\gamma}^{1836}) + b \cdot I_{ec}^{b} \cdot X \cdot I_{\gamma}^{1836})} = N_0 \cdot 0{,}989(10) \quad (6)$$

Where: $a;b$ - electron capture probability to the 2 and 1 level for ^{88}Y, $I_{ec}^{a};I_{ec}^{b}$ - emission probability of KX-radiation due to electron capture to the 2 and 1 level, $I_{\gamma}^{898};I_{\gamma}^{1836}$ - γ-radiation emission probability with energies 898 keV and 1836 keV, $X = \varepsilon_{\gamma 1}^{898} / \varepsilon_{\gamma 1}^{1836}$ - ratio of γ-radiation registration efficiency with energy 898 keV to the γ-radiation registration efficiency with energy 1836 keV in the selected «window» (≥400 keV).

The value of X=1,00(1) was calculated by Monte-Carlo method (6) for selected distances source-detector. The maximum contribution of X in correction standard uncertainty in the right part of the equation (6) was equal 0,1%.

For ^{125}I

$$\frac{N_1 \cdot N_2}{N_c} = N_0 \cdot \frac{(K_1 \cdot I_{\gamma} + I_{ec} + I_{ic}) \cdot (K_2 \cdot I_{\gamma} + I_{ec} + I_{ic})}{(I_{\gamma} \cdot I_{ec} \cdot (K_1 + K_2) + 2 \cdot I_{ec} \cdot I_{ic})} = N_0 \cdot 2.0064(42) \qquad (7)$$

Where: $K_1 = \varepsilon_{\gamma 1} / \varepsilon_{kx1}; K_2 = \varepsilon_{\gamma 2} / \varepsilon_{kx2}$ - ratio of γ-radiation registration efficiency to the KX-radiation registration efficiency in each KX-detector in the selected energy «window» (≥15 keV).

The value of K_1=K_2 = 1,215(45) were calculated by Monte-Carlo method (6) for selected distances source-detector. The maximum contribution of K_1 and K_2 in correction standard uncertainty in the right part of the equation (7) was equal 0,25%.

For ^{57}Co 14,4, 122 и 136 keV γ-transitions are taken into account.

$$\frac{N_1 \cdot (N_2 - N_2^{\gamma})}{N_c} = N_0 \cdot \frac{(K \cdot a \cdot I_{\gamma}^{136} + b \cdot I_{\gamma}^{122}) \cdot (I_{ec} + a \cdot I_{ic}^{136} + b \cdot I_{ic}^{122} + b \cdot I_{ic}^{14})}{(K \cdot a \cdot I_{ec} \cdot I_{\gamma}^{136} + b \cdot I_{ec} \cdot I_{\gamma}^{122} + b \cdot I_{\gamma}^{122} \cdot I_{ic}^{14})} = N_0 \cdot 1.016(9)$$

(7)

Where: $a;b$ - transition probability from 2 to 0 and from 2 to 1 level for ^{57}Co, $I_{\gamma}^{122};I_{\gamma}^{136}$ - γ-radiation emission probability with energies 122 keV and 136 keV, $I_{ic}^{14};I_{ic}^{122};I_{ic}^{136}$ - KX-radiation emission probability due to γ-radiation internal conversion with energies 14,4 keV, 122 keV and 136 keV, $K = \varepsilon_{\gamma 1}^{136} / \varepsilon_{\gamma 1}^{122}$ - ratio of γ-radiation registration efficiency with energy 136 keV to the γ-radiation registration efficiency with energy 122 keV in the selected energy «window» (≥80 keV).

The value of K=1,060(4) was calculated by Monte-Carlo method (6) for selected distances source-detector. The maximum contribution of K in correction standard uncertainty in the right part of the equation (9) was equal 0,2%

Results

The ^{54}Mn, ^{57}Co, ^{65}Zn, ^{88}Y and ^{139}Ce radionuclide activity in film covered sources OSGI type was measured using KX-γ coincidences method. The results obtained are presented in table 1. In the last column of table 1 the deflection of these results from the same obtained by national activity standard of Russia Federation. The ^{54}Mn №5123 and ^{57}Co №5171 sources are the witnesses of international comparisons (7,8). Other radionuclides activity measurements results was traced to the results of key comparisons (9,10,11).

Table 1. The results.

Nuclide	Source number	Date	KX-γ coincidences activity, Bq	Uc, k=2 %	Activity, measured on the standard, Bq	Uc, k=2 %	Deflection, %
^{54}Mn	5123	01.01.09	12 956	1.0	12 860	1.0	0.8
	41-06	01.01.09	36 363	1.0	36 462	2.0	-0.3
	384-08	01.01.09	72 186	1.1	72 451	2.0	-0.4
^{65}Zn	5201	01.01.09	5 593	2.1	5 524	2.0	1.2
	16-08	01.01.09	41 130	2.1	41 221	2.0	-0.2
	387-08	01.01.09	90 294	2.1	90 236	2.0	0.1
	350-07	01.01.09	171 258	2.2	171 755	2.0	-0.3
^{88}Y	15-08	01.01.09	14 004	2.3	14 172	2.0	-1.2
	24-06	01.01.09	38 236	2.3	38 549	2.0	-0.8
	469-08	01.01.10	97 031	2.3	98 369	2.0	-1.4
^{139}Ce	14-08	01.01.09	18 446	1.8	18 340	2.0	0.6
	47-06	01.01.09	67 000	1.8	66 007	2.0	1.5
	471-06	01.01.10	150 746	1.9	149 330	2.0	1.0
^{57}Co	5171	01.01.09	8 229	2.0	8 108	1.0	1.5
	01	01.01.09	205 224	2.1	206 731	2.0	-0.7

^{125}I activity was measured in 5 film covered sources, prepared from one solution with well known masses. A detailed description of sources preparation procedure is given in (3). The specific activity is calculated as the average value of 5 results. The result obtained is given in comparison with specific activity (table 2), measured on the national activity standard of Russia Federation with KX-γ coincidences method.

Table 2. ^{125}I Specific activity.

Nuclide	Date	KX-γ coincidences specific activity, Bq/mg	Uc, k=2 %	Specific activity, measured on the standard, Bq/mg	Uc, k=2 %	Deflection, %
I-152	21.05.09	3084	2.0	3135	2.0	1.6

Discussion

Table 1 and 2 data analysis shows, results obtained using described prototype are in good agreement with national standard results. It means spectrometer-radiometer MKGB-01 can be used for activity measurements without preliminary calibration.
Moreover setup described makes it possible to measure radionuclides activity directly in mass-produced film covered sources (12).

On the basis of this investigations an original KX-γ coincidences setup construction is proposed (MKGB-01 units based primary converter, background shielding system, positioning device, secondary electronics).

Conclusions

A concept of creation of the setup, implements the KX-γ coincidences method, based on commercially available spectrometer-radiometer MKGB-01 is developed. The main requirements to the future KX-γ coincidences setup are formulated.

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Comparison of two techniques for low-level tritium measurement – gas proportional and liquid scintillation counting

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Abstract

Measurement of low-level tritium activities in natural/non-polluted waters, e.g. in precipitation and groundwater, requires special techniques for water pretreatment and detection of low-level radioactivity. Two methods for low-level tritium measurement have been developed in Zagreb Radiocarbon and Tritium Laboratory: gas proportional counting (GPC) and liquid scintillation counting (LSC). Method of electrolytic enrichment of water samples with tritium followed by measurement by LSC *Quantulus 1220* has been developed in 2008. Here we present the comparison of the two measurement techniques.

Our Laboratory participated in the intercomparison study organized by the IAEA, TRIC2008. The data evaluation of six samples revealed that all results measured by LSC are accepted, as well as four GPC results, while two GPC results of samples with low tritium activity (<2 TU) are in the warning level.

Introduction

Atmospheric thermonuclear tests during 1950es and 1960es introduced large amount of anthropogenic tritium into the atmosphere. The peak of tritium activity with up to 10 000 TU has been reached in 1963 in a monthly rain in North America. The anthropogenic tritium has been used as a global tracer in hydrogeology and meteorology studies (Rożanski at al. 1991). Tritium activity in surface, groundwater and precipitation was measured by a gas proportional counting (GPC) technique, and later on by liquid scintillation counting (LSC). However, the present tritium activity almost approached natural levels (5 – 10 TU), and measurement of samples without tritium enrichment is not precise enough any more.

In Zagreb Radiocarbon and Tritium Laboratory two measurement techniques of low-level tritium activities in natural/non-polluted waters, e.g. in precipitation and groundwater, have been used: gas proportional counting (GPC) and liquid scintillation counting (LSC). GPC technique for measurement of non-enriched samples has been used since 1978. A method of tritium measurement by LSC *Quantulus 1220* using electrolytic enrichment of water samples has been developed in 2008.

Material and methods

CH_4 is used as a counting gas in a multi-wire GPC. It is obtained by reaction of water (50 ml) with aluminium carbide at 150°C (Horvatinčić, 1980). The counting energy window is set to energies between 1 keV and 10 keV to obtain the best figure of merit. Gas quality control has been performed by simultaneous monitoring of the count rate above the tritium channel, i.e., above 20 keV (Krajcar Bronić et al. 1986). The limit of detection is 2.5 TU.

System of electrolytic enrichment consists of 20 cells of 500 ml volume (anodes: stainless steel, cathodes: mild steel) and equipments for primary and secondary distillations. The first step of sample pretreatment is primary distillation. If conductivity of distilled samples is <50 µS/cm, samples are ready for the enrichment procedure. Otherwise, samples have to be distilled again. For each enrichment run 3 spike and 2 dead-water (DW) samples are used for system control. Enrichment procedure lasts for 8 days (1420 Ah). Final volume of water sample after electrolysis is 19±1 ml. The enrichment factor (E) and the enrichment parameter (P) for each electrolysis run have been calculated using initial and final mass of water in cells and individual count rates of spike water before and after enrichment. The average enrichment factor (E) of several electrolysis runs is 21.4 ± 0.9, and the P parameter reached a value of 0.95.

Mixture of 8 ml of water and 12 ml of scintillation cocktail *Ultima Gold LLT* in plastic vials is used for counting in LSC. The measurement run contains 15 samples, 3 enriched spikes, 2 enriched DW samples, 1 non-enriched spike, 2 non-enriched DWs and a standard sample (activity 11 552 TU on 7.7.2008). The limit of detection is 0.3 to 0.5 TU, depending on the measurement duration.

Results

The results of GPC and LSC measurements of IAEA TRIC2008 intercomparison samples are presented in Table 1. The results were evaluated with the z-score and Final score evaluations. The z-score values were calculated using measured laboratory values (A_{lab}) and their sigma values (σ_{lab}) and IAEA true values (A_{IAEA}):

$$z = \frac{A_{lab} - A_{IAEA}}{\sigma_{lab}}$$

The Final score evaluations were determined by the evaluation criteria for both trueness and precision of the measurements (Gröning et al, 2009). In the final evaluation, both scores are combined and a result must obtain ‘Acceptable’ score in both criteria to be assigned final score ‘Acceptable’.

According to the Final score evaluations all our LSC results are acceptable while two GPC results of samples T14 and T15 with low tritium activity are in warning level. The z-score evaluations showed that most of GPC results are lower than the IAEA true values (mean z-score is -0.44) and in the case of LSC results the deviations from the IAEA true values is smaller (mean z-score is -0.29). The comparison of both GPC and LSC results with the IAEA true values is presented in Fig.1.

Table 1. Comparison of GPC and LSC values measured in our Laboratory with IAEA TRIC2008 intercomparison results. The z-score and Final score evaluations are presented for both methods (A – acceptable, W – warning)

IAEA code	IAEA true value (TU)	GPC				LSC			
		Lab. code	Lab value (TU)	z-score	Final score	Lab. code	Lab value (TU)	z-score	Final score
T14	1.54 ± 0.05	T-3867	0.21 ± 0.81	-1.64	W	T-3906	1.25 ± 0.3	-0.97	A
T15	4.07 ± 0.05	T-3868	3.50 ± 0.96	-0.59	A	T-3907	4.11 ± 0.3	0.13	A
T16	7.74 ± 0.06	T-3869	6.60 ± 0.99	-1.15	A	T-3908	7.42 ± 0.3	-1.06	A
T17	14.46 ± 0.08	T-3870	14.23 ± 0.67	-0.34	A	T-3909	14.44 ± 0.4	-0.05	A
T18	0.67 ± 0.05	T-3871	0.03 ± 0.82	-0.78	W	T-3910	0.57 ± 0.3	-0.33	A
T19	568.7 ± 2.3	T-3872	579.24 ±5.70	1.85	A	T-3911	576.0 ± 13	0.56	A

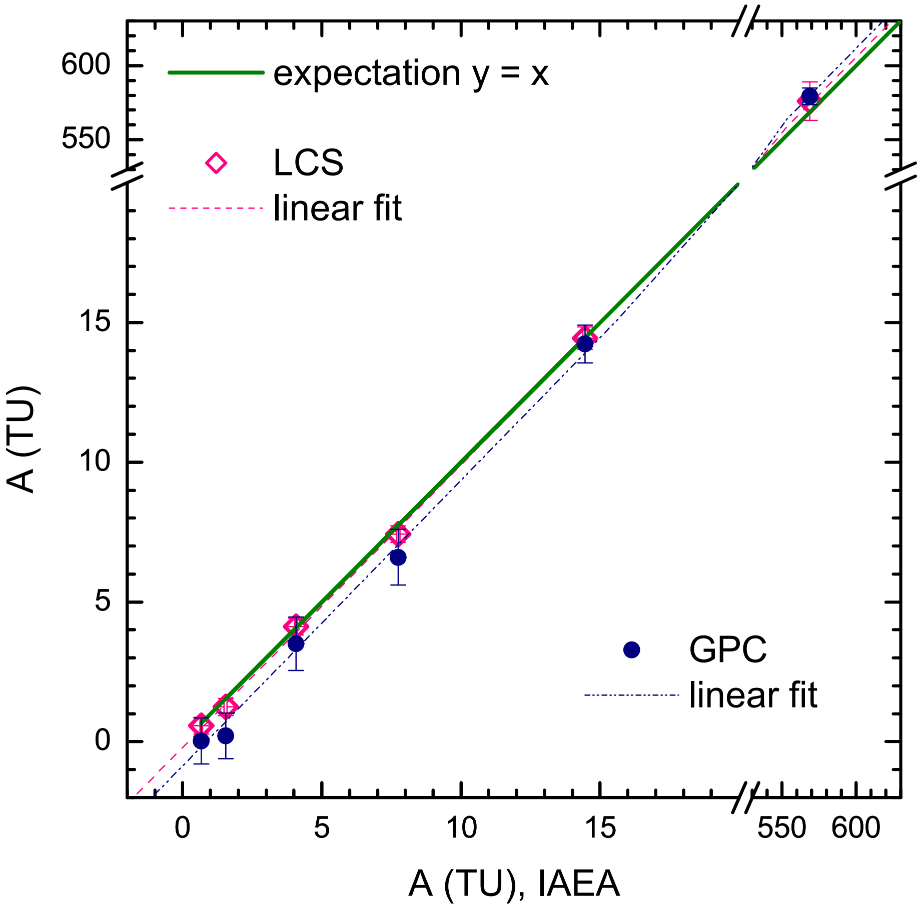

Fig. 1. The comparison of tritium activities of TRIC2008 intercomparison samples measured by GPC and LSC techniques with the IAEA true values.

For additional testing of both systems we measured tritium activity in several precipitation and groundwater samples by both GPC and LSC techniques. The results are presented in Fig. 2. Here we can see that all LSC results have smaller errors then GPC results. For the samples with low tritium activity (<10 TU, mostly groundwater) the GPC results are mainly lower then the LSC results while for the higher tritium activities (precipitation samples) the GPC results are lower than the LSC results. If we take into account also the IAEA intercomparison results obtained by both methods, we can conclude that for the low tritium activity the GPC system is not acceptable because of higher/worse detection limit (2.5 TU). LSC system with detection limit of 0.3 – 0.5 TU and with better precision (Table 1) is more suitable for most natural water samples including precipitation and groundwater samples.

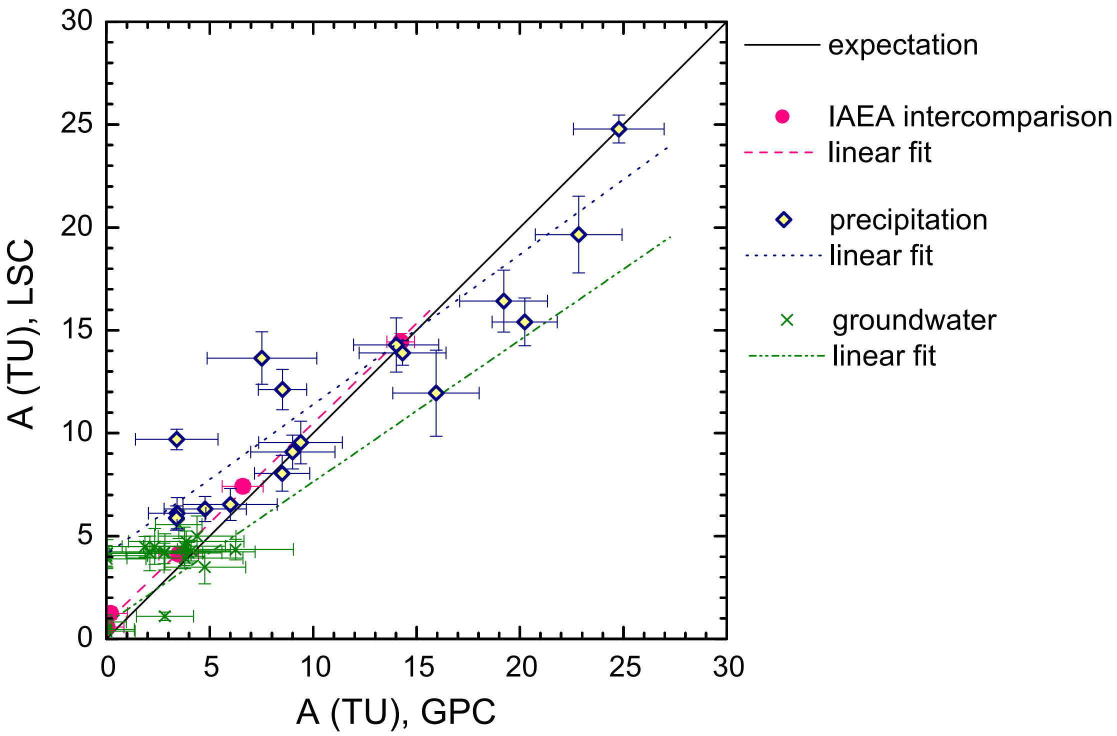

Fig. 2. Comparison of GPC and LSC results of groundwater and precipitation samples, as well as TRIC2008 intercomparison samples.

Conclusion

The data evaluation of six TRIC2008 samples (IAEA intercomparison) revealed that all results measured by LSC are accepted, and for GPC results four results are accepted and two results of low tritium activity (<2 TU) are in the warning level. Measurements of groundwater and precipitation samples show that GPC values are still valid for higher tritium activities (>10 TU), while lower activities can not be measured properly without enrichment and followed by LSC measurement.

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Multi-screen diffusion battery for radon progeny dispersion analysis

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Abstract

The screen-type diffusion battery method was enhanced for size distribution measurements of radon progeny air particles. The measurement technique implies multisection arrangement of several dozens of screen stages and a backup filter during sampling with subsequent activity measurement of each component. Variations on quantity of screens, flow rate and screen meshness allows to obtain required cut-off diameters of the stages and essential operating size range in general. Serial mounting of screen blocks with continually narrowing penetration curve is also used. Optimal parameters were adjusted for radon progeny size distribution researches.

Size distribution measurements of radon progeny in the range of 1-400 nm were performed in the laboratory radon box with high radon equilibrium-equivalent concentration (5-20 kBq/m^3). Generally accepted modes of unattached and aerosol fractions were confirmed. The smallest mode is represented by the primary free particles (geometric mean diameter 1.75 nm, geometrical standard deviation $\sigma_g = 2.0$). It appeared that the nucleation mode consists of three modes (8.5, 20 and 40 nm with σ_g 1.5, 1.2 and 1.7, respectively). Two modes of the accumulation mode were also observed (120 and 300 nm with σ_g 1.5 and 1.2, respectively).

Knowledge of dispersion behaviour of natural radioactive aerosols and gases is critical for understanding human exposure and for appropriate measurement technique development, especially in the case of unattached fraction. Results obtained in the work have particular value from this point.

Introduction

Natural radioactive gas radon and its progeny form the most significant radiological hazard on the general population round the world. Inhalation dose due to radon progeny is highly dependent on the size of inhaled particles. Newly generated short-lived radon decay products usually exist as positively charged ions that rapidly attach on atmospheric aerosol particles and form radioactive aerosol. So the size distribution of radon progeny is directly related to the natural particle sizes that primarily assist at the atmosphere.

In order to have an appropriate estimate of radon dose one should determine aerosol size distribution along with the radon concentration. Traditional measurement instruments for dispersion analysis include diffusion battery and cascade impactor. The first is used for measurements in nanometer region, while the second – for micrometer region. In this work a screen-type diffusion battery was designed for the size distribution measurements in the wide range from fractions of nm to several hundreds of nm.

Investigations were conducted in the laboratory radon box in Radon laboratory of Ural State Technical University (Ekaterinburg, Russian Federation). High radon concentration values in the box allowed performing aerosol particle size distribution measurements with high accuracy.

Material and methods

Most commonly used methods for size distribution measurements of atmospheric ultrafine particles are based on their diffusion properties. The smaller particles have higher diffusivities than large ones and diffuse more rapidly to surfaces. In this work we use screen-type diffusion battery method optimized for wide range of particle sizes.

The measurement system consists of a number of wire screens with different penetration characteristics and several backup filters arranged sequentially in a series configuration. Activity collected on each wire screen and filter is measured after sampling with semiconductor alpha counter.

Three different wire screen types that appeared to be optimal for experiments were chosen. In Table 1 the main screen characteristics are summarized. Analytic aerosol filters with standard characteristics were used as backup filters.

Table 1. Wire screen characteristics.

Type	Mesh	Wire diameter, mm	Screen opening, mm	Screen thickness, mm	Screen mass, g	Density, g/cm^3
1	35	0.25	0.5	0.51	2.13	7.8
2	150	0.65	0.1	0.14	0.69	8.5
3	350	0.03	0.04	0.07	0.31	7.8

Particle penetration through a wire screen was described by semi-empirical equations of filtration theory (Cheng et al. 1980). Penetration through the screen depends on its characteristics, geometry, air flow velocity and particle size. From theoretical calculations and experimental testing of diffusion batteries with different screen combinations two main configurations were adjusted for wide range aerosol dispersion analysis (see Table 2).

It should be noticed that for Type II battery ultrafine particles with sizes less than 10 nanometers get stuck already on the first wire-screens. It allows to neglect a control of tube losses, which is significant only for slow motion of ultrafine particles.

Table 2. Multi-screen diffusion battery configurations.

Type	Screen mesh	Quantity of screens	Operating flow rate, l/min	Operating AMTD range, nm
I	35 150 350	5 5 14	25	0.78-27
II	150 350	6 28	0.8	11-400

Sampling procedures were performed in the laboratory radon box with a volume of 4 m^3. Radon was generated inside the box by placing a standard source of Ra with activity of 518 kBq and emanation coefficient 0.26. Equilibrium conditions between radon and its progeny were established before the sampling. The radon equilibrium-equivalent concentration in the box atmosphere was in the range from 4500 to 18000 Bq/m^3. The volumetric flow rate during sampling was kept constant.

Diffusion battery was placed inside the box via locking system in order to neglect particle concentration losses. By use of flexible tubes the device was connected to the air pump with adjustable volumetric flow rate. Sampling procedure was implemented during 10-40 minutes. Individual activity measurements of collecting elements were performed after 40 minutes delay in order to neglect the influence of equilibrium shift between short-lived radon daughters. The primary measurement results were presented as a set of equilibrium-equivalent concentrations, calculated from individual activities of each diffusion battery collecting element according to the Kusnetz method (Kusnetz 1956).

It is well known that aerosol particle distribution in the air is well described by the polymodal lognormal distribution. In the case of unimodal lognormal distribution integral probability function plotted on the log-probability graph is used for definition of the activity median thermodynamic diameter (Hinds 1999). The probability scale of log-probability graph compresses the percent scale near the median (50% point) and expands the scale near the ends such that a cumulative plot of a lognormal distribution will yield a straight line. A log-probability plot can be constructed by plotting the logarithm of diameter versus the probability function, expressed in cumulative percentages. This probability function is defined as:

$$f(D_i) = f(D_{i-1}) - \frac{C_{\text{eq},i}}{\sum\limits_{k=1} C_{\text{eq},k}} \cdot 100\%,$$

where D_i is a 50% cut-off diameter of i wire-screen, $C_{\text{eq},i}$ is an equilibrium-equivalent concentration, corresponding to i wire-screen. The value of the probability function $f(D_1)$ for the first wire-screen is fixed to 100%.

It is supposed that every screen of diffusion battery catches particles related to several modes of aerosol particle distribution. Probability function curve plotted for the full set of collecting elements have some inflections. By the analytical method each diffusion battery type was divided to several sections with specific number of wire screens. Thus, integral probability function curve of every section was linear and corresponded to lognormal distribution of one mode. An example of the probability function curves for one diffusion battery is presented in figure 1.

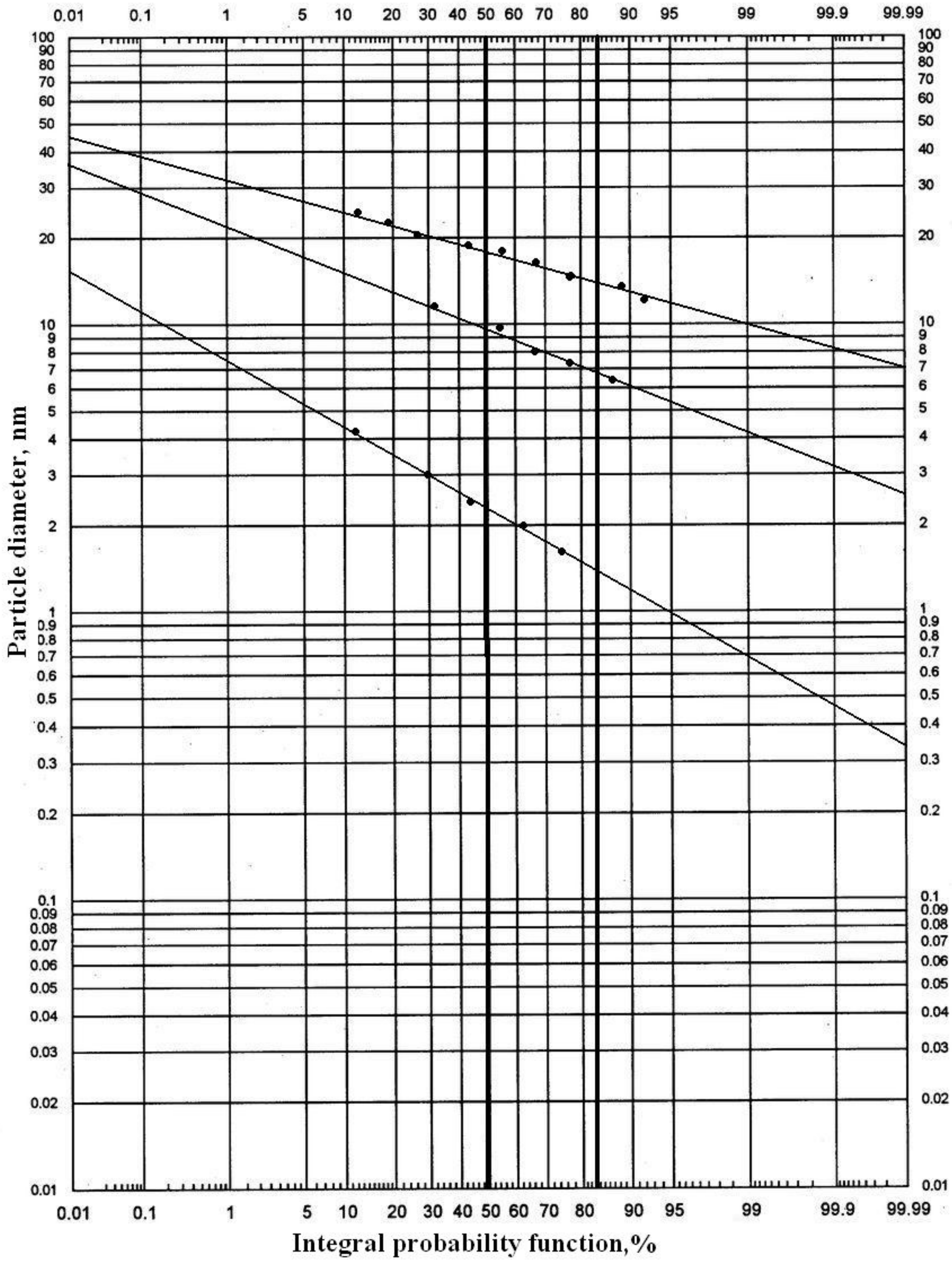

Fig. 1. Typical view of integral probability function for diffusion battery type I.

For a lognormal distribution the geometric standard deviation is defined as:

$$\sigma_{\mathrm{g}} = \frac{d_{50\%}}{d_{84\%}},$$

where $d_{50\%}$ and $d_{84\%}$ are diameters, associated with values of probability function 50% and 84%, respectively.

Results and discussion

In the previous research with primary version of diffusion battery and aerosol cascade impactor four modes of radon progeny particle activity size distribution were observed (Salomatova 2009). In the present research variations of screen quantity, flow rate and screen meshness essentially expanded the overall diffusion battery operating size range. It allows to exclude impactor application which led to large uncertainty. New diffusion battery method produces more detailed data on aerosol dispersion.

For each experiment performed in the radon box total equilibrium-equivalent concentration, activity median thermodynamic diameter and geometric standard deviation were calculated. Six modes of radon progeny particles activity size distribution were defined. Characteristics of the defined modes are presented in the Table 3. Obtained radon progeny particles activity size distribution is shown on the Figure 2.

Table 3. Distribution modes characteristics.

Mode number	AMTD, nm	Geometric standard deviation	Fraction
1	1.75	2.0	0.4
2	8.5	1.5	0.03
3	20	1.2	0.07
4	40	1.7	0.1
5	120	1.5	0.26
6	300	1.2	0.14

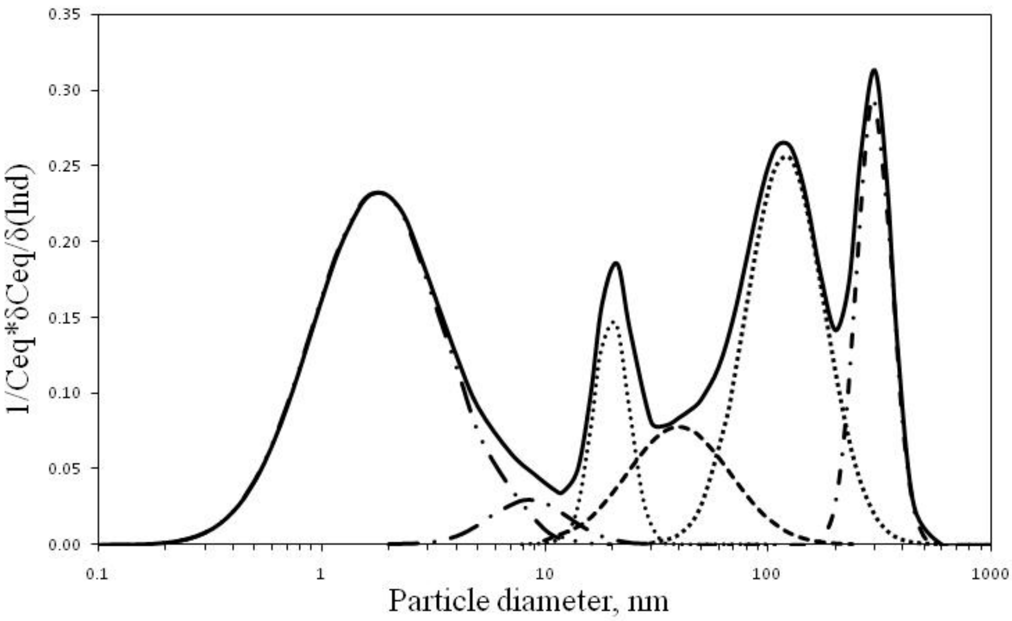

Fig. 2. Radon progeny activity size distribution.

A comparison with independent measurements of activity size distribution in earlier research (Porstendorfer 1996) is shown on the Figure 3.

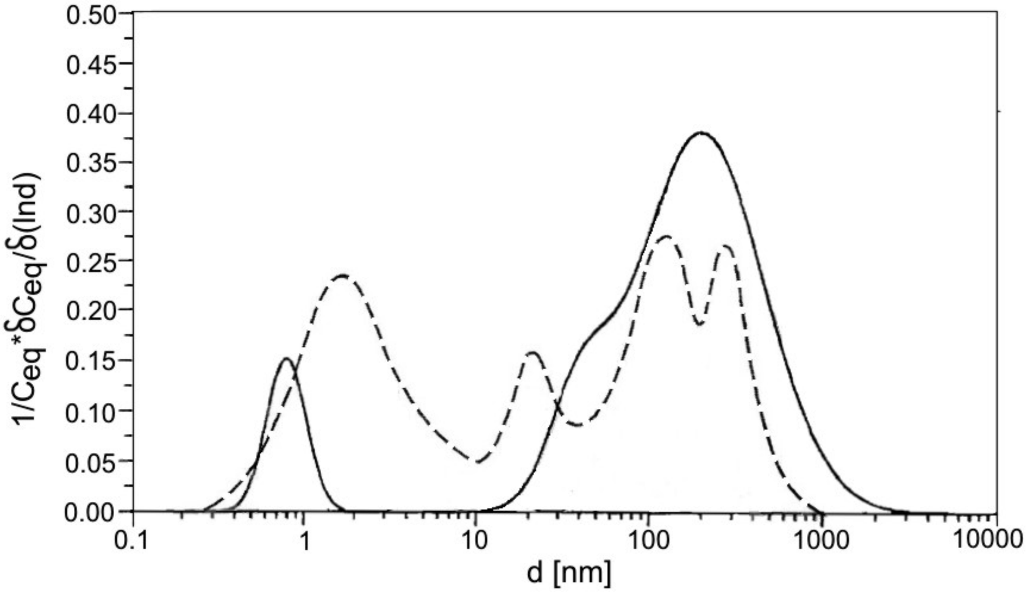

Fig. 3. Measured activity size distribution (dotted line) in comparison with typical distribution (solid line) for indoor air in dwellings (Porstendorfer 1996).

Air particles modes according to established classification could be identified as follows: 2 nm mode represents an unattached fraction of free radon progeny atoms; next five modes are formed by coagulation of unattached fraction atoms with atmospheric particles of different sizes. Thus, 8 nm corresponds to the group of natural particles, formed by the homogeneous nucleation; 20 nm and 40 nm modes are based on the particles, formed not only by homogeneous nucleation, but also condensation (heterogeneous nucleation). The largest modes 120 nm and 300 nm correspond to condensation particles.

On this basis it is possible to make an assumption that radon progeny particles activity size distribution can be an indicator of natural aerosol particles size distribution.

Conclusions

Several conclusions can be made as a result of the work performed.

The screen-type diffusion battery method was enhanced for radon progeny particle activity size distribution in the range from 1 to 400 nanometers with reduced geometric standard deviation in comparison with previous experiments with diffusion battery and cascade impactor.

Modes of the radon progeny particle activity size distribution corresponds to modes of the natural soluble particle size distribution. It is possible to measure concentration and size distribution of the fine soluble particles in the atmosphere through registration of radon progeny particles, coagulated with natural aerosols, by means of standard radiometric techniques.

The influence of atmospheric conditions parameters, such as humidity, temperature, aerosol particle concentration etc, is supposed to be investigated in the following work.

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Evaluating on-site monitoring cart conceptually developed for radiation workplace

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Abstract

In our previous study, an on-site radiation monitoring cart was developed by inspecting apparatuses necessary for radiation monitoring. These apparatuses were selected from those commercially available. It was found that all the equipment could be installed on an appropriately sized monitoring cart. Since the cart was too heavy, key methods for reducing the mass of the cart were suggested. The present study follows the previous study, and literature search was carried out to evaluate the performance of the cart from the viewpoint of radiation detection limits and radioisotope concentration. As a result, radiation detection limits were sufficiently lower than the legal limits stipulated by law in radiation workplaces radioisotope surface contamination, and radioisotope concentration in air, which are stipulated in the Japanese radiation law: The Law Concerning Prevention from Radiation Hazard due to Radioisotope etc.

1. Introduction

For developing an on-site radiation monitoring cart, radiation measurement apparatuses used for monitoring radiation at a radiation facility have been investigated recently. In our previous study (Kawano and Nishimura 2009), these apparatuses were properly selected from those commercially available, and they were investigated to see if they could be compactly installed on an appropriately-sized to be manhandled on-site monitoring cart. Furthermore, the feasibility of the cart and problems to be resolved were examined. It was found that all the equipment necessary for radiation monitoring could be installed on a monitoring cart of 164 cm (W) × 150 cm (H) × 80 cm (D), but the cart was too heavy. Thus, a more lightweight solution was required. Four key methods for reducing the mass of the cart were suggested.

Following the above-mentioned previous study, the cart was evaluated from the viewpoints of radiation detection limit and radioisotope concentration, assuming the selected apparatuses were installed onto the cart. The radiation detection limit was investigated through literature research, and compared with those stipulated in The Law Concerning Prevention from Radiation Hazard due to Radioisotope etc. of Japan (the Japanese radiation law is used as a abbreviated expression below) to evaluate the performance of the cart. It was concluded that the cart conceptually designed using only commercially available radiation measurement systems had a sufficient detection capability for radiation monitoring in radiation facilities including radiation workplaces and radiation-controlled areas.

2. Conceptually designed monitoring cart

2.1. Radiation monitoring in work places

In radiation workplaces that mean radiation workplaces and radiation-controlled areas, and controlled areas means boundary of controlled areas in the present study, radiation monitoring has to be performed at regular intervals. According to the Japanese radiation law, three measurement obligations, the radiation dose in radiation workplaces, radioisotope surface contamination, and radioisotope concentration in air, are stipulated for radiation monitoring at a radiation workplace. In Table 1, radiation and radionuclides to be monitored and legal limits stipulated in the Japanese radiation law as typical cases are summarized.

Table 1. Regulation limits for work place and controlled areas.

Measurement item	Radiation or Nuclides	Legal limits	
		Work place	Controlled area
Radiation dose	Gamma ray Neutron	1000 µSv/40 h (25 µSv/h)	1300 µSv/520 h (2.5 µSv/h)
Radioisotope surface contamination	Alpha Others	4.0 Bq/cm^2 40 Bq/cm^2	0.4 Bq/cm^2 4.0 Bq/cm^2
Radioisotope concentration in air	^{3}H ^{14}C ^{32}P ^{35}S ^{125}I	5.0E-1 Bq/cm^3 4.0E-2 Bq/ cm^3 7.0E-3 Bq/cm^3 2.0E-2 Bq/cm^3 1.0E-3 Bq/cm^3	5.0E-2 Bq/ cm^3 4.0E-3 Bq/ cm^3 7.0E-4 Bq/ cm^3 2.0E-3 Bq/ cm^3 1.0E-4 Bq/ cm^3

Radiation dose limits are 1000 µSv/a week for a radiation workplace and 1300 µSv/3 months for a controlled area, assuming one week and three months are 40 and 520 hr (5 days/week, 8 hour/day and 13 weeks/3 months), the radiation dose limits correspond to 25 and 2.5 µSv/h, respectively. These dose limits are normally unrelated to radiation quality, gamma rays, neutrons, and beta rays. In the case of radioisotope surface contamination, the limits are stipulated under two categories, alpha-ray-emitting nuclides and other radiation-emitting nuclides. For alpha-ray-emitting nuclides, 4.0 and 0.4 Bq/cm^2 are the surface contamination limits for a workplace and a controlled area, respectively. For the other radiation-emitting nuclides, the limits are ten times larger, 40 and 4.0 Bq/cm^2. In the case of radioisotope concentration in air, limits are stipulated for respective radioisotopes. In Table 1, the limits of radioisotope concentration in air are listed for the five typical nuclides of ^{3}H, ^{14}C, ^{32}P, ^{125}I, and ^{35}S. These nuclides are used often in radiation facilities in Japan. Radioisotope concentration in water is not stipulated by the Japanese radiation law because eating and drinking are not permitted in radiation facilities, including controlled areas.

2.2. Construction of on-site radiation monitoring cart

The conceptually designed on-site radiation monitoring cart discussed in the previous study is shown in Fig. 1, containing plan and side views. All the apparatuses used for monitoring radiation dose, surface contamination, and radioisotope concentration are mounted on a battery-powered rover rack, which one person can operate like an electric car. The dimensions of the rover rack are estimated to be 164 cm (W) × 150 cm (H) × 80 cm (D) . The rover rack has a running speed of 3.5 km/h and a running distance of 3.5 km when the battery is fully charged.

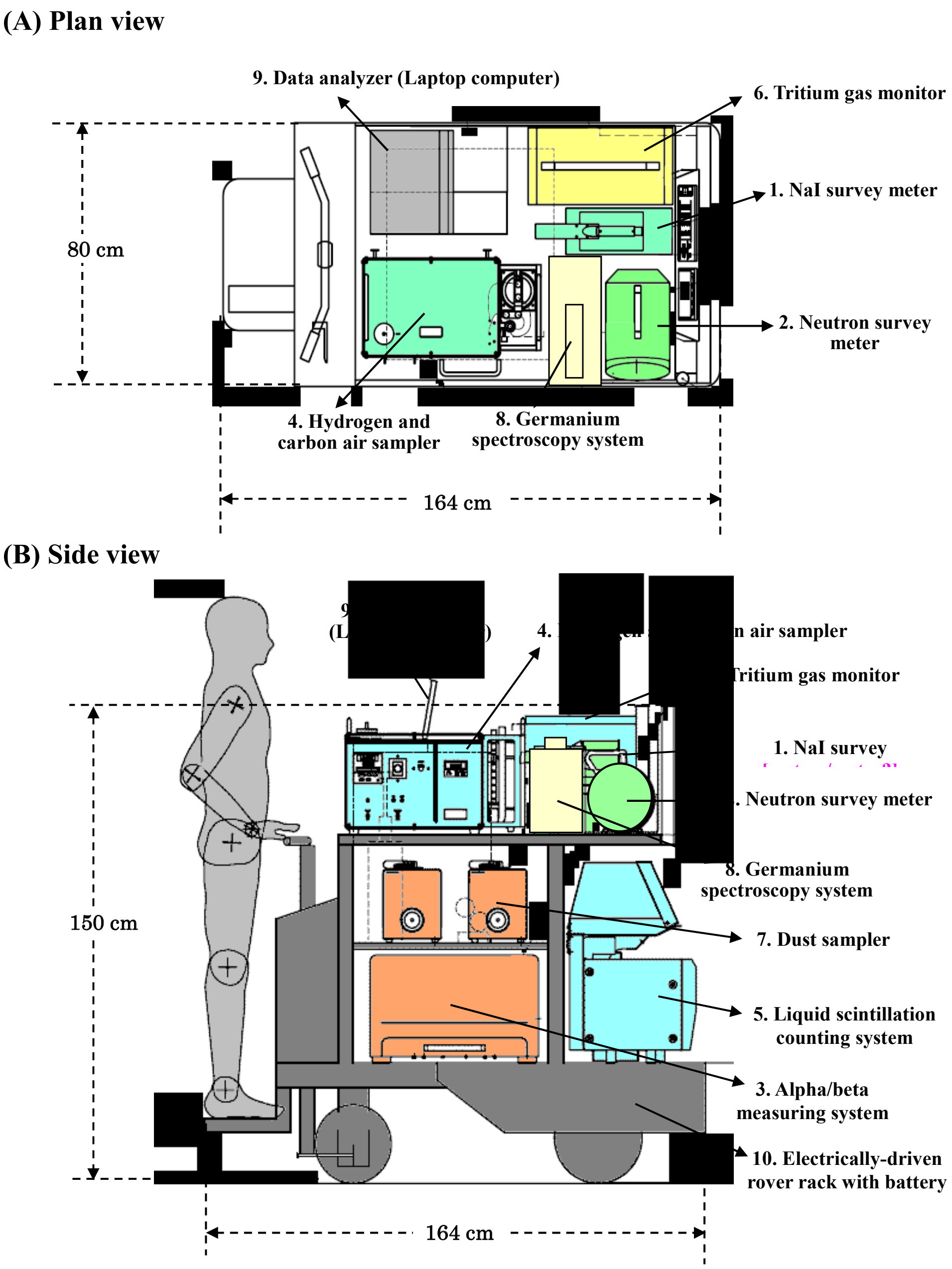

Fig. 1. Illustration of radiation monitoring cart.

(1) Monitoring radiation dose for gamma (X) rays and neutrons

A NaI(Tl) scintillation survey meter and a neutron survey meter are used to measure the radiation dose in radiation workplaces. The neutron survey meter is installed in a ^{3}He proportional counter for detecting neutron radiation. These survey meters are portable and can be replaced with other types of conventional survey meters, such as a Geiger Mueller (GM) survey meter and a tritium survey meter, as the need arises. For this radiation monitoring, alpha rays are ignored because their range in air is short and their penetrating power is weak.

(2) Monitoring radioisotope surface contamination

There are two methods for monitoring radioisotope surface contamination. One is a survey method and the other is a smear method. In the survey method, the above-mentioned survey meters, NaI(Tl) scintillation, neutron, GM, and tritium, can be used to directly detect radioisotope surface contamination. In the smear method, a smear paper is used to wipe up radioactive contamination from floors and the surfaces of devices, tools, etc. The radioactivity of contamination collected on the smear paper can be measured using the liquid scintillation counting system, alpha/beta radiation measurement system, or germanium spectrometer. The liquid scintillation counting system is used to measure tritium and carbon-14 contamination with high sensitivity. The alpha/beta radiation measurement system has a ZnS(Ag) scintillation detector and a plastic scintillation detector for measuring alpha and beta radiation, respectively. The germanium spectrometer is used to analyze gamma-ray-emitting nuclides contained in radioactive contamination. The smear method is highly sensitive but requires time a somewhat longer than the survey methods since it is an indirect measurement.

(3) Radioisotope concentration in air

There are also two methods for monitoring radioisotope contamination in air. One method is continuously measuring radioisotope concentration in air using a vent-type detector. An example of this type of monitor is a tritium gas monitor, which is installed in a vented ionization chamber and can provide live data. In the other method, a tritium and carbon air sampler and a dust sampler are used to collect radioisotope contained in air. With the hydrogen and carbon air sampler, tritium and carbon-14 contained in air are collected in the chemical forms of water and carbon dioxide by a cooled collection tube with dry ice-ethanol and a monoethanolamine tube, respectively. The radioactivity of the tritium collected in water and that of carbon-14 collected with monoethanolamine are measured using the liquid scintillation counting system mentioned above. The dust sampler is used to collect other radioisotopes that are bound to dust suspended in air on a filter paper or an activated charcoal filter. Radioactivity trapped on the filter can be measured using the liquid scintillation counting system, the alpha/beta radiation measurement system, or the germanium spectrometer, which is similar to the smear method.

A conventional laptop computer is installed on the on-site radiation monitoring cart as an on-site data analyzer. The computer can perform data acquisition, data storage, and data display. Consequently, the on-site radiation monitoring cart has the same functions as a conventional radiation measurement room. Furthermore, the cart is movable, allowing it to be brought into the radiation laboratory to carry out on-site radiation monitoring and removed from the radiation laboratory on an as-needed basis. The cart is essentially a portable radiation measurement room.

3. Evaluation of detection limit

To certify that the cart is effective in detecting these radiation limits, a literature search was carried out. Table 2 lists the detection limits and their ratios to the legal limits set by the Japanese radiation law for the respective apparatuses conceptually installed for monitoring radiation dose in radiation workplaces, radioisotope surface contamination, and radioisotope concentration in air. For radioisotope concentration in air, detection limits are shown for five typical nuclides, ^{3}H and ^{14}C, ^{32}P, ^{125}I, and ^{35}S.

Table 2. Comparison of detection and regulation limits.

Measurement item	Radiation or Nuclides	Apparatuses installed	Detection limit	Detection limit/Legal limit	
				Work place	Controlled area
Radiation dose	Gamma ray	NaI(Tl) scintillation survey meters	0.1 μSv/h	4.003-03	4.00E-02
	Neutron	Neutron survey meters	0.01 μSv/h	4.00E-04	4.00E-03
Radioisotope surface contamination	Alpha Others	Alpha/beta measuring system	0.04 Bq/cm^2 0.03 Bq/cm^2	0.01 7.50E-04	0.1 7.50E-03
Radioisotope concentration in air	^{3}H	Tritium and carbon air sampler and	2.9E-5 Bq/ cm^3	5.80E-05	5.80E-04
	^{14}C	Liquid scintillation counting system	8.0E-6 Bq/cm^3	2.00E-04	2.00E-03
	^{3}H	Tritium gas monitor	1.1E-2 Bq/ cm^3	0.022	0.22
	^{32}P ^{35}S ^{125}I	Dust sampler and alpha/beta measuring system	3.8E-6 Bq/ cm^3 2.0E-6 Bq/ cm^3 2.5E-6Bq/ cm^3	5.43E-04 1.00E-04 2.50E-03	5.43E-03 1.00E-03 2.50E-02

3.1. Radiation dose

For radiation monitoring in radiation workplaces, a NaI(Tl) scintillation survey meter and a neutron survey meter are used. Detection limits of the meters are 0.1 µSv/h and 0.01 µSv/h, respectively, as shown in Table 2. These detection limits can be compared to the legal limits shown in Table 1, the ratios are derived as 4.00E-03(1/250) and 4.00E-02(1/25) for gamma ray radiation and 4.00E-04(1/2500) and 4.00E-03(1/250) for neutron radiation in radiation workplace s and controlled areas. These results suggest that the radiation detection limits of these survey meters are sufficiently lower than those stipulated by law.

3.2. Surface contamination

Assuming the alpha/beta measuring system is regularly used to measure radioactive surface contamination with the smear method, the detection limits are 0.04 Bq/cm^2 for alpha-particle-emitting nuclides and 0.03 Bq/cm^2 for other nuclides. Comparing these detection limits to the legal limits for alpha-particle-emitting nuclides, the ratios are derived as 0.01(1/100) and 0.1(1/10) for radiation workplaces and controlled areas. For other nuclides, the ratios are 7.50E-04(1/1000) and 7.50E-03(1/100) for radiation workplaces and controlled areas. The radiation limits of the smear method are sufficiently lower than the legal limits.

3.3. Radioisotope concentration in air

To measure the ^{3}H and ^{14}C concentrations in air, a tritium and a carbon air sampler and a liquid scintillation counting system are used. The detection limits of both nuclides are 2.9E-5 Bq/cm^3 and 8.0E-6 Bq/cm^3, respectively. The detection limits are smaller than the regulation limits, and the ratios are less than 5.80E-05(1/1000) and 5.80E-04(1/100) for radiation workplace s and controlled areas. If only tritium is measured, a tritium gas monitor can be used. The detection limit of the monitor is 1.1E-2 Bq/ cm^3. The ratios of the detection limits to the regulation limits are less than 0.022(1/45) and 0.22(1/4.5) for radiation workplace s and controlled areas.

For monitoring other nuclide concentrations in air, the dust sampler is used to collect dust in a filter. The filtrated dust containing radioisotopes can be measured using the alpha/beta measuring system. The detection limits of these measurements are 3.8E-6, 2.0E-6, and 2.5E-6 Bq/ cm^3 for the more commonly used nuclides ^{32}P, ^{125}I, and ^{35}S. The ratios of the detection limits to the regulation limits are less than 2.5E-03(1/400) and 2.5E-02 (1/40) for radiation workplace s and controlled areas.

The above results show that the on-site radiation monitoring cart conceptually designed using only commercially available radiation measurement systems performs well from the view point of radiation detection limits when used for monitoring radiation workplace s, including controlled areas.

4. Summary

An on-site radiation monitoring cart was developed by conceptually installing necessary apparatuses including a NaI(Tl) scintillation survey meter, neutron survey meter, alpha/beta measuring system, tritium and carbon air sampler, liquid scintillation counting system, dust sampler, and a tritium gas monitor. These apparatuses were all selected from those commercially available. The size of the cart was 635 mm × 800 mm × 1500 mm.

The performance of the cart was evaluated by inspecting its radiation detection limit. The ratios of the detection limits to the legal limits set by the Japanese radiation law were examined. As a result, the detection limit of the cart is, at worst, 1/100 and 1/10 smaller than the legal limits for radiation monitoring of radioisotope surface contamination, radioisotope concentration in air, and in radiation workplace s and controlled areas. The cart has sufficiently detected radiation in a radiation facility.

Reference

Kawano T. and Nishimura K. Conceptual Design of On-Site Monitoring Cart Available for Fusion Facility. 9th International Symposium on Fusion Nuclear Technology, October 11-16, 2009, P2-145, Dalian Chine, (2009)

A new AMS system for actinides isotopic ratio measurements at CIRCE (Caserta, Italy)

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Abstract

Anthropogenic long-lived alfa-emitting radionuclides have been (and still are) released in the environment by nuclear testing, nuclear accidents, and operations of fuel reprocessing and plant decommissioning. Among these ^{239,240}Pu and ^{236}U are the most significant. The Accelerator Mass Spectrometry (AMS) technique surmounts the limitations of other techniques used.

In Italy no Nuclear Power Plant (NPP) is in operation, but the four dismissed NPPs are now being decommissioned; moreover, new generation of NPPs are reconsidered for future installations. Both the operations demand accurate investigations about the possible contamination by radioactive releases of nuclear sites and neighboring territory and of structural materials of the reactors. The monitoring activity requests ultrasensitive methodologies for the detection and quantification of ultralow activity radionuclides. Among radionuclides the alpha-emitting isotopes of actinide elements are the most critical also because of their toxicity. The radiation counting methods, due to the long half lives, do not provide the necessary sensitivity and, in some cases, resolution (e.g. ^{239}Pu-^{240}Pu).

The present contribution reports on the state of the art of a project, carried out at the AMS facility in the Center for Isotopic Research on Cultural and Environmental Heritage (CIRCE, Caserta, Italy), aiming to establish an ultrasensitive system for the measurement of concentrations and isotopic abundances of U and Pu, and to exploit this technique for the characterization, from the point of view of the contamination by U and Pu, of both environmental samples from the Garigliano NPP site and samples representative of the structural materials of the building and of the reactor of the same plant. At the same time the system provides a tool able to fulfill the analytical needs of IAEA for the campaigns for nuclear safeguards against illegal nuclear activities and the military use of weaponry with depleted uranium.

Introduction

Anthropogenic uranium and plutonium, and fission and activation products were widespread around the world due to human nuclear activities. Different sources of radionuclide contamination affect the environment on various scales of magnitude: plutonium from the atmospheric weapons test fallout is detectable globally in soils and sediments (Roca et al, 1989). In particular, the operation and decommissioning activities of a NPP could lead to airborne and liquid releases of radionuclides that would be mixed with contributions from larger and world scale sources. Useful tools to solve among different contributions are the isotopic ratios: $^{236}U/^{238}U$, $^{238}Pu/^{239+240}Pu$, $^{240}Pu/^{239}Pu$, $^{241}Pu/^{239}Pu$, $^{242}Pu/^{239}Pu$, given that the relative concentrations of plutonium and uranium isotopes depend on the nature of the source material and on its subsequent irradiation history. ^{239}Pu is produced from ^{238}U via neutron capture and subsequent beta decay of the resulting short-lived ^{239}U ($t_{1/2}$ = 23.47 min). In weapon-grade plutonium, ^{239}Pu is the most abundant isotope ($^{240}Pu/^{239}Pu < 0.07$). In weapon test fallout, $^{240}Pu/^{239}Pu$ varies, depending on the test parameters, in the range (0.10-0.35), the integrated ratio for the Northern hemisphere being about 0.18 (Koide et al., 1985); significantly different values (0.035-0.05) are found in Mururoa and Fangataufa atoll sediment, because of the peculiar nature of French testing (Chiappini et al., 1989).

In nuclear reactors, due to the different composition of fuels, uranium enrichment and burn-up degree, characteristic relative abundances of plutonium isotopes will be obtained: $^{240}Pu/^{239}Pu$ increases with irradiation time, which, in turn affects $^{238}Pu/^{239+240}Pu$. This ratio is useful to resolve between different sources in case they show similar $^{240}Pu/^{239}Pu$: e.g., irradiated nuclear fuel in a PWR with 7-20% ^{235}U and burn-up 1.4-3.9 GW·d reaches $^{240}Pu/^{239}Pu$ isotopic ratios of 0.13, a value, that could be ascribed also to global fall out; on the other side, these two sources show quite different $^{238}Pu/^{239+240}Pu$ activity ratio (0.025-0.04 for the global fallout and 0.45 for that nuclear fuel), (Buessler et al., 1995). Another valuable tool to identify a nuclear reactor origin of a radionuclide contamination is $^{236}U/^{238}U$ isotopic ratio. The dominant ^{236}U mode of formation is the capture of a thermal neutron by ^{235}U, a secondary contribution being the alpha decay of 240Pu; its concentration in nature has been heavily increased as a consequence of irradiation of enriched uranium in nuclear reactors. Several orders of magnitude of difference between the $^{236}U/^{238}U$ isotopic ratios in naturally-occurring uranium (10^{-9} to 10^{-14}) and in spent nuclear fuel (10^{-2} to 10^{-4}) imply that also a small contamination from irradiated nuclear fuel in natural sample is able to increase the $^{236}U/^{238}U$ measured in the whole sample. The measurement of U and Pu isotopic ratios requires the sensitivity of mass spectrometric techniques because ^{236}U and plutonium are present in environmental samples at ultra trace levels (^{236}U concentration is quoted to be in the order of pg/kg or fg/kg and ^{239}Pu around 100 pg/kg (Perelygin et al.,1997) and are long-lived radionuclides. To measure $^{236}U/^{238}U$ at natural level the unique able technique is the AMS, because of its capability to suppress background of isobaric molecules as it doesn't suffer of isobaric interferences from $^{235}UH^+$. The hydrides formation in ICPMS devices defines the detection limits of this technique for $^{236}U/^{238}U$ to 10^{-7}. Also, because of the interference from $^{238}UH^+$, the measurement of $^{239}Pu/^{240}Pu$ by ICPMS requires a higher decontamination of the plutonium fraction from uranium than AMS needs. However, no chemical procedure is efficient enough to separate uranium and plutonium fractions to allow the mass spectrometric measure of ^{238}Pu, being the concentration of ^{238}U seven orders of magnitude higher than that of ^{238}Pu. Alpha-spectroscopy remains the only suitable technique for the measurement of ^{238}Pu concentration.

Material and methods

Sample preparation

The extraction of uranium and plutonium was performed following the procedures described in Kritil et al. (1979), applied to soil and sediment samples and adapted to our needs (Quinto et al., 2009). The samples were dried in a muffle furnace at 110°C, homogenized and combusted at 450 °C overnight. ^{236}Pu (30 mBq = 3.8×10^6 atoms) and ^{232}U (86 mBq = 2.7×10^8 atoms) were added as reference isotopes and also as radiotracers to determine the radiochemical yield of the extraction. The samples (2-12 g) were leached in 7.2 M HNO_3 and 0.5 g $NaNO_2$ for at least 3 h. The residue was centrifuged off and the supernatant was stored; fresh acid was added to complete the washing of the matrix. The combined leachates were brought to dryness and the residue was dissolved in 7.2 M HNO3. After adding of 0.2 g $NaNO_2$ the solution was heated until gas formation ceased and, after cooling, was loaded on an anion exchange column (6 ml Dowex 1x2, 100-200 µm particle size, previously conditioned with 7.2 M HNO_3). Plutonium present as Pu (IV) is retained on the resin. Uranium, mostly present as U (VI) is washed from the column with 45 ml of 7.2 M HNO_3. The original solution (now free of Pu) and the washing solutions were combined and kept for uranium determination. The column was washed with 30 ml of 32% HCl to remove thorium. Finally, plutonium was eluted with 50 ml of a mixture of 0.36 M HCl and 0.014 M HF into a teflon beaker containing already 5 ml 65% HNO_3. This solution was brought to dryness, then evaporated three times in 5 ml 65% HNO_3 with some drops of 30% H_2O_2 and another three times in 5 ml 37% HCl. The residue was dissolved in 6 ml fuming HCl, and subdivided in two halves for AMS measurement and alpha-spectrometry, respectively. The uranium fraction contained large amounts of iron. Therefore, the solution was evaporated to dryness and the residue dissolved in 20 ml 8 M HCl, from which iron was eliminated by diisipropylether liquid-liquid extraction. The iron free uranium fraction was evaporated to dryness, dissolved and evaporated twice in 50 ml 37% HCl, and finally dissolved in 80 ml 8M HCl. This solution was brought to a second anion exchange column (Dowex 1x2, 100-200 µm particle size, in 8 M HCl). Calcium and thorium were washed out with 50 ml 8 M HCl, then uranium was eluted with 90 ml 0.1 M HCl. The uranium solution was treated the same way as the plutonium solution; two subsamples for AMS and alpha-spectrometry were prepared. The chemical form of the AMS sample is $U+Fe_2O_3$ and $Pu+Fe_2O_3$, obtained by iron hydroxide co-precipitation and subsequent conversion to iron oxide by combustion at 700 °C. Uranium and plutonium fractions in iron oxide matrix were mixed with a similar volume of silver powder and pressed in aluminium sample holders suitable for the ion source of accelerator system. For alpha-spectrometry, plutonium and uranium fractions where co-precipitated with NdF_3 (Hindman, 1983) and deposited on membrane filters suitable for alpha-spectroscopy (Quinto et al., 2009).

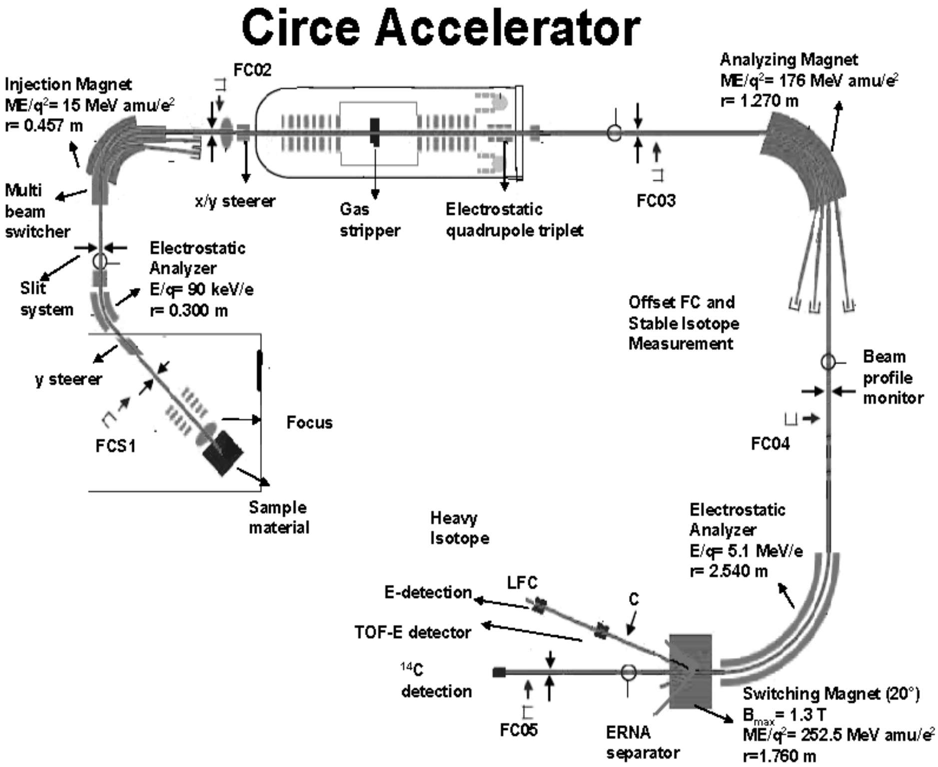

Fig. 1. The scheme of the +3 MV tandem accelerator of Pelletron type installed at the CIRCE.

The AMS measurement

The +3 MV CIRCE tandem accelerator of Pelletron type (Terrasi et al. 2008), similar to others installed in the past years in Europe and USA, is shown in Fig.1. Since the installation of CIRCE in 2005, we have extended the measuring capabilities of the original system with the installation of a switching magnet and a heavy isotope beamline (De Cesare et al., 2009; De Cesare et al., 2010).

The caesium sputter ion source is a 40-sample MC-SNICS. A typical output from 1 mm diameter samples pressed in Al cathodes for $^{12}C^-$ ions is 30 μA at 6 kV cathode voltage and a total injection energy of 67 keV and 0.3 μA for $^{238}U^{16}O^-$ molecules with a total injection energy of 50 keV. These ions are energy selected by a ±45° spherical electrostatic analyzer (r = 30 cm, plate gap = 5 cm) that is operated up to ±15 kV. Maximum electric field strength is 6 kV/cm, resulting in an energy/charge state ratio of 90 keV. The 90° double focusing Low Energy (LE) injection magnet (r = 0.457 m, vacuum gap= 48 mm, ME/q^2 = 15 MeV· amu) allows high resolution mass analysis for all stable isotopes in the periodic table. The insulated stainless steel chamber can be biased up to 15 kV for beam sequencing (e.g. between ^{12}C, ^{13}C, ^{14}C or between $^{238}U^{16}O$, $^{236}U^{16}O^-$).

The accelerator is contained inside a vessel filled with sulphur hexafluoride (SF6) at a pressure of about 86 psig. The terminal voltage (TV) is achieved and stabilized by GVM feedback on the charging system high voltage supply. The TV can be kept constant at 3.000 MV with a ripple of 1 kV. In the stripper, Argon (Ar) is recirculated by two turbo-pumps; the working pressure is about ~13 mTorr for $^{14}C^{3+}$ at 2.550 MV and ~7.3 mTorr for $^{238}U^{5+}$ at 2.875 MV, with an offset of ~6 mTorr

The double focusing 90° High Energy (HE) bending magnet has r= 1.27 m, ME/q^2 = 176 MeV.amu and M/ΔM = 725, with slit opening of ±1 mm both at object and image points, so that, e.g. $^{236}U^{5+}$ at 3 MV can be analyzed. The two 45° electrostatic spherical analyzers (r= 2.54 m and gap = 3 cm) are operated up to ±60 kV; energy resolution is E/ΔE = 700 for typical beam size.

Finally the selected ions are counted in a surface barrier detector.

In order to assess laboratory background for ^{236}U measurements, diluted samples of Joachimsthal uranium (0, 4, 40, 400 and ~104 μg) were prepared for each run. We used the U^{5+} current as an estimate of the uranium content. The effect of a mass-dependent background correction is almost negligible. The ^{236}U count rate, calibrated by the dilution series, was used to estimate the ^{236}U concentration. Results of such measurement are reported in Quinto et al. 2009.

In order to assess the accuracy of the $^{240}Pu/^{239}Pu$ isotopic ratio measurement, a Reference Material, IAEA 368 (sediment sample collected in the Pacific Ocean at Mururoa Atoll, a former French nuclear weapons testing site), was measured in three runs. Each measure of $^{240}Pu/^{239}Pu$ isotopic ratio was in agreement with the certified interval of values (0.035 to 0.05).

Results and discussion

Measurements have been performed with a position sensitive silicon detector with an active area of 58×58 mm², positioned 1.9 m downstream the 4 mm diameter collimator, positioned in turn 80 cm from the switching magnet. To measure the $^{236}U/^{238}U$ ratio a tuning of the transport elements up to the final detector is performed in order to maximize the ion optical transmission. The ^{236}U events are counted in the final detector and ^{238}U current is measured in FC04.

The tuning is made by setting the parameters of the beam line of the ^{238}U current to the Last Faraday Cup (LFC) positioned in front of the detector, using as target material uranium oxide (U_3O_8) obtained from Uranyl Nitrate ($UO_2(NO_3)_2{}^{\cdot 6}$ H_2O) baked at 800°. This material originated from the uranium mines "K. k. Uranfabric Joachimisthal", and is the VERA (Vienna Environmental Research Accelerator) in-house U standard. In our case, the sample preparation provides uranium in form of UxOy+Fe_2O_3 from which xUyO- is sputtered. To select different injected masses the injection magnet bouncing system is used. The stripping yield for $^{238}U^{5+}$achieved by the Ar gas stripper of CIRCE is around 3.1%. At 3 MV terminal voltage, ME/q^2 is ~ 170 MeV·amu/e^2 for $^{238}U^{5+}$ and ME/q^2 is ~ 221 MeV·amu/e^2 for $^{238}U^{4+}$; since the double focusing HE magnet reaches a maximal ME/q^2= 176 MeV·amu, the 5+ represents the lowest charge state which can be bent by the HE magnet. For heavy ion measurements, the object and image slits of the injection magnet are closed to ±1 mm, the slits of the HE magnet are closed to ±2 mm and a collimator of 4 mm diameter is positioned just at the waist of the ESA. The current transmission between FCO4 and LFC is ~ 80 %. Once the setup for the pilot beam $^{238}U^{5+}$ is found, the voltage at the chamber of the injection magnet, the terminal voltage and the voltage of the analyzer ESA are scaled to the other wanted mass ($^{236}U^{5+}$).

The preliminary measurements for the isotopic ratio $^{236}U/^{238}U$ were performed by cycling between ^{238}U, which is measured by means of the current in the Faraday cup just after the HE magnet FC04, and ^{236}U, that is measured in the final 16-strips surface barrier detector. We measured 2 samples: one with a nominal ratio $^{236}U/^{238}U \sim 10^{-8}$, obtained by adding a spike of ^{236}U to the KkU VERA in-house U standard, and the KkU itself with $^{236}U/^{238}U = (6.98 \pm 0.32) \times 10^{-11}$ (Steier et al. 2008). Also, samples with a "chemistry blank" have been prepared by the same treatment as the others. The contribution of the latter is of the order of 1 % of the KkU sample signal. Moreover, comparison with the spatial distribution obtained when the collimator was removed shows that interference from ions of higher magnetic rigidity with respect to $^{236}U^{5+}$ are largely suppressed by the 4 mm collimator; one can estimate that the residual interference background below the $^{236}U^{5+}$ peak, obtained as relative difference between the sum of the 4 central strips of the KkU (4mm C. out) and KkU (4mm C. in), is less than 80 % of the total, which is in turn the interference measured without the collimator. That indicates that "interference + chemistry" background in our $^{236}U/^{238}U$ measurements is $< 5\times10^{-11}$. This value has to be compared with previous result (3×10^{-9}, De Cesare et al 2009) obtained at 0°, before the installation of the switching magnet. De Cesare et al., (2009) have obtained a lower limit of $(9\pm1)\times10^{-9}$ for the spike sample isotopic ratio, in agreement with the expected value. These results imply that with a background $< 5\times10^{-11}$, we are able to measure samples with an isotopic ratio value $\geq 10^{-10}$, without the use of an Energy Time Of Fligth system (TOF-E). In order to obtain the best $^{236}U/^{238}U$ isotopic ratio sensitivity of 10^{-12} (Steier et al., 2004), in samples including about 1 mg of U, a TOF-E system will be included. An upgraded of the CIRCE facility has been planned including the TOF-E system, with a flight path of 1.5 m, aiming to reach the necessary $^{236}U/^{238}U$ isotopic ratio sensitivity.

Further investigations are needed to identify the amount of ^{238}U that is under the peak of $^{236}U^{5+}$ and the amount of ^{235}U, that is also present. A Time of Flight–Energy (TOF-E) detection system (De Cesare et al. 2007; De Cesare et al. 2009) will be used, to discriminate between ^{236}U, ^{238}U and ^{235}U, with the start signal for the TOF measurement given by a MCP (MicroChannel Plate), in electrostatic mirror configuration, positioned about 70 cm from the 4 mm collimator. The energy information is given by the 16-strips silicon detector, which also provides the timing for the stop signal.

As an example of application to investigate the possible contamination of Garigliano Nuclear Power Plant (Caserta, Italy) activities, some soil samples were collected for the measurement of actinides in the neighbour of the plant. The values of activity and isotope ratios in environmental samples measured with AMS allow to identify the origin of the activity, due to the sensitivity of the technique used. The isotope ratio of $^{236}U/^{238}U$ and the specific activity of ^{236}U and ^{239}Pu in soils is shown in Figure 2, where are also reported the results of ^{137}Cs specific activity obtained from gamma spectrometry measurements.

It is evident a correlation between the specific activities of the three radionuclides, indicating a possible common origin. The isotope ratio values ranging between 4.5 10^{-10} and 1.3 10^{-8}. These values are compatible with the value measured in a control sample

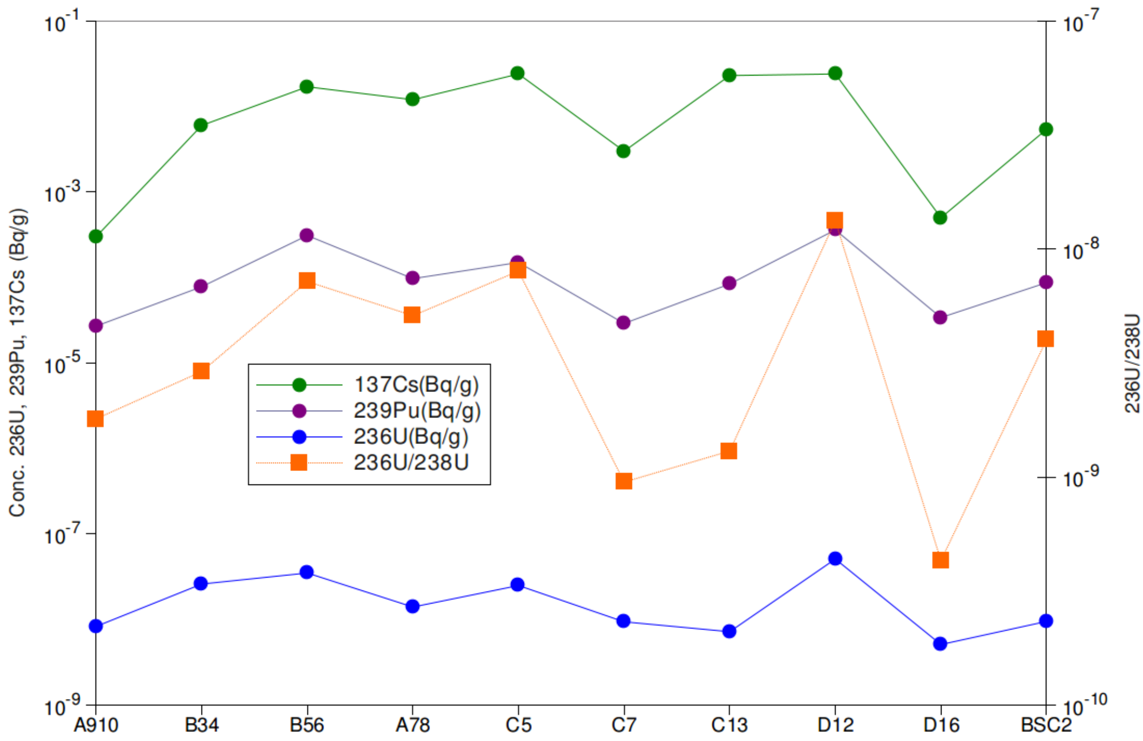

Figure 2. Results of ^{236}U, ^{239}Pu and ^{137}Cs specific activities and of ^{236}U/^{238}U ratio obtained on soil samples collected around the Garigliano Nuclear Power Plant (Caserta, Italy).

collected in the Sele plain (BSC2 sample), about 100 km far from the Garigliano NPP. Moreover, these values are consistent with values of global fallout [Sakaguchi et al., 2009] and are significantly lower than those measured in contaminated sites [Boulyga & Heumann, 2006]. Measurements of samples representative of the structural materials of the building and of the reactor of the same plant are also measured, but data not still available.

Conclusions

In this work an overview of the upgraded CIRCE facility for actinides measurements has been described in order to reach the necessary ^{236}U/^{238}U isotopic ratio sensitivity. The results of CIRCE laboratory are not far from those of ANU e VERA laboratories, but improvements are planning. Applications of method are carried out on soil sample to investigate possible contamination due to a NPP activities.

This system provides a tool able to fulfill the analytical needs of IAEA for the campaigns for nuclear safeguards against illegal nuclear activities and the military use of weaponry with depleted uranium.

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Neural network method for activity measuring in environmental samples

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Neural network method for activity measuring in environmental samples. Theses.

The spectrometric method is one of the most popular methods for activity measuring in environmental samples. It provides application of semi-conductor and scintillation spectrometers. Despite high measurement accuracy, high price and some difficulties in operation make semi-conductor detectors difficult to apply. Scintillation spectrometers, which are cheaper, are applied to determine activity of natural radionuclides. But continuous spectrum image on the monitor permits the spectrometers to measure not so many radionuclides and, thus, it restricts application of the scintillation spectrometers.

One of the ways to solve the problem is to use sophisticated mathematical analysis of decomposition of scintillation spectra. Examination of different spectrum analyses and devices, produced by means of these methods, reveals that sometimes the above solution is quite improper and has weak solution stability. Besides, some problems appear while applying these methods in field devices, which are to carry out scale analysis without a computer. As a result, simplified methods are applied, which reduce accuracy of activity determination.

The current article describes the method of decomposition of scintillation spectra into spectrum components by means of artificial neural networks. The aim of the above method is to determine activity of radionuclides in the source under measurement according to its radiation spectrum. The spectrum is acquired by means of scintillation spectrometer. If applying a laboratory spectrometer, which, as a rule, is equipped with a computer, the spectrum is transferred from the ADC to the PC, which operates with the above method. If applying field spectrometers, spectrum is to be processed by either a smart sensor or operating control unit. Activity calculation algorithm is equal in both cases.

Introduction

Nowadays industry often has bad unwholesome influence on the environment and a human being. That is why environmental friendliness is one of the quality factors of production. It causes constant development of special equipment and methods for environmental monitoring. Radioactivity stands out from other quality factors to be controlled. High radiation contamination risks, on the one hand, and strict safety regulations, on the other hand, draw special attention to metrological parameters of modern equipment of radiation monitoring.

The spectrometric method is one of the most popular methods for activity measuring in environmental samples. It provides application of semi-conductor and scintillation spectrometers. Despite high measurement accuracy, high price and some difficulties in operation make semi-conductor detectors difficult to apply. Scintillation spectrometers, which are cheaper, are applied to determine activity of natural radionuclides. But continuous spectrum image on the monitor permits the spectrometers to measure not so many radionuclides and, thus, it restricts application of the scintillation spectrometers.

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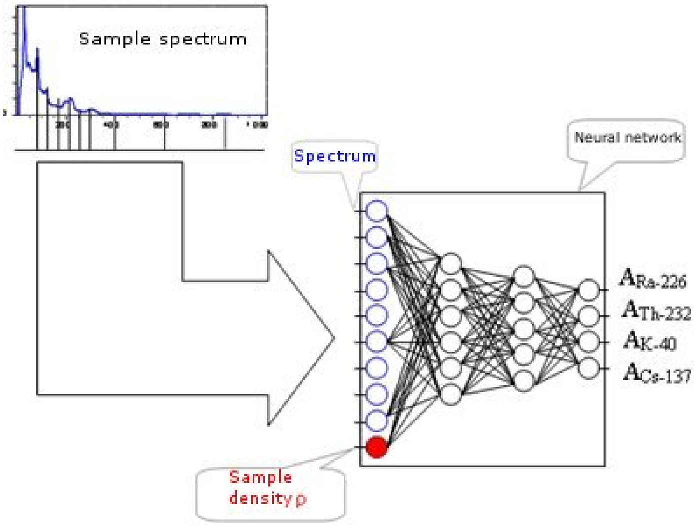
Activity calculation of radionuclides

Since the spectrometric method under discussion is relative, calibration of the equipment is required. According to the neural network theory such calibration is called learning.

The step-by-step algorithm of activity calculation follows. Learning (calibration) is described in part 3 of the current article.

First of all, we describe input data. We have a spectrum of a working source N collected by means of a spectrometer. The source (sample) is measured in the preset geometry with the certain density $\rho_{и}$ and life time. We have also a background spectrum $N_{ф}$ measured under the same conditions. Background is usually measured by means of a spectrometer either with an empty shielding chamber or with an empty sample dish in the chamber.



Pic.1. Diagram of activity measurement in samples by means the neural network method.

To reduce the spectrum to the energy scale, we should know energy calibration. Modern spectrometers provide the relation between a spectrum channel No. and energy, which are linearly dependent as follows:

$$E_i = E_0 + k \cdot i \quad (1)$$

where E_i – energy corresponding to the *i*-channel, keV; E_0, k – coefficients of energy calibration fixed during the device setting.

Data gained in the course of learning (calibration) is weights of neural network W, scales of each network input M_{in} and output M_{out} and energy windows where summing up R_E is carried out.

During calibration network parameters should be set up, including the number of layers and neurons in each layer and a type of activation function with parameters. The number of network inputs depends on the number of selected windows, plus one input for density. The number of network outputs corresponds to the number of radionuclides under detection.

First of all, we prepare input data for network calculation.

In the spectrum channels we define corresponding windows for each specified energy window. According to Eq.1:

$$R_{Lj} = \frac{R_{ELj} - E_0}{k} \quad , \text{ j=1..n} \quad (2)$$

where R_{Lj} – channel No. in the spectrum of the left edge of the *j*-window; R_{ELj} – energy of the left edge of the *j*-window, keV; n – number of windows.

The similar equation is used for the right edge:

$$R_{Rj} = \frac{R_{ERj} - E_0}{k} \quad , \text{ j=1..n} \quad (3)$$

where R_{Rj} – channel No. in the spectrum of the right edge of a *j*-window; R_{ERj} – energy of the right edge of a *j*-window, keV.

We define total counting rate in the specified windows for the source working spectrum and background spectrum:

$$S_j = \frac{1}{t} \cdot \sum_{i=R_{Lj}}^{R_{Rj}} N_i \quad , \text{ j=1..n} \tag{4}$$

$$S_{\phi j} = \frac{1}{t_{\phi}} \cdot \sum_{i=R_{Lj}}^{R_{Rj}} N_{\phi i} \quad , \text{ j=1..n} \tag{5}$$

where S_j and $S_{фj}$ – counting rates in the j-window of working and background spectra correspondingly, imp/sec; t и $t_ф$ – real measurement time of working and background spectra correspondingly, sec. [1, 4].

Now we define input values of the neural network as follows:

$$X_j = \frac{S_j - S_{\phi j}}{M_{in\, j}} \quad , \text{ j=1..n} \tag{6}$$

where X_j – value of the j-output of the neural network; $M_{in\,j}$ – scale value for the j-output. Eq. 6 reveals that the value X_j can vary from 0 to 1, which is caused by principles and architecture of neural network.

The number of network inputs m=n+1, with another special input for density, is defined as follows:

$$X_m = \frac{\rho_u}{M_{in\, m}} \tag{7}$$

where $\rho_и$ – density of the sample under measurement, g/cm^3; $M_{in\ m}$ - scale for density input, g/cm^3.

Input data is gathered now and should be put in calculation of the neural network. Network calculation means definition of states and outputs of neurons within inner layers and the last output layer [2]. These are outputs of the last layer, which are desired unscaled activities of radionuclides. One should remember that the number of outputs is not fixed and defined according to research tasks. It means that if activity of a certain radionuclide is to be determined in the sample with a radionuclide mixture, the network can have one output corresponding to a specified radionuclide. In this case network learning is carried out by means of spectra of unmeasured radionuclides.

Network calculation is made in a layer-by-layer manner. Calculation of neuron states for the first layer is not necessary, since their outputs, or network inputs, are known. Thus, neuron states for the second layer are calculated as follows:

$$C_j^{II} = \sum_{i=1}^{P_1} X_i \cdot W_{1,i,j} \quad , \text{ j=1..}P_2 \tag{8}$$

where P_1 - number of neurons in the first layer, similar to the number of network outputs; P_2 – number of neurons in the second layer; $W_{1,i,j}$ – weight between the i-neuron of the first layer and j-neuron of the second layer, the index *l* depicts a layer, which has a link output (the first layer for Eq. 8).

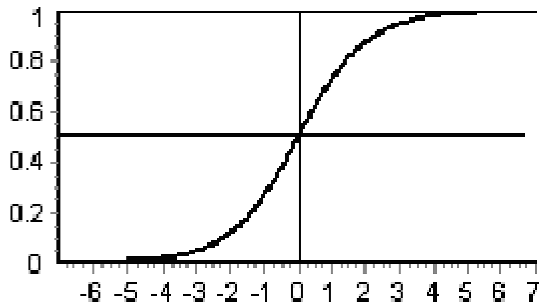
Neuron output is regarded as the function of its state:

$$y_j^{II} = f(C_j^{II}) \text{ , j=1..P}_2 \tag{9}$$

where y_j^{II} - output of the j-neuron in the second layer; f – activation function, defined as:

$$f(c) = \frac{1}{1+\exp(-\alpha \cdot c)} \tag{10}$$

where c – argument of the function or, in this case, neuron state; α – coefficient, which determines the slope of the function graph (Pic.2). The coefficient α is determined in the course of calibration and can vary from 0.5 to 3.



Pic. 2. Sigmoid activation function [3].

Despite vast variety of activation functions, the research revealed that the above type of the activation function provides particularly successful and accurate calculation.

Neuron states for the third layer are calculated as follows:

$$C_j^{III} = \sum_{i=1}^{P_2} y_i^{II} \cdot W_{2,i,j} \text{ , j=1..P}_3 \tag{11}$$

As shown by Eq. 11, the network inputs are replaced by the outputs of the second layer. Thus, we have got the following equation, similar to Eq. 9:

$$y_j^{III} = f(C_j^{III}) \text{ , j=1..P}_3 \tag{12}$$

where y_j^{III} - output of the j-neuron in the third layer.

The network for the rest of layers, including the output layer [2, 3], is calculated on the same principle as illustrated above. In this case, the output layer is described as follows:

$$C_j^{Z} = \sum_{i=1}^{P_{Z-1}} y_i^{Z-1} \cdot W_{Z-1,i,j} \tag{13}$$

$$y_j^{Z} = f(C_j^{Z}) \text{ , j=1..P}_Z \tag{14}$$

where y_j^Z - output of the j-neuron in the last layer; Z – number of layers in the network.

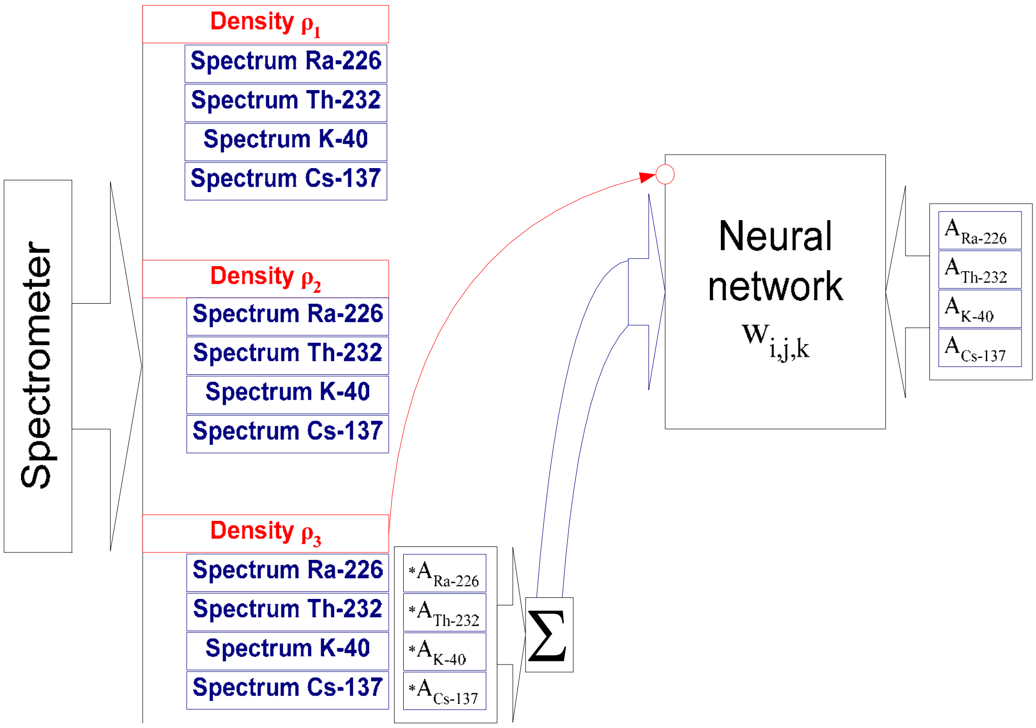
To turn directly to calculation of radionuclides activities, output scales determined during the calibration should be used.

$$A_j = (y_j^Z - k_{\kappa p}) \cdot (\frac{M_{out\ j}}{k_{yM}}) \quad , \ j=1..P_Z \tag{15}$$

where A_j – activity of the j-neuron in order, Bq; $M_{out\ j}$ - scale for j-output of the network; $k_{\kappa p}$ –boundary coefficient, which enables not to use the non-linear range of a sigmoid function and is defined as 0.1; k_{yM} – coefficient introduced to correct a scale disturbed by the boundary coefficient. More detailed description of the above coefficients follows.

Calibration (learning)

To train the neural network, calibration spectra are required, which should be spectra of separate radionuclides measured in calibration samples with different densities (Pic.3).



Pic. 3. Example of choosing calibration spectra for learning.

These available calibration spectra should be included into the learning sample, where each component is a spectrum with different radionuclide content and density. To get such spectra, the additivity concept of simple spectra of radionuclides is to be applied. Thus, the equation for a sample spectrum with activities A_j, where j=1..n (n – number of nuclides in a source) will be the following:

$$Q_i = Q_{\phi_i} + \sum_{j=1}^{n} q_{ij} \cdot A_j \;, \quad \text{i=1..v} \tag{16}$$

where Q_i – counting rate in the i-channel of the mixed spectrum of the sample, imp/sec; $Q_{\phi i}$ – counting rate in the i-channel of the background spectrum, imp/sec; q_{ij} – counting rate in the i-channel for the simple spectrum of the corresponding j-radionuclide, imp; v- number of channels in a spectrum. Eq. 16 can be applied provided that energy calibration of both simple spectra and the background spectrum is similar. If the above condition is not met, the value of energy calibration is to be made equal for all the spectra by means of Eq. 1. Besides, one should take into account that simple spectra *q* are collected for samples with one and the same density.

Simple spectra *q* are calculated as follows:

$$q_{ij} = \frac{\dfrac{N_i}{t} - \dfrac{N_{\phi_i}}{t_\phi}}{A_j} \;, \quad \text{i=1..v, j=1..n} \tag{17}$$

where N_i– counting in the i-channel in the real j-radionuclide spectrum, imp; $N_{\phi i}$– counting in the i-channel in the background spectrum, imp; A_j – sample activity of the j-radionuclide, Bq; t and t_ϕ – life time for measuring source spectra and background spectra correspondingly, sec [4].

As shown by Eq. 15, while changing the value A_j we can acquire the required spectrum Q for a non-existent sample with the density similar to that one, for which simple spectra q were collected. Thus, Eq. 15 can be put down as follows:

$$Q_i^k = Q_{\phi_i} + \sum_{j=1}^{n} q_{ij}^k \cdot A_j^k \;, \quad \text{i=1..v , k=1..r} \tag{18}$$

where r – number of densities; k – density No. in order.

Thus, Eq. 17 is the main one for making both learning and testing samples. The variable k, i.e. density, and variables A_j are subject to variation, or the so-called randomization. And this is the acquired spectrum Q, which is the learning element. There can be as many similar elements as possible. It means that the more information the neural network learns, the more accurate and precise results are achieved.

Now we describe the learning algorithm in details. We have simulated spectra of radionuclide mixtures Q and the activity vector A_B corresponding to each of the spectra. Thus, according to the algorithm described in part 2 of the current article, initialization of the matrix weights W should be carried out by means random variables from 1 to 0.3 [2]. Then, we put different spectra from the learning sample Q in random order to the network input, calculate forward the network by means of Eqs. 1 – 15 and, as a result, get the simulated vector of radionuclide activity A_M. In this case we have already got activity vector A_B from the learning sample for the spectrum Q.

The matrix weights are corrected in the opposite way to the network distribution. Thus, an auxiliary j-variable can be illustrated as follows:

$$g_j^Z = y_j^Z \cdot (1 - y_j^Z) \cdot (y_j^Z - y_{Bj}) \;, \quad \text{j=1..}P_Z \tag{19}$$

where y_j^Z - j-neuron output of the last layer; Z – number of network layers; $y_{B\,j}$ - target value of the j-output, calculated according to the following:

$$y_{B\,j} = \frac{A_{B\,j}}{M_{out\,j}} \quad , \; j=1..P_Z \tag{20}$$

where $M_{out\,j}$ - scale for j-output of the network.

Now we calculate the coefficient of weight correction:

$$\Delta W^T_{Z-1,i,j} = -\eta \cdot g_j^Z \cdot y_i^{Z-1} + \mu \cdot \Delta W^{T-1}_{Z-1,i,j}\,, \quad i=1..P_{Z-1}\,, j=1..P_Z \tag{21}$$

where T- number of learning iteration, if T=1 (i.e. at the beginning of learning) variables $\Delta W^{T-1}_{Z-1,i,j}$ are equal to 0; η – parameter, which determines learning rate, can vary from 0 to 1; μ – learning coefficient (Part 2) can vary from 0 to 1. The minus sign "-" in front of the first component of Eq.21 is caused by the fact that the weight changes in the opposite way to that one indicated by the derivative of error surface.

Weight correction is carried out according to the following equation:

$$W^T_{Z-1,i,j} = W^{T-1}_{Z-1,i,j} + \Delta W^T_{Z-1,i,j} \quad , \quad i=1..P_{Z-1}\; j=1..P_Z \tag{22}$$

The above weight correction (Eqs. 19 – 22) is carried out in the opposite way to the network distribution for each pair of layers. This approach is called "backward error distribution".

For the next pair of inner (hidden) layers (***not for the last and next to last ones***) weights change in the following way:

$$g_j^{Z-1} = y_j^{Z-1} \cdot (1 - y_j^{Z-1}) \cdot \delta_j \quad , \; j=1..P_{Z-1} \tag{23}$$

where

$$\delta_j = \sum_{k=1}^{P_Z} g_k^Z \cdot W^T_{Z-1,j,k} \tag{24}$$

Eq. 23 differs from Eq. 19 on the error of a hidden layer δ_j , which does not correspond directly to target output values. That is why weight correction of the hidden layer is proportional to its "contribution" to the error of the next layer.

Then, weight correction is carried out in a similar way to Eqs. 21 and 22:

$$\Delta W^T_{Z-2,i,j} = -\eta \cdot g_j^{Z-1} \cdot y_i^{Z-2} + \mu \cdot \Delta W^{T-1}_{Z-2,i,j}\,, \; i=1..P_{Z-2}\,, j=1..P_{Z-1} \tag{25}$$

$$W^T_{Z-2,i,j} = W^{T-1}_{Z-2,i,j} + \Delta W^T_{Z-2,i,j} \quad , \quad i=1..P_{Z-2}\; j=1..P_{Z-1} \tag{26}$$

As far as next layers are concerned, weights are transformed according to Eqs. 23 – 26 if correction is carried out from the end of the network to its beginning [2, 3].

After correction completion, another spectrum Q with a density parameter is put into the input the network to start learning. The network is calculated forward and the results are compared with target values of the vector y_{Bj} specified by the spectrum Q. Then, learning is carried out in the opposite way. There can be as many learning iterates as possible (about $5 \cdot 10^6$). It stands to reason that only a computer can calculate the above. The higher capacity computers have, the greater interest to artificial neural networks is aroused in different spheres of physics.

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Whole body counting with large plastic scintillators as a tool in emergency preparedness – determination of total efficiency and energy resolution

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Abstract
The measured total efficiencies for large plastic scintillators (NE 102 A), 91.5 × 76.0 × 24.5 cm^3, has been compared with Monte Carlo calculated total efficiencies. For ^{54}Mn the results show a good agreement except close to detector edges. To investigate light losses the Monte Carlo calculated deposited energy spectrum was converted into emitted light spectrum. The amount of emitted light was then changed depending on where the energy was deposited in the scintillation material. The emitted light per unit energy was calculated with Birks formula, with kB set to 9.65 mg cm^{-2} MeV^{-1}. Even for small light losses the emitted light spectrum resolution was affected.

Introduction and Theory
Increased demands on emergency preparedness have lead to a renewed interest in whole body counting. It is hard to know beforehand the energy and source distributions that will be of interest in a situation and a fast and accurate calibration method is therefore sought after. One possible solution is to do a Monte Carlo calibration of the whole body counter, WBC, system.

The University of Gothenburg has a low-activity laboratory equipped with two WBCs, one with four large plastic scintillators (NE 102A) and one with two large NaI(Tl)-detectors. The ambition is to do a Monte Carlo calibration for both systems, and presented in this work are the preliminary results for the plastic scintillator system.

To verify the WBC model, calculated total efficiencies for several nuclides were compared with measured. This was a relatively straight forward work; a much more demanding task is to obtain a calculated energy deposition spectrum whose energy resolution is comparable to a measured spectrum.

In scintillator spectrometry systems a spectrum is broadened due to multiple processes between the initial energy deposition and the final signal from the photomultiplier tube, PMT. The processes can be divided into five major parts (Valentine 1998): 1) production of scintillation photons; 2) the scintillation photons are collected at the PMT photocathode; 3) production of photoelectrons; 4) the photoelectrons are collected at the first dynode in the PMT; and 5) multiplication in the

PMT. Breitenberger (1955) has given a statistical description of the spectrum broadening in scintillator spectrometers, equation 1, using a theoretical model where the steps 2-4 are combined into a single transfer function. The relative variance V of the number of electrons Q to arrive at the PMT anode is given by

$$V_Q = V_T + \frac{1+V_M}{\overline{X}\overline{T}} \tag{1}$$

Each index indicates the quantity that contributes to the variance: X is the number of photons, M the multiplication including after-effects in the PMT and T the probability for one photocathode electron to reach the first dynode in the PMT per scintillation photon. The number of scintillation photons/MeV is known for NE 102A and V_M should, according to Breitenberger, be considered an instrumental parameter that is adjusted for each individual case. To determine T for the plastic scintillators is still a work in progress and investigated in this work are the effects on spectrum broadening due to nonlinear light yield and loss of scintillation photons, for example by light absorption in the scintillation material.

To convert the calculated deposited energy into emitted light the following relation first proposed by Birks (1964) was used

$$dL/dr = S(dE/dr)(1+kB(dE/dr))^{-1} \Leftrightarrow dL/dE = S(1+kB(dE/dr))^{-1} \tag{2}$$

where r is the range in the scintillator expressed in g cm^{-2}, dL/dr the specific fluorescence, S the absolute scintillation efficiency, dE/dr the specific energy loss, kB a constant expressed in mg cm^{-2} MeV^{-1}, and dL/dE the emitted fluorescent light per unit deposited energy. Birk's equation accounts for nonlinear light yield.

The code used was MCNPX 2.60, Monte Carlo N-Particle eXtended. The calculated energy depositions spectra are scored with MCNPX's f8 tally; it is possible to use a Gaussian energy broadening, GEB, function together with the f8 tally. However, since the broadening process consists of several steps, each with its own probability and variance, which are not necessarily Gaussian, the ambition is to use Breitenberger's equation instead of the GEB function.

Material and methods

The WBC consists of four large plastic scintillators made of NE 102A (NE 102A is equivalent to BC-400) each measuring 91.5 × 76.0 × 24.5 cm^3. Each detector is equipped with two PMTs (EMI 9545A) mounted on a perspex light guide and the stainless steel detector housing has a 0.397 mm copper window. The signal from the photomultiplier tubes are amplified (Canberra Amplifier 816A) and the output fed to an Ortec pulse-height analyser (Ortec ASPEC-927). Fig. 1 shows the detector configuration.

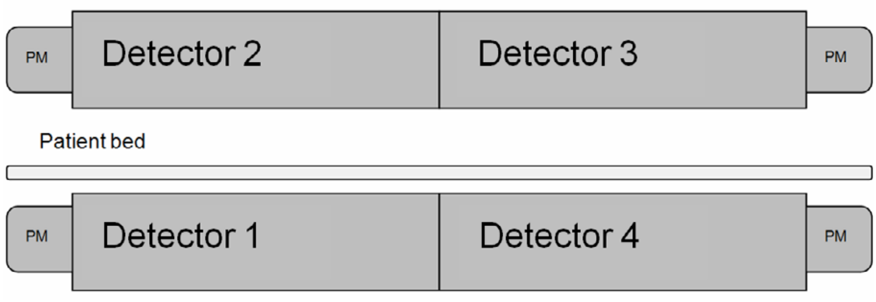

Fig. 1. Two detectors above and two below the patient bed gives, roughly, a 3π geometry. PM stands for photomultiplier tube. Medial direction: along the patient bed, -100 to 100 cm; lateral direction: across patient bed, -40 to 40 cm.

The Monte Carlo model of the WBC included the scintillation material, detector housing with Cu window, light guides, patient bed and a 3 mm lead wall coating of the surrounding steel chamber. Adding more wall or floor material gave a very little difference in total efficiency and it was therefore excluded in the model.

Four button sources were used: ^{137}Cs, ^{54}Mn, ^{65}Zn and ^{60}Co; only the result for ^{54}Mn is presented here.

Total efficiency

The total efficiency was measured and simulated for the source placed along the axis of symmetry in medial and lateral direction with 10 cm intervals. Medial direction: along the patient bed; lateral direction: across patient bed. The total number of counts was compared to the calculated number of counts scored with the f8 tally.

Spectrum broadening

The calculated deposited energy in MeV was converted into MeVee (MeV electron equivalent, a deposited energy of 1 MeV generates 1 MeVee in light output at linear light yield). This was done using the MCNPX's pulse height light tally, PHL. When using PHL, *dL/dE* for all energies has to be stated in the MCNPX input file. *dL/dE* was calculated with equation 2, where *S* was set to 3.0 (Mukhopadhyaya 2004); *dE/dr* was calculated with ESTAR (NIST 2005). For high electron energies the response is linear for NE 102A (Saint-Gobain Crystals 2005) and *kB* was changed until the ^{54}Mn photo absorption energy peak at 0.835 MeV lead to a photo absorption emitted light peak at 0.835 MeVee.

To investigate how light losses, depending on energy deposition site, affected the emitted light spectrum, each detector material cell in the MCNPX model was divided in medial direction into 9 smaller cells, all with the same material properties but with a different *dL/dE* depending on the distance from the light guide. The emitted light per particle track was scored with the f8 tally, and when used together with PHL, f8 sums the emitted light per particle track in MeVee from all 9 cells. Light absorption or light leaving the detector material was represented by multiplying *dL/dE* with a factor A_i (i=1:9; $A_i \leq 1$). A was calculated with the equation $A_i=-Cx_i+1$ where C is a constant and x_i the distance from the light guide to the middle of each of the 9 detector cells. However, this it is not necessarily considered an accurate description of the detector material. The value for C was changed in steps of 0.0005 from 0.0005 to 0.0025, for the box furthest away from the light guide A_9 varied from 0.956 to 0.778.

Results

Total efficiency

In fig. 2 and 3 the medial and lateral total efficiencies for ^{54}Mn are showed for detector 1. The results show a good agreement between measured and calculated efficiencies except close to the detector edges.

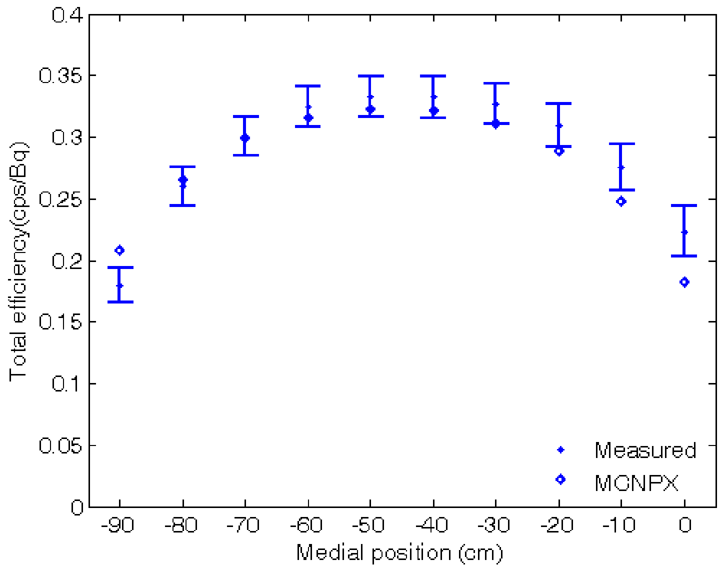

Fig. 2. The total efficiencies for ^{54}Mn placed at different positions along the medial direction. Lateral position was 0 cm in all measurements.

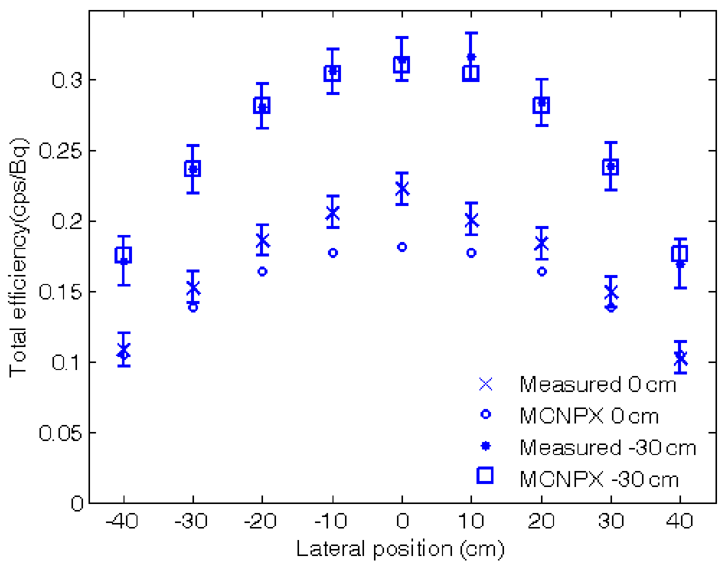

Fig. 3. The total efficiencies for ^{54}Mn placed at different positions along the lateral direction. Medial position was 0 and -30 cm, respectively.

Spectrum broadening

For kB=9.65 mg cm^{-2} MeV^{-1} the photo absorption energy peak at 0.835 MeV lead to a photo absorption emitted light peak at 0.835 MeVee, for lower energies small differences were noticed between linear and nonlinear light yield.

For detector 1, the deposited energy spectrum, and thereby also the emitted light spectra for all A_i=1, were nearly identical when the source was placed at -30 and -60 cm in medial direction, lateral position 0 cm. Fig. 4 and 5 show the emitted light spectrum for both positions for all Ai<1. In fig. 4 C=0.0005 and in fig. 5 C=0.0025. It is obvious that even for a small assumed absorption the emitted light spectra differs for the two source positions and the photon absorption emitted light peaks are shifted.

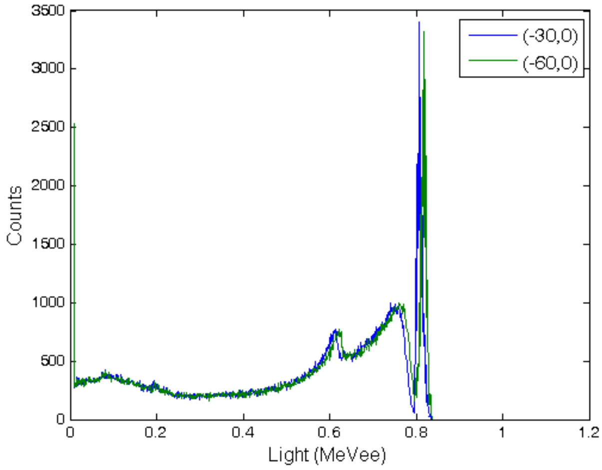

Fig. 4. The blue and green spectra show the emitted light in MeVee for ^{54}Mn placed at -30 respectively -60 cm in medial direction, lateral 0 cm for both. *C*=0.0005 and *A*=0.956 for the detector cell furthest away from the light guide, i.e. a small light loss.

The multiple peaks for C=0.0025 is due to the relative large differences in A between the 9 detector cells. If there was a continuing decrease of A the spectrum would most likely be more smeared without narrow peaks.

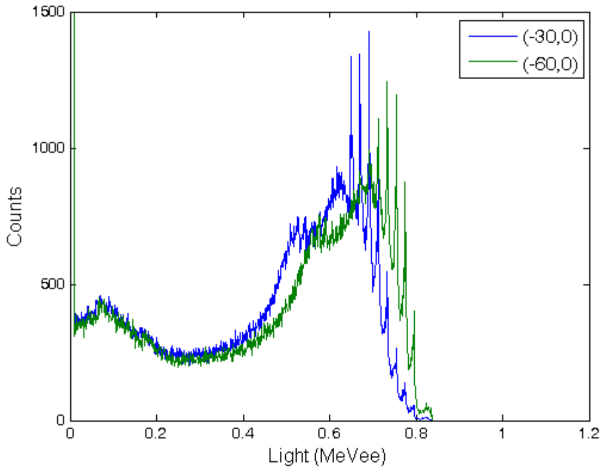

Fig. 5. The blue and green spectra show the emitted light in MeVee for ^{54}Mn placed at -30 respectively -60 cm in medial direction, lateral 0 cm for both. *C*=0.0025 and *A*=0.778 for the cell furthest away from the light guide, i.e. a rather high light loss.

Discussion

The comparison between calculated and total efficiency showed a good agreement for most source positions. Close to the detector edges the accuracy of the source position is quite important, and some deviations between measured and calculated efficiencies could be caused by a difference in stated and actual distance between detector housing and scintillation material. Also, when the source was placed at medial positions, it was assumed that the displacement was only in the medial direction when calculating the uncertainties, and vice versa for the lateral positions. Further analyses of the data results will give larger uncertainties and the calculated results might then be within the measured uncertainties limits.

Birk's constant *kB* was determined to be 9.65 mg cm^{-2} MeV^{-1}, for which there is a nonlinear response for energies below the photo absorption emitted light peak. But, there are other formulas describing the nonlinear light yield and it is also possible to use a linear response for all energies, the question is: what is most accurate? Birk's formula might not be the most accurate description, but it was easy to use together with the PHL tally, and for a detector described in this work the eventual nonlinear effects on spectrum broadening will most likely disappear in other more dominating broadening effects such as multiplication in the PMTs. Even if it is barely noticeable, the results does not imply that any formula would do, but it ought to be acceptable to use Birk's formula as long as one is aware of the assumptions that lie behind equation 2, which is carefully described by Birks (1964).

The results for different *A*, trying to mimic light absorption and light leaving the detector material, show that even for a small light loss the spectrum will be affected. For such large detectors as in this case it seems unlikely that light losses cannot be ignored and must be determined more exactly if an accurate spectrum broadening function is to be made. If so, a better representation of light losses must be done, the method described here gave crude results and how *A* is to be calculated must relate to the physical properties of the material. The attenuation length for NE 102A is 250 cm (Saint-Gobain 2005). The attenuation is described with Beer's law, but the conditions in the plastic scintillators, where light can be reflected, differs from the conditions described in Beer's law where incoming light is assumed to be parallel and attenuation is defined as light absorbed or scattered out of a narrow beam. This suggests that the absorption length would be longer than 250 cm, but how long and if the absorption increases linearly with length is not known. One possibility would be to do a Monte Carlo calculation of the light's path inside the detector material, but then one must know the reflecting index for the material surrounding the plastic scintillators and how long a particle track should be followed. The results show that the light losses are important, but at this moment it is yet to be determined how it should be modelled. The PHL tally has the advantage of being relatively easy and straight forward to use, and once the relationship between light losses and absorption site is more known, it ought to be possible to use the PHL tally to account for it.

Conclusion

Monte Carlo calculated total efficiencies correspond well with measured for several source positions relative to the detector. The effects of light losses are noticed in emitted light spectrum resolution, and for large detectors this ought to be properly accounted for when doing a spectrum broadening.

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Examination of patients using the whole body gamma ray counter that is calibrated according to the individual anatomy properties

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Abstract
Recently G.H. Kramer, K. Capello and E. Cardenas-Mendez [1] concluded that "whole body counters can now be calibrated more accurately using voxel phantoms, but the improvement is not as great as one might expect".

However, we have a slightly different view on the usefulness of voxel phantoms during the whole body counter-assisted examination of patients. Although the merits of voxel phantoms during calibration of whole body counters using the standard voxel phantoms are really not very significant, during the examination of patients the properties of individual anatomy can introduce considerable corrections into the whole body counting efficiencies (for low-energy gamma emitters, it may lead to the 2-fold increase/decrease of the assessed activity).

As an example, we performed whole body counter calibration and examination of patients, using both standard plastic phantom (developed by the Livermore laboratory, USA) and voxel phantom method (using the OEDIPE software developed at IRSN, France, and the well-known Monte Carlo code MCNP [2]). During our studies, we obtained the correction factors that are needed for the transition from the Livermore plastic phantom to the individual voxel phantom.

Introduction
Monitoring of internal radiation exposure for involved workers is one of the most actual problems in radiation hygiene of nuclear industry. Usually, the routine monitoring is performed using the so-called whole body and organ counters (WBC), i.e. generally semiconductor, more rarely scintillation gamma spectrometers that register the gamma rays, which are emitted by the incorporated radionuclides. In vivo monitoring data are especially important to make solutions on the medical measures that are to be taken in case of accidental radionuclide contamination of the nuclear workers. Modern in vivo counters possess high counting efficiencies for gamma rays through wide energy range, and provide high resolution in terms of absorbed energy values. However, they require the calibration procedure, i.e. determination of spectrometer counting efficiency (in other words, a relational factor between activities of incorporated radionuclides and counting intensities in the pulse-height-spectra of the gamma spectrometers).

The most biologically significant incorporated radionuclides (like ^{239}Pu and ^{241}Am) emit low-energy (13-60 keV) gamma rays. As a result, determination of WBC counting efficiencies is hampered due to intensive absorption and scattering of radiation in the patient's body. To circumvent these obstacles, one could use mathematical simulation (particularly Monte Carlo method as almost the only valid calculation method for such a purpose) rather than measurement of reference phantoms. However, such calculations require an adequate model of the patient's anatomy. This anatomy model could be described in terms of small rectangular boxes (voxels) with certain density and chemical composition.

Material and methods

The OEDIPE software

To provide the adequacy of mathematical simulation regarding to the physical measurements, the voxel representation of the patient's anatomy is retrieved from the x-ray computed tomography (CT) or magnetic resonance imaging (MRI) data. The software OEDIPE has been developed for creation of voxel phantoms of individual patients to conduct MCNP calculations. MCNP is a general-purpose Monte Carlo code, which performs transport calculations for gamma rays, neutrons and electrons in wide energy range. MCNP describes the transport geometry in terms of first- and second-order surfaces, as well as toroidal surfaces, thus enabling voxel representation of the patient's anatomy.

The MCNP input data file is automatically written by the OEDIPE interface. It contains the information on the patient's body in voxel representation, source characteristics, detector materials and geometry (described in terms of standard surfaces rather than voxels), the tally of interest (pulse-height-spectrum of the detector or dose distribution inside the body). To accelerate the calculations, the neighbor voxels with the same density and chemical composition are coupled into larger rectangular boxes using the maximal rectangular area technique. To visualize the transport geometry written in the MCNP input data file, the Sabrina software is used.

Whole body counting lab at FMBC

At the Burnasyan Federal Medical Biophysical Center (FMBC, Moscow, Russia), four Canberra germanium WBCs are placed into a shielded canyon, which shielding is made of exceptionally old steel. This steel was produced in 1930s, i.e. before the nuclear weapon tests had begun. Thus, this special shielding, which does not contain the artificial radionuclides, provides low radiation background inside the canyon. Moreover, the air inside the canyon is being ventilated during the in vivo examination, thus decreasing the radon activity, and creating the exceptionally low background intensity.

The preliminary test of the experimental equipment, measurement technique and voxel phantom-assisted calibration of whole body counters was done by whole body counting of plastic torso phantom developed by Livermore National Lab. ^{241}Am was to be measured using the Canberra whole body counters. This radionuclide is usually incorporated together with other actinides in the lungs of nuclear workers; it has the relatively high (59.54 keV) energy of major portion of emitted gamma rays. During the measurements, perforated lungs of the Livermore phantom contained 27 capsules with ^{241}Am of 3165 Bq total activity. Measurements were conducted by two spectrometers placed at the "chest" of the phantom in from of right and left lung.

Using these WBC devices, we examined exam three patients of the FMBC clinic (see Fig.1). Case histories of all patients (retired employees of Seversk Chemical Plant (Seversk town, Tomsk region) indicate to both inhalation (under routine operations) and wound intake (puncture wounds of hands during industrial operations) of non-soluble actinides, including ^{241}Am compounds in 1960-ties. Within long period of time, patients were regularly examined (natural excreta analysis, whole body counting) and treated (intravenous administration of Ca-DTPA (Pentacin), soft tissue dissection in the wound site) in medical facilities of FMBA of Russia (former 3rd Major Department of the USSR Ministry of Health).

To obtain the voxel phantoms of the patients as well as the Livermore plastic mannequin, we scanned the ex-workers and the Livermore phantom using the Toshiba Aquillion CT machine.

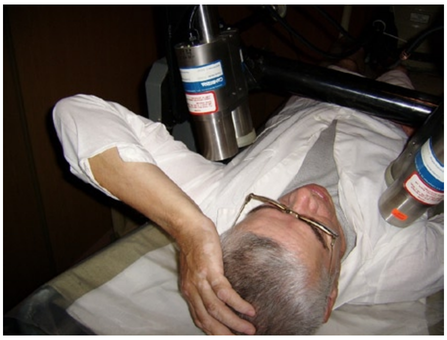
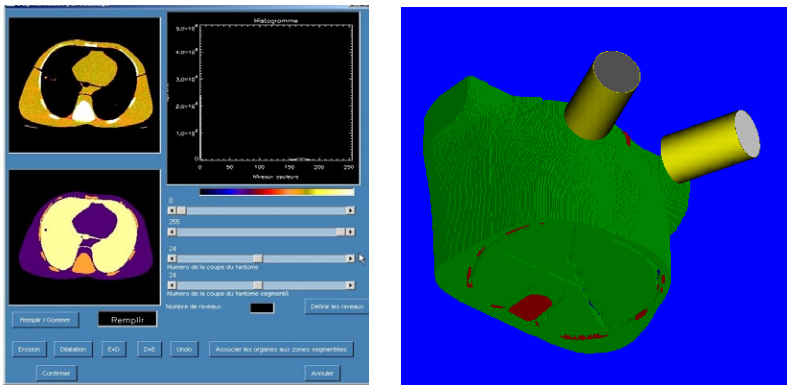


Fig. 1. Lung examination in patients at FMBC clinic.

Results and Discussion

Experiments with the Livermore plastic phantoms

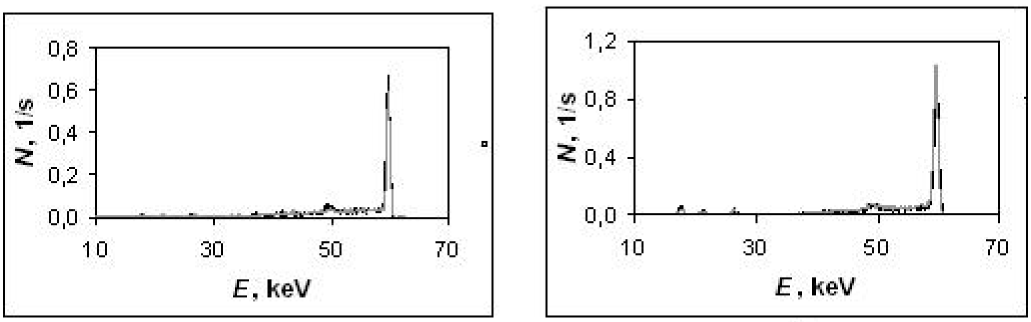
The Livermore phantom counting conditions were simulated using the MCNP software in voxel geometry (256x256x51 voxel resolution) that was retrieved using the OEDIPE software from the CT images obtained at the Burnaysan FMBC clinic. Fig. 2 provides results of OEDIPE software calculations: separation of major organs in the phantom image (panel a) and radiation transfer geometry record in the input data file of MCNP code (panel b). MCNP calculations for plastic phantom have used the source term as multiple points; the point positions were determined by the capsule places in the CT images of the phantom.



a b

Fig. 2. The Livermore phantom CT images processing for WBC calibration. Panel a: separation of organs and tissues applying OEDIPE software. Left upper window: initial CT image and segmented image below (colors are related to the tissue types); right upper window: spectrum of initial image according to the pixel brightness. Panel b: radiation transfer visualization with Sabrina software, which was recorded by OEDIPE software as the input data file for MCNP calculations. The color corresponds to the type of the tissue or substance. Number of voxels: 256×256×51.

The experimental and calculated spectra of whole body counter detectors placed in front of left (panel a) and right (panel b) lung of the phantom are shown by Fig. 3. Table 1 contains data on recalculation factor between the intensity of the photo peak (59.54 keV) and the total activity of ^{241}Am for right and left detector twins. Calculations did not use either counting efficiency correction versus the absorbed energy or spectrum standardization, so the calculated values of Fig. 2 and Table 1 are absolute ones rather than relative ones. The average quadratic deviation in the maximum pulse counting channel for calculated spectrum is about 6% for 1 million histories of random tests, which requires about 20 minutes of calculation time at personal computer (3 GHz, 512 Mb RAM).



a b

Fig. 3. Experimental (light colored) and calculated (dark colored) spectra of the left (panel a) and right (panel b) counting of ^{241}Am gamma radiation for lungs of Livermore National Lab plastic phantom.

Table 1. Experimental and calculated factors for recalculation of the photo peak (59,54 keV) intensity and the total activity of ^{241}Am for Livermore phantom and detectors placed in front of left and right lung.

Detector	Calculation (*C*), $s^{-1}\cdot kBq^{-1}$	Experiment (*E*), $s^{-1}\cdot kBq^{-1}$	Difference *D* = (*C*-*E*)/*C**100%
Left	0.85	0.78	+9
Right	1.41	1.55	-6

The comparison of calculation/measurement demonstrates that calculations have 10% and better accuracy to reproduce absorbed energy spectrum of Canberra Industries detectors both in the photo peak and in the Compton scattering range.

Examination of retired nuclear workers

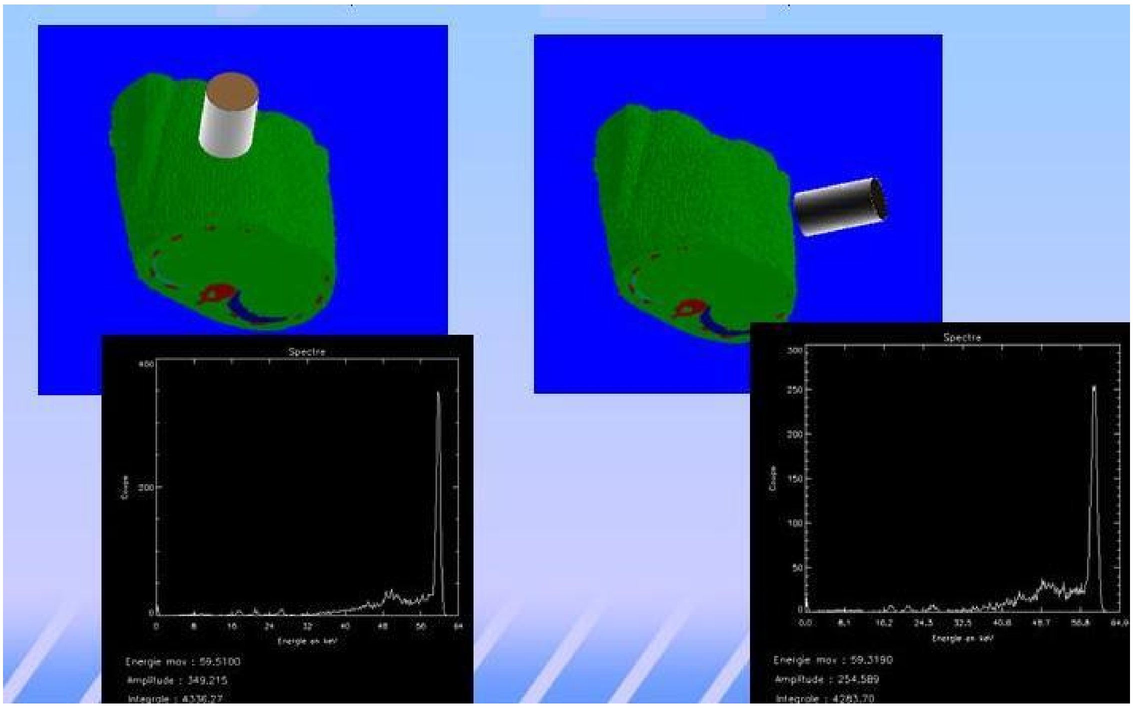


Fig. 4. ŒDIPE computer code application to assess re-calculation factors for counts in the spectrometry channels into ^{241}Am incorporated activity for one of retired employee of atomic industry. In the top: MCNP phantoms (Sabrina software applied for visualization). In the bottom: absorbed energy spectra calculated according to MCNP computer code for homogeneous lung distribution of ^{241}Am of one unit of activity (ŒDIPE software applied for visualization).

To determine ^{241}Am calibration factors in different spectrometry channels, ŒDIPE software was used basing upon CT chest images of patients (256x256pixel resolution for each slice) to create their MCNP voxel phantoms (Fig. 4, on the top). When preparing MCNP phantoms, the incorporated radionuclide distribution was adopted to be homogeneous in each lung.

Applying MCNP4c2 computer code, absorbed energy spectra were calculated for each of detectors placed in front of left and right lung of each patient; the homogeneously distributed ^{241}Am source of one unit of the activity was adopted. The photopeak intensity corresponding to the energy of major portion of photons emitted by ^{241}Am (59.54 keV) has provided the assessment of re-calculation factors.

Experimental calibration factors were determined by Livermore phantom measurement and set of ^{241}Am point sources placed in the "lungs" of phantom. ^{241}Am and ^{239}Pu measurement results are given by Table 2 for patients.

Table 2. ^{241}Am and ^{239}Pu measurement results for patients

Patient	Patient B.		Patient S.		Patient O.	
Lung	left	right	left	right	left	right
Full absorption peak intensity at 59,56 кэВ (N), c^{-1}	10,3	1,8	< 0, 1	0,80	0,43	1,25
Individual (OEDIPE+MCNP) assessment of ^{241}Am lung burden, ($A_C = N/K_C$), кБк	5,1	0,90	< 0, 04	0,31	0,39	0,95
Assessment of ^{241}Am lung burden using the Livermore calibration phantom, ($A_L = N/K_L$), кБк	6,9	0,6	< 0, 07	0,26	0,28	0,40
Relative error in lung burden assessment of ^{241}Am activity, $\eta = (AC - AL)/AC \cdot 100\%$	-35%	+33%	-75%	+16%	+28%	+57%
Correction factor $K_u=A_L/A_C$	1,35	0,67	1,75	0,84	0,72	0,42

The current value of radionuclide burden is proportional to the current value of the internal exposure dose rate. Concerning the internal exposure dose accumulated within long period of time since the examination and, moreover, concerning the retrospective internal exposure dose assessment, such values depend on biokinetics parameters of incorporated actinides, which are subjected to the significant individual variability as well as depend on medical measures accelerating radionuclide body excretion.

Besides, neither the exact portions of different intake pathways, nor the initial isotope composition of the intake are not known well. Large number of unknown factors affecting the internal exposure accumulated dose makes its assessment very approximate. To increase the accuracy and reliability of accumulated dose assessment, it is convenient to elaborate the periodical monitoring of radionuclide burdens in different organs and tissues of the patient.

Conclusions

1. The developed computational and experimental method for internal exposure monitoring was implemented for whole body counting of incorporated actinide patients.
2. The original technique for incorporated radionuclide in vivo measurement is elaborated, which not only increases the measurement accuracy significantly (with excluding the error component related to the incompliance of plastic phantom structure to the individual patient's anatomy) but also provides the uncertainty assessment of routine measurement technique based on plastic phantom calibration for WBC measurement result interpretation.
3. ŒDIPE computer code also provides the internal exposure dose distribution, which finding was tested, when solving industrial hygiene task (radioactive contaminated wound assessment).
4. The application of radiometric monitoring combined with Monte Carlo computational method as the long term program for persons involved in the internal exposure accident victims would significantly increase the reliability of in vivo monitoring of incorporated radionuclides and assessments of cumulated doses of internal exposure.

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An extrapolation ionization chamber for α-ray detection and measurement

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Abstract

The extrapolation ionization chamber is a special type of the cavity ionization chamber, developed by Böhm (Böhm, 1983) for the absolute measurement of the absorbed dose rate in β-ray beams. The cavity theory was first created for the absorbed dose measurement in photon radiation, but it was also created for charged particles. In the radiation metrology laboratory from IFIN-HH we built such an extrapolation chamber for β-ray, but we tried to find how it would work when irradiated with α-ray.

Firstly, we obtained the experimental curves of the ionization current versus the polarizing voltage, *U*, the former have quite a net saturation region. Then, we measured the saturation current of the chamber for different values of the distance between the electrodes (i.e. different values of the sensitive volume of the chamber). When we represented the graph of the ionization (saturation) current against the distance between the chamber's electrodes, we noticed that this curve is quite analogous to the same curve obtained for β-ray irradiation.

So, by analogy, we concluded that for this chamber, the main requirements of the cavity theory are also fulfilled when the chamber is irradiated with α-ray.

This paper presents the results of the measurements concerning the $I=I(U)$ and $I=I(x)$ characteristic curves of the extrapolation chamber when it is irradiated with α-ray from a Pu-239 radioactive source. Some concluding remarks are also included, concerning the possibility of developing an absolute method for the measuring the absorbed dose for α-ray, based on the cavity theory.

Introduction

The detection and measurement of α-rays are procedures that repeat the general requirements that are imposed to ionizing radiations. In the same time, they must take notice of the characteristics of this type of radiations.α-rays are He nuclei (and therefore have a big rest mass- He^4_2 and (2+) electrical charge) that undergo Coulombian interactions which make them release energy quickly into the medium they pass through. It is for this reason that the path of the α-rays in various mediums is considerably smaller (for example 3.75 cm through air). In solid mediums, α-rays have much smaller paths and are completely absorbed in depths of tens or hundreds of microns.

The properties mentioned above impose that specific methods are elaborated and adequate detectors are designed. One of the methods for the detection and measurement of ionizing radiation is the ionometric method, which is based on the collecting and measuring of the electrical charges produced by ionization inside the gas from the sensitive volume of the detector.

As far as α-rays are concerned, it is necessary that the detector (for example an ionization chamber) have a sufficiently thin window that permits the passing of particles without significant loss of energy; hence, the particles that reach the sensitive volume can produce a sufficient number of ionizations. The ionizations lead to the outcome of an electrical signal that can be measured with adequate precision at the output of the detector.

Initially, the cavity theory, which correlates the generation of ionization electrical charges inside the gas from a cavity, with the energy released in the material surrounding the cavity, was elaborated for photon radiation. Further on the theory was adapted also for radiations made of particles with rest mass and electrical charge Trott (Trott, 1966).

The most complex elaboration of the cavity theory for β-particles was presented by Böhm (Böhm, 1983) he also designed an ionization cavity-chamber with variable volume (the extrapolation chamber). The latter is destined to measure the absolute value of the released energy (meaning the absorbed dose) in a particular medium for the β-radiation.

Material and methods

This type of ionization chamber (extrapolation chamber) was also put into practice by the Radiation Metrology, Testing and Dosimetry Team (CMRID) of the Horia Hulubei National Institute for R&D in Physics and Nuclear Engineering, IFIN-HH.

Description of the ionization (extrapolation) chamber built by IFIN-HH

The ionization chamber has a cylinder shape with plane-parallel electrodes Fig. 1. The voltage electrode (polarization electrode) is made up of an electroconductory layer (an aluminium layer with a thickness of a few microns), which is placed on one of the sides of a very thin sheet of Mylar (70 mg/cm^2); this electrode is also the input window of the detector. The collector electrode is made up of a graphite layer placed on a polystyrene block. This graphite layer is electrically isolated from the guard electrode (guard ring). The latter consists of a circular layer of graphite. The guard ring is connected to the detector's ground and is designed to define the sensitive volume of the detector from an electrical point of view.

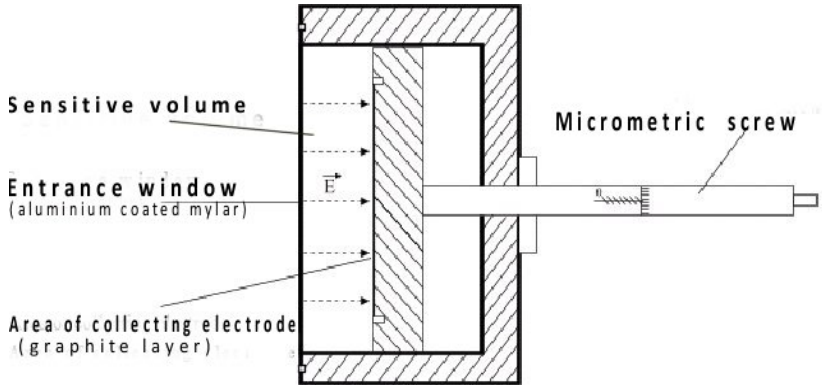
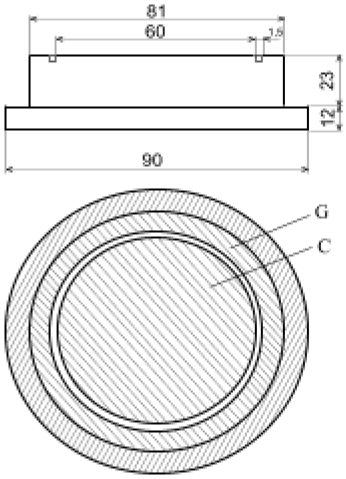


Fig. 1. Schematic view of the extrapolation chamber.

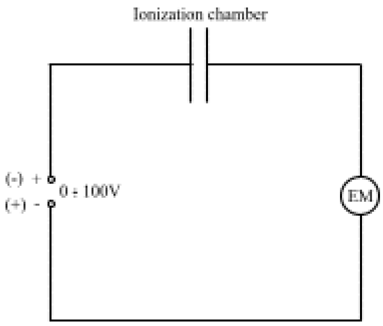
The sensitive volume of the detector can be modified mechanically by varying the distance between the polarization electrode and the collector electrode with a micrometric screw, in the range of 5-16 mm. In Fig. 2 we present the shape and size of the electrodes.



C – collecting aria, G – guard ring electrode

Fig. 2. The shape and size of the electrodes of the extrapolation chamber.

In figure 3 we present the electrical circuit used for measuring the ionization charge (or current) of the ionization chamber, Bercea et al (Bercea et al. 2008a, Bercea et al. 2008b).



EM - electrometer

Fig. 3. The electrical circuit for measuring the ionization charge / current.

The experiment

As we have shown in our introduction, in order to calculate the rate of the absorbed dose, $\dot{D}_\alpha$, we use the relationship (1). This includes the slope of the curve ionization current versus distance *I=I(x),* $\left(\frac{\partial I}{\partial x}\right)$, in the range where it is linear. This is extrapolated for $x \rightarrow 0$, meaning where the perturbator effect on the radiation that interacts with the solid medium that surrounds the cavity filled with gas (air) disappears.

$$\dot{D}_\alpha = K\frac{W}{eS\rho}\left(\frac{\partial I}{\partial x}\right)_{x\rightarrow 0} \qquad (1)$$

K - constant which takes in to account the units of measure
W - the average energy required to produce the air ionization
e - the elementary electric charge
S - the aria of the collecting electrode
ρ - the air density in the sensitive volume of the ionization chamber

In order to see to what extent the ionization chamber made by IFIN-HH acts according to the cavity theory, which refers to α-radiation, the following experiments were made.

The experimental drawing of the curves ionization current versus polarization voltage, *I=I(U),* for a given source of α-radiations in a well-defined geometry. The geometry corresponded to the various distances between the polarization electrode and the collector electrode, *x*, and also the detecting of the range of values of the polarization voltage, values within which the detector is functional.

Choosing a value for the polarization voltage that lies within the saturation range for all the values of the distance between the electrodes, *x*, the graphical representation of the curve that gives the ionization current *I* versus the distance between the electrodes of the chamber, is determined experimentally.

Results

The measurement of the ionization current

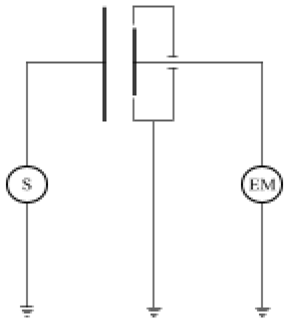
The ionization current generated by the detector was measured in the presence of a source of α-rays, in a fixed irradiation geometry. The α-ray source used in this case was a Pu^{239} source having the following characteristics:

– -extended, circular source with a diameter of 120 mm;
– -emission rate: ε= 13 601 α/s in 2πsr.

The source was positioned with the aid of a special support ring placed at a distance of 2 mm from the surface of the window of the ionization chamber.

The ionization current was measured with a Keithley 617 electrometer.

The arrangement used for measuring the ionization chamber is given in Fig. 4.



S – voltage source; EM - electrometer

Fig. 4. The experimental arrangement for measuring the ionization current.

Considering the order of magnitude of the ionization current (about 1 pA) during the measurements, certain precautions had to be made, in order to reduce the influence of the working conditions on the results of the measurement. Thus, in the working chamber, the temperature was held constant (20 ± 2º C); any actions that may cause powerful noises or vibrations were avoided, as well as the movement of people in the proximity of the measurement stand (less than 1.5 meters from the stand).

Before installing the radioactive source on the stand, we measured the current of the ionization chamber for the value “0” of the polarization voltage. The value we obtained in this case was ±5.6 x 10^{-15}A. After installing the Pu^{239} source on the stand, we waited 2 hours before starting the measurements.

The first important observation was that for U = 0 in the presence of the radioactive source, the ionization chamber presented a positive current , whose values depended on the value of the distance between the electrodes of the chamber (Table 1). A first assumption is that it could be due to the alpha particles reaching the collecting electrode, in absence of any polarization.

Table 1. The values of the ionization current (I) for 3 values of the air gap, for *U* = 0.

x (mm)	I (A)
5	$1.0074 \cdot 10^{-12}$
7	$1.960 \cdot 10^{-12}$
9	$0.2388 \cdot 10^{-12}$

For the three values chosen for the distance between the electrodes, x, we experimentally obtained the characteristic curves I=I(U). The results we obtained are presented in Fig. 5.

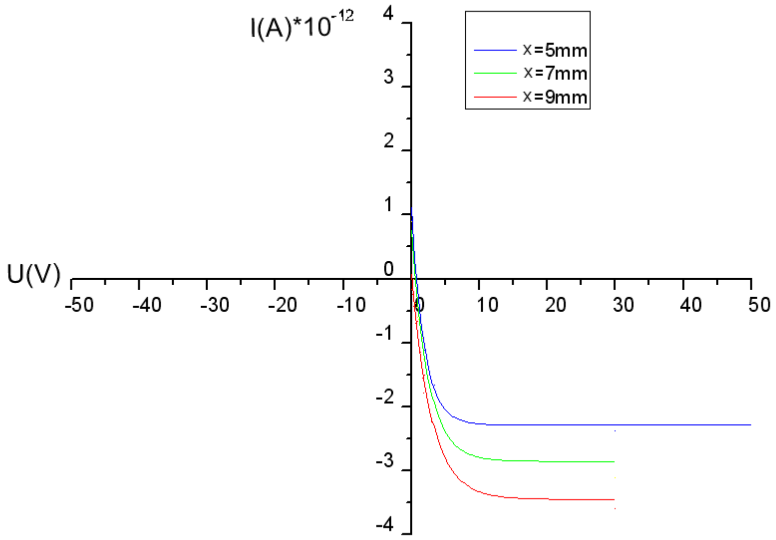


Fig. 5. The I=I(U) curves for three values of x.

For the value of the polarization tension U=15 V situated in the saturation region of the I = $I(U)$ curve we made measurements of the ionization current and we also obtained the curve that expresses the ionizing current versus the distance between the electrodes (the gap depths), x, respectively the volume of the chamber (V), considering the fact that $V = S*x$, and the surface of the collector electrode S = constant.

The results obtained are presented in table 2 and the curve that resulted is presented in Fig. 6.

Table 2. The ionization current corresponding value of the gap depth (*x*).

x (mm)	$I*10^{-12}$ (A)	x (mm)	$I*10^{-12}$ (A)	x (mm)	$I*10^{-12}$ (A)
1	0.48	6	2.8	11	4.4
1.5	0.7	6.5	3	11.5	4.5
2	1	7	3.1	12	4.5
2.5	1.22	7.5	3.2	12.5	4.5
3	1.52	8	3.4	13	4.5
3.5	1.7	8.5	3.6	13.5	4.5
4	2	9	3.8	14	4.5
4.5	2.3	9.5	4	14.5	4.5
5	2.4	10	4.2	15	4.5
5.5	2.5	10.5	4.3		

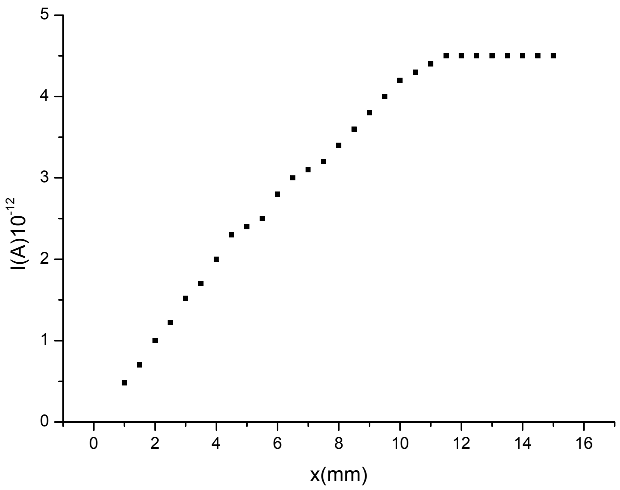


Fig. 6. The *I* = *I(x)* characteristic curve.

Discussion

The analysis of the results obtained regarding the $I=I(U)$ characteristic of the ionization chamber, in the presence of a α-radiation field, as above presented Fig. 5, leads to the conclusion that in the presence of these radiations, the detector has a typical behaviour. The ionization current reaches the saturation value fairly quickly for all the possible values of the distance between the electrodes.

In saturation conditions, we have concluded that the $I = I(x)$ curve also has a typical behaviour for this type of detector, respectively, for $x \rightarrow 0$, the ionization current is proportional to the distance between the electrodes, x, in the range 0 to 4.5 mm, such as the Figure 6 shows, and thus, with the sensitive volume of the chamber. This behaviour indicates the fact that, in the presence of a α-radiation field, the ionization chamber with a variable volume produced by IFIN-HH has the same behaviour as when irradiated with β-radiations.

Conclusions

The experimental results we obtained, especially the shape of the $I=I(x)$ curve, which is linear within the range 0 to 4 .5 mm, and consequently when $x \rightarrow 0$, are an experimental confirmation for the fact that the cavity theory, as presented for the β-radiations /1/, may also be used, in the same experimental configuration, for measuring the absorbed doze in α-radiation fields and beams. All that is left is that further research leads to the identification of all the quantities necessary for obtaining the absorbed doze of α-rays. This relation would connect the absorbed dose rate $\dot{D}_{\alpha}$, to the variation of the ionization current, $\left(\frac{\partial I}{\partial x}\right)$, which would be analogous to the relation for the β ray.

Acknowledgements

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Mesenchymal stem cells as drug cells for radiation burn treatment

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Abstract

Local radiation syndrome is marked by necrosis that may extend to the deep subcutaneous structures. Today, treatment is surgery, excision, graft and flap with sometimes bad results. It has been suggested that Mesenchymal Stem Cells (MSC) therapy could be used in order to treat numerous tissue lesions. We have performed a novel therapeutic approach of local radiation syndrome by using local autologous MSC therapy combined to surgery. For this purpose, autologous bone marrow cells were collected from the unexposed iliac crest. For GMP production, MSC were expanded in a closed system (MacoPharma partnership) containing an innovative serum free medium supplemented with human platelet lysate as previously described (Doucet et al., J. Cell Physiol., 2005). Quality control assays evidenced that expanded cell population retained typical MSC characteristics and did not exhibit chromosomal abnormalities. As previously demonstrated, MSC produced many cytokines and growth factors which could have a critical role in improving the healing process by counteracting the local inflammatory waves and by promoting the autologous skin engraftment.

We believe that MSC act as drug cells delivering in situ in the lesion growth factors which contribute to the healing of the lesion. We have also demonstrated that after in vitro cell activation, the conditioned medium of MSC exhibited a similar effect on wound healing than that obtained with freshly expanded MSC. In case of caryotypic abnormalities occurring after in vitro MSC expansion, the use of MSC conditioned medium could be considered as a relevant alternative of MSC therapy. Other sources of MSC such as adipose tissue, gingival mucosa are also taken in consideration in view of setting up an allogeneic stem cell bank.

A new therapeutic approach for radiation burns combining surgery and mesenchymal stem cell administrations: About four cases

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Abstract

The physiopathological mechanisms of severe radiation burns are well described and their treatments are well codified but very difficult to perform with an important functional and vital risk. We present four patients with accidental local radiation burns and propose a new therapeutic approach combining surgery and local stem cell therapy.

The first patient had local radiation burns of left fingers and left buttock (Chile, 2005). We performed early excision of the irradiated part of the buttock after dosimetric reconstruction. We covered the buttock and the fingers with full thickness skin graft. Autologous Mesenchymal Stem Cells were locally administrated in the lesion in combination with surgery. The second patient had a very important radiation-induced skin necrosis located to the left arm from the shoulder to the elbow (Senegal, 2007). The surgical procedure used a pedicle latissimus dorsi muscle flap and a proximal forearm ante bracchial flap after a very large excision of skin and triceps muscle. Several Stem Cell administrations were combined to the surgery after many bone marrow collections. The third patient had a local radiation burn of the hands (Tunisia, 2008). Full thickness skin grafts were combined with local stem cells administrations. The fourth current case presented a local radiation-induced burn of the limb and was also treated by surgery and local stem cell therapy (Equator, 2009). We obtained a complete and stable healing in all cases. Stem cell therapy using autologous expanded MSC has to be considered as an adjuvant treatment of the surgery corresponding to excision of necrosis tissues and flap reconstructions.

Our results demonstrate that this new therapeutic procedure using surgery and local stem cell therapy is very promising. We believe that this innovative approach could improve the treatment of local radiation burns in term of functional and vital results.

Experience of mesenchyme cell's therapy in case of severe local radiation (x-ray) injure of back tissues

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Abstract

On the 13-th of January of 2008 patient, 61 years old, addressed to admission department of Burnazian Federal Medical Biophysical Center of Federal Medical Biological Agency.

Patient asked for medical help to different hospitals and at 350-th day after exposure diagnose of local radiation injure (LRI) was defined. Unsuccessful try of iterative autografting was made. At 592-nd day after exposure patient was hospitalized to Burnazian Federal Medical Biophysical Center of Federal Medical Biological Agency.

The result of conservative therapy was decreasing volume of fibrin and non-active growth of granulations. In connection with depth and area of wound conservative therapy was not successful, patient was need surgical treatment. But his cardiac pathology did not allow surgical treatment.

This situation was the reason to looking for new strategies of treatment of LRI.

Scientific dates and little clinical experience allowed to use autologous mesenchyme cell therapy to cover wounds in cases of LRI in combination with conservative therapy.

Result of using of new technology was active growth of granulation, active peripheral epithelization and total epithelization of ulcer.

In conclusion, this clinical case shows opportunity to use modern hi-tech medical methods in combination with conservative therapy to treat consequences of severe local radiation injures. At present this strategy of medical management is preferable method for patients with severe LRI and with contraindication to surgery.

Introduction

On the 13-th of January of 2008 patient, 61 years old, addressed to admission department of Burnazian Federal Medical Biophysical Center of Federal Medical Biological Agency. Anamnesis that he told: during long time (about 30 years) he severe from coronary heart disease, in 1996 survived coronary artery bypass grafting, in 2000 - cardiac infarction. In July of 2006 because of frank coronary atherosclerosis and severe angina pectoris try of stents installation was made under fluoroscopic control. Because of technical and anatomical troubles length of operation was about 3½-4 hours.

During early postoperative period patient noticed appearing of itch and little vesicles on interscapular region. In 2 months this symptoms regressed and changed to infiltration and pain syndrome.

Patient asked for medical help to polyclinic near his home, situation was defined like dermatitis with unclear etiology and non-specific treatment was applied.

At 280-th day after exposure black crust appeared in the center of infiltration. Patient used applications with different tinctures, as a result black crust was removed and ulcer was fined under it.

Patient asked for medical help to different hospitals and at 350-th day after exposure diagnose of local radiation injure (LRI) was defined. Unsuccessful try of iterative autografting was made. At 592-nd day after exposure patient was hospitalized to Burnazian Federal Medical Biophysical Center of Federal Medical Biological Agency.

St. Localis: there was ulcer on interscapular region, measurement 5x5x4 cm, bottom of ulcer covered by fibrin, skin around ulcer with fibrous and atrophic changes, palpation aroung ulcer was without pain syndrome.

On the base of anamnesis and objective status diagnose was defined: consequence of severe LRI of back tissues after x-ray exposure.

To define severity and area of local radiation injury following procedures were carried out:

- Radioisotopic study of thorax
- Electron spin resonance of thorax
- Cytogenetic analysis of lymphocytes
- Ultrasonic scanning of back tissues
- Bacteriological analysis of wound microflora

On the base of calculation and results of cytogenetic analysis local dose was about 20-25 Gy without total exposure. The patient was treated according to common pathogenetic mechanisms of LRI:

- Local therapy (specific non-adhesive bandage with antiseptics and antibiotics)
- Disaggregating therapy
- Stimulation of regeneration
- Disintoxication therapy
- Antibiotic therapy

Wound was contaminated with pseudomonas aeruginosa, which was eliminated by systemic antibiotic therapy.

The result of conservative therapy was decreasing volume of fibrin and non-active growth of granulations. In connection with depth and area of wound conservative therapy was not successful, patient was need surgical treatment. But his cardiac pathology did not allow surgical treatment.

This situation was the reason to looking for new strategies of treatment of LRI.

Scientific dates and little clinical experience allowed to use mesenchyme cell's therapy to cover wounds in cases of LRI in combination with conservative therapy. For this aim scientific protocol was worked out, which was passed by ethical committee of

Burnazian Federal Medical Biophysical Center of Federal Medical Biological Agency. Patient subscribed assent to use new medical technology for treating his disease. After that 20 ml of fat was taken. In sterile conditions according standard method, stem mesenchyme cells was separated and cultivated in autologous serum to quantity 1-1,5*10^6. After that 3-5*105 of them was cleaned and injected around ulcer in 10 points.

Results

Result of using of new technology was active growth of granulation, active peripheral epithelization and decreasing of wound measurement on 90% during 1 month.

For stimulation of epithelization of wound with lot of granulation was made successful iterative autografting. We got total epithelization of ulcer during following 3 weeks.

Conclusion

This clinical case shows opportunity to use modern hi-tech medical methods in combination with conservative therapy to treat consequences of severe local radiation injures. At present this strategy of medical management is preferable method for patients with severe LRI and with contraindication to surgery.

New haematological criteria of acute radiation sickness severity

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Abstract
Diagnostic criteria of acute radiation sickness (ARS), which based on peripheral blood cells, were described in numerous publications. However, not all suggested indices meet the requirements of accurate and early determination of ARS severity and only some of them could be used for integral estimation of hematopoiesis damage caused by radiation. Retrospective analysis of hemograms was carried out in 136 patients who suffered from Chernobyl accident. From this amount 32 patients survived ARS grade 1 (mild), 31 grade 2 (moderate), 8 grade 3 (severe), and 65 persons was irradiated in doses lower 1 Gy. As the new criteria we suggest following indices: a day of granulocytes primary decrease to 2×10^9, 1×10^9 and 0.5×10^9 L^{-1}, and a day of thrombocytes primary decrease to 100×10^9, 50×10^9 and 30×10^9 L^{-1}. The ANOVA statistical analysis showed that the patients, who were discriminated by ARS grade, demonstrated highly significant difference of time period from irradiation (day 0) till granulocytes and thrombocytes came down to above-mentioned levels. For integral estimation of hematopoiesis damage it is suggested the index of summary deficit of peripheral blood granulocytes and thrombocytes count between 10th and 55th day after the irradiation. It was calculated as the square of a figure (or several figures) on a graph, which basis is a straight line meaning the low normal level of granulocytes (2×10^9 L^{-1}) or thrombocytes (150×10^9 L^{-1}) count. Slanting lines granulocytes and thrombocytes decrease or increase on the graph serve as the figure sides. Any calculation follows the graphic presentation of these mature cells dynamics. The summary cells deficit significantly correlated with doses of irradiation. The criteria that were suggested enabled to estimate the severity of radiation damage and start the adequate treatment already on the early stage of ARS.

Introduction
The Chernobyl accident showed that in a case of unforeseen radiation exposure, resulting in ARS development in man, methods of biological dosimetry become the only available for estimation of radiation dose and a rate of sickness severity. Peripheral blood indices are used for this purpose more frequent. Thus G. Andrews (Andrews et al. 1965; Andrews 1980) was one of the firsts who suggested using peripheral blood

lymphocytes count on 1st and 2nd days after radiation exposure for evaluation of whole-body radiation dose and radiation injury rate. Research workers of Moscow Institute of biophysics (Gus'kova et al. 1987; Baranov et al. 1995) used nomographic charts for absorbed radiation dose estimation. These nomograms are based on lymphocytes content in peripheral blood during the first 9 days after irradiation. These researchers determined ARS grade by leukocytes count on the 8-9th days after exposure. D. Densow et al. (1996) consider that it is possible to predict a depth of hemopoiesis injury by granulocytes content in peripheral blood between the 4th and 12th days following irradiation. In accordance with gradation, which they suggested, the injury of hemopoietic tissue changes from transient slight granulocytopenia to entire depression of hemopoiesis.

However all above-mentioned methods of biological dosimetry are not absolutely reliable because they demonstrate essential discrepancy in dose value and ARS grade if applies in the same patient. This fact makes the search of new criteria for estimation of radiation exposure rate and ARS grade is sufficiently topical.

Material and methods

The retrospective analysis of patient's histories was done for the purpose of peripheral blood cells count during first two month after the irradiation. There was 71 patients in the study program who suffered ARS in a result of the Chernobyl accident, including 32 persons survived ARS grade 1 (mild), 31 ARS grade 2 (moderate) and 8 ARS grade 3 (severe). ARS grade determined according to classification had been worked out by A. Gus'kova et al. (Mettler et al. 2007). Additionally 65 patients without clinical and haematological signs of ARS were included in the study (ARS 0). Their whole-body radiation doses were evaluated cytogenetically by count of dicentric in peripheral blood lymphocytes (Gus'kova et al. 1987). Age-specific and dosimetric characteristic of the examined patients are submitted in table 1.

Table 1. Age-specific and dosimetric characteristic of the examined patients.

Indices	ARS 0	ARS grade 1	ARS grade 2	ARS grade 3
Number of patients	65	32	31	8
Gender: M / F	61 / 4	29 / 3	30 / 1	8 / 0
Age at exposure, years mean ± SD min – max 95% confidence interval	 34.9±10.1 20.8-59.4 32.4-37.4	 33.6±8.8 17.6-56.3 30.4-36.7	 38.7±13.9 20.7-79.3 33.6-43.7	 35.8±12.5 20.4-55.1 25.3-46.3
Whole-body radiation dose, Gy n[a] mean ± SD min – max 95% confidence interval	 0.3±0.2 0.1-0.6 0.2-0.4	 1.0±0.6 0.1-3.3 0.7-1.2	 2.4±0.9 0.5-4.2 2.0-2.7	 5.4±1.0 3.9-7.1 4.4-6.3

Note: a – number of patients with the dose evaluated in 1986

Results

As the new haematological criteria for ARS grade evaluation there were studied the following indices: a day of granulocytes decrease to 2×10^9 (lower norm limit), 1×10^9

(the agranulocytosis point), and 0.5×10^9 L^{-1} ("the day of 500 neutrophils" according to Gus'kova et al. 1987), as well as granulocytes the most minimal level and the day, when it came, and a day of thrombocytes decrease to 100×10^9, 50×10^9 and 30×10^9 L^{-1} (the last two points mean the critical level of thrombocytes when risk of spontaneous internal bleeding and haemorrhages becomes high), and thrombocytes the most minimal level and the day when it came.

Statistical analysis ANOVA (analysis of variances) revealed the high significance of intergroup difference between days after exposure when granulocytes decreased to afore-mentioned levels (table 2).

Table 2. The duration of time period from irradiation till the day when peripheral blood granulocytes decreased to certain levels (mean ± SD).

ARS grade	Day after irradiation when granulocytes decreased to:				Granulocytes minimal level, $\times10^9$ L^{-1}
	2×10^9 L^{-1}	1×10^9 L^{-1}	0.5×10^9 L^{-1}	the most minimum	
1	27±8 (n=32)	31±7 (n=23)	32±3 (n=10)	36±4 (n=32)	0.7±0.5 (n=32)
2	13±6 (n=30)	19±7 (n=30)	24±5 (n=30)	29±4 (n=30)	0.1±0.1 (n=31)
3	9±5 (n=8)	14±5 (n=8)	16±5 (n=8)	22±3 (n=8)	0 (n=8)
ANOVA, F (p)	40.0 (0)	29.9 (0)	26.3 (0)	47.8 (0)	47.8 (0)

The time periods when peripheral blood thrombocytes decreased to 100×10^9, 50×10^9, 30×10^9 L^{-1} and the most minimal levels differed significantly as well as thrombocytes the most minimal values in patients with various ARS grades (table 3).

Table 3. The duration of time period from irradiation till the day when peripheral blood thrombocytes decreased to certain levels (mean ± SD).

ARS grade	Day after irradiation when thrombocytes decreased to:				Thrombocytes minimal level, $\times10^9$ L^{-1}
	100×10^9 L^{-1}	50×10^9 L^{-1}	30×10^9 L^{-1}	the most minimum	
1	23±5 (n=31)	26±5 (n=24)	28±4 (n=10)	29±5 (n=32)	36±24 (n=32)
2	15±3 (n=28)	18±3 (n=28)	21±4 (n=30)	26±4 (n=31)	10±6 (n=30)
3	10±2 (n=7)	13±1 (n=7)	14±2 (n=7)	22±5 (n=8)	3±3 (n=8)
ANOVA, F (p)	43.3 (0)	41.5 (0)	37.9 (0)	23.0 (0)	10.1 (0)

The high statistical significance in difference of above-mentioned time periods in patients had survived ARS of various severity made possible to use these indices as diagnostic criteria for ARS. Based on results of statistical analysis it was found by empirical way the fluctuation ranges for granulocytes and thrombocytes time indices (table 4).

The figure 1 demonstrates an example how to determine ARS grade by the way of finding on granulocytes dynamics chart the points that correspond the days when these peripheral blood cells decreased to certain levels. Thus granulocytes decreased to 2×10^9 L^{-1} on 7th day, to 1×10^9 L^{-1} on 15th day, to 0.5×10^9 L^{-1} on 21st day, to the minimal level on 25th day. According to suggested criteria this hemogram corresponds to ARS grade 3.

Table 4. Haematological criteria of ARS grades.

Indices	ARS grade 1	ARS grade 2	ARS grade 3
Day of granulocytes decrease to:			
2×10^9 L^{-1}	19-33	8-18	6-7
1×10^9 L^{-1}	29-41	18-28	9-17
0.5×10^9 L^{-1}	32-42	23-31	12-22
minimal value	34-44	26-33	17-25
Day of thrombocytes decrease to:			
100×10^9 L^{-1}	20-34	13-19	6-12
50×10^9 L^{-1}	22-36	15-21	12-14
30×10^9 L^{-1}	26-35	18-25	13-17
minimal value	32-44	25-31	16-24
Minimal values:			
granulocytes ($\times10^9$ L^{-1})	300-1500	0-300	0
thrombocytes ($\times10^9$ L^{-1})	20-80	10-20	0-8

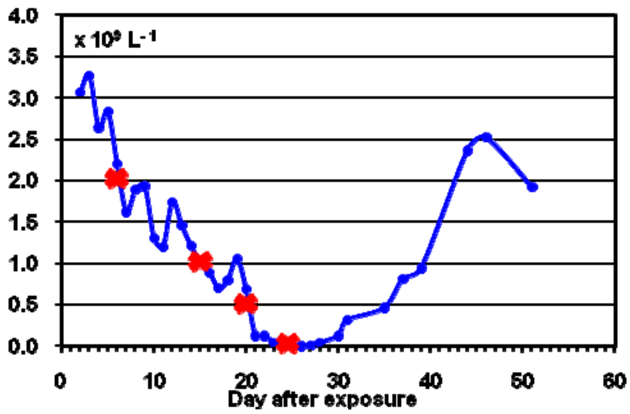

Fig. 1. Granulocytes dynamics in ARS grade 3 survivors (red crosses indicate the days when cells levels decrease to 2×10^9, 1×10^9, 0.5×10^9 L^{-1} and to minimal value)

ARS grade largely reflects a patient's overall clinical status than the depth of hemopoiesis injury. The value of granulo- and thrombocytopenia in ARS survivors of the same grade could vary both during cytopenia rising and the recovery of these cells count in peripheral blood. The deeper and continuously granulo- and thrombocytopenia was the stronger radiation injury of hemopoiesis, and vice versa. Therefore the quantitative characteristics of cytopenia, which is per se a deficit of peripheral blood cells caused by radiosensitive cell loss, may be used for estimation of hematopoiesis radiation injury.

For this purpose hemograms of 65 ARS survivors was analyzed to determine overall granulocytes and thrombocytes deficit in peripheral blood from the 10th till 55th days after exposure. The choice of such time interval are explained by several reasons: first, not all victims visit hospitals or out-patient departments after radiation exposure right away and are examined by physician (as it was during the Chernobyl accident); second, before the 10th day the great possibility of granulocytes abortive elevation exists in patients suffered from ARS grade 1 or 2; third, in the case of successful clinical outcome and hemopoiesis recovery the majority of patients leave a hospital not later the 60th day after exposure.

An index value, which characterizes granulocytes and thrombocytes deficit, was calculated on a graph that showed the dynamics of corresponding peripheral blood cells. This index is equal to square of figure (or several figures) that are restricted by the graph curve from sides and the bottom, and by straight line from the top (fig. 2). This line signifies the lower normal level of blood cells content, which is 2×10^9 L^{-1} for granulocytes and 150×10^9 L^{-1} for thrombocytes. The way of square calculation on graphs is described in statistical manuals (Bland 1995).

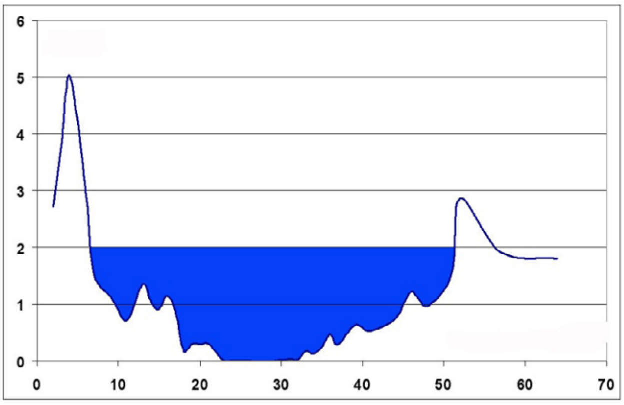

Fig. 2. Granulocytes dynamics in patient K., who survived ARS grade 2 (Y-axis – ×10^9 L^{-1}; X-axis – days after exposure; the blue coloured area indicates granulocytes deficit from the 10th till 55th day after exposure).

A day granulocytes deficit is equal to difference between 2×10^9 and actual cell count in a litre of peripheral blood. The sum of deficit value per every day signifies the overall cells deficit, which is easily measured in conventional units. On the fig. 2 the blue coloured square means granulocytes deficit in ARS grade 2 survivor. This deficit is 60.1 units. Thrombocytes deficit in the same patients is equal to 3987 units (fig. 3). To do the values of granulocytes and thrombocytes deficit more comparable this index was divided on 100. Therefore the summary deficit of both blood cells is the following: 60.1 + 39.87 = 99.97 units.

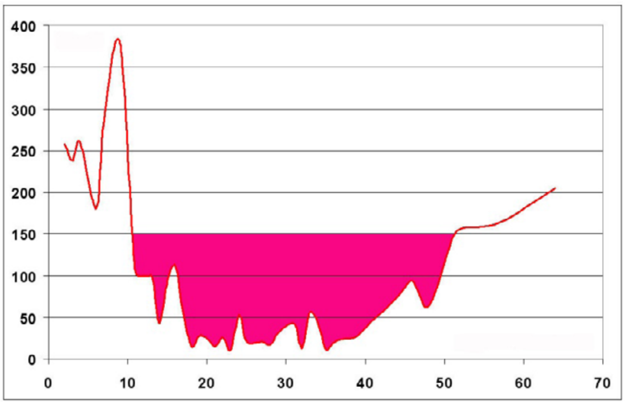

Fig. 3. Thrombocytes dynamics in patient K., who survived ARS grade 2 (Y-axis – ×10^9 L^{-1}; X-axis – days after exposure; the red coloured area indicates thrombocytes deficit from the 10th till 55th day after exposure).

Interrelation between summary deficit of granulocytes and thrombocytes, and whole-body radiation dose was analyzed and showed significant correlation measured by Spearman's, Pearson's and Kendall's methods (table 5).

Table 5. Correlation between summary deficit of granulocytes and thrombocytes and whole-body radiation dose (n=66).

Method	Correlation coefficient	p
Pearson	0.695	0
Kendall	0.608	0
Spearman	0.795	0

From 65 patients with radiation doses less than 1 Gy 41 persons had repeated short-term episodes of granulo- and thrombocytopenia from the 10^{th} till 55^{th} day after exposure. The summary deficit of peripheral blood granulocytes and thrombocytes was estimated in these patients. The values of the summary deficit for ARS survivors and patients with doses lower 1 Gy are showed in table 6. As far as they have a high significance of intergroup difference it was worked out the criteria for evaluation of hemopoiesis radiation injury that based on the mean values of deficit indices and confidence intervals. Thus the summary deficit of granulocytes and thrombocytes 0-20 units means that the hemopoiesis radiation damage could not cause ARS. The deficit value 21-40 units is attributed to ARS grade 1, 41-70 to ARS grade 2, 71-100 to ARS grade 3 and greater than 100 to ARS grade 4.

Table 6. Granulocytes and thrombocytes deficit in ARS survivors and patients with radiation doses lower 1 Gy.

Indices	ARS 0	ARS grade1	ARS grade 2	ARS grade3	ANOVA, F (p)
Number of patients	41	31	27	7	
Granulocytes deficit, units: mean ± SD min – max 95% confidence interval	 2.1±4.3 0-28.1 1.8-5.0	 17.0±12.3 0.2-46.2 12.5-21.6	 43.8±11.3 21.8-65.2 39.4-48.3	 42.7±8.4 31.0-57.3 34.9-50.5	111.3 (0)
Thrombocytes deficit, units: mean ± SD min – max 95% confidence interval	 2.2±3.4 0-21.9 1.4-3.0	 15.6±7.3 2.1-32.3 13.0-18.2	 28.2±8.3 7.7-44.8 24.9-31.5	 37.3±6.6 28.3-46.1 31.2-43.4	177.3 (0)
Summary deficit, units: mean ± SD min – max 95% confidence interval	 4.3±5.8 0-29.8 2.9-5.7	 32.1±17.1 4.5-70.1 26.0-38.3	 72.0±18.0 29.5-100.0 64.9-79.1	 80.0±11.0 66.6-91.4 69.8-90.1	230.0 (0)

Discussion

In the case of accidental irradiation of man it is not always possible to estimate radiation dose by the methods of physical dosimetry as well as calculate it by the way of experimental imitation of accidental situation. Therefore in this situation the preference belongs to biological dosimetry methods of dose estimation that based on quantitative and qualitative homeostasis changes. Some methods as electron paramagnetic resonance spectrometry of teeth enamel and evaluation of stable chromosomal aberration in peripheral blood lymphocytes are expensive enough, require special equipment and take much time for analysis. In view of such circumstances haematological indices are used wide in clinical practice for dose determination and estimation of radiation injury rate. Peripheral blood analysis requires no trouble and biological material, as capillary or venous blood, is gotten easy. The terms of analysis procedure are short enough so prompt result is expectable that is very important under condition when ARS threatens a patient's life.

All known hematologic indices, which are used for ARS grade estimation, can be divided on two groups: (1) criteria that have a diagnostic importance during the primary reaction or in latent period, and (2) indices that are used in the period of ARS manifestation.

The first group of criteria consists of indices that based on peripheral blood lymphocytes and leukocyte count during the first 10 days after irradiation. However the methods accuracy is under doubt. Thus according to L. Suvorova et al. (1991), ARS grade, which was determined by blood lymphocytes count between the 3rd and 6th day after the irradiation, had true prognosis in 50% of cases. Our results proved this data. The percentage of patients with false ARS grade, which had been diagnosed by leukocytes count, was 58%. When both lymphocytes and leukocytes count was used, not more severe ARS grade but slighter grade was mistakably diagnosed. This means that some patients, who will demonstrate the haematological signs of ARS (based on granulocytes and thrombocytes count) 20-30 days after irradiation, during first decade remains without the true diagnosis.

During the period of ARS clinical manifestation peripheral blood leukocytes and thrombocytes count is the main diagnostic criteria of this disease and the basis of it classification (Yarmonenko et al. 2004). According to this grading leukocytes fall to 3-1.5×10^9, 1.5-0.5×10^9, 0.5-0.1×10^9 L^{-1} and 0, and thrombocytes fall to 40-100×10^9, 20-40×10^9, 10-30×10^9 L^{-1} and 0 corresponds to 1, 2, 3 и 4 ARS grades. However these criteria are not early ones due to they are applicable only on a peak of ARS clinical symptoms.

Other authors (Anno et al. 1989) suggested to be guided by following levels of granulocytes and thrombocytes count: the doses 1-2 Gy cause granulocytes fall up to 2-4.5×10^9 and thrombocytes one to 80-180×10^9 L^{-1}, 2-3.5 Gy to 0.5×10^9 and 80×10^9 L^{-1}, respectively; the doses 3.5-5.5 Gy and higher result in the cells drop till zero.

Other method of dose estimation starting from 1 Gy are based on nomographic chart consisting of several curves of neutrophils and thrombocytes dynamics from the beginning of irradiation till the 60th day (Gus'kova et al. 1987, 1989). The advantage of this method is a dose evaluation on ARS early stage. However the dose that is determined by neutrophils count on the first days after irradiation not always coincides with ones estimated by thrombocytes count, and differs from the dose that is calculated by these cells minimal level in peripheral blood i.e. on late stage of the disease.

So today there are no accurate and sure methods for ARS grade and dose estimation due to all represented haematological criteria have two essential shortcomings: first, they are not designated for estimation of the dose that lower 1 Gy i.e. less than ARS threshold of development; second, they do not allow to accurately determine ARS grade on the early stage of the disease, only on the peak of clinical manifestation.

The diagnostic criteria of ARS that we suggested are based on time of granulocytes and thrombocytes decrease till the certain levels. These criteria have not a claim on high accuracy but are the still one method for ARS grade determination on the early stage of it development. Thus the day when granulocytes drop to 2×10^9 and thrombocytes to 100×10^9 L^{-1} lets to estimate ARS grade in the latent period of the disease. The day of granulocytes decrease to 1×10^9 or 0.5×10^9 and thrombocytes to 50×10^9 or 30×10^9 L^{-1} falls on manifestation period.

Estimating the state of hematopoiesis in patients, who did not suffered from ARS, we came to the conclusion that they had the radiation injury of hematopoietic tissue too. Although the threshold for ARS development is the dose 1 Gy of X-ray or gamma-irradiation, Y. Moskalev (1991) considered that the dose 0.65-0.7 Gy is the minimal portion of energy that is capable of damage blood stem cells. Other authors (Neumann et al. 1981; Uckun et al. 1989) thought that D_0 for a pluripotent stem cell (CFU-GEMM) varies from 0.54 till 0.94 Gy. Radiosensitivity of progenitors of different lineages (granulo-, mono-, and erythrocytopoiesis) is slightly less than the blood stem cell but nevertheless their threshold does not exceed 1 Gy.

G. Gruzdev (1988) showed that starting from the whole-body exposure of 0.5 Gy signs of bone marrow injury in man appear but haematological indices slightly deviate from normal values. This injury is not enough for clinical manifestation of radiation bone marrow syndrome. Our data confirms this statement. Thus from 65 patients with radiation doses lower 1 Gy 41 persons had granulocytes and thrombocytes dynamics that was not corresponded to typical ARS haematological picture but these patients demonstrated transient (single or repeatable) episodes of granulo- and thrombocytopenia. No one known haematological criterion is enabled to estimate neither the radiation dose nor severity of bone marrow injury for these people.

The methods of hematopoiesis radiation injury based on estimation of granulocytes and thrombocytes deficit was applied to a patient who had been irradiated in dose 0.3 Gy (fig. 4). Granulocytes deficit was 6.8 units and thrombocytes one 8.5 units. So the summary deficit was 15.3 units what is less than lower level of ARS grade 1. Therefore this method is applicable not only for ARS survivors but for persons who are undergone to irradiation in dose range lower 1 Gy. It is only the obligatory condition that blood analysis should be done regularly for not to miss cytopenia episodes.

Conclusions

It was worked out the two new types of haematological criteria for estimation of radiation injury rate in man. They are based on changes of peripheral blood granulocytes and thrombocytes count after the exposure. The first method enables to determine ARS grade (the rate of severity) in latent period and during manifestation of disease. Its main advantage is the enhancement of possibility to diagnose ARS grade on the early stage of the disease not excluding the other similar methods but together with them.

The second criterion is not applicable for early ARS diagnosis but gives the possibility to estimate the rate of hemopoiesis radiation injury both in ARS survivors and persons who were irradiated in doses lower 1 Gy and had transient granulo- and thrombocytopenia.

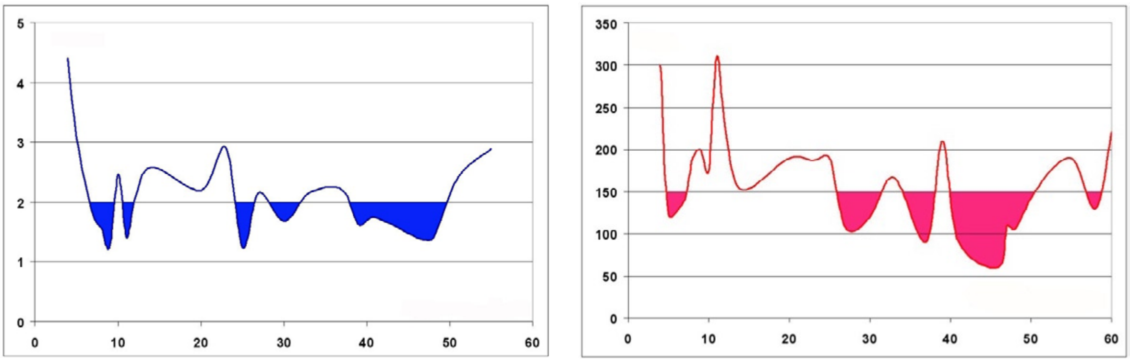

Fig. 4. Dynamics of granulocytes (left) and thrombocytes (right) in patient B. with whole-body radiation dose 0.3 Gy (conventional signs the same as on fig. 2 and 3)

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Calixarene nanoemulsion: a new treatment for uranium skin contamination

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Abstract

Despite protection means implemented in the nuclear industry, some internal contamination cases of workers by actinides such as uranium may occur either by inhalation, ingestion or penetration through wounds or intact skin. In case of cutaneous contamination, no specific emergency treatment exists although it may induce a high internal exposure of contaminated individuals. In this context, we have developed a new topical pharmaceutical form dedicated to emergency treatment of uranium skin contamination. In this order, we have decided to take advantage of the particular chelation properties of calixarene molecules developed in our laboratory for bioassay analyses purpose. The objective of the present work is to integrate calixarene molecules in a topical form for uranium skin decontamination. We have thus developed an oil in water (O/W) nanoemulsion containing these calixarene molecules. The physico-chemical properties of this nanoemulsion have been characterized and its efficiency for uranium extraction has been evaluated by *in vitro* and *ex vivo* experiments.

The characterization of the calixarene nanoemulsion has been performed by oily droplets size, zeta potential and pH measurements, as a function of calixarene concentration. The results suggest that calixarene molecules are present at the surface of the oily droplets. Thus, calixarene molecules are potentially available to trap uranyl ions present in an aqueous contaminated solution. This was confirmed by the *in vitro* evaluation of the calixarene nanoemulsion efficiency by ultrafiltration techniques. Indeed, more than 80% of uranium can be extracted by the calixarene nanoemulsion from an aqueous contaminated solution in optimized experimental conditions (pH, volumes and calixarene:uranium stoechiometry). Then, uranium percutaneous diffusion kinetics over 24 hours experiments, on intact or excoriated pig ear skin samples using Franz cells system, have shown that the application of the calixarene nanoemulsion immediately after the contamination quantitatively (98%) inhibits the uranium cutaneous transfer. Concurrent analysis of uranium distribution in skin samples by SIMS technique showed no significant accumulation of uranium or calixarene-uranium complex through the different layers of the skin, except within the *stratum corneum* in case of intact and non treated skin. A delayed application of the calixarène nanoemulsion (5 min, 15 min or 30 min) after the contamination on excoriated skin

samples is still efficient since uranium transfer is reduced by up to 75%. Thus, for optimal efficiency, the treatments should be applied the fastest after the contamination.

In conclusion, this study has successfully demonstrated the efficiency of the calixarene nanoemulsion constituting a promising system to treat uranium contaminated skin.

Introduction

Internal contamination of power plants workers by actinides may occur by inhalation, ingestion or penetration through wounds or intact skin. Although the most current contamination pathway is inhalation (Gerber and Thomas 1992), the contamination through intact or wounded skins is still a big concern, since it may induce a high internal exposure of contaminated individuals after translocation of the radionuclides from the contamination site into the body (De Rey et al. 1983, Lopez et al. 2000, Petitot et al. 2004, Petitot, Frelon et al. 2007, Petitot, Gautier et al. 2007). The current medical care in case of skin contamination only occurs after transfer of the victim to the medical unit of the nuclear site and it only consists in local decontamination of the wound by rinsing with soaped water or a 25% calcic salt of diethylene triamine pentaacetic acid (Ca-DTPA) solution. This procedure is often followed by a decorporation treatment with intravenous injection of 1 g of Ca-DTPA in 4 mL (ASN 2008) to reduce the risks of tissue damage and induction of cancer. A surgical excision of the contaminated tissues can be needed in order to remove residual radionuclides and prevent their penetration into the body (Gerber and Thomas 1992, Bailey et al. 2003). Nevertheless, Ca-DTPA exhibiting a lack of selectivity and affinity for uranium in biological medium, these procedures are ineffective in case of uranium contamination that may induce strong kidney toxicity (Lopez et al. 2000, Métivier 2001). Some chelating agents more specific to uranium have been developed in order to decorporate uranium from the body (Durbin 2008). Except the biphosphonate molecule series (Houpert et al. 2004, Yang, et al. 2005, Xu et al. 2008), none of those chelating agents have been used in pharmaceutical forms dedicated to skin decontamination. In addition, the specific uranium chelation by the biphosphonate molecules within the proposed formulations has not been demonstrated.

In this context, we have developed a new topical pharmaceutical form dedicated to emergency treatment of uranium skin contamination. We have taken advantage of the particular chelation properties of a calixarene molecule (Fig. 1) developed in our laboratory for radiotoxicological analyses purpose. The selectivity, affinity and extraction efficiency of uranium present in trace in biological media by this calixarene have been successfully shown in previous works (Boulet et al. 2006). The objective of the present study is to integrate these calixarene molecules in a topical form for skin decontamination. We have thus developed an oil in water (O/W) nanoemulsion displaying calixarene molecules at the interface between the oily and external aqueous phase. The physicochemical properties of this nanoemulsion have been characterized and its efficiency for uranium extraction has been evaluated *in vitro* by using an adapted ultrafiltration method and *ex vivo* with first experiments on intact and excoriated pig ear skin explants in Franz diffusion cells.

Material and methods

Materials

Calixarene molecule (Fig. 1) was synthesized as described in the patent (Duval et al. 2006). Other compounds used for preparing calixarene nanoemulsions were paraffin oil (d = 0.86), (VWR, Fontenay-sous-Bois, France), non ionic surfactants sorbitan monooleate (Span® 80) and polyoxyethylene glycol sorbitan monooleate (Tween® 80), purchased from Sigma-Aldrich (Saint-Quentin-Fallavier, France) and water obtained from a Milli-Q® Synergy 185 water purification system (Millipore, Saint-Quentin-en-Yvelines, France).

Uranium-contaminated solutions were prepared by diluting a standard depleted uranium solution (1,000 $mg.L^{-1}$ in 2% HNO_3, SPEX Certiprep, Horiba Jobin Yvon, Longjumeaux, France). The pH of the contaminated solutions was adjusted to 5 with 0.01 M acetate buffer. A bismuth (^{209}Bi) stock standard solution at 100 $\mu g.L^{-1}$ (from a 10 $mg.L^{-1}$ single element standard, SPEX Certiprep, Horiba Jobin Yvon, Longjumeaux, France) and a multielemental standard solution containing depleted uranium at 1 $\mu g.L^{-1}$ (from a 10 $mg.L^{-1}$ tuning solution SPEX Certiprep, Horiba Jobin Yvon, Longjumeaux, France) were prepared in 2% HNO_3 (from a 67% HNO_3 stock solution, Normatom, VWR, Fontenay-sous-Bois, France) for Inductively Coupled Mass Spectrometry (ICP-MS) measurements. All reagents were used as received without any further purification. Pig ears used in Franz cell diffusion experiments were purchased from Guy Harang abattoir (France) with the authorization of the French Departmental Directorate of Veterinary Services of Hauts-de-Seine. Pig ears were stored at -20°C until use.

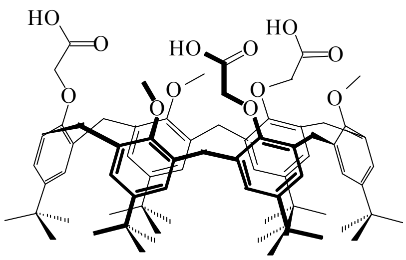

Fig.1. Structure of 1,3,5-OCH_3-2,4,6-OCH_2COOH-*p-tert*butylcalix[6]arene.

Methods

Preparation and characterization of calixarene nanoémulsions

Oil in water calixarene nanoemulsions were prepared by the emulsion inversion point method as previously described (Spagnul et al. 2010). The amounts of surfactants, oil and water were respectively 5%, 20% and 75% (w/w). The dispersed oil droplet size and zeta potential were both measured with a Nanosizer/Zetasizer® ZS apparatus (Malvern Instruments, Worcestershire, U.K.), and pH measurements were carried out with a MeterLab/pH M240 pH-meter (Radiometer Copenhagen, Denmark). These parameters were determined as a function of the calixarene concentration in the nanoemulsion.

In vitro calixarene nanoemulsion efficiency study

Equal volumes of uranium-contaminated solution (40 µg.L^{-1}, pH 5) and calixarene loaded nanoemulsion (2 mg.g^{-1}) were mixed and the mixture was kept under static condition during definite times. Aliquots of the mixture were put into Microcon® centrifugal filter devices (3,000 Da, Millipore, Saint-Quentin-en-Yvelines, France) and centrifuged at 6,000 rpm for 30 minutes at 20°C. The uranium extraction by the calixarene nanoemulsion was determined by quantifying free uranium present in the aqueous filtrate phase recovered by ultrafiltration-ultracentrifugation of the mixture using Inductively Coupled Mass Spectrometry (ICP-MS) technique (Spagnul et al. 2010).

Ex vivo calixarene nanoemulsion efficiency study on pig ear skin explants

The diffusion kinetics of uranium through intact and wounded skins was studied after application or no application of the calixarene nanoemulsion treatment in Franz diffusion cell assays using MicroettePlusTM apparatus (Hanson Research Corp., Chatsworth, California, USA). Full thickness skin pieces were removed from pig ears and wound was generated by removing the *stratum corneum* of the skin by tape stripping using 60 standard D-Squame® discs (Monaderme®, Monaco) pressed onto the skin using a D-Squame® applicator that provides a constant 150 g cm^{-2} pressure (Monaderme®, Monaco). Each skin sample was put on receptor compartment of the diffusion cell, its dermal face in contact with a 0.025 M carbonate buffer receptor medium (Sigma-Aldrich Saint-Quentin-Fallavier, France), covered with the donor compartment and the whole device was fixed with a clamp. The contamination was made by depositing 600 µL of a 10 mg.L^{-1} uranyl nitrate solution at pH 5 on the skin sample in the donor compartment of the Franz diffusion cell. Skin decontamination was evaluated by depositing 600 µL of 4 mg.g^{-1} calixarene nanoemulsion in the donor compartment immediately after the skin contamination. The dosage form was also applied 5, 15 and 30 minutes after the skin contamination step on excoriated contaminated skin samples. 24 hours later, uranium diffusion was then evaluated by uranium quantification in the receptor fluid. The Receptor medium was kept under stirring and at the constant temperature of (33.5 ± 0.1)°C during the experiment. All experiments were conducted under occlusive conditions.

ICP-MS uranium analysis

Quantification of ^{238}U content in the aqueous samples was made by ICP-MS measurements using optimized protocols originally designed for human urine samples: aliquots were properly diluted in 2% HNO_3 and ^{209}Bi was added as internal standard at 1 µg.L^{-1} (Baglan et al. 1999, Bouvier-Capely et al. 2003, Bouvier-Capely et al. 2004). As diluted solutions were prepared by weighing, the related uncertainties can be neglected compared to the statistical ones. The combination of the statistical errors was made using the conventional law of uncertainty propagation. The ICP-MS signal was optimized at mass 238 before each measurement series.

Localisation of uranium in skin biopsies by SIMS microscopy

At the end of the 24 hours Franz cell diffusion experiments, skin biopsies were thoroughly rinsed with water and fixed in a solution containing 6% glutaraldehyde in a sodium cacodylate buffer at 4°C during one day. Skin samples were then dehydrated in various propylene oxide and ethanol baths, permeated with a propylene oxide/Epon mixture and then embedded in pure EPON-type resin. Skin sections (0.9 µm) of embedded samples were cut and laid on silicium or gold holders for SIMS (Secondary Ion Mass Spectrometry) analysis. The SIMS microscopy, which allows the elemental and isotopic analysis of a solid surface, is based on the sputtering of a few atomic layers from the surface of a sample by O^{2+} primary ions beam bombardment. The ejected secondary ions are collected and analysed by mass spectrometry. Then, compositional images of the surface are formed from the secondary ions spectra. A more detailed description of the physical phenomenon is provided in the literature (Castaing and Slodzian, 1962; Tessier et al., 2009). For each skin area analysed, $^{40}Ca^+$ or $^{23}Na^+$ images gave the histological structure of the skin and $^{238}U^+$ images showed uranium localisation within the cutaneous structures. The SIMS analyses were performed with a CAMECA IMS 4F-E7 instrument (Gennevilliers, France).

Results and discussion

Characterization of the calixarene nanoemulsion

A previous study on the physicochemical characterization of the calixarene nanoemulsion showed that the oil droplet size is nearly 175 nm and decreases significantly ($p < 0.05$) to 150 nm from a calixarene concentration in the nanoemulsion of 2 mg g^{-1} (Spagnul et al. 2010). Tension measurements at the oil/water interface showed that calixarene molecules have surfactant properties which can explain the droplet size diminution after the calixarene molecules addition in the nanoemulsion.

The zeta potential measurement of the dispersed oily droplets showed that increasing calixarene concentration in the nanoemulsion leads to a zeta potential decrease from -13 mV to -50 mV until a 4 mg g^{-1} calixarene concentration at which the zeta potential value did not change significantly from its -50 mV minimal value (Spagnul et al. 2010). This zeta potential decrease due to the emergence of negative charges at the surface of the oily droplets can be explained by the presence of calixarene molecules at the oil droplets surface. Indeed, the carboxylic functions of calixarene molecules may be partially deprotonated at the nanoemulsion aqueous phase pH which is about of 4.5 (Boulet 2006).

These results tend to demonstrate that calixarene molecules are at the surface of the oil droplets under ionised form. Calixarene molecules are thus potentially available to extract uranyl ions from an aqueous uranium contaminated solution.

In vitro evaluation of uranium extraction by the calixarene nanoemulsion

The decontamination potential of the calixarene nanoemulsion was first evaluated by keeping under static contact a mixture of equal volumes of 2 mg.g^{-1} calixarene loaded nanoemulsion with a 40 µg.L^{-1} nitrate uranyl solution at pH 5, ensuring a 10,000-fold calixarene molar excess. The extraction ability of the calixarene nanoemulsion was compared to that of a solution of calixarene in paraffin oil in the same conditions of calixarene excess (molar ratio calixarene/uranium = 10,000). The results presented in Table 1 indicate that the calixarene nanoemulsion is able to extract $(83.4 \pm 0.6)\%$ of uranium of the contamination solution after only 5 minutes of contact (total contact time of 35 minutes after ultrafiltration) and the optimum of uranium extraction is quickly achieved. The presence of calixarene in the nanoemulsion is itself largely responsible for the extraction ability as the nanoemulsion without calixarene extracts only about 20% of uranium at equilibrium. The use of a more simple galenic form consisting in a solution of calixarene in paraffin oil only allows the extraction of ~7% of uranium from the contaminated solution. This last result shows that the emulsified system ensures a high uranium extraction yield due to the larger oil/water interfacial contact. The calixarene nanoemulsion form seems thus to be well suited for the decontamination of an aqueous solution contaminated by uranium.

Table 1. Uranium extraction (%) by paraffin oil and nanoemulsion in presence or not of calixarene, as a function of contact time between uranium solution and nanoemulsion or paraffin oil, in static conditions.

	U extraction (%)		
Contact time	Calixarene loaded Nanoemulsion	Nanoemulsion without calixarene	Calixarene in paraffin oil
5 min	83.4 ± 0.6	23.2 ± 0.8	-
1 hour	87.4 ± 0.7	16.7 ± 4.9	7.0 ± 0.9
4 hours	80.0 ± 3.9	25.3 ± 8.8	8.3 ± 0.3

Ex vivo evaluation of uranium extraction by the calixarene nanoemulsion

From the good uranium extraction yields obtained by the calixarene nanoemulsion in the previous *in vitro* tests, the calixarene nanoemulsion extraction efficiency for skin decontamination was evaluated. A 4 $mg.g^{-1}$ calixarene loaded nanoemulsion was deposited in the donor compartment of the Franz diffusion cells on pig ear skin explants immediately after a contamination with a 10 $mg.L^{-1}$ uranyl nitrate solution. A concentration of 4 $mg.g^{-1}$ of calixarene in the nanoemulsion was chosen in order to allow an excess of calixarene molecules compared to uranium ions (molar ratio calixarene/uranium = 80) and ensure a good uranium extraction. The uranium diffusion through the skin was followed during 24 hours under occlusive conditions and as a function of the application or no application of the calixarene nanoemulsion immediately after the contamination. The results obtained on intact skin 24 hours after the contamination are presented in Fig. 2-A. Intact skin seems to be a good barrier against uranium penetration as only 0.026 ± 0.006 % of the deposited uranium diffused through the skin 24 hours after the contamination. The application of the calixarene nanoemulsion immediately after the contamination allowed an uranium diffusion decrease of (98.4 ± 5.3)%. A reduction of uranium diffusion is also observed after application of nanoemulsion without calixarene, which can be partly explained by the dilution effect of the contaminated solution with the nanoemulsion aqueous phase and by the electrostatic interactions between uranium ions and the surfactants.

At the issue of the 24 hours Franz cell diffusion experiments, intact skin biopsies were analysed by SIMS microscopy (Fig. 2-B). The images and mass spectra showed that after the application of the calixarene nanoemulsion on uranium contaminated skin explants, uranium was no more significantly detected in skin structures whereas the radionuclide is significantly retained in the *stratum corneum* layer of the epidermis without treatment (Fig. 2-B) or after application of the nanoemulsion without calixarene (not shown) .

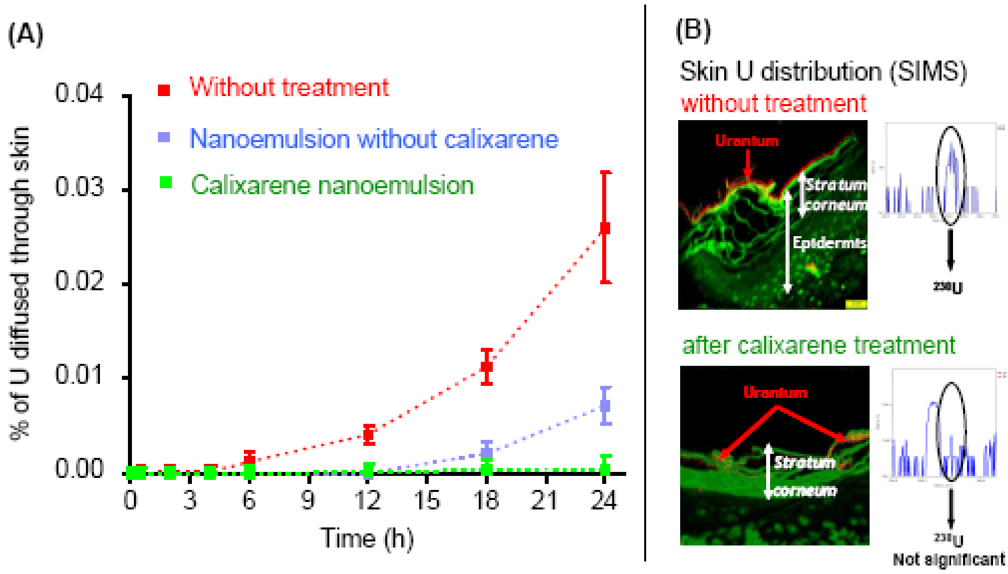

Fig.2. (A) Uranium percutaneous diffusion kinetics through intact skin in absence of treatment (■) or after application of nanoemulsion without (■) or with 4mg.g^{-1} calixarene (■). The treatment was applied immediately after the contamination step and the percentage of initially deposited uranium that diffused through intact skin biopsies was recorded during 24 hours. Each point corresponds to the mean of 10 assays ± standard deviation.
(B) Ionic images and corresponding mass spectra of intact skin biopsies without treatment (up) and immediately treated after the contamination step by calixarene nanoemulsion (down). Biopsy was analysed by SIMS 24 hours after the beginning of Franz cell experiments. In the ionic image, $^{238}U^+$ image (red) was superposed to $^{23}Na^+$ (green) image (200 µm x 200 µm image field).

In the case of excoriated contaminated skin (Fig. 3-A), (44.3 ± 6.4)% of the deposited uranium diffused through the skin, which is significantly higher than what was observed for intact skin. The prompt application of the calixarene nanoemulsion treatment conducted to an uranium diffusion decrease of (97.7 ± 0.8)% after 24 hours. The calixarene nanoemulsion is thus still efficient in case of contaminated wounds.

After 24 hours of Franz cell diffusion experiments, SIMS images and mass spectra showed that no significant uranium accumulation was detected in the excoriated skin structures after the application of nanoemulsion with calixarene on contaminated excoriated skins (Fig. 3-B). Uranium traces were found to be retained in the areas where a few residual *stratum corneum* layers remained after the tape stripping step.

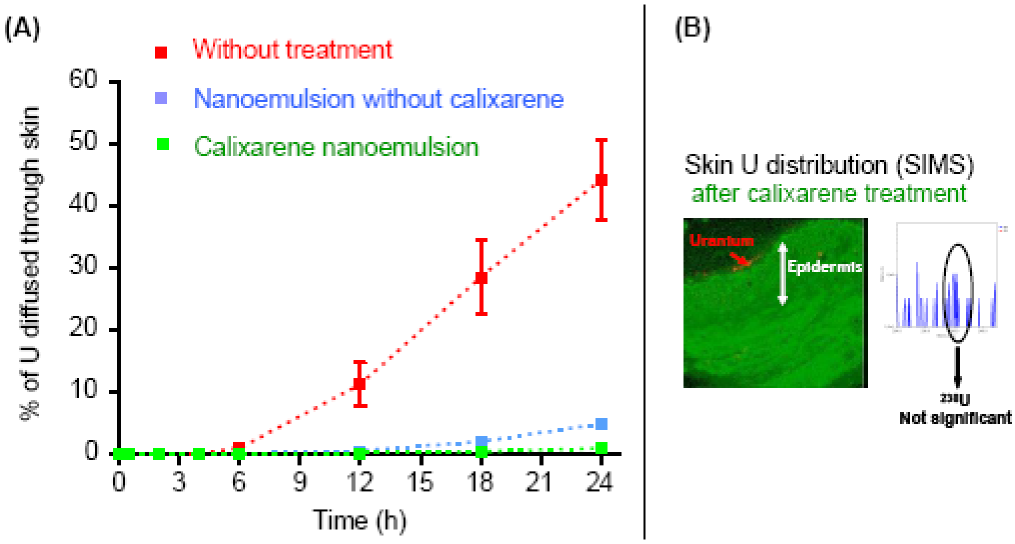

Fig.3. (A) Uranium percutaneous diffusion kinetics through excoriated skin in absence of treatment (■) or after application of nanoemulsion without (■) or with 4mg.g^{-1} calixarene (■). The treatment was applied immediately after the contamination step and the percentage of initially deposited uranium that diffused through intact skin biopsies was recorded during 24 hours. Each point corresponds to the mean of 10 assays ± standard deviation.
(B) SIMS mass spectrum and ionic image of excoriated contaminated skin biopsy treated immediately after the contamination step by calixarène nanoemulsion. Biopsy was analysed by SIMS 24 hours after the beginning of Franz cell experiments. In the ionic image, $^{238}U^{+}$ image (red) was superposed to $^{23}Na^{+}$ (green) image (200 µm x 200 µm image field).

The effect of time elapsed between the skin contamination by uranium and the application of the treatment on the uranium diffusion kinetics was studied on excoriated contaminated skin by delaying the application of the calixarene nanoemulsion by 5, 15 and 30 minutes. As illustrated in figure 4, uranium diffusion kinetics was roughly the same when the calixarene nanoemulsion was applied 5, 15 or 30 minutes after the contamination step. In these three cases, the calixarene nanoemulsion allowed a 3.5-fold uranium diffusion percentage decrease after 24 hours as compared to the absence of treatment: indeed, 11.55±2.79% in average of deposited uranium diffused after 24h (Fig. 4). Thus calixarene nanoemulsion is less efficient in case of delayed application but still allows a very significant uranium diffusion decrease as compared to the absence of treatment application. This reduction of efficiency being approximately the same whatever the delay of application (5 to 30 minutes), we assume that a part of the deposited uranium fastly penetrates in the upper excoriated skin structures, saturates them and then diffused passively through the skin to later reach the receptor medium. Experiments are in progress to better understand this phenomenon. From these results, it appears that it is preferable to treat the cutaneous contamination as soon as possible to ensure an optimal decontamination efficiency of the calixarene nanoemulsion.

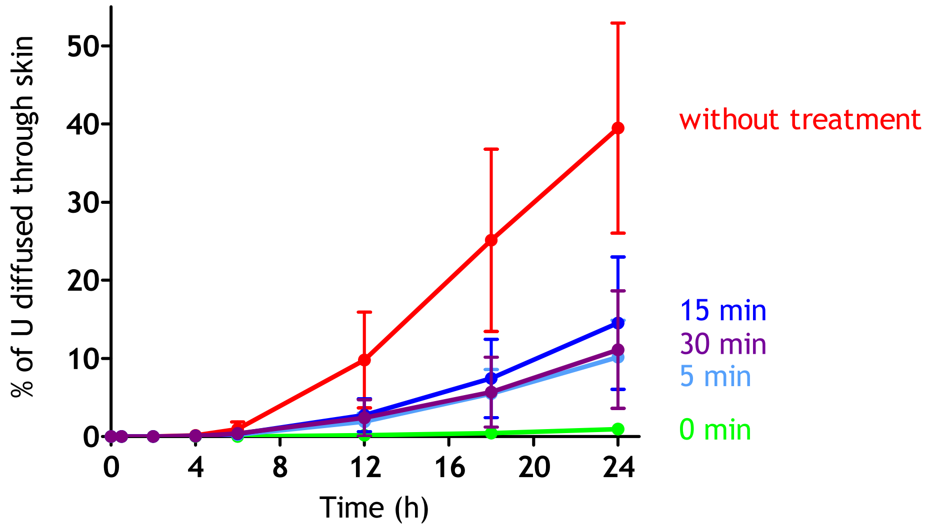

Fig. 4. Uranium percutaneous diffusion kinetics through excoriated skin in absence of treatment (■) and after immediate (■) application or 5 minutes (■), 15 minutes (■) and 30 minutes (■) delayed application of calixarene nanoemulsion. The percentage of initially deposited uranium that diffused through excoriated skin biopsies was recorded during 24 hours. Each point corresponds to the mean of 10 assays ± standard deviation.

Conclusions

In summary, our study demonstrates that the calixarene nanoemulsion allows decreasing almost quantitatively uranium percutaneous diffusion by trapping and retaining uranyl ions at the surface of the skin. This patented formulation (Bouvier-Capely et al., 2008) thus constitutes a promising therapeutic approach for the local treatment of intact and wounded skin contaminated by transferable uranium compounds. For optimal efficiency, the treatment should be applied as fast as possible after the contamination. As the calixarene molecules used in this study also efficiently chelate other actinides, we planned to evaluate the calixarene nanoemulsion efficiency on skin contaminated by other actinides such as plutonium and americium and to perform *in vivo* studies.

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Premature chromosome condensation (PCC) assay for dose assessment in large radiological accidents

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Abstract

A dose calibration curve for a practical PCC-ring assay was established and applied in a simulated, mass casualty accident. The PCC assay was validated against the conventional dicentric assay. A linear relationship was established for PCC rings after Co-60 irradiation with doses up to 20 Gy. In the simulated accident experiment, 62 blood samples were analysed with both the PCC ring and the conventional dicentric assay, applying a triage approach allowing crude dose estimate through analysis of a relatively small number of cells. Samples received various uniform and non-uniform (10-40% partial body) irradiations up to doses of 13 Gy. The results indicated that both assays yielded good dose estimates for the whole body exposure scenario, although in the lower dose range (0-5 Gy) dicentric scoring resulted in more accurate whole-body estimates, whereas PCC rings were better in the high-dose range (above 8 Gy). Both assays succeeded poorly in identifying partial body exposures, most likely due to the low numbers of cells scored in the triage mode. In conclusion, the study confirmed that the PCC ring assay is suitable for use as a biodosimeter following whole-body exposure to high doses of radiation. However, there are limitations for its use in the triage of people exposed to high, partial body doses.

Introduction

Several markers of exposure are available for biological dosimetry purposes. The dicentric assay is generally considered to be the gold standard of biodosimetric methods (Blakeley et al. 2005). The assay can be applied for dose estimation after whole- and partial-body exposure. The disadvantages of the dicentric assay are the time consuming and technically demanding analyses and limitations in assessing doses in excess of about 5–8 Gy due to impaired cell proliferation (Sasaki and Norman, 1966).

Impaired cell proliferation after high-dose exposures can be circumvented by the method of premature chromosome condensation (PCC). The original PCC technique is based on fusion of interphase lymphocytes with mitotic cells (Waldren and Johnson, 1974; Hittelman and Rao, 1974; Pantelias and Maillie 1983). Later it was discovered that PCC could be induced in G2-phase lymphocytes with the help of Calyculin A or okadaic acid— both phosphatase inhibitors. Kanda et al. (1999) suggested scoring PCC rings for the purpose of biological dosimetry after high doses of radiation. They obtained a good dose–response curve for PCC rings in the dose range up to 20 Gy of X rays. Similar results were reported by Lamadrid et al. (2007).

In large radiological accidents, fast and reliable dose assessment may be crucial for early decision of medical care and methods providing reliable triage method for assessing exposed and nonexposed people are needed. To increase the throughput of the dicentric assay, Lloyd et al. (2000) suggested a triage approach in which a reduced number of cells would be scored. For a whole-body exposure up to a dose of 8 Gy, they obtained a satisfactory result from scoring 20 metaphases per subject. After partial-body exposure with irradiated fractions in the range of 50–97%, a satisfactory result was obtained when 50 cells per subject were analyzed. In a mass casualty accident, people may be exposed to doses in excess of 5–8 Gy. One possible scenario is that of a very strong γ-ray source (for example, a source used for industrial radiography) hidden below a seat of a public transport vehicle. For highly exposed individuals, the dicentric assay may fail due to impaired division of lymphocytes. Thus there is a need to validate the PPC method for such exposure scenarios to cover the range of potential exposures.

In the present study, we have established a PCC ring calibration curve for γ-radiation validated it for use in a mass casualty accident. The validation was conducted by comparison to the dicentric method in a triage exercise that simulated an accident scenario involving a large number of casualties. Each case obtained an individual uniform or non-uniform irradiation.

Material and methods

PCC and dicentric dose response curves

Whole blood was irradiated with the following doses: 0, 1, 2.5, 5, 10, 15, and 20 Gy. All irradiations were performed with a ^{60}Co gamma source at 0.3 Gy/ min in a water bath with a fixed temperature of 37 ºC. The blood samples were further incubated for 2 hours at 37°C to allow for repair of DNA damage before lymphocytes were isolated. The dicentric dose response curves used in this study were developed previously in STUK (Lindholm et al. 1998) and FOI + NRPA (Stricklin et al. 2005).

Simulation of mass casualty accident scenario

A scenario of malevolent use of radiation was simulated by in vitro irradiation of blood samples. The simulation of doses was planned by a third party. The scenario was based on the assumption that a very strong gamma source had been hidden in a public transport vehicle, giving rise to a dose rate of 20 Gy/h close to the source and 2 Gy/h at 5 m distance. In this scenario, 62 persons with clinical symptoms and potential exposure were evaluated. Specifically, 62 blood samples were exposed to different doses at STUK, using the same irradiation conditions as described above. For the purpose of the simulated scenario, blood samples were collected from 14 volunteer donors in compliance with institutional guidelines for research on human samples.

Forty four samples were exposed in a homogeneous manner (simulation of whole body exposure) to the following doses (numbers of exposed cases are given in parentheses): 0.0 (7), 0.4 (3), 0.8 (2), 1.2 (3), 1.6 (2), 2.1 (3), 2.7 (2), 3.2 (3), 3.6 (2), 4.4 (2), 4.9 (3), 5.7 (3), 6.2 (2), 7.0 (2), 8.0 (1), 9.0 (1), 10.0 (2), 13.0 (1). 18 samples were irradiated in a non homogeneous manner (simulation of partial body exposure). The latter involved mixing of irradiated and non-irradiated blood in the following proportions (given doses in brackets): 10% (8, 9 and 13 Gy); 15% (7, 8, 8 and 10 Gy); 20% (9, 9, 9, 10, 13 and 13 Gy); 25% (7 and 7 Gy); 30% (10 and 13 Gy) and 40% (7 Gy) of irradiated blood.

Lymphocyte culture and analysis

Lymphocytes were separated and cultured at a density of $0.5x10^6$/ ml for successful cell growth. In the PCC protocol okadaic acid (final concentration 500 nM) was added for the last 1 hour in a 48 hour cell culture. In the dicentric assay colcemid (final concentration 0.2 µg/ml) was added for the last 2.5 h of the 48 h cultures. Cells were harvested according to standard cytogenetic procedures and stained with Giemsa. Analysis was performed in a blind manner on coded slides.

For all PCC analyses, fast scoring approach, ie. inclusion of intact cells without counting centromeres, was applied. PCC cells in any cell cycle phase were scored and rings were analysed irrespective of the presence of a centromere, in accordance with Kanda et al. (1999). Large PCC rings with either a clear open space or large spherical rings with a diameter greater than the width of chromosomes were scored.

For the simulated accident scenario, the analysis of dicentric samples was performed according to the routine in the respective laboratory. Scoring criteria for dicentrics were according to IAEA (2001), ie. metaphases with 46 centromeres were approved, and thirty dicentrics or 50 metaphases were scored according to Lloyd et al. (2000). From PCC preparations, 50 rings or 300 PCC cells per sample were analysed.

Statistical analyses, dose assessment and dose categories

The PCC ring data was fitted by the maximum likelihood method (Papworth 1975). The distribution of PCC rings scored for each dose was analysed with the help of the software NETA, which identifies distributions as Poisson, Neyman-type A or none of the two (Morand et al. 2008). Appropriate confidence limits (Neymanian or Poissonian) for the PCC yields were also calculated using NETA.

The calibration curve established in the current work was used to estimate the doses based on PCC rings. For dicentrics, each laboratory used its own calibration curve (Lindholm et al. 1998; Stricklin et al. 2005) that covered dose range 0-5 Gy. The fit of the estimated doses to the given dose was evaluated by the chi-square goodness of fit test. For samples irradiated in a partial body exposure scenario the dose to the exposed fraction of cells was calculated by the Dolphin method (IAEA 2001). No account was taken of the fact that, following homogeneous exposure of blood, PCC rings tend to show overdispersed distributions.

Results

PCC ring frequencies were evaluated at both STUK and FOI for seven dose points. Full data of this study have been published earlier (Lindholm et al. 2010). Data from the two laboratories were not statistically different and were therefore pooled. The distributions of PCC rings were generally overdispersed, and were best fitted to Neyman type A distribution after doses of 5, 10 and 15 Gy. Poisson distributions were observed after 2.5 and 20 Gy, while the distribution after 1 Gy was neither Poisson nor Neyman. The PCC data were fitted to a linear equation. The fit coefficients are shown in table 1, along with the linear-quadratic fit coefficients for dicentrics. Figure 1 demonstrates the linear fit of PCC rings to dose. The 1 Gy dose point, which was distinctly lower (marked with a square symbol in the figure) was excluded from the fit. The dose response of PCC rings showed a slight downward curvature at 20 Gy. The linear fit was chosen for practical reasons to represent the PCC ring dose relationship to be used for dose assessment of large scale accidents. However, it does lead to underestimation of dose in the dose range above 15 Gy.

Table 1. Dose coefficients used for dose estimations.

	Control (± S.E.)	Linear coefficient (± S.E.)	Quadratic coefficient (± S.E.)
PCC rings	0.0018 ± 0.0001	0.049 ± 0.001	
Dicentrics, STUK	0.0005 ± 0.0002	0.013 ± 0.0043	0.054 ± 0.003
Dicentrics, FOI and NRPA	0.0016 ± 0.001	0.014 ± 0.006	0.065 ± 0.003

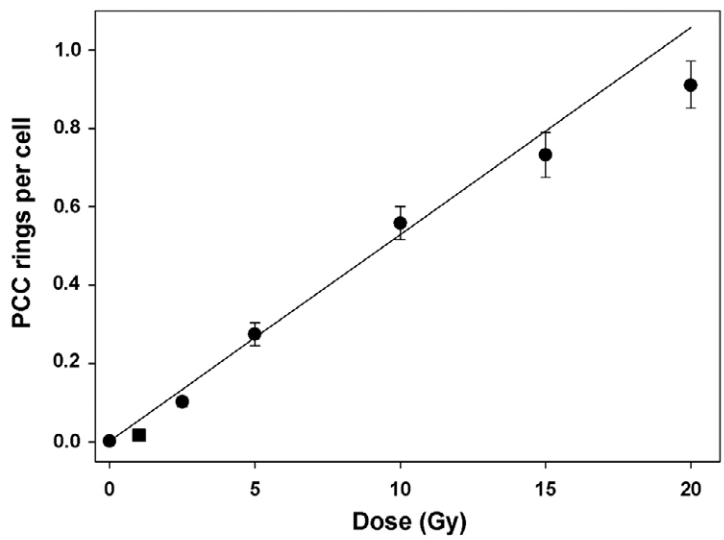

Fig. 1. PCC ring data fitted to a linear model. Square symbol displays the data point at 1 Gy which was omitted in the fitting procedure. Error bars represent 95% confidence limits.

For the accident simulation exercise, estimated doses for samples of uniform exposures and calculated from dicentric and PCC ring frequencies are plotted against the given dose in Figure 2. The chi square values of the goodness of fit are 13.86 for dicentrics and 20.00 for PCC rings (with 41 degrees of freedom), indicating a better fit of the dicentric data. Triage analysis of dicentrics resulted in a better dose estimate in the dose range below 6 Gy which is explained by the dose range of dicentric curves that reaches up to only 5 Gy and dose estimates above this are extrapolations. The PCC ring assay resulted in a slight underestimation of doses in the range 2-6 Gy.

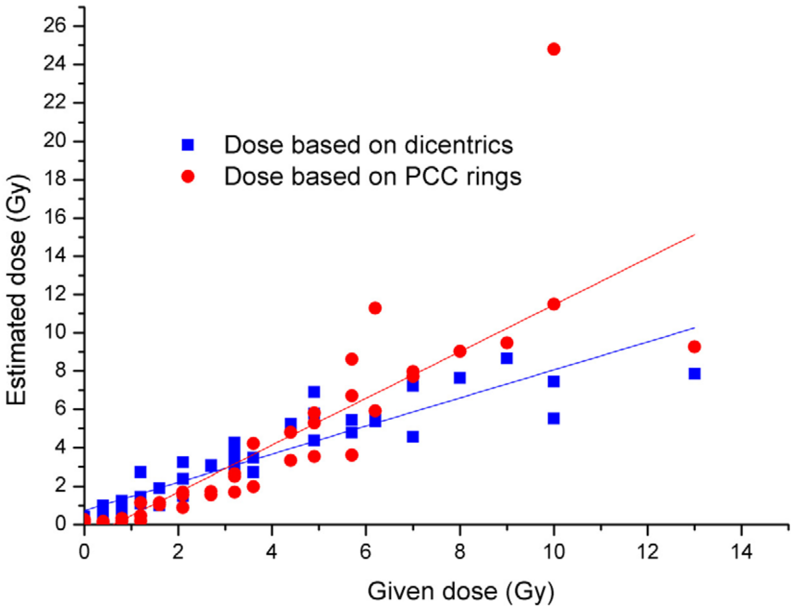

Figure 2. Dose estimates based on dicentric and PCC ring frequencies plotted against the given dose for 46 samples exposed in a homogeneous manner. Solid lines represent linear fits (Rdicentrics = 0.73 ; RPCC rings = 1.26) to the data points. Due to overlapping, all individual plots can not be differentiated.

Figure 3 demonstrates how well the triage scoring of dicentrics and PCC rings allowed to identify exposed and non-exposed cases. Of non-exposed samples, one case was missed with the dicentric assay and two cases were not correctly recognized by PCC rings. In the lower dose categories, the dicentric assay was more efficient in placing the cases into the correct category than the PCC assay. Above the dose of 6 Gy, PCC ring scoring was superior to dicentrics in identifying the correct category. Taking all dose categories into account, data from the dicentric assay were slightly albeit not significantly better than from the PCC assay in placing doses into the right category (75% vs. 59%). The same was true with respect to the 95% confidence limits: 75% for dicentrics vs. 64% for PCC rings.

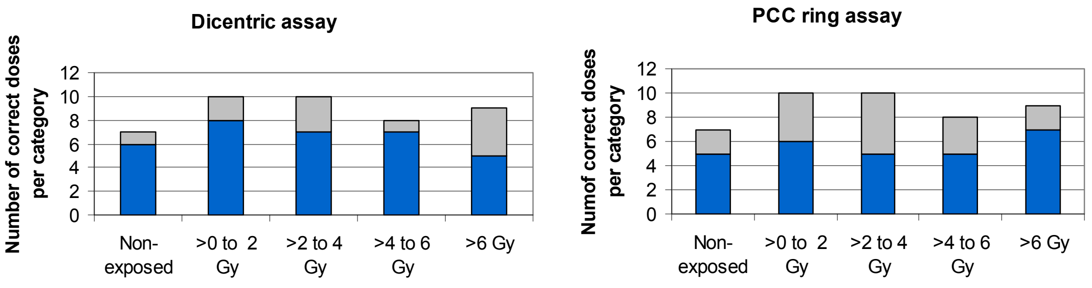

Figure 3. Homogeneously exposed cases correctly estimated by dicentrics and PCC rings. Numbers of cases (in dark blue) where the 95% confidence limits of the dose estimate encompassed the given dose.

Partial body exposure was identified by a significant overdispersion of dicentrics and PCC rings. Analysis of dicentrics allowed identification of 61% of cases, while analysis of PCC rings accommodated 56% (data not shown). There was no trend of an improved identification with increasing size of exposed blood fraction. Both assays were inefficient in estimating the given partial body doses. Also, no correlation between the doses estimated by the two assays was observed.

Discussion

The evaluation of the PCC ring assay in parallel with the dicentric method for feasibility in triage dose assessment demonstrated that for whole body exposure, the dicentric scoring resulted in somewhat more accurate estimates in the lower dose range (below 6 Gy), whereas PCC rings performed better in the high-dose range (above 6 Gy). The difference is best explained by inherent characteristics of the calibration curves. Underestimations observed at low doses using PCC rings are most likely due to the sub-linear dose-response as indicated with the data obtained at 1 Gy. A more accurate analysis with several dose points in the low dose range would be required to obtain adequate calibration. The published studies of PCC ring dose-response for low LET radiation have not included doses below 5 Gy (Kanda et al. 1999, Lamadrid et al. 2007), since the aim of the PCC studies is mainly to investigate high doses. The obvious explanation of the underestimation of high doses applying the dicentric assay lies in the saturation of dicentric aberrations at 8-10 Gy, caused by the inhibition of cell proliferation (Sasaki and Norman 1966). We observed a saturation of PCC rings at doses of 15 Gy and higher. A similar effect was reported by Kanda et al. (1999) but, interestingly, not by Lamadrid et al. (2007) who reported 2.5 times lower PCC ring yields than the linear coefficient obtained in our study using the same radiation quality. This implies that there are clear differences in scoring of PCC rings between the two studies, a fact that may influence the observed yield of rings at high dose range.

An attempt was made to assess the correct placement of homogeneously irradiated samples in dose categories by checking if the 95% confidence limits of the dose estimate encompassed the given dose. Dicentric assay was superior in the categories spanning dose range below 6 Gy whereas PCC rings performed better in categories above 6 Gy.

Lloyd et al. (2000) also tested the efficacy of the triage dicentric assay to identify inhomogeneous exposure. They simulated exposure scenarios where 50 – 97 % of blood was irradiated. According to our accident scenario, the fraction of irradiated blood could be below that, and we simulated exposures where 10-40% of blood was irradiated. The results from this study showed that both assays, applied in triage mode, were not effective in identifying the exposure as partial body: only 61% of the cases were correctly identified by the dicentric assay and 55 % by the PCC ring assay. As a result, the attempt to identify the doses given to the exposed fractions of blood was not successful. In general, the dose was either missed completely or underestimated by the dicentric assay and overestimated by the PCC ring assay. Although the number of inhomogeneously exposed samples was very low for extensive conclusions, the results seem to indicate the limitation of both triage assays to detect partial body exposures where the fraction of irradiated blood is below 40%.

Comparing the efficiency of two biodosimetry assays for triage purposes also includes the aspect of time. In the current study, scoring times were not systematically followed but time recording of occasional sample analyses indicated no radical differences in performing fast scoring to 300 PCC cells in comparison to analysis of 50 metaphases with centromere counting.

PCC rings demonstrated non-Poisson distribution after most of the doses in establishing the dose calibration curve and this observation induces problems with respect to identification of partial-body exposures. We think that the overdispersion of is due to scoring of rings in both metaphases and G2 cells. Because differentiating between G2 phase and early metaphase cells is often impossible, we scored PCC rings in cells from both phases of the cell cycle. Consequently, the yield of PCC rings obtained is representative for two phases of the cell cycle: the "normally damaged" mitotic cells and the "highly damaged" G2 cells.

In a mass casualty accident it is essential that a fast and reliable triage method for crude assessment of radiation dose covering a large dose range is available. Our study confirmed that the PCC ring assay is suitable for use as a biodosimeter following whole-body exposure to high doses of radiation. However, the technique appears to possess limitations in the triage of people exposed to high, partial body doses.

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The French Defense radiation protection service (SPRA) and the national response in case of a radiological accident

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Abstract

The French Defense radiation protection service (SPRA) is located on the site of the Percy military hospital which is well known for the treatment of radio-contaminated and irradiated wounded.

The SPRA is involved in many military and civilian training courses, especially to teach the principles of medical response in radiation accidents (triage of absolute and relative emergencies, specific technics and drugs, psychological aspects). Intervention in case of a radiological or nuclear event is also one of the major missions.

First of all, the SPRA has to control all the Centers for the treatment of radio-contaminated wounded in France (CTBRC). During the exercises performed by the French Navy or the Air Force (transportation of a nuclear weapon, accident on a ballistic missile submarine), the SPRA provides hygiene and safety support to Ministry of Defense and advises the headquarters.

If necessary, the SPRA is permanently able to send, by road or by aircraft, an expert team and an analysis team with mobile laboratories (spectrometric and radiochemical analysis) in France or overseas if requested by military or civilian authorities.

The SPRA has initiated for the French armed forces health service conventions with institutions like the French Atomic Energy Commission (CEA), EDF Group (Electricité De France) or the French Radiation Protection and Nuclear Safety Institute (IRSN) in order to supply the best medical care for an ionizing radiation victim in the nearest CTBRC or in the Percy military hospital.

Radioprotective efficiency from consecutive application of indralin and interleukin-1β at the acute irradiation

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Abstract

Experiments in 316 male mice (CBA x C57Bl) F1 have shown, that consecutive application of indralin in a dose of 200 mg/kg 15 minutes before irradiation and interleukin-1β in a dose of 50 mkg / kg in 15 minutes after irradiation increases the survival rate of the mice irradiated in $LD_{50\text{-}90/30}$ for 40-60 %, reduces the depth of postradiation infringements of hemopoiesis, increases proliferative activity of hemopoietic cells. It allows to reduce expressiveness of leuko-, lympho- and neutrophilopenia in early terms after irradiation and to speed up restoration of the contents of cells in peripheral blood. The stimulating effect of consecutive application of indralin and interleukin-1β is shown also in prevention of decrease in activity of myeloperoxydase and alkaline phosphatase, maintenance of higher level of glycogen in neutrophils of peripheral blood of the irradiated animals.

Introduction

The existing system of medical radiation protection is based on the complex of therapeutic and preventive arrangements aimed at the preservation of life and health of people exposed to ionizing radiation. Decrease in the intensity of adverse effects of irradiation in dangerous for human being doses is achieved by application of special pharmacological drugs of preventive or therapeutic action (Bump E., Malaker K. 1997; Kuna P. et al. 2004; Grebenyuk A. et al. 2007; Baranov A., Rozhdestvensky L. 2008).

Indralin is one of the most effective modern radioprotectors which was successfully applied for protection of participants in the rectification of the consequences of the accident at the Chernobyl Atomic Power Station (Vasin M. 2006). One of the most prospective radioprotective substances from the group of means and methods of early therapy of radiation injuries is interleukin-1 (IL-1) (Neta R. 1997; Grebenyuk A. et al. 2005; Rozhdestvensky L. et al. 2008).

The efficiency of consecutive application of radioprotectors and modern means of early therapy of acute radiation syndrome has not been investigated by now. The most prospective of these schemes is the application of radioprotector indralin and recombinant interleukin-1β (IL-1β), which is applied in Russia as a means of emergency therapy of radiation injures.

The purpose of the research was the experimental estimation of efficiency of consecutive application of indralin and interleukin-1β at acute irradiation.

Material and methods

Experiments were carried out in 316 male mice (CBA x C57Bl) F1. The animals were housed in standard conditions of vivarium. The rules of humane treatment of experimental animals were observed during experiments.

Indralin was entered *per os* (with the help of a probe into a stomach) in a dose of 200 mg/kg 15 minutes before irradiation. Interleukin-1β was injected intraperitoneally in a dose of 50 mkg/kg 15 mines after irradiation.

The estimation of radioprotective and therapy effects of preparations was carried out by studying 30-days survival rate and average life expectancy of the dead animals and by estimating the number of hemopoietic cells in endogenous and exogenous colony formation techniques (Till J, McCulloch E. 1991). The total number of leukocytes and differential white blood cells count were estimated with the help of standard methods. Cytochemical determination of myeloperoxidase, alkaline phosphatase and glycogen was carried out in peripheral blood neutrophils.

The animals were exposed to X-ray irradiation on the RUM-17 unit with the following conditions: voltage 180 kV, current 15 mA, filter 0.5 mm Cu + 1.0 mm Al, source to skin distance 50 cm, dose rate 0.292 mA/kg (52.2 R/min), one-way irradiation, the direction of exposure: back-to-chest. Dosimetric support of experiments was carried out with the help of a dosimeter ID-11 and calculation method.

The data obtained in experiments were put to the standard statistical analysis with average value and error of mean calculations with the help of the computer program Statistica 6. Fisher's exact test and Mann-Whitney U-test were taken for the estimation of the reliability of the differences of mean values. Probability $p<0.05$ and more was considered to be reasonable for a conclusion about statistical confidence of the differences of the obtained data.

Results

The data from Table 1 show that consecutive application of indralin and IL-1β led to the significant increase in the number of mice survived after exposure to radiation in comparison with the control group and groups of animals which received these preparations separately.

Table 1. Effect of indralin and interleukin-1β on the survival rate of mice (CBA x C57Bl) F1, exposed to X-ray irradiation in doses of 6.3, 6.8 and 7.3 Gy.

Dose, Gy	Experimental group	Number of mice in group	Survival rate, %
6.3	Irradiation (control)	10	50
	Indralin + irradiation	10	60
	Irradiation + IL-1β	10	60
	Indralin + irradiation + IL-1β	10	90 *
6.8	Irradiation (control)	12	25
	Indralin + irradiation	12	40
	Irradiation + IL-1β	12	25
	Indralin + irradiation + IL-1β	12	83.3 *
7.3	Irradiation (control)	15	0
	Indralin + irradiation	15	33.3 *
	Irradiation + IL-1β	15	20
	Indralin + irradiation + IL-1β	15	46.7 *
* p < 0.05 in comparison with irradiation (Fisher's exact test).			

At the same time none of the studied schemes of preventive treatment and therapy of radiation injures hasn't had a significant effect on average life expectancy of animals died after irradiation. The major part of the mice died on the 10-14th day after irradiation which is typical for small laboratory animals with bone marrow form of acute radiation syndrome.

The estimation of radioprotective and therapy efficacy of separate and combined use of indralin and IL-1β on the number of hematopoietic early precursors was carried out in endogenous and exogenous colony formation techniques.

It has been found out that indralin, IL-1β and their combination prevented postradiation hemopoiesis depression. The number of colonies on the spleens of mice-hybrids, estimated in endogenous colony formation technique, was higher in the group of mice which were injected with the combination of preparations than in the groups of animals which received indralin or IL-1β only (Tab. 2).

Table 2. Effect of indralin and interleukin-1β on the number of the 9-th day colony-forming units grown on the spleens of X-ray irradiated mice (CBA x C57Bl) F1 in the endogenous colony formation technique.

Experimental group	Dose, Gy		
	6.5	7.0	7.5
Irradiation (control)	3.9 ± 0.8	2.8 ± 0.9	1.9 ± 0.8
Indralin + irradiation	12.0 ± 0.0 *	9.7± 1.2 *	6.8 ± 1.8 *
Irradiation + IL-1β	9.5 ± 2.0 *	9.2 ± 1.8 *	6.7 ± 1.2 *
Indralin + irradiation + IL-1β	16.0 ± 0.0 * # "	12.1 ± 1.0 * # "	10.5 ± 0.5 * # "
* p < 0.05 in comparison with irradiation (Mann-Whitney U-test); # p < 0.05 in comparison with irradiation + indralin; " p < 0.05 in comparison with irradiation + IL-1β.			

The data in Table 2 demonstrate that radiation exposure caused dose-dependent decrease in the number of endogenous colonies on the spleens of irradiated mice-hybrids**.** The largest number of colonies grown on the spleens of irradiated mice (i.e. the largest number of hemopoietic stem cells which retain viable) was registered in animals which received both preparations. For example, consecutive preventive application of indralin and early therapeutic application of IL-1β led to fourfold increase in the number of colonies on the spleens in mice-hybrids exposed to radiation in a dose of 6.5 Gy. When exposed to radiation in a dose of 7.5 Gy, consecutive application of indralin and IL-1β led to more than fivefold increase in the number of endogenous colonies in comparison with the animals of the control group (mice exposed to radiation without pharmacological protection) and to 1.5 time increase in comparison with the separate use of indralin or IL-1β .

Moreover, it has been found out that the separate use of indralin or IL-1β didn't have a significant effect on the number of the 9-th day colony-forming units (CFU-S_9) on the spleens of irradiated male mice (CBA x C57Bl) F1 detected in the exogenous colony formation technique. At the same time the number of CFU-S_9 of irradiated mice-hybrids injected with indralin and IL-1β was 2 times more than the number of CFU-S_9 in the control group (irradiation), 1.4 times more than in animals injected with indralin only and 1.7 times more than in animals injected with IL-1β only (Tab. 3).

Table 3. Effect of indralin and interleukin-1β on the number of myelocaryocytes and number of the 9-th day colony-forming units grown on the spleens (CFU-S_9) of X-ray irradiated mice (CBA x C57Bl) F1 in the exogenous colony formation technique.

Experimental group	Number of myelocaryocytes, x 10^6/femor	Number of CFU-S_9	
		in one femor	on 10^6 myelocaryocytes
Irradiation (control)	18.3	64 ± 16	3.5 ± 0.9
Indralin + irradiation	20.0	88 ± 24	4.4 ± 1.2
Irradiation + IL-1β	20.8	72 ± 16	3.5 ± 0.8
Indralin + irradiation + IL-1β	22.3	136 ± 8 *	6.1 ± 0.4 *
Endogenous level	23.8	320 ± 100	13.4 ± 4.2
Indralin	20.8	520 ± 60	25.0 ± 2.9
IL-1β	22.8	340 ± 40	14.9 ± 1.8
Indralin + IL-1β	25.3	400 ± 140	15.8 ± 5.5
* $p < 0.05$ in comparison with irradiation (Mann-Whitney U-test)			

Moreover, it has been found out that combined introduction of the indralin and IL-1β had a positive effect on hemopoiesis of irradiated mice estimated by the dynamics of the number of leukocytes in peripheral blood during all follow-up period (Fig. 1).

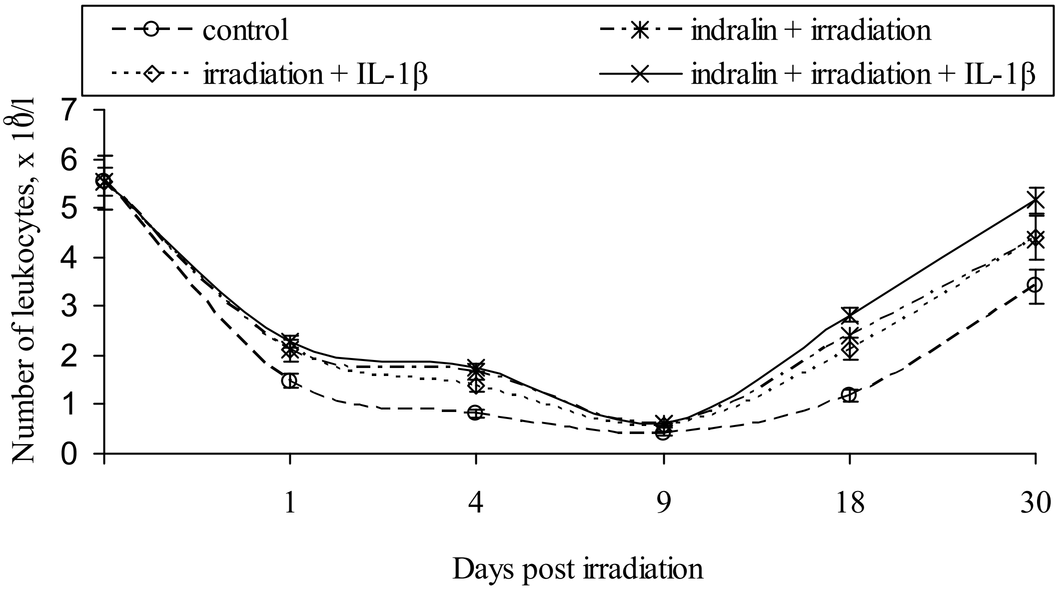

Fig. 1. Effect of indralin and interleukin-1β on the dynamics of the total number of leukocytes of peripheral blood of mice (CBA x C57Bl) F1 exposed to radiation in a dose of 8 Gy.

For example, even on the first day after irradiation the number of leukocytes in the blood of animals which received both preparations was higher than in the control group (50%) and in groups with separate injection of indralin or IL-1β.

Consecutive preventive application of indralin and early therapeutic application of IL-1β significantly prevented the decrease in the total number of neutrophils of peripheral blood of mice irradiated in a dose of 8.0 Gy during all follow-up period and didn’t differ from the background count on the 20-th day.

In addition, both separate and consecutive application of indralin and IL-1β stimulated the functional capacity of peripheral blood neutrophils of irradiated animals which was manifested in significant increase in the content of myeloperoxidase, alkaline phosphatase and glycogen in these cells in comparison with control groups.

The data in Fig. 2 demonstrate that glycogen content in neutrophils of mice which were consecutively injected with the radioprotector and hemopoiesis stimulator was higher than in irradiated mice during all follow-up period, but did not differ from those in groups of mice injected with indralin or IL-1β only.

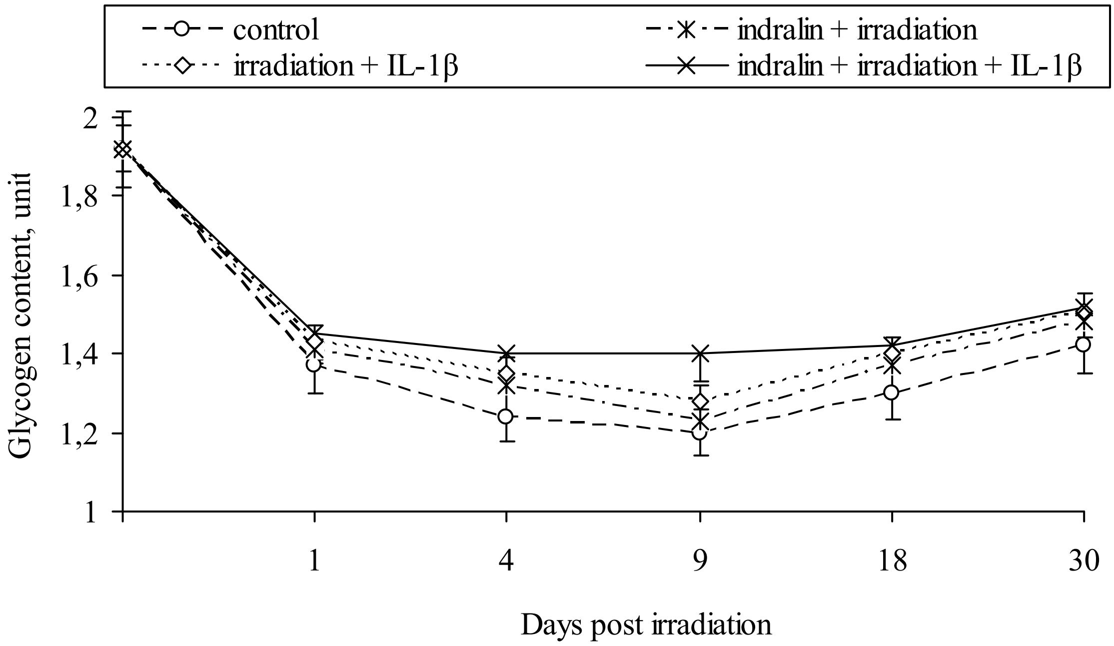

Fig. 2. Effect of indralin and interleukin-1β on the content of glycogen in peripheral blood neutrophils of mice (CBA x C57Bl) F1, exposed to radiation in a dose of 8 Gy.

Discussion

Our findings on radioprotective efficiency of indralin and early therapeutic activity of IL-1β agree with those of the other authors (Neta R. 1997; Legeza V., Chigareva N. 1999; Rozhdestvensky L. et al. 2008; Vasin M. et al. 2008), supplement them and increase the sum of knowledge of the mechanism of therapeutic action of these preparations.

It was found out that application of indralin 15 minutes before exposure to radiation effectively protects mice irradiated in $LD_{50\text{-}100/30}$ against death increasing their survival rate by 10-30%, has a positive effect on bone marrow hemopoiesis of irradiated animals, reduces the intensity of leuko-, lympho- and neutrophilophenia in early terms after irradiation, accelerates restoration of the contents of cells in peripheral blood. High radioprotection level on application of indralin has been established earlier in experimental studies in many species of laboratory animals: mice, rats, Syrian hamsters, guinea pigs, rabbits, dogs and monkeys (Vasin M. et al. 2008). Radioprotective effect of this preparation is based on its ability to reduce oxygen content in the cell, thereby reducing the intensity of “oxygen effect” and manifestation of oxidative stress. For hemopoiesis stimulator IL-1β this mechanism is not essential (Neta R. 1997). However, its introduction to mice after exposure to radiation can increase radioprotective efficacy of indralin, which is estimated both by the survival criterion and the condition of bone marrow hemopoiesis. For example, consecutive preventive application of indralin and early therapeutic application of IL-1β can increase the survival rate in mice irradiated in $LD_{50\text{-}100/30}$ by 40-60%, although do not have a significant influence on average life expectancy of dead animals.

In our opinion, one of the most important mechanisms, which form the basis of radioprotective effect of combined use of these preparations, is that IL-1β activates proliferation and differentiation of hemopoietic stem cells in a great number preserved in the bone marrow of irradiated mice due to the protective action of indralin. This statement can be confirmed by the fact that both separate application of IL-1β and combined application of IL-1β and indralin reduce intensity of postradiation disorders in bone marrow hemopoiesis, increase proliferation activity of hematopoietic stem cells, which finally leads to faster recovery of absolute content of cells in peripheral blood.

The fact that IL-1β mainly has an effect on the hemopoietic stem pool retaining some of the cells from death and/or intensifying its proliferation has been shown earlier (Neta R. 1997; Grebenyuk A. et al. 2005; Rozhdestvensky L. et al. 2008). However, in the course of our investigations it was found out that the largest number of CFU-C_9 in irradiated mice can be observed in condition of combined application of indralin before exposure to radiation and IL-1β after irradiation.

Conclusions

The results of the research have shown the effectiveness of consecutive preventive administration of indralin and early therapeutic application of IL-1β for protection against acute irradiation. In this case, hemopoiesis stimulator IL-1β increases the radioprotective efficacy of indralin estimated by the survival criterion of irradiated mice, the condition of bone marrow hemopoiesis and functional activity of peripheral blood leukocytes.

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Hospital response plan for radiation emergencies: the project of Careggi University Hospital in Firenze

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Abstract

The plan described in this paper outlines the procedures used by Careggi University Hospital in Firenze in response to incidents where individuals are exposed to radioactive material either in case of radiological accidents (for instance during transportation, industrial or medical activities) or in case of malevolent use of radioactive sources.

The main challenge in planning a radiological emergency response is the proper definition of departments and people involved, their own duties and responsibilities, the effective coordination among them, also taking into account the pre-existing hospital structures and concurrent routine activities. Moreover in a mass casualty incident, many people may come to an healthcare facility injured, contaminated, or not. Most of them are likely to be minimally injured or simply concerned about potential contamination, therefore needing just an ambulatory care or measurement. In any case the correct answer to everyone has to be given without interfering with the other activities of the Emergency Department.

In this plan the functions, duties and responsibilities of staff people of every involved departments are reported in detail. The plan covers all the operating procedures starting from the radiation emergency alert till the closure of emergency operation, including the waste disposal.

The main topics of the plan are the preliminary hospital activation, the contamination assessment procedure, the possible decontamination of patients, the radiation protection of the staff. Special sections are devoted to define a list of radiation measurement instruments, emergency supplies and pharmaceuticals to be stored in the Emergency Department in order to properly handle the response.

This plan is part of a more general CBR hospital response plan, and a first step towards a regional response plan to radiological emergencies through the Italian National Health Service structures.

Introduction

The plan has the goal to analyse and describe the hospital response to a radiological emergency, defining departments involved, duties and responsibilities, facilities, materials and instrumentation required.

The plan takes into account the already planned response to conventional emergency and it is intended as a component of a more comprehensive CBR plan.

Main goals of the hospital response to the emergency are:

- the rapid identification of the type of emergency
- the subsequent activation of the response
- the interaction with other external structures
- the coordination with other regional highly specialized hospitals.

Material and methods

Florence University Hospital (Azienda Ospedaliero-Universitaria Careggi, AOUC) is located in Tuscany Region, Italy, and it is one of the main hospitals of the Italian National Health Service (NHS). It is tightly integrated within the Florence University, thus maximizing the synergies between health care, research and teaching. The hospital is organized as a campus with different buildings, that together house over 5,800 employees, 1650 beds, 54000 in-patients, 22000 day hospital admissions and nearly 10 million medical examinations per year.

The hospital hosts nuclear medicine, radiodiagnostic and radiotherapy and Health Physics departments, where health care workers are highly trained in radiation protection. RPE responsibilities are included in the Health Physics Dept. activities.

Available facilities and instruments are a gamma spectroscopy system for isotope identification and activity measurement, a portable gamma spectrometer, TLD and electronic dosimeters, contamination and exposure measurement instruments, decontamination kits, gamma probes.

The Emergency Department is located centrally with respect to other buildings, and has enough outdoor spaces to host a radiation assessment mobile tent and a decontamination mobile tent as well.

A Tuscany region reference centre for toxicology and an internal occupational health department charged to deal with all the medical aspects related to exposure to ionising radiation are also present. Bioassay labs can provide immediate assessment of blood contamination as well as the other clinical blood parameters.

Considered scenarios

At present no nuclear power plant is operating in Italy, so the considered scenarios are:

- incident in industrial facilities
- incident in medical facilities
- incidents during transport of radioactive material
- orphan sources incidents
- malevolent use of radioactive material

Involved departments

The departments/services involved in the emergency response are:

- Emergency Department (ED): resources are necessary for the treatment of patients suffering from injuries in combination with possible radiation contamination or exposure (combined injuries).
- Health Physics Department: for radiation protection, radiation screening, isotope identification and any dosimetric assessment
- Toxicology and occupational health services to provide physician competent in radiation health effects
- Technical services department for security, mobility and logistics operation
- Press Office
- Psychology team

Results

Allocation of roles

An Hospital Operation Command Team (HOCT) has to supervise the emergency operations and has to be alerted by the ED manager.

The components of the HOCT are the hospital medical director, the ED director, the Radiation Protection Expert, the technology services director and the nurse staff director.

The proper management of people arriving to the hospital is provided by the security staff.

The contamination screening of people arriving to the hospital is performed by Health Physics staff, that assures at least one Medical Physics Expert (MPE) and one Radiology Technician on call at any time.

Patient medical treatment is provided by the ED personnel. In order to reduce the spread of contamination and to allow the routine operation, a section of the ED is isolated and dedicated completely to the radiation emergency.

Information workflow

The radiation emergency plan is activated at once as a notification is transmitted by First Responders (Fire Department).

From this moment on the medical operations are lead by the ED, while the HOCT coordinates the global hospital response. The information workflow is shown in Fig.1

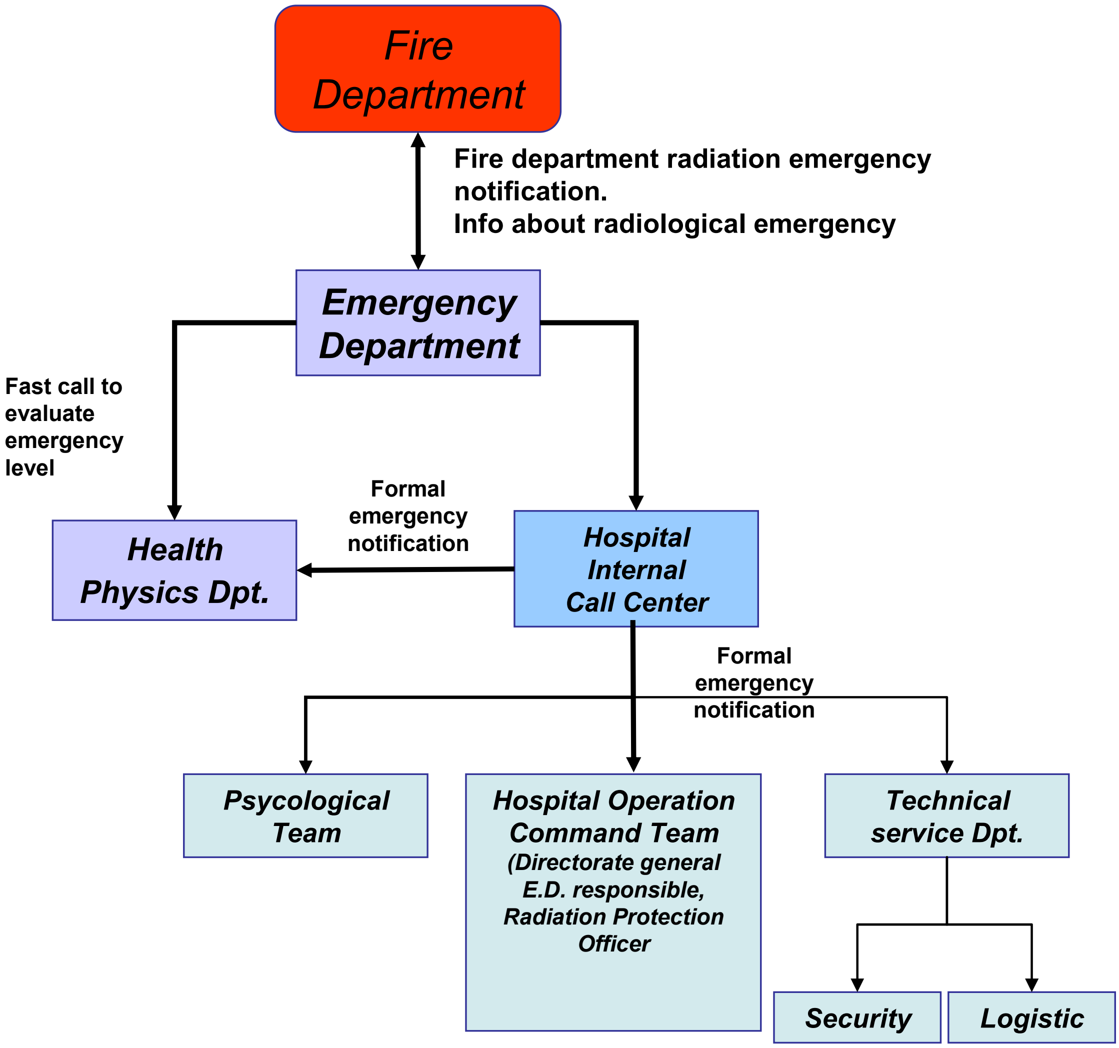

Fig. 1. Information workflow in case of radiological emergency.

Patient admission

In case of radiation emergency, every access to the hospital is closed but one in order to properly filter the accessing people. At the entrance of the ED people involved in the radiological emergency are redirected to the pertinent special ED section. All other kind of access is limited until the first response to the emergency is declared over by the HOCT.

The radiation screening and decontamination are performed outside the buildings, in dedicated mobile tents (Fig.2).

Personnel Training

People involved in the emergency plan (health care professionals, security and logistics staff, psychology team, HOCT) are required to attend training courses.

The training course includes a general section about the radiation protection and the plan structure, and a more specific session devoted to health care staff only.

The goal of the training is to provide a basic overview of the most important issues in an effort to raise the level of confidence and lower the level of anxiety of the staff who may be called to deal with this kind of event.

The training covers the following topics:

- Characteristics of ionizing radiation and radioactive materials
- Differentiation between radiation exposure and radioactive material contamination
- Causes of radiation exposure and contamination
- Staff radiation protection procedures and practices
- Facility preparation
- Patient assessment and management of radioactive material contamination and radiation injuries
- Health effects of radiation exposure
- Facility recovery and waste disposal

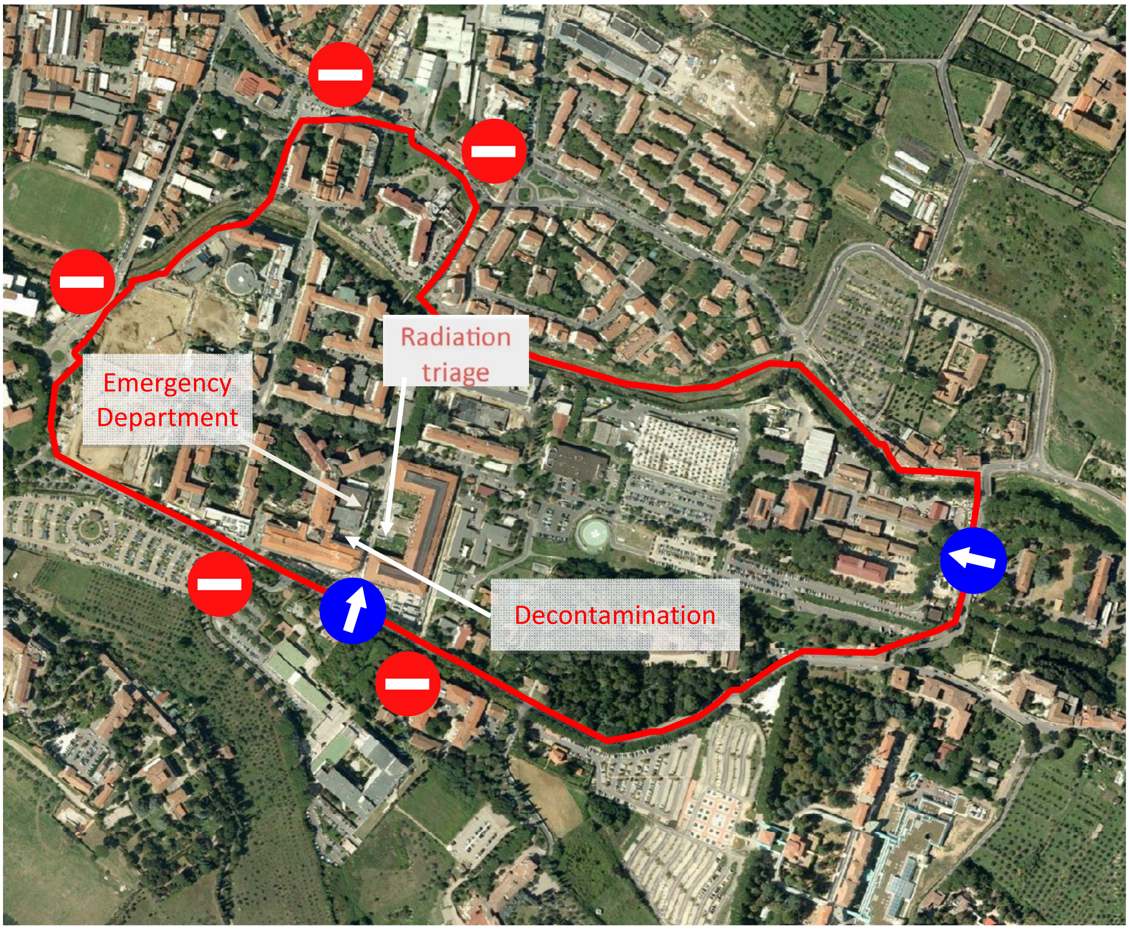

Fig. 2. Plan of the hospital, with closed access in case of radiological emergency. The access close to the ED is devoted to people involved in the radiological emergency, the other access to patients.

Conclusions

The proposed plan covers the organisational aspects, defines the professional roles, charges the departments and structures to provide effective response in case of radiological emergency.

This plan is part of a more general CBR hospital response plan, and a first step towards a regional response plan to radiological emergencies through the Italian National Health Service structures.

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Occupational medicine professionals and radiation accidents

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Abstract

Protection of the public exposed to nuclear/radiological (NR) agents in medical or industrial installations is required in many countries by Regulatory and Health Authorities. National Authorities require training in all aspects of occupational and public exposures to ionizing radiation, as well. Specific aspect of public exposure is in the case of accidents or terrorist/malevolent acts. Many countries increase their efforts to prepare medical workers for the intentional release of NR agents within the civilian population by terrorists or accidents on nuclear/radiological installations. Presumed targets for terrorist attack are urban or suburban heavily populated locations. They are usually industrial parts of the cities, and bring additional risk correlated with industrial installations. For many people, the first point of contact with harmful agents could be their workplace, and therefore occupational medicine professionals (OMP) are considered as potential first responders to a terrorist attack. They are educated in health surveillance of personnel occupationally exposed to low-doses of harmful agents, including radiation. Still, the lack of training for medical response in radiation emergency has been recognized. Therefore, OMP must be additionally trained on the specific challenges of terrorism and emerging situations, including how to recognize and isolate, treat and track individuals who will need prompt and appropriate care, especially in the case of mass casualties. OMP from MMA, as members of NRCB team, are involved in different radiation emergencies: from decommission, depleted uranium, to terrorism, and therefore additional educational courses for students, general practitioners, and OMP have been organized. In this paper, our experiences and the main topics of this program are presented.

Introduction

Ionizing radiation has been used for the benefit of humankind for a long time. Still, it`s harmful effects, overexposure and accidents are the first associations of radiation for more the 60% of adults from EU. This attitude is probably caused by horrible experience from Hiroshima and Nagasaki, as well as some other accidents.

More recently, NR terrorism became an actual problem. Modern society is known by global anti-terrorist activity. It seems like there has never been so much activity on reducing nuclear threat, and at the same time, never bigger possibility of it being used.

That was the main reason for global anti-nuclear activity and efforts for improvement of public protection against malevolent use of NR material.

Many countries in the world created wide and detailed national plans for emergency preparedness in the case of NR accident, as well as for NR attack. One of the most important elements of preparedness is adequate education of the responders. Medical workers are unavoidable members of the team. Even though radiation effects are usually studied in medical schools and numerous specialization courses it seems that physicians are not sufficiently prepared for the emergency in case of NR accident, especially in the case of the mass causalities. Therefore, the global attitude is that the improvement of knowledge and skills of medical workers in this area are needed for providing help and treatment to the public.

Material and methods

In this paper, programs of additional education for OMP, young general practitioners on compulsory military service and doctors on specialization courses in Military Medical Academy are presented. They are based on IAEA recommendations and adapted to previous knowledge of the participants and their possible role and responsibilities in the NR accidents.

Results and Discussion

Potential sources of nuclear/radiological accidents

here are several potential sources of nuclear/radiological accidents. At the first place, the source of the accident could be nuclear installations due to damage or human mistake. These accidents are rare; they involve a lot of people and demand the activity of many services. The second potential source of the accident could be lost sources, so called *orphaned sources*. These accidents are not so rare, and the number of people involved is different depending on scenario.

More recently, important potential sources became NR terrorist devices. In a last decade, numerous blackmails and threats involving these devices were denounced worldwide. Until now, no cases of terrorist actions involving radioactive material have been known. Nevertheless, efforts to prevent them are still needed. There are two groups of devices: improvised nuclear devices (IND) and radiological dispersal device (RDD). It is presumed that attack by IND is less possible, because they are harder to produce, transport and provide with sufficient nuclear material. The most possible scenario is attack by RDD, which is easy to produce, transport and use. Most of all, it is easy to provide a smallest quantity of radioactive material needed for production of RDD. The highest advantage of RDD is the fear and panic they cause, even if they contain the smallest quantity of radioactive material.

Potential targets for terrorist nuclear/radiological attack

Presumed targets for terrorist attack are urban or suburban heavily populated locations. They are usually business or industrial parts of the cities, or specific places like underground, harbors, airports, theatres, even objects like hospitals, kinder-gardens, and schools. Finally, nuclear or industrial installations could be marked as targets and induce mixed-hazards situation.

For many adults, the first point of contact with harmful agents could be their workplace, and therefore occupational medicine professionals (OMP) are considered as potential first responders to a terrorist attack.

Emergency response

Emergency response in these situations is complex and demands activity of numerous services. Medical workers are unavoidable members of the rescue teams on all levels, from the first response to final hospital treatment. The aim of the protection and response is to prevent deterministic effects and minimize the possibility of stochastic effects.

Final treatment for wounded and sick people who are irradiated or contaminated in NR attack should be provided in specialized hospitals. In order to prevent pressure on these hospitals it is crucial to organize qualified medical teams on-site and on pre-hospital level. They will provide first aid and triage. Therefore, it is extremely important to educate first responders and to organize the entire network. It is presumed that city emergency medical teams would react, but in the case of attack they could be slowed down or even stopped due to panic, traffic troubles or any other reason. In the case of NR attack with several explosions, and the unknown number of radiation sources, numerous emergency teams are needed. Therefore, it would be rationale to provide emergency medical response from the nearest medical professionals.

Why OMP

OMP are trained to follow-up harmful effects of many physical, chemical, biological and psychological factors from the workplace. They are also trained to detect and measure doses or levels of these agents and effects, as well as to assess the risk of exposure to them. Most of OMP are situated in the medical centers in the industrial areas or factories. They are responsible for medical treatment of the workers on daily basis. In the accidental situation of any kind, OMP are the nearest medical teams to provide help on-site. Besides the emergency medical treatment, OMP teams could be useful for a few different reasons.

OMP services settled in industrial areas are supplied by materials necessary for emergency situations connected with industrial process. OMP are completely aware of all the physical, chemical pollutions released from the industrial process. They are usually equipped with detection and dosimetric tools and have a date base of previous measurements taken before the accident, so they can notice any significant difference and increase in pollutions. These data could be of a great importance for the activity planning and risk assessment. OMP are familiar with early notification of accidents, as well as the chain of action and responsibility. This service has a data base of workers health. In the accidental situations they can provide previous results and help in treatment planning. Occupational medicine departments have epidemiological data which could be of great importance in the evaluation of the accident.

Still, further education of OMP is needed, because the main goal of their interest is professional exposure and the effects of chronic or prolonged exposure to low doses or low level doses to harmful agents from working environment. Besides that, health surveillance is based on previously planned activities. On the contrary, in the accidental situations they are faced with sudden exposure of many people to high doses/levels of harmful agent/s. That implies necessity for further education of OMP. Good educational programs should be specific and adapted to target groups, to their assignments and responsibilities in the accidental situations. One of the most important assignments is first medical aid on-site of the accident, so the training and education of the first responders are crucial for the good help, as well as the protection of responders.

Courses in Military Medical Academy

Military Medical Academy organizes medical education for different profiles of medical workers. Basic NRCB course is compulsory for all the specializations. The main topics of this course vary for different specializations, from preventive medicine to clinical sciences.

OMPs course

Considering OMP as very important for the mentioned accidental situations, courses on urgent toxicology in National Center for Poison Control are organized, including the basic principles of emergency toxicological procedures and detection of toxic substances. The second part of the course consists of: basic principles of detection, prevention and protection in the case of bioterrorist acts, as well as basic principles of microbiological diagnosis, clinical signs and treatment. The third part of the course is radiation protection. Emergency medical response is the main goal, and also detection of different kinds of ionizing radiation, basic dosimetry (field and personal), decontamination, treatment of acute radiation syndrome and skin injuries. At last, OMPs are well trained to protect themselves and members of their team. This program is sufficient to prepare them for all-hazard situations. Further trainings are needed to refresh their knowledge and skills.

After this program our OMP could be involved in post-accidental follow-up of late consequences of harmful agents, as well. Besides the contamination of the people, contamination of the food, water and the environment is a potential consequence of the radiological accident/attack. Radiation protection department, as a part of the Occupational medicine, is responsible for the monitoring of these goods. The results of radiometric analysis are crucial for further activities of the OMP.

NRCB for doctors on compulsory military service

Young doctors-general practitioners on compulsory military service are obligated to attend additional NRCB course. Main goal of this course is medical management in accidental situations with mass causalities. These groups are trained for field work. Besides the basic topics mentioned above, they are trained to organize decontamination of large groups of people in a tent which is the part of the field military hospital. They are trained for the first aid, triage, transport and basic treatment of victims in NR accident.

These courses are part of our efforts to provide a network of adequately trained first responders. For the real improvement, regularly repeated trainings and preparedness are needed. Increased spectrum of NR devices and possible scenarios of NR attacks, in turbulent political situation, imply further development of emergency preparedness, not only on local, but also, on global level.

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Brain tumour risk in relation to mobile telephone use: results of the INTERPHONE international case-control study

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Abstract

The rapid increase in mobile telephone use has generated concern about possible health risks related to radiofrequency electromagnetic fields from this technology. An interview-based case-control study with 2708 glioma and 2409 meningioma cases and matched controls was conducted in 13 countries using a common protocol.

A significantly decreased risk was seen in relation to ever having been a regular mobile phone user both for glioma (odds ratio (OR) 0.81, 95% confidence interval (CI): 0.70,0.94) and meningioma (OR 0.79; 95% CI: 0.68, 0.91), possibly reflecting participation bias or other methodological limitations. Odds ratios were below 1.0 for all deciles of lifetime number of phone calls and nine deciles of cumulative call time. While there was no evidence of dose-response, a significantly increased risk of glioma was seen among users in the highest decile of cumulative call time (OR 1.40; 95% CI: 1.03, 1.89). That risk was greatest among subjects with tumours in the temporal lobe, where RF absorption is generally highest, and among subjects who reported using their phones on the side of the head where their tumour occurred. Self-reports of phone use are, however, subject to considerable recall error. Sensitivity analyses conducted to evaluate the robustness of the findings generally showed similar results.

Overall, no increase in risk of either glioma or meningioma was observed in association with use of mobile phones. There were suggestions of an increased risk of glioma at the highest exposure levels, but biases and errors prevent a causal interpretation. The possible effects of long-term heavy use of mobile phones require further investigation.

UV-A radiation enhances melanoma metastasis in mice

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Extended abstract

Ultraviolet (UV) radiation is considered the major factor in skin cancer development (1). Due to the increasing popularity of skin tanning lamps, the potentially deleterious effects of solarium-derived UV radiation have become a public health concern, since the use of solaria has been linked to the development of melanoma, a particularly aggressive subtype of skin cancer (2,3). Melanoma is characterized by high risk of hematogenous metastases in the early stages of disease and it is the major reason for melanoma mortality. An understanding of the physiological consequences of UV exposure is of crucial importance in the prevention of melanoma. The possibility that UV radiation may affect melanoma metastasis has not been addressed widely, although it is known that some of the UV-induced cellular effects, such as the systemic immunosuppression (4,5), increased expression of matrix degrading enzymes (6) and adhesion molecules (7,8), are events that can mediate the autonomous growth of melanoma and possibly enhance the metastatic potential of the melanoma cells.

We have investigated the effect of solarium-derived UV-A (320-400 nm) irradiation on the metastatic capacity of mouse melanoma reporting possible link between UV radiation exposure and melanoma metastasis both *in vitro* (9) and *in vivo* (10). Previously we have shown that *in vitro* UV-A radiation enhances the metastatic properties of mouse melanoma B16 cell lines by increasing the adhesiveness of melanoma cells to endothelium and changing expression of adhesion molecules. We have also shown that *in vivo* UV-A exposure of mice increases the formation of melanoma lung metastases in C57BL/6 mice injected *i.v.* with low-metastatic B16-F1 cells. Obtained results have confirmed that mice, that were *i.v.* injected with B16-F1 cells and exposed to UV-A, developed 14 days after treatment 4-times more of lung metastases as compared with the non-exposed group. However, the *in vitro* exposure of melanoma cells, prior to injection into mice, lead to induction only of 1.5-times more metastases as compared with the animals injected with non-irradiated cells. Therefore, UV-A-induced changes in the adhesive properties of melanoma cells cannot, alone, account for metastasis increase observed after *in vivo* exposure of mice.

The fate of the melanoma cells in mice circulation and the body during the first hours as well as days after melanoma cell injection remains obscure in the used mouse model. Although *in vitro* systems to assess melanoma metastasis exist, these systems do not adequately reflect the complexity of the in vivo microenvironment. Similarly, the rapid changes in cell surface molecule expression that occur in dynamic systems are most effectively observed in real time *in vivo*. Therefore the next step was to investigate the metastatic process in real time and to determine how an *in vivo* UV exposure of the mouse affects the melanoma cell circulation kinetics, cell trafficking to the host-organs, and their long-term ability to form metastases utilizing different *in vivo* molecular imaging technologies.

In vivo flow cytometer (IVFC) is an excellent tool with the capability to count circulating fluorescently labeled cells in living animals (11,12), since the migration of cell population of the interest can be observed in the native environment without the need to draw blood samples. Since the circulation kinetics of the B16 melanoma cells in this experimental metastasis model was totally unknown, the depletion rate of the fluorescent-labeled cells was determined using the IVFC technique. Melanoma cells were cleared very fast from mouse circulation and mainly for this reason no significant changes in the cell clearance were found between the UV-exposed mice as compared to the non-exposed controls.

To assess the localization of the B16 melanoma cells after the UV irradiation, the short term trafficking into the lungs during the first hours after melanoma cell injection was studied by non-invasive *in vivo* biochemiluminiscense system that allows the whole body imaging. Bioluminescence imaging showed increased melanoma cell arrest in the lungs 60 minutes after UV-A exposure compared to controls. As a long-term effect of the UV exposure on melanoma cells, UV-A irradiation was also shown to enhance remarkably the proliferation of B16-F1 cells in the lung parenchyma as compared to the non-exposed group during the next 15 days.

As conclusion we suggest that UV-A-derived increase of metastasis *in vivo* might be a combination of enhanced adhesion to pulmonary vasculature and enhanced proliferation that both contribute to the observed UV-A effect on melanoma metastasis. These results presented in this study suggest that, if occurring also in humans, exposure to UVA radiation during extensive sunbathing or solaria tanning periods might have the potential to cause increase in the hematogenous melanoma metastasis in patients with metastatic disease, but this needs to be confirmed in the future studies.

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Investigation of sun habits in Sweden 2005–2009

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Abstract

Between the years 2005-2009 a questionnaire has been used by the Swedish Radiation Safety Authority in order to see how much UV-radiation the Swedish population is exposed to. This Questionnaire has also made it possible to learn about how the sun habits for the Swedish population, their attitudes and knowledge about UV-related matter. A description of the method used will be presented together with result and discussion about what can be said about the years the questionnaire has been used. The questionnaire has been distributed to 2000 persons in Sweden and is collected by the Statistics Sweden, which also summarizes the results.

Introduction

Sweden has an environmental goal which states that the annual skin cancer incidence shall not be higher in the year 2020 than it was in 2000. A reason for setting up this goal is the fact that skin cancer is one of the fastest increasing types of cancer in Sweden and the main cause for this is the exposure of UV-radiation.

This goal is followed by two indicators, the annual number of skin cancer incidence and the exposure of UV-radiation of the population. The disadvantage of the annual skin cancer incidence as an indicator is that it does not correlate with the exposure of UV-radiation today. The delay between the damage and the onset of skin cancer makes it hard to draw any conclusions from the exposure today and the skin cancer incidence. In order see the actual exposure of UV-radiation one cannot rely on the incoming radiation alone but the crucial part is the behaviour and attitude when it comes to the sun or solarium.

This paper will present the model and questionnaire being used in order to estimate the actual exposure of the Swedish population as well as get knowledge about sun habits.

The indicator for the environmental work reflects how much UV radiation people are exposed to, depending on their behaviour. There are two interesting aspects of exposure to consider, first, the total exposure for the population (population exposure) and the extent of the exposure (burns). Both of these aspects are linked to the risk of getting skin cancer.

The questionnaire also provides information on attitudes toward sun exposure and various protective behaviours, knowledge about ultraviolet radiation and skin cancer. The work with the questionnaire began in autumn 2004 with the participant Katarina

Yuen, Lars-Erik Paulsson and Ulf Wester from SSI[1] and Richard Bränström from Karolinska Institutet (KI) and Helena Bäckström from Statistics Sweden (SCB).

Material and methods

When the purpose of the questionnaire was established the variables that would be used was determined. This section presents the approach when the exposure model and questionnaire was designed. As a first step a decision was made to construct at model that was rough but more easy to work with instead of a model with high precision which would increase the cost and workload. Because of the roughness in the model it is best to compare the results only from the same questionnaire, even though the exposure can be expressed in physical units. The exposure can be based on the following factors:

The duration of our time outside when the sun is reasonably strong.
What we use to protect ourselves from the radiation.
Exposure from other sources than the sun. (Solarium).

The model

The model takes into account behavioural aspects of the exposure and not weather conditions. It is based on MED, minimal erythemal dose where MED is the dose of UV-radiation that barely gives a redness on the skin and should in principle be decided for every individual and situation. Here MED is set as 210 J_{CIE}/m^2 (weighted erythemal dose). The coefficients used in the model has been chosen in order to give the exposure a physical quantity. The simplifications and assumptions in the model make it difficult to speak about the calculated exposure in terms of MED. Therefore the exposure is simply named the exposure in this paper. The exposure can vary between 0 and 1372,5.

Sunprotection

The protection from the sun can be divided into three components, surface, intensity and frequency. Surface means how large part of the skin that are being protected. For example some clothes only cover parts of the body, and the parts that are unprotected is exposed to the radiation that the current environment. The parts that are covered can still be exposed to the radiation depending on the fabric. Intensity takes into account the intensity of the sun and how much the UV-radiation is filtered by the protection. This concerns sparse, thin fabrics and its sun protective properties, badly applicated sunscreen etc. Frequency is how often a protection is being used.

The questions in the questionnaire is for a general situation with beautiful summer weather, i.e. no division for different situations as working time, free time or time abroad. The model assumes mainly free time. But different weights are used for free time and work time since it is reasonable to assume more clothes and shadow in a working environment. Since the model lacks precision, an estimate for the protective properties for different sun protective means as the size of hats, densities of the clothes etc. is built upon reasoning rather than measurement. For example shorts and t-shirts

[1] In July 2008 the Swedish Nuclear Power Inspectorate (SKI) and the Swedish Radiation Protection Institute (SSI) merged into the Swedish Radiation Safety Authority (SSM)

give a protection up to 50 %, scattered radiation can still reach people in the shadow and a protection of about 50 % can be assumed.

One question is concerning how many hours one usually spends outside a day with beautiful weather. The question is limited between 10 am and 4 pm between May and August. A division is made between a working day and a free day/weekend in Sweden and a free day abroad. Weeks of vacation in Sweden and abroad can also be extracted.

Even though the current weather conditions was not included in the model maps from the SMHI (Swedish Meteorological and Hydrological Institute) were studied for the years 1961 – 1990 in order to get a reasonable estimation for weather reduction. From these maps the weather reduction was set as 0,5.

The Exposure outside

The average UV-index in Sweden for the period of interest was approximately 3. One UV-index represents 90 J/(m^2h) erythemal UV-radiation against a horizontal surface i.e. 0,4 MED/h. Thus UV-index 3 corresponds to 1,3 MED/h. But for a realistic situation a person does not expose the entire body to UV-radiation at the same time. The fact that radiation before 10 am and after 4 pm can be of importance 1 MED/h was chosen for the exposure in Sweden and 2 MED/h for the exposure abroad.

Today, most of the sun bed used are full body tanning devices and most people tan with very little clothes. In this case it is reasonable to consider the entire body exposed to the radiation. In the model 2,5 MED/h was used for one tanning session in a solarium.

Results

The exposure

The exposure has been calculated according to the above mentioned model and is shown in figure1 and 2 for the years 2005-2009. During these years the exposure is relatively constant and no tendencies can be seen. The contribution from trips abroad to the exposure is relatively high. If you compare the different age groups from 2009 it is clear that the group with ages 18-24 has the highest degree of exposure. Generally the exposure is decreasing with increasing age.

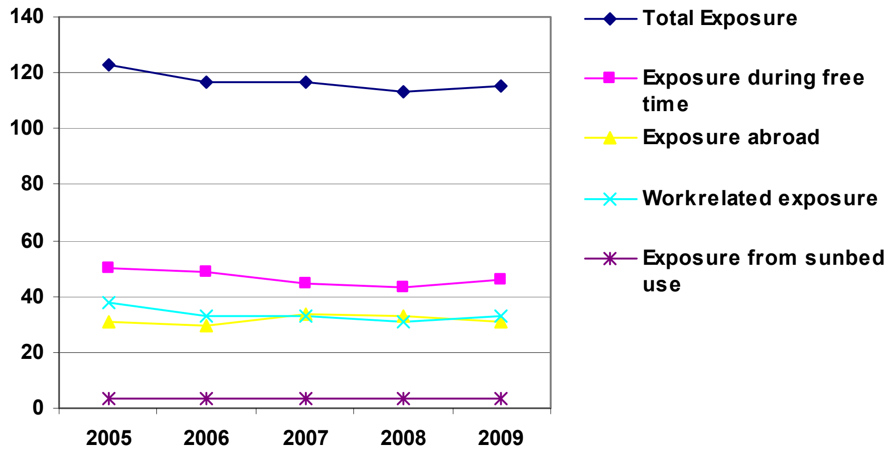

Fig. 1. UV-exposure for the years 2005-2009.

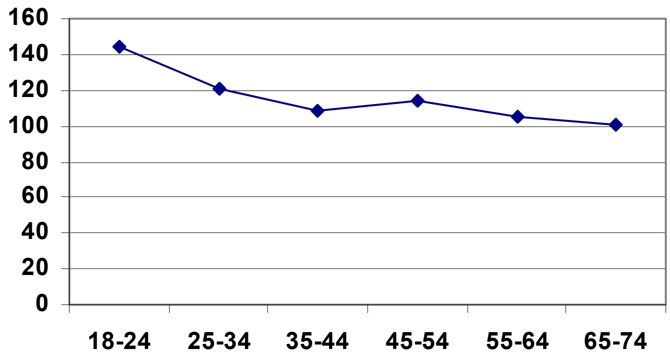

Fig. 2. UV-exposure for the year 2009 divided into different age groups.

Burns

Apart from the total exposure of UV-radiation the burns from UV-exposure is a riskfactor for skincancer. The most common places for getting a sunburn were a garden or balcony and a beach in sweden. Being burned abroad and on a lake was almost just as common.

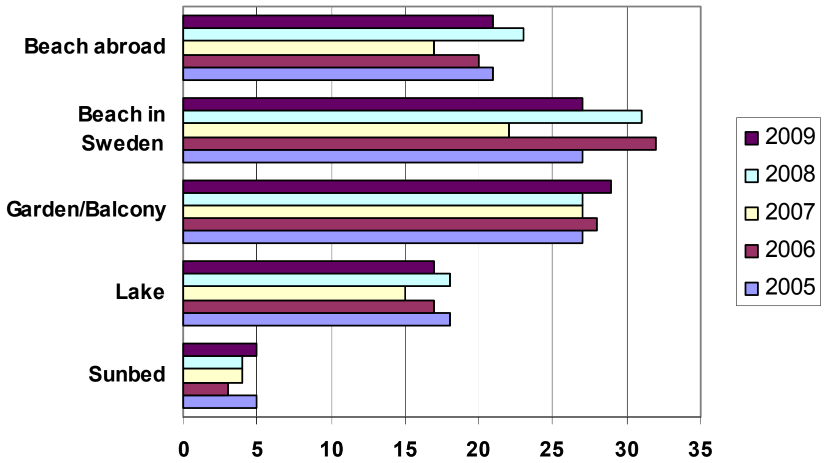

Fig. 3. Percentage of people suffering from sunburn at different locations for the years 2005-2009.

Protection

There are several ways to protect oneself from the sun. For example you could stay in the shade, avoid being outside in the middle of the day, using sunglasses, wearing clothes, cap or a hat or use sunscreen. As is shown below the most common protective behaviour is to wear clothes and the least common behaviour is to avoid being outside in the middle of the day.

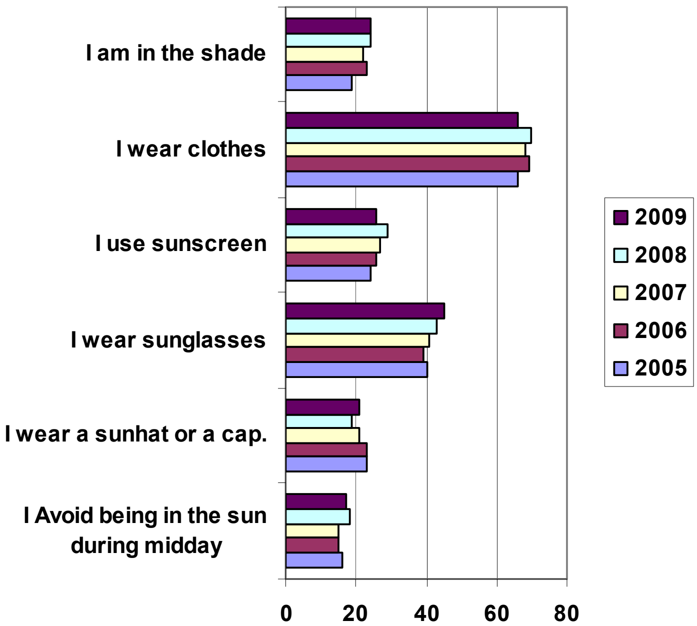

Fig. 4. Percentage of the respondents that always or often protect oneself in different ways.

Attitudes

The most important factor for how much UV-radiation the population is exposed to comes from the behaviour in the sun. There has not been any drastic change between the years 2005 and 2009. Most people enjoy being in the sun and try not to burn themselves. About two thirds think they look better and feel healthier when they have a tan and these statements are truer for women than men. One third worries about getting skin cancer. For the different age groups one can see that the youngest age group compare to the older ones think that using clothes as sun protection is more uncomfortable but the oldest age groups think using a sunscreen is more uncomfortable than the youngest age group.

Knowledge

The knowledge about the sun, skin cancer and other related topics is relatively high. Almost everyone knows that the sun can cause skin cancer and that children are more sensitive to the rays of the sun. There are no drastic differences between the age groups but one might see tendencies that women has a little more knowledge about matters that concerns health.

Discussion

A summary of the questionnaire and its results gives a relatively good overview of the sun habits in Sweden. But so far it is hard to see any obvious trends. Because of this, the questionnaire will not be sent out annually but every second or third year. Not unexpectedly the group with the greatest risk behaviour is the people in the youngest age group. The reasons why people act in a way they know can be harmful most likely depend on several factors. In terms of acute injuries there is only sunburn and possibly some eye damage, but these usually heal reasonable quick and therefore it is probably not a great concern for most people. The long time effect of UV-exposure and burns, skin cancer and cataract, usually lies way ahead in the future and does not affect the behaviour when you go to the beach or work in the garden.

Apart from the indicator for the UV-exposure and knowledge about sun habits and attitudes the data from the questionnaire has been used in press releases every year it has been sent out. For example:

2005: “Every second person burn during sun vacation!”
2006: “Every fourth Swede burn in sun when gardening!”
2007: “Risks well known, but sun sense practice rare (1 of 3)!”
2008: “Small improvement of sun habits! Sunburns down 6%”
2009: “Sun bed burns doubled among young persons!”

Conclusions

The Questionnaire will not be sent out annually but every second or third year. This gives the opportunity to make more specific studies that can focus on the potential risk groups. Since the risk behaviour is the “worst” in the youngest age group generally it would be interesting to see how the behaviour is for even younger ages. In order for the authority to give proper advice and information it is necessary to continue this kind of study.

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Role of modulation in the biological effects of radiofrequency radiation

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Abstract

The biological effects of modulated radiofrequency (RF) radiation have been a subject of debate since early publications more than thirty years ago, suggesting that relatively weak amplitude-modulated (AM) RF electromagnetic fields have specific biological effects different from the well-known thermal effects of strong radiofrequency energy. This discussion has been recently activated by the increasing human exposure to RF radiation from wireless communication systems. Modulation is used in all wireless communication systems to enable the signal to carry information. A previous review in 1998 indicated that experimental evidence for modulation-specific effects of RF energy is weak. This paper reviews recent studies (published after 1998) on the biological effects of modulated RF radiation. The focus is on studies that have compared the effects of modulated and unmodulated (continuous-wave, CW) RF fields; studies that have used only modulated or only CW signals are not included. While the majority of recent studies have reported no modulation-specific effects, there are a few interesting exceptions that warrant follow-up studies.

Location of glioma in relation to mobile phone use

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Abstract

The objective of our study was to evaluate whether the gliomas within the brain differ between mobile phone users and never-users. The energy absorbed from the radiofrequency (RF) electromagnetic fields of mobile phones depends strongly on the distance from the source. We would expect gliomas among users to be located nearer to the source of exposure i.e. mobile phones if such exposure increases the risk of gliomas. We used case-case analysis to evaluate whether location of the gliomas in the brain is related to the source of RF exposure. The study methods applied were novel and provide an improved approach to studying focal effects in the etiology of gliomas. By utilizing information on the tumor location instead of only amount of mobile phone use as in earlier studies, the case-case method enables focusing on SAR distribution of RF field. This offers a possibility to study biologically and physically more meaningful and refined hypotheses. The data consisted of 888 gliomas from seven countries with tumor mid-points assigned by neuroradiologists on a three-dimensional (1 x 1 x 1 cm) grid based on radiological images. The typical position of the mobile phone while used was assumed to be in the line from the external acoustic meatus to the corner of the mouth and distance was computed between this line and the mid-point of the tumor. The data analyses were made using unconditional logistic regression with distance as a categorical outcome (5 cm as a cut-point) in the case-case approach. In the case-case analyses never-regular users and those using mobile phone at the opposite side as the tumor had their gliomas nearest to the exposure line, but the differences were statistically non-significant. Our results do not suggest gliomas being located in excess in those parts of the brain with the highest exposure to the RF field of mobile phones.

Background

Mobile phones emit radiofrequency (RF) electromagnetic fields. RF fields have not been shown to be tumorigenic, but their biological and health effects have not been firmly established so far (Ahlbom et al. 2009). The energy absorbed from the RF fields of mobile phones depends strongly on distance from the source decreasing to one tenth in 5 cm of tissue (Cardis et al. 2008).

As the RF field emitted by the mobile phones penetrates the brain in a highly localized fashion, a local effect restricted to the part of the brain closest to the handset would be expected, if there is one. Instead of concentrating on crude indicators of phone use (such as self-reported number and duration of calls) as in most previous studies, utilizing the accurate tumor location enables focusing on the postulated distribution of the RF field within the brain, thus offering a biologically and physically more meaningful and more specific approach. However, only a few studies have assessed the location of glioma in relation to mobile phone use (Takebayashi et al. 2008, Hartikka et al. 2009), and these studies are based on small sample sizes.

The objective of the study was to evaluate whether gliomas among phone users occur preferentially in the areas of the brain having the highest RF exposure from the handset.

Methods

This study is based on data from seven European centers (nationwide studies in Denmark, Finland, Norway, Sweden; Bielefeld, Heidelberg and Mainz regions in Germany; Rome, Italy; and Thames region of Southeast England) within the Interphone study, an international collaborative case-control study, whose main objective was to assess whether mobile phones increase the risk of brain tumors (Cardis et al. 2007).

The study included 888 gliomas (61% of all glioma cases diagnosed during the study-period) diagnosed between September 2000 and January 2004 with tumor mid-point(s) assigned on a three-dimensional grid (1 x 1 x 1 cm). These mid-point(s) were defined by neuroradiologists, blind to the data on mobile phone, based on radiologic images.

The main exposure indicator in the analyses was the shortest estimated distance from the mid-point of the glioma to the putative source of exposure, i.e. typical location of the phone. The exposure line was assigned as a line from the external orifice of the ear canal to the corner of the mouth with the entire phone regarded as the source of exposure (as most phones have an integrated antenna with the whole body of the phone emitting an RF field). To avoid potential recall bias, distance was calculated to the nearest source of exposure on the same side as the glioma was located, irrespective of the patient's reported typical side of use.

Case-case analyses were carried out using unconditional logistic regression with distance between the mid-point of the glioma and the putative source of exposure (location of the phone) as a binary outcome (≤5 cm, >5 cm). Exposure indicators analyzed included regular use of mobile phone (regular use defined as ≥1 call/week for ≥6 months, use in the 6 months prior to glioma diagnosis excluded), cumulative call-time (divided into tertiles: 0.001-46 hours, 46-339 h and >339 h, with median of 133 hours and maximum of 20,000 hours), laterality (preferred side of use) and duration of use (cut-points chosen to correspond to those in previous studies: 6 months to 5 years, 5 to 9 years and 10 or more years of use). Never-regular use of a mobile phone was the unexposed reference category in all analyses. All analyses were adjusted for country, sex, age group and socioeconomic status.

Results

Information on mobile phone use was obtained from 98% of the cases, with 57% regular mobile phone users and 43% reporting no regular use. Laterality of use was known for 490 cases (99% of all regular users).

The mean distance between the tumor mid-point and the phone (for all gliomas, 6.25 cm) did not vary substantially by the indicators of mobile phone use. The mean distance was somewhat shorter among cases who had not used a phone regularly (6.19 cm) and those reporting the preferred side of use as contralateral to the tumor (6.29 cm) in comparison to regular (6.29 cm) and ipsilateral (6.37 cm) use. The mean distance was slightly longer for those with the highest cumulative call-time or those having used mobile phone for more than ten years, but the differences were not significant.

In the case-case analysis, using unconditional logistic regression with categorical distance as the outcome (≤5 cm, >5 cm), non-significantly decreased ORs for gliomas located within 5 cm of the presumed phone location were found in regular users compared with never-regular users (OR 0.87 (95% CI 0.63-1.20)). The odds ratios for higher exposure were below unity for all exposure variables in these analyses, indicating no excess in the highly exposed parts among regular vs. never-regular users.

Conclusion

Our results do not support the hypothesis of gliomas among users of mobile phones being preferentially located in the parts of the brain with the highest RF exposure from mobile phones. In the case-case analyses, gliomas among never-regular users, representing lower RF exposures, had a shorter mean distance between tumor mid-point and the presumed source of exposure than regular users. Consistent findings indicating no increased frequency of gliomas in the areas closest to the typical position of the mobile phone handset were observed with all exposure characteristics, e.g. cumulative call-time and duration of use.

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Hyperthermia-induced proliferative response in human cancer cell lines is counteracted by a 2.2 GHz pulsed signal

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Abstract
The present study describes the cell growth response of two human cancer cell lines, HepG2 and NB69, to 24-h simultaneous exposure to two physical agents: mild hyperthermia and 2.2-GHz, pulse-modulated, radar-like radiofrequency (RF) signals. The samples were sham-exposed or RF-exposed simultaneously inside two identical waveguides placed in a CO_2 incubator set at 37 ºC (standard temperature) or 38 ºC (mild hyperthermia). A complete discretized model of the setup was created for numerical dosimetry using FDTD software SEMCAD X. The average dose of RF radiation absorbed by the cultures was calculated to be subthermal ($\Delta T<0.1$ ºC). At the end of the 24 h treatment the cell growth was analyzed through Trypan blue exclusion and cytometry. At 37 ºC the NB69 line responded to the RF exposure with a consistent reduction in cell number (13.5% % below controls; $p<0.001$) together with slight but significant changes in the kinetics of the cell cycle. In contrast, the HepG2 cell growth and cell cycle were not changed after the RF treatment. The + 1 ºC thermal stimulus alone induced significant increases in the number of cells in both NB69 and HepG2 lines (17.9% $p< 0.05$ and 18.8%, $p < 0.01$ above controls at 37 ºC, respectively). This cytoproliferative, thermally-induced response was blocked in both cell lines by the simultaneous exposure to RF. Consequently, the results indicate that under standard temperature conditions the HepG2 line is not responsive to the cytostatic effect exerted by the RF exposure in NB69 cells, whereas, under thermal stimulation of cell proliferation, both lines showed a similar, cytostatic response to RF, likely to be mediated by changes in the cell cycle. Studies are in progress investigating the mechanistic and molecular basis of the herein described cellular response.

Introduction
The study of the potential hazards of chronic exposure to weak, radiofrequency (RF) electromagnetic fields in residential or occupational environments has been addressed by a number of epidemiological studies (see Hietanen, 2006; Hardell and Sage, 2008 for recent reviews) and experimental investigations (Garaj-Vrhovac and Orescanin,

2009; Xu et al., 2010). RF fields in the GHz range are of particular interest because of the rapid implantation of recent and emerging telecommunication technologies. International regulatory bodies like ICNIRP or the European Council (ICNIRP, 1998; EU Council Recommendation, 1999) have developed their standards for protection of the public and workers against non ionizing radiation on the basis that only RF doses inducing thermal increases $\Delta T \geq 1$ °C in the exposed tissues can be considered detrimental to humans. There is general agreement that the energy at which most RF telecommunication signals are emitted are too low to induce significant thermal increases in the tissues of potentially exposed individuals (Hietanen, 2006). Consequently, it has been widely assumed that under normal conditions in residential and occupational environments, the exposure to RF radiation at subthermal doses can be considered safe. This assumption is supported by data from a large body of experimental studies that have failed to detect potential harmful effects of weak RF signals on a number of in vitro biological models (Higashicubo et al., 2001; Chauhan et al., 2007; Prisco et al., 2008).

On the other hand, several experimental studies have reported effects on cell cycle control and apoptosis (Marinelli et al., 2004; Joubert et al., 2008), on gene expression (Remondini et al., 2006; Buttiglione et al., 2007), and tumorigenesis (Repacholi et al., 1997) in different biosystems exposed to RF at doses estimated to be subthermal. This data would provide limited support to some epidemiological results indicating that chronic exposure to weak RF fields could represent a risk factor in the etiology of a number of ailments, ranging from perceived electromagnetic hypersensitivity or neurological symptoms to neurodegenerative diseases or different cancer types (Rubin et al., 2006; Sage and Carpenter, 2009). Taken together, both blocks of epidemiological and experimental evidence seem to be indicative that different, presently undetermined factors may be responsible for the controversial data obtained so far. Even if subthermal mechanisms of RF interaction existed, at high power densities the thermal effects would prevail, leading to adverse consequences. At lower exposure levels biological effects have been reported but, due to limitations in modeling and dosimetry, the possibility cannot be ruled out that thermal mechanisms could also intervene in the biological response. Under the present circumstances there is general consensus that experimental studies using sensitive biological models are crucial to the study of the thermal/subthermal mechanisms underlying the bioeffects of weak RF fields. The present study investigates the cell growth response of two human cancer cell lines, HepG2 from hepatocarcinoma and NB69 from neuroblastoma, that have been reported to be sensitive to RF fields (Hernández-Bule et al., 2007; Trillo et al., 2009). The cells were exposed for 24 hours to a 2.2 GHz, pulse-modulated radar-like signal with high instantaneous amplitude and very low average power. Little is known on the biological interactions of this type of signals, and their potential relevance to the human health needs to be more efficiently addressed by the standards for radiation protection. Additionally, in order to determine whether a potential response to the RF exposure could be mediated by thermal phenomena, a set of experiments was conducted in which cell cultures were exposed to the RF treatment and/or incubated under mild hyperthermia (38 °C).

Material and methods

Exposure System

The schematic view of the RF exposure equipment is shown in Figure 1. The system has been described in a previous paper by Varela et al. (2010). Briefly, the system consists of: 1) A signal generator (MCL 15156, MCL Inc, USA) in the 2.0 – 2.5 GHz band, with specification for CW power ≥ 35 W and pulse duration ≥ 5 µs; 2) A modulator that controls the pulse duration and repetition rate; 3) A double rectangular waveguide applicator (95 x 45 mm section, 500 mm in length) connected to a high power matched load; 4) Ancillary equipment composed of a RF counter HP-5347A, a power meter HP-432A and a detector for pulse shape control.

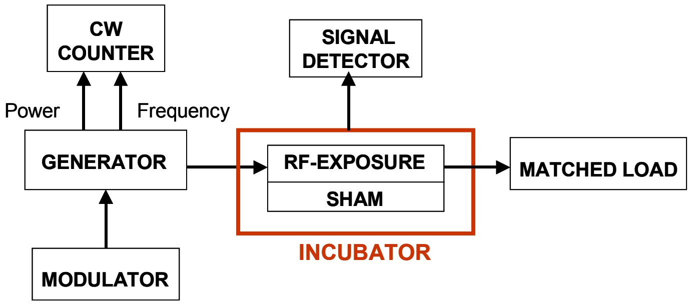

Figure 1.- Diagram describing the configuration of the System: The electromagnetic signal is generated by a signal generator sweep oscillator ranging 1 - 3 GHz. The RF applicator is located inside a CO_2 incubator and is composed of two waveguides, one for RF-exposure and another for sham-exposed control samples.

The RF applicator

During RF- or sham-exposure the samples are placed on a double-shelf Teflon support holding eight Petri dishes, inside each of the two wave guides (Fig. 2). Both guides have a lateral, copper hinged section and are closed by slotted short circuits equipped with fans, for temperature homogeneity and proper atmosphere exchange within the guides. A coaxial probe in the centre of the top wall waveguide is used for monitoring the parameters of the RF signal applied to the cells. Since all in vitro procedures have to be conducted in sterile conditions, the waveguides, sensors, connectors, holders and wiring were designed and built to withstand the 9-hour incubator routine for inner decontamination at 90 °C and RH > 95%.

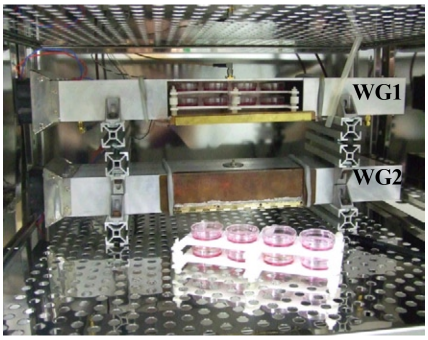

Figure 2. Waveguide pair for simultaneous RF- and sham-exposure within a CO_2 incubator. The trapdoor to the upper waveguide (WG1) has been opened to show the holder with Petri dishes. The 8-dish holder corresponding to WG2 is displayed outside the guide.

RF signal and dosimetry

The dosimetric procedures and results have been described in detail by Varela et al. (2010). Briefly, a 28 W (CW), 2.2-GHz, pulse-modulated (5 µs pulse duration, 100 Hz repetition rate) signal was applied to cell cultures grown in 35-mm Ø plastic Petri dishes, where the cells formed a monolayer on the bottom of the dishes. Dosimetric calculations were performed after a homogeneous model of the exposed materials, considering the dishes filled with 1.5 ml of lossy medium with relative dielectric constant $\varepsilon r = 77.5$ and electric conductivity $\sigma = 2.3$ S/m, and taking into account the presence of a meniscus at the surface of the culture medium. The Finite-Difference Time-Domain (FDTD) method was used to calculate the SAR distribution in the cell cultures through commercial software SEMCAD X (Schmidt & Partner Eng. AG, Zurich, Switzerland). The space-averaged SAR for CW exposure was calculated to be 46 W/kg. Since the samples were exposed to 5-µs pulses at a 100 pps repetition rate, the estimated time-averaged SAR would be 23 mW/kg. Considering that the experiments were conducted under strict temperature control and taking into account the damping thermal effect of the heat diffusion, we can assume that in our experiments the RF stimulus was applied at a dose well below those at which significant thermal effects can be expected.

Cell Culture

The human hepatocarcinoma cell line HepG2 obtained from the European Collection of Cell Cultures (ECACC, Salisbury, UK) was grown in Dulbecco's modified Eagle's medium (D-MEM, BioWhittaker-Lonza, Verviers, Belgium) supplemented with 10% heat-inactivated Foetal Calf Serum (FCS, GIBCO-Invitrogen, Paisley, Scotland, UK), 2 mM L-glutamine, and 100 U/ml penicillin - streptomycin. The human neuroblastoma cell line NB69, obtained from ECACC, was cultured in D-MEM medium supplemented with 15% heat-inactivated FCS, 4 mM L-Glutamine, 100 U/ml penicillin – streptomycin and 0.25 µg/ml amphotericin B. The media were always pre-heated at 37 ºC in order to avoid abrupt temperature changes that could induce cellular stress. In all experimental runs, four days prior to RF exposure, HepG2 or NB69 cells were aliquoted from a single parental flask to individual 35-mm Ø Petri dishes at densities of 9×10^4 or 7×10^4 cells per millilitre of medium, respectively, and grown in a humidified atmosphere with 5% CO_2 at 37 ºC.

Cell Growth and cell Viability Analyses

HepG2 or NB69 cells at a 60% confluence (day 4 after plating), were transferred in sterile conditions to the experimental incubators and grown inside/outside energized/unenergized waveguides. In all experimental runs, two cell culture samples were incubated simultaneously: A group of 8 Petri dishes was grown inside WG1 (Fig. 2) and an identical, 8-dish group was incubated inside WG2. In a set of experiments investigating the cellular response to mild hyperthermia, a third group of 8 control dishes was grown simultaneously inside an identical, separate CO_2 incubator, at a 37 ºC temperature. At the end of the 24-hour incubation the cells were detached from the dishes, resuspended in 1 ml of medium and studied for viability and proliferation. The number of alive and dead cells was determined through haemocytometer analysis of 50-

µl aliquots. Each sample was double-counted using the Trypan blue exclusion method. All experimental procedures and analyses were conducted in the blind.

Flow cytometry (FACS)

Cell samples grown for 24 hours with or without RF stimulation were harvested and fixed in 70% ethyl alcohol at room temperature for 4 hours. Subsequently, the cells were treated with RNAse-DNAse free (100 µg/µl; Roche) and stained with the fluorescent dye propidium iodide (IP, 20 µg/ml; Boehringer-Manheim) for 1 hour at room temperature. The relative fractions of cellular subpopulations in different phases of the cell cycle were determined through quantification of DNA content by flow cytometry, using Becton-Dickinson FACScan (FACScalibur, Becton Dickinson, Franklin Lakes, NJ, USA). Cell cycle parameters were determined by CellQuest 3.2 Software (Becton Dickinson Immunocytometry Systems, San Jose, CA, USA). Twenty thousand cells per sample were analyzed.

Statistical Analysis

Data was expressed as mean ± standard error (SEM) of at least three independent replicates, using GraphPad Prism software (GraphPad Software, Inc., San Diego, CA). The ANOVA test followed by two tailed Student's t-test was applied. Statistical significance was set at $p < 0.05$.

Results

Viability and cell growth response to the RF signal at standard temperature

Fig. 3 summarizes the cell growth response of the HepG2 and NB69 cell lines to the RF treatment at standard conditions of temperature (37 ºC). In the HepG2 line, at the end of the 24-h RF treatment no significant differences were observed with respect to sham-exposed controls. In contrast, NB69 cells responded to the same RF treatment with a consistent, statistically significant reduction (13.5% below sham, $p<0.001$) in the total number of cells (Fig. 3). This effect was accompanied with a modest, thought statistically significant increase in the proportion of necrotic cells in the exposed samples (15.9 ± 2.1%; $p<0.01$) when compared to the respective sham-exposed samples (spontaneous death rate: 13.9 ± 2.0%).

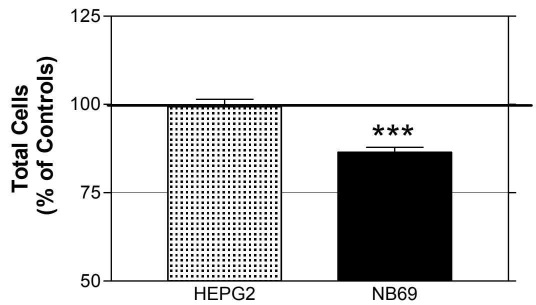

Figure 3. Comparative growth response after 24 h of RF- or sham-exposure. Means ± SEM of a total of 6 (HepG2) and 8 (NB69) independent replicates, with 8 RF and 8 sham dishes per replicate. Normalized data over the total cell number in the corresponding group of samples incubated simultaneously inside the non-stimulated waveguide (sham-exposed). In NB69, the overall effect was statistically significant ($p<0.001$) with respect to sham-exposed samples.

Cell Growth response to treatments with RF and/or mild hyperthermia

Figure 4 summarizes the cell growth response of the HepG2 and NB69 cancer lines when exposed to the RF signal under conditions of mild hyperthermia (38 ºC). At the end of the 24-h incubation at 38 ºC the sham-exposed samples of both cell lines showed significant increases in total cells (18.8% in HepG2, $p< 0.01$, and 17.9% $p< 0.05$ in NB69) when compared to controls incubated at 37 ºC. Such a temperature-induced effect was not observed when the thermal treatment was applied simultaneously with the RF exposure. When compared to their sham-exposed controls incubated at 38 ºC, the RF-exposed samples kept simultaneously in the same incubator showed significant reduction in the total cell number at the end of the 24-h treatment, both in HepG2 (13.7% below sham, $p<0.01$) and in NB69 (10.1% below sham, $p<0.01$).

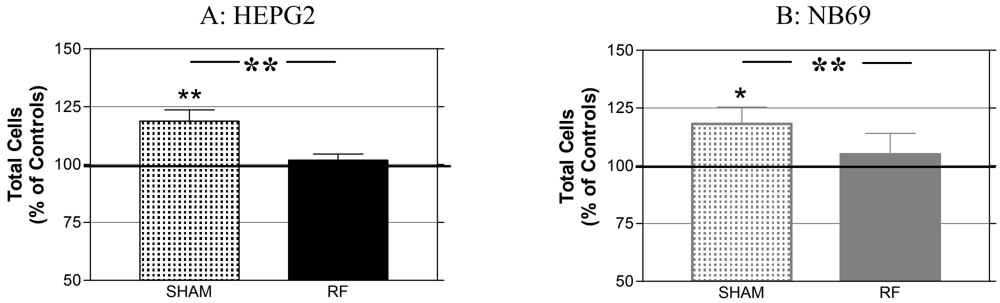

Figure 4. Growth response of HepG2 and NB69 cells after 24 h of simultaneous treatment with mild hyperthermia and RF- or sham-exposure. A total of 5 independent replicates were performed per cell line, with 8 dishes exposed to RF at 38 ºC, 8 sham dishes incubated at 38 ºC and 8 controls growth at 37 ºC, per replicate. Means ± SEM. Data normalized over the total cell number in the corresponding group of controls incubated simultaneously at 37 ºC.

Changes in the kinetics of the cell cycle in response to the RF stimulus

FACS analyses were conducted in order to investigate whether the differential growth responses in HepG2 and NB69 could be due, at least in part, to RF-induced alterations in the cell cycle regulation. No significant changes were observed in the distribution of HepG2 cells in the different phases of the cycle after RF- exposure (data not shown). In contrast, the NB69 line responded to the RF treatment with slight but consistent and statistically significant increases in the percent of cells in G0/G1, both under standard temperature and mild hyperthermia (6% and 3.4 % over controls, respectively, $p < 0.05$; Fig. 5). Also, a significant increase in the percent of cells in G2/M phases (9% over controls, $p < 0.05$) was observed when RF was applied at standard temperature, but not at 38 ºC.

As for the influence of the thermal treatment alone, it induced significant changes in the cell cycle progression of both lines; resulting in shortening of the S-phase in samples incubated at 38 ºC (approximately 30% below controls at 37 ºC, $p < 0.05$). The simultaneous treatment with RF blocks this thermally-induced response, and only small but significant increases in the G0/G1 phases were observed (Fig. 5B).

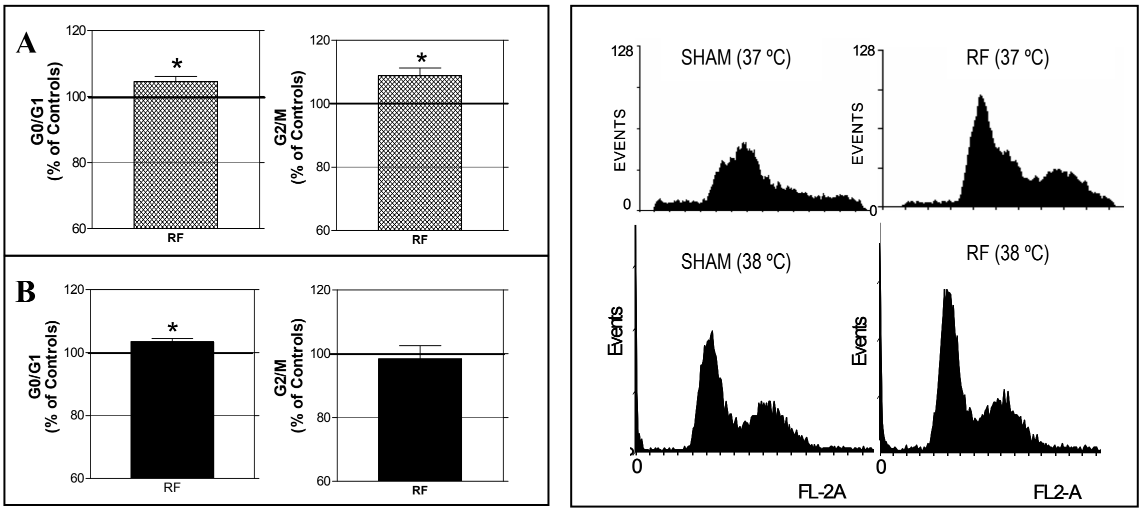

Figure 5. Left: Changes in the percent of NB69 cells in G0/G1 and G2/M phases of the cell cycle in response to RF exposure. A total of 4 independent replicates with 8 RF and 8 Sham dishes per replicate were carried out at (A) standard temperature (37 ºC) and (B) Mild hyperthermia (38 ºC). The data (Means ± SEM) are normalized over the respective sham-exposed samples. Right: Four flow cytometry histograms, representative of RF-exposed and Sham-exposed samples of the line NB69, incubated at 37 ºC or 38 ºC.

Discussion

The biological effects of low power or short-pulse RF signals that do not induce significant hyperthermia in the exposed tissues have not been investigated sufficiently. A more complete knowledge of the non thermal or subthermal effects of the RF radiation on biological systems is of utmost interest from two biomedical points of view. On the one hand, an increasing number of medical applications based on RF treatments have been developed in recent years. A better understanding of the biological responses to RF involving phenomena other than thermal could be useful to the development of new treatments, including co adjuvant therapies in oncology, traumatology or pain relieving, among others. Furthermore, it is widely accepted that hyperthermia can enhance the sensitivity of cells to radiation and drugs, and mild hyperthermia (by 1–2 °C) has been reported to induce heat-shock proteins, increase biophylaxis and immunocompetence, prevent stress and fatigue, and reduce depression or anxiety, which could contribute to diminish adverse reactions to chemotherapy and potentiate its antitumor effects (Skitzki et al., 2009; Yamada et al., 2009).

On the other hand, the level of exposure to environmental RF emissions has progressively increased in developed countries. A positive correlation between RF exposure and tumorigenesis has been suggested by a number of epidemiological studies (Hardell et al., 2008; Sadetzki et al., 2008; Khurana et al., 2009). These results have received limited support from experimental data indicating that RF fields can induce DNA breaks (Garaj-Vrhovac and Orescanin, 2009), chromosome aberration (Mazor et al., 2008), as well as alteration of gene expression (Remondini et al., 2006; Zhang et al., 2008). Despite of the fact that these in vivo and in vitro results could be indicative of possible damaging effects of RF fields, other experimental results do not support such a possibility. Thus, whether or not the reported biological responses may be induced through subthermal phenomena, is presently a controversial issue that needs to be elucidated.

Altered cell proliferation is among the most sensitive phenomena currently used to study the cellular response to environmental carcinogens. However, the available evidence on RF effects on cell proliferation is scarce and the results are conflicting (Higashicubo et al., 2001; Pérez-Castejón et al., 2009). The present study addresses the hypothesis that RF signals at subthermal doses could influence cancer progression by increasing or accelerating cell proliferation in two human cancer cell lines. We examined cell proliferation following exposure to RF radiation at standard incubation temperature (37 ºC) or at mild hyperthermia (38 ºC), and analyzed comparatively the cell responses induced by a 2.2-GHz pulse-modulated, S-band radar-like RF signal with high instantaneous amplitude and very low average power, and by a 1 ºC hyperthermia caused by the corresponding increase of the incubation temperature.

In the NB69 line the RF treatment at standard temperature resulted in significantly decreased cell number (13.5% below sham-exposed samples), accompanied with significantly increased proportions of cells in the phases G0/G1 and G2/M of the cell cycle. Also, a modest but consistent and statistically significant increase in the rate of necrosis (2% over sham) was observed in the NB69 line; however, this subtle cytotoxic effect would have little impact on the overall cell growth rate. The other human cancer line tested, HepG2, did not respond to the 24-h exposure to the same RF treatment when it was applied at standard temperature of 37 ºC. However, under +1 ºC hyperthermia, which induced significantly increased cell growth in sham-exposed NB69 and HepG2 cultures (about 18% over control samples at 37 ºC), both cell lines responded to the RF signal with significant ($p < 0.01$) reductions in cell growth, (10.1 % below sham-exposed in NB69 and 13.7 % in HepG2). In NB69 this RF-induced reduction in cell growth at 38 ºC was accompanied with a modest, though significant increase in the fraction of cells in phases G0/G1.

A similar RF effect on cell proliferation was observed by Velizarov et al. (1999) in transformed human epithelial amniotic cells exposed to a 960 MHz microwave radiation. These authors reported that at incubation temperature of 39 ºC a significant reduction in cell proliferation occurred in the exposed cells when compared to non-exposed (control) samples. However, in contrast to our results, the treatment with mild hyperthermia alone did not change significantly the proliferation of the amniotic cells with respect to control samples incubated at 37 ºC. Taken together, the results of both studies indicate that, when administered alone, mild hyperthermia induces different responses in different human cell types; whereas exposure to subthermal doses of RF signals in the GHz range could elicit a common, antiproliferative response in different, thermally stimulated cell lines.

Additionally, in both NB69 and HepG2 cell lines the RF treatment neutralized the thermally-induced proliferative response observed after incubation at 38 ºC. This result could explain the apparent lack of response reported by other authors in cells exposed to relative high, thermal doses of RF radiation (see, for instance, Takashima et al., 2006). It is possible that in some of those studies a mild thermal effect had occurred in fact, that would be counterbalanced by an electromagnetically-induced response. In other words, the thermal electromagnetic signal would have induced two opposite, simultaneous responses resulting in a null or non detectable overall effect.

In conclusion, a 24-h treatment with subthermal doses of 2.2 GHz, pulse-modulated RF radiation elicited different responses in two human cancer cell lines. The NB69 line was proven responsive to the RF stimulus, exhibiting significant reduction in the number of cells, both under standard incubation temperature (37 ºC) and under mild hyperthermia (38ºC). Such antiproliferative effect was associated to cell cycle alterations. The HepG2 cell line was not responsive to the RF signal when applied at standard temperature conditions. However, at a temperature of 38 ºC, the RF stimulus induced in HepG2 an antiproliferative response similar to that in NB69, the cultures reaching growth rates equivalent to those of controls incubated at 37 ºC. Additionally, the thermal treatment alone induced significant shortening of the S-phase in both lines, which resulted in significant acceleration in the cell cycle progression and increased cell number when compared to samples incubated at 37 ºC. The simultaneous treatment with RF neutralized this thermally-induced response, indicating that the electromagnetic and the thermal stimuli might exert opposite effects on cell proliferation and cell cycle progression. As a whole, the antiproliferative responses elicited in two different human cancer cell lines are not supportive of the hypothesis that repeated exposure to weak, RF electromagnetic signals in the GHz range can exert carcinogenic effects on transformed or initiated human cells. Further research would be necessary in order to elucidate whether or not the described, RF-induced cytostatic effects might be relevant to the potential development of emerging, electromagnetic-based therapeutic strategies.

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Exposure of the French population to 50 Hz magnetic fields: EXPERS study

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Abstract

To study the exposure of the French population to 50 Hz magnetic fields (MF), two samples (children and adults) representative of this population were created. Each person wore an EMDEX II measuring and recording the MF to which he/she was exposed during 24h, and has progressively filled in a timetable and a questionnaire with information about themselves and their homes. When returning the meter, the pollster recorded the GPS coordinates of their homes.

In total, 978 series of MF were validated for children and 1054 for adults. The arithmetic and geometric means observed were respectively 0.09 and 0.02 µT for children and 0.14 and 0.03 µT for adults.

Introduction

The magnetic field (MF) at extremely low frequency (ELF) have been suspected, for around 30 years, to be responsible for several pathologies in humans, more precisely, childhood leukemia (Wertheimer et al., 1979). In 2001, the International Agency for Research on Cancer (IARC) classified ELF MF in the category II-B ("possibly carcinogenic to human").

The last collective assessment by international expert groups (WHO 2007, SCENHIR 2009) concluded that the last major questioning concerning ELF MF is the statistic correlation observed in several meta-analysis between the increase of childhood leukemia risk and a MF exposure higher that 0.4µT in means over 24h (Ahlbom et al., 2000), without any causal relation.

The exposure of the French population to these types of fields is known only very approximately. In 2007 the Health Ministry initiated a study about the exposure to 50 Hz MF of a representative sample of the French population (1000 children from 0 to 14 years and 1000 adults of 15 years and over). We present here the results of this study called "EXPERS" for EXposure of the PERSson.

Material and methods

Recruitment of the volunteers

One of the problematics linked to this study was to create a sample of 1000 children and 1000 adults based on random sampling method. For this a call for tender was launched and MV2 Conseil was chosen to conduct this work and collect the data.

This phase of data collection was conducted in 3 campaigns (February-April 2007, October 2007-April 2008 and October 2008-January 2009).

MV2 created a database made up of a file with 95 362 phone numbers taken in a totally random way from the general file of phone numbers attributed in France, except professional phone numbers. The selection of individuals was based on these numbers, respecting the distribution of the population in the 22 regions[1] of metropolitan France according to the 2006 census (www.insee.fr).

Data collection

Each volunteer recruited wore an EMDEX II (Enertech, USA) measuring and recording every 3 seconds the 40-800 Hz MF during 24h. Figure 1 gives an example of measurements recorded by a volunteer.

Among the equivalent models on the market, we chose this meter because it enables the distinction between the 50 Hz component and the harmonics, which is useful in analysing the different sources of MF. Moreover, we have checked that it was not disturbed by electromagnetic fields emitted by GSM. (Magne et al., 2006).

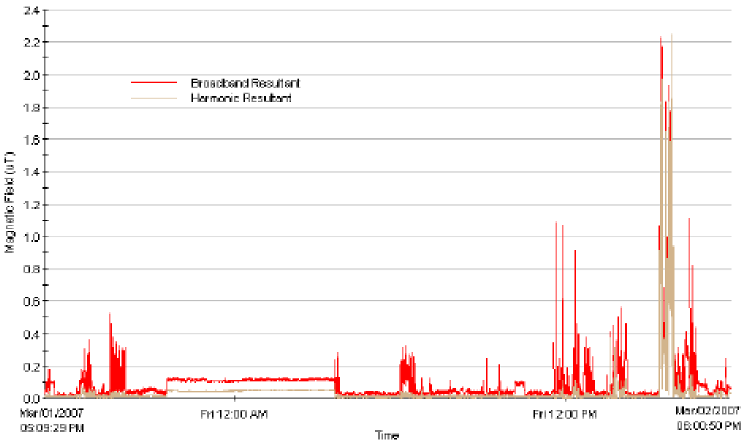

Figure 1. example of MF record for a volunteer.

Each volunteer progressively completed a timetable in which he noted his activities, with starting and ending time, the location and the electric appliances used.

When returning the meter, the pollster filled in with the help of the volunteer a questionnaire containing information relative to the volunteer (age, sex, profession, …), his home (year built, number of years spent in the home, number of people living in the home, energy type, heating method of home and water, …)

[1] France is divided into 22 regions and 96 departments.

He also noted the GPS coordinates at the volunteer's home front door. These coordinates were transmitted to ERDF (the French electricity distribution network operator) and RTE (the French electricity transport network operator) in order to identify all the electric networks close to each home. The criteria of identification of electric networks around homes are given in another communication (Magne et al., 2010). The results given here concern overhead and underground high voltage networks, from 63 to 400 kV, and the electrified train networks (data June 2009).

Description of the database

The number of volunteers having participated in the MF measurements was 2 148 (2.25 % of the phone numbers called). In total, the information relative to 2 048 people (989 children and 1 059 adults) was validated by MV2 Conseil.

When looking at the distribution of the database created, we notice that 11 departments out of 96 are not represented (note that we did not fix any quota per department).

After checking the compatibility of MF series and timetables, 16 series were deleted. The sample analysed was composed of 978 children and 1054 adults.

Results

Descriptive analysis

The arithmetic means (MA) and geometric means (MG) observed are respectively 0.09 and 0.02 µT for children and 0.14 and 0.03 µT for adults. Figure 2 gives the distribution of MA for both populations.

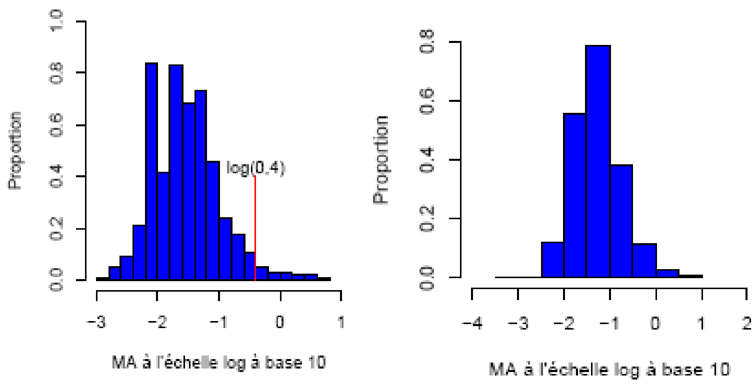

Figure 2. distribution of MA over 24h (children/adults).

On the whole, 3.1% of children observed an arithmetic mean higher than 0.4 µT. Two of them observed a geometric mean higher than this value.

When considering the exposures outside the period of sleep, the mean exposures for children are 0.05µT (MA) and 0.02 µT (MG). Eleven children (1.1%) have recorded a MA higher than 0.4 µT.

The mean exposure for adults are 0.10 µT (MA) and 0.03 µT (MG).

A comparison of mean exposures was performed for both populations with the help of rank tests. Figure 3 gives an example.

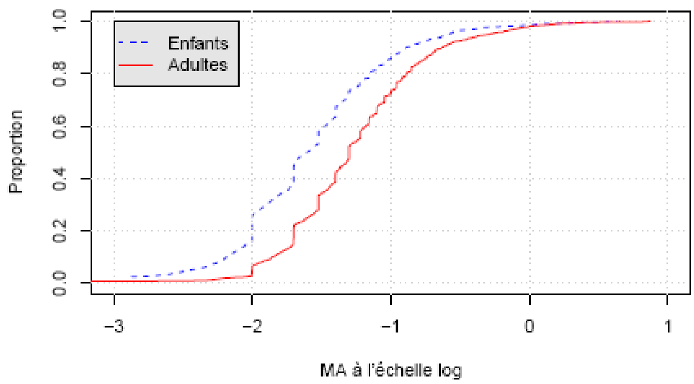

Figure 3. distribution function of 24h MA for children and adults.

The results show that:

- children are less exposed than adults,
- children are more exposed at home than outside, while it is the opposite for adults,
- at home, both populations are more exposed during the day than during the night (it is the opposite for adults in MG),
- regarding electric networks:
  - for both populations, the mean exposure (at home and over 24h) are higher for the volunteers living close to electric networks than for those living far away from these networks,
  - the MA (at home and over 24h) are not different for children living close to high voltage networks and for those living close to train networks,
  - the MG at home (with and without sleep period) are higher for children living close to high voltage networks than for those living close to train networks (it is the opposite for MG over 24h),
  - the MA and the MG (at home and over 24h) are not different for adults living close to high voltage networks and for those living close to train networks,
- people living in Ile-de-France are more exposed than in the other regions.

Characterization of the exposures

We are seeking to characterize the mean exposures in function of the data collected. A study on the data of the first phase has shown that there exists relationships of dependence between the mean exposures and some explicative variables. In order to characterize theses structures, we chose firstly linear models. These models gave very low levels of explained variance. This can be explained by the fact that the studied variables are not the only ones to influence the exposure, or by the fact that the relationship is not linear. For further analysis, we thus decided to use non parametric multidimensional models.

The factors identified as influencing the mean exposure are:

- the age,
- the density of population of the department,
- to have placed the EMDEX II close to a clock radio (for 24h exposure),
- living in a city of more than 2 000 inhabitants,
- living in a building,
- the presence close to the home of overhead power lines or of electric train networks,
- the time spent on a computer,
- the time spent in shopping centres,
- the time spent on train transports,
- the time spend in non-electric transport,
- the time spent at school.

All are not factors influencing both populations, nor both means, nor exposure over 24h and outside period of sleep.

The level of variance explained remains very low (between 10 and 30 %). The models obtained are thus non predictive.

Search of exposure classes

We are seeking to separate the population studied in several classes, by regrouping those who have the most similar exposures. Each series of MF is described by 7 indicators: maximum value, 3rd quartile, MA, MG, median, standard deviation and RCMS (Rate Change of Metric Standardized, indicator measuring the temporal variability of the signal). A hierarchic classification is then applied on these indicators centered and reduced.

Figure 4 gives an example of dendogram of classification. For each population and each scenario (24h or outside period of sleep), we chose 3 classes of exposure.

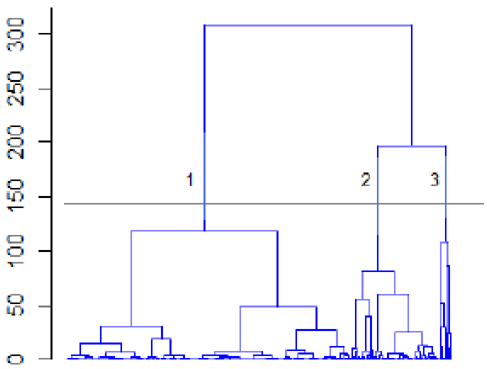

Figure 4. example of classification (exposition aver 24h for adults).

After we used a logistic regression in order to identify the variables leading a subject to belong to the most exposed class. The results depend on the scenario, type of mean and population considered.

The main factors which increase the probability to belong to the most exposed classes are:
- to have placed the EMDEX II close to a clock radio (for 24h exposure),
- the presence close to the home of overhead power lines or of electric train networks,
- living in a city of more than 2 000 inhabitants and the density of population of the department (for adults),
- the time spent on train transports,
- the time spent in shopping centres (for adults).

The main factors which decrease the probability to belong to the most exposed classes are:
- the time spent in non-electric transport,
- the time spent at school.

Discussion

The mean time to recruit one volunteer was 70 minutes. The recruitment of children was even more difficult than those of adults. This led to modifying the recruitment protocol by privileging children from the second campaign.

Tests of homogeneity have been performed in the 11 departments without subjects comparing to the corresponding regions: we looked at if the probability of selecting an individual in these departments is the same as the probability of selecting an individual in the corresponding regions. The statistical tests show that the fact of having no volunteers in these 11 departments is completely by chance.

One the one hand, the analysis of volunteer profiles does not show any difference in the ratio boy/girl in the children compared to the French population. On the other hand, women are more represented than men in our sample. In the same way, the profile of ages show a deficit of children under 6 years of age, and a surplus of 35 to 50 year old adults. This can be explained by the modification of recruitment protocol: we suggested to the adult to record the measurements at the same time as the child, often it was probably the mother who accepted.

On the whole, 3.1% of children observed an arithmetic mean higher than 0.4 µT. Two of them observed a geometric mean higher than this value. These percentages are higher than what is given in the literature, that is why we have tried to explain these high exposures. This phenomena has appeared during the first measurement campaign. We have then noticed that the high exposures corresponded to signals with values which were sometimes high (several µT) and constant during the night, and a ratio of harmonics around 1/3. These signals corresponded to magnetic fields emitted close to clock radios. Additional investigations into clock radios have shown that:
- the MF level vary strongly from one model to another,
- the MF source is the transformer, which is located in the clock radio or deported in the socket,
- the level of MF decrease very rapidly with the distance,
- at 50 cm, the magnetic field emitted by the clock radio is negligible.

So the highest 24h measurements can be explained by the presence of clock radios during the night. But are these measurements representative of the exposure of the person? In order to avoid measuring MF in contact with clock radios, we asked in the following campaigns to respect a distance of 50 cm between the EMDEX II and any electric appliances during the night, and we asked the volunteers whether they had respected this requirement. All did not do it!

When considering the exposures outside the period of sleep, 11 children (1.1%) have recorded a MA higher than 0.4 µT. This result is coherent with the literature.

The 24h measurements may thus include a measurement bias which overestimates the exposure. That is why we studied the exposure over 24h and the exposure outside the period of sleep.

Conclusions

Beyond the difficulties encountered by MV2 Conseil to perform the measurements, we can remember that 3.1% of the children have observed an arithmetic mean higher than 0.4 µT. The main exposure sources for these high exposures are clock radio (the real exposure of the person was overestimated). When looking at the exposures outside the time of sleep, 1.1% of children have observed an arithmetic mean higher than 0.4 µT. This value is consistent with the literature.

Children are less exposed than adults. An explanation could be the fact that they move less during school periods.

With the descriptors calculated on the series, each type of population is divided in 3 classes of exposure. The factors identified in terms of higher means or belonging to most exposed classes are mainly having put the EMDEX II close to a clock radio, having his home close to overhead power lines or to electric train networks, the time spent on train transports, in commercial centres, on a computer or watching television, living in a city of more than 2 000 inhabitants or in a building. These factors depend on the population considered (adult or children), the type of mean (arithmetic or geometric), and the scenario (over 24h or outside period of sleep).

The analysis of mean exposures has shown than the variables retained do not alone allow to characterize these means.

The work presented here will be continued by including the presence or not of distribution lines and substations. This information could improve the characterization of mean exposures in terms of explained variance.

Another possible use of these data is the validation of physical models of assessment of magnetic fields.

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Impact of Post-Processing in human body dosimetry exposed to 50 Hz magnetic field

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Abstract
In order to comply with the requirements on the exposure of workers to electromagnetic fields, numerical simulation is necessary because the currents induced in the body are very difficult to measure. Induced currents are computed by solving Maxwell equations by means of different numerical methods (FEM, BEM, FIT, etc...) operated on a panel of models of human body which are available with variable geometrical sharpness. To be able to compare the numerical accuracy of these models, we need to understand, for each numerical method, the impact of the quality of the meshing on the reliability of the maximal computed value of the induced current.

In this study, we apply four quality criteria on four types of methods working in the ELF domain, applied to one single mesh offering medium sharpness. By analyzing the maximal induced currents in a particular exposure case to vertical induction, we found that the efficiency of our criteria is highly dependent on the method of resolution in parts of the body which are too coarsely meshed.

Introduction
The European Directive 2004/40/CE dealing with electromagnetic fields for workers defines two reference values: the AV (Action Value) which is a warning threshold, and the ELV (Exposure Limit Value) which have not to be exceeded for biological reasons. We can compare both values via simple models such as on Fig. 1.

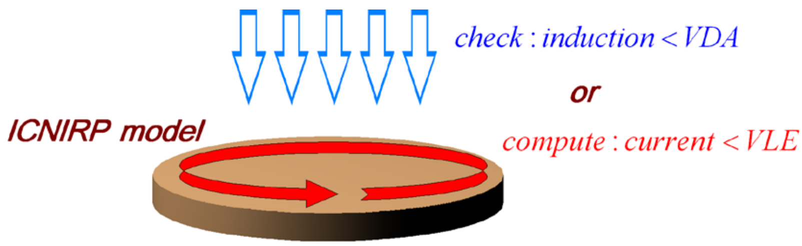


Fig. 1. Simple ICNIRP exposure model illustrating AV and ELV.

The magnetic and electric field can be easily measured *in situ* and then compared with the AV which is $500\mu T$ at $50 Hz$ for the magnetic induction and $10 kV/m$ for the electric field, while the ELV corresponds to the RMS value of the current density averaged over $1 cm^2$ surface normal to the current vector, and is considered only in the CNS (Central Nervous System). To evaluate such a value inside the body, and compare it to the $10 mA/m^2$ threshold, not only a numerical simulation is required, but the computation must ensure reliable results all over the CNS organs, which depends on the association of a robust resolution method and a anatomically realistic modelling of the body (Ducreux et al 2009).

The numerical methods to solve Maxwell equations in ELF domain

In order to so solve the Maxwell equations for the induced currents in the human body at 50 Hz, we considered four different formulations : two of them are general purpose FEM (Finite Element Method) for industrial applications of the QSA domain (Quasi-Static Approximation), the third one is a dedicated method derived from the first one, and the last one is derived from the FIT (Finite Integration Technique).

Formulation 1: general purpose magnetic FEM “A-Phi”

The unknowns are the magnetic vector potential $\mathbf{A}$ in the entire space and the scalar electric potential φ in the body. The computer code we used for that formulation is *Code_Carmel3D* (Henneron et al. 2007), which is well fitted to magnetic hysteretic media in electrical machines accounting rotation and circuit coupling. That method is the most ancient and the most practised in ELF domain of electrotechnics.

Formulation 2: general purpose electric FEM “T-Omega”

The unknowns are the electric vector potential $\mathbf{T}$ in the body and the scalar electric potential Ω in the whole space. The tools we have selected for that formulation are *Code_Carmel3D* and Maxwell3D (Miller et al. 2008). That second method is comparable with the first one, while in this context it seems to be more accurate, as it ensures strongly the normal continuity of the vector current at interfaces (which is much more interesting than the analogous condition on the magnetic flux density imposed by the A-Phi formulation).

Formulation 3: a human body dedicated magnetic FEM “Phi-A”

In case of human body, the magnetic vector potential $\mathbf{A}$ depends much more on the incident field than on the induced currents. So it may be sufficient to compute the electric scalar potential φ in the body in response to an incident magnetic potential vector $\mathbf{A}$ associated with the incident field, considered as a volume source term. The resulting advantage is a very cheap computation. The code we used for that formulation is an academic one, based on the library Getfem++ (Renard et al 2010).

Formulation 4: a « FEM-like » Finite Integration Technique

Generally, the FIT is devoted to the high frequency domain, because it solves the complete Maxwell equations, and moreover it can work directly with short wavelengths on the high resolution voxel-phantoms. Recently, this method was made usable with linear tetrahedrons, within non-structured meshes, by running the CST-EM Studio code

(Weiland et al 2008), and in addition, a low frequency domain solver is now proposed, so that the FIT can easily compete in our human body ELF benchmark.

Benchmarking our formulations on a simple meshing test case

First we compared the formulations on a very simple problem, i.e. an non-magnetic ellipsoïd with an homogeneous conductivity of $0.2\ \mathrm{S/m}$, subjected to an uniform $50\mathrm{Hz}$ incident magnetic induction of $500\mu\mathrm{T}$ directed along the small axis shown on Fig.2.

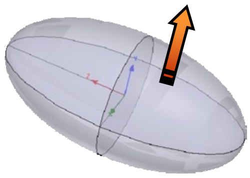


Fig. 2. Configuration of the test.

For such a system, we easily obtain the following analytical expression of the maximal current $J_{\max}$:

$$J_{\max} = \omega B \frac{ab^2}{a^2 + b^2}$$

Where a and b are the major axes, B is induction and ω its angular frequency. Then we can evaluate the impact of the mesh refinement on the local maximal value of the induced current computed with the four formulations. For that purpose, we treated different axis dimensions for the two types of meshes of Fig. 3.

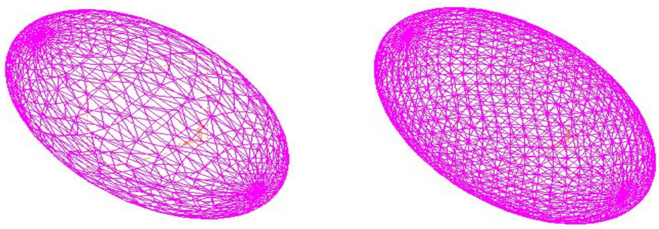


Fig. 3. Coarse mesh (3000 tetras) and fine mesh (15000 tetras).

As we can see on Table 1, the refinement of the mesh fit the formulations when they are of the same family (i.e. 1 with 4 and 2 with 3) but it cannot make all the four formulations converge to the exact result.

Table 1. Maximum induced current density ($\mathrm{mA/m^2}$) : case of the 60x30 cm ellipsoid.

	Formulation	Coarse mesh	Fine mesh
1)	Carmel ($A-\varphi$)	8.29	5.30
2)	Carmel ($T-\Omega$)	5.46	5.18
	Maxwell3D ($T-\Omega$)	4.87	5.18
3)	Getfem++ ($\varphi - A$)	7.50	5.30
4)	CST (FIT)	6.04	5.19
Analytical solution		**5.30**	

Benchmarking our formulations on a particular human body model

The « formulation sensitivity » that we found in the above elementary test is much more significant on a realistic human body mesh such as shown on Fig. 4.

Fig. 4. Studied human body mesh (provided by Ansoft).

This model contains around 100000 tetras (70% for the body) and 21 organs for which we can see on Fig. 5 that their size, shape and conductivity are ill-assorted.

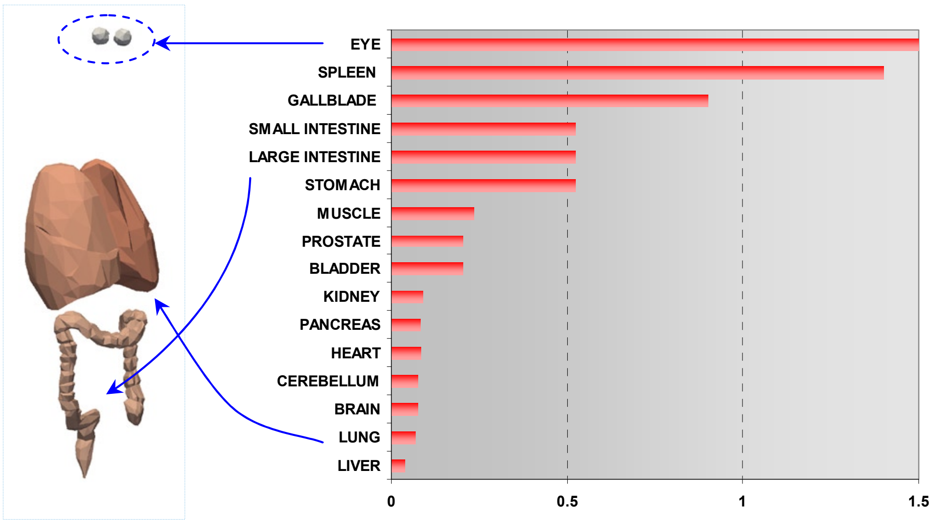


Fig. 5. Some components of the human body model and their conductivity (S/m).

Assuming a uniform vertical incident field of $1\,\mu T$ at $50 Hz$, we computed the maximal induced current in each organ obtained by the four formulations. The results show important discrepancies (see “EYE” on Fig. 6), which are not located into the high conductivity organs only (see “PROSTATE” and “PANCREAS” on Fig. 6). In all the conductive organs, the two general purpose formulations 1 and 2 are relatively close together compared to the derived formulations 3 and 4. Nevertheless, Fig. 6 shows that formulation 1 and 2 differ in the muscle, that suggest some meshing effect (muscle has a complex shape because it is defined as the whole body less all other organs).

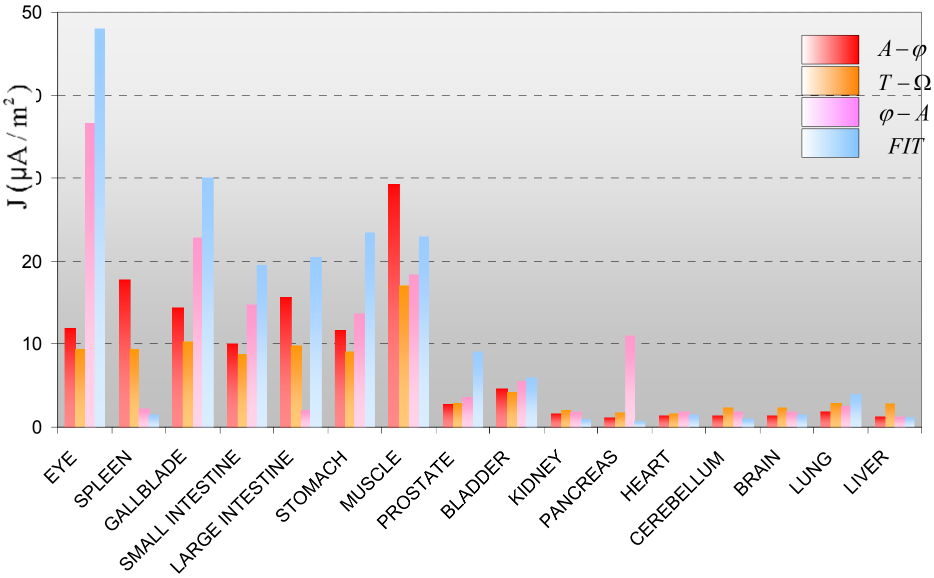


Fig. 6. Maximum induced current density in organs.

Post-processing of the results using mesh quality criteria

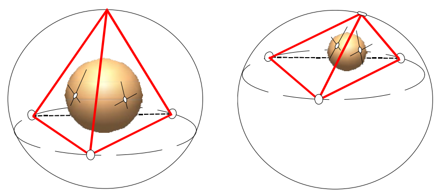
In order to quantify the impact of the mesh quality and of the formulation on the reliability of the maximum computed current, we can use local error estimators. As the computation provides one current value for each tetrahedron, we associate the local error with a so called "quality criterion" assigned to each tetrahedron, which can be estimated "a priori" or "a posteriori" (i.e. before or after the computation) – see table 2.

Table 2. "A priori" and "a posteriori" error estimators.

Error estimator	a priori	a posteriori
mesh dependant	geometrical	statistical
formulation dependant	spectral	energetical

A priori geometrical criterion

This very well known criterion is based on the shape of the tetrahedron. It is build with the ratio of the radius of in-sphere R_{int} to the radius of circumscribed sphere R_{ext} as shown in Fig. 7. So the quality q is unity for an equilateral tetrahedron and is zero for a flat one (*sliver*).



$$q = 3\frac{R_{int}}{R_{ext}}$$

Fig. 7. Example of a "good" and of a "bad" tetrahedral geometry

A priori spectral criterion

This criterion is based on the numerical conditioning of the elementary matrix. In that sense, it includes some physical properties of the problem, such as the static or dynamic nature of the magnetic response. Hereafter the elementary matrix for the $A-\varphi$ formulation is considered (the dynamic term is the one which depends on the skin depth δ). N_A and N_φ are respectively the edge and nodal shape functions for tetrahedra:

$$\mu M(\delta) = \begin{bmatrix} (rotN_A)^2 & 0 \\ 0 & 0 \end{bmatrix} + \frac{i}{\delta^2} \begin{bmatrix} N_A^{\ 2} & \frac{1}{i\omega} N_A^T gradN_\varphi \\ \frac{1}{i\omega} gradN_\varphi^T N_A & \frac{1}{\omega^2}(gradN_\varphi)^2 \end{bmatrix}$$

Once the above matrices are gauged, we easily compute the non-zero eigenvalues (there are 3 non-zero eigenvalues for the first static matrix and 6 for the second dynamic matrix). Then we deduce the condition number ρ and draw the static and dynamic quality criteria q by the following classical form :

$$q = 1 - \left(\frac{\sqrt{\rho} - 1}{\sqrt{\rho} + 1} \right)^2$$

We found that for the $A-\varphi$ formulation, the static and dynamic criteria are very similar except for very large distortions for which the static criterion is slightly more selective. Moreover, after a scaling of 30% accounting that “perfect regular” shape is equilateral for geometrical quality and rectangular isosceles for spectral quality, one observes (Fig. 8) that both “a priori” criteria are quite similar. This fact can be explained by arguing that for flat elements have a nearly singular jacobian matrix, which is used for computing the derivatives of shape functions. Therefore, the more the jacobian matrix is close to be singular, the more the resulting elementary matrix will be ill-conditioned.

Fig. 8. Bad elements selected by “a priori” criteria : geometrical (a), spectral (b) for the A-Phi formulation.

A posteriori statistical criterion

This criterion is based on the distribution of the values computed for the entire mesh. Generally, it can be assumed that the few last percentiles of the distribution are non-significant because these extreme values are due to a poor numerical behaviour of the badly shaped elements. This very simple approach works in most cases.

As expected, the rejected poor elements depend on the formulation. The Fig. 9 compares the last deciles of the current obtained with both general formulations $A-\varphi$ and $T-\Omega$, and exhibits an exploding divergence for the last 2%, what implies that aberrant elements are sensitive to the formulation : on Fig. 10 (a), the tetrahedrons filtered with $A-\varphi$ are mainly located in the interior of the body, so they are supposed to be bad elements, while on Fig. 10 (b) the same filtering with $T-\Omega$ eliminates some tetras on the surface of the trunk, although they seem to be "true" high current elements.

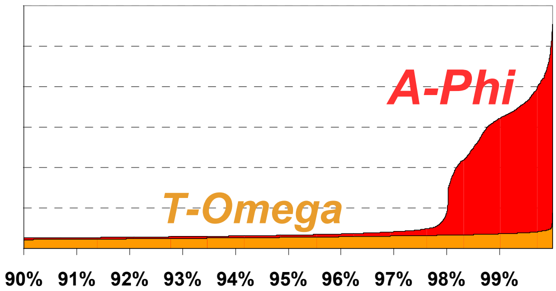


Fig. 9. Diverging last decile of currents distribution.

A posteriori energetical criterion

This criterion is based on the detailed comparison of both general formulations $A-\varphi$ and $T-\Omega$, which are supposed to bound the exact solution (Ren et al, 2002). A measure of the distance between both solutions may be evaluated for each tetrahedron via some energetic normalisation of the residue of constitutive laws, either conductive or magnetic, such as defined in last column of Table 3.

Table 3. Combination of solutions for the constitutive law checking.

formulationr	solution field	derived field	constitutive law
$A-\varphi$	$B_{A-\varphi} = rot\ A$	$E_{A-\varphi} = -i\omega A - grad\varphi$	$J_{T-\Omega} = \sigma E_{A-\varphi}$
$T-\Omega$	$J_{T-\Omega} = rot\ T$	$H_{T-\Omega} = T - grad\Omega$	$B_{A-\varphi} = \mu H_{T-\Omega}$

We can adjust that norm so that the quality criterion on each element returns unity if the residue is null and returns zero if the relative residue is for instance more than 100%. We found the second criterion (magnetic law) is unity in the whole body, what justifies the basic hypothesis of the $\varphi-A$ formulation, i.e. scattered induction is negligible before the incident one. Moreover, we can note on Fig. 10 (c) that the first criterion (conduction law) is much more selective than the statistic criterion, what indicates that both "a posteriori" criteria are far to be equivalent.

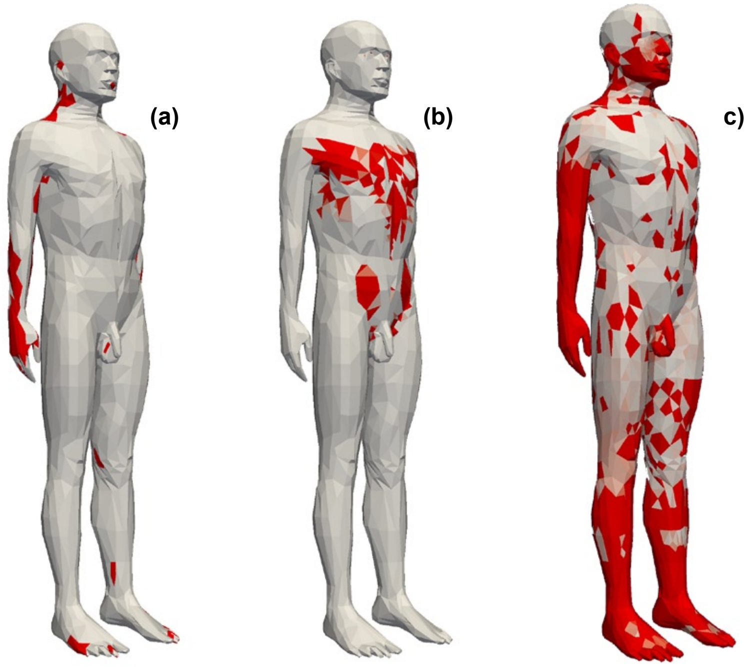


Fig. 10. Bad tetrahedrons selected by our "a posteriori" criteria : statistical A-Phi (a), statistical T-Omega (b), energetical (c).

As pointed before, the case of the muscle is very particular, since in our computational phantom, it is meshed as the complementary volume of other organs, each of them presenting more or less a convex geometry. Therefore, there are a lot of thin interface layers, such those spotted on Fig. 11. There interfaces are not clearly detected as badly meshed by our "a priori" criteria, but reveal nevertheless a poor "a posteriori" quality, as shown on graph of Fig. 11.

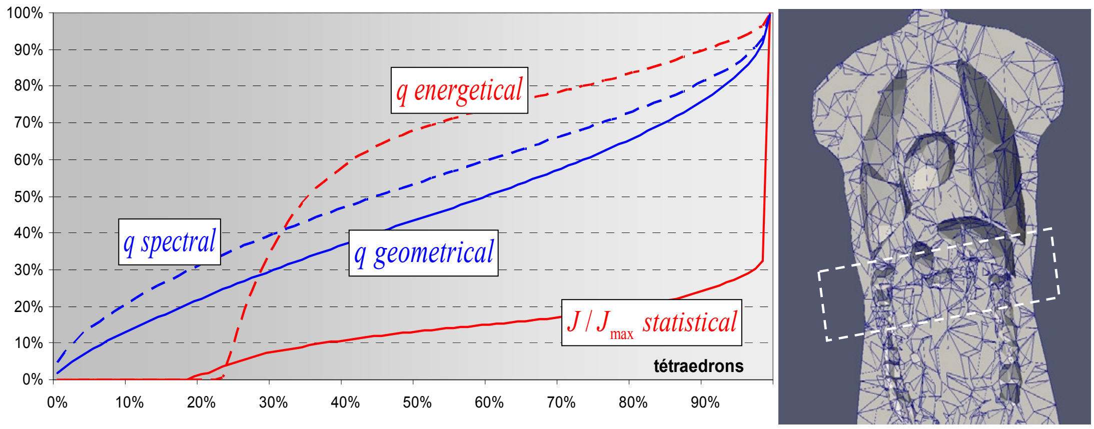


Fig. 11. Case of the "muscle" region. Left : distribution of criteria and right: cut plane in the muscle organ including narrow layers interfacing heart / left lung / intestine / liver / right lung.

Conclusions

Several finite element models of human body and different formulations are available to solve the problem of induced currents by ELF electric and magnetic fields. But till now, there is no general consensus on what combination formulation + mesh is able to provide a reliable maximal current.

We applied four existing formulations on a single human body mesh, two of them working in general purpose codes, the two others being adapted from Maxwell equations to ELF human body configuration. We noticed some discrepancies between results of these formulations, mainly in the high current zones (conductive organs) and in the badly meshed parts of the model.

Then, we defined four quality criteria in order to post-process these results, two of them so-called "a priori" depending on the mesh and the two others called "a posteriori" depending on the results. We found that the "a priori" criteria detect the bad shaped elements but not the high current elements. The "a posteriori" criteria can detect both the contorted elements and the high currents elements at a time, but they are highly sensitive to the formulation in complex organs such as muscle. For that reason, a mesh refinement should be tried in the inter-organ zones before testing energetical criteria.

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Electric properties of mammalian tissues: ex vivo results from 1 Mhz to 1 GHz

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Abstract
Electromagnetic radiations may interact with biological tissues by the way of biomedical devices for therapeutic or diagnostic purposes (hyperthermia, ablatherapy, NMRI,…). Another situation is related to electromagnetic fields sources in the daily life such as mobile or public or domestic devices. Many studies are currently done to determine their potential effects leading sometime to controversy between the published results. The electric properties of biological tissues are frequency dependence and major parameters that governs these interactions inside the human body.

Techniques for both invasive and non-invasive assessment of tissue characterization were proposed since many decades. Different methods are used to determine them among which the so called bioimpedance.

This paper presents and discuss results of ex vivo measurements on mammalian tissues done less than two hours after excision in a surgery department at the Nancy Cancer Center (France). Dielectric permittivity and electrical conductivity of female breast human tissues in the frequency range from 1 MHz to 1 GHz were measured. They are compared to previously published results and the differences discussed according to the influencing parameters.

Introduction
Interactions between electromagnetic field (EMF) and biological tissue find applications in therapeutic or diagnosis methods. On the other hand, development of devices radiating electromagnetic field, like mobile phone, led to the questions of their possible biological effects. These so called EMF bio-effects have led to controversial models for electromagnetic (EM) dosimetry simulation. One reason is that these models are depending on the knowledge of the electric characteristics of the human body. Electromagnetic properties of biological tissues are thus fundamental parameters necessary for any research in bioelectromagnetics.

Many techniques and methods for measuring the bio impedance exist and are primarily motivated by the frequency band of interest (Burdette & Cain, 1980 ; Foster & Schwan, 1996). Interest in the EM properties of biological tissues began more than one century ago (Schwan & Kay, 1957 ; Foster & Schwan, 1986). Bioimpedance spectroscopy is one of the no-destructive, low-invasive and most promising techniques for biological tissues characterization (Rigaud et al, 1996).

Up to now, the values provided by the literature are not always reliable since these data are sparse and disseminated over the frequency range with great variations even for a same biological organ (Gabriel et al., 1996 ; Surowiec et al., 1985). This is not the only major factor and other parameters had to be taken into account for reliability of the experimental results . Moreover, the measurements taken from a sample are likely to be affected by multiple influencing factors such as the contact with the membrane, the temperature at which the measurement is performed, the pressure on the sample, duration post mortem or the structure of the tissue.

The measures on biological tissue are particular for two reasons : firstly because of variations between the samples under test, on the other hand because of the diversity of influencing factors and the non-availability of references for the sensor calibration (Nadi, 2008). Bio impedance measurements need to compensate for the influencing factors related to biological aspects and those related to metrology and instrumentation. This is a crucial factor at low frequency. The dielectric behaviour of biological tissues is dependent on their nature and on the frequency of interest (Schwan & Kay, 1957). They are deformable, heterogeneous, anisotropic and may be solid or liquid. As example of the parameters influencing the measures, the temperature is one whose effects on the variations of the conductivity and permittivity have been previously investigated and is now well known (Foster & Schwan, 1996 ; Jaspard & Nadi, 2002). Precautions taking account of these influencing parameters are required to ensure metrological reliability of the results by adequate corrections.

Comparative studies remain thus a real challenge. In this paper, ex vivo dielectric data are presented and discussed for female breast tissue.

Material and methods

Measurement cell

The experimental set up is based on a material analyser (HP 4291A). Its measurement cell was modified to allows measurement of small biological samples (Gagny, 1998). Data acquisition is done throught an IEEE connection to a PC. This material analyser works on an original method V/I extended to a large frequency band, up to 1 GHz. The main principle of the cell is similar to the geometry proposed by Von Hippel (Von Hippel, 1954).

The experimental set up and the measurement cell are presented on figure 1 & 2. It is constituted by two circular electrodes of which the upper one could be adjusted throught a precision micrometer. The lower electrode is surrounded by a ring guard and ground planes.

Cell for measuring ex vivo samples

A small support (relative permittivity of 1.1 ; loss tangent δ 0.01) permit to fix the position of the sample on the edge outside the circular lower electrode (Gagny, 1998) in order to cover it completely. According to the cell manufacturer specifications the optimal dimensions of the sample placed in the middle of the measurement cell have a diameter higher than 15 mm and a thickness equal or less than 3 mm. This is one of the technical influencing parameters that could affect the reproducibility but it is not discussed here.

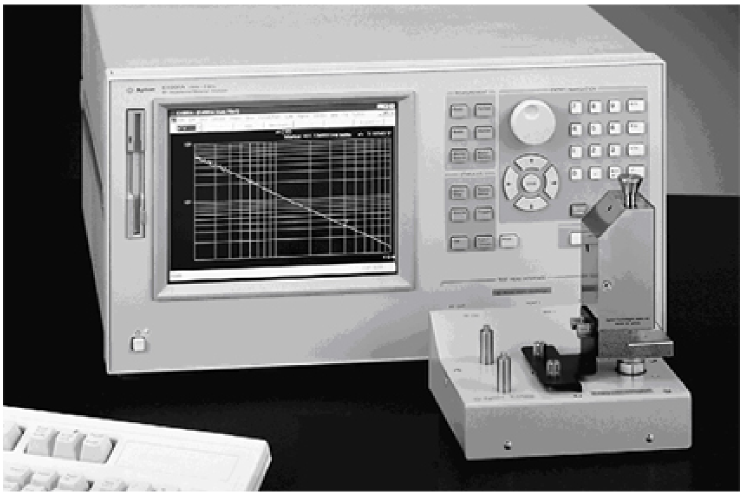

Figure 1.View of the material analyser HP 4291A and the measurement cell.

Eight tissues samples used in this ex vivo study were obtained from surgical ablation procedure on female breast tumor performed in the Cancer Center at the University hospital of Nancy (Gagny, 1998). The samples were excised in tissues surrounding and inside the tumor in the thirty minutes following the surgical procedure by the surgeon. Dimensions of the samples obtained were limited by the clinical constraints since they were given to us on the side to the medical analysis. The necessity to preserve the quality of the tissues excised for biological and medical analysis of the tumor effective excision is one of these constraints. The measurements were performed at ambiant temperature in the maximum of two hours following the surgery, the experimental set up being placed in a room close to the surgery department.

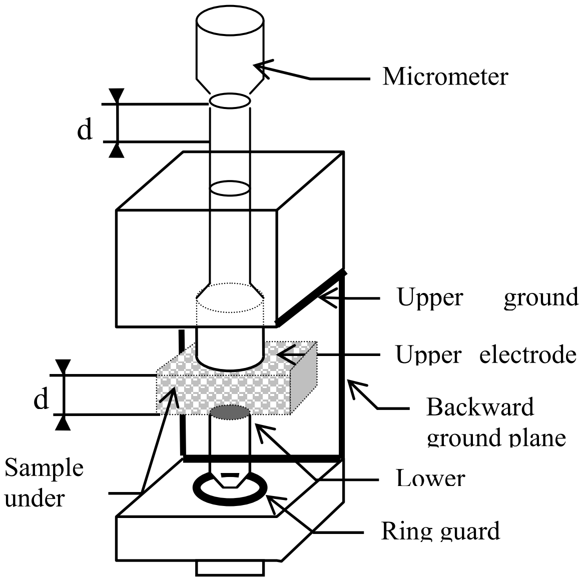

Figure 2. Modified measurement cell for ex vivo electromagnetic characterization.

Method

The instrumentation measures a complex admittance of a 3 points calibrated cell in which the sample is inserted between the two circular electrodes. Theoretical model of the measurement cell allows to solve the inverse problem in order to deduce the permittivity and the conductivity from the measurements. The electric parameters are deduced from the real and imaginary parts of the measure.

$$\varepsilon_r^* = \frac{Y_m^* . d}{j\omega \varepsilon_0 S(1+E_{edges})}$$

where

$E_{edges} = 454 . d^{0.825} . \varepsilon_r^{-0.554}$ is a factor for compensation of edges effect

d : distance between the electrodes [m]

ε_r^* : complex permittivity

Y_m^* : complex admittance

ω : pulsation [rad/s]

ε_0: vaccum or air permittivity

S : surface of the circular electrodes [m^2]

In the low frequency band, one must take account of major influencing parameters such as the well known phenomena of electrodes polarization (Foster & Schwan, 1986). Other metrological parameters like the difficulty to calibrate the measurement cell, the variations of biological parameters such as the anisotropy and heteregoneity of the samples affect the reproducibility of the results.

One advantage of the material analyser HP 4291A is that it operates with the V/I procedure extended in a broad band, typically from 1 MHz up to 1 GHz. The automated data acquisition permit to scan this broad band with a logarithmic step giving 201 frequency between 1MHz et 1GHz. All measurements were made between 21 and 25 °C.

Results and discussion

Ex vivo samples

A systematic spectroscopic measurements were done for different samples and compared to the previous results obtained by other authors.The samples were classified according to their nature (muscle, fat) and dimensions. We present below the mean value deduced from measures done on a single sample (mostly non fat) and the mean value for measures on three similar samples with different dimensions and structure (mostly fat).

Measurement of electric conductivity and dielectric permittivity

Among the datas obtained with our method, only the results that could be compared to others for ex vivo human breast are presented on this paper. Measurements for permittivity and conductivity of breast muscle and breast fat samples are presented on figure 3 & 4 and compared to those from Schwan (Schwan & Kay, 1957 ; Stoyes et al.,

1982 ; Foster & Schwan, 1986), Joines (Joines et al., 1994) and Gabriel (Gabriel et al, 1996). One goal is to check the high differences that can be observed between results from one research team to another in the low frequency range.
Concerning our results, as explained above, a spectroscopic logarithmic measures were done between 1 MHz and 1 GHz. For the clarity, the standard deviation is given only for 15 points.

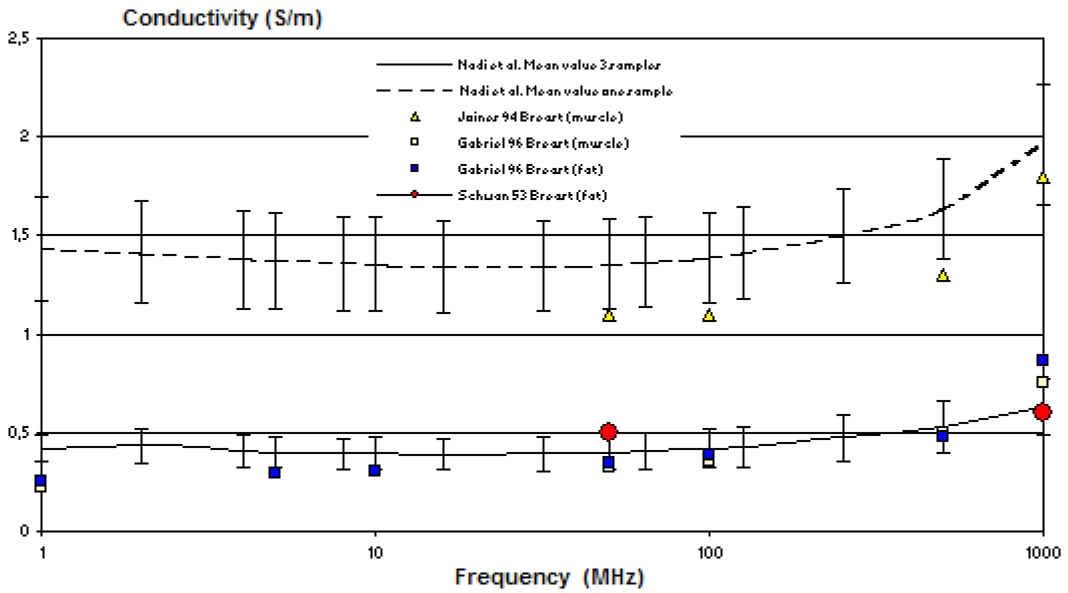

Fig. 3. Electric conductivity of muscle and fat from female breast samples versus frequency compared to previous datas

One can observe that the set of measurements obtained with the mean value from different samples (mostly fatty) is similar to the results given by Schwan (fat) and Gabriel (muscle and fat). However, on can observe that Gabriel's measures on fat and muscle breast are very close. The difference between her datas for fat and muscle are less different than between Joines datas for muscle and her datas for muscle even at high frequency. This is strange and we have no explanation for such high difference.

A mean value was deduced from a set of measures on the same sample (mostly muscle, plain line). The second curve (dashed line) corresponds to the mean value obtained from three different samples and are far from those of Schwan (fat) and Gabriel (muscle and fat). Joines (muscle) datas are also overestimated compared to the other results. However his datas seem similar to our results obtained by mean value on one sample (mostly muscle).

The most important differences for both the conductivity and the permittivity are situated below 10 MHz. It is well known that in this frequency range current conduction through tissue is mainly determined by the tissue structure, i.e. the extra- and intra-cellular compartments and the insulating cell membranes. Thus the nature of the biological tissue is more sensitive at low frequency. On the other hand, the polarizarion effect on the electrodes is also more important below 1 MHz.

Obviously, according to the conditions of both the procedure and the preparation of the sample and its nature, the time after excision, it is normal to find such differences from an author to another.

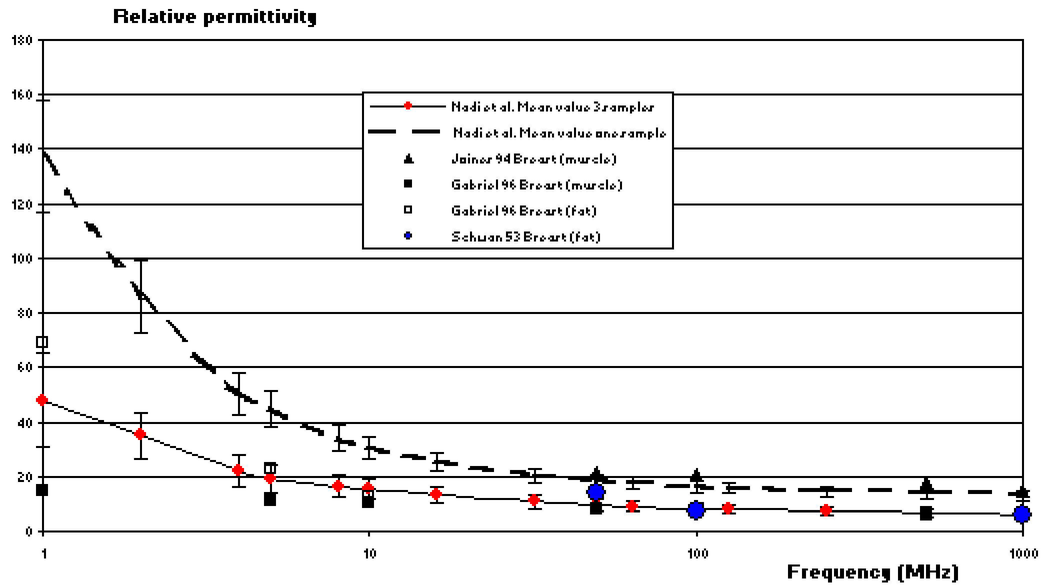

Fig. 4. Dielectric permittivity of muscle and fat from female breast samples versus frequency compared to previous datas

The fact that results are obtained by using different instruments, conditions and procedures (40's up to 80's for Schwan) could explain partly these differences. Schwan used for his early datas manual bridges that did not allow a systematic spectroscopic measurement. This is one reason why some of his first results are sparse and focused on specific frequency. Results from Gabriel were deduced by compilation of datas from the literature and measurements in her laboratory. Joines datas are not overestimated but the differences are certainly due to the nature of the sample used and its preservation after excision. These differences could be by evidence be explained by the importance of fat and muscle heterogeneity of the breast samples and their dimensions that have been used by each author. Over 10 MHz one can accept the hypothesis that all the values are in the same order for the permittivity but this is not true for the conductivity since the difference is greater than 100% between fatty and non fatty breast muscle.

Conclusion

Electric characterization of female breast samples measured ex vivo two hours after the surgery were presented and compared to previous datas. The electrical conductivity and relative permittivity of mammalian human tissues were measured at frequencies from 1 MHz to 1 GHz. The measurements were made using a material analyzer whose measurement cell was adapted to receive small human samples. All measurements were made between 21 and 25 °C.

These results were compared to previous well known datas. Obviously, the differences and variations are due to biological species under test as well as to the metrological difficulties due to the interface between the electrodes and the biological tissue at low frequency.

Metrological parameters affect the reproducibility of the results according to the medium under test and the procedure used by each research team. These differences and variations are due to biological species or organ under test as well as to the metrological difficulties related to the interface between the electrodes and the biological tissue). For mammalian tissues that interest us here, it is necessary to class their heterogeneity as fatty or non fatty since the difference could be over 100% from a sample to another. Other influencing parameters had to be taken into account. Experimental results cannot be rigorously compared as regard to the differences between the natures of the ex vivo sample under test and to the physical and metrological constraints. Comparative studies remain thus a difficult challenge. However, the measurements must stay inside a domain of confidence based on a mean value with standard deviation. At high frequency over 10 MHz, it is possible to admit that the permittivity values are of the same order for fat and muscle breast samples but this is not true for the conductivity.

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Occupational exposure to electromagnetic fields in electrotherapy services and possible related health effects

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Abstract

Aim: The assessment of possible adverse health effects in occupational exposure to electromagnetic fields (EMF) generated by electrotherapy equipments (ETE).

Methods: Exposure evaluation included ergonomical analysis and EMF measurements (static and low-frequency magnetic flux density, low- and high-frequency electric and magnetic strengths). Health status analysis comprised anamnesis, clinical and neurological examinations, exposure and subjective symptoms questionnaires. Psychological tests aim to put in evidence subtle changes of nervous system activity.

Results: 38 electrotherapists vs. 82 matched controls were studied. Magnetic and electric field measured levels generally didn't exceed ICNIRP Guidelines. However, when personnel get closer to the applicators, higher local exposure occurs. Hands and head seem to be the higher exposed. The number of treated patients and the different electrotherapy procedures induce variations in exposure duration. We met three generations of ETE and stray fields seem to be important in older ones. The newer devices show significant lower levels of non-intentional exposure. Health investigations show mainly nervous system subjective symptoms and signs (asthenia, memory and attention disturbances, irritability, vegetative disorders, headaches, dizziness, etc.). These findings seem to be more frequent in exposed and seems to be correlated with the exposure length.

Conclusions: Significant levels of EMF occupational exposure levels were found. Higher exposures are attributable to practice procedures, to peculiar electrotherapy procedures, to bad positioning of ETE, and to older generations of ETE. The lack of risk knowledge is an important factor for some exposure situations, but it could be corrected. Health findings point out to possible effects at higher levels but further studies should be done. Risk awareness policies are to be performed for both employees and decision factors.

Analysis of electric network data in the EXPERS study

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Abstract

The Health Ministry initiated a study on the personal exposure of the French population to 50 Hz magnetic field. One of the points of interest of this study is the contribution of each exposure source to the total exposure. We focus here on electric networks. The different data of the study, which includes a little more than 2000 subjects, allow us to identify for some subjects the presence of one or several networks close to home. The distance for taking such networks into account depends on the type of network: we considered all distribution and transport networks, as well as electric train networks. From magnetic fields measurements, after we checked whether the identified networks are really influencing the exposure of subjects. The results depend on the type of network.

Introduction

The French Ministry of Health initiated a study on the personal exposure to 50 Hz magnetic field of the French population (Souques et al. 2008). The global results of this study, called EXPERS, are presented in another communication (Bedja et al. 2010). A point of interest is the contribution of each source of exposure to the total exposure. We focus here on electric networks.

Material and methods

MV2, poll institute in charge of the collection of data, collected for each subject:

- 24h magnetic field measurement
- timetable
- general questionnaire about home and electric appliances used
- GPS coordinates of the home
- Address of the home

For each type of electric network (low to very high voltage), we determined the maximum distance at which the magnetic field generated by the electric network remains theoretically measurable in a residence. This distance was calculated for overhead power lines of voltage 63kV and higher by simulating magnetic fields generated by the different types of networks, in the middle of the span and with the annual mean current. The distance kept is the one where the mean calculated magnetic field is less than 0.1 µT. The magnetic field depends also on the geometry of the line and the distance kept was the largest of the distances calculated for different geometries.

Note that for train networks, a distance of 200 m was arbitrarily chosen. The transport like metro or tramway were not taken into account.

All subjects within these corridors were arbitrarily classified as "exposed" to magnetic fields generated by electric networks. The address of the home was converted into Lambert II étendu coordinates (French Lambert coordinates) by RTE (the French electricity transport network operator). The results of this calculation was validated by comparison to GPS coordinates measured.

The "exposed" subjects were identified from the addresses of subjects of EXPERS study by the network operators using their geographical data bases (all data not yet available for ERDF (the French electricity distribution network operator), the low and medium grid operator in France).

However, as the width of the corridor is overestimated, it is important to know, for the subjects arbitrary classified "exposed", whether their magnetic field measurements show or not the influence of electric networks. Indeed, the variation over one day of the field generated by electric networks is quite characteristic, and a visual check generally allows to confirm or not the contribution of one (or several) electric lines to the magnetic field record.

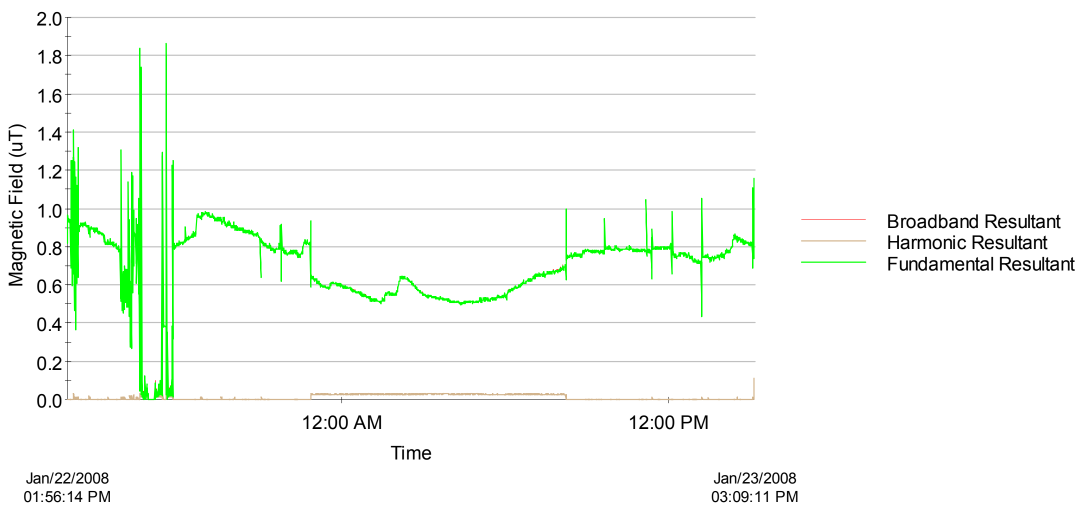

Figure 1. example of a subject really exposed to the magnetic field generated by a high voltage power line.

Figure 1 gives the example of a subject where the source is a high voltage overhead power line. The characteristics are a signal with little noise and proportional to a load curve of a power line, i.e. maximum at the end of the day, decreasing during the night then increasing again in the morning.

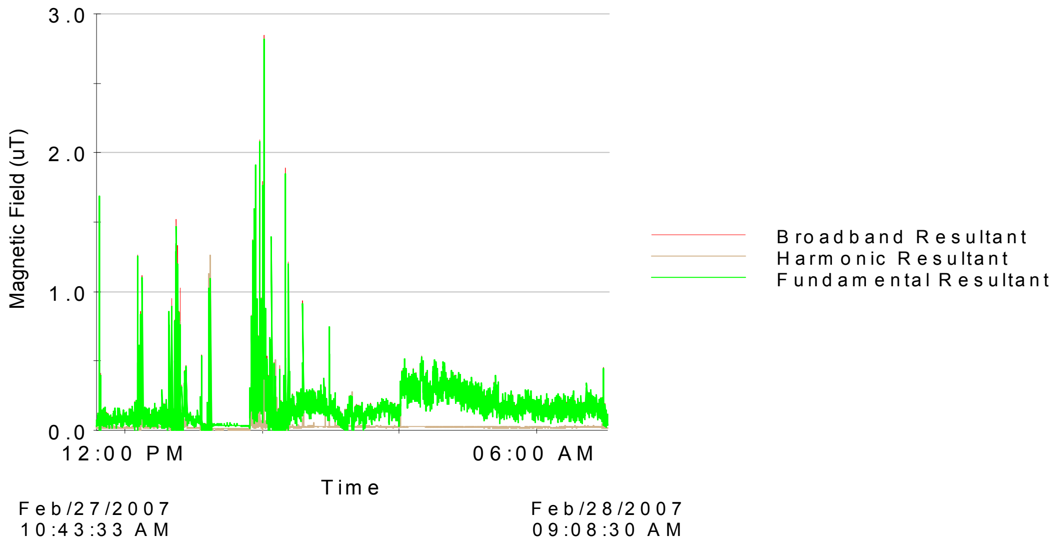

Figure 2. example of a subject really exposed to the magnetic field generated by a middle voltage power line.

Figure 2 gives an example representative of a subject where the source could be a middle voltage underground network. The signal presents the trend of a load curve during the night, but is noisy.

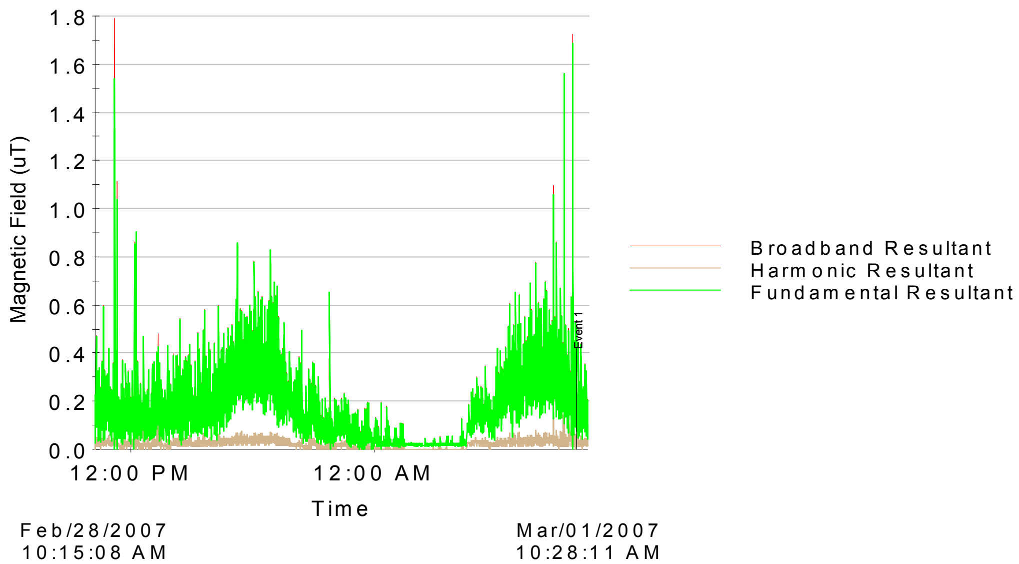

Figure 3. example of a subject really exposed to the magnetic field generated by a train line.

Figure 3 gives the example of a subject where the source is a train line. The characteristics are: a very noisy signal, proportional to traffic (maximum the evening and the morning, zero during the night), and with a ratio of harmonics above zero.

Signals are obviously not all as clear as in the examples given above. A lot are barely detectable and at the limit of ground noise. In order to privilege a conservative approach, we noted as “really exposed” all subjects as soon as we noted a very low tendency at the limit of ground noise.

Results

The EXPERS magnetic field database contains 2048 volunteers and 1581 addresses. There are less addresses than subjects because an adult and a child of the same family could participate in the study.

Table 1 gives the corridor width taken into account around electric networks, in function of the type of networks.

The position of RTE and ERDF networks is known with a precision of 10 m. For electric train networks, a distance of 200 m was chosen because of a lower precision on the position of networks, determined from IGN[1] data, with a precision around 100 m.

Table 1. Definition of corridors around electric networks.

Type of network	Distance (m)
400 kV overhead line	200
225 kV overhead line	120
150 kV overhead line	100
63 and 90 kV overhead line	70
Low voltage to 20 kV overhead line	20
Train line	200
Underground line 225 kV	20
Underground line 63 to 150 kV	20
Underground line low voltage to 20 kV	20
MV/LV substation	20

The number of « exposed » subjects varies between 9 for 400 kV lines (table 2), 162 for train lines (table 3) to more than 1024 for low voltage underground lines (table 4).

Table 2. Distribution of subjects « exposed » to magnetic field generated by RTE networks (data February 2010).

	Number of addresses	Number of individuals	remarks
Overhead line 400 kV	7	9	6 addresses (8 individuals) « exposed » to several lines
Overhead line 225 kV	9	13	
Overhead line 63 to 150 kV	20	26	
Total Overhead line	35	46	
Underground line 225 kV	16	17	5 addresses « exposed » to several lines
Underground line 63 to 150 kV	18	20	
Total Underground line	34	37	

[1] Institut Geographique National

Table 3. Distribution of subjects « exposed » to magnetic field generated by train lines.

	Number of addresses	Number of individuals	remarks
Train line	128	162	70 individuals are in Ile-de-France - 2 individuals with TGV line

Table 4. Distribution of subjects « exposed » to magnetic field generated by ERDF networks (data available today for 1565 individuals over 2032).

	Number of addresses	Number of individuals	remarks
Overhead line 20 kV	25	35	Provisional figures
Overhead low voltage line		620	
Total Overhead line		639	
Underground line 20 kV	455	543	
Underground low voltage line		880	
Total Underground line		1024	
MV/LV substation	60	66	
From which MV/LV substation in building	23	23	

Table 5. Analysis of the magnetic field measurements for subjects « exposed » to magnetic field generated by electric networks.

	Number of individuals	Measurements with influence of an electric network	Remarks
Overhead line 400 kV	9	7	
Overhead line 225 kV	13	11	
Overhead line 63 to 150 kV	26	8	
Underground line 225 kV	17	6	
Underground line 63 to 150 kV	20	8	
Train line	162	37	2/3 in Ile-de-France
Overhead line 20 kV	35	2	Provisional figures (analysis ongoing)
Low voltage overhead line	620	At least 54	
Underground line 20 kV	543	116	
Low voltage underground line	880	At least 115	
MV/LV substation	66	12	
Fro which MV/LV substation in building	23	4	

The visual analysis of measurements for all “exposed” subjects (table 5) shows that the magnetic field generated by the networks was detected in a proportion varying from 85 % of subjects “exposed” to magnetic field generated by very high voltage overhead power lines to 20 % of subjects exposed to magnetic field generated by train lines and less than 10 % of subjects “exposed” to magnetic field generated by 20kV overhead lines.

Discussion

This result must however be balanced by the following remark: the magnetic field measured is the result of the summation of magnetic fields emitted by all magnetic field sources, and not only by electric networks detailed here. As such the results are quite reliable for high voltage overhead lines and train lines whose signals are well characterised on the measurements, but for underground networks, substations and low voltage lines, the signal is at the limit of ground noise and non specific.

For example, for the individuals noted “exposed” to a magnetic field generated by MV/LV substation in buildings, and whose measurements are reflecting an electric network, no one is living at immediate proximity of a substation: the magnetic field “seen” on the measurement is thus not due to the substation.

In the same way, a part of the individuals noted « exposed » to a magnetic field generated by an underground line are living in buildings on a floor too high for underground networks be able to influence their exposure: it could be in this case the electric networks of riser shafts in buildings. A more detailed analysis will be necessary on this point.

Conclusions

This paper shows that:

– the part of the population whose exposure to 50 Hz magnetic field is influenced by high voltage power lines is small,
– the criteria of distance chosen in this study is maximizing and thus overestimates logically the number of people whose exposure to 50 Hz magnetic field is influenced by electric networks,
– it is not conclusive that underground electric networks are really the source of exposure seen in some measurements.

This study will continue with the analysis of all ERDF network data.

To discriminate the real influence of underground networks, it should be necessary to:

– look at the influence of the parameter “floor of the building”,
– look at the measurements of individuals noted as “non exposed” to magnetic field emitted by an electric network, to see whether all are presenting exposures non influenced by electric networks or whether if some are presenting exposures influenced by electric networks, which would accredit the hypothesis of the influence of electric networks from riser shafts in buildings.

Acknowledgment

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On implementation of the new methodologies concerning measurement of occupational electromagnetic field levels

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Abstract

Measurement of electromagnetic field levels is a very important step in assessing human exposure to electromagnetic fields present in the working environment. After the year 2000, the methodologies on electromagnetic field measurement have been much developed and improved and, at present, there is a big number of technical standards dedicated to various exposure situations.

The implementation of the new methodology on electromagnetic field measurement in the national practice might be difficult, expensive, time consuming and it requires high qualified and special trained personnel. Considering the big change in the methodology to be applied, we analyze the limitations, difficulties and errors that might alter the quality of field measurement and exposure assessment.

To overcome the temporary lack regarding the good knowledge and clear understanding of the new methods, the authors propose a simple strategy consisting in a few steps. A general methodology was elaborated to be assimilated by the personnel involved in this domain, before learning complex methods. Standardized models of measurement report have been proposed. To ensure a solid background, training activities of measurement operators should be carried out by experts.

In agreement with European demands, some specific measures have been proposed to be taken concerning information dissemination, consultancy activities and, if the case, classification of measurement service providers into basic level and high level services. As an example, our activities in Romania in this domain are briefly described.

Microlens formation as protective mechanism against direct laser radiation

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Abstract

Laser radiation poses a significant threat to human eyes. Depending on power, even scattered radiation can produce damage (mostly to retina), not to speak of direct laser beam. The usual method of protection is through different kinds of filters (absorption, reflection, polarization, holographic) inside goggles or protective windows. Without exception all types of filters protect human eye from scattered or otherwise diffuse reflected laser light. All manufacturers of laser goggles specify that the eye is not protected against direct laser beam. We describe a novel principle of eye protection which is based on expanding laser beam by the filter material itself. In other words, when the laser beam hits the filter material, negative (diverging) microlens is formed almost immediately. The beam is strongly expanded by the microlens, thus reducing the energy density to tolerable level. To test the idea we have developed a suitable material – tothema sensitized gelatin. Tothema is a trade-name of a mixture of gluconates used to treat anemia. Mixture is added to ordinary cooking gelatin, rendering the material sensitive to laser radiation. Sensitivity was further enhanced at 532 nm by additionally doping material with eosin or betanin. Experiments have shown that upon irradiation, strong microlens (focal lengths far below -1 mm) forms quickly and the laser beam diverges, reducing the overall energy density. Results of measurements of dynamics of the process are shown.

Biological hazard of low-intensity radio frequency electromagnetic fields – Model experiments with protozoa

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Abstract
The effects of low-intensity 1 GHz (a mobile connection frequency) and 10 GHz (radar and satellite communications) electromagnetic radiation with energy flux density (EFD) of 5, 10 and 50 μW/cm^2 on a laboratory population of unicellular aquatic organisms – ciliates *Spirostomum ambiguum* were studied. The effects were registered by the criterion of change in spontaneous motor activity (SMA). SMA of ciliater was observed under the microscope. Two lines were deposited in the eyepiece of the microscope, crossing each other at right angles. A quantitative measure of motor activity of each ciliate was a number of intersections of the lines on the eyepiece of the microscope for 1 min. To do this, each ciliate was placed individually in a drop of water in a special cell with holes 5 mm in diameter and 2 mm in depth.

Sp.ambiguum used for a long time to estimate the negative effect of chemical and physical factors on the environment. These ciliates have proved to be a sensitive bioindicator of low-intensity influences. Our research has shown that electromagnetic radiation reduces motor activity of ciliates already at such low levels of exposure as 10, 45 minutes and 8–9 hours respectively with EFD 50, 10 and 5 μW/sm^2. The negative effect did not depend on a frequency of electromagnetic radiation in the diapason from 1 to 10 GHz. The effect had a threshold character. After reaching the threshold, the negative effect did not change with exposure and had a mass character. The level of SMA in the population of ciliates decreased at 40%. The negative effect transmitted to descendants of irradiated ciliates. The results were confirmed by independent experiments involving more than 5000 *Spirostomum*.

Our results are interesting for practical using of protozoa's behavioral reactions for testing of biological hazard of electromagnetic pollution of aquatic environments. It is of interest as the general problem of electromagnetic activity on the biota.

Introduction
It is known that not only sufficiently powerful fluxes of energy (in tenths of W/cm^2) of radio frequency electromagnetic fields (EMF) have had a negative impact on organisms. The negative biological effect occurs when the energy flux density (EFD) is comparable with the level in tens and units of μW/cm^2. Such levels of EFD are

observed in many workplaces and places of human habitation. Interest in the biological effects of low intensity electromagnetic fields has increased significantly with the advent of mobile communications.

Experts did not have general opinion about safety of radio frequency electromagnetic radiation with nonthermal power in spite of the duration studies with help of different biological subjects. There are opposing points of view. This is reflected in huge differences of the maximum allowable levels (MAL) of electromagnetic fields which are recommended in different countries. Until recently, differences in the MAL for the population in different countries were to 1000 times. In Russia the MAL installed as 10 $\mu W/cm^2$ (Sanitary rules and norms 2.1.8/2.2.4.1190-03, Russia), that is less than in Europe and in the USA. These regulations, however, are not sufficiently substantiated. The experimental data are contradictory. The theoretical justification is absent.

The biological risk of electromagnetic radiation has been studied with help of very labor-intensive methods. It hampered the expansion of research. We used an express method which is based on a quantitative assessment of changes in spontaneous motor activity of aquatic ciliates *Spirostomum ambiguum* Ehrbg (Tushmalova et al 1998). The toxicity of pesticides, heavy metals, phenols and other factors, including those at low doses of agents, as well as the differentiation of spirostomum's motor activity by alpha- and beta-irradiation were detected within 1-2 h (Tushmalova et al 1998; Tseplin et al 2005; Sarapultseva 2008). High sensitivity of this method was shown in all cited works. We carried out several studies of the influences of radio frequency electromagnetic fields on the motor activity of *Sp.ambiguum* (Sarapultseva et al 2008; Sarapultseva et al 2009). The results, in our opinion, have the theoretical and practical interest. We found negative effects of low-intensity 1 GHz (a mobile connection frequency) and 10 GHz (a frequency of radars, satellite television, radio, wireless computer networks, satellite navigation, etc) electromagnetic radiation. The negative effects of electromagnetic radiation i) was manifested at low energy flux density, ii) did not change with increasing exposure of EMF in a wide time range, and iii) was inherited during vegetative reproduction.

Let's consider the obtained results.

Material and methods

Ciliates were cultured at 20 ± 2^{0}C in the biological test tubes with tap water, devoid of chlorine. Installations for the electromagnetic irradiation of biological samples consisted of generators, antennas and cables which were giving energy in the mode of continuous generation. The energy flux density had regulated in accordance to the established relationship for each frequency electromagnetic radiation.

Ciliates were irradiated in the mass culture in Petri's plastic dishes with a diameter of 4 cm and 0.3 cm layer of water. The time of exposure in the electromagnetic field was dependent on the energy flux density and ranged from 5 minutes to 16 hours. The control groups of ciliates were in similar conditions, but without the influence of electromagnetic fields.

Quantify changes in motor activity of ciliates was carried out immediately after irradiation. To do this, 20 ciliates were selected randomly from mass culture, which was irradiated with the frequency of 1 or 10 GHz and energy flux density of 5 or 10 or 50 μW/cm^2 and during from 5 min to 16 h. Spontaneous motor activity (SMA) of ciliates from the control groups was tested at the same time. Ciliates from irradiated and control groups were placed individually in a drop of water in special plastic plate with holes of 5 mm in diameter and 1-2 mm in depth. SMA of each ciliate was observed under the microscope. At the eyepiece of the microscope were did two lines crossing each other perpendicularly. A quantitative measure of motor activity of each ciliate was the number of intersections of the lines for 1 min.

The ciliates after irradiation were continued to cultivate at optimal condition in order to clarify the question of inheritance of negative effects of low irradiation. The effect of irradiation was estimated by reduce of motor activity in the experimental groups of ciliates. These changes were measured immediately just after irradiation and at the 4th, 14th, 21st and 30th days.

It was held on 3–5 series of observations at each case. Thus, it was tested more then 5 000 ciliates. Assessing the reliability of experimental data was carried out with help of the nonparametric Fisher test.

Results

Study of biological electromagnetic radiation effects at 1 GHz with a flux density of energy 10 μW/cm^2

First, consider the results of low-intensity electromagnetic radiation at a frequency of mobile communications (1 GHz) with the maximum allowable in Russia energy flux density 10 μW/cm^2. The change in ciliates' motor activity after radiofrequency exposure duration 15–60 minutes you can see in the Table 1.

Table 1. Changing in ciliates' motor activity (in absolute terms and in % relative to control) just after radiofrequency exposure duration 0.25-1 h at 1 GHz (EFD 10 μW/cm^2).

The exposure time, minutes	$M \pm m$	%, $(M_{exp}/M_c)\cdot 100$
0 (Control)	2.15 ± 0.18	100.0 ± 8.4
15	2.20 ± 0.17	102.3 ± 7.7
30	1.80 ± 0.21	83.7 ± 11.6
45	1.20 ± 0.15*	55,8 ± 12.5*
1	1.25 ± 0.17*	58.1 ± 13.6*

* $p \leq 0.05$
$M \pm m$ – average motor activity with a standard error
M_{exp}* – *motor activity in experiment
M_c* – *motor activity in control

Statistically significant ($p \leq 0.05$) decrease in motor activity of ciliates occurs "abruptly" after 45 minutes stay in the EMF at 1 GHz and EFD 10 μW/cm^2. Table 2 presents the results of observations in 3 series of experiments at the threshold exposure time of 45 minutes.

From Table 2 shows that the average SMA of ciliates in the experimental groups is significantly lower than SMA in the controls groups. Change in motor activity is well reproduced in all series of experiments and differs by 40–45% of control.

In subsequent experiments the effect was studied with increasing exposure in EMF. At Figure 1 shows the results of change in motor activity of ciliates immediately after RF exposure in a wide time interval from 15 minutes to 10 hours.

Table 2. Changing in ciliates' motor activity in control and after exposure at 1 GHz during 45 minutes in three series of experiments.

№ Experiments	Control, $M_c \pm m_c$	Experiment, $M_{exp} \pm m_{exp}$	%, $(M_{exp}/M_c) \cdot 100$
1	2.10 ± 0.18	1.20 ± 0.15*	57.1 ± 12.5*
2	2.25 ± 0.18	1.32 ± 0.17*	59.2 ± 13.0*
3	2.20 ± 0.15	1.15 ± 0.19*	52.3 ± 16.5*
Average	2.18 ± 0.16	1.22 ± 0.17	55.9 ± 13.9

*** $p \leq 0.05$**
***$M \pm m$* – average motor activity with a standard error**
***M_{exp}* – motor activity in experiment**
***M_c* – motor activity in control**

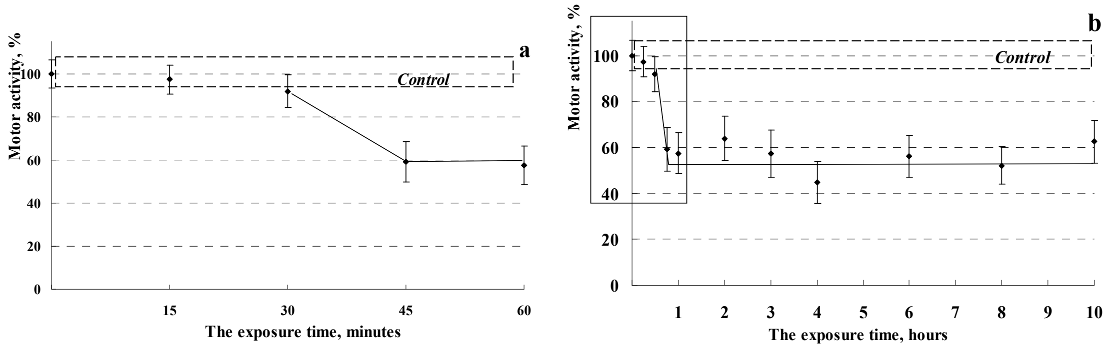

Fig.1. Changing in ciliates' motor activity (in% relative to control) at the electromagnetic irradiation with 1 GHz and 10 μW/cm² duration up to 60 minutes (a) and up to 10 hours (b).

The differences in ciliates' motor activity between the experimental groups and controls are evident, equal and don't change with increasing exposure from 0.75 to 10 h.

Study of biological electromagnetic radiation effects at 10 GHz with a flux density of energy 10 μW/cm²

At Table 3 shows the changes in motor activity of ciliates immediately after exposure of up to 10 hours at a frequency of 10 GHz.

Statistically significant ($p \leq 0.05$) decrease in motor activity occurs "abruptly" after 45 minutes stay in the EMF. That time coincides with the time of the "safe" location in the EMF with the parameters of mobile communication (Table 1 and 2). Table 3 also shows that the negative effect produced by radiofrequency exposure within 45 minutes, didn't change even with increasing exposure time to 10 hours.

Save a negative effect in the generations of ciliates

Let's consider the results of motor activity's changes in the offspring of irradiated ciliates. Figure 2 shows the change in motor activity of ciliates on the 4th (a), 14th (b), 21st (a) and 30th (d) days after exposure.

Table 3. Changes in ciliates' motor activity (in absolute terms and in% relative to control) immediately after radiofrequency exposure duration of 0.25–10 h at 10 GHz (EFD 10 μW/cm^2).

The exposure time, h	*M* ± *m*	%, $(M_{exp}/M_c)\cdot 100$
0 (Control)	2.14 ± 0.06	100.0 ± 2.8
0.25	2.09± 0.10	97.7 ± 4.8
0.50	1.95 ± 0.09	91.1 ± 4.6
0.75	1.33 ± 0.08*	62.2 ± 6.0*
1	1.24 ± 0.06*	57.9 ± 4.8*
2	1.19 ± 0.10*	55.6 ± 8.4*
3	1.13 ± 0.07*	52.8 ± 6.2*
4	1.08 ± 0.06*	50.5 ± 5.6*
6	1.34 ± 0.08*	62.6 ± 5.9*
8	1.42 ± 0.12*	66.4 ± 8.5*
10	1.42 ± 0.09*	66.4 ± 6.3*

*** p ≤ 0.05**

***M* ± *m* – Average motor activity with a standard error**
M_{exp}* – *motor activity in experiment
M_c* – *motor activity in control

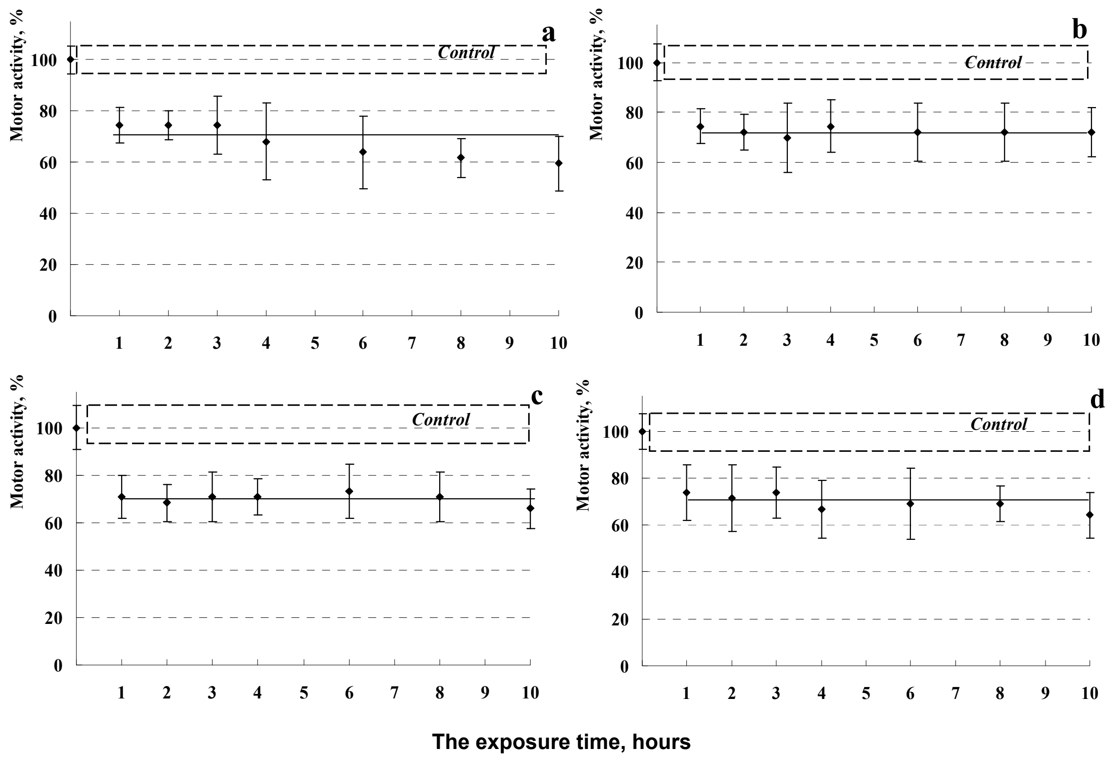

Fig. 2. Changes in ciliates' motor activity (in% relative to control) after radio frequency exposure at 10 GHz and 10 μW/cm^2 on the 4th (a), 14th (b), 21st (c) and 30th (d) days

Figure 1 shows that the significant decrease ($p<0.05$) in SMA was approximately the same at all times (immediately and in distant periods after exposure) and at all period of irradiation. That effect took place at exposure from 1 to 10 h. The effect was recorded on the 30th days after irradiation. During that time the ciliates replaced about 10–12 generations (average length of the cell cycle was about 2 days (Wichterman R., 1953). It follows that the effect can transmitted to descendants of irradiated cells.

Study of biological hazards of electromagnetic radiation at different radio frequency energy flux density

Figure 3 shows the results of changes in motor activity of ciliates at electromagnetic radiation at a frequency of mobile communications (1 GHz) with EFD 5 μW/cm^2.

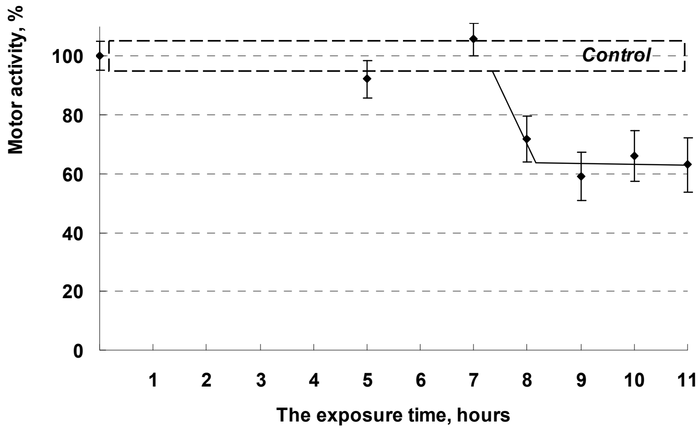

Fig. 3. Changing of ciliates' motor activity in electromagnetic field with EFD 5 μW/cm^2 at 1 GHz up to 11 hours.

Recall that such energy flux density is in 2 times less than the maximum allowable level in Russia. However, Figure 3 shows that the ciliates' motor activity reduces about the same as it is in the electromagnetic field with the energy flux density of 10 μW/cm^2. Decreased of motor activity occurs, however, after much longer exposure, after 8–9 h. The effect in that case when the exposure reached a threshold no longer depends on the duration of exposure. The saltatory nature of the effect in these experiments was clearly registrated. If a 7-hour exposure was not causing reduction of spontaneous motor activity, the 8-hour exposure caused this effect. Increased exposure time up to 9, 10 and 11 h hadn't influence on the degree of change in ciliates' motor activity. The time which ciliates spent in the electromagnetic field with a lower energy flux density of 5 μW/cm^2 (when the changes in ciliates' motor activity did not occur) increased in 16 times (from 45 min to 8-9 h) compared with the exposure time in the EMF with the energy flux density of 10 μW/cm^2.

The overall pattern and degree of changes in SMA were the same after irradiation at a frequency of 1 GHz with an even higher energy flux density – 50 μW/cm^2. The effect was absent in a certain time range, but then the motor activity of the experimental ciliates abruptly decreased by about 35–40%, as happened in previous cases. In the case when the energy flux density was 50 μW/cm^2, irregularities in the motion ciliates occurred with exposure of no more than 10 minutes.

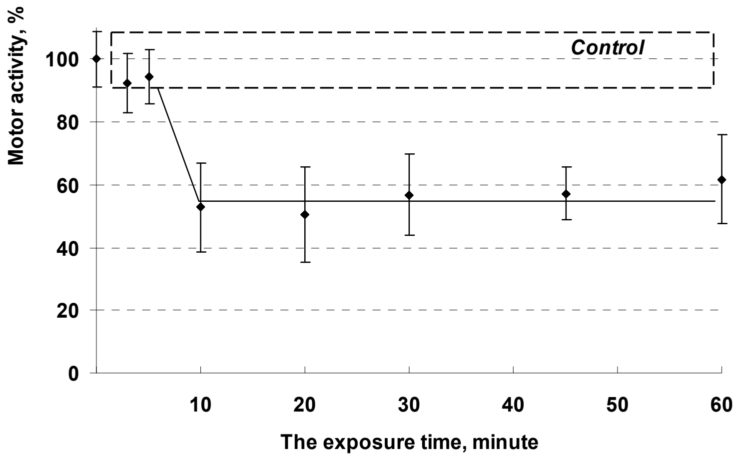

Fig. 4. Changing of ciliates' motor activity in electromagnetic field with EFD 50 μW/cm^2 at 1 GHz with exposure of 1 minutes up to 1 hours.

Next, we evaluated the biological effectiveness of EMF at a frequency of 10 GHz. The degree of inhibition of ciliates' motor activity was significant and approximately equal and didn't changed with increase of energy flux density from 5 up to 50 μW/sm^2. However, the time of occurrence of dysfunction movement depended on the energy flux density. The safe duration of exposure at 10 GHz was about 8–9 h, 45 and 10 minutes respectively with the EFD of 5, 10 and 50 μW/cm^2 as was observed at a frequency of 1 GHz (figure 5).

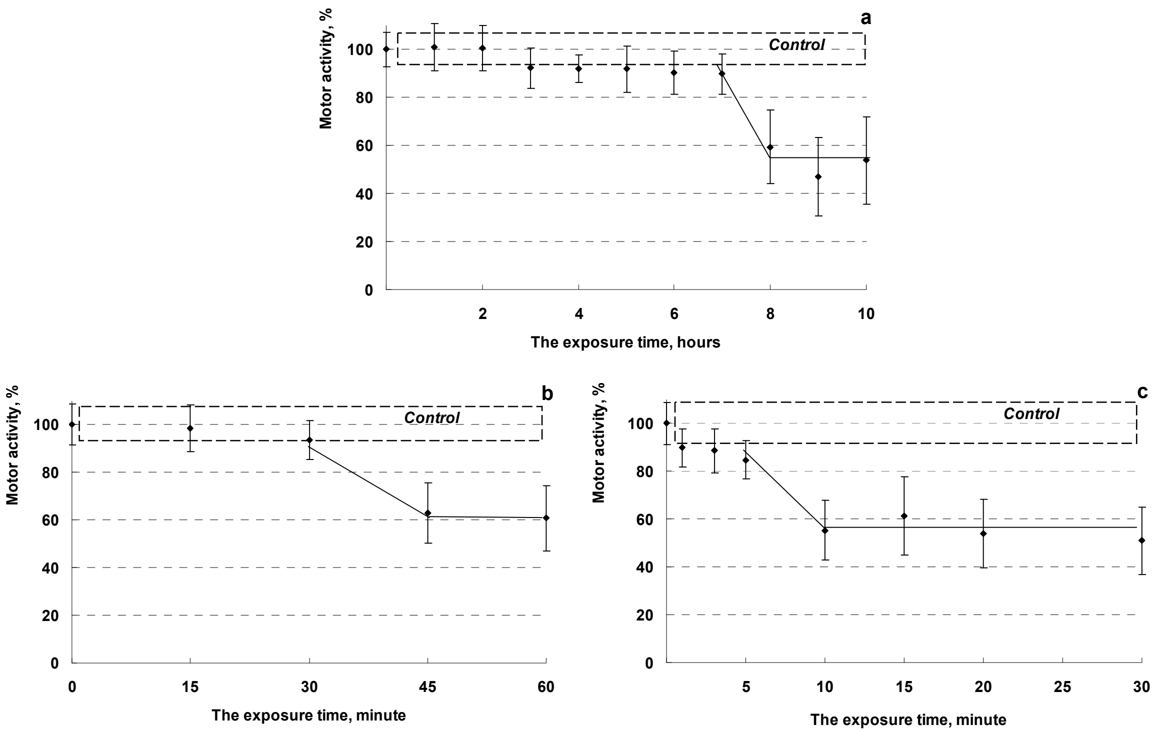

Fig. 5. Changing of ciliates' motor activity in electromagnetic field at 10 GHz with EFD 5 (a), 10 (b) and 50 (c) μW/cm^2.

From Figure 5 shows that the extent and nature of the effect of electromagnetic radiation at a frequency of 10 GHz was not appreciably different from those which were found after irradiation at the frequency of mobile communications 1 GHz (Fig.3 – 4 and Tabl.3). Decrease in ciliates' motor activity was approximately 40%. The degree of change in the ciliates motor activity wasn't depended of the time of exposure after reaching a threshold.

Discussion

The study shows that even very low levels of electromagnetic radiation of radio frequencies, which is widely used by people, could harm the function of motion of the representative biota. A massive inheritable independent from duration of electromagnetic irradiation a decrease of ciliates' motor activity is unusual phenomenon for low-intensity non-ionizing radiation.

However, the results are quite consistent with the data obtained by us after γ-irradiation *Sp.ambiguum* in a wide range of doses, including the low doses (Sarapultseva et al., 2008). A mass effect of decreasing the viability of individuals in populations of other species of ciliates (*Paramecium caudatum, Climacostomum virens*), amoebae and mammalian cells after exposure to low doses of radiation was described also in the book of I.Bychkovskaya et al (Bychkovskaya et al., 2006 http://irbb.ucoz.ru). These violations persisted in remote periods after irradiation – inherited. This proves the reality of the effect and assumes a possible common mechanism of biological action of low-dose of electromagnetic radiation of different nature.

The data about the occurrence of negative effect in motor activity of ciliates after relatively low electromagnetic effects in the early periods after irradiation and irreversible transfer to the progeny rule out the possibility of mutational nature of these changes. We assume that one possible mechanism is a change in the structure of DNA. It is the basis of epigenetic inheritance (Jablonka E. et al., 2008).

Despite numerous studies to date were not significantly determined the basic mechanisms (except thermal) effects of electromagnetic radiation on biological subjects. There are recent theoretical developments and some, though not indisputable facts that the electromagnetic radiation is not only a background against which the unfolding basic biological processes, but it plays a decisive role in all processes of exchange. Several studies show that water plays a decisive role in the effects of electromagnetic radiation on biota. The observed biological effects of low doses of electromagnetic radiation may be due to the formation of reactive oxygen species in the aquatic environment. This is accompanied by stronger oxidative process and the formation of a large number of free radicals that bind to protein components (especially in the tissues which involved in metabolism). It is known that free-radical processes violate the metabolism of DNA, causing mutations. It violates the complementarily of the structure of DNA, breaking hydrogen bonds; induce chromosomal and genomic disorders. Such processes are for non-thermal electromagnetic radiation. The specificity of the non-thermal radiofrequency electromagnetic field is determined of the resonance character of the energy and informational nature. The primary effect is realized at the cellular level and is associated with elements of cytoplasmic membrane, in particular, molecules of proteins and enzymes with a significant electric dipole moment.

Conclusions

We obtained the data that the negative effect of electromagnetic radiation to ciliates' motor activity takes place already at the maximum allowable, and even in two times lower than maximum allowable limit of energy flux density. It is shown that the nature and extent of damage from electromagnetic radiation with frequencies differing by ten times (1 and 10 GHz) are virtually identical. Those are of particular importance in terms of EMF practical application. This is interesting in connection with the problem of standardization of EMF on biota.

Analysis of the new data will be interesting as a fundamental aspect in the discussion of dose-independent reactions, which were obtained in different biological subjects after low radiation (Bychkovskaya et al., 2006 http://irbb.ucoz.ru; Sarapultseva et al., 2008; Sarapultseva, 2008; Sarapultseva et al, 2010). In this case, however, remains an open question how formed the equal response to exposure of low doses of different agents if those agents are very differ among themselves on transferred energy. This question requires further experimental studies.

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Sunbed-usage by 12–23 year old in Iceland 2004–2009

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Abstract

In Iceland, polls have been conducted every year since 2004 among 1800 randomly selected individuals aged 12-23 years inquiring whether they used sunbeds in the previous 12 months. The intention was to determine the level of sunbed use by the young and to monitor the effectiveness of a yearly information campaign on the possible consequences of such usage.

On average; approximately 20% of those between ages 12 and 15 said they used sunbeds as compared to 52% of those aged 16-19 years. For the whole group of 12-23 years old, 41% reported using sunbeds.

A statistically significant 20% reduction in the number of users took place for the whole group aged 12-23. However, the reduction was smaller and statistically insignificant for those aged 12-15.

Introduction

In Europe, young people below age of 18 have long been discouraged from using sunbeds by mandatory user instructions in sunbed saloons. The effectiveness of this discouragement is not well known since data on the usage by the youngest users is scarce. Published information on sunbed use normally only spans age groups 18 year and older.

Concerns over a rising incidence of melanoma in Iceland, especially among young women, led to the formation of a UV-task group in 2004 by the Icelandic Radiation Safety Authority, the Cancer Society, the Association of Dermatologists, the Directorate of Health and the Public Health Institute of Iceland. Advertisements from sunbed saloons aimed at children preparing for their confirmation ceremony at the age of 13 to 14, prompted the group to employ Capacent-Gallup, a professional polling agency to monitor the sunbed usage of young people with annual user-polls.

Advertisements, prepared by an advertising agency (ENNEMM) were published on the 10th of March 2004 in newspapers with an open letter to parents, signed by the directors of the above named health authorities, see Figure 1.

Fig. 1. Advertisement aimed at parents. A confirmation figure with an imprint of sunbed-goggles in its face. The text asks parents to consider the health consequences of cosmetic tanning.

The advertisements were followed up by interviews with health specialists in the news-media. In the following years, similar activities have been repeated. In Figure 2 is an example of an interactive Internet banner prepared in 2008 to grab the attention of the young. This effort had support from newspapers, radio and TV-stations and received much attention.

Fig. 2. An Internet banner aimed at youths with a title in slang-language followed by a simple message: "Young people should not use sunbeds". When the lever is moved the smiley goes through several stages of burning (three are shown).

Material and methods

Data on sunbed usage was collected by Capacent-Gallup in Reykjavik, using 1800 randomly selected individuals aged 12-23, each year in March to May. During the first years, individuals were contacted by telephone, but presently the majority is polled through the Internet.

In the first few years, answers were received from almost 70% of those selected but this number is down to almost 50% in the most recent polls. This is however not believed to skew the results because the reason for declining response is not believed to be linked to sunbed usage. The questions on sunbeds are among many others that are mostly relevant to product marketing and responses to all polls have been decreasing in recent years. For children younger than 18 years, permission from parents must be sought which probably reduces the response rate.

When averages are made over different groups (sex, age, living in or outside Reykjavik), they are corrected for differences in group response.

Results

Usage percentages are presented in Table 1. Those who are 12 to 15 years old have not finished their compulsory schooling and are grouped together. Most of those aged 16-19 are in high school / junior college (Icelandic: menntaskóli, Danish: gymnasium) and many of those aged 20-23 are university students.

Table 1. Percentages of different age groups that said they used sunbed the previous 12 months.

Year	12 -15 year	16-19 year	20-23 year	12-23 year +/- 95% CI
2004	25%	56%	61%	46% +/- 3%
2005	13%	49%	57%	38% +/- 3%
2006	19%	51%	55%	40% +/- 3%
2007	21%	52%	52%	40% +/- 3%
2008	20%	56%	55%	42% +/- 3%
2009	21%	47%	43%	37% +/- 3%
Averages	20%	52%	54%	41%

The young people were also asked how often they used sunbeds. As could be expected, the age group 12-15, reported less use compared to users 16 years and older. Among the former, who used sunbeds at least once, around 15% used them monthly or more often, while some 28% of users in age group 16-19 and 24% of those aged 20-23 did the same (averages for the whole period). A corresponding number for 16-75 years old in 2004-2007 is reported as 19% in another study [1].

Figure 3 shows how the number of sunbed users has decreased for the population of young people aged 12-23. The decrease is 20% from the year 2004 to 2009 and statistically significant.

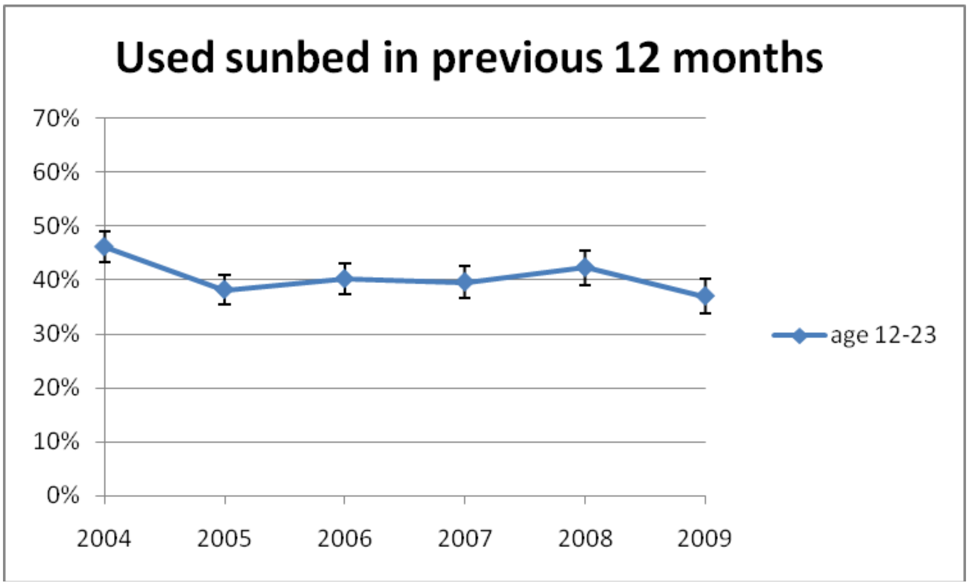

Fig. 3. Percentage of young people (age 12-23) that said they used sunbeds at least once the previous 12 months, both sexes and rural locations included. A 95% confidence interval is shown.

Figure 4 displays the change over time in sunbed usage for the three groups. The number of users 12-15 year old was almost halved the first year of active information dissemination, but it has grown since to similar level as it was earlier (the difference is statistically not significant). The number of sunbed users in age group 20-23 year old was however reduced by 30%.

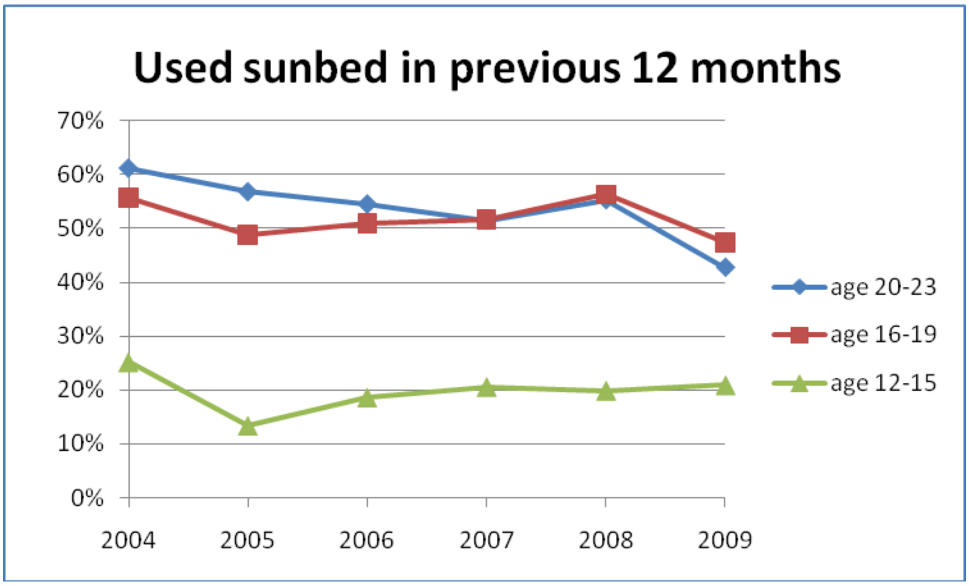

Fig. 4. Sunbed usage of three age groups. Confidence intervals are not shown but can be estimated to be ~√3 wider than in Figure 3.

Discussion

The results presented here are in line with data gathered by different means, see reference [1]. Information on sunbed numbers indicates that sunbed usage has been very common in Iceland but it has been reduced in recent years. It does not come as a surprise that sunbed usage was sharply reduced by children at confirmation age in 2004, it is more surprising how resilient this usage has later proven to be. The drop in sunbed usage reported in 2009 seems mostly to have affected adults and could be due to the sudden economic turmoil in Iceland in the late 2008.

The results seem to be internally consistent but there is no other data on the usage by 12-15 years old in Iceland with which to compare it.

In 2005 Nordic radiation protection authorities issued a common statement [2] strongly advising young people not to use sunbeds. The radiation protection agencies of Iceland, Norway, Finland and Sweden issued another statement 11th of November 2009 recommending regulation of tanning facilities that includes the prohibition of use for people below 18 year of age [3]. This recommendation was unanimously supported by the Icelandic UV-task group.

The government of Iceland decided in March 2010 to introduce a legislation to ban sunbed use by people less than 18 years of age. This legislation is now in preparation.

Conclusions

According to this study, 20% of 12 through 15 years old Icelanders used sunbeds in Iceland in 2004-2009. It has proven difficult to permanently decrease this usage through information campaigns. The campaigns have however been effective in reducing usage by adults who were not their primary target.

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Evaluation of low frequency magnetic field exposure system for ICDs for in vitro studies

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Abstract

This paper presents an in-vitro characterization set up of implantable cardiovertor defibrillator (ICD) immunity. A set up has been modeled by using a Helmholtz coil and an electric ad hoc interface. It generates and controls uniform magnetic fields. A comparison was made between the file that contains the parameters of emitted disturbances at a given time and the time of the occurrence of each event recorded by the ICD. This has allowed us to deduce the signal level that caused false detections in one case, or inhibition of the detection in the other case. Evidence is that the higher the level of disturbance, or when the frequency is in the "cardiac" frequency band, the number of false detections increases. One result is an amplitude "window effect" for the detection and dysfunction.

Introduction

Risks of electromagnetic fields (EMF) exposure associated with patients bearing medical implants such as the implantable cardiovertor defibrillator (ICD) are under a concern since this population is increasing. Cardiac devices are the most common medical implants(Nuegell et al 1996, Occhetta et al 1999) and are subject to specific international standards. However, since the European directive 2004-04 CE on the limitation of a worker's exposure to EMFs, supplementing the 1999/519/CE recommendation, many questions remain unanswered. Establishing a risk assessment procedure for workers with medical implants is very complex due to the diversity of potential situations involving EMF interactions in the professional environment. The effect of radio frequencies (RF) on ICD's has been widely investigated (Inrich et al 1996, Guertin et al 2007) and there are many papers in the literature describing the electromagnetic interference between ICDs and RF systems such as medical devices (X-rays), cellular phones and security systems (Inrich et al 1996, Mathew et al 1997, Shellock et al 2007).In addition, the effects of extremely low frequency (ELF) fields (e.g those generated by power lines for transport such as railways or for power supplying) have also been considered. The ICD bearer, until a few years ago, was typically excluded from his job after his implantation. Since ICD may concern also young people now, this situation implicates a real social problem when they come back to their work after surgery. Research has been performed on ICDs in the proximity of high voltage power lines (Souques et al 2002) and the susceptibility of implantable medical devices to the EMFs generated by electric security systems has also been investigated, highlighting transient anomalies (McIvor et al 1995, Mugico et al 2000). Nowadays, the average age of ICD bearers is decreasing, therefore, an ICD bearer could be working in a factory where high power machines generate EMFs. For example, in the proximity of high power welder machines (75 kVA), the magnetic field ranges from 100 µT (at 1.5 m) to 2000 µT (at 0.15 m) (Cooper et al 2002, Fetter et al 1996). In this sense, particular effort is devoted to analyze the electromagnetic interference (EMI) effects of ELF fields on cardiac implants, both from an experimental (Della Chiarra 2006) as well as from a theoretical simulation approach (Dawson et al 2002) .The exposure to uniform electric field and contact currents has also been investigated(Dawson et al 2000). Assessment of human exposure at the workplace for persons bearing active implantable medical devices in electromagnetic fields is thus a current challenge for all the concerned electrical industries. Previous relevant studies have already been conducted on pacemakers in vitro at the Laboratory of Electronic Instrumentation of Nancy (Nancy University) for frequencies of 50/60 Hz, (Scmitt et al 2005). In this paper we describe a method for the in vitro characterization of the immunity of ICD to electromagnetic interferences at low frequencies (50 Hz-60Hz).

Methods and materials

An experimental set up (Katrib et al 2009) using a source for environmentally relevant interferences consists of a programmable low frequency generator connected to a power amplifier then to a transformer to deliver a sinusoidal current connected to Helmholtz coils (uniform magnetic fields) to simulate the EMF. Helmholz coils and the device under test (DUT) are placed in a Faraday cage (fig 1).

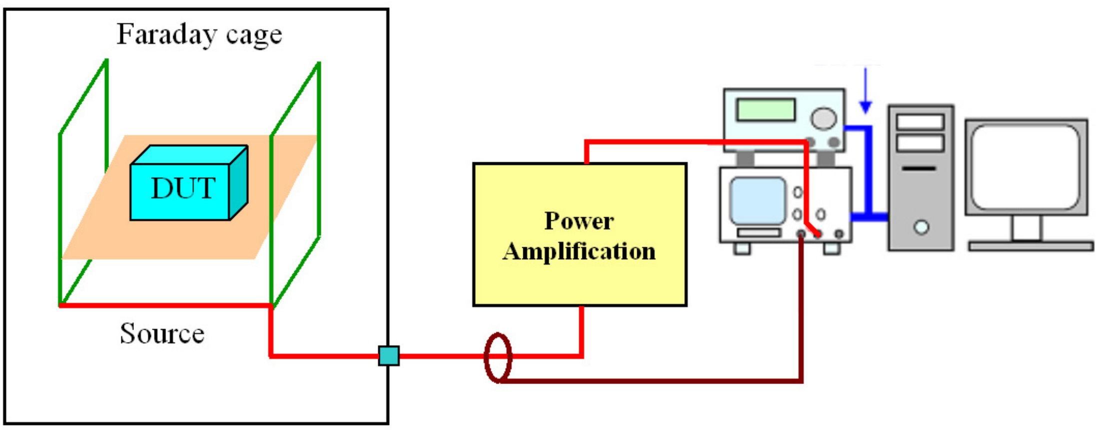

Figure 1 Experimental set up for testing the ICD immunity.

The entire set up is controlled with a PC by an IEEE488 connection and a data acquisition program that can control and communicate with a piece of test equipment to adjust the amplitude levels and frequencies of the EMF. The acquisition of these parameters is synchronized with the clock of the DUT, by a telemetric controller. The oscilloscope allows the monitoring of the signals during routine settings. The exposure system presented here offers the possibility to investigate the immunity of ICD to magnetic fields by placing the device, together with its leads, inside a homogeneous testing field. A Plexiglas support, holding the ICD and the leads in a well-defined position, is placed between the induction coils so that the DUT is positioned in the homogenous field volume for tests. This support may be used in the air as well as in a Plexiglas tank that can be filled with saline solution or gelatin mimicking electrical properties biological tissues (Marchal et al 1989). In latest study Silny (Scholten et al 2001) estimated the closed loop formed by the metallic housing and the tissues to 225 cm ^{2}, in unipolar detection. In this case big phantom was needed (45*30*24.5 cm) to simulate a realistic situation. In the bipolar detection we don't have this problem as the distance between proximal and distal electrodes does not exceed 2 cm. Our proposed tank simulator has a height of 12 cm and a length of 28 cm. Finally, the induction coils are placed on a wooden support to avoid distortion of the magnetic field.

Results

Figure 2 presents the electrocardiogram (ECG) registered when testing the ICD with a lure heart signal simulating a ventricular fibrillation. In this study, we apply an arrhythmic signal (fibrillation or tachycardia) during 15s followed by a normal rythme signal during 45s, so that the ICD is able to recover between each arrhythmic episode.

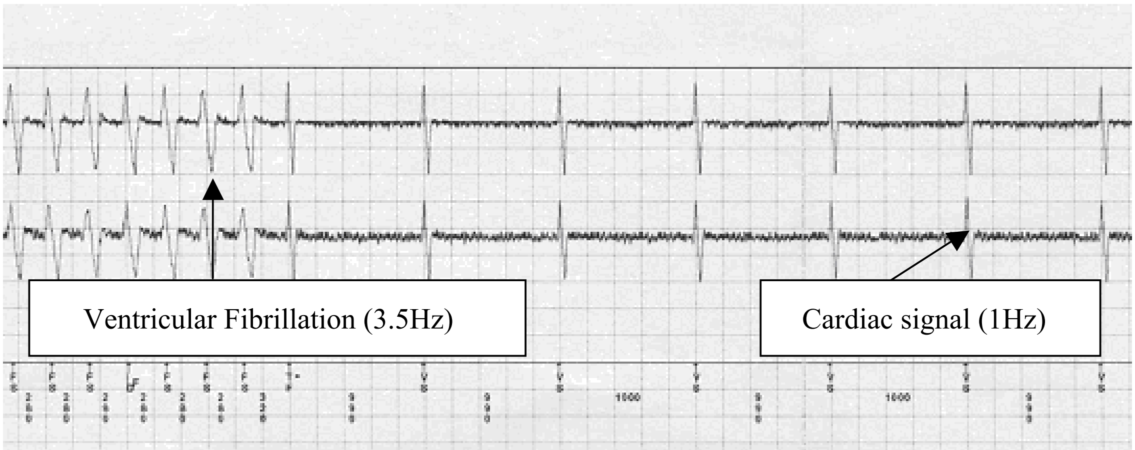

Figure 2 Chart of ECG during a simulated fibrillation episode.

Once the physiological signals are set up, we study the behaviour of ICDs in the VVI mode in order to plot the immunity of ICD versus applied magnetic fields. For magnetic fields up to 1000 µT, which is 2 times higher than the standard threshold for occupational exposure to magnetic fields, no interaction was found. Above 1000 µT, figure 3 and 4 presents the behaviour of ICD exposed to different inclinations angles of the magnetic field with respect to perpendicular exposure. One can remark that when the inclination angle increased, the disruptive level decreased. At an inclination angle of 19°, the inhibition (pathologic signal undetected) occurred at the disturbance level of

2570 μT for 50 Hz and 2640 μT for 60 Hz. On the other hand at 90°, the inhibition began at 1750 μT for 50 Hz and 2020 μT for 60 Hz. We still remark that most of the ICD are non-sensitive to magnetic fields below 1400 μT, where the induced voltage does not exceed the limit of 2 mV according to the standard (EN54505). All the interference threshold curves measured until today have shown that the filter structure or input/output systems used for ICDs have a major influence on their behaviours faced with EMF. The pass band filter structures were found the most sensitive at frequencies between 10 Hz and 100 Hz, and especially at 60 Hz where we can see a lot of interferences. Bipolar electrodes are the most effective remedy for low frequency interferences compared with unipolar electrodes. The present research results substantially improve the currently unsatisfactory situation with regard to the assessment of the safety of persons with active implantable medical devices who are exposed to magnetic fields in the low frequency ranges at their workplace.

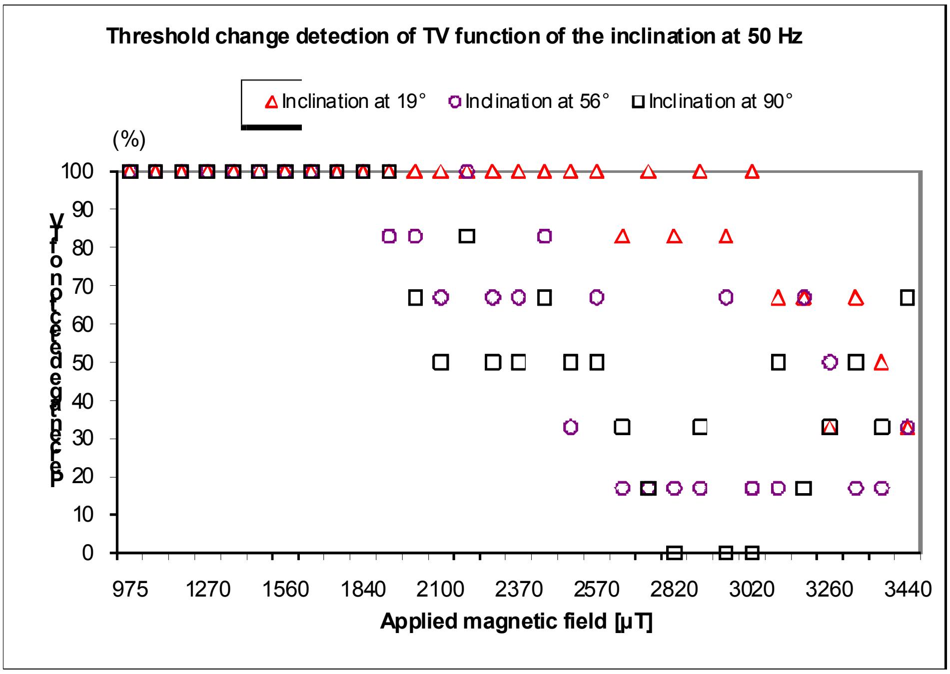

Figure 3. Behaviour of ICD in relation to magnetic field inclination at 50Hz.

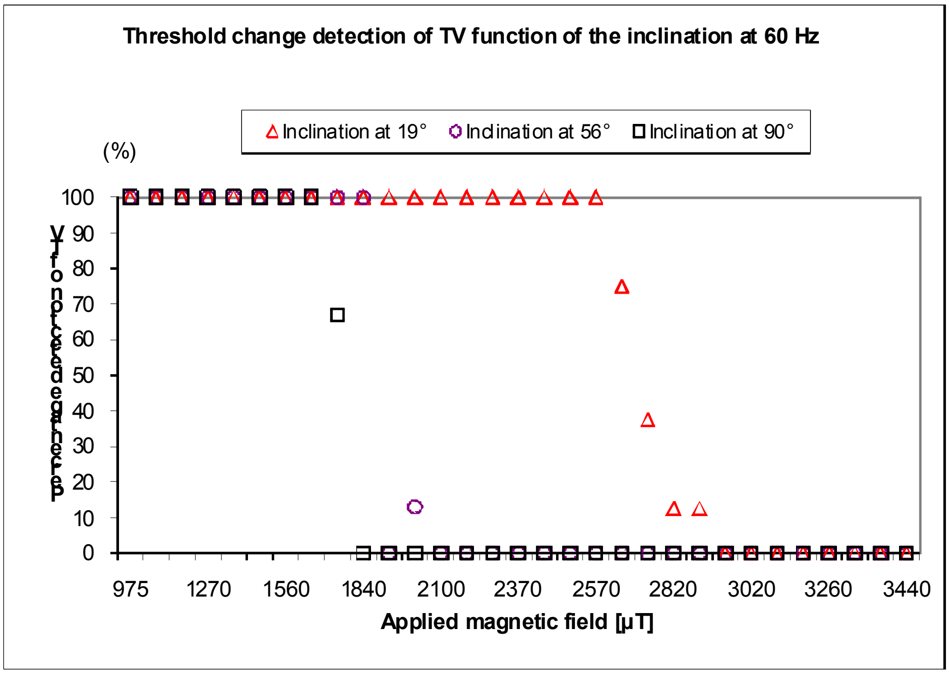

Figure 4. Behaviour of ICD in relation to magnetic field inclination at 60Hz.

Discussion

In this paper we present a protocol and setup dedicated to test the immunity of ICD exposed to low frequency homogenous magnetic fields. We present the methodology used to generate a pathological signal, and we show the influence of the inclination angle on the implant at 50 Hz and 60 Hz. In future work we will test more ICDs at test frequencies up to 100 KHz, and it would be efficient to also test a modulated signal as stated in the standard (EN54505).

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The main directions of the Russian efforts in radiation safety and health protection

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Abstract
The status of radiation protection and radiation safety at the Russian hazardous enterprises is assessed as satisfactory. According to data of long-term monitoring, annual dose to nuclear workers is not higher than 2.5 mSv. Contribution of the nuclear industry to the public doses is not higher than some parts of the percent. Over many years, any excess radionuclide discharge/effluent has not been registered and chemical contamination of the common air and water environmental media do not exceed the permissible concentrations. Indexes of the public health are similar to those for the whole Russian Federation. Chronic occupational morbidity rates are lower than 2.0 cases per 10 000 workers. The acute occupational pathology occurs very seldom, mainly due to incidents. Over 60 years of the nuclear industry history, 754 persons have been radiation injured, 350 persons of them had diagnosis acute radiation disease; amount of radiation-induced deaths was 71 (taking into account consequences of the Chernobyl accident). Today, Russia must develop the national system for control and keeping radiation protection at the required level. This paper will give some examples of the current radiation protection problems important for public health and scientific results of research on the health effects of long-term, low-dose radiation exposure.

Introduction
The Foundations for the state policy in nuclear and radiation safety assurance in the Russian Federation for the period till 2010 and future perspectives (approved by the RF President 4 December 2003) serve as a basis for development of radiation safety assurance and public health protection in respect to impact of radiation factors in Russia. The Foundations for the state policy are implemented in work of the federal authorities. The Federal Medical Biological Agency (FMBA of Russia) is one of them. Since 1946 up to now, the FMBA of Russia develops practically all issues of radiation safety and health protection in Russia.

The FMBA of Russia is a federal authority, which was assigned the responsibility for medical sanitary support and state sanitary epidemiological supervision both of workers from some industrial institutions with especially hazardous work conditions and residents of some Russian areas. The listing of such organizations includes: the state atomic energy corporation "Rosatom", state shipbuilding corporation, federal agency of marine and river transport. They have nuclear and radiation hazardous facilities and cover 22 settlements situated in the supervision areas of the mentioned facilities. Moreover, the FMBA of Russia regulates safety during nuclear energy using and it is one of the customers and coordinator of the medical part of the Federal target program «Nuclear and radiation safety assurance for 2008 and for the period till 2015».

The structure of the FMBA of Russia includes research institutes specialized in radiation medicine and biology; clinics; institutions under the sanitary epidemiological supervision system, which institutions are located according to the principle of optimum dislocation with respect to organizations with especially hazardous work conditions. Thus, three interconnected components – science, medical support and regulation – predetermine the foundation for radiation safety assurance system. The special feature of this system is its complex functioning: from surveillance, control and supervision of work conditions at radiation hazardous facilities and population living at the areas of their allocation, medical aid rendering and to scientific support of radiation safety and health protection problems.

Current status of radiation safety

Annual analysis of the operator's radiation passports and territorial bodies' reports shows that today, ionizing radiation is not the key health affecting factor both for workers of radiation facilities and population living at the supervision areas. Over many years, excess radionuclide discharges/effluents have not been registered. Contamination of common air and water environmental media with different chemicals do not exceed the maximum permissible concentrations. Radiation safety conditions at the facilities under the State Corporation "Rosatom" are evaluated as satisfactory. Long-term monitoring shows that annual doses to nuclear workers do not exceed 2.5 mSv (Fig. 1).

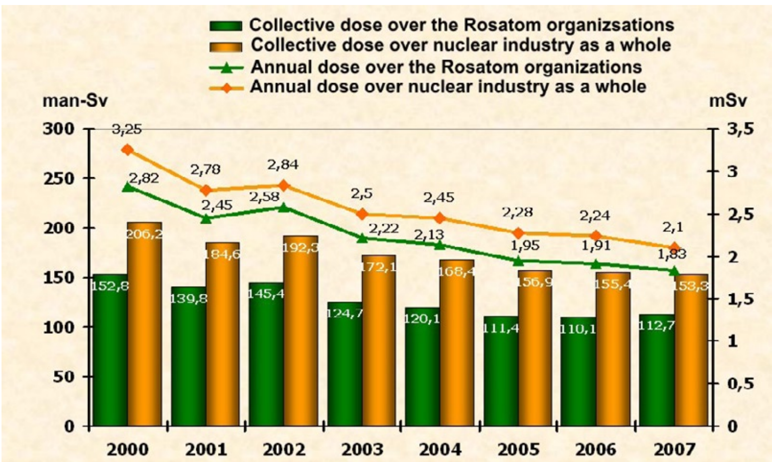

Fig. 1. Dynamics of annual and collective doses to nuclear workers of Russia.

Comparison of some demographic indexes of the public health in Russia as a whole and those for the contingent under the FAMB of Russia service (Table 1), shows that the birth rate here is lower than the similar index over the Russian Federation. This fact can be explained by the age structure of residents in the close territorial formations, where the older residents dominate. The morbidity found is also higher a bit because of improved medical care. At the same time the death and infant death rates are much lower than the similar indexes over Russia due to timely revelation and treatment of different pathology.

Table 1. The public health indexes for the population under the FMBA of Russia service.

Index (over 1000)	Over FMBA of Russia	Over Russia as a whole
Birthrate	8.8	11.3
Public morbidity:		
All residents	1740	1703
Adult residents	1600	1360
Total mortality	11.5	14.7
Infant mortality	6.9	9.4

Over the recent years, indexes of the chronic occupational morbidity at the facilities serviced are lower than those over the Russian Federation. The acute occupational pathology occurs very seldom, mainly due to emergencies.

Table 2 includes data on clinically significant consequences of radiological incidents in territory of the former USSR and Russia, according to information from the FMBC Register. Over the 60-year period of the nuclear industry existence, 754 persons were radiation injured, among them 350 cases of the acute radiation disease has been registered; amount of radiation induced deaths was 71 (taking into account consequences of the accident at the Chernobyl nuclear power plant).

Table 2. Information on clinically significant consequences of the radiological incidents in territory of the former USSR and Russia (as of 01.01.2009).

Classification of incidents	Number of victims with clinically significant consequences		
	Total	Among them: acute radiation disease	Among them: died
Radioisotope installation induced incidents	170	51	16
X-ray installation induced incidents	50	-	-
Reactor incidents (without the accident at the Chernobyl NPP)	82	73	13
Incidents at the PA «Mayak» (1949–1956)	168	-	-
Accidents at nuclear submarines	133	85	12
Other incidents	17	7	2
Accident at the Chernobyl NPP, 1986	134	134	28
Total	754	350	71

The public effective doses originated from operation of facilities in the supervision area were lower than minimum value of 0.01 mSv.

Radiation safety problems significant for health protection

There are some problems in the nuclear industry connected with the ageing primary funds; isolation of some shallow facilities for radioactive waste storage from the environment; large volume of liquid radioactive waste; a lack of reserve in the facilities for the spent nuclear fuel storage at some nuclear power plants; and radioactive waste storage facility under such plants management etc.

Some problems originate from past defense activity of the specific facilities and radiological accidents there. Unfortunately, such accidents resulted in severe consequences for workers, population and environment. First of all, we mean here "the first-born of the nuclear industry" – the production association "Mayak", which got the very difficult research and industrial tasks to produce nuclear weapon at the late 1940s.

The South Urals problems

The most serious consequences for the public health were induced by the radioactive waste discharges into Techa River and the accident at the PA "Mayak" of 1957. Since 1949, some part of the waste water containing radioactive substances, the gross activity of which was about 3 mln. Ci, has been released into Techa River and till July 1951, such releases were practically uncontrolled. This resulted in high radioactive contamination of the river system and radiation exposure to residents of the coastal settlements at doses of 4 - 5 Sv, which induced 940 cases of the chronic radiation disease over the first two years after the releases. In 1957, the explosion of the tank for the liquid radioactive waste storage resulted in radioactive contamination of the large area of the Urals region (the Eastern-Urals radioactive track) and the public exposure at 0.9 Sv.

Table 3 includes the summarized specification of the PA "Mayak" register. Inspection of the health conditions of this contingent revealed such late effects as increasing mortality due to malignant neoplasms and leukemia of workers being employed over 1948-1958. Over those years, learning of the new technology was accompanied with significant over-exposure of workers. After 1958, the cancer death rate was twice lower in comparison with average national data.

Table 3. The register of the PA "Mayak".

Indexes	Time of employment		
	1948-1958	1959-1972	1948-1972
Total number of individuals	12.692	6.154	18.846
Number of persons-Plutonium carriers	9.423	4.997	14.420
Number of persons with known life status	92.1%	96.3%	93.5%
Dose, cGy	112.8	18.9	83.4
Amount of Plutonium accumulated, kBq	2.5	0.4	1.7

The experts from the Urals Research and Practical Centre of Radiation Medicine and South-Urals Institute of Biophysics made the following conclusions:

1. The late radiation-induced effects such as increasing death due to malignant neoplasms and leukemia have been registered both of some residents of the Techa

River coastal villages, in Chelyabinsk region, and injured to the highest doses over the period of the highest releases, and of workers employed over 1948-1958.

2. The long-term observation of descendants of the exposed persons living at the coastal areas of Techa River and in the territory of the Eastern-Urals track did not demonstrate any changes in the public health.
3. The radiation situation around the Techa Reservoir System is now under control and causes radiation exposure neither to workers nor to the population, except for the critical population group of Muslyumovo village.
4. Because of increasing filtration of radioactive substances from the Techa Reservoir System, stabilization of concentrations in the river water is being registered. At that, the ^{90}Sr specific activity within the Muslyumovo village area is permanently higher than the intervention level established by the NRB-99 (5 $Bq \cdot l^{-1}$). The ^{137}Cs concentrations in the river water are lower and generally do not exceed the maximum level of the ^{137}Cs content in water of 11 $Bq \cdot l^{-1}$.
5. The ^{90}Sr specific activity in milk became 10-15 times lower since 1958 up to now. At the same time, since 2003 the steady tendency to the ^{90}Sr increasing in milk is obvious due to increased concentration of this radionuclide in the river water. In samples of the cow milk being pastured in the floodplain and drunk from the river the ^{90}Sr content in milk is higher than the permissible limit.

In this light, life in the settlements at the Techa River, especially near Muslyumovo village is potentially hazardous. In 2006, the FMBA of Russia made the associated conclusions and recommendations and sent them to the Rosatom. Our conclusions were used to justify the decision making, jointly with the Chelyabinsk regional authorities, to re-settle the residents from this village to the safer place.

Peaceful nuclear explosions

Today, evaluation and regulation of radiation safety in the regions affected by the peaceful nuclear explosions is very important radiation hygienic problem. More than 80 underground nuclear explosions have been made in territory of 16 Russian Federation subjects over the period since 1965 to 1988, with the peaceful purposes. At that, 79 explosions were confined (camouflet), after which neither radioactivity release into atmosphere nor environmental contamination assumed to be taken place. At the same time, during two explosions rather significant release of radioactive products into atmosphere and environmental contamination really took place. So, after Craton-3 explosion of 1978 in the Republic of Sakha (Yakutia), the radioactive track has been generated, the length of which was several kilometers. A wide range of problems are to be solved, including:

- Uncertainties in the legal status of the sites where the peaceful nuclear explosions were being performed;
- A lack of principles for distribution of responsibilities for safety at such sites;
- The public concern in the vicinity of such sites, in addition to concern of workers from the facilities functioning at such sites.

Radon in dwellings and on the industrial floor

In some subjects of the Russian Federation, the population is under significant exposure originated from high contents of radon 222, thoron 220 and their short-lived airborne progenies in dwellings. The situation at two areas under the FMBA of Russia service is unsatisfactory. These are: Octyabrsky village of Chita region and Lermontov city of the Stavropol Territory.

Octyabrsky village of Chita region began to be built in 1964, as a temporary residence of members of the geological prospecting expedition and builders. The village appeared to be situated at the area of the mining dump of the uranium mining-milling facility, as the Priargunsky mining chemical association was being built. It was situated just in the health protection zone of the facility surrounded by the units of the mining complex and other industrial units of the Combine. The experts of the Burnasyan Federal Medical Biophysical Centre inspected the radiation hygienic situation in Octyabrsky village. The excess of the radon regulation (200 Bq/m3) has been registered in 39 % dwellings (Figure 2). The average dose induced by radon intake in rooms was 14 mSv/year. The State Corporation "Rosatom" and Zabajkalsky Government decided to resettle 2000 residents to the safer place.

Unfortunately, over quite long-term period, the FMBA of Russia cannot decide the similar problem in Lermontov city of the Stavropol Territory, where about 1000 dwellings are radon-unhealthy, i.e., radon concentration in dwellings are higher than the authorized norms, so, the residents of the first and ground floors are under excessive exposure.

The FMBA of Russia recognizes challenges in radiation protection of workers and the public originated from the former "uranium" legacy of the USSR. Such problems assume to be solved under the inter-state target program of the European and Asian Economic Community (EurAsEC) «Reclamation of Territories of the EurAsEC states affected by impact of uranium mining and milling facilities».

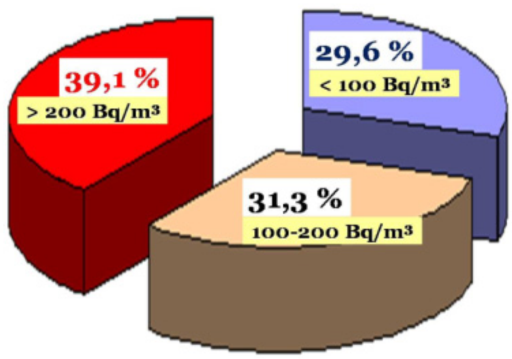

Fig. 2. Radon equivalent equilibrium activity concentration in dwellings, %.

The uranium mines located at the Russian area require the special attention and efforts of the FMBA of Russia. It is well-known that uranium mining and enrichment in mines is one of the most dose forming operations. So, at the Priargunsky mining chemical combine, the permissible annual dose limits established for workers are systematically excessive during uranium mining, mainly, due to excessive contents of radon and its decay products at the working area (the mine). At that, number of workers injured to annual dose higher than 20 mSv increases with years: over 2006 - 30, over 2007- 101 persons. In addition, according to the expert assessments of the FMBA of

Russia, the lung cancer of the uranium miners is 4 times more frequent in comparison with the mean population data.

Consequences of the accident at the Chernobyl NPP

According to data of the United Nations Scientific Committee on Effects of Atomic Radiation, in fact, the single science based consequence of radiation exposure to the public resulted from the accident at the Chernobyl NPP is significant increasing incidence of the thyroid cancer in persons who were children at the moment of the accident. Maximum absorbed dose to the thyroid was 50 Gy. Among those being exposed in childhood, more than 6000 cases of the thyroid cancer have been registered (by 2005, 15 deaths has been registered). Therefore, the FMBA of Russia solves such important health protection problems as arrangement of medical dose register for the Russian residents exposed to emergency radioiodine fallouts of 1986, reconstruction of doses to the thyroid, and maintenance of the register including information on persons participated in the accident consequences mitigation.

Peaceful nuclear renaissance

After the accident at the Chernobyl NPP, the requirements for the NPP safety became much stricter. The main public requirement was to assure guaranteed safety not only for the plant workers, but also for the residents of the NPP impact area. In this light, one of the most relevant scientific and practical tasks is to justify the public dose levels safe in respect to potential induction of late stochastic (cancer and genetic) consequences. For the purpose of such assessment, some large-scale radiation hygienic and epidemiological studies are to be performed at different stages of the NPP operational life. In this light, the FMBA of Russia develops the special unified method for performance of overall monitoring of radiation hygienic situation and public health in the vicinity of nuclear power plants. The methods suggested will help to compare the observed change of the health and environment condition indexes with the ”background” values, to evaluate their dynamical trends, to give quantitative risk assessments of radiation exposure, to justify suggestions on improvement of the public and environmental protection conception and to assure safe operation of the NPP.

Nuclear legacy

The FMBA of Russia jointly with the experts from the Norwegian Radiation Protection Authority participates in environmental remediation of radiation hazardous sites in Murmansk region of the Northwest Russia. These operations include management of the spent nuclear fuel (SNF), solid and liquid radioactive waste (RW) both accumulated over the period of the Navy activity and originated from dismantlement of nuclear submarines and above-water ships equipped with nuclear powered installations. As a result, some regulatory documents have been developed and radio-ecological monitoring is being carried out at the sites for the SNF and RW temporary storage (STSs) located in Andreeva Bay on the Kola Peninsula and in Gremikha village in the coastal stripe of the Barents Sea.

The analysis of radiation-hygienic situation taking account of the data obtained, leads to the conclusion that a long and extensive remediation program is necessary after removal of SNF and RW from the STS’s. Having in mind up-to-date approaches to radiation protection, identification of remediation strategies has been focused on justification of reduction of the occupational and public exposure originated from the residual contamination. The following have been defined (Figure 3):

- Conservation (storage under surveillance) – excludes the potential threat of contamination of the STS territory, water area and air media. A guarded area is arranged and continuous radiation monitoring is carried out.
- Conversion (renovation) – suggests subsequent use of the STS territories and facilities in compliance with the existing regulatory documents regulating the radiation impact on the personnel and public under normal conditions of radioactive source management. Limited use of the territory in combination with remediation measures and radiation monitoring ("brown lawn" concept) is envisaged.
- Liquidation – suggests stage-by-stage dismantlement and removal of equipment, withdrawal of RW, including contaminated environmental media, and assures the limited exposure dose to the critical group of the public at the level 1 mSv/y ("greenfield" or unlimited use conception).

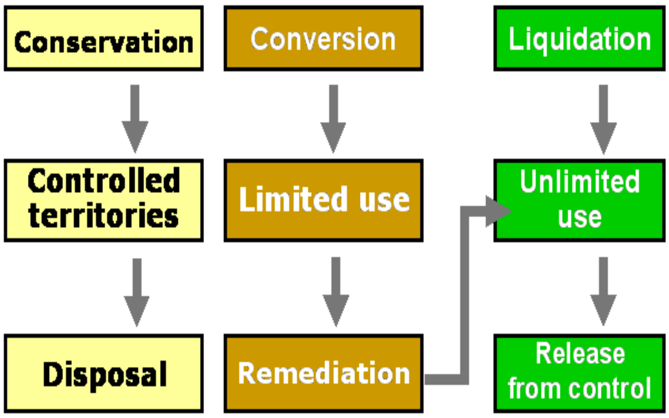

Fig. 3. Restoration options at the sites for the SNF and RW temporary storage.

FMBA of Russia approved the norms for the main options of the STS remediation. They have been developed on the basis of the relevant Russian laws and standards, taking into account the current radiation situation at the STS as well as up-to-date international recommendations and good experience in the field of contaminated area remediation in other States.

Conclusions

The available Russian and international legislative and regulative legal basis as a whole helps to solve problems of radiation safety assurance during the special and hazardous operations, enhances and supplements the principal legislative documents. At the same time, some difficult problems in the regulatory documents are to be solved with respect to operation of the national nuclear industry:

- Remediation of sites contaminated following past nuclear activity and "nuclear legacy";

- Decommissioning of nuclear facilities or justification of prolonged operational life of the functioning ones;
- Radioactive waste management;
- Support of effective emergency response;
- Difficulties in adequate application of the optimization principle.

Russia is not a passive onlooker of the world radiation safety problems to be solved according to the UNSCEAR priorities. They include:

- Medical exposure to patients;
- Radiation levels and consequences of nuclear power production;
- Exposure originated from the natural radiation sources;
- Improved understanding of consequences of radiation exposure at low dose rate.

Today, the accumulated scientific potential of the experts from the FMBA of Russia, the experience in radiation safety regulation and medical sanitary support, as well as the factual materials available enable to solve the most difficult tasks in radiation safety.

Spanish national campaign for the search and recovery of orphan radioactive sources

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Abstract
This paper aims to describe the main approaches, the steps taken, the experience gained and the results obtained in the campaign for the "Search and Recovery of Orphan Radioactive Sources" that took place in Spain between February 2007 and December 2009. The paper aims to share the experience gained with others who are considering or are already involved in similar campaigns, and to enable opinions to be exchanged with those responsible for such campaigns in other countries. The campaign was initiated by a decision of the Spanish Ministry of Industry, Tourism and Trade (Ministerio de Industria, Turismo y Comercio, MITYC) with the expert assistance of the Nuclear Security Council (Consejo de Seguridad Nuclear, CSN) in compliance with the national legislation currently in force regarding the control of highly active and orphan radioactive sources, and was commissioned to the Spanish National Company for Radioactive Waste Management (Empresa Nacional de Residuos Radiactivos,S.A., ENRESA). The campaign tried to seek and recover the largest possible number of orphan radioactive sources and involved the collaboration of various different agents and organisations where such sources could be found. The paper provides details regarding the number and radiological characteristics of the sources which have been recovered in Spain during the campaign. Experience shows that, even though a suitable regulatory framework exists, as is the case in our country, orphan radioactive sources may still exist for various reasons. Several other actions taken place in Spain previously or in parallel to this campaign were and continue to be very positive in regards to the recovery of control of the majority of radioactive sources existing in the country. It has to be emphasised that in this subject complementary international concerted efforts are always needed and welcome.

Introduction
According to the Spanish regulation, an orphan source is a radioactive source with an activity above the exemption level established in legislation, which is not under regulatory control, either because it has never been or because it has been abandoned, lost, misplaced, stolen, or transferred without proper authorization.

Orphan radioactive sources have caused deaths and injuries in several countries, when members of the public, without knowledge of their potential risk, have found and handled them in an inadequate way. There is also concern that orphan sources may be acquired for malevolent purposes. Finally, the economic impact of an accidental fusion of an orphan source within the metal industry can be considerable. All these reasons have led many countries to consider making efforts to improve the control over them.

Orphan sources are mentioned in the European Union Legislation. The Council Directive 203/122/Euratom on the control of high-activity sealed radioactive sources and orphan sources, Art 9.4, states that "Member States shall ensure that campaigns are organized, as appropriate, to recover orphan sources left behind from past activities".

Orphan sources are also mentioned in Spanish Legislation. The Royal Decree 229/2006 on high-activity sealed radioactive sources and orphan sources, Art 11.4 says that "the Ministry of Industry, Tourism and Trade , with the expert assistance of the Nuclear Safety Council , shall organize, as appropriate, campaigns to recover orphan sources left behind from past activities".

Financial aspects are also mentioned in both legislative documents. According to CE Directive, Art 10, member states "shall ensure that, on the basis of arrangements to be decided by them, a system of financial security is established or any other equivalent means to cover the intervention costs relating to the recovery of orphan sources". Similarly Royal Decree 229/2006, Art 12 says: "the costs stemming from managing orphan radioactive sources, in addition to the measures taken to recover such sources or to deal with radiological emergencies caused by them, shall be defrayed by the last owner of the source, providing that the owner can be identified. If identification is not possible, these costs shall be covered by the owner of the facilities in which the source is detected".

This specific national campaign was launched in Spain as a result of the initiative of the MITYC, with the expert advice of the CSN. The implementation of the campaign was entrusted to ENRESA, and it was developed between February 2007 and December 2009.

The aim of the campaign was the search and recovery of the largest possible number of orphan radioactive sources existing in the country, based on the voluntary cooperation of those agents and organizations where the sources were or might be found, and taking in consideration the Spanish experience and "nuclear" history.

Some of the main characteristics of the Spanish National Campaign were that it was conceived as a preventive action (working in parallel with other on-going corrective campaigns as the metal industries "Protocol"); it focused on past activities, giving preference to radioactive sources of higher activity but covering all type of sources; it was addressed to all sectors of use or potential use; and was geographically broad- based (national). The campaign was funded by the General State Budget but was not free for users or owners. Fares applied were the same as those charged by ENRESA for the management of disused sources in regulated facilities.

Material and working procedure

Since the starting date of the campaign, several core activities were carried out, following a basic initial and very general publicity performed by the MITYC, including the addition of specific information on the campaign in the electronic Web pages of MITYC, CSN and ENRESA.

The initial step taken in the campaign was the creation of an Advisory Committee to support the task of ENRESA, formed by senior members from different organizations, selected for their experience. This Committee was basic for the review of the current and past history of the regulatory control applied, including the management of disused sources in Spain; the detailed analysis of other actions promoted by Spanish authorities in the past regarding orphan and vulnerable sources; the data gathering and evaluation of the historical national information available regarding the uses of the radioactive sources; and the identification of fields of activity likely to have used and continue to posses orphan sources. This Committee met in several occasions during the campaign for a follow up of the action plan established.

a) Review of the current and past history of the regulatory control and management of disused sources in Spain

According to IAEA guidance it is always important for orphan sources campaigns the evaluation of the regulatory history in the country.

Spain has since 1982 a well established Regulatory Agency, the CSN. Before this date, from 1964 to 1982, the Nuclear Energy Board (Junta de Energía Nuclear, JEN) used to act as the Regulatory Agency.

There has always been a rather good degree of control of radioactive sources in the country. Regulations have been implemented and applied since long ago. Licenses granted to users included the identification of the specific radionuclides and activities authorized, as well as provisions for the complete life-cycle of radioactive sources. The facility closure permissions were not granted until it could be demonstrated that the radioactive material was transferred to an authorized final recipient. There were also import controls of sources and regular inspections to users. Of course, the degree of regulatory control of radioactive sources has been stricter since 1982 and even more since 2006 when RD229 was approved.

The collection of disused radioactive sources for their safe management, including disposal, was the initial responsibility of the JEN, until ENRESA was created in 1986. Although the management system was not really “complete” in Spain until 1986, the system and operational capabilities needed to recover such sources have been in existence much earlier and legal requirements and attempts have always been applied to return radioactive sources to the original supplier, although this has not always been possible. Since approval of RD229 in 2006, agreements with supplier and bank guarantees are needed to get approval for the use of whatever high activity sources. Individual inventories of these sources are also fully available.

The conclusion of this evaluation was that the possibility in some degree, of sources existing in the country from past activities, could not be discarded.

b) Analysis of other actions promoted by Spanish authorities in the past regarding orphan and vulnerable sources

The first actions aimed to recover control of orphan and vulnerable sources were initially performed by the JEN since very early (mostly devoted to the radium needles and tubes in the medical sector) and were additionally pursued since the creation of the CSN in 1982. Since 1986 all the field actions were basically assigned to ENRESA.

The following is a brief summary of the most relevant activities promoted in the past:

– The general campaign for the removal of radium-226 sources, being carried out since 1970's, which has resulted to date in several thousand sources collected and finally re-exported to an authorized management facility in USA, after facing many practical difficulties (dispersion, private use, etc).
– The campaign for the removal of lightning rods headers, based on a change in the regulatory system in 1988. This has also faced quite a lot of practical difficulties (lack of reliable information; dispersion; private use, etc) and has collected to date more than 22500 rods. At present it is practically finished but some 100 headers per year are still being removed. The radioactive sources collected after dismantling the headers were basically sent for recycling to UK.
– The campaign for the removal of teletherapy headers, limited in scope and duration (11 headers between 1989 and 1991) and essentially conceived a preventive action. Since then on, disused units are being re-exported as part of the supply process of new equipments.
– The Protocol for the Radiological Surveillance of metallic materials, signed in November 1999 by the Spanish Government and the industrial sectors, most actively involved in the recycling of scrap metal. The protocol was later endorsed by the main trade unions. More than 130 individual companies have signed the protocol to date. By the end of 2008, almost one thousand different radioactive materials had been detected (around 100 per year). Although a great part of them were just material containing Naturally Occurring Radioactive Materials (NORM), several true radioactive sources have been found and removed from the system.
– The Transfer Authorization, which is an administrative procedure used over decades by the Spanish Authorities as needed, to regain control over whatever radioactive material found "uncontrolled", by authorizing the transfer of responsibility over such materials from the existing possessor to an authorized agent for its further management. To date over 200 transfer authorizations had been granted by the end of 2008 and ENRESA has managed to recover 246 radioactive sources.

As a consequence of all these actions carried in Spain since long ago, the expectations in this specific campaign to find a considerable number of orphan sources was low, altough a "confirmation" was considered worthy.

c) Data gathering and evaluation of the historical national information regarding the use of radioactive sources

From the available official documentation and with the help of the Advisory Committee and the individual relevant people identified, who were involved in the first uses of radioactive sources in Spain, a rather good vision of the use of the radioactive sources in the early years (50´s to 80´s, before the CSN was created) was possible.

Since the 1980´s official inventories of the practices and applications of sources in Spain, and the radionuclides and activities within the practices, were available and could be accessed.

In Spain there are about 1300 regulated facilities of different types. Since the objective of the campaign was the collection with preference of the higher activity sources, a review of users of these sources in the different fields was performed, giving priority to the early pioneers. From this study the following uses of radioactive sources were selected: teletherapy, braquitherapy, blood irradiation, irradiators for research, military applications, calibration laboratories, gammagraphy, geophysical and soil controls, and industrial applications.

Spain has not been a producer or exporter of radioactive sources except for some limited production by the JEN in the early years for local use. Most sources used in Spain were (and still are) imported. This is another condition for the results of this campaign expected not to be huge.

d) Identification of fields of activity likely to posses orphan sources

From the data collected, the following group of users likely to have orphan sources was identified:

- Old public and private hospital and clinics
- Old universities with scientific and technical studies
- Old research centres
- Service companies
- Industries
- Manufacturers and suppliers
- Warehouses
- Waste disposal companies

e) National contacts

The approaches to the identified objectives were made using the following channels:

– Government Ministries and Regional Government Authorities responsible for areas, such as Radiation Safety, Nuclear Power, Education, Research, Defence, Health, Environment, Mines, Agriculture, Transport, Industry and Development.
– Non governmental organizations, professional organizations and societies, cooperative institutions, technical service organizations, other technical organizations and trade groups.
– Official Schools related to medical, sanitary, scientific and technological activities.
– Business Associations in the most representative sectors of the industry.
– Industries within the country by sector of activity (chemical, mineral, oil and gas, paper, glass, wood, tobacco, drinks, etc.)
– Service companies importing sources temporally for specific jobs (non destructive testing, geophysical and soil controls)
– Suppliers of radioactive material or equipment, detectors and laboratory equipment. Companies supplying electro medicine equipments and businesses related to radiation protection
– Transport companies and Customs control organizations

– Specific individuals identified by the Committee with historical knowledge of the first uses of radiation sources in the country (mostly in the medical sector)

The contacts with professionals of these organizations were made using all available means: telephone, letter, direct interview, etc. Neither the broadcast media nor the public directly were ever addressed in the campaign. Distribution of the information through medical, scientific, and industrial associations was very useful to achieve a "multiplier" effect in its re-transmission to members and its dissemination via their own regular media (electronic pages, magazines, publications, etc). Additional promotion of the campaign was made in several ordinary fora organized by these associations and societies as well as in courses related (even if laterally) to the topic of the campaign.

f) International contacts

Contacts with international agencies with experience on similar situations were of great value for the campaign. A visit was paid to the International Atomic Energy Agency (IAEA) to know programs being carried out in the field and to search for potential direct contacts. The IAEA is giving technical advice and support all over the world for recovering orphan sources and holds several geographically specific campaigns. A meeting was also held in Madrid with the USA National Nuclear Security Administration (NNSA) with the aim of sharing information about the initiatives being followed by the United States in the world to recover radioactive sources and to study ways of cooperation. The NNSA has been doing enormous economic and human efforts to give many countries all over the world material, means and technical support to recover uncontrolled sources.

g) Other steps taken in the campaign:

– Creation of easily accessible and understandable documents containing graphic information for an easy identification of orphan sources.
– Distribution of graphic and textual material to institutions, organizations, companies and associations.
– Attendance to congresses and meetings for the promotion of the campaign
– Bibliographical searches, Internet searches and consultation of international publications to trace possible locations for orphan sources.

h) Types of search

In the campaign two types of search were performed:

– Non physical or administrative searches, consisted basically in "detective work" for the collection and analysis of information in order to attempt to identify sectors likely to hold orphan sources.
  Some of the key documents analyzed were:
  - Data on orphan sources collected in the past (national campaigns, transfer resolutions)
  - Historical records of disposal of sources
  - Detections of national origin in the metal industry
  - Lists of early supplies in the country (old files).
  - Data on known lost and found sources
  - Historical documents

- Specific examination of sectors of use.
- Lists of attendees to meetings and congresses of interest
- Mailing lists of specialized societies and associations
- Official reports

– Physical searches, using radiation monitors and according to ENRESA searching plans to locate sources, were carried out in a few facilities suspected to have orphan sources and willing to collaborate in the campaign (warehouses in ports or airports, abandoned facilities, etc).

Results

Attention was paid to all contacts, communications and responses received, whether or not related to genuine orphan radioactive sources. In some cases, the demands were redirected to ENRESA's normal answering system for the ordinary management for radioactive waste or other alternative routes, according to the characteristics of the declared material.

Non-specific consultations were received about disposal routes for:

– X-rays equipment. In Spain specific companies make destruction of tubes.
– Smoke detectors, not covered by the campaign because their management is regulated by RD 208/05, transposing the corresponding European Directive, as electric and electronic equipment.
– Lightning conductors, for which a specific campaign was still active.
– Disused regulated sources.
– Exempt sources, not the objective of the campaign at an early stage but with some allowances later on due to specific circumstances.
– Sources found in those industries and field of activity already covered by the "Protocol" in place for the metal recycling industries.

Each recovery of orphan sources included the following actions, performed using the capabilities available in ENRESA as part of its regular business:

– Analysis of the available information and technical visit to complete the detailed description of the radiological (and any other relevant) characteristics of the radioactive sources declared.
– Acquisition of graphical images of the sources and equipments.
– Preparation of the sources for its provisional storage and future transport.
– Collection of radioactive sources from the declaring premises and transport to ENRESA's facilities in El Cabril (Cordoba).

First sources were recovered in the campaign by the end of 2007, after getting from the MITYC the generic transfer authorization (an administrative requirement whenever a "radioactive material" is to be transferred from one to another possessor).

By the end of the campaign more than 300 radioactive sources have been declared and mostly recovered. In addition to these sources, other radioactive materials have also been recovered, including unsealed sources of various radionuclides, equipments with depleted uranium shielding and several other contaminated pieces.

The data below show the results at the end of December 2009. These are not yet fully the "definitive" results as they do not include the sources declared by the end of December 2009, collected in the first months of 2010.

Classification of sources collected by radionuclide

Radionuclide most frequently found in the campaign was Cs-137, followed by Ra-226, Co-57, Sr-90, Ni-63 y H-3.(Table 1.)

Table 1. Classification of sources collected by radionuclide.

Number of sources	Radionuclide	%
80	Cs-137	30,08
40	Ra-226	15,04
35	Co-57	13,16
21	Sr-90	7,89
18	Ni-63	6,77
14	H-3	5,26
10	Po-210	3,76
9	Th-232	3,38
5	Co-60	1,88
4	C-14	1,50
4	Ba-133	1,50
3	Ra-226/Be	1,13
3	Am-241/Be	1,13
3	Pm-147	1,13
3	Sn-119	1,13
14	Various	5,26
266		100

The figure shows expected values with some exceptions: Co-57, a short lived radionuclide used for calibration of gamma cameras in Nuclear Medicine services; Ni-63, included in laboratory equipment for gas chromatography and Po-210 used in static eliminators in the industry.

Classification of sources collected by activity

Most of the radioactive sources collected in the campaign had activities lower than 1MBq when declared. Nearly 100 sources had activities above this value. The highest single activity was 76000 MBq and the total activity was above 100000 MBq.(Table 2)

Table 2. Classification of sources collected by activity.

Activity	Number of sources
< 1 MBq	168
1- 10 MBq	22
10 -100 MBq	31
100 -1000 MBq	36
1000 - 10000 MBq	8
10000 - 100000 MBq	1
TOTAL	266

Most active sources collected in the campaign included industrial gauging sources, Ra-226 tubes and needles, sources for the measurement of humidity and density of soil, braquitherapy eye and surface applicators and many types of checking and calibration sources.

Activities shown are referred to the date of collection of the sources.

Classification by sector of activity

Sources have been collected in the different types of facilities, namely medical, industrial, research, etc. in a similar percentage (Table 3)

Table 3. Classification of sources collected by sector of activity.

Type	Number of facilities	Number of sources	% facilities
Industrial	24	74	32,43
Research	22	73	29,73
Medical	22	100	29,73
Others	6	19	8,11
Total	74	266	100

The the distribution of sources and facilities is similar, but the sources with the highest activities were collected at industrial facilities.

Classification of the sources collected in the campaign according to IAEA categories

Table 4. Classification of sources collected by IAEA category.

Category	nº sources	%
Exempted	70	26,32
Category 5	184	69,17
Category 4	12	4,51
Total	266	100

One fourth of the radioactive sources collected were exempted when declared, although they were not always exempted in origin. In spite of this, they were collected.

Around 200 sources were not exempted, being mainly categorized as 4 y 5, although some of them were category 3 in origin (Table 4)

Geographical distribution of the sources collected

Although the campaign was addressed to the whole Spanish territory, most of the sources were collected in Madrid area, for two basic reasons: a) Spain was a very “centralised” country in those years and historically the first uses of radiation were promoted from the JEN in Madrid; and b) in some case (army) the sources found were firstly collected in their central services in Madrid, before being transferred to ENRESA lately.

Discussion

Reasons for the existence of orphan sources

The analysis case by case of the reasons found for the existence of orphan radioactive sources shows the following results:

- Sources were an historical legacy (used before regulatory control was put in place)
- Costs of disposal were very high
- No disposal routes were available
- Deliberate fears of regulatory requirements (a few cases)
- Sources had been used in regulated installations without having been declared

Other remarks about the orphan sources recovered

- The sources recovered in the campaign did not represent, in any case, a security risk or a high safety risk at the locations in which they were found. No evidence of malicious practice was detected.
- Work was performed in an effective and discrete way. There was not any negative media impact associated to the campaign. Great interest and collaboration was offered by the different sectors addressed.
- In most of the cases owners covered the cost of the management applied regularly in Spain. In some specific cases holders of the sources found it extremely difficult to meet the costs involved. None of the sources was left behind due to the cost issue.
- The campaign was a good opportunity for owners of orphan sources to “clear houses” in a relatively easy way.

Some lessons learned in the campaign

- The usefulness of carrying out physical inventories in the facilities, including most ancient materials, to find out orphan sources.
- The highly recommended prompt disposal by the regulated installations of disused sources, considered “vulnerable” as they may became “orphan” rather easily.
- Exempt sources were not the object of the campaign, but were recovered in quite a few cases for different reasons. Many users expressed their doubts as how to dispose off this type of sources, even when clear rules were available in the country.

The ending date of the campaign, December 2009, refers only to the pro-active actions carried out specifically. From this date on, the Spanish regulatory system maintains mechanisms to allow ENRESA to continue recovering all orphan radioactive sources declared in future, following specific administrative procedures, as it has been doing up to date

Conclusions

Spanish experience showed that, even though a suitable regulatory framework exists and though it is implemented efficiently by the supervisory authorities, as is the case in our country, orphan radioactive sources may still exist. Experience has shown the usefulness of this type of "proactive" campaign in spite of previous related efforts.

International complementary concerted efforts are always needed and are highly recommended to improve control of orphan sources, because, although there may be a well established regulatory system, no country is completely isolated and free of risk with respect to the inadvertent international movement of radioactive material.

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Mitigation of exposure to radon by household water treatment

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Abstract

Water distribution networks are obliged to carry out monitoring of the water quality. Presently they cover over 90% of the water supply in Finland. There are, however, about 500,000 Finns that rely on private wells for their daily water supply. One of the contaminants that impair the water quality of private wells, especially those drilled in bedrock, is radon. There are presently 6,600–9,000 private wells in which radon concentration exceeds the recommended maximum concentration of 1,000 Bq/L. Due to inaccessible water distribution networks household water treatment units remain the only viable option for mitigating the exposure. In this paper, household water treatment principles for removing radon from water are described together with doses and risks related to waterborne radon among Finnish well-owners.

Introduction

Radon is a radioactive gas that originates from naturally occurring uranium that has been present in the earth's crust since its formation. Due to its solubility to water, it is found in all groundwaters. Water supply plants are obliged to carry out statutory monitoring of the water quality and the maximum concentration set for radon is 300 Bq/L (STUK 1994). Decree 401/2001 by Ministry of Social Affairs and Health recommends 1,000 Bq/L as the maximum concentration of radon in domestic wells. Radon causes radiation exposure through ingestion and through inhalation because radon is partly released into indoor air during water usage. These exposure pathways increase mainly the risk of stomach and lung cancer (NRC 1999).

Systematic surveys of the occurrence of radon in Finnish groundwaters began at the end of the 1960s. In the first phase, bedrock water samples from drilled wells in the Helsinki area were surveyed (Kahlos & Asikainen 1973). Radon concentrations were generally high, with a mean of 1,600 Bq/L, which prompted the surveying of waterborne radon throughout Finland. Based on over 1,500 water samples, the mean radon concentrations in groundwater wells and bedrock water wells at that time were assessed as 60 and 630 Bq/L, respectively (Asikainen & Kahlos 1980).

In the 1980s, the surveys were mainly directed at private wells, which were investigated in co-operation between STUK and Geological Survey of Finland, which performed nationwide studies on the physico-chemical and microbiological quality of well waters (Juntunen 1991). Since that time, radioactivity in all new groundwater sources are measured before they are connected to water distribution networks. In a more recent population-based random study, in which 184 water samples from dug wells and 288 from drilled wells were analysed, the observed mean concentrations of radon were 50 and 460 Bq/L, respectively. In dug wells, all measured radon concentrations were below 1,000 Bq/L, but 10.8% of drilled wells exceeded this level (Vesterbacka *et al.* 2005).

The availability of water services has steadily increased, especially in the more densely populated southern Finland, where the highest radon concentrations are generally found. Finland presently has a population of 5.3 million, about 4.8 million of whom have now access to water distribution services. The remaining 500,000 rely on private wells or have formed small co-operatives for abstracting water (Isomäki *et al.* 2007).

The number of drilled wells has steadily increased. In the 1950s, one well out of 1,000 was a drilled well, but by the early 1990s the portion had increased to 23% (Korkka-Niemi 2001). According to a survey conducted in 2007, the portion of drilled wells was already 32.5% (Vienonen 2007). According to data collected by Muikku *et al.* (2009), the portion of drilled wells was 43.9%. From these, we can estimate that the number of users of drilled wells is 160,000–220,000.

At the end of 2007, 1.01 million households with a total of 2.65 million residents were living in one-family houses (Statistics Finland 2008). From these data, we can estimate that the number of drilled wells in permanent use is about 61,000–83,000 and, considering the concentration distribution of radon, the number of wells where radon concentrations exceeds 1,000 Bq/L is 6,600–9,000.

Methods for removing radon from drinking water

Research on radon removal from water supplies was initiated in the 1970s in Czechoslovakia. Aeration was found an effective method for removing radon out of water. A removal efficiency of 99% was reported for 8 minutes aeration time using an air-to-water ratio of 1:8 (Hanslik *et al*. 1978). In the early 1980s, several aeration techniques and activated carbon adsorption were tested in Sweden. Among the tested methods, aeration under atmospheric pressure was found the most effective with a removal efficiency of up to 75% (Hedberg *et al.* 1982). In the USA, similar studies were also begun in the early 1980s. Three methods were tested and found effective: granular activated carbon adsorption, diffused aeration and spray aeration (Lowry 1983).

Removal studies in Finland were initiated in the 1990s in collaboration between STUK, the Finnish Environment Institute and Helsinki University of Technology. From the first project onwards, the main objective was to bring suitable and reliable treatment units to the market. Therefore, companies specializing in water treatment have been invited to participate in the research projects (Myllymäki *et al.* 1999). In 1997, a European research project financed by the European Commission started to investigate removal of naturally occurring radionuclides from ground water (Annanmäki & Turtiainen 2000). In order to obtain long-term experiences of their operation, a follow-up project was initiated and the units mounted during the previous project were monitored until the end of 2002 (Vesterbacka *et al.* 2008).

Aeration technique

As a dissolved gas, radon can be removed from water by aeration similar to e.g. carbon dioxide. Three alternative principles are customarily employed in aeration: a large number of air bubbles are produced in the water (diffused aeration); water is sprayed into air as small droplets (spray aeration); or a large surface is created between water and air on an inert material (tower aeration). Domestic aeration units typically employ a combination of the first two principles and the aeration time needed per water batch is 4–10 minutes. Most commercial systems work under atmospheric pressure and hence a water pump and a pressure tank are needed after the aeration unit to give the water enough pressure to deliver it throughout the plumbing. Water flow from a well through an aeration unit into a pressure tank is electronically controlled (Turtiainen 2009). The cost of acquiring aeration units is two to three times as high as that of an activated carbon unit. This difference, however, will be partly compensated by the higher operating costs resulting from the need to replace the carbon batch every three years (Mäkeläinen & Turtiainen 2003).

Low-cost alternatives where aeration takes place in the bore hole have also been introduced. In total, aeration units from eight manufacturers have been tested as of spring 2010. The lowest radon removal has been recorded for aeration in the bore hole, where efficiencies varied from 3% to 77% (average efficiency during 100 litres flow). More sophisticated aeration units, however, were all able to remove more than 90% of radon (Turtiainen 2009). Presently, four brands are available on the Finnish market, all with adequate radon removal efficiency.

Activated carbon adsorption

As a non-polar gas, radon is effectively adsorbed on activated carbon. Activated carbon units sold in Finland are typically pressure vessels with carbon bed volumes of 39–105 litres. An automatic backwash system is employed in cases where the influent contains large amounts of iron, manganese or humic substances that may cause clogging. The units are always installed to treat all household water so that exposure to radon through inhalation is prevented, as well. The units operate passively under normal plumbing pressure (2–6 bar) and hence require no additional pumps (Turtiainen *et al*. 2000a). Therefore, the acquisition price of the unit remains low compared to the aeration technique.

The maximum amount of radon accumulated on the carbon bed depends on the water flow rate and the radon concentration in raw water, and is reached in about tree weeks after commissioning of the unit. The short-lived progeny of radon that emit gamma radiation are also retained on carbon and reach equilibrium with radon. Due to resultant external gamma radiation from the activated carbon units they cannot be installed inside residential buildings and their use should be limited to radon concentrations below 5,000 Bq/L (Annanmäki *et al.* 2000).

The radon removal efficiency of activated carbon units was followed at ten households for a period of 3 to 9 years. The carbon bed was replaced with a fresh batch in three locations during the follow-up. In the selection of the households, different types of water were covered, including iron- and manganese-rich water and water with a high content of organic carbon. Radon concentrations in raw water were 1,500–7,400 Bq/L (Vesterbacka *et al.* 2008). All units removed more that 90% of radon, and most of them nearly 100%. Some units showed a decline in removal efficiency over time, and hence to ensure adequate removal efficiency it was recommended that the carbon batches be replaced every three years. Water quality, both microbiological and chemical, remained good. Iron, lead-210, polonium-210 and organic substances were partly removed by activated carbon treatment (Turtiainen *et al*. 2000b).

Customer insights on radon removal units

STUK recommends aeration technique as the primary household water treatment for removing radon because the units do not cause external radiation or produce any type of radioactive waste. The maintenance costs of aeration units are also somewhat lower than activated carbon units which require replacement of carbon at three years' interval. Replacement of carbon may sometimes be neglected by the home-owners.

According to a survey among customers that use household water treatment for removing radon, activated carbon adsorption is more widely employed, with a 70% share of all units (Vesterbacka *et al.* 2008). The reason for this is evidently the more affordable acquisition price. Another survey conducted in 2008 among 53 well-owners implied that most people are ready to take required measures against radon when they receive information. Only nine of the interviewed did not intend to react (Turtiainen & Salonen 2010).

Effective doses and related risks

Dose conversion factor for radon ingestion by adults reported by NRC (1999) is 3.5×10^{-9} Sv/Bq. The factor increases with decreasing age being 4.0×10^{-8} Sv/Bq for infants. Drinking water consumption has been investigated among the adult Finnish population in the age groups 25–34, 35–44, 45–54, 55–64 and 65–74 years. The average daily intakes among female and male populations representing these age groups were 0.667–0.929 and 0.441–0.676 litres, respectively (Paturi *et al.* 2008). By assuming even water ingestion during the day and by subtracting time spent outside homes (work, school, travelling, etc.) the annual intake of water at home among female and male population of these age groups has been assessed as 230–260 and 140–190 L/a, respectively (Turtiainen & Salonen 2010). According to the survey by Vesterbacka *et al.* (2005), the mean concentration of radon in those wells where radon exceeds 1,000 Bq/L, is 2,700 Bq/L.

Radon is partly released into indoor air during water usage. The proportion of waterborne radon concentration in indoor air to radon concentration in water is expressed as transfer factor (Nazaroff *et al.* 1987). Parameters, such as ventilation rate, indoor air volume, total water usage rate and proportion of radon released from water during certain types of water usage, affect the value of transfer factor. NRC (1999) recommends that a factor value of 1.0×10^{-4} be used in dose assessments (1,000 Bq/L of radon in water causes an excess radon concentration of 100 Bq/m^3 in indoor air). In Finnish one-family houses, both water usage rate and ventilation rate are lower than those in American houses. The modelled value (0.36×10^{-4}) is supported by actual measurements, which give a mean value of 0.39×10^{-4} (Turtiainen & Salonen 2010). The mean excess radon concentration in indoor air can thus be assessed as 100 Bq/m^3 in houses where the recommended maximum value for radon in water is exceeded. ICRP65 recommends a dose conversion factor of 6×10^{-9} Sv/(Bq h m^3) for radon in indoor air of homes with an equilibrium factor of 0.4 (ICRP 1994). These values together with a 0.8 indoor occupancy factor will be used in the following.

By using the values discussed above, the mean collective effective dose from ingestion can be estimated as 31–42 man Sv/a (weighed by population per gender and age group) among users of drilled well water in which radon exceeds 1,000 Bq/L. By using risk of 5% per Sievert it can be roughly assessed that two additional fatal cancer cases per year are attributable to ingested radon (ICRP 2007). The collective effective dose, 31–42 man Sv/a, is also obtained from dose assessments for inhalation, which may add one or two fatal cancer cases to the previous figure.

The number of household water treatment units installed for removing radon is presently about 1,000. Assuming an average removal efficiency of 95% for the removal units, the effective dose through both ingestion and inhalation of radon has been reduced by 3.4 mSv per year on average among those who have purchased them. Expressed as collective effective dose, the reduction is about 9 man Sv per year.

Discussion

In normal cases, the individual lifetime risks related to radon in drinking water are small. Even among those who use water in which radon exceeds 1,000 Bq/L, the mean lifetime risk is below 1%. Extreme concentrations of radon in well water are, however, occasionally found. Probably the highest concentration of radon ever found in private wells, 130,000 Bq/L, was located in Southern Finland in 2007. The family of five had been using the water regularly for a period of 10 years. The effective dose from ingested radon to the family members was thus about 1 Sv each. Hence, the assessed risk of one of the family members getting cancer attributable to ingested radon is about 30%.

Owing to long distances to water mains from the estate or due to projected low end-point consumption, joining the public water services may not be possible for homes with high radon concentration in the well water. The government has recognised the lack of water services (both drinking water and waste water) in rural areas and has accordingly channelled additional funs to extend the existing networks. There are, however, areas where the networks will not be available in near future. In these cases, household water treatment has provided an effective means to mitigate radon exposure. Acquisition of a water treatment unit is straightforward and the units sold today have generally been reliable and customer satisfaction has been good. This was made possible particularly by engaging the companies in research and development of the techniques from the first phase on.

Statistically, the number of cancer cases related to the radon in well water is only a few per year. Citizens, however, should have the right to be informed about this risk, which is also implemented in national legislation. Radiation is a sensitive issue for many, especially for families with children. Public communication campaigns or newspaper articles about risks relating to well water have always expedited radon measurements at laboratories and subsequently resulted in increased sales of the treatment units. Public memory, however, is short and public communication should be enhanced when the number of water samples sent for radon measurement start to decline. There are still 15,000 to 21,000 Finns that continue to be exposed to waterborne radon without necessarily being aware of it.

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Dose assessment for tritium releases during normal operation of NPP

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Abstract

Radiation Doses (RD) codes for estimate of impact of air effluents during normal operation of Czech NPP assume that all atmospheric tritium releases are emitted from ventilation stacks only. Transfer of liquid tritium is modeled by diffusion process in river only. In reality, in the case of Dukovany NPP, cooling water in cooling towers contain significant activity of tritium, as water is coming from reservoir with liquid effluents. Until now no special attention was devoted to the transfer of water vapor with tritium from cooling towers, because it was supposed that the tritium released from NPP was already taken into account in the calculation of doses from hydrosphere; estimation of doses for Dukovany NPP was therefore conservative. The aim of present research is to find out how conservative is the present way of estimation of doses from tritium and a realistic approach will be sought. A computer code MHTO has been developed to assess tritium doses to the general public. The code enables to simulate the behavior of tritium in the environment released into the atmosphere and hydrosphere under normal operation of NPPs. The code can calculate the doses for the four forms: tritium gas (HT), tritiated water vapor (HTO), water drops and organically bounded tritium (OBT). Models in this code consist of the tritium transfer model including reemission of HTO from soil to the atmosphere. The comparison of results obtained from calculations using RD and MHTO codes prove that when taking into account the releases of tritium from cooling towers, codes RD provide a value of maximal individual effective dose from hydrosphere about 30% less and the dose from atmosphere is about 20% higher in comparison to the case without releases from cooling towers. At the same time, the collective dose is approximately 30% lower as in the case without releases from cooling towers.

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The legacy of uranium mines – Pluralist Expertise Group experiment on uranium mines in Limousin (France)

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Abstract

Context: In France, as mining and milling operations drew to a close, from the 80s until 2001, AREVA-NC has carried out with the administration important work on remediation and rehabilitation of more than 200 sites to assure the protection of the population and the environment, with continuous monitoring of the environment, which is ongoing. The closure and the remediation over this period have caused some concern in the public and NGOs. This issue is of particular concern in Limousin, the area that most contributed to uranium mining in France. Among actions taken by the French Government and Regional authorities, the Pluralist Expertise Group (GEP) concerning the legacy of uranium mines in Limousin was created after a joint mission letter from Ministries of environment, industry and health (2005).

Method: The group brings together experts from various technical fields including French institutes, the industrial operator, local and national NGOs, independent experts and foreign experts. Four multi-partite working groups (WG) have been set up to discuss the current status and management options for the sites:

– WG1: Inventory of substances and transfers in the environment
– WG2: Impact and surveillance
– WG3: Regulations and long-term control and management
– WG4: Analysis of measurements.

The purpose of this paper is to describe the work of WG2.

Results: At first the WG2 has drawn up a status report of the current situation, both at national and international level, for its topics (risk for the environment, impact

on the population, health and ecological surveillance). Then the experts of the WG2 have put forward some recommendations concerning the actual management of the uranium mines in France. New methods in the field of environmental impact, sanitary screening and dosimetric assessment have been proposed and tested in Limousin. The details of WG2 will be published at the end of this year (*www.gep-nucleaire.org*).

Radiation situation in Moscow and public doses due to man-made radiation exposure

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Abstract

Over 2002-2007, radiation situation resulted from global fallout in Moscow has been assessed in terms of ^{137}Cs and ^{90}Sr contents in more than 500 samples of the environmental media. Such assessment showed that ^{137}Cs and ^{90}Sr contents in atmospheric precipitation complied with the average Russian levels, while annual activity concentration of these radionuclides in common air is six orders of a magnitude lower than permissible values regulated by the radiation safety standards.

^{137}Cs content in foodstuffs intaken by Moscow residents varies over the range from 0.1 up to 1.4 Bq/kg(l), while that of ^{90}Sr - from 0.08 up to 0.13 Bq/kg(l),; this is much lower than permissible specific activities. Over the years under observation, excess ^{137}Cs and ^{90}Sr contents in comparison with the permissible ones have been registered in wild food samples (mushrooms, wild berries). This is due to the fact that such kinds of foodstuffs entered to Moscow from the regions affected by the accident at the Chernobyl nuclear power plant.

The selective individual dose monitoring using thermoluminescent dosimeters showed that annual effective dose induced by external exposure of natural and cosmogenic radionuclides to Moscow residents is not higher than 1 mSv/a. Contribution of man-made radiation exposure is not higher than 5%. Effective dose induced by internal man-made radiation exposure to Moscow residents due to food intake over the inspected time period is 13 μSv/a on average.

Materia Study of global fallouts of nuclear explosion radioactive products at the area of Moscow began in 1957.

At the first stage, the research task was to determine concentration and isotope composition a radioactive aerosols in near-earth layer of atmosphere. Later, te objective of this work was to study radiation situation within Moscow area and to calculate public doses due to the global fallouts.

Development of the methods for radiochemical analysis promoted determination of ^{90}Sr and ^{137}Cs contents in the environmental media. Introduction of gamma-spectrometry methods enabled to specify complex compound of radioactive substances in the environmental media. Learning of methods for identification of radioactive substances in foods helps to evaluate radionuclide intake and to assess dose to the public, which is the main criterion for the radiation situation assessment.

In this light, the following tasks have been set:

1. To collect and analyze different parameters of the radiation situation, in particular, the extent of radionuclide contamination of the prime foodstuffs.
2. To study the site contamination with ^{90}Sr and ^{137}Cs radionuclides on the base of data on their contents in common air and fallouts, as well as gamma background levels in-situ.
3. To specify the food ration structure of the Moscow resident per capita taking into account the age structure of intake.
4. To evaluate ^{90}Sr and ^{137}Cs intake via ration.
5. To calculate effective doses to the public due to global fallouts.
6. To identify the critical age group.

So, our study included determination of radionuclide contents in the environmental media:

– Common air (aspiration samples);
– Atmospheric fallouts (sedimentation fallouts)
– Foodstuffs.

Sampling was being carried out according to the requirements of te regulative and legislative documents.

A value of the atmospheric fallout activity in Moscow was being determined on the base of precipitation analysis being collected using the special dishes (cuvette) (sedimentation filters), as well as radioactivity determination of common air (aspiration filters).

In samples of common air and fallouts, sum of beta active radionuclides was being determined using the radiometry method.

Over the reported period, gross beta activity of atmospheric fallouts on the geological substate varied from 4,0 to 33 Bq/m^2, while gross beta activity of radionuclides in the near-earth atmospheric air was at the level of 110 – 127 µBq/m^3. Results of gamma background measurements varied from 9 to 12 µR/h. Such data are in good compliance with data presented in radiation hygienic passports of Moscow and with data provided by the MosNPO “Radon”.

Samples of foodstuffs and drinking water were being examined using gamma spectrometry, radiochemistry and radiometry methods.

To evaluate intake of radioactive substances via foods by the residents of Moscow, data on food ration structure for Moscow residents have been examined. They resulted from the findings of selective inspections of the domestic budgets being performed by the Goscomstat of Russia.

The prime food groups include: bread and cereal products, milk and dairies, potato, vegetables and melons, meat and meat products, fish and fish products, as well as fruits and berries.

The foodstuffs were provided via the Moscow trade network and markets.

Study of the Goscomstat materials on the food ration structure enabled to conclude that over the recent years the food structure of Moscow residents remained practically unchanged and is still at the same level.

Table 1 includes data on intake of the prime food groups by the residents of different age groups.

Table 1. Mean intake of the prime food groups by the residents of different age groups in Moscow over 2000-2007, kg/days.

Food group	Per capita on average	Adults >17 years old	1-2 years old	2-7 years old	7-12 years old	12-17 years old
Bread and cereal products	0,327	0,392	0,157	0,188	0,441	0,376
Milk and dairies *	0,519	0,622	0,776	0,776	0,467	0,624
Potato	0,183	0,220	0,121	0,133	0,195	0,195
Vegetables and melons	0,192	0,231	0,144	0,216	0,331	0,188
Meat	0,237	0,285	0,119	0,154	0,225	0,237
Fish	0,043	0,053	0,006	0,021	0,032	0,032
Fruits and berries	0,126	0,151	0,094	0,141	0,216	0,123

*except for tallow oil

To evaluate ^{90}Sr and ^{137}Cs contents in foodstuffs included in the food ration of Moscow residents, results of direct examinations performed by the experts from the Burnasyan FMBC have been considered.

Table 2. Average contents of ^{90}Sr and ^{137}Cs in Moscow foodstuffs over the period 2000-2007, Bq/kg,l.

Foodstuffs	2000 – 2007	
	^{90}Sr	^{137}Cs
Bread and cereal products	0,10	0,27
Milk and dairies	0,07	0,39
Meat and meat products	0,14	0,32
Potato	0,07	0,24
Vegetables and melons	0,09	0,08
Fish and fish products	0,14	0,32
Fruits and berries	0,14	0,18
Mushrooms	0,17	2,1
Tea	5,6	4,0
Drinking water	0,002	0,001

Over the full period of surveillance, the specific activity of food samples collected in Moscow for the purpose of analysis was lower than the established regulations.

The level of radionuclide contents in foodstuffs being intaken by the resident of the Moscow region, is at the level of the averaged Russian data, while in some foodstuffs it is even lower (bread, milk, potato) (Table 3).

Table 3. Mean contents of ^{90}Sr and ^{137}Cs in foodstuffs in the Russian regions over the period 2000-2004, Bq/kg, l [12].

Foodstuffs	2000 – 2007	
	^{90}Sr	^{137}Cs
Bread and cereal products	0,13	1,4
Milk and dairies	0,12	0,70
Meat and meat products	0,06	0,30
Potato	0,08	0,70
Vegetables and melons	0,09	0,03
Fish and fish products	0,25	0,28
Fruits and berries	0,12	0,04
Mushrooms	3,7	39,0
Drinking water	0,002	0,001

Contents of ^{137}Cs and ^{90}Sr in foodstuffs and in water being intaken by the Moscow residents, is induced by the global fallouts of ^{137}Cs and ^{90}Sr originated from nuclear weapon tests and the accident at the Chernobyl NPP of 1986.

^{90}Sr and ^{137}Cs intake of the Moscow residents via the foodstuffs has been calculated on the base of mean contents of such radionuclides in foodstuffs collected over 2004 and 2006, and mean food intake by different age public groups over the period 2000-2007.

Table 4. Daily and annual intake of ^{90}Sr and ^{137}Cs via foodstuffs by different age groups of the Moscow residents over 2003-2006.

Age		Years					
		2003		2004		2006	
		^{90}Sr	^{137}Cs	^{90}Sr	^{137}Cs	^{90}Sr	^{137}Cs
1-2 years old	Bq/day	0,131	0,241	0,100	0,243	0,138	0,293
	Bq/day	47,82	87,97	36,5	88,65	50,37	106,95
2-7 years old	Bq/day	0,160	0,274	0,118	0,276	0,167	0,327
	Bq/day	58,40	100,0	43,07	100,6	60,96	119,36
7-12 years old	Bq/day	0,190	0,469	0,150	0,302	0,220	0,364
	Bq/day	69,35	171,20	54,75	110,23	80,30	132,86
12-17 years old	Bq/day	0,174	0,408	0,132	0,303	0,196	0,362
	Bq/day	63,51	148,90	48,11	110,74	71,54	132,13
Adults	Bq/day	0,204	0,437	0,146	0,336	0,219	0,394
>17 years old	Bq/day	74,46	159,50	53,11	122,78	79,94	143,81

When calculating radionuclide intake via the food ration components, thye drinking water intake was taken into account (daily):

1,4 l – for adults and children of 12-17 years old;

0,9 l – for children of 7 - 12 years old;

0,7 l – for children of 2 - 7 years old;

0,5 l – for children of 1 - 2 years old.

Radionuclide intake via drinking water has been calculated on the base of the mean level of ^{137}Cs and ^{90}Sr contents equal to 0,001 and 0,002 Bq/l respectively.

The public dose in Moscow is originated from the natural and artificial sources of ionizing radiation. The main dose-forming factors for the Moscow population are the natural sources.

As for ^{137}Cs and ^{90}Sr, the main pathways of intake are ingestion and exposure originated from fallouts on the geological substate.

When calculating effective internal doses induced by ^{90}Sr and ^{137}Cs, being intaken by the population of Moscow via the foodstuffs and water, annual radionuclide intake values via the foodstuffs by the different age groups of population and dose coefficients for ^{90}Sr and ^{137}Cs for each age group are taken into account.

For persons included in any group, the effective dose calculation assumes:

Intake of only local foodstuffs (provided through the Moscow trade network);

Average food intake by members of the appropriate age group (kg/day).

Annual (committed) effective internal dose ($E_,$) is assessed in terms of radionuclide intake by the local population via the local foods:

$$E = 10^{-3} \cdot \sum_k d_k \cdot \sum_n (M_n \cdot A_{kn}), \text{ mSv/year,}$$

where:

d_k – dose coefficient for food intake of the k-th radionuclide by the adult residents according to Annex II, Sv/Bq;

M_n - mean annual intake of n-th foodstuff, kg/year;

A_{kn} - annual specific activity of the k-th radionuclide in the n-th foodstuff, Bq/kg.

Gross effective internal dose to the adult residents of Moscow over the inspected period is given in Table 5.

Table 5. Gross effective dose to the adult residents of Moscow over 2003 - 2007, μSv/year

Exposure pathway	Radionuclide									
	2003		2004		2005		2006		2007	
	^{90}Sr	^{137}Cs	^{90}Sr	^{137}Cs	^{90}Sr	^{137}Cs	^{90}Sr	^{137}Cs	^{90}Sr	^{137}Cs
Internal exposure via foods	2,31	2,25	1,5	1,6	2,85	2,92	2,24	1,87	2,76	2,59
Internal exposure via drinking water	0,84	0,42	1,09	0,3	0,03	0,07	0,03	0,07	0,03	0,07
Total	3,15	2,67	2,59	1,9	2,88	2,99	2,27	1,94	2,79	2,66

Gross effective dose to the adult public of Moscow due to internal exposure varies from 3,1 to 5,77 μSv/year. Gross effective dose to the critical group – children of 12 – 17 years old – is not higher than 6,5 μSv/year.

Doses originated from the drinking water intake are very low and depend upon the age special features of water intake: minimum - 0,063 μSv/year for children of 1-2 years old and maximum- 0,148 μSv/year for the critical group - 12-17 years old.

Thus, gross effective dose to the Moscow residents due to global fallouts is not higher than 13 μSv. Even taken into account dose induced by external gamma radiation originated from the natural radionuclides, gross effective dose to the Moscow residents is much lower than the public dose limit regulated by the national regulative documents (1 mSv).

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Ionizing radiation exposure of the Belgian population in 2006

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Abstract

The radiation exposure of the Belgian population in 2006 was calculated with the methods of the UNSCEAR 2000 report. The annual average effective dose is estimated at 4.6 mSv, of which 2.5 mSv is from natural sources and 2.1 mSv from applications of ionizing radiation; more than 95% from medical imaging. Radiotherapeutic exposures are not accounted for in this overview.

The contribution from diagnostic radiology practice was calculated by multiplying an estimated average dose per type of X-ray examination with the National Health Insurance data on the number of examinations. The exposure is dominated by CT, which provides about 60% of the effective dose; the number of CT-scans increased by 77% between 1997 and 2006 to 155 per 1000 people. An extensive measurement campaign on the patient doses in interventional and cardiological radiology showed effective doses between 5.7 mSv for diagnostic angiography and 15.3 mSv for an average PTCA. The Belgian government has recently launched a project to monitor therapeutic procedures for deterministic effects and to guide optimization in these practices. The effective dose from diagnostic radiology is estimated at 1.88 mSv/y. Better dose estimates will be available after the publication of diagnostic reference levels from the first generalized and legally obliged patient dose survey.

The number of diagnostic administrations of radiopharmaceuticals to patients in Belgium was 52 per 1000 people in 2006 resulting in an effective dose from nuclear medicine of 0.22 mSv/y.The estimated contributions from natural sources in 2006 are the same as in 2001.

The average exposure in Belgium has doubled over the last 110 years, from approximately 2.3 mSv in 1895 to 4.6 mSv in 2006. Of this increase 0.2 mSv comes from natural sources and 2.1 mSv from medical applications. During the same period the average life expectancy in Belgium for man increased from 48 to 77 years and for women from 51 to 83 years resulting in a 3- to 4-fold increase of the lifetime exposure.

The risk perception of the population from ionizing radiation is strongly correlated to the perceived possibility of potential exposure, with a high concern for nuclear waste management and a low concern for medical and natural exposure.

Introduction

Man has always been exposed to natural radiation arising from the earth as well as from outer space (cosmic radiation). Since the discovery of X-rays by Röntgen in 1895 people are also exposed to man-made sources of ionizing radiation. Attention was first drawn to the medical possibilities of diagnosis of bone injuries and imaging of retained bullets received on the battlefield. Diagnostic radiology was in the 20th century supplemented with nuclear medicine and radiotherapy. Outside the medical field industrial and military applications of ionizing radiation were developed, but their collective impact on the population exposure remained limited. Medical uses of radiation continue to increase as techniques develop and become more widely disseminated. As part of this trend, high dose procedures (particularly CT scanning) are increasingly being used to the point that in several countries, including Japan and the USA, medical has displaced natural sources of radiation as the largest overall component of ionizing radiation exposure.

The effective doses to the Belgian population in 2001 were presented at the IRPA 11 conference in Madrid (Vanmarcke 2004) with the methods given in the UNSCEAR 2000 report. In this paper, the assessment is revised using data for 2006.

Medical radiation exposures

Medical imaging has seen the introduction of digital detectors. This has led to 2 main advantages: (1) most detectors can work at a lower dose level than film-screen systems, and (2) digital detectors can be used for new applications, such as high resolution (3D) imaging. This evolution is a logical reply to ever increasing demands for more complex and new imaging procedures and a more efficient use of dose budgets. Next to these arguments, upgrading of technology is governed by financial aspects and work flow issues. Guidance or optimization of dose settings is not yet routine practice in our hospitals. Recent patient dose studies organized in the frame of the medical directive (97/34/Euratom) and its Belgian implementation confirmed a large dose spread.

The amount of radiation per procedure indicates an increasing trend and as a result also the dose to the patient.

- Higher activities can be administered in nuclear medicine to improve image quality. The increasing use of various isotopes with short half-lives, such as technetium-99m (T½: 6 hours), leaves the dose to the patient and the environment within a reasonable margin.
- High doses can be delivered to patients in interventional radiology, sometimes resulting in skin injuries.
- The introduction of digital techniques could reduce the dose provided that the imaging process is strictly controlled. Overexposures that lead to images too dark for interpretation are virtually impossible in digital radiology. The search for better image quality, sometimes introduced without any justification procedure, increases exposure. This is especially the case in CT that has seen the introduction of high resolution cardiac angiography, CT perfusion, dual energy applications, etc..

Very high doses are delivered precisely to tumor volumes in radiotherapy to eradicate disease, mostly cancer, while minimizing the irradiation of the surrounding healthy tissue. The quantity effective dose is inappropriate for characterizing these exposures, in which levels of irradiation are by intent high enough to cause deterministic effects. Radiotherapeutic exposures are therefore not accounted for in this overview. Incidents however have stressed the importance of quality assurance and optimization.

In the frame of these developments, Flemish and Belgian data were collected for the yearly report on the environment and nature in Flanders (MIRA 2007) and were compared to the world average values of the UNSCEAR 2000 report.

Diagnostic radiology

According to National Health Insurance data (RIZIV/INAMI), the average inhabitant of Belgium was subject to 1.2 X-ray examinations in 2006 (excluding dental X-rays) (MIRA 2007). The trend in the annual frequency is shown in fig. 1. The number of examinations remained stable between 1997 and 2006 thanks to efforts of the National Health Insurance to keep the costs under control.

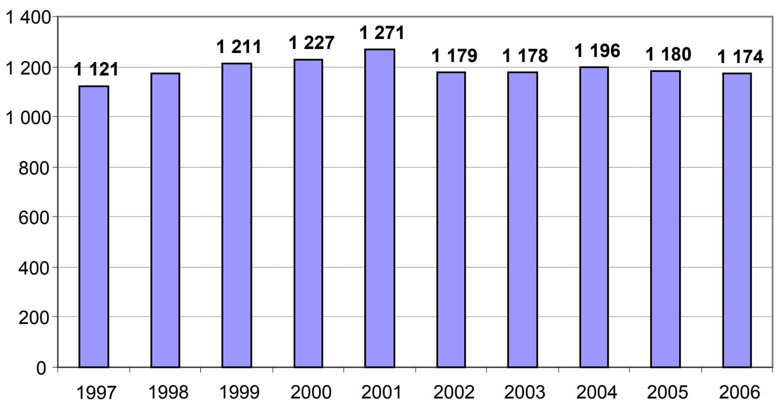
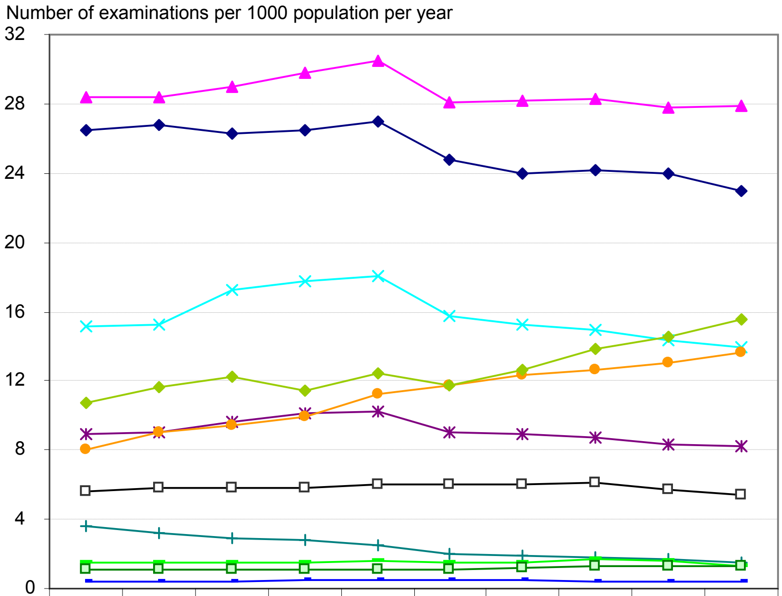


Fig. 1. 10-year trend for total number of X-ray examinations per 1000 inhabitants in Belgium (excluding dental X-rays).

More detailed National Health Insurance data show a shift between the types of examinations (fig. 2). Traditional techniques, such as spine and GI-tract examinations (included in “fluoroscopic examinations”) are being replaced by CT (Computed Tomography) or MR (Magnetic Resonance imaging). The number of CT-scans increased by 77% between 1997 and 2006 to 155 per 1000 people. The figure shows also a significant increase in the use of mammography since 2001, when the Flemish government provided free breast cancer screening for women aged 50 to 69 years. Chest examinations remained constant at a high level. The field of vascular imaging has progressed rapidly over the last decades. Blood vessels are visualized for diagnostic or therapeutic purposes. Diagnostic X-ray procedures are being replaced by non invasive imaging such as MR angiography or CT angiography, but the number of (higher dose) interventional procedures is rising. Vascular imaging should be closely monitored, because the exposure to the patient and the medical team can be very high, especially during interventional procedures. Everyday, doctors improve their skills and knowledge in this field, so that ever smaller vessels or more complex medical problems find a solution. Interventional procedures are considered less risky for the patient than traditional “open” surgery and shorten the duration of hospitalization.



	1997	1998	1999	2000	2001	2002	2003	2004	2005	2006
Chest	265.	267.	263.	264.	269.	247.	239.	242.	239.	229.
Bed	56.	58.	58.	58.	60.	60.	60.	61.	57.	54.
Limbs and joints	284.	283.	290.	297.	304.	281.	282.	282.	277.	278.
Head – abdomen – hip	151.	152.	172.	177.	180.	157.	152.	149.	143.	139.
Spine	89.	90.	96.	101.	101.	90.	89.	87.	83.	82.
Mammography (clin. +	79.	90.	94.	99.	112.	117.	123.	126.	130.	136.5
Fluoroscopic examination	35.9	31.6	29.1	27.9	25.2	20.4	19.5	18.5	16.8	15.3
Arthrography	3.6	3.8	4.2	4.6	5.1	4.6	4.6	4.2	4.2	4.4
Angiography	14.8	14.9	15.0	15.3	15.8	15.2	15.2	16.6	15.9	13.5
CT	106.8	116.8	122.1	114.6	124.2	117.8	126.1	138.2	145.8	155.2
Operating room	10.7	10.8	10.9	10.9	11.5	11.4	12.3	13.1	13.0	13.4

Fig. 2. Trends in diagnostic radiology practice in Belgium between 1997 and 2006 (Bosmans 2007).

As there are only a few data on the doses delivered to patients in radiology in Belgium available, we had to rely partially on data from other sources. The average effective dose per type of examination from four different sources is compared in table 1. In the right column, the effective doses, used for the calculations in the report on the environment and nature in Flanders, are given (MIRA 2007). They were derived from local studies and supplemented by international literature (Bosmans 2007). In Belgium, doses from conventional X-ray examinations are relatively high due to the extensive use of fluoroscopy for positioning of the patient. The selected value for the average CT dose is 7.2 mSv, which is somewhat less than the value from the UNSCEAR 2000 report of 8.8 mSv. The value was calculated from the relative frequency and the effective dose per type of CT examination given in table 2.

The values that were used to estimate the exposure from interventional procedures were retrieved from an extensive national study (Bleeser 2008, Smans 2008, Bogaert 2008). In the frame of this multicenter study, effective patient doses were calculated for diagnostic imaging of the lower limbs and the most frequent cardiac examinations. This study also allowed to estimate effective doses for the other types of frequent examinations. Today, our regulatory authority, the Federal Agency of Nuclear Control, requires that dose data of all dynamic X-ray procedures are registered. How to exploit these data for optimization is not yet clear.

Table 1. Comparison of effective doses from diagnostic X-ray examinations (mSv).

Examination	UNSCEAR 1993	UNSCEAR 2000	Mol 2001	RIVM 2005	MIRA 2007
Chest	0.14	0.14	0.15	0.07	**0.31**
Limbs and joints	0.06	0.06	..		**0.032**
Spine	1.7	1.8	2.6	0.35	**2.6**
Pelvis and hip	1.2	0.83	..	0.28	**0.83**
Head	0.16	0.07	..		**0.14**
Abdomen	1.1	0.53	0.92	0.45	**0.92**
GI tract	5.7	5.0	..	4.8	**12.9**
Cholecystography	1.5	2.3	..		**2.3**
Urography	3.1	3.7	7.9	3.2	**7.2**
Angiography	6.8	12.0	..	12.2	**Diagn.: 5.7 Intervent.: 9.8**
PTCA	...	22.0	..		**15.3**
Mammography	1.0	0.51	..	0.4	**0.34**
CT	**4.3**	**8.8**	**7.7**	**1.3 – 11**	**7.2**

Table 2. Average patient dose from CT examinations (MIRA 2007).

Type of CT examination	Average effective dose mSv
CT skull	1.5
CT skull base	1.7
CT abdomen and/or chest	12.7
CT spine	5.7
CT limbs and joints including angiography	1.5
Average CT dose	**7.2**

Figure 3 shows the average effective dose distribution of the Flemish population in 2006 from diagnostic X-ray examinations. This distribution is derived from the frequency of the various medical procedures and the average radiation dose for that procedure from table 1 (MIRA 2007). The exposure is dominated by CT, which provides 59 % of the effective dose. The average annual dose from diagnostic radiology in Flanders is estimated at 1.75 mSv in 2006 and in Belgium at 1.88 mSv. The difference is mainly due to a slightly lower number of CT scans in Flanders (142 per 1000 population) than in Belgium (155 per 1000 population).

Figure 4 shows the temporal trends in patient dose from diagnostic radiology in Belgium. The average annual dose from CT increased by 77% between 1997 and 2006. The increase of the CT dose is partly compensated by a decrease in conventional examinations. Most remarkably, the installation of a large number of MR systems has not stopped the expansion of the number of CT examinations.

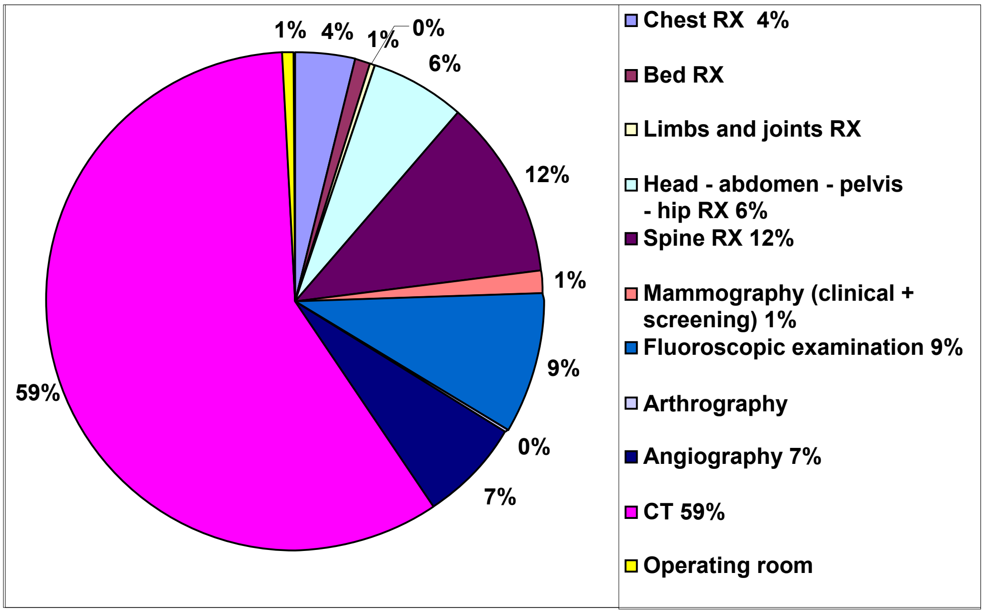


Fig. 3. Dose Distribution from diagnostic X-ray examinations in Flanders in 2006 (Bosmans 2007).

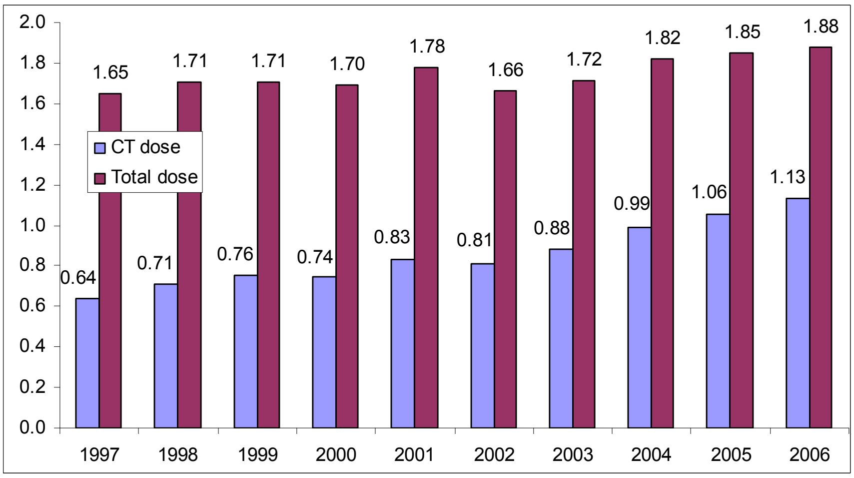


Fig. 4. Trends in annual effective dose (in mSv per year) from diagnostic radiological examinations in Belgium (1997-2006). The large and increasing share from CT is given separately.

Nuclear medicine

In nuclear medicine, radionuclide preparations are administrated to patients for diagnosis or to a much lesser extent for therapy. The number of diagnostic administrations of radiopharmaceuticals to patients in Flanders and Belgium was respectively 40 and 52 per 1000 population per year in 2006 (National Health Insurance data). The large academic hospitals in Brussels, which attract a lot of Flemish patients, could explain the difference. The number of diagnostic nuclear medicine procedures in Belgium is high. The UNSCEAR 2000 estimate for countries with an advanced health care system is 19 per 1000 population per year.

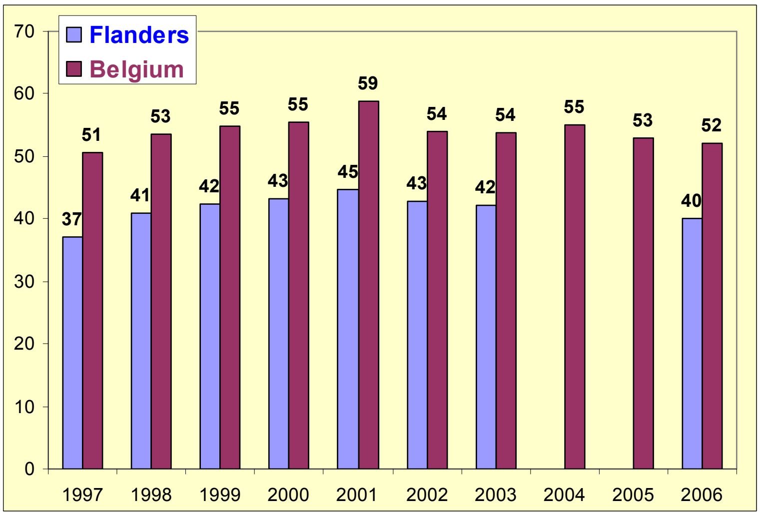


Fig. 5. 10-year trend in diagnostic nuclear medicine practice in Belgium (1997-2006) and Flanders (1997-2003 and 2006).

Positron Emission Tomography (PET), where the radionuclide fluorine-18 (T½: 110 min.) is usually administered in patients, is a relatively new technique, which has proven its value for a number of practices, such as the detection of brain tumors. There are currently several dozen PET scanners in Belgium, including several PET-CT. The dose to the patient from a PET examination varies between 5 and 8 mSv. In case of PET-CT the CT dose has to be added. This CT may be either a full CT, which in most cases replaces another CT, or an examination to improve PET image reconstruction. In the latter case, the dose may be relatively limited. In any case PET-CT is a high dose imaging procedure. A Flemish study showed that technologists who manipulate the F-18 syringes may receive doses to fingers and hands above the limit of 500 mSv/y (Berus 2004).

Technetium-99m dominates the spectrum of the radiopharmaceuticals. The typical dose for an examination is between 1 and 10 mSv and the administered Tc-99m activity between 100 and 900 MBq. The Belgian Society for Nuclear Medicine has published guidelines for the reference administered activities for diagnostic nuclear medicine procedures on their website:

http://www.belnuc.be/images/stories/PDF_guidelines/reference_activities.pdf

A survey at 19 nuclear medicine departments in Flanders estimated the average dose per diagnostic procedure at 4.2 mSv (De Geest 2002). This value, multiplied by the number of examinations in 2006, results in an average dose of 0.17 mSv/y in Flanders and 0.22 mSv/y in Belgium. The corresponding UNSCEAR 2000 estimate for countries with an advanced health care system is 4.3 mSv per procedure and 19 procedures per 1000 population, which corresponds with an average dose of 0.08 mSv/y.

Medical radiation exposures in Belgium in 2006

The average effective dose from diagnostic medical imaging in Belgium in 2006 is estimated at 2.1 mSv/y: 1.88 mSv/y from X-ray examinations and 0.22 mSv/y from nuclear medicine procedures. The corresponding values for Flanders are 1.92 mSv/y:

1.75 and 0.17 mSv/y. The collective medical exposure for the whole population in 2006 is estimated to be:

- Belgium: 0.0021 x 10 511 382 = 22 000 manSv;
- Flanders: 0.00192 x 6 078 600 = 11 700 manSv.

The increasing medical exposure could be responsible for hundreds of excess cancer cases each year in Belgium. It should be noted however that the age distribution of patients differs from the age distribution of the general population on which ICRP-risk coefficients are still based. In digital radiology departments, examination specific age distributions could be retrieved from the image databases, but this is not a routine task in the imaging department.

Exposures from natural radiation sources

The estimated contributions from natural sources in 2006 are the same as in 2001. The average radon concentration in Belgium is estimated indoors at 48 Bq/m³ and outdoors at 10 Bq/m³. With the UNSCEAR dose conversion factor of 9 (nSv/h)/(Bq/m³) (in terms of radon decay products) and a small contribution from radon gas dissolved in blood, a radon dose of 1.35 mSv is calculated. The annual exposure to cosmic radiation is estimated at 0.35 mSv, including a small contribution from air travel and holidays (for instance winter sports). Finally, the external and internal exposures from K-40 and the natural decay series are assessed at 0.4 and 0.3 mSv respectively and the thoron exposure at 0.1 mSv.

Sources and trends of radiation exposure in Belgium

The radiation exposure of the Belgian and Flemish population from natural and man-made sources is compared in table 3 to the average exposure for countries with an advanced health care system from the UNSCEAR 2000 report. The average annual dose in Belgium is 4.6 mSv and in Flanders 4.1 mSv. The largest contribution comes from diagnostic medical examinations, which is estimated, on the basis of social security data, to be 2.10 mSv in Belgium and 1.92 mSv in Flanders. The second largest contribution is from radon exposure.

The average effective dose in Belgium has doubled over the last 110 years from approximately 2.3 mSv/y in 1895 to 4.6 mSv/y in 2006. Of this increase 0.2 mSv/y comes from natural sources and 2.1 mSv/y from medical applications.

- A slow increase of the indoor radon exposure from about 1.15 mSv/y in 1895 to 1.35 mSv/y in 2006. This is due to the reduction in ventilation in modern homes and to the use of building materials with enhanced radium levels, such as phosphogypsum and fly ash. 10 to 100 times higher radon concentrations have been found in some dwellings, mostly located in radon prone areas.
- A small increase of the cosmic radiation of about 0.05 mSv/y from air travel and winter sports. 10 times higher cosmic ray exposures are possible for frequent flyers.
- A strong increase in the medical diagnostic use of ionizing radiation from 0 mSv/y in 1985 to 2.1 mSv/y in 2006. 10 to 100 times higher values are possible for particular patients.
- A small contribution from all other man-made sources of less than 0.05 mSv/y.

During the same period (1895 - 2006) the average life expectancy in Belgium for man increased from 48 to 77 years and for women from 51 to 83 years. The combined effect of these two trends resulted in an increase of the lifetime exposure by a factor of 3 to 4:

- for men from 110 mSv in 1895 to 354 mSv in 2006 and;
- for women from 117 mSv in 1895 to 382 mSv in 2006.

Table 3. Average exposure from radiation sources in Belgium, Flanders and in developed countries with an advanced health care system (Vanmarcke 2004; MIRA 2007; UNSCEAR 2000).

Source	Average annual effective dose Belgium mSv/y	Flanders mSv/y	Worldwide mSv/y
Natural radiation			
Cosmic radiation	0.35	0.3	0.4
External terrestrial radiation	0.4	0.4	0.5
Radon	1.35	1.0	1.1
Thoron	0.1	0.1	0.1
Internal exposures other than radon	0.3	0.3	0.3
Total (rounded value)	*2.5*	*2.1*	*2.4*
Man-made			
Radiology	1.88	1.75	1.2*
Nuclear medicine	0.22	0.17	0.08*
Other man-made exposures	< 0.05	< 0.05	< 0.05
Total (rounded value)	*2.1*	*2.0*	*1.3*
Total (rounded value)	**4.6**	**4.1**	**3.7**

* Developed countries with an advanced health care system

Risk perception from ionizing radiation

The third risk barometer survey of the perception of the Belgian population (Perko 2010) confirmed that there exists no generalized fear for nuclear activities. Compared to 2006 a slight rise (5%) of risk perception was noticed for most items. Industrial risks (industrial accidents, industrial waste) are evaluated much lower than personal and lifestyle risks. The paradox that the sources with the highest population exposures, such as medical X-rays and natural radiation, are perceived as less dangerous than nuclear power plants, which have a low population exposure in normal operation, continues to exist. However the perception of medical and radon risk is higher than in previous barometers. The risk perception for terrorism, chemical or nuclear waste or accidents was 30 to 40% lower in 2006 compared to 2002. These lower values were for the largest part replicated in the 2009 survey.

The predistribution of iodine for prophylaxis in accidental conditions was widely supported. About 16% of the Belgian population remembers the iodine release incident in August 2008 in IRE, Fleurus. Most people believe the consequences of this release to be more serious than what is communicated by the authorities, and this holds for both the general population and the subset of people who lives near the installation. There remains a significant lack of confidence in the provision of correct information by the authorities on risks related to accidents in nuclear installations or radioactive waste.

Higher confidence exists where authorities have been active and often present in the media.

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Sharing an environmental monitoring network: The AREVA Tricastin experience

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Abstract

Set up at the beginning of the site's operations, in 1962, the monitoring of the radiological environmental impact of the AREVA site of Tricastin has evolved over time to more specifically meet the multiple objectives of environmental monitoring: 1) to prove the respect of the commitments required by the authorities ; 2) to be able to detect a drift in the observed levels ; 3) to enable the assessment of impacts of industrial activities ; 4) to ensure the balance between environmental quality and practical uses by local population ; and 5) to inform the public of the radiological state of the environment. Thousands of data were acquired on the radioactivity of all environmental compartments as well as on the functioning of local ecosystems. Today, the Network of Environmental Monitoring of AREVA Tricastin goes beyond the requirements of routine monitoring to provide innovative solutions for monitoring the radioactivity (especially for uranium) in the environment.

Introduction

To assess the potential impact of the nuclear facilities on its environment, AREVA has set up a monitoring network on and around the site, since the start of its activities. Since 1962, numerous data have been accumulated, to know the functioning of local ecosystems, to assess the background of environmental radioactivity and trends in the different environmental compartments. An ambitious collection and structuring of historical data is underway to have a comprehensive vision of the past activities of the site.

Currently, samples and measurements of environmental radiation monitoring are managed by an AREVA common environmental monitoring network. The sampling is made inside and outside the site according to an agreed schedule which is controlled monthly by administrative authorities (ASN[1], DSND[2] and DREAL[3]). In addition to these regulatory requirements, AREVA also performs a targeted monitoring to strengthen the survey and to increase knowledge of the environmental impact of industrial activities on the site, according to its values and environmental policy. Moreover, an increased monitoring program was set up in July 2008 following the

[1] French Nuclear Safety Authority for civil facilities
[2] French Nuclear Safety Authority for military facilities
[3] French Regional Authority for BPI

accidental discharge of SOCATRI. It was gradually alleviated after the demonstration of a speedy return to the original situation before the incident.

Nuclear operations on the Tricastin AREVA site and regulatory commitments

The following institutions - AREVA NC Pierrelatte, COMURHEX Pierrelatte, EURODIF Production, FBFC Pierrelatte, Enrichment Corporation Tricastin (SET) and SOCATRI- are part of the upstream activities of the AREVA Company. They provide services for worldwide electricians: nuclear conversion, enrichment of uranium and nuclear fuel fabrication.

The AREVA Tricastin site now includes five nuclear operators of AREVA Company (AREVA NC -formerly COGEMA-, COMURHEX, EURODIF Production, SOCATRI and SET), a facility of FBFC, some research activities (units of the Atomic Energy Commission and units of the Institute for Radiological Protection and Nuclear Safety) and the facility labelled BCOT of the NPP EDF-Tricastin. To cover all these activities, there are several nuclear plants and a secret nuclear plant, each supervised by regulatory texts which authorise the nuclear releases and require environmental monitoring. In addition, other regulatory texts complete the requirements of environmental monitoring for the BPI of the site. Environmental monitoring of the site is therefore based on a set of texts, sometimes depending on various administrative authorities (ASN DSND, DREAL), and whose release dates are different. Their updates are dependent on the administrative development of each facility; the harmonization of environmental monitoring at the site is therefore a real challenge.

The pooling process initiated in 2005 is now operational. An agreement signed between the various operators of the site and the Head of the Tricastin site governs the operation of a common network of environmental monitoring. A computer database manages all the results which are previously validated by the two laboratories of the site dedicated to the measurement of environmental samples. The overall environmental monitoring network has evolved to be fully compatible with ISO 17025 standards which establish competency requirements for testing and calibrations, including sampling in the environment.

Current results of the environmental monitoring network

As illustrated in Figure 1, monitoring of radioactivity in the Tricastin environment interested in all media (air, water, soils, flora and fauna, food from local agriculture). The samplings are made at frequencies appropriated (daily, weekly, monthly, quarterly, and semi-annually) according to the substance analysed and / or the monitored parameter.

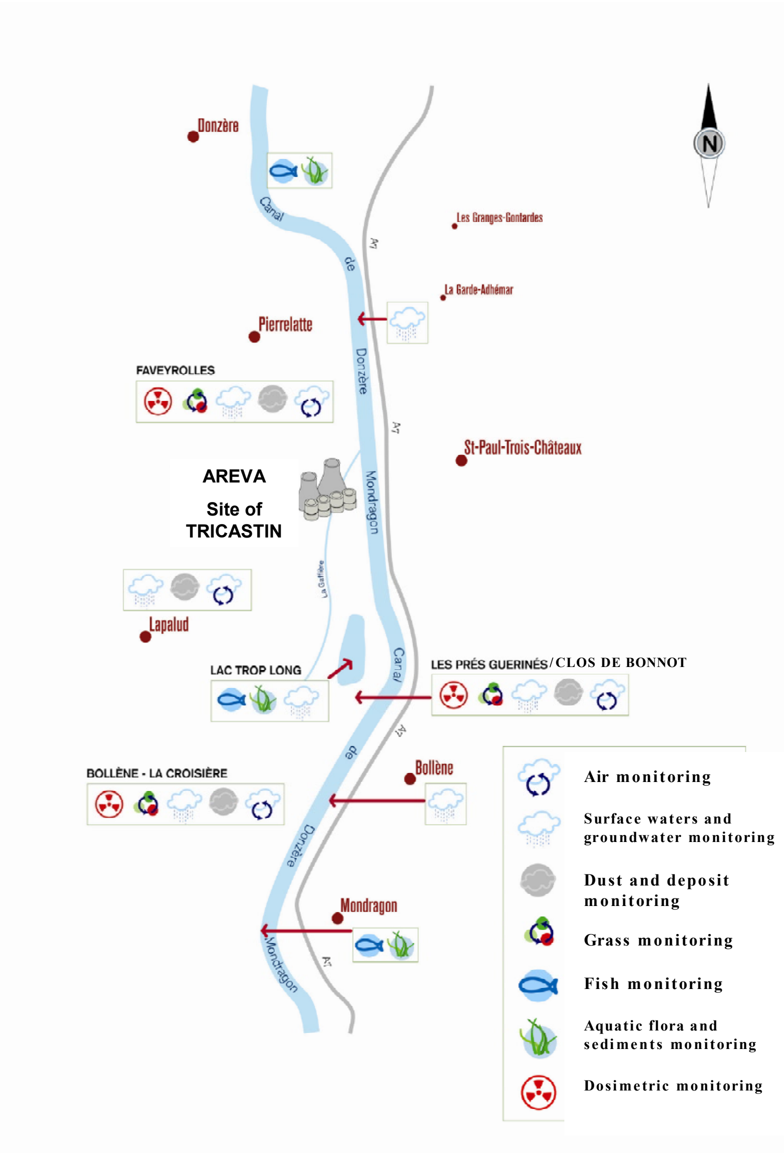

Fig. 1. Simplified diagram of the environmental monitoring network.

a. Air monitoring

The air monitoring network consists of eleven air control stations, including one outlet dust exhaust air and a ground recovery device for the precipitation. Seven of these stations are located within the site and four of them off-site, as close as possible to population groups living along the site. Each station is equipped with an external automatic air sampler for determining the radiological activity; an Owen gauge for the collection of rainfall ; and for two of them, with a bubbling system for the collection of tritium and carbon 14.

Air monitoring allows the measurement of the background radiation naturally emitted by the environment and the detection of possible influence of industrial activities of the Tricastin site. In France, the natural background is about 0.8 mSv per year [IRSN, 2009], due to cosmic and telluric rays. Around the Tricastin site (outside the site fence and at the reference stations), levels are generally measured at the local natural background level (80 nSv / hr or 0.7 mSv per year), excepted for a portion of the road labelled “Chemin des Agriculteurs” (Path of Farmers), at the east of the AREVA site, between the nuclear power plants of EDF and AREVA facilities. At this point, the combined operations of both nuclear sites induced a measurable cumulated dose which however is less than 1 mSv, in the theoretical hypothesis of a permanent stand at this barrier during the entire year (Figure 2).

A meteorological station equipped with a mast height of 100 m completes the air monitoring devices and allows the measurement collection 24/7.

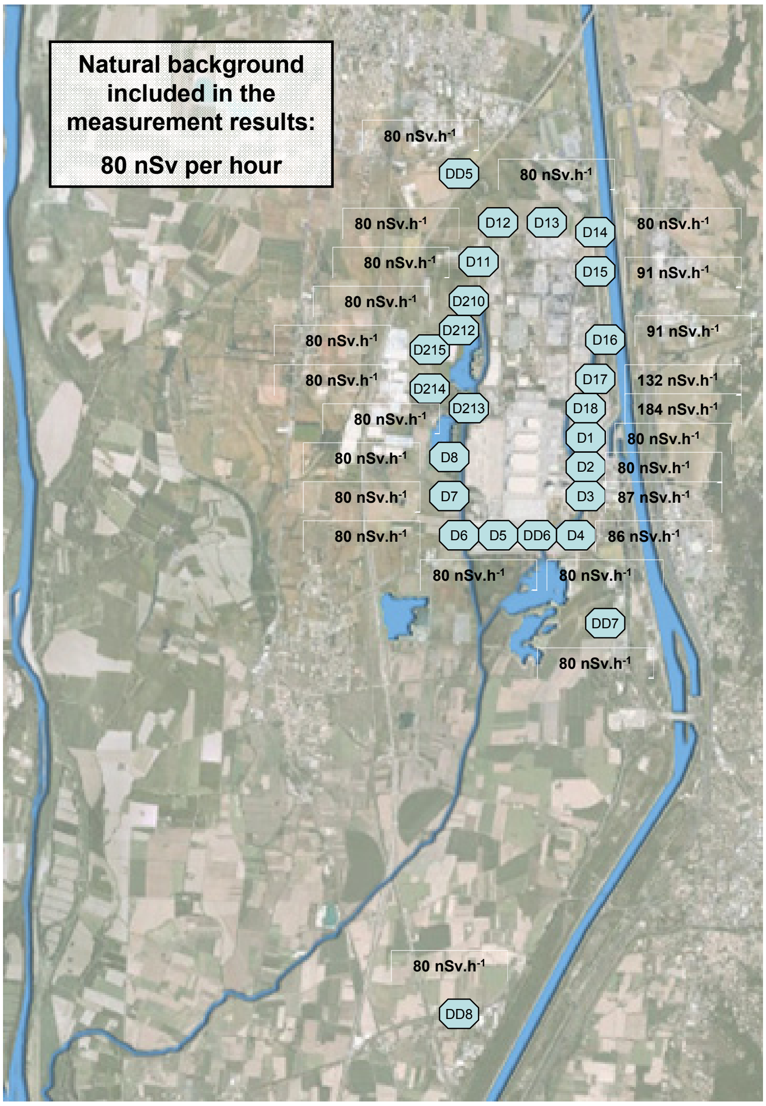

Fig. 2. Map of the mean annual dose rates measured in 2009 (background dose rate included).

b. Terrestrial monitoring

The site provides regulatory oversight of the monitoring of terrestrial radioactivity. At least a monthly analysis of plants (grass), at four points in the vicinity of the site, is performed. A mini-garden has been established for each reference station to ensure the availability of grass. On these samples, monthly measures of gross beta activity, uranium content and potassium-40 activity are performed. In addition, a measurement campaign on major agricultural products and on the land surface layer is made yearly.

The monitoring of radioactivity in the grass allows both to measure the background due to the natural occurrence in plants of radioactive elements (such as potassium 40, uranium and its progeny), and, to detect a possible influence of industrial activities of the Tricastin site. The plants uranium content is naturally very variable. The order of magnitude is of several mg per kg fresh weight but can reach several tens of micrograms per kg fresh weight for certain herbs [Anke, 2009]. Therefore, a possible influence of site operations is difficult to highlight and a PhD in collaboration with IRSN was initiated in 2009 to obtain more detailed background noise in local products.

c. Aquatic monitoring

The monitoring of the aquatic environment concerns (Figure 1):

- Surface waters: upstream and downstream of the Donzère-Mondragon channel, which is the outfall of the mayor discharge processes of the site, upstream and downstream of the rivers Gaffière and Mayre Girarde, downstream of the river Lauzon and in the Lake "Trop Long";
- Groundwater: today some fifty regulatory points are monthly analysed,
- The drinking waters: annual measurements of gross alpha and gross beta activities and concentrations of potassium-40 and uranium are performed for the cities of Pierrelatte, Bollène and Lapalud;
- This monitoring is supplemented by an annual campaign of sampling and analysis of aquatic sediments, flora and aquatic fauna.

Waters monitoring allow measuring the background level due to the natural presence of radioactive elements in water (e.g. uranium and its progeny, and potassium-40) and detecting a possible influence of the site of Tricastin. The rivers have a varying concentration of uranium in their water, about 0.02 to 6 $\mu g.l^{-1}$ [IPSN, 2001], but locally, the groundwater from granites can naturally achieve much higher values, such as 100 $\mu g.l^{-1}$ [ASN / Department of Health and Sports / IRSN, 2009].

For surface waters, the measured values in 2009 show no influence of AREVA Tricastin site on alpha activity measured in the overall Donzère-Mondragon channel and in the Mayre Girarde River (Figure 3). However, traces of the site's activities are visible inside and outside the site, in the Gaffière River. This influence is also visible in the Trop Long Lake and in the Lauzon River. These results are similar to those obtained for uranium and the overall beta activity show broadly similar trends.

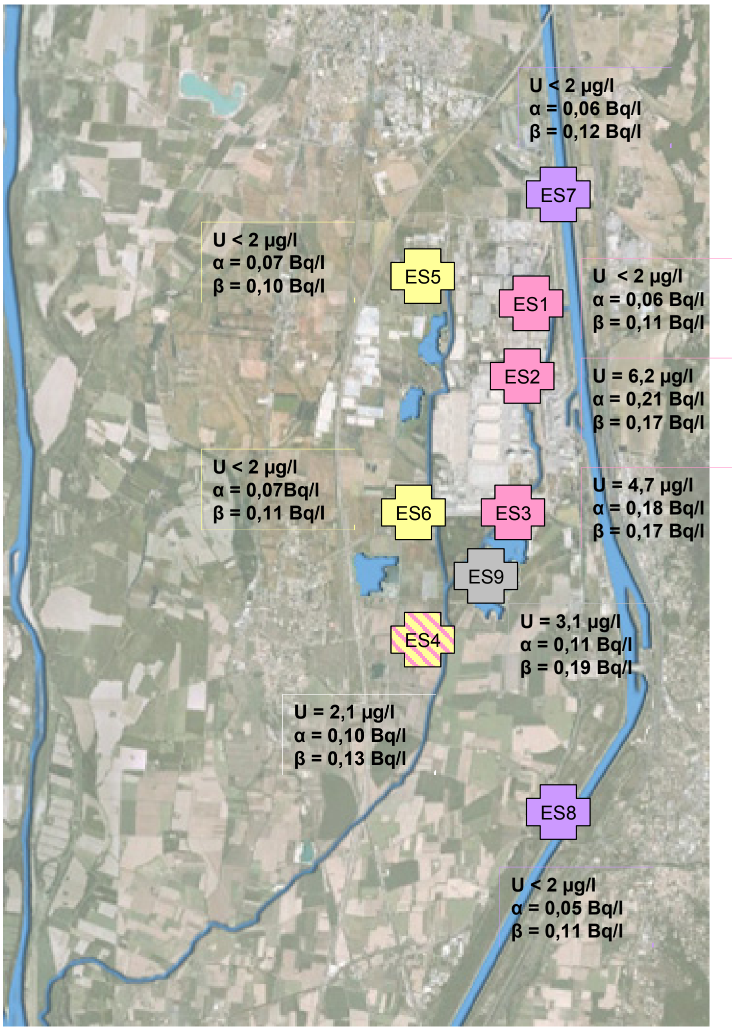

Fig. 3. Results of uranium, gross alpha and gross beta measurements obtained in 2009 in the surface waters.

For groundwater, most regulatory piezometers inside and outside the site show an average of about 1 to 6 $\mu g.l^{-1}$ of uranium, except two piezometers located within the property, with higher values (of the order of several tens of micrograms per litre explained by the past activities of the site.

Regarding the monitoring of the groundwater south of Tricastin, extensive works have been undertaken since 2007 in collaboration with AREVA, ASN, IRSN and DDASS Drome and Vaucluse[4] within a working group which was subsequently extended to other stakeholders. The first measurements campaign has highlighted the existence of a spatial variation of uranium content in the water and the existence of a few points where the uranium concentrations in groundwater exceed significantly the natural background (values between 4 and 10 $\mu g.l^{-1}$ l, instead of about 1 $\mu g.l^{-1}$). The complete inventory of private wells has led to a better spatial definition of the zone with higher levels than the local background and expand the range of values encountered (now values between 1 and 22.7 $\mu g.l^{-1}$).

[4] French regional agencies for Health

AREVA Tricastin invests more than 2 million euros for its regulatory environmental monitoring network, which, in 2009, accounted for 300 monitoring points outside and inside the site, about 14,000 specimens sampled and 27,000 radiological and chemical analyses (water, air, sediment, plants and fish). To carry out and analyze these results, AREVA Tricastin relies on specific teams dedicated to environmental sampling assisted by two laboratories approved by the French Nuclear Authority in the framework of the National Network for Measurement of Radioactivity Environment (RNME).

Beyond knowledge acquisition, AREVA Tricastin sought to reinforce and develop the outline and dialogue with local stakeholders around these issues, by organizing, for example, specific presentations to the attention of the city councils and meetings with nearby residents of the site. Meanwhile, AREVA regularly informs the local Commission of Information (CLIGEET) of the state of the site environment (food survey presented March 21, 2007, state of the water and soil presented July 4 and November 21, 2008, measurement campaign of the radioactivity of private water wells presented July 4, 2008 ...).

In addition, a selection of measurement results representative of the different environmental compartments is updated monthly on the website: http://www.areva.com. Furthermore, the results of measurements of radioactivity into the environment are made available to the public each month via the website RNME established by the ASN and IRSN, at the following address: http:/ / www.mesure-radioactivite.fr

The dosimetric impact on population due to the releases of the site is estimated from the activity of radioactive liquid and gases released during the past year. The computer code COMODORE South-East [Altran, 2007] which includes a dispersion model, is used to assess the transfer of radioactivity into the environment, particularly in the food chain. This assessment is performed according to standards that are the subject of international experts' consensus. For many radionuclides, their level in the environment is well below the analytical detection limits. Therefore, the knowledge of added radioactivity contribution in the environment is not always accessible by measurement. That is why the dosimetric impact can only be assessed by modelling. These calculations take into account all the pathways by which radionuclides can reach humans (air, water, food, etc.) according to the lifestyles and dietary habits of populations. A survey was conducted around the Tricastin site in 2004-2005 to determine precisely the parameters related to dietary habits of these populations.

The evaluation of the dosimetric impact of discharges is made for groups of people likely to be the most exposed among the residents around the Tricastin site. This population group is so called "reference group". For the Tricastin site, because of the plurality of air discharge outlets of different facilities, and because of the local weather conditions, there are several population groups that meet these criteria:

- People in the hamlet of Faveyrolles, north of the site,
- People in the hamlet of Pré Guériné, south of the site,
- Local residents of Bollène-La Croisière, south of the site,
- People in the hamlet of Clos de Bonnot, south of the site.

The establishment of reference stations network of environmental monitoring of the site is perfectly consistent with the location of these groups (Figure 4).

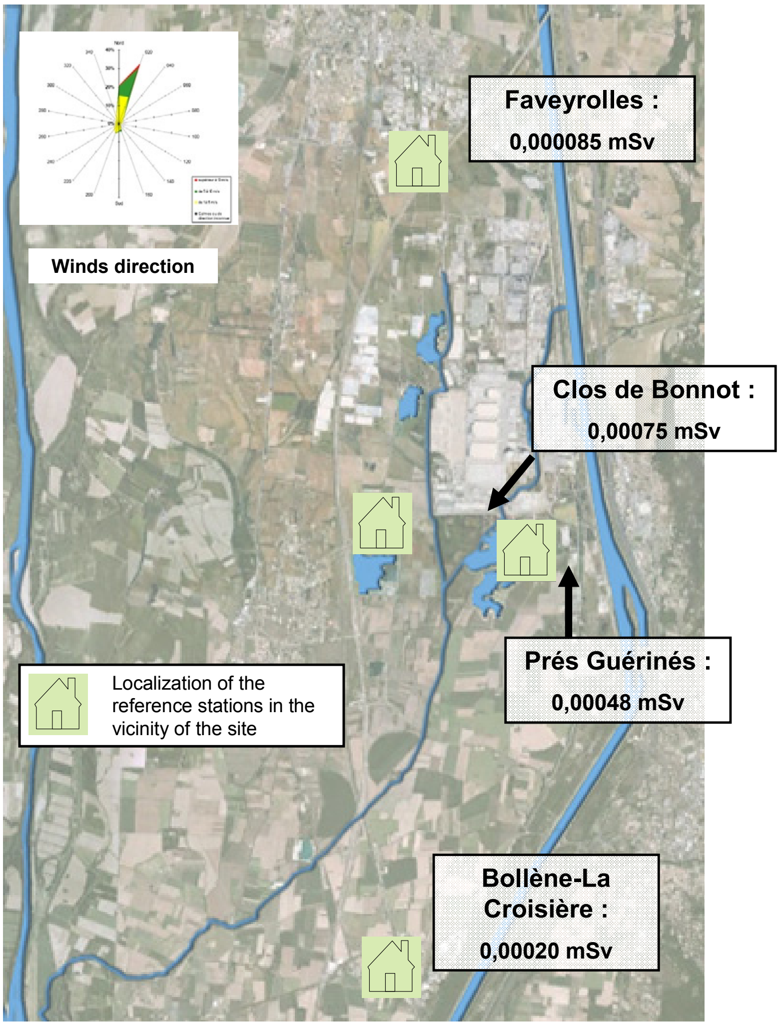

Fig. 4. Map of population groups living along the site and associated doses estimated in 2009.

Historically, in the discharge permits, the reference group labelled “Prés Guériné” was considered as the global benchmark for monitoring the dosimetric impact on people of all the releases of the AREVA Tricastin facilities. Today, although the differences between the calculations for Prés Guériné and Clos de Bonnot hamlets are weak, the population group of Clos de Bonnot shows the highest annual dose.

The annual dose due to actual authorized liquid and gaseous discharge in 2009 was 0.0007 mSv for the population group most potentially exposed (Clos de Bonnot). For other age groups, the doses are lower than those of adults. Considering also the other nuclear facilities nearby, which may contribute to the exposure of these reference groups, these values remain of the order of a thousandth of the limit set by the legislation, which is 1 mSv per year for member of the public.

Future developments

Today, the environmental monitoring network of AREVA Tricastin keep on evolving to go beyond the requirements of routine monitoring and to detect traces of increasingly low radioactivity in the environment. The main projects concern the development of metrology tools to refine as much as possible knowledge of potential impacts on humans and the environment:

- Several research and development projects were launched in 2009 by AREVA Tricastin to develop prototype devices for immediate detection of releases of uranium at concentrations of a few tens of micrograms per litre while the tools currently available only detect, in real time, levels of the order of mg.l-1.

- In autumn 2009, AREVA Tricastin started a research thesis in partnership with IRSN for the measurement of uranium and its progeny in the food chain. In parallel, seasonal campaigns of measurement of uranium in food began in the summer of 2009. The aim is thus to characterize more accurately the background levels in local products but also to respond to societal demands on the levels in the food and agricultural products marketed.

- An inventory of flora and fauna was undertaken in 2009. This study will not only complete the inventory of ecosystems near the site, but also will also define, if possible, species to watch as indicator of the healthy state of the local ecosystems.

Finally, an extensive restoration project of the environmental monitoring records of the site is underway. All these data will be soon structured through a GIS (Geographic Information System) to perform in depth analyses.

Conclusion

Set up from the start of the site in 1962, the environmental monitoring network of the AREVA Tricastin site has evolved over time to meet more and more precisely the multiple objectives of environmental monitoring ; 1) to show compliance with authorities' requirements ; 2) to be able to detect a drift in the observed levels ; 3) to enable the assessment of impacts of industrial activities ; 4) to ensure the balance between environmental quality and industrial practices ; and 5) to inform the public of the radiological state of the environment. Thus, thousands of data were acquired on the radioactivity of all environmental compartments as well as on the functioning of local ecosystems (groundwater piezometric levels, temperatures, flows of rivers ...). Today, the environmental monitoring network of AREVA Tricastin goes beyond the requirements of routine monitoring to provide innovative solutions for monitoring the radioactivity (especially for uranium) in the environment.

The public opening of the national network for measuring environmental radioactivity (RNME) on the 1st of February 2010 helps to enhance transparency and credibility of the operators on the quality and reliability of the measurements made in the framework of regulatory monitoring programs.

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^{137}Cs activity concentrations in Polish meats – Current status and dose assessment for consumers

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Abstract
According to the Commission recommendation and the national law, the Polish Veterinary Inspection implemented a programme of radioactivity monitoring in a wide range of foodstuffs of animal origin. Samples of beef, lamb, pork, poultry, and game meat were taken at meat processing facilities and then transported to regional laboratories for analyses. Radiocaesium activity concentrations were determined by gamma-ray spectrometry using scintillation (NaI(Tl)) detectors. Generally, the radiocaesium activity concentrations in of beef, lamb, pork, and poultry were very low, and, in most of the measurements, reached MDA values. The slightly elevated radiocaesium activity concentrations, observed in several beef samples, may be partly explained by semi-free rearing of these animals. In some game meat samples fairly high levels of ^{137}Cs were still noted. Ingestion of a large portion of game meat in a daily diet by hunters' families may increase ^{137}Cs uptake in this group of consumers. Due to low radiocaesium activity concentrations in meat from domestic animals, the mean effective doses may be considered very low. The annual effective dose received by individuals in Poland and attributed to ^{137}Cs intake with meat was estimated to be a few μSv. Somewhat higher effective dose was calculated for hunters' families as a result of increased game consumption. Therefore, taking into account the permitted annual public dose, consumption of Polish meats are truly safe in terms of contamination with radiocaesium.

Introduction
Atmospheric nuclear weapon tests and accidents at nuclear installations are the main sources of anthropogenic radioactive contaminations of the environment. Radioactive nuclides may affect human beings throughout the food chain. Among the radionuclides released, the most important are considered those of radiocaesium, particularly ^{137}Cs. This radioisotope has a relatively long half-live (30.17 years), and as a chemical and metabolic analogue of potassium is accumulated in muscle tissues regarded as a main food product of animal origin.

An increased level of radiocaesium was noted in many foodstuffs in the early years after the Chernobyl nuclear accident. The radiocaesium levels have remained high in products such as game meat, mushrooms and berries. It is postulated that the transfer of radiocaesium from soil via plants and mushrooms to animals is much higher in forest ecosystems than in agricultural environments, and the radionuclide decrease is very slow (Johanson and Bergström 1994, Kiefer et al. 1996, Kostiainen 2007, Strebl and Tataruch 2007).

One of the most important routes of radionuclide penetration into human organisms seems to be the alimentary tract. Therefore, according to the EU and national law, the Polish Veterinary Inspection implemented a programme of radioactivity monitoring in a wide range of foodstuffs (beef, lamb, pork, poultry, game meat, fish, eggs and milk). Similar programmes are also conducted in other EU member states (Kameník et al. 2009, Madruga 2008, Varga 2006).

The objective of this work was to determine the current levels of radiocaesium in Polish meats, and assess the effective doses for consumers.

Material and methods

Samples were taken at farms or food processing facilities and then transported to laboratories for analyses. The sampling scheme in the regions of Poland in 2008 is shown on Figure 1.

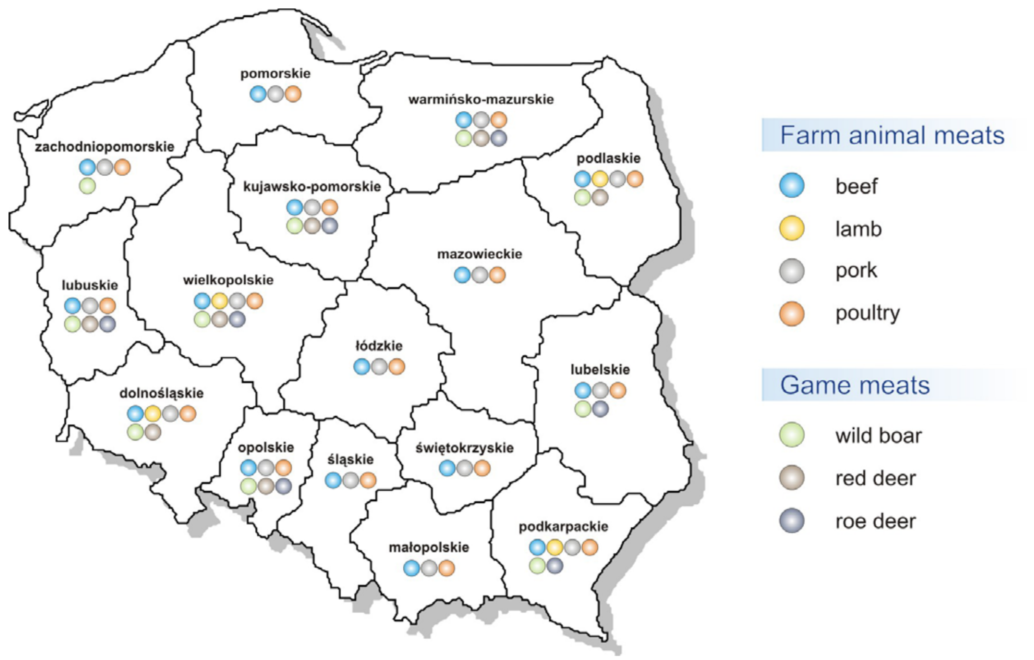

Fig. 1. Sampling in the regions of Poland in 2008.

Muscle samples were chopped, ground and put in 450 cm^3 Marinelli beakers. The sample geometry was similar to that of a source used for the detector calibrations in laboratories.

^{137}Cs and ^{134}Cs activity concentrations were determined by a gamma-ray spectrometry technique using scintillation detectors (NaI(Tl), 2’’) (Scionix, The Netherlands) placed in lead shieldings. Counting time was set to 72,000 s. Gamma-ray spectra were acquired and evaluated using the Genie 2000 software (Canberra, USA).

For radiation dose assessment originating from ^{137}Cs, a dose conversion factor of $1.3 \cdot 10^{-8}$ Sv/Bq was applied to assess the effective doses received by consumers (International Atomic Energy Agency 1996). The effective dose was defined by the equation.

$$H_{\mathrm{E}} = A_k m D_{k\,\mathrm{f}(k)}$$

where H_{E} is the effective dose (Sv); A_k – the concentration of k radionuclide (Bq/kg); m – the quantity of consumed food (kg) and $D_{kf(k)}$ is the dose conversion factor for k radionuclide (Vilic et al. 2005).

Results

Radiocaesium activity concentrations

The ^{134}Cs activity concentrations were negligible (always below MDA values) in all the samples measured.

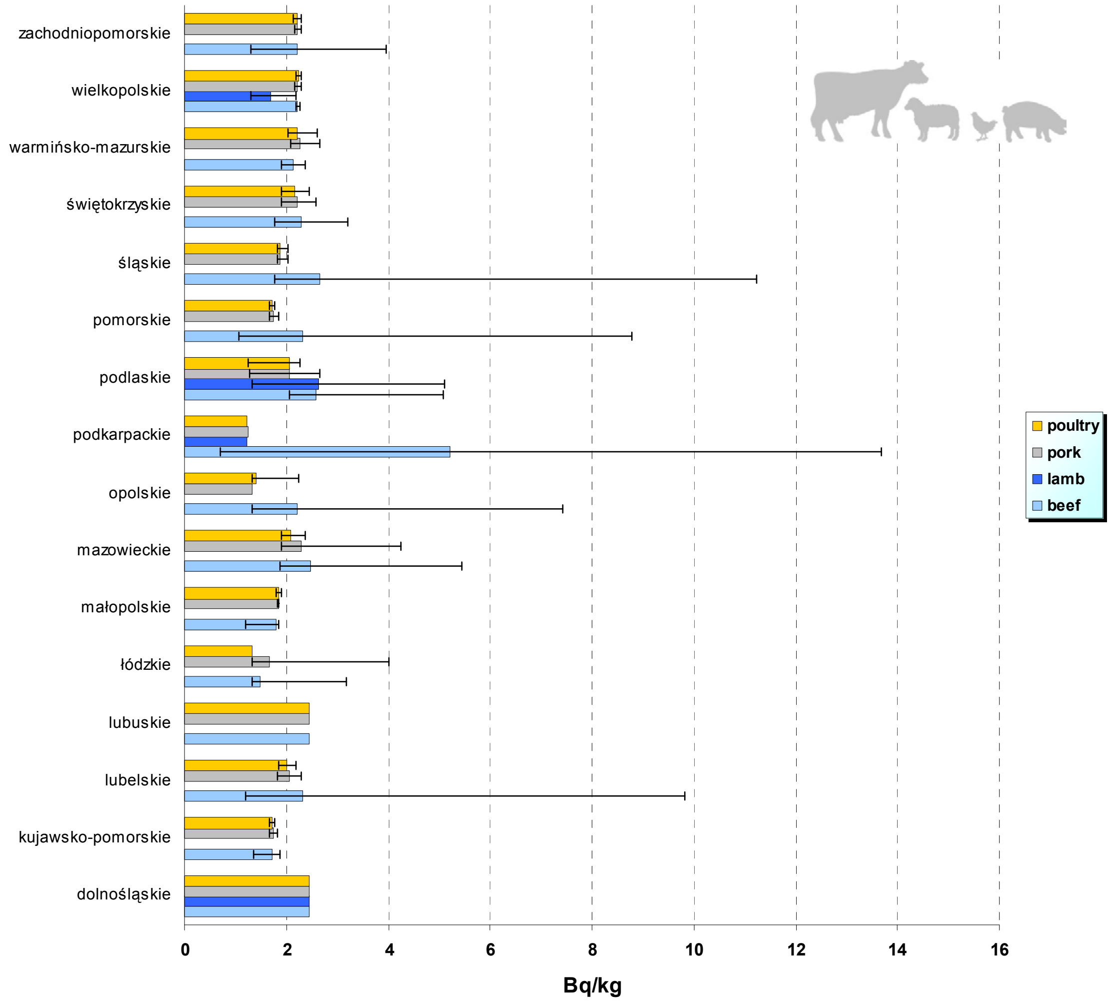

Fig. 2. ^{137}Cs activity concentrations in farm animal meats from the regions of Poland in 2008 (mean and range; no error bars – only 1 sample analysed or all the activity values were below MDA).

The ^{137}Cs activity concentrations were below MDA values in most cases. It should be stressed that because of small differences in analytical equipment, mostly in lead shields, the detection limits for caesium radioisotopes varied between laboratories. The results of ^{137}Cs measurements (averages and ranges) in farm animal and game meats are given in Fig. 2 and Fig. 3, respectively. The results are presented on a fresh-weight basis. As it can be seen, the mean values were about 2 Bq/kg in farm animal meats.

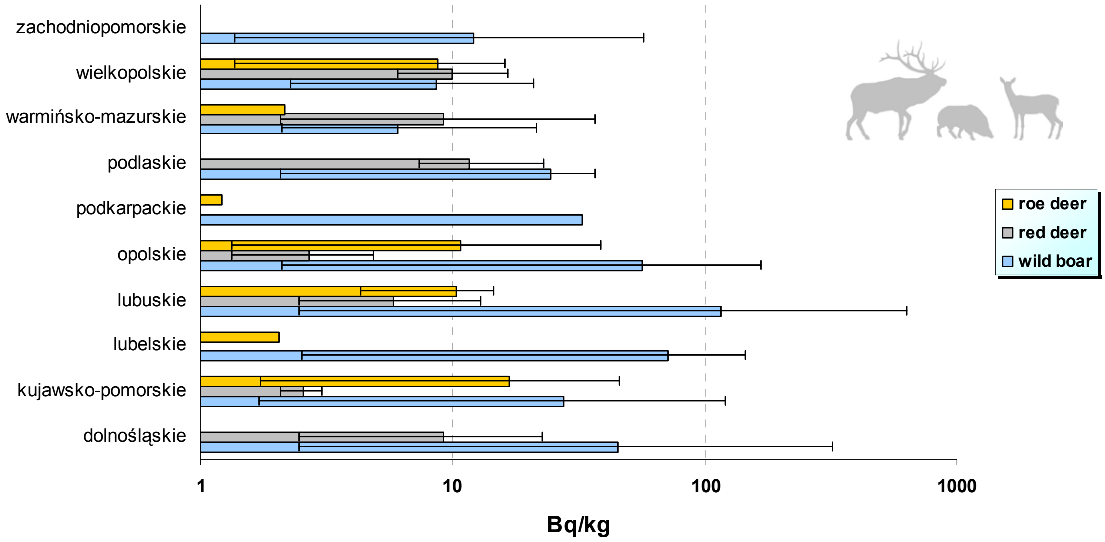

Fig. 3. ^{137}Cs activity concentrations in game meats from the regions of Poland in 2008 (mean and range; logarithmic scale; no error bars – only 1 sample analysed).

The ^{137}Cs activity concentrations were very variable between and among the game meat samples. Even in the same region they ranged from MDA values to more than 100 Bq/kg. The highest ^{137}Cs radioactivity concentration (> 600 Bq/kg) was noted in a wild boar meat sample.

Radiation dose assessment

Since the ^{137}Cs activity concentrations were below the detection limits in the majority of samples, the effective doses originating from meat consumption have been overestimated.

Table 1. Effective doses from consuming 1 kg of farm animal meat.

Farm animal meat	H_E per kg of farm animal meat (μSv)
beef	0.031
lamb	0.026
pork	0.026
poultry	0.025
combined	0.027

Mean effective doses from consuming 1 kg of farm animal meat and game meat are presented in Table 1 and Table 2, respectively. The mean effective dose originating from game meat was almost 10 times higher than that from farm animal meat. The highest mean effective dose was calculated for wild boar meat.

Table 2. Effective doses from consuming 1 kg of game meat.

Game animal meat	H_E per kg of game meat (µSv)
wild boar	0.52
red deer	0.09
roe deer	0.10
combined	0.24

For the wild boar sample with the ^{137}Cs activity concentration exceeding 600 Bq/kg, an effective dose of 8.26 µSv per kg was calculated.

Discussion

Due to a relatively short physical half-life (2.06 years) and a lower deposition after the Chernobyl accident (De Cort et al. 2001), the current radioactivity concentrations of ^{134}Cs in foodstuffs are negligible, and its contribution to the effective doses received by consumers may be omitted.

The highest radiocaesium activity concentrations in meat were observed in the summer of 1986. In subsequent years, contamination with radiocaesium decreased steadily. Since 1994, ^{134}Cs radioactivity concentrations have been below the detection limit in most samples. Mean ^{137}Cs radioactivity concentrations reached a level of 2–3 Bq/kg. Moreover, the differences in radiocaesium activity concentrations among the meats examined were less visible (Muszynski et al. 2003).

The ^{137}Cs activity concentrations in meat measured in 2008 were similar to those noted in previous years. The highest values were found for game meats. There is a significant variation in the ^{137}Cs activity concentrations between and among game animal species. It may be attributed to different contamination levels even in neighbouring areas and different feeding habits of game animals (Vilic et al. 2005).

In Poland after the Chernobyl accident, the highest ^{137}Cs activity concentration (227 Bq/kg – mean, 4–4,357 Bq/kg – range) was observed in roe deer followed by red deer and wild boar. During the next five years, radiocaesium activity concentrations steadily decreased to tens Bq/kg (Grabowski et al. 1994). In our study, the measured activity concentration values were even lower. Similarly to the Austrian findings (Strebl and Tataruch 2007), in recent years the average ^{137}Cs activity concentration in Poland has been higher in meat from wild boar than that in meat from roe deer or red deer.

It is well documented that some mushrooms, readily consumed by animals, show high ability to accumulate radiocaesium (Bem et al. 1990, Grabowski et al. 1994, Malinowska et al. 2006, Mietelski et al. 1994, Pietrzak-Flis et al. 1996). Bay bolete (*Xerocomus badius*), one of the most wide-spread mushroom species in Poland, reveals a unique radiocaesium accumulation feature. Pigments from the bay bolete cap cuticle (badion A and norbadion A) can complex potassium and its chemical analog caesium (Aumann et al. 1989). Deer truffle (*Elaphomyces granulatus*), which also contains particularly high levels of radiocaesium, may be another radionuclide source for game animals, particularly for wild boars (Hohmann and Huckschlag 2005). These authors studied a wild boar stomach content to test the influence of *E. granulatus* intake on wild boar radiocaesium contamination. They found deer truffles in significantly higher proportions in stomachs of animals with maximum contamination levels. Moreover, animals consuming deer truffles might digest contaminated soil components.

Between 1998 and 2008 German and Belarusian researches analysed more than 600 wild boar meat samples from Baden-Württemberg. The muscle radiocaesium levels varied widely from less than 5 to more than 8 000 Bq/kg (fw). The investigators observed maximum values of ^{137}Cs activity concentrations during the winter periods. They connected this pronounced maximum with the consumption of highly contaminated deer truffles available for wild boar in winter (Semizhon et al. 2009). These mushrooms grow in a depth of 6–8 cm in spruce forest soil corresponding to Oh/Ah (organic layer/mineral layer) horizon where the peak of ^{137}Cs activity was observed (Pietrzak-Flis et al. 1996).

The MDA values (1–2 Bq/kg), which were taken for determination of radiation doses, are characteristic for analytical systems used by the laboratories participating in veterinary radioactivity monitoring in Poland. In Hungary, γ-ray spectrometry using NaI(Tl) scintillation detectors are considered as a screening method. For ultimate measurements, γ-ray spectrometry using high purity germanium detectors (HPGe) is applied. The typical limit of detection for meat is in the order of magnitude of 10^{-1} or 10^{-2} Bq/kg fresh weight (Varga et al. 2006). So, a broader implementation of this technique in Poland should produce more reliable results of γ-emitting radionuclide measurements and calculation of the radiation doses for consumers.

Assuming that the mean annual consumption of farm animal meat is around 75 kg and the mean effective dose is 0.027 μSv per kg, an individual in the Polish population may receive the annual effective dose of about 2 μSv.

Lithuanian researchers calculated the average annual effective dose caused by ^{137}Cs at a level of 0.02 μSv, whereas the effective dose derived from natural ^{40}K was 0.14 mSv. So, the main contributors to the total annual effective dose seem to be naturally occurring radionuclides (Butkus et al. 2006). Even in the Ukrainians with the maximum intake of ^{137}Cs, the annual effective dose of 2.7 mSv was estimated (Shiraishi et al. 2008). This value corresponds to the natural radiation dose (2.4 mSv) for worldwide inhabitants (United Nations Scientific Committee on the Effects of Atomic Radiation 1988).

Our studies showed that in some wild boar meat samples the ^{137}Cs radioactivity concentrations may still exceed 600 Bq/kg (maximum permitted level) even more than 20 years after the Chernobyl accident. However, average consumption of game meat in Poland is very low and the contribution of this sort of food to the effective doses received by consumers is negligible, with the exception of hunters and poachers and their families. Supposing that the mean annual consumption of game meat in these populations is around 20 kg (Haldimann et al. 2002), the average annual effective dose of 4.8 μSv was calculated.

Conclusions

Considering the permitted annual public dose at a level of 1 mSv, the consumption of farm animal meat is truly safe in terms of contamination with radiocaesium.

Some precautions may be taken in populations with elevated intake of natural food products.

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Public doses due to tritium emissions from Cernavoda NPP

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Abstract

Heavy water reactors have a large tritium load. In a CANDU reactor most of the tritium is formed in the thermal-neutron-capture reaction, 2H(n,ƒ×)3H which occurs both in the moderator and heat transfer system. Very small amounts of tritiated heavy water may escape from moderator and heat-transport systems of CANDU reactors during maintenance and normal operation. Tritium emissions of Cernavoda NPP were continuously monitored. The tritium concentrations in environmental samples were monitored as part of routine program and public doses were calculated. This paper presents the supplementary tritium doses for a member of public, estimated using HTO concentrations in environmental samples and OBT doses estimated based on Candu Owners Group studies ¡§OBT/HTO Ratio in plants¡¨ and ¡§Contribution of Organically Bound Tritium to total Tritium Dose¡¨ .

New Swedish regulations for clearance of materials, rooms, buildings and land

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Abstract

The Swedish Radiation Safety Authority (SSM) has developed new regulations for clearance of materials, rooms, buildings and land. The new regulations develop and widen the scope of the current Swedish regulations on the release of goods and oil from nuclear facilities. The new regulations put requirement on the procedures for clearance of materials, rooms, buildings and land during all stages of nuclear and non-nuclear practices. The proposed clearance levels are based on recommendations from the European Commission. The regulations are scheduled to enter into force in January 2011.

Introduction

The Swedish Radiation Safety Authority (SSM) is the regulatory and supervisory authority for radiation protection and nuclear safety in Sweden. SSM has developed new regulations for clearance of materials, rooms, buildings and land. The regulations specify under what conditions objects that may have been contaminated[1] by radionuclides in licensed practices with ionizing radiation, can be released from further regulatory control by SSM. Thereby materials and resources can be used and managed in a rational way, in a manner that is acceptable from a radiation protection point of view.

The regulations will replace the current Swedish regulations for release of goods and oil from nuclear facilities, SSMFS 2008:39 (Ref 1) and introduce new rules for clearance of materials from non-nuclear practices as well as new rules for clearance of rooms, buildings and land. The work builds on work performed by the former Swedish Radiation Protection Authority (SSI). A draft version of the regulations was sent for a broad national review in 2006 and a large number of comments were received (see Ref 2 for a description of the draft version). SSM developed the regulations further and a new version was sent for a broad national review in 2009. The regulations have thereafter been adjusted and clarified in some aspects and will soon (spring 2010) be notified to the European Commission (EC) according to the Euratom treaty, before being decided by SSM. The regulations are planned to enter into force in January 2011.

[1] In this paper, the term contamination means any occurence of radionuclides on the surface or in the bulk of an object that is caused by the actual practice.

After the national review in 2006, the Swedish nuclear industry took an initiative to develop a common handbook or code of practice for clearance procedures (Ref 3). The handbook is planned to be issued by the Swedish Nuclear Fuel and Waste Management Company (SKB), following SSM's decision on the regulations.

Background

The work to revise the current regulations for clearance of goods and oil from nuclear facilities (Ref 1) was initiated for several reasons. The main reasons were:

- The methods for management of conventional wastes have changed substantially since the current clearance levels were developed.
- The current regulations were originally not intended for large amount of materials arising from decommissioning of nuclear installations.
- There are currently no general clearance regulations for non-nuclear practices in Sweden.
- A need for harmonization with international recommendations from the EC and the IAEA (Refs 4, 5, 6)

The work was prepared for several years by following the international development of clearance recommendations and regulations. The actual revision project started in 2004 with an evaluation of the current regulations. The evaluation was based on inspections of the clearance procedures at the Swedish nuclear installations (four NPP:s, one fuel fabrication plant and one site with material test reactors, hot-cells for material investigations and radioactive waste management facilities). Another important preparatory work was the investigation of the applicability of the EC recommendations for buildings and building rubble in Sweden (Ref 7). Also, some of the requirements were developed on the basis of experiences from decommissioning projects in Sweden and other countries. During the work, several seminars have been held to discuss specific topics on clearance with the licensees and other interested parties. On several occasions, information and progress reports have been given to radiation protection experts of the licensees. International experiences have been included in the work, by invitation of experts from Germany and Spain to seminars in Sweden, and by participation on international conferences.

The regulations

The regulations contain the following subsections, which will be presented below.

- Introductory requirements
- Area of application
- Clearance
- Measures for clearance
- Clearance levels
- Rules for implementation of clearance levels
- Competence
- Reporting
- Other clearance options and exceptions

Introductory requirements

The introductory requirements describe the purpose of the regulations and give a definition of the term clearance. It is stated that clearance means that the Radiation Protection Act (1988:220) and the Act (1984:3) on Nuclear Activities shall no longer be applied to materials, rooms, buildings and land that may have been contaminated by radioactive substances in such practices with ionizing radiation that are subject to licensing. Thus, it is important to note that the regulations only deal with exemptions from the acts on *radiation protection* and *nuclear safety*, in the case of potentially contaminated objects due to *licensed practices*. Other legislation concerned with hazards from radioactive substances might thus still be applicable, for example legislation on transport of radioactive materials and legislation on non-proliferation of nuclear materials.

Area of application

The area of application is specified both in terms of what is included in the scope of the regulations and what is excluded. Included are: Materials, including waste, rooms, buildings and land that *may* have been contaminated by radioactive substances in practices with ionizing radiation that are or have been carried out under a license according to the Radiation Protection Act or the Act on Nuclear Activities.

Excluded are:

1. releases of radioactive substances to air or water,
2. naturally occurring radioactive substances that are not part of the license for the actual practice with ionizing radiation,
3. practices that only involve naturally occurring radioactive substances and that are carried out without any purpose of use of the radioactive, fissile or fertile properties of radioactive substances,
4. nuclear medicine.

Thus, only radionuclides that have been part of the licensed practice have to be considered when applying the regulations. Nuclear medicine is excluded since contaminated objects from such practice need special consideration and are dealt with in other regulations from SSM (for example contaminated clothes and sheets).

Clearance

Clearance of materials is treated differently than clearance of rooms, buildings and land. For materials, it is stated that a material is subject to clearance when it has been checked in accordance with the regulations and it has been concluded by the licensee that the content of radioactive substances is less than the clearance levels given in the regulations. Thus a mandate is given to the licensee to perform clearance under own responsibility, without any obligatory supervision or decisions by the authority.

For rooms, buildings and land that shall no longer be used for the actual practice, it is stated that the licensee shall take all measures that are needed to achieve clearance. An application on clearance shall thereafter be given in to SSM, who decides on clearance. To allow for some flexibility, the licensee may temporarily use rooms and buildings that have been checked for contamination for other purposes without any decision on clearance by SSM.

Measures for clearance

An obvious requirement is that objects have to be checked for radioactive substances before clearance. The regulations state that the checks shall be based on measurements. Thus it is not sufficient to only estimate the occurrence of radioactive substances based on for example knowledge of the possible ways of contamination of the object. However, it is allowed to apply nuclide vectors or correlation factors between the occurring nuclides if the vectors or factors have been established from measurements on samples that can be considered representative for the actual composition of radionuclides.

It is further stated that the methods and extent of the checks shall be adapted to the risk of contamination, the extent of contamination, and to the characteristics of the potentially contaminated object.

Also, the methods shall correspond with Swedish or international standards, or guiding documents from SSM. Although no such guiding documents are planned by SSM today, they could be issued as a complement to the standards or to override the standards (if some part of a standard for some reason would be considered not appropriate).

Some decontamination efforts may be warranted to keep the amount of radionuclides in cleared materials, and thus also the potential exposure of members of the public, as low as reasonably achievable. It is therefore stated, as an advice attached to the regulations, that decontamination should be considered and that easily removable contamination should be removed before clearance. In another advice attached to the regulation, it is stated that systems, equipment and components should be removed before clearance of rooms and buildings, in order to facilitate decontamination and activity checks of the remaining structures.

An important prerequisite for allowing the licensee to perform clearance under own responsibility is to ensure that it is performed in a well-planned, qualified and systematic manner. In order to achieve this, it is stated that the licensee shall develop a written control program that shall specify the methods and who is authorized to conduct the checks. The control program shall also describe how quality assurance, self-assessment and documentation of results is done. For clearance of rooms, buildings and land as well as for clearance of more than 100 tonnes of materials per year, the control program shall be given in to SSM. Thereby, SSM has the possibility to review the program, judge if it is appropriate and give comments and suggestions for improvements. If needed, SSM can also require the licensee to change or improve the program in a specified manner.

Also, to facilitate supervision by SSM, it is stated that the performing of checks as well as the results of checks shall be documented and that the documents shall be preserved for ten years after clearance. The only exception from this rule concerns tools and equipment that are cleared after temporary use in the practice (se below).

Furthermore, the regulations explicitly forbid dilution of contaminated materials with the purpose to achieve clearance. It is also stated that liquids shall be cleaned from radioactive particles as far as practically achievable, in order to prevent exposure of the public due to accumulation of radionuclides in filters or in sediments.

Clearance levels

As required in the European Community Basic Safety Standards (Ref 8), the clearance levels are based on the basic criteria for exemption (Annex 1 of Ref 8). The criteria are specified as a maximum effective dose to any member of the public of the order of 10 microsieverts per year and a collective effective dose of less than about 1 mansievert per year (or assessment that clearance is the optimum option).

Clearance levels are specified for four different cases:

1. Materials for unrestricted further management (reuse, recycling or disposal)
2. Oil and hazardous waste for incineration or disposal
3. Rooms and buildings for reuse
4. Buildings for demolition

For materials, oil and waste, nuclide specific clearance levels for activity concentration (becquerel per gram) are given for 196 nuclides. For materials for unrestricted further management (category 1 above), the values are those established in European Commission recommendation RP 122, Ref 4. For oil and hazardous waste for incineration or disposal (category 2 above), the values are 10 times higher for all nuclides (except for three nuclides for which the clearance values are the same as those recommended in RP 122 in order to not exceed the exemption values of the European BSS, Ref 8).

Clearance levels are also stated for surface contamination on materials. As in the current regulations (Ref 1), the values are 40 kBq/m2 in total for beta and gamma emitting radionuclides and 4 kBq/m2 for alpha emitting radionuclides.

In general, both the levels for activity concentration and the levels for surface contamination have to be complied with for clearance. The only exception is for tools and equipment that have been used temporarily in the practice and that are intended to be used also after clearance (i.e. not being sent for demolition or recycling). Another prerequisite for this exception is that contamination shall only be expected to be found on accessible surfaces.

For rooms and buildings (categories 3 and 4 above), nuclide specific clearance levels are given for 104 nuclides. The values are those established in European Commission recommendation RP 113, Ref 5, Table 1 (for reuse or demolition) and Table 2 (for demolition only).

No clearance levels are given for land in the regulations. The current approach is that such clearance levels will be decided by SSM on application by the licensee based on site specific considerations.

Rules for implementation of clearance levels

In the regulations, some rules are given on how the clearance levels shall be interpreted and implemented. Firstly, the summation formula shall be used for contamination that consists of several nuclides, see for example Ref 4. Moreover, the activity concentration may be determined as a mean value for up to 1000 kg (in the case of materials) or 1 m^2

(in the case of room and building surfaces, i.e. walls, floors and ceilings) when comparing with the clearance levels.

Surface contamination on materials may be determined as a mean value for up to 0.03 m2 when comparing with the clearance levels.

As mentioned in the beginning of the paper, the Swedish nuclear industry has jointly developed a guiding document for clearance. The document gives further guidance on several different issues that should be considered in the clearance process.

Competence

There is a general requirement that the involved personnel shall have sufficient competence for the measures that they take and the judgements that they make in the clearance process. This is further specified by requirements on knowledge of which radionuclides that occur in the practice and to what extent they may occur as contamination. The personnel shall also be educated in harmful effects and risks with ionizing radiation, in rules and routines for clearance and in methods for sampling and measurement, including inaccuracies and limitations of the methods.

Reporting

A yearly report shall be given in to SSM in order to facilitate SSM's supervision and retrospective follow up on the use of the clearance option for management of materials and waste. The report can also serve the purpose of public insight in the process of clearance. The report shall specify the amounts and kinds of materials that have been subject to clearance, the nuclide specific concentration of radioactive substances in the materials, and, in the case of oil and hazardous wastes, the receivers of the materials. No report is required from licensees that have cleared less than 1000 kg of materials during the past calendar year, in order to reduce the administrative burden on licensees.

Other clearance options and exceptions

SSM can, on a case-by-case basis, decide on other clearance levels than those specified in the regulations. This requires an application by the licensee supported by an analysis of the radiological consequences of the proposed management or use of the object(s) after clearance. If there are circumstances that limit the possible ways of public exposure, higher clearance levels may be accepted.

SSM can also permit exceptions from the requirements in the regulations, if there are special reasons, as long as the purpose of the regulations is not violated.

Discussion

Several difficult issues have to be dealt with when regulating and implementing clearance procedures. A general feature of the necessary considerations is that they include a large portion of judgement.

The derivation of proper clearance levels is a fundamental issue that is based on judgement of the expected or potential exposure that should be accepted from cleared objects. Generally, such judgement leads to the establishment of a dose criterion. In international guidance and legislation, 10 microsieverts per year is commonly used, since it is generally regarded as trivial. In some cases, for example in Ref 6, consideration is also made about low probability exposure scenarios, with the application of higher dose criteria in such cases (1 mSv/year in Ref 6).

To derive clearance levels, considerations must be made about expected and potential scenarios for exposure. This step also includes a large portion of judgement and considerations about the practices that can give rise to contaminated objects as well as about the further destiny of cleared objects (for example in the selection of considered radionuclides, identification of scenarios and choice of parameters and computational models for exposure calculations). Special attention is normally given to such realistic scenarios that would cause a higher exposure than other foreseeable scenarios. This means that the most probable doses to members of the public should be well below the dose criterion and that the dose criterion should only be exceeded in fairly extreme cases.

Internationally, a lot of work has been done on clearance levels for activity concentration in materials, which has been a good help in the development of the new Swedish regulations. However, for surface contamination there is very sparse guidance. Thus, there is a need for international recommendations on general clearance levels for surface contamination. There is also a need for further harmonization between clearance levels and transport regulations (see Ref 9).

The treatment of naturally occurring radionuclides poses a special problem when it comes to clearance. There are practices that involve naturally occurring radioactive materials (NORM) for which the application of the 10 microsievert criterion would cause restrictions on the operations and on the treatment of residual materials that would probably not be motivated from a radiation protection point of view, taking into account the natural abundance of NORM. Instead, higher dose criteria (300 microsieverts per year, Ref 10) or clearance levels corresponding to higher doses (Ref 6) are internationally recommended.

However, for practices in the nuclear fuel cycle, where NORM has been processed in view of its fissile properties, the 10 microsievert criterion should be applied, as for other radionuclides in the nuclear fuel cycle (fission and activation products)[2]. In this case the natural abundance of NORM causes detection problems, which might mean that less confidence level has to be accepted for some nuclides, or even that higher clearance levels can be applied, based on optimization of radiation protection (cf Ref 4, footnote of Table 1).

Once the clearance levels have been established, the question is how to show compliance with the levels. As mentioned earlier, this involves judgement about the potential contamination of the object, selection of appropriate methods for determination of the contamination and proper interpretation of the results. Considerations must be made about which nuclides shall be determined, their expected distribution, the properties of the contaminated materials, confidence levels and detection limits of measurements, logistics, calibration of equipment, quality assurance, etc. Many of these issues have to be solved in a way that is adapted to the actual circumstances. In the new Swedish regulation, this step in the procedure is a

[2] Actually, this is the main reason for the decision to use the clearance levels recommended in Ref 122 instead of Ref IAEA. This is further described in Ref 9.

responsibility of the licensee to a large extent. This emphasizes the importance of skills and competence of the personnel responsible for clearance.

Clearance levels are often regarded as "perfectly safe" levels of contamination which can be disregarded from a radiation protection point of view. However, application of the principle of optimization of radiation protection means that some efforts for decontamination before clearance may be warranted (Ref 9). It could also be warranted to direct the release of materials into the society in such a way that exposure is avoided.

Clearance procedures are closely connected to means and routines for control of radioactive materials and waste within a practice. To avoid exposure of the public or deterioration of public confidence it is crucial that no radioactive material, on false grounds, leaves the practice as cleared material. To prevent this, the clearance procedures should involve additional "lines of defence" to identify and prevent such illicit clearance. This could include random additional checks and gate monitors for goods leaving the site

Conclusions

Over a period of several years, SSM has developed new regulations for clearance of materials, rooms, buildings and land. The work has involved extensive consultation with stakeholders and information to licensees and other interested parties. In parallel, the Swedish nuclear industry has developed a code of practice as a common platform for implementation of the requirements in the management systems of each company. The regulations, the industry code of practice, as well as the awareness and common understanding that has developed during the process, should provide a good basis for robust procedures so that clearance will be done in a reliable and transparent manner.

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The approach to assessing doses to humans in the Posiva safety case

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Abstract

Posiva Oy is responsible for implementing a final disposal programme in Finland for spent nuclear fuel. The next step of the nuclear licensing is by the end of 2012 submit a construction licence application for a KBS-3 type of repository at the Olkiluoto site. Currently, a safety case is produced to support this application, were a preliminary version was completed in 2009. Assessing doses to humans is a key part in order to assess compliance with regulatory requirements on the long-term safety. A prospective deterministic dose assessment, based on the ICRP concept of assessing doses to the *representative person* has been developed. Exposure characteristics are based on site-specific conditions and regional land use. For example, the number of exposed persons is limited by the site-specific capabilities to produce food and drinking water. Due to the temporal scale of the assessment, the surface environment will undergo significant development and many generations may be exposed. This is taken into account by deriving full dose distributions (the dose to each potentially exposed individual) for all generations during the assessment time window. Doses to representative persons, both for the more highly exposed individuals and for the other people in the exposed population, are then identified from the dose distributions. This paper present the methodology applied in the interim safety case to derive annual doses to humans from concentrations of radionuclides in environmental media, resulting dose distributions from the calculation cases analysed, and the doses used when assessing compliance with the regulatory dose constraints for humans.

Introduction

Posiva Oy (Posiva) was established in 1995 by the two Finnish nuclear power companies, Teollisuuden Voima Oyj (TVO) and Fortum Power and Heat Oy (Fortum), to implement the final disposal programme for spent nuclear fuel and to carry out the related research, technical design and development. The spent nuclear fuel is planned to be disposed of in a KBS-3 type of repository to be constructed at a depth of about 400 metres in the crystalline bedrock at the Olkiluoto site. Posiva is currently preparing for the next step of the nuclear licensing of the repository, which involves submitting the construction licence application for a spent fuel repository by the end of 2012. A safety case will be produced to support this licence application; the Posiva plan for conducting the safety case is documented in (Posiva 2008).

A key contributor to the safety case is the biosphere assessment. The overall aims of the biosphere assessment are describe the future, present, and relevant past conditions at, and prevailing processes in, the surface environment of the Olkiluoto site; to model the transport and fate of radionuclides hypothetically released from the repository through the geosphere to the surface environment; and to assess possible radiological consequences to humans and other biota. For the dose assessment, a graded approach based on three tiers has been applied, where Tier 1 and Tier 2 involve conducting generic evaluations to screen out radionuclides that have insignificant radiological consequences. Tier 3 is based on based on site-specific state-of-the-art radionuclide transport modelling and dose assessment.

The regulatory requirements for the long-term safety of a geological repository in Finland are set out in detail in the Radiation and Nuclear Safety Authority's (STUK) Guide YVL E.5 on disposal of nuclear waste (STUK 2009). The criteria for the time window to be addressed in the biosphere assessment and the criteria for protection of humans within this time window are stated as follows (STUK 2009): *"In any assessment period, during which the radiation exposure of humans can be assessed with sufficient reliability, and which shall extend at a minimum over several millennia: 1) the annual dose to the most exposed people shall remain below the value of 0.1 mSv, and 2) the average annual doses to other people shall remain insignificantly low."*. For the larger groups, no fixed dose constraint is set, but the acceptability of the doses depends on the number of exposed people, and they shall not exceed values from one hundredth to one tenth of the constraint for the most exposed people (STUK 2009).

The graded approach to dose assessment has been developed for, and applied in, the recently produced interim safety case (Hjerpe et al. 2010, Posiva 2010). This paper presents the methodology for assessing doses to humans in Tier 3 and selected results from the calculation cases analysed for the interim safety case.

Methodology

The main objective of the assessment of doses to humans is to assess compliance with the above-mentioned regulatory dose constraints defined in terms of annual effective doses. The ICRP (International Commission on Radiological Protection) has provided guidance (ICRP 2007a) on how to assess individual doses for the purpose of establishing compliance with the ICRP recommendations (ICRP 2007b). However, ICRP states that the guidance for the protection of future individuals in the case of disposal of long-lived radioactive waste as provided in (ICRP 2000) remains valid. ICRP (2000) recommends, in the context of assessing doses to those most likely to receive the highest doses, that exposures should be assessed on the basis of the mean annual doses received by the *critical group*. The concept adopted by Posiva is based on the guidance in (ICRP 2007a) on how to assess the dose to the 'representative person'; this is considered to be consistent with the (ICRP 2000), since the 'representative person' is considered to be equivalent to an 'average member of the critical group'. The dose assessment applied in the Posiva interim safety case, below referred to as *the dose assessment*, is presented below and described in detail in (Hjerpe et al. 2010).

The dose assessment is a general process (Fig. 1) for using results from radionuclide transport modelling to make dose estimates that are suitable for assessing compliance with annual effective dose constraints. The methodology for assessing doses to humans described here focuses on three blue middle boxes in Fig. 1.

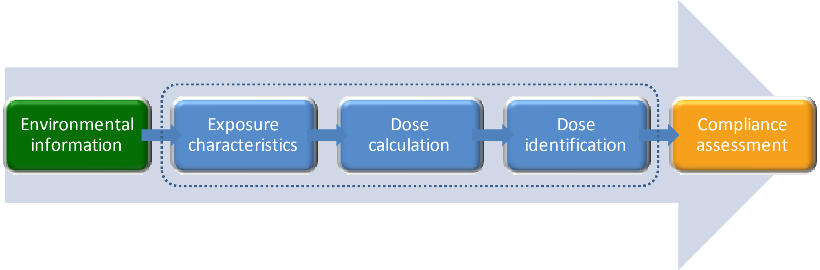

Fig. 1. General process for assessing the doses to humans.

Environmental information

The key environmental information needed for the dose assessments is the time-dependent and radionuclide-specific radioactivity concentrations in environmental media delivered from the radionuclide transport modelling process of the biosphere assessment. This information is given for each biosphere object in the landscape model (the landscape model is the site-specific and time-dependent radionuclide transport model for the surface environment containing connected biosphere objects, where a biosphere object is a sub-area of the whole modelled area described by one, or more, ecosystem types and a data set).

In addition, production rates of individual food products, to derive the total productivity of edibles, and aggregated concentration ratios of radionuclides from environmental media to edible food products, are also needed when calculating food ingestion doses. The productivity of edibles is calculated by summing over all plant parts and animal products normally consumed by man. The aggregated concentration ratios are radionuclide- and ecosystem-specific and derived as the productivity-weighted average of the soil-to-edible concentration ratios for all edibles produced in the ecosystem in question.

Exposure characteristics

A specific set of individual characteristics is selected, including, among others, food and water intake rates. In general, diet, occupancy, and other information needed to estimate exposure can be referred to as ‘habit data’ (ICRP 2007a). When assigning the exposure characteristics, care has been exercised to prevent excessive conservatism. The key features in the selected exposure characteristics, and the selection of habit data, can be summarised as follows:

– the exposure characteristics aim at being reasonable and sustainable, in line with the concept of assessing doses to the representative person (ICRP 2007a),
– all main exposure pathways are considered: ingestion of food, ingestion of water, inhalation and external exposure,
– the number of exposed individuals is limited by the capability of the biosphere objects to produce food and drinking water, by the size of suitable residential areas from a present-day perspective, and by present demography,

– average present-day intake rates are assumed for the representative person,
– cautious assumptions are made for occupancy data (e.g., time spent outdoors),
– a very cautious assumption is made for usage of local resources (i.e., all foodstuffs and water originate from contaminated areas), and
– exposed individuals have no preferences regarding food.

Dose calculation

The dose calculation method applied follows the concept described in (Avila & Bergström 2006), based on values of food energy intake (carbon) and water intake given by the ICRP for Reference Man (ICRP 1975, 2002). The doses arising from intake of contaminated water and food, inhalation of contaminated air, and external exposure from contaminated areas are calculated by using dose coefficients. The dose coefficients used for ingestion and inhalation are based on the values recommended by (ICRP 1996) for adults. The dose coefficients used for external radiation are the values for radionuclides uniformly distributed to an infinite depth and an effectively infinite lateral extent of the contamination in soil from (U.S. EPA 1993). The approach is to calculate *landscape doses*, defined as follows:

– the *landscape dose* (E_{L}) is the pathway-, radionuclide- and biosphere object-specific annual effective dose to an individual.

In the dose calculation, the exposure pathways inhalation and external exposure are combined into the same dose quantity ($E_{\mathrm{L,IE}}$), whereas the dose contributions from food ingestion ($E_{\mathrm{L,F}}$) and water consumption ($E_{\mathrm{L,W}}$) are kept separate. The reason for this is that the exposures from external and inhalation pathways occur simultaneously at the same location, whereas it is not always the case that the consumed food and water originate from the same location. This approach gives the possibility to calculate the dose to an individual who, for example, e.g. lives in a forest, drinks water from a nearby lake and eats food produced on a nearby cropland. For all biosphere objects, landscape doses are calculated for the different pathways as follows:

– *Landscape dose from food ingestion* ($E_{\mathrm{L,F}}$) is the product of the annual intake of a radionuclide from food ingestion and the corresponding dose coefficient for ingestion.
– *Landscape dose from water consumption* ($E_{\mathrm{L,W}}$) is the product of the annual intake of a radionuclide from water consumption and the corresponding dose coefficient for ingestion; $E_{\mathrm{L,W}}$ is calculated for lakes and rivers.
– *Landscape dose from inhalation and external exposure* ($E_{\mathrm{L,IE}}$) is the sum of doses due to inhalation and external exposure from terrestrial objects (forest, wetland and cropland). The contribution from inhalation is the product of the annual intake of a radionuclide via inhalation and the corresponding dose coefficient for inhalation. The contribution from external exposure is the product of the activity concentration in soil and the corresponding dose coefficient for external exposure.

Dose identification

The pathway-specific dose contributions are combined to obtain the "total" dose to each individual in the exposed population. The approach to identifying the dose from all pathways, denoted as the *annual landscape dose*, E_{ALD}, is to identify the dose maxima from each pathway, and sum these. Thus, the dose to the most exposed individual is the sum of the doses from ingestion of the most contaminated food, ingestion of the most contaminated water, and the dose from inhalation and external exposure arising from staying all time in the biosphere object giving the highest dose. When deriving E_{ALD} it is also required that the annual consumption of food and water originates from the exposed area. Thus, the dose may be a sum of contributions from different biosphere objects. For example, if a forest object is producing the most contaminated food, but not enough to satisfy the annual food demand of an individual, it is then assumed that this individual eats everything produced from this forest and the remaining food needed is taken from the biosphere object producing the second most contaminated food.

The approach is to derive E_{ALD} to each exposed individual and the distribution of E_{ALD} between the different individuals of an exposed population is referred to as the *dose distribution*. The highest numbers of people that can satisfy their annual food demand, drinking water demand, and demand regarding residential areas, by utilising contaminated food, water and land are referred to as maximum sustainable populations; these are derived for each exposure pathway. For food ingestion, the carbon content in all contaminated edibles produced during one year divided by the average annual carbon demand by an individual is the maximum sustainable population. For water intake, the population is derived as the available drinking water from all contaminated lakes and rivers divided by the average annual water intake by an individual, taking into account that not all available water is utilised as drinking water. Finally, for inhalation and external exposure, the population is derived by dividing the total suitable residential area by an assumed population density. Here, only forest objects are included and the population density is cautiously selected as the present-day urban density in Helsinki (national maximum).

The derivation of the dose distribution starts by deriving E_{ALD} to the most exposed individual and then subtracting what caused that exposure from the landscape model (one individual's annual demand of food or water, or the size of the residential area). Then, the E_{ALD} to the second most exposed individual is derived, excluding the food, water and residential area the most exposed individual "used" from the landscape. This procedure is performed iteratively until all three exposure pathways have been the dominating pathway E_{ALD} at least once, or until a pre-selected upper limit on the size of the exposed population is reached. This approach to derive the dose distribution ensures that no potentially highly exposed groups are excluded.

The derived dose distribution then forms the basis for identifying annual landscape doses to

– a representative person from the most exposed group, which is the sub-group of the above identified exposed population that receives the highest doses, and
– a representative person for the other people in the exposed population.

For probabilistic assessments, (ICRP 2007a) recommends using the 95th dose percentile as the basis for selection of the most exposed group and that the characteristics of these people need to be explored in the case when relevant dose constraints might be exceeded by a few tens of people or more. Even though the present Posiva dose assessment is deterministic, a similar approach for selecting the most exposed group is applied here since a distribution of doses is derived. The most exposed group is defined as consisting of the smallest of: 1) the 5% most exposed persons in the dose distribution, and 2) a few tens of people, here selected as the 50 most exposed persons. As the final step, the E_{ALD} to two representative persons are identified as follows:

– the annual landscape dose to a representative person within the most exposed group, E_{group}, is the average E_{ALD} for all individuals in the most exposed group.
– the annual landscape dose to a representative person among the other individuals in the exposed population, E_{pop}, is the average E_{ALD} in the exposed population, excluding the most exposed group.

Compliance assessment

In order to assess compliance with the regulatory dose constraints, the whole assessment time window needs to be considered. This is done by identifying, for each generation, the doses to representative persons. Then,

– the highest value, over all generations, of E_{group} is used to determine compliance with the regulatory dose constraints to the most exposed people, and
– the highest value, over all generations, of E_{pop} is used to determine compliance with the regulatory dose constraints to other people.

Results

The methodology described above was applied in the biosphere assessment, summarised in (Hjerpe et al. 2010), in the interim safety case, summarised in (Posiva 2010); this section presents the key results. Dose distributions were derived based on the outcome of the landscape modelling, for which the key input was the derived radionuclide release rates from the geosphere to the biosphere, excluding those radionuclides that were identified from the screening study (Hjerpe et al. 2010) as having insignificant radiological consequences. These release rates are results from nine *repository calculation cases* (RCC), presented in detail in the most recent assessment for a KBS-3V repository (Nykyri et al. 2008). All RCCs addressed have their origin in the same type of repository scenario – assuming releases from a single canister with an initial penetrating defect at the time of emplacement. In addition, each RCC is analysed with three *biosphere calculation cases* (BCC), each using different configurations of the landscape model; the difference is the spatial distribution of the releases from the geosphere, assuming that the releases originates from different repository panels in each BCC. Thus, a total of 27 combinations of calculation cases were analysed. In addition, a number of sensitivity BCCs was analysed to assess the uncertainties introduced in the biosphere assessment process. All results are presented in (Hjerpe et al. 2010); the results presented here are limited to the results from the most realistic cases (the three BCCs in conjunction with the base RCC for the scenario considered) and the cases resulting in the highest doses used in the compliance assessment. The base RCC case, Sh1, assumes (Nykyri et al. 2008) a hole with 1 mm diameter in the copper overpack reaching the insert and the time when water enters a canister through a defect and the dissolution of fuel matrix starts immediately after the emplacement of the canister. The RCC leading to highest doses (see below) are Sh4 Q, in which it is assumed that size of the defect is 4 mm and flow rates are higher for all release routes in the near-field and geosphere. The three BCC are based on the repository panels, denoted as Panels A, B and C, needed for spent fuel from the units currently in operation or under construction, based on the 2006 repository layout, (Kirkkomäki 2007).

The dose distributions for the repository base case Sh1 analysed with the BCC with releases from panel A and for the case Sh4 Q analysed with the BCC with releases from panel are shown in Fig. 2. The annual landscape doses to a representative person for the most exposed group, E_{group}, and to a representative person for other people, E_{pop}, for the RCCs Sh1 and Sh4 Q analysed with the three BCCs are shown in Fig. 3. Table 1 presents the dose maxima, and the year the maxima occur, for the annual landscape doses shown in Fig. 3.

Discussion

The derived dose distributions (Fig. 2) show that the E_{ALD} can be roughly divided into three exposure levels:

1) the E_{ALD} maxima, received only by one or a few individuals. The doses to these most exposed individuals are mainly caused by ingestion of food from forest or wetland objects receiving direct releases from the geosphere (Hjerpe et al. 2010). Since the productivity of edibles is low in these objects, this highest exposure level is only affecting one or a few individuals,
2) an intermediate level, received by some tens to hundreds of individuals. These doses are caused by ingestion of food from croplands, with high productivity of edibles, that are irrigated with contaminated water (Hjerpe et al. 2010), and
3) a lower level, received by some thousands of individuals. The doses to the individuals in this group are totally dominated by ingestion of contaminated water. These individuals do not get any dose from food ingestion, since it is assumed that all contaminated food has been consumed by individuals of the two groups above.

In Table 1, the annual dose maximum to a representative person for the most exposed group, E_{group}, ranges from about 8×10^{-7} to 3×10^{-5} mSv and the dose maximum to a representative person for other people, E_{pop}, ranges from about 2×10^{-8} to 5×10^{-6} mSv.

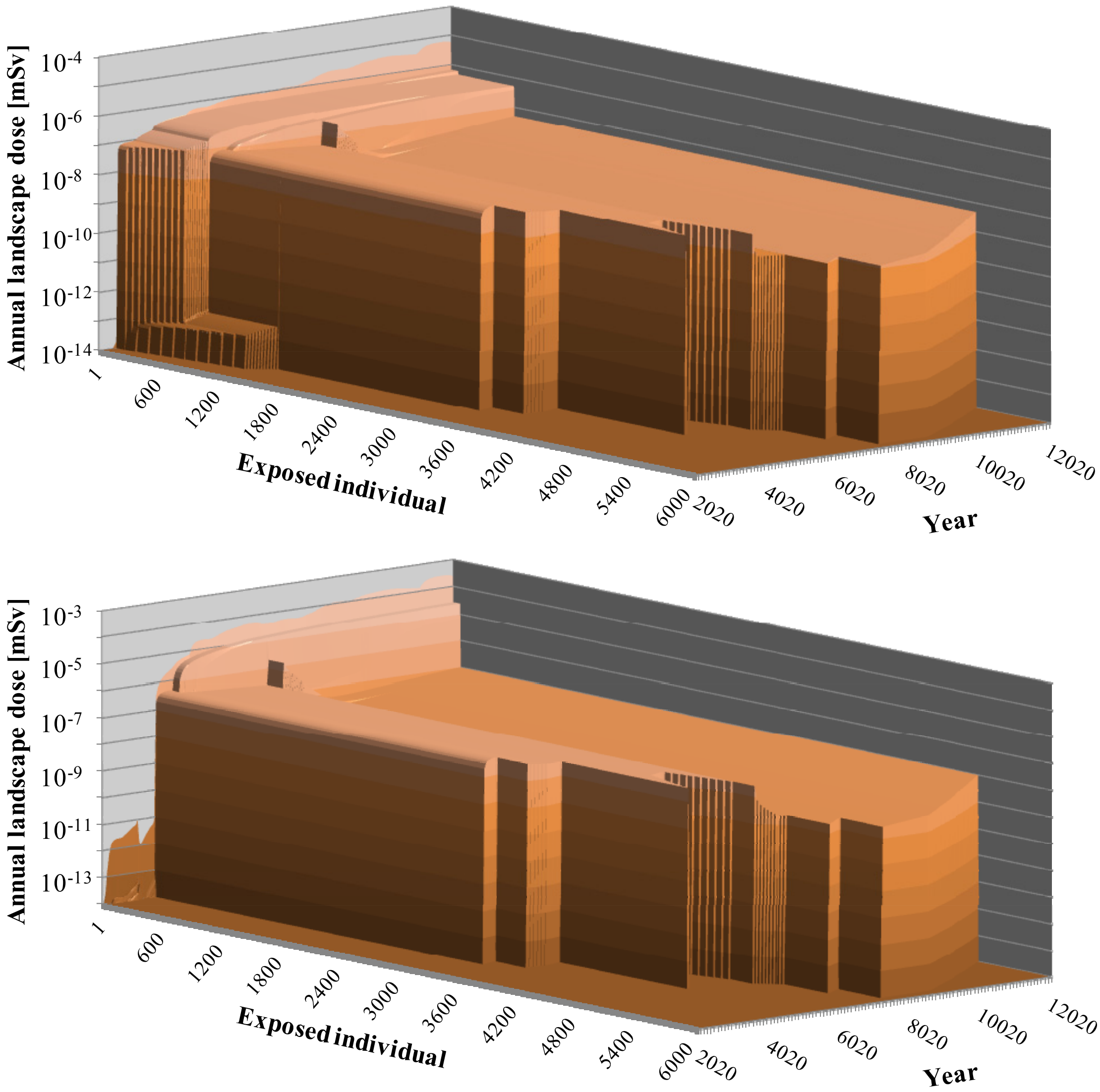

Fig. 2. The dose distributions for the repository base case Sh1 analysed with the BCC with releases from panel A (top) and for the case Sh4 Q analysed with the BCC with releases from panel C (below).

Table 1. Annual landscape dose maxima to a representative person for the most exposed group, (E_{group}) and other people (E_{pop}) for selected RCCs analysed with the BCCs and the year the maxima occur.

RCC	BCC	E_{group} [mSv]	Year	E_{pop} [mSv]	Year
Sh1	Panel A	7.9×10^{-7}	11 920	1.2×10^{-7}	3 970
	Panel B	1.6×10^{-6}	11 870	2.0×10^{-8}	12 020
	Panel C	3.9×10^{-6}	11 870	2.9×10^{-8}	3 570
Sh4 Q	Panel A	1.4×10^{-5}	6 570	4.7×10^{-6}	3 970
	Panel B	1.2×10^{-5}	11 820	2.0×10^{-7}	3 570
	Panel C	3.1×10^{-5}	11 820	4.5×10^{-7}	3 920

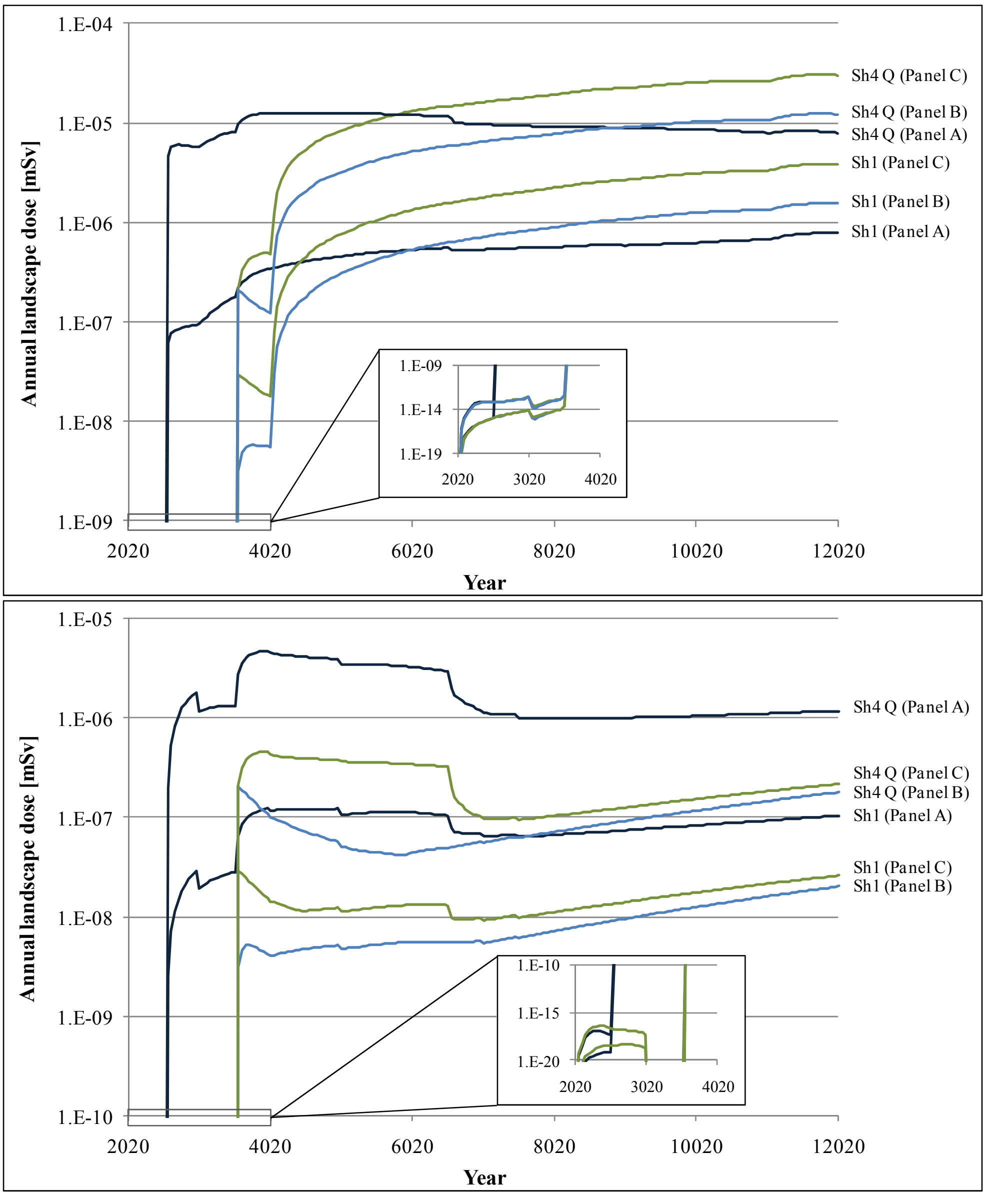

Fig. 3. Annual landscape doses to a representative person for the most exposed group, E_{group}, (top) and to a representative person for other people, E_{pop}, (below) for selected calculation cases.

Conclusions

This paper has presented the methodology for assessing doses to humans adopted in the Posiva safety case. It is considered that this concept is adequate for the purpose of determining compliance with the regulatory dose criteria. Results from applying the dose assessment methodology in the interim safety case have also been presented. The results show that the derived E_{group} maximum is more than three orders of magnitude below the regulatory annual dose constraint for the most exposed people. Furthermore, the results show that the derived E_{pop} maximum is more than two orders of magnitude below the regulatory annual dose constraint.

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Study of the site dose rate of the spent fuel storage facility with add-on shielding

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Abstract
The site dose rate of a spent fuel storage facility is always the focus of public concern in Taiwan. For the strict limitation of the site dose rate committed by the nuclear power plant, an add-on shielding design is needed for the spent fuel storage facility. The purpose of this study is to estimate the site dose rate of the spent fuel storage facility with variant thickness of the add-on shielding design. The simulation model of the storage cask and the add-on shielding was constructed in detail. Neutron and photon sources of the spent fuel in the nuclear power plant was also considered in the simulation. The discrete ordinate code DORT was adopted to calculate the dose rate at the surface of the storage cask, and the SKYSHINE III code was used to estimate the site dose rate according to the fluence results calculated by DORT. Six designed thickness of the add-on shielding were simulated, which include: (1) cask without add-on shielding; (2) 10 cm concrete add-on shielding at side of the cask and 5 cm concrete at top of it; (3) 25 cm concrete at the side of the cask and 20 cm at top of it; (4) 30 cm concrete at the side of the cask and 25 cm concrete at top of it; (5) 35 cm concrete at the side of the cask and 30 cm concrete at top of it; (6) 45 cm concrete at side of the cask and 40 cm concrete at top of it. Besides a single storage cask, a storage facility layout composed of 42 casks was also simulated for the site dose rate estimation. With the simulation results, the tendency of the dose rates at the surface of the cask and those at the distances for different add-on shielding designs were discussed. It is noticed that the dominant contribution of the dose rate at the cask surface comes from the photon source, but in the far distance the dominant dose contribution varies with the add-on shielding thickness. The add-on shielding design provide a simple and practical solution to the spent fuel storage facility for the condition of the nearby site boundary in Taiwan. The simulation results could offer information for the storage facility design in the future.

Introduction
The radiation protection project concerned about the site dose rate of a spent-fuel storage facility is always a major concern for a nuclear power plant. Therefore, the Taipower Company planned to construct an independent spent-fuel storage installation (ISFSI) for the midterm storage of the spent fuel. The strict limitation about the

geographical condition and the site dose rate committed by the nuclear power plant has to be considered in Taiwan. To solve this problem, an add-on shielding design is needed for the spent fuel storage facility. The purpose of this study is to simulate the layout of the storage facility for different shielding designs and skyshine calculations.

Material and methods

For the shielding analysis, a model of an ISFSI storage cask is developed. The ISFSI storage cask includes a canister loaded with 56 spent-fuel assemblies and a vertical cylindrical container. The actual limitation of the site dose rate is 0.05 mSv / yr, which is promised value in the report of environmental impact assessment. Hence, there is additional concrete shielding outside the side and the top of the storage cask to ensure the promised site dose rate. Single cask with different add-on concrete design were simulated, which include: (1) cask without add-on shielding; (2) 10 cm concrete add-on shielding at side of the cask and 5 cm concrete at top of it; (3) 25 cm concrete at the side of the cask and 20 cm at top of it; (4) 30 cm concrete at the side of the cask and 25 cm concrete at top of it; (5) 35 cm concrete at the side of the cask and 30 cm concrete at top of it; (6) 45 cm concrete at side of the cask and 40 cm concrete at top of it. The decay heat calculated by SAS2H/ORIGEN-S simulation code is 9.7 kW per cask. For the site dose rate calculation, the discrete ordinate code DORT was adopted to calculate the dose rate at the surface of the storage cask with the cross-section library CASK-81. The output flux data of the DORT simulation were transferred to equivalent point sources for SKYSHINE III calculation. Figure 1 shows the layout of the ISFSI facility with 13 simulation position, and figure 2 shows the position of the 42 casks in the layout.

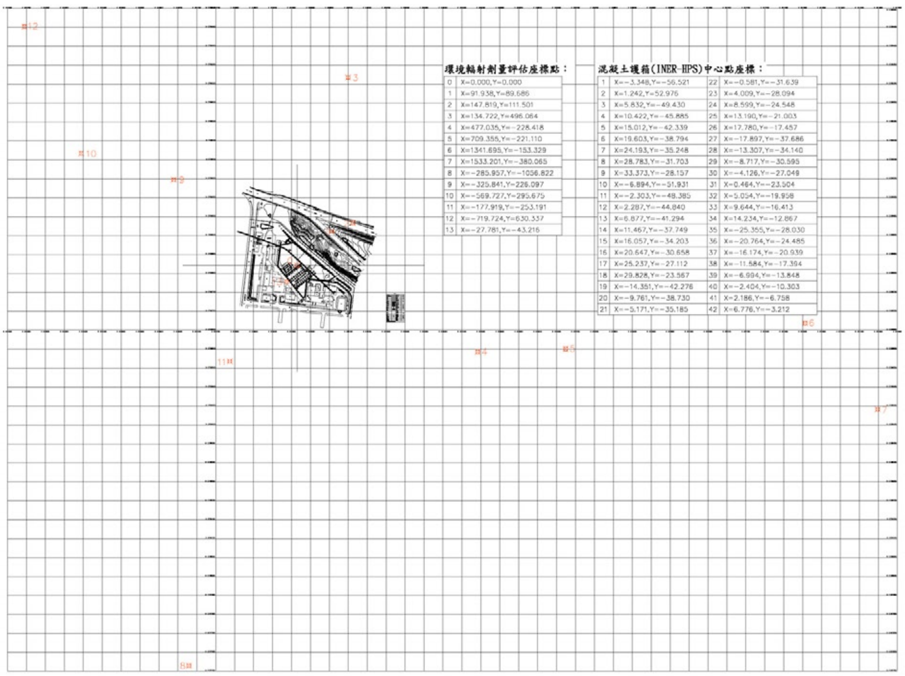

Fig. 1. The layout of the ISFSI facility with 13 simulation position in Taipower Company.

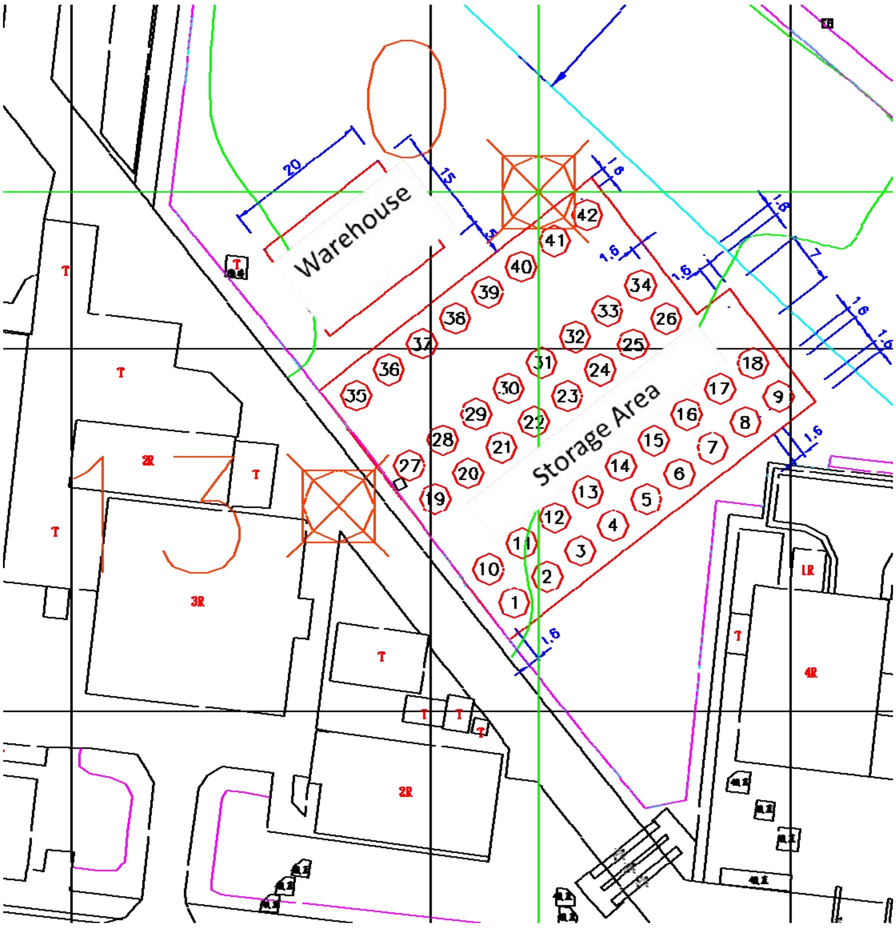

Fig. 2. The position of the ISFSI facility with 42 storage casks.

Results

The maximum surface dose rates at the top and at the surface of the storage cask with six-different shielding thickness design are listed in Table 1. It shows the dose rates contribution from fuel neutron, secondary photon, fuel photon, and structure photon, respectively. On the other hand, a storage facility layout composed of 42 casks was also calculated by SKYSHINE III at various distances. Table 2 shows the side dose rate for 13 estimated positions in Figure 1 with the add-on shielding thickness of 10 cm concrete at side and 5 cm concrete at top. From simulation position 1 to position 12, the side dose rate limitations are complied with the 0.05 mSv/hr. On the other hand, the position 13 is the nearest point from ISFSI facility, and the promised value is 2.5μSv/hr of environmental impact assessment. The total dose rate is 0.166μSv/hr (14.3mSv/yr) in position 13, and it is also complied with the promised value.

Table 1. The maximum surface dose rates calculated by DORT with six-different shielding thickness design.

Additional Concrete Thickness(cm)	Position	Dose rate(mSv/hr)				
		Fuel neutron	Secondary photon	Fuel photon	Structure photon	Total dose
Side:0,Top:0	Side	2.85E-03	6.98E-03	5.57E-02	1.47E-03	6.70E-02
	Top	8.91E-02	1.12E-03	3.10E-03	1.33E-01	2.26E-01
Side:10,Top:5	Side	1.13E-03	3.65E-03	1.58E-02	4.73E-04	2.13E-02
	Top	1.33E-02	2.16E-03	3.39E-03	1.41E-02	2.80E-02
Side:25,Top:20	Side	3.13E-04	1.29E-03	1.91E-03	8.79E-05	3.60E-03
	Top	2.36E-03	1.26E-03	1.22E-04	7.18E-03	1.09E-02
Side:30,Top:25	Side	1.35E-04	7.83E-04	7.85E-04	3.92E-05	1.75E-03
	Top	1.14E-03	1.03E-03	5.57E-05	3.55E-03	5.77E-03
Side:35,Top:30	Side	8.48E-05	3.58E-04	5.94E-04	2.11E-05	1.06E-03
	Top	5.55E-04	7.83E-04	2.63E-05	1.80E-03	3.16E-03
Side:45,Top:40	Side	3.04E-05	1.71E-04	2.27E-04	2.11E-05	4.41E-04
	Top	1.36E-04	4.10E-04	6.04E-06	4.67E-04	1.02E-03

Table 2. The side dose rates with the add-on shielding thickness of 10 cm concrete at side and 5 cm concrete at top.

Position	Side dose (mSv/yr)	Top dose (mSv/yr)	Total dose (mSv/yr)	Design criterion
1	4.13E-02	4.96E-03	4.63E-02	0.05 mSv/vr
2	1.44E-02	2.23E-03	1.66E-02	0.05 mSv/yr
3	7.52E-04	5.92E-05	8.11E-04	0.05 mSv/yr
4	7.92E-04	7.76E-05	8.70E-04	0.05 mSv/yr
5	1.54E-04	1.36E-05	1.68E-04	0.05 mSv/yr
6	3.45E-06	3.04E-07	3.76E-06	0.05 mSv/yr
7	1.12E-06	1.01E-07	1.22E-06	0.05 mSv/yr
8	1.86E-05	1.44E-06	2.00E-05	0.05 mSv/yr
9	1.81E-03	1.80E-04	1.99E-03	0.05 mSv/yr
10	2.30E-04	2.24E-05	2.53E-04	0.05 mSv/yr
11	6.69E-03	7.29E-04	7.41E-03	0.05 mSv/yr
12	2.28E-05	2.43E-06	2.52E-05	0.05 mSv/yr
13	1.35E+01	8.06E-01	1.43E+01	2.5µSv/hr

Conclusions

In this study, the ISFSI facility was modelled and simulated by DORT simulation code for the radiation protection purpose. In order to satisfy the promised side dose rate limitation, the add-on shielding designs were necessary. Six designed thickness of the add-on shielding were simulated in this study. By analysing the dose rate contribution of the storage casks in the facility layout, the add-on shielding thickness of 10 cm concrete at side and 5 cm concrete at top was the best suggestion for the ISFSI design.

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Using equivalent point source to evaluate steel shielding cover for turbine buildings

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Abstract

Skyshine issue is one of major concerns of the public dose in turbine buildings for a BWR nuclear power plant. This paper evaluate the thickness of steel shielding slabs on the top and the side of gas turbines to satisfy the public dose criteria that the power plant committed for an under-construction nuclear power plant in Taiwan.

Included in this work is combination of the QADCG/INER III point kernel radiation transport code and SKYSHINE III procedure. The QADCG/INER III is used to convert volumetric source into equivalent point source. The equivalent point source is applied to SKYSHINE III code to estimate the public dose with the energy spectrum of N-16 under open roof condition. The QADCG/INER III code also used to evaluate the thickness of steel slab between the low pressure gas turbine and the electric generator.

To satisfied the dose criteria, the thickness of steel slabs are 20 cm and 25 cm on the top and the side of the turbine set respectively.

Introduction

For the sakes of routine maintenance and heat transfer, sometimes it will not use fixed material such as concrete to construct the roof of the turbine building in a BWR nuclear power plant. In this case, the skyshine radiation from the top of the turbine building will be the major exposure pathway to the public dose. This project evaluated the thickness of steel slab covered on the top of the turbine sets for an under-construction nuclear power plant in Taiwan to reduce the skyshine effect. The thickness of steel slab located between the low pressure gas turbine and the electric generator was also calculated to avoid direct radiation through the off-site area.

In the aspect of skyshine dose estimation, it was evaluated by the SKYSHINE-III code. This dedicated skyshine code deals with point sources and applies the integral line-beam method for the calculation of skyshine dose. The EPRI NP-243 report provided a procedure to convert the volumetric source into an equivalent point source by using point-kernel code. The turbines and main steam pipes are dealt with independent volumetric sources, and N-16 is the major radionuclide within these equipments. These volumetric sources were converted into equivalent point source via

QADCG/INER III code. Those point sources positions were assumed in the center of the corresponding equipments respectively.

Another issue is to evaluate the thickness of the steel shielding slab between the low pressure turbine and the electric generator. It faced the off-site direction where the public could get close to. The direct radiation is the major concern in this case. The QADCG/INER III point-kernel code was applied to estimate the shielding requirement.

The public dose criteria committed by the nuclear power plant is 0.05mSv/y per unit.

Material and method

There are one high pressure turbine and three low pressure turbine in the turbine building. The main steam pipes concerned in this work are one cross-under pipe and three cross-over pipes connected upon the high and low pressure turbines respectively. Table 1 shows the geometry and activity of those equipments.

Table 1. The source terms geometry and activity.

Equipment	Radius (cm)	Length (cm)	Steam density (g/cm3)	Specific activity (µCi/g)	Total activity (Ci)
HPT	94.5	610	0.0364	72.18	4.496E+01
LPT1	150	636	0.005	75.43	1.696E+01
LPT2	150	636	0.005	75.43	1.696E+01
LPT3	150	636	0.005	75.43	1.696E+01
HPT_X-under pipe	46.66	1520	0.0077	73.38	5.874E+00
LPT1_X-over pipe	46.66	1500	0.0053	73.38	3.990E+00
LPT2_X-over pipe	46.66	1500	0.0053	73.38	3.990E+00
LPT3_X-over pipe	46.66	1500	0.0053	73.38	3.990E+00

The QADCG/INER III is applied to calculate dose values caused by the volumetric source on relatively large distance above turbines in different polar angles while the steel slab is present. In the same detecting points, dose values are calculated for point source of unit activity without steel cover. Both the average dose values are obtained by integrating the dose value for each detector over angle and then dividing by the length of the angle interval. The source strength of equivalent point source is the ratio of average dose of volumetric source to the average dose of unit activity point source. The equivalent point source strength is applied to SKYSHINE III code to estimate the public dose with the energy spectrum of N-16 (6.2MeV) under open roof condition. Figure 1 shows the simplified illustration to convert volumetric source into equivalent point source.

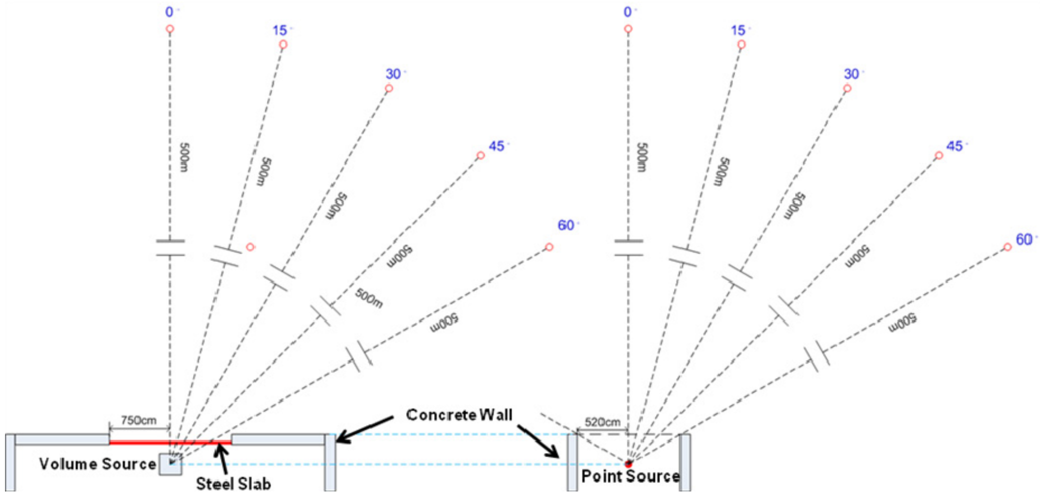

Fig. 1. A simplified illustration to convert volumetric source into equivalent point source.

The ratios of volumetric source to point source with 20 cm steel slab covered on the top and the activities of the equivalent point sources are listed in Table 2. Figure 2 is the schematic diagram applied to the Skyshine III simulation geometry model. On the right side of the Figure 2 is the 3-D model, and the left one is the side view diagram. The low pressure turbine (TLB3) is toward the off-site directory. Figure 3 shows the relative location of turbine set and assessing points in off-site.

Table 2. The volumetric source to point source ratios and the activities of equivalent point source with 20 cm steel slab on the top.

Source terms	Volumetric source to point source ratio	Activities of equivalent point source (Ci)
HPT	5.348E-03	2.405E-01
LPT1	5.568E-03	9.450E-02
LPT2	5.568E-03	9.450E-02
LPT3	5.568E-03	9.450E-02
HPT Xunder Pipes	5.506E-03	3.234E-02
LPT1 Xover Pipes	5.510E-03	2.198E-02
LPT2 Xover Pipes	5.510E-03	2.198E-02
LPT3 Xover Pipes	5.510E-03	2.198E-02

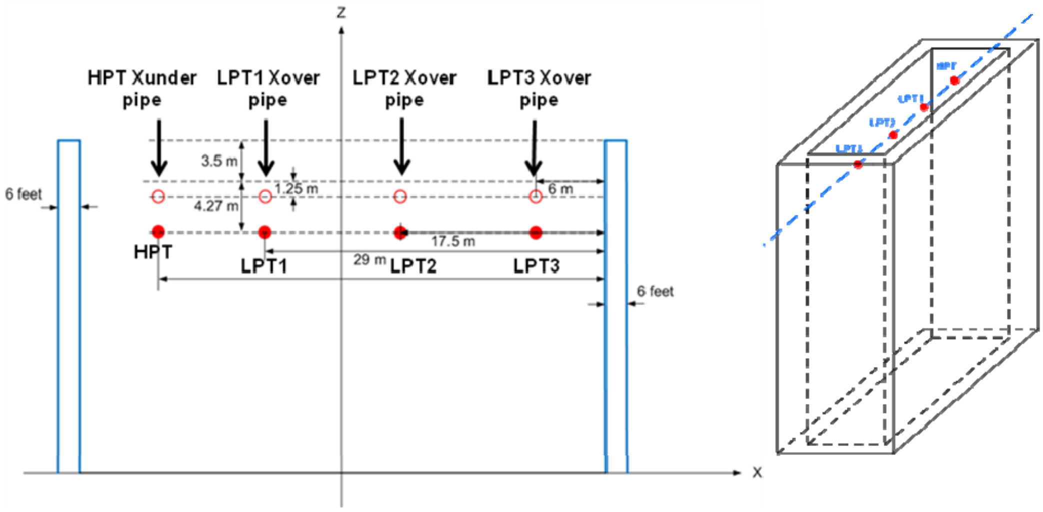

Fig. 2. The geometry model applied to Skyshine III simulation.

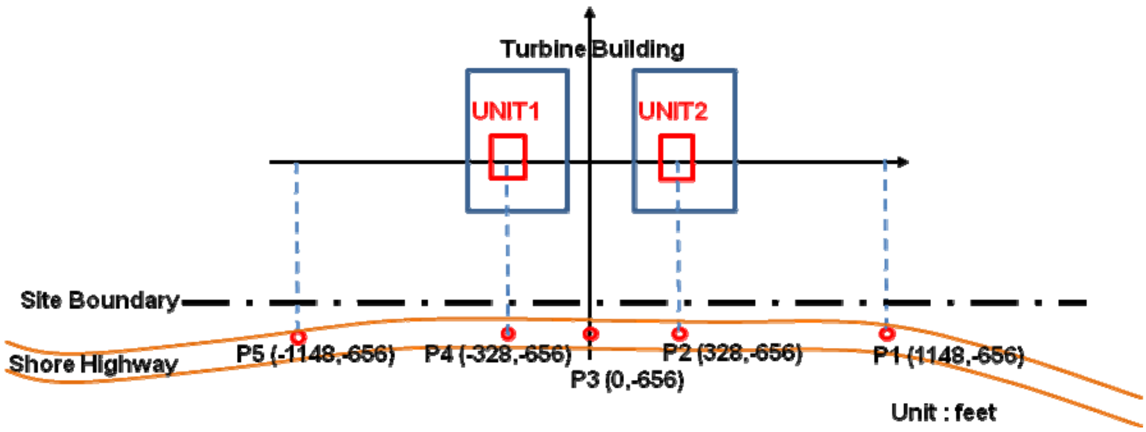

Fig. 3. The relative location of turbine set and assessing points.

Owing to the self-shielding effect and the shielded by the electric generator, the direct radiation from turbines is negligible. The considered source terms for direct radiation evaluation are the main steam pipes upon the turbines. Figure 4 illustrates the side view and top view of the geometry model applied to the QADCG/INER III simulation. Both the turbine set and the electric generator are simplified to homogeneous rectangular solids with density of 2.62 g/cm3 steel material.

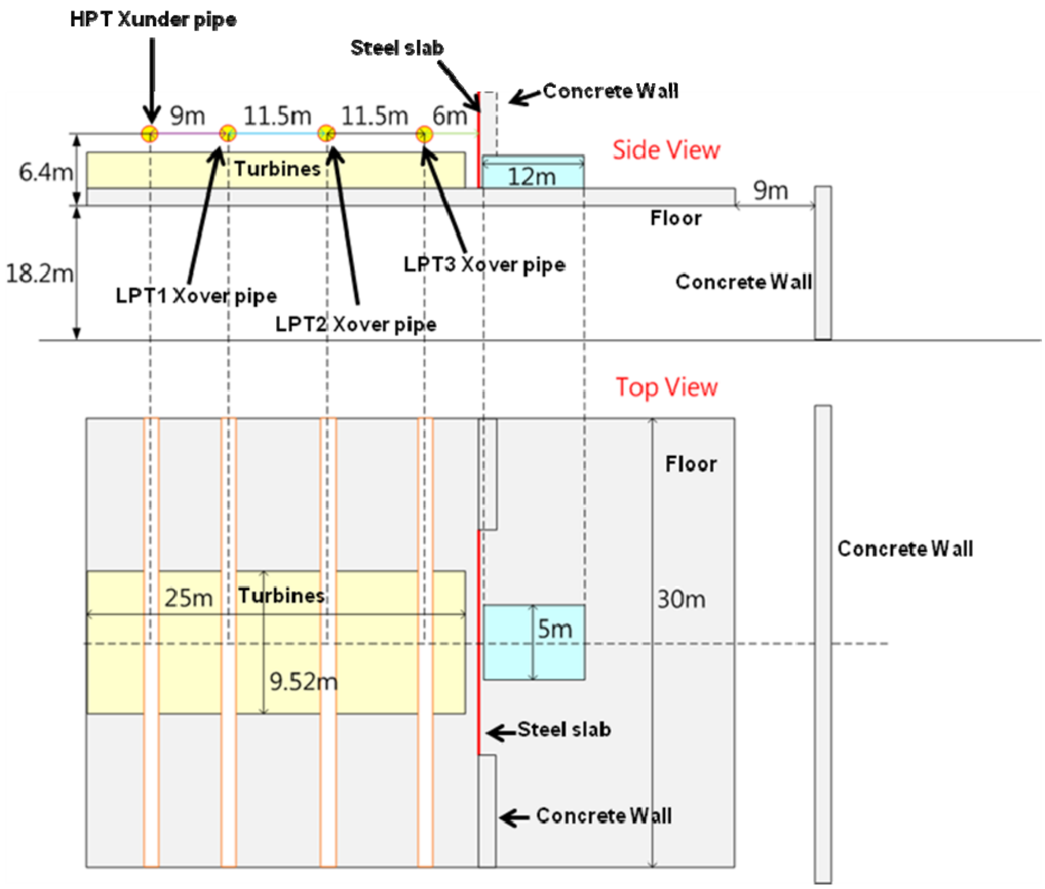

Fig. 4. The geometry model for QADCG/INER III simulation.

Results and discussion

Unit 1 and unit 2 are geometric symmetry to the five off-site assessing points for skyshine dose evaluation. Unit 2 was applied to simulate the skyshine dose for the public in this work. Table 3 is the skyshine dose rate estimating results for unit 2 with 20 cm shielding steel slab on the top. It includes the dose contribution caused by turbines and the main steam pipes. The total shyshine dose of all source terms lists at the bottom of the table. The maximum value 0.0237 mSv/y lies in the assessing point 2.

Table 3. The skyshine dose rates of off-site assessing points with 20 cm steel slab on the top (mSv/y).

Equipment	1	2	3	4	5
HPT	4.640E-03	9.724E-03	4.337E-03	1.348E-03	3.782E-04
LPT1	1.969E-03	4.017E-03	1.821E-03	5.762E-04	1.606E-04
LPT2	1.851E-03	3.196E-03	1.657E-03	5.429E-04	1.544E-04
LPT3	1.363E-03	1.904E-03	1.047E-03	3.691E-04	1.280E-04
HPT_Xunder pipe	9.091E-04	1.813E-03	8.198E-04	2.654E-04	7.843E-05
LPT1_Xover pipe	6.264E-04	1.215E-03	5.501E-04	1.823E-04	5.543E-05
LPT2_Xover pipe	6.214E-04	1.128E-03	5.704E-04	1.950E-04	5.852E-05
LPT3_Xover pipe	5.097E-04	7.086E-04	4.087E-04	1.507E-04	4.948E-05
Total equipments	1.249E-02	2.371E-02	1.121E-02	3.630E-03	1.063E-03

The thickness of steel slab between low pressure turbine and the electric generator is 25 cm. The direct dose rates of all main steam pipes in different distance along the centerline of the turbine set shows in the second column of Table 4. The distance of 150 m indicates location on the shoreline highway in off-site. The third column is skyshine dose rates of turbines and main steam pipes with 20 cm steel slab on the top in the same distance. The total dose to the public including skyshine and direction radiation lists in the fourth column.

In the distance less the 180 m, the direct radiation will be shielded by the electric generator, floor and the building wall as illustrated in Figure 5. The dose rate is quite small in the near side. The maximum direct dose and the total dose are occurred in about 220 m, and the dose decreases with increasing distance farther than that. Considering both the dose contributions of skyshine and direct radiation, the maximum total dose rate 0.0489 mSv/yr are not appeared in the nearest distance from site boundary.

To satisfy the requirement of dose criteria (0.05 mSv/yr per site) that committed by the nuclear power plant, the thickness of steel slab covered on the top on the turbine set needs 20 cm. The steel slab located between the low pressure turbine and the electric generator needs 25 cm in thickness.

Table 4. The direct radiation dose rate with 25 cm side steel slab, skyshine dose rate with 20 cm top steel slab and the total public dose rate.

ALL PIPES	Side steel slab thickness 25 cm	Top steel slab thickness 20 cm	Total Dose (mSv/yr)
Distance	Direct Dose (mSv/yr)	Skyshine (mSv/yr)	
150 m	4.736E-07	2.371E-02	2.371E-02
160 m	4.202E-07	2.137E-02	2.137E-02
170 m	1.670E-05	1.950E-02	1.951E-02
180 m	1.522E-02	1.776E-02	3.298E-02
190 m	2.925E-02	1.618E-02	4.543E-02
200 m	2.805E-02	1.485E-02	4.290E-02
210 m	3.329E-02	1.358E-02	4.687E-02
220 m	3.640E-02	1.248E-02	4.887E-02
230 m	3.612E-02	1.146E-02	4.759E-02
240 m	3.542E-02	1.055E-02	4.598E-02
250 m	3.555E-02	9.712E-03	4.526E-02
260 m	3.499E-02	8.965E-03	4.396E-02
270 m	3.416E-02	8.286E-03	4.245E-02
280 m	3.296E-02	7.666E-03	4.063E-02
290 m	3.151E-02	7.073E-03	3.859E-02
300 m	2.913E-02	6.578E-03	3.570E-02
320 m	2.494E-02	5.725E-03	3.066E-02
340 m	2.132E-02	5.052E-03	2.637E-02
360 m	1.824E-02	4.387E-03	2.262E-02
380 m	1.579E-02	3.846E-03	1.964E-02
400 m	1.374E-02	3.414E-03	1.716E-02

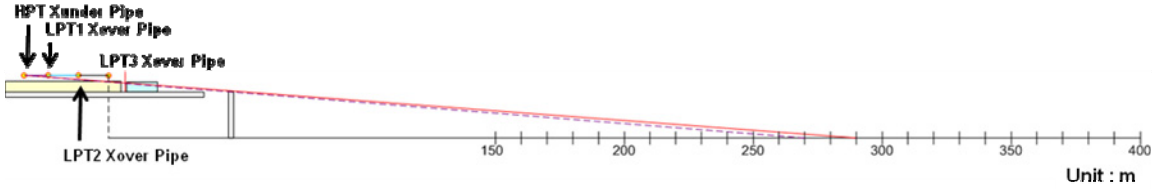

Fig. 5. The illustration of main stem pipes and off-site assessing points in different distance.

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Shielding analysis for Proton Therapy Center in Prague, Czech Republic

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Abstract
The study of radiation fields around the cyclotron room has been made for the proton therapy center in Prague. The proton cyclotron of energy 230 MeV will be installed there as well as additional components located along the beampath. The ambient doses have been computed by taking into account 3 major neutron and photon sources - cyclotron, energy degrader and its collimator.

The fluxes of neutrons/photons produced by protons have been computed using Monte Carlo code MCNPX^{TM} 2.5.0. Energy and angular distributions have been computed for various targets and proton beam energies. Different physical models of high-energy proton interactions have been studied too. The annual charge at the cyclotron exit has been based on "patient model" (distribution of treatment type) established for similar therapy center in Essen (Germany). The cyclotron extraction efficiency of 50% has been assumed; the beam losses inside the cyclotron amount to 82200 nA.h/year. By convention, it has been considered no beam losses in the degrader, i.e. all the losses of degrader and collimator have been attributed to the absorption in the collimator.

As the cyclotron room is located underground, the outside concrete walls of the cyclotron room (West and South walls) have reduced thickness followed by soil. The annual doses have also been computed inside the fixed beam room (towards East), at the maze exit (towards North), inside the main control room (North, next to the maze), and above the cyclotron room roof. Calculated 3D dose distributions have been used to verify the radiation shielding design and the radiation protection optimalization.

Introduction
Proton therapy is becoming an increasingly popular modality in radiation therapy, most often for the treatment of malignant and non-malignant tumours. Known advantages of proton therapy is its ability to deliver radiation dose with high precision to the tumour while surrounding healthy tissue and organs are irradiated minimally. The cost of the building can be over 100 million EUR and the shielding can have a significant impact on the cost, on the other hand must be designed to adequately protect personnel

and the public. The primary purpose of the study is to design a shield that will be in agreement with the current legislation requirements.

The design of the Prague Proton Therapy Center is similar to the existing facility in Essen, Germany. Thus, some calculations in this article are also independent assessment of proven shielding design.

In this work, the statistical approach has been used for verification of the shielding design. The fluxes of secondary radiation (neutrons and photons) produced by protons have been computed using Monte Carlo code MCNPX^{TM}, version 2.5.0. The ambient dose equivalents at selected critical points around the shielding structure have been calculated using fluence-to-dose conversion factors from the International Commission on Radiological Protection (ICRP) Report No. 74. Influence of the high-energy nuclear interaction model choice has been tested as well.

Material and methods

Shielding design aims

The primary design aims are to limit annual dose equivalent behind the shielding walls to <1mSv (outside of proton therapy facility) and <20mSv (inside of the proton therapy facility, i.e. in control rooms, in corridors, etc), respectively. Dose equivalents have been estimated assuming a 16 hours daily shift, 6 working days per week; all year round (Stichelbaut, 2008). All calculations have been also based on technology contractor information as well as model of patient and distribution of treatment type according to the intended operation of the Therapy Center (Stichelbaut, 2007; Stichelbaut, 2008).

Main sources of radiation and spectra of secondary radiation around them

This work is aimed to verified shielding barriers around the room with accelerator – cyclotron - accelerating protons to the energy of 230MeV. Thus, the main sources in the cyclotron room are cyclotron and energy selection system. The energy selection system is composed of two main components - degrader and its collimator. The degrader is made of graphite (density 1.8 g/cm^3) and is designed to slowing down the proton energy to energies that are indicated for the treatment (190MeV, 150MeV, 110MeV, and 70MeV). The collimator is made of tantalum (density 16.6 g/cm^3) and is used for narrowing the scattered proton beam after the degrader (Stichelbaut, 2007; Stichelbaut, 2008).

The cyclotron - The cyclotron extraction efficiency is assumed to ~50%, i.e. ~50% of the beam is lost in the cyclotron (Stichelbaut, 2007). All beam losses in cyclotron have been assumed at the extraction radius, i.e. at the full energy 230MeV, where the proton beam interacts with a copper material. The thickness of a copper layer (in all directions) in simulations is more than range of protons of energy 230MeV in a given material/copper. Cyclotron itself is shielded in all directions at least of 40 cm of iron (density 7.86 g/cm3), i.e. a iron layer of 40 cm thickness has been considered in simulations as well.

The graphite degrader - Secondary radiation spectra have been calculated for the narrow proton beam of energy 230MeV incident perpendicular the middle of the base of the graphite cylinder. Dimensions of these targets/cylinders are shown in Table 1. In all cases the radius of cylinder is greater than range of protons of energy 230MeV and the length of the cylinder is designed so that the proton beam energy at the exit of the cylinder is equal to the desired energy (used for treatment).

Table 1. Parameters of the cylinder target used in simulation instead of graphite degrader.

Energy of protons at the end of the target/cylinder	190 MeV	150 MeV	110 MeV	70 MeV
cylinder radius [cm]	22,385 cm	22,385 cm	22,385 cm	22,385 cm
cylinder lenght [cm]	5,686 cm	10,634 cm	14,739 cm	17,865 cm

The collimator - Dimensions of target/cylinder substituting for the purposes of simulation the collimator are shown in Table 2. In this case radius and length of the target/cylinder are equal to each other and both dimensions are greater than range of proton incident a tantalum target/cylinder. Also in case of collimator, the proton beam has been considered as narrow and as a perpendicular incident a base of the target/cylinder.

Table 2. Parameters of the cylinder target used in simulation instead of tantalum collimator.

Energy of protons entranced the target/cylinder	70 MeV	110 MeV	150 MeV	190 MeV	230 MeV
cylinder radius [cm]	0,5599 cm	1,2122 cm	2,0515 cm	3,0503 cm	4,1877 cm
cylinder lenght [cm]	0,5599 cm	1,2122 cm	2,0515 cm	3,0503 cm	4,1877 cm

In simulations, the geometric center of a cylinder/target has been places to the origin of coordinate system. A cylinder axis coincides with the x-axis as well as proton beam incident the target/cylinder in the middle of cylinder base. The cylinders were surrounded by air. At the sphere of radius of 1 meter have been registered secondary particles emitted from the target. As the main components of the secondary radiation have been identified neutrons and photons. In Figures 1 and 2 there are shown multiplicities of particles emitted from targets. Examples of some energy and angular spectra of neutrons from various sources of radiation are shown in Figures 3 - 5. Generally, neutron spectra cover the entire energy interval from zero energy up to energy of proton impinging on a target. As can be seen in Figures 6 and 7, energy spectra of photons are not so wide – spectra finished, maximally, at several tens of MeV. Also multiplicity of photons are lower than neutron ones. All simulations there were three different interaction models of high energy particles included in MCNP 2.5.0 – Bertini, CEM2K and INCL4 (Pelowitz , 2005).

In all other (not shown in this paper) cases (different energy of incident protons) have had energy and angular spectrum a similar character, for both neutrons and photons as well. Energy and angular spectra suggest that selection of an appropriate model for shielding calculation/verification is not so easy with these results. Use of any of nuclear interaction model has its pros and cons. One leads to a greater multiplicity of secondary particles in the entire energy spectrum, while another just only at high energies (hundreds of MeV). Thus, all mentioned nuclear interaction models have been tested and compared for shielding calculations/verifications as well.

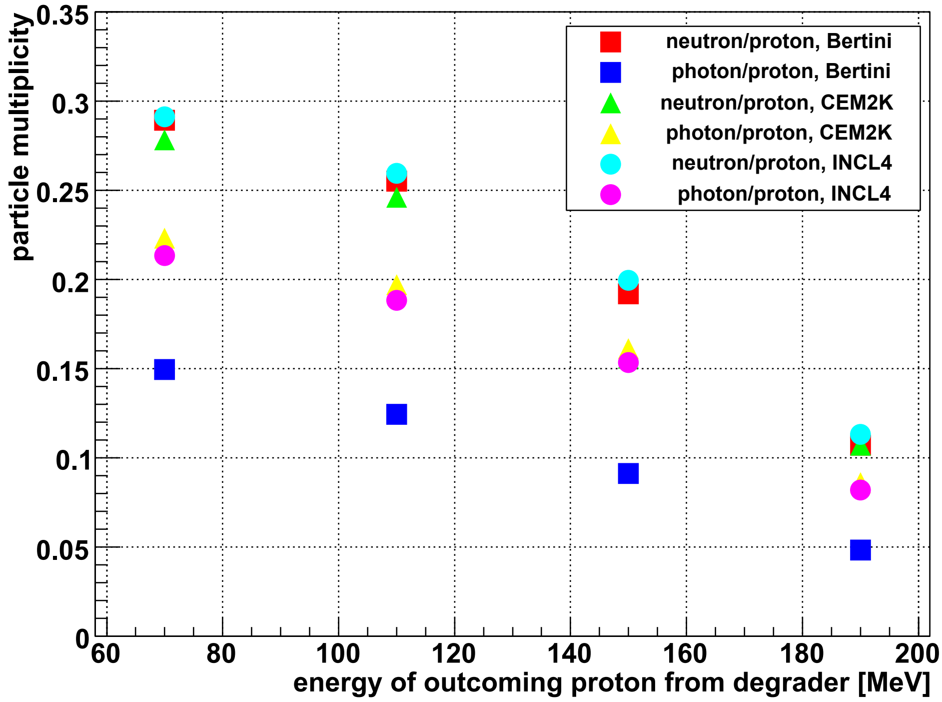

Figure 1. Multiplicities of particle emitted from cylindrical target made of graphite (degrader).

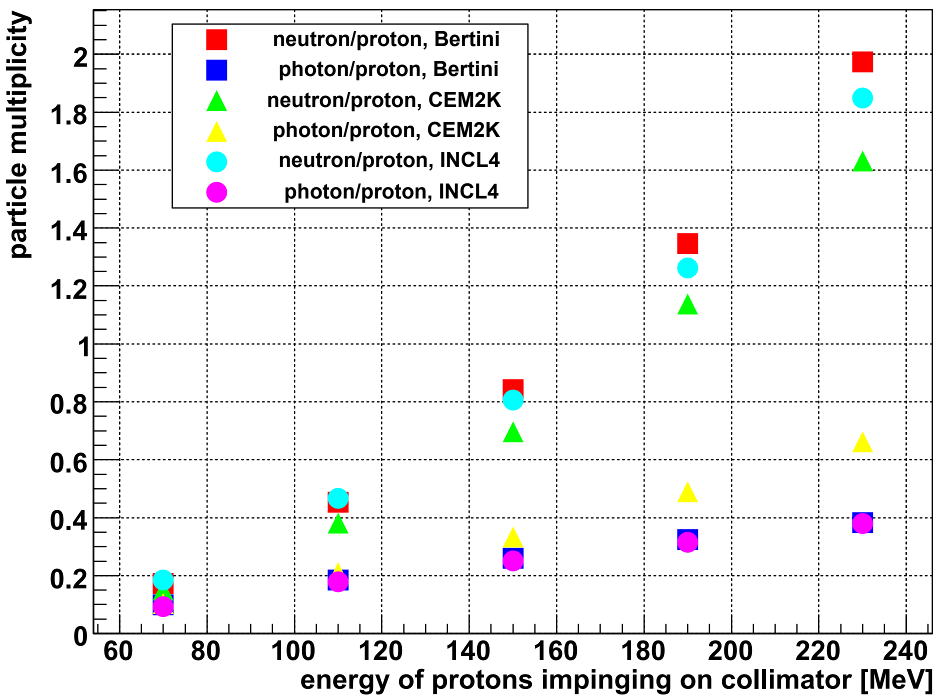

Figure 2. Multiplicities of particle emitted from cylindrical target made of tantalum (collimator).

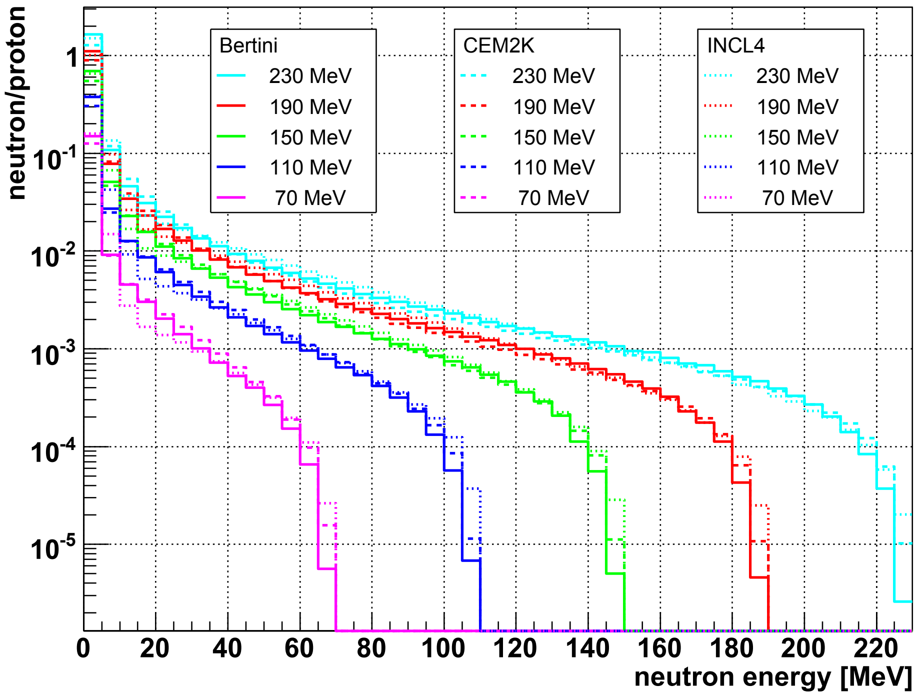

Figure 3. Examples of energy spectra of neutrons emitted from tantalum target (collimator) for various nuclear interactions calculation models.

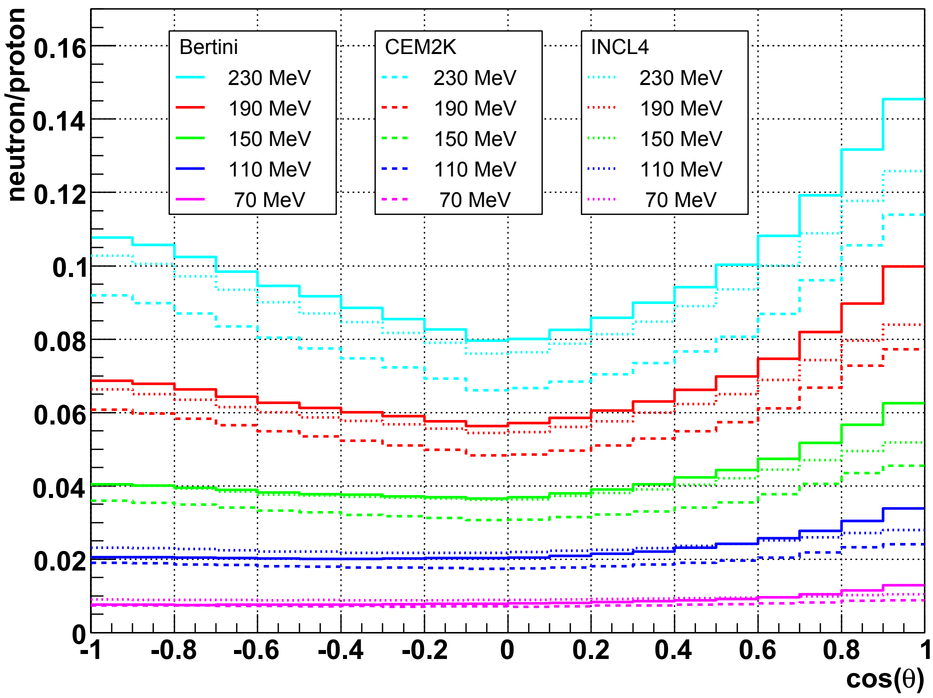

Figure 4. Examples of angular spectra of neutrons emitted from tantalum target (collimator) for various nuclear interactions calculation models.

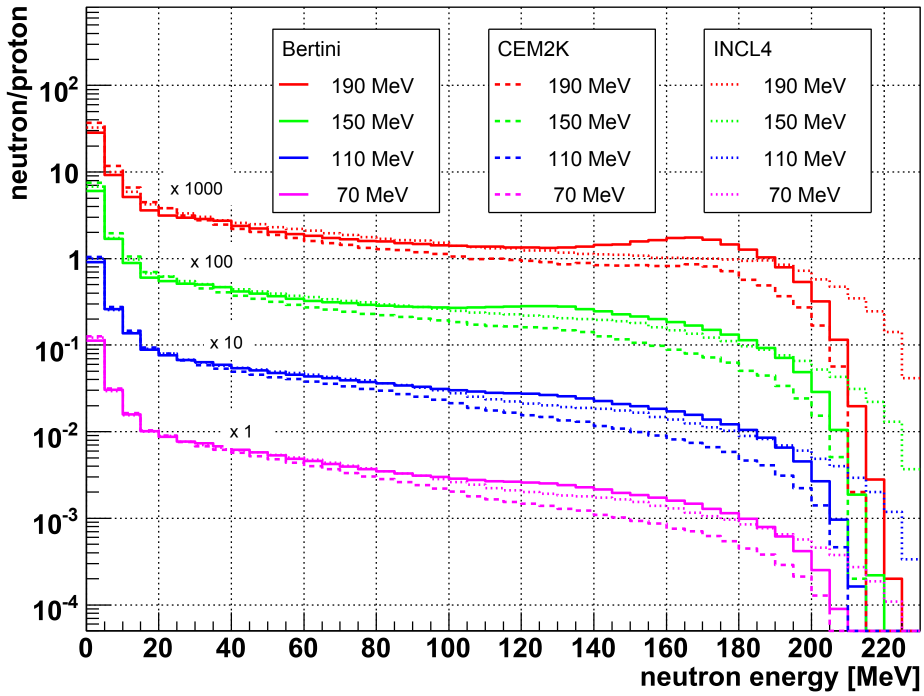

Figure 5. Examples of energy spectra of neutrons emitted from graphite target (degrader) for various nuclear interactions calculation models.

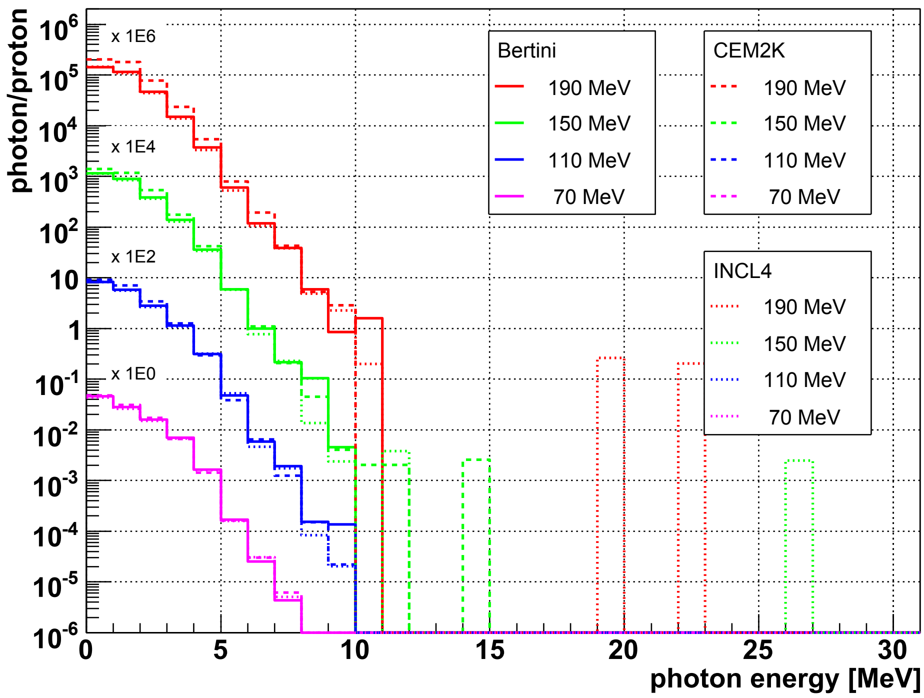

Figure 6. Examples of energy spectra of photons emitted from tantalum target (collimator) for various nuclear interactions calculation models.

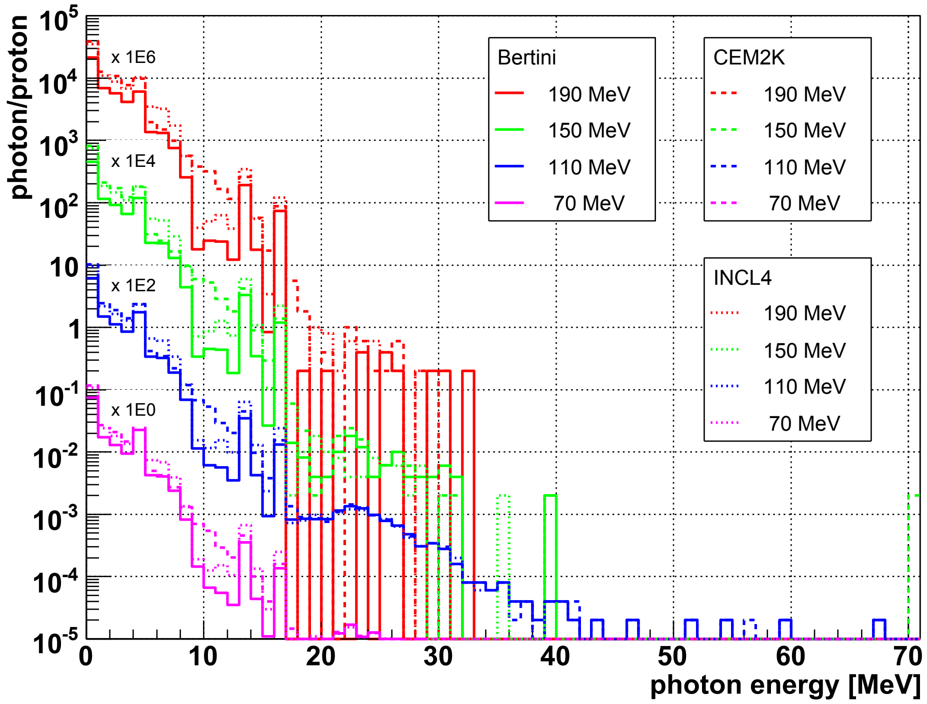

Figure 7. Examples of energy spectra of photons emitted from graphite target (degrader) for various nuclear interactions calculation models.

Shielding calculations

Figures 9 and 10 show the 2D distributions of the calculated values of ambient dose equivalent H*(10) [mSv/year] around the room (see 3D drawing of the object Proton Therapy Center in Prague) with cyclotron, etc. Cross-sections are located in a vertical plane perpendicular to the walls. In the calculations has been included only neutron component of the secondary radiation. Photon component gives only approximately 1 percent of contribution to the ambient H*(10). For evaluation of the dose equivalent have been used fluence-to-dose conversions factors for neutrons from ICRP-74. For results has also been taken into account an expected annual capacity of the Proton Therapy Center due to the number and types of therapeutic interventions (Stichelbaut, 2008).

Three main sources of radiation have been taken into account for shielding verification – cyclotron, degrader and its collimator. These components and their simplification for simulation have been described above. Accelerator efficiency is considered to 50 percent. Conservative approach where beam losses are uniformly disturbed around the cyclotron at extraction radius has been used. Measured transmission factors for degrader and collimator have been used for calculation. By convention, it has been considered that there are no beam looses in the degrader (it means transmission of the degrader is equal to 100 percent) and all beam losses of the degrader and collimator have been attributed to collimator transmission. This transmission depends on the energy of proton beam after degrader as well as on the beam delivery mode during treatment. In the Proton Therapy Center will be for treatments used both active scanning and passive scattering. According to the information from contractor, use of collimator transmission factors for active scanning is more conservative approach (Stichelbaut, 2007; Stichelbaut, 2008).

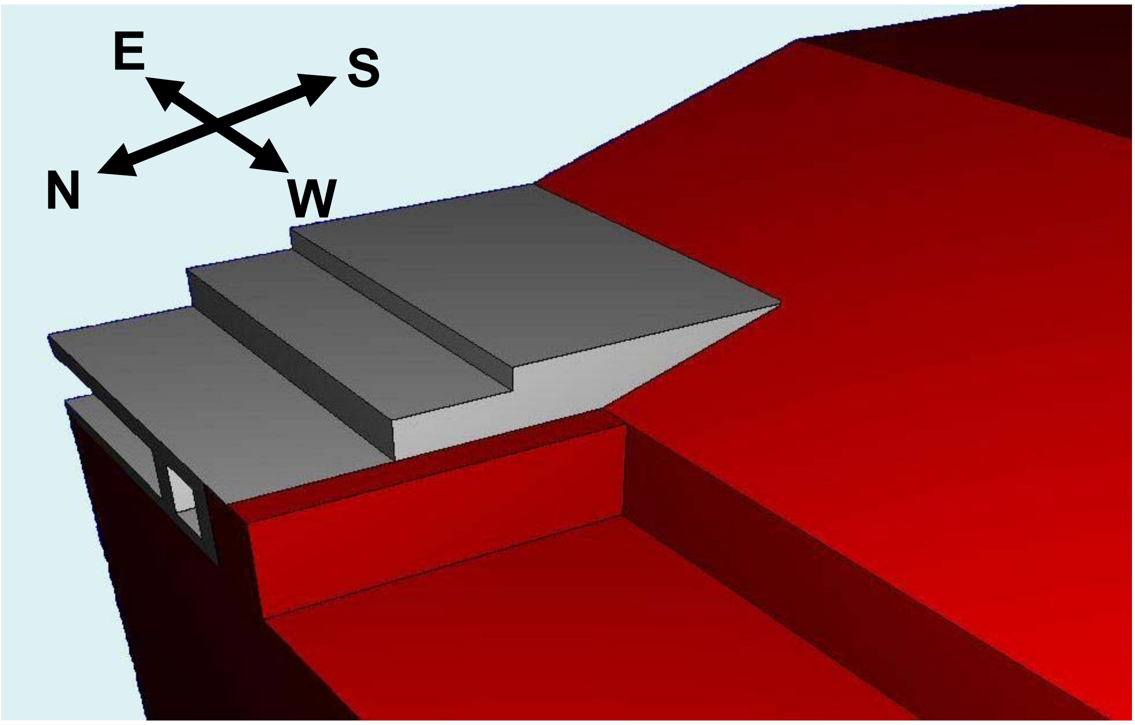

Figure 8. 3D model of the room with a cyclotron and its accessories (grey = concrete, red = soil).

As was mentioned above, for verification of shielding have been used all three models of interaction of high energy particles in the MCNP - Bertini, CEM2K and INCL4 – (Pelowitz, 2005). All models give comparable/similar results with any major deviations from the relevant radiation protection with an impact on legislative requirements compliance.

Discussion

Shielding barriers (concrete and eventually soil) of the cyclotron room in the Proton Therapy Center in Prague are according to the simulation sufficient. There are also some border-line cases in a few localized areas, for example terraces above the entrance maze or place behind the east wall. In these critical areas will be no permanent presence of persons, but it is recommended that they should be properly labelled.

The area directly above the cyclotron and the energy selection system (degrader and collimator) requires also special attention. The area above cyclotron is constructed as a removable cyclotron hatch which must be provided for lowering the accelerator into the building. That means there is not concrete plate, but only some concrete blocks. So, it should be paid attention when imposing these blocks and try to compose these blocks without large air gaps and/or to fill gaps with another shielding material. The area directly above degrader and collimator is also problematic. To achieve the annual dose equivalent <1mSv/year a wall of thickness 4.20m of concrete is needed and space available in this area is only 2.20m. To increase the ceiling it should be possible to use iron or steel material as a shielding. The NCRP Report No. 144 (2003) recommends that any iron or steel shielding should be backed by at least 0.61 m of concrete (since the cross section for the interaction of neutrons with iron falls rapidly below about 5MeV).

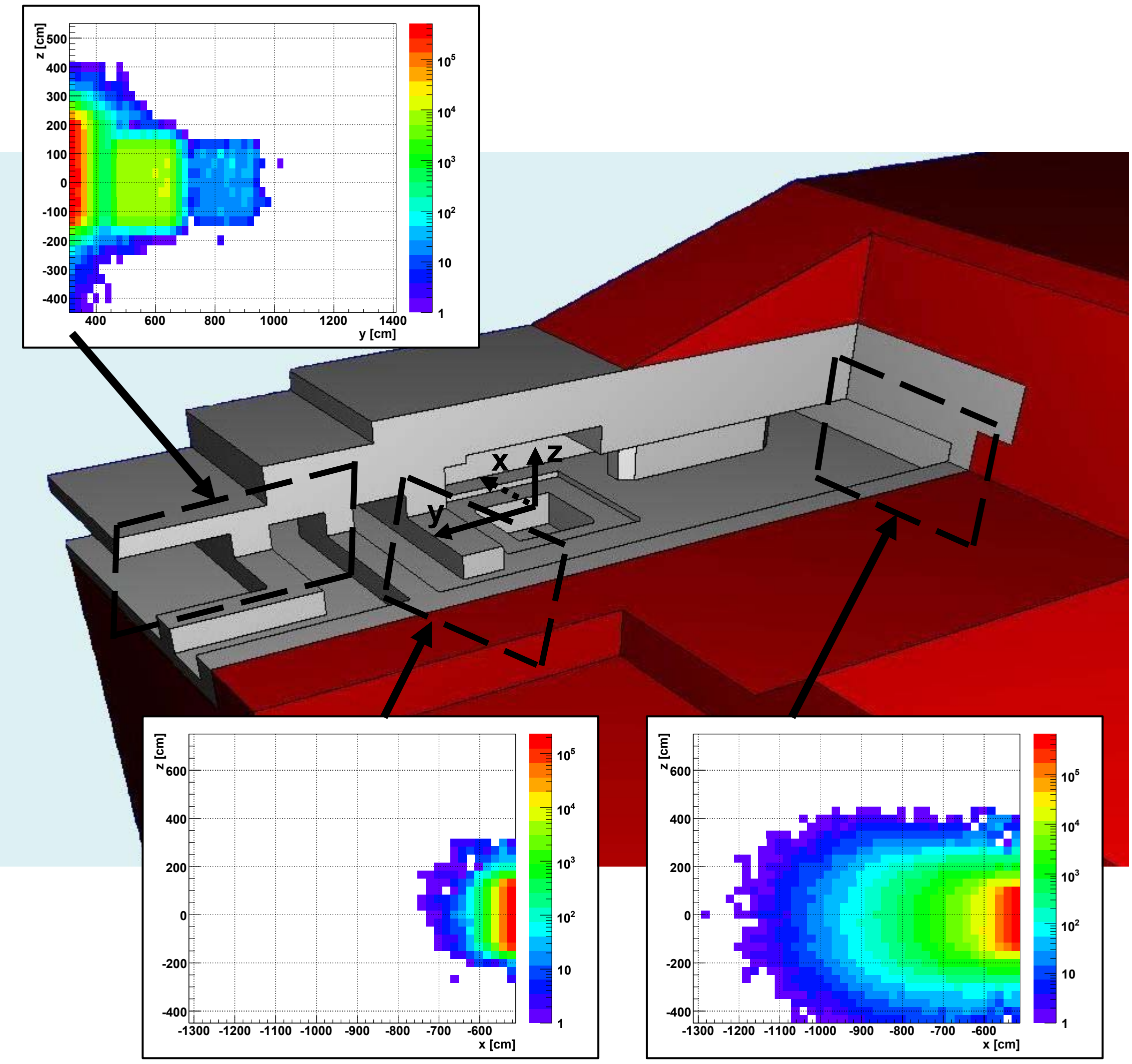

Figure 9. H*(10) [mSv/year] in selected areas of west wall and maze of the cyclotron room.

Conclusions

Shielding barriers of the room with cyclotron and energy selection system have been verified. Thicknesses of the concrete wall proposed in the project are sufficient and meet the requirements of current legislation in the Czech Republic in almost all areas. The problematic areas are described above, in section Discussion.

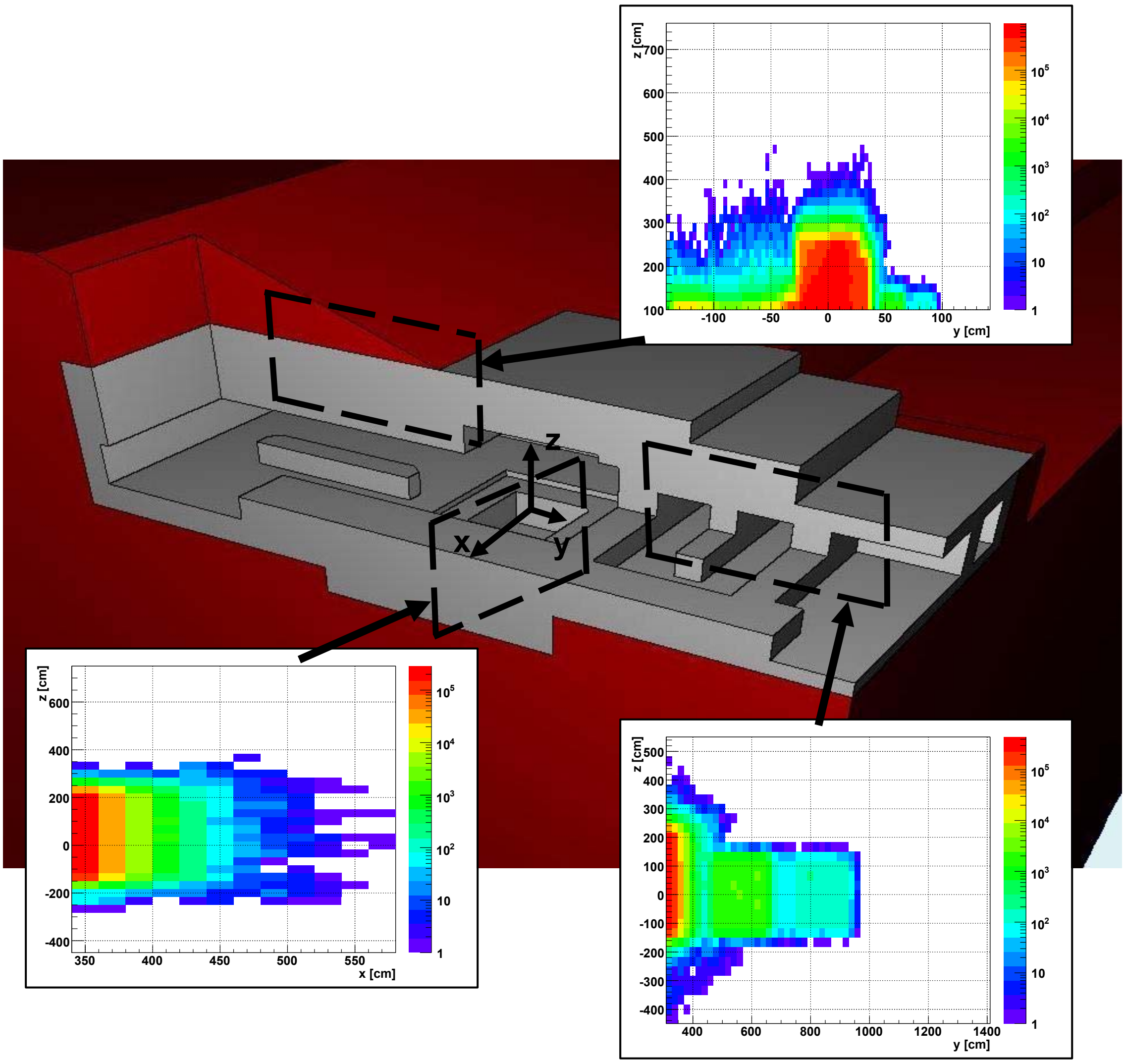

Figure 10. H*(10) [mSv/year] in selected areas of east wall, maze and ceiling of the cyclotron room.

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Ambient radiation monitoring in a corridor configuration

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Abstract

Real-time monitoring used of high sensitivity detectors to measure radiation levels in public access area at the Nuclear Medicine Department (NMD) is the prime interest of this investigation. The results of this study are used to verify and to improve the facility layout, patient routings and administrative control measures. The NMD outpatients, with an initial dose of up to 740 MBq (20 mCi) per case, may wait around and incidentally congest in one place that could cause an unexpected higher exposure level in public access areas. In this surveillance study, the ambient radiation time-profile and peak dose rates have been characterized for the transit hallway or corridor where non-NMD medical staff or patients' relatives may present with the source-bearing patients. In order not to interfere with daily NMD operations, a high sensitivity, collimated NaI(Tl) gamma spectroscopy system is deployed at the far end of the hallway to monitor the ambient radiation levels. The time-profile of the radiation levels were derived from a video-taped gamma spectroscopy using time-stripping, consecutive unfolding method. Another digital G-M dosimeter, which has peak rate locked-in capability, is also adopted for tracking the highest ambient radiation dose rate in this transit hallway to alert the medical staff whenever unexpected NMD patients are crowded at one place. Typical 10-min average count rates, based on a 12-hour observing time from 7am to 7pm, are evaluated from this remote collimated detector. From these 10-min average count rate data, we found the ambient radiation levels at some time intervals can reach up to more than 35 times of the natural background level. Also, the peak-rate dosimeter at this corridor has once registered to a peak value of 40.4 micro-sievert/hr. Based on these radiation monitoring data, suggestions to improve the NMD daily operation and the needed administrative changes have been reviewed and accepted by the management of the hospital. Some of these recommendations had been implemented, including waiting time reduced for the patients, less chairs and more satellite waiting areas and permanent survey meters installed with preset light/audio alarm features.

Introduction

Ambient radiation levels in a corridor patients waiting area have been greatly reduced after being remodelled, of our Nuclear Medicine Department (NMD) based on the ALARA consideration. The NMD outpatients, with an initial dose of up to 740 MBq (20 mCi) per case, may wait around and incidentally congest in one place that could cause an unexpected higher exposure level in public access areas. Completed ambient radiation monitoring of our NMD before remodelling had been characterized and published earlier by the same authors elsewhere. (Lai 2001; Lai el al.2003) In this new surveillance study after remodelling, the ambient radiation time-profile has been re-evaluated by using a video-assisted, high sensitivity collimated NaI(Tl) detector system. The peak dose rates in hourly basis and the accumulated daily dose are also demonstrated by real-time, hand-held digital survey dosimeters.

Material and methods

The NMD corridor is situated at the first floor of the six-story Medical Technology Building with a reception room located near the main entrance of the building. A public elevator and a stairway for the whole building are next to the reception room across from the hallway. Before remodeling, the building hallway or transit corridor, measured about 3 meter wide and 40 meter long, has been provided with seating chairs for patients and their relatives to use. The transit corridor divides the first floor into three gamma scan rooms on one side and the drug-administered room and a 4th-scan room on the opposite side. (see Figure 1)

In order not to interfere with daily NMD operations, a high sensitivity, collimated NaI(Tl) MCA gamma spectroscopy system is deployed at the far end of the hallway which is used for emergency exit only. The ambient radiation imposed by the NMD patients is determined by an MCA integral counting mode with a region of interest (ROI) from 60 keV to 2 MeV. In this surveillance study, the ambient radiation time-profile and peak dose rates have been characterized for the transit hallway or corridor where non-NMD medical staff or patients' relatives may present with the source-bearing patients. This MCA detector system is converted from a conventional thyroid up-take probe that can remotely monitor the ambient radiation at a greater distance. The viewing solid angle of the detector through the collimator, which has to cover most the hallway and seated patients, can be verified by a laser pointer shining through the detector assembly with it's detector temporarily removed. The gamma spectroscopy system is energy-calibrated using an americium-241 and a cesium-137 sources for low energy at 60 keV and high energy at 661 keV, respectively. The NMD patients waiting corridor and radiation monitoring instrumentation layout is depicted in Figure 2.

After remodelled, we have removed all the seating chairs from the hallway and divided patients that are waiting for gamma scans into two separate waiting rooms. We also redesigned our facility with a 0.5-cm thick Pb wall and larger space separations between rooms that include a PET/CT scan room, two separated 18FDG i.v. injection rooms, and a delayed-phase patient waiting room. The time-profile of the radiation levels were derived from a video-taped gamma spectroscopy using time-stripping, consecutive unfolding method. An internet-based video-exposure technique has also been used for this ambient radiation measurement. (Huang 2005) Energy-compensated, hand-held digital G-M survey meters are used to measure accumulated environment doses during the surveillance day. One of the G-M dosimeters, which can perform a peak rate locked-in feature and pre-set alarm capabilities, was also adopted for tracking the highest ambient radiation dose rates in the corridor hallway area. New look of the public transit corridor at the same location is shown in Figure 3 for comparison.

Results

With our newly improved facility in operation, we have demonstrated the NMD corridor area average daily count rate has dropped from 16,140 cpm to 3,333 cpm on one Wednesday surveillance study, before and after remodelling, respectively. Detector absolute efficiency calibration data, using a standard Tc-99m point source, at various distances are listed in Table1. During a benchmark testing of the same collimated detector, typical natural background is at about 900 cpm that gives a detection sensitivity of about 4.5 mCi when a Tc-99m point source is located at 30 meters distance. Real-time, radiation time-profile measurements during the surveillance day are illustrated in Table 2.

The hourly peak dose rate detected in patient waiting areas has also reduced to a factor of about two, from maximum dose rate of 40.4 micro-Sv/h to 23.4 micro-Sv/h, during one worst scenario. (See Table 3) Bold fonts in this table denote the exposure dose rates exceed 10 micro-sievert per hour.

Fig. 1. Hallway of the Nuclear Medicine Department before remodeling.

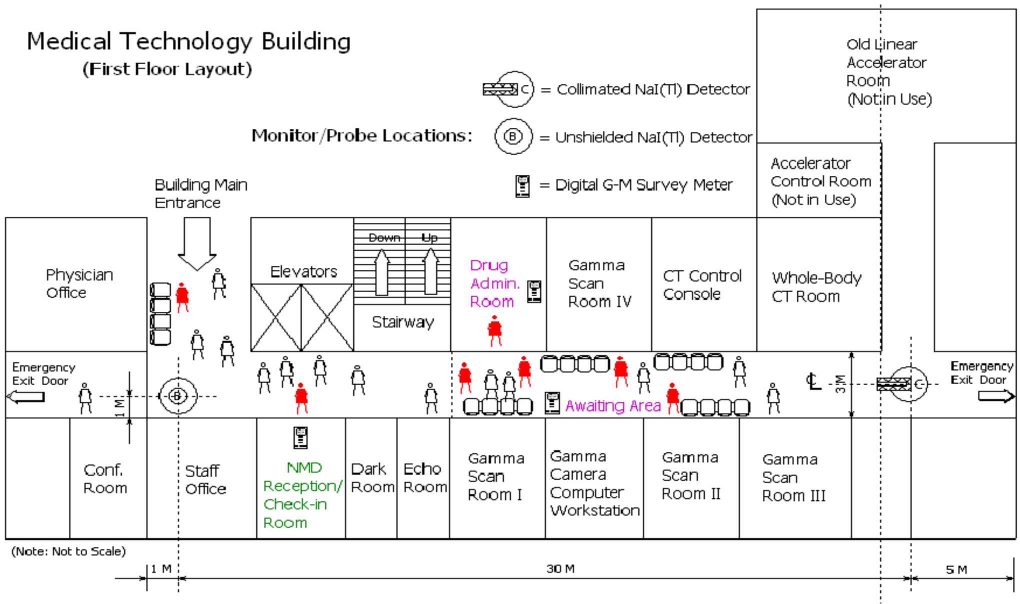

Fig. 2. Layout of the old NMD patients waiting corridor area and the ambient radiation monitoring instruments deployed.

Fig. 3. Hallway of the Nuclear Medicine Department after remodeling.

Table 1. Normalized detector absolute efficiency calibration data at various distances in the NMD corridor geometry. (see Fig. 2 for detection configuration).

Distance (meter)	Collimated NaI(Tl) Detector (cpm/mCi)	Unshielded Na(Tl) Detector (cpm/mCi)
5	10636	19422
10	2659	4029
15	1096	1499
20	537.6	728.8
25	318.6	477.8
30	203.1	314.8

Table 2. Real-time radiation surveillance study at the NMD corridor area.

Clock time interval (Surveillance Day)	MCA Count Rate (ave. hourly; cpm)	Peak Dose Rate (microsievert/h.)	Accumulated Daily Dose (microsievert)
7am-8am	1200	0.31	0.18
8am-9am	2138	5.86	0.33
9am-10am	3870	9.20	0.85
10am-11am	1965	0.83	1.20
11am-12am	3964	**13.1**	1.66
12am-1pm	8824	6.04	2.39
1pm-2pm	4196	1.02	2.64
2pm-3pm	7291	0.41	3.54
3pm-4pm	1649	0.34	3.62
4pm-5pm	923	0.28	3.82
5pm-6pm	903	0.30	4.08

Table 3. Sampling peak dose rates (in microSievert/h) at patient waiting rooms.

Awaiting Room* Before Remodeling	Time Interval	Awaiting Room #1* After Remodeling	Awaiting Room #2* After Remodeling
0.24	8am – 9am	0.90	0.22
23.6	9am – 10am	**23.4**	4.23
40.4	10am – 11am	6.41	4.47
10.7	11am – 12am	**15.4**	1.34
20.2	12am – 1pm	6.20	0.40
10.1	1pm – 2pm	0.74	0.38
13.7	2pm – 3pm	2.26	0.28
10.2	3pm - 4pm	0.36	0.23
0.61	4pm - 5pm	0.21	0.28

*Note: Patients awaiting room increases to two separate rooms after remodelling.

Discussion

In the past, the environmental dose assessments within our hospital were based on a monthly integral dose measurement only. Due to mobility nature of the NMD patients, it is difficult to predict collectively the true radiation levels in public access areas at a given time. In this report we have demonstrated with a much higher sensitive detector in a real-time basis to quantify this radiation level at our work environment and measured peak dose rates at specific interested spots that may be occupied by the general public. Based on these radiation monitoring data, suggestions to improve the NMD daily operation and the needed administrative changes have been proposed and reviewed by the management of the hospital. Some of these recommendations include waiting time reduced for the patients, less chairs and more space separation provided in the hallway and permanent survey meters installed with preset light/audio alarm capabilities, etc.

Conclusions

The great reduction of the environment dose was achieved mainly by using larger room space with thicker lead wall, from previous 2-mm to new 5 mm in lead thickness, and by increasing patient waiting rooms/areas with less chairs available in each seating location. Other NMD administrative control measure of our dose reduction program has also been emphasized in better patient routing, scheduling and less waiting time for the diagnostic patients.

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Contribution to the national survey of population exposure from selected X-ray medical examinations in Slovakia

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Abstract

EC Medical Exposure Directive 97/43/EURATOM requires in the Article 12 that the Member States will determine the distribution of individual dose estimates for relevant groups of the population. Large differences in the published data from various countries are mainly the consequence of variations in the equipment used, in the adopted choice of the examination categories, in the measured dose quantities and in the limited sample size due to the financial problems with the collection of the data for such survey. Even the dose values collected for one certain examination type show large uncertainties following the imaging technologies and mistakes in protocol standards. The recently published "Guidance on Estimating Population Doses from Medical X-ray Procedures" (EC Report No.154/2008), helped also in Slovakia to specify examination types and to harmonize the methodology for gathering the representative effective doses and annual frequencies, in the framework of the professional bodies involved in public health and radiation protection, for the first stage of the population dose estimates. The poster presentation will show the preliminary results for several examination categories (starting with mammography).

Introduction

The European Commission initiated a study for reviewing the situation regarding implementation of *Article 12* of 97/43 EC Directive [1]: „Estimation of patient dose" where it is proposed that Member states shall ensure that the distribution of individual dose estimates from medical exposure should be determined for the population and for relevant reference groups of the population, accepting exposures from medical X-ray imaging procedures.

Following European legislation, all radiological facilities should regularly supply information on patient doses, to be able to check reference levels and to determine the contribution to the collective dose. For determination of the trends in collective dose and for estimation of per caput doses from medical X-ray examinations was recently published European Guidance on Estimating Population Doses from Medical X-ray Procedures [2].

Objectives of population dose estimation in agreement with EC 154/2008 are:

- to observe trends in collective dose and/or per caput dose from medical X-ray examinations
- to determine the contribution of different imaging modalities to the collective dose
- to determine the relationship between the frequency of different types and their contribution to the collective dose
- to compare the contribution of medical X ray with that of other sources to population exposure

Similarly to the published population dose assessments from medical X-rays in several European countries (3, 4), also in Slovakia initial surveys of information about the population exposures from various types of diagnostic examinations has been collected. This information was available mainly due to the activities of the Slovak Commission of Quality Assurance in Radiology, working by the Slovak Ministry of Health. The main objective of the activities of the Commission are presented in realization of quality audits for reaching the same level of clinical image evaluations and equal technical quality program implementation.

In our presentation we haw collected the results of radiation load of patients undergoing mammography examinations, as an example of our procedure by assessing the patient doses for national dose inventories. In the cohort of 7 934 patients we show the age distribution, distribution of breast thickness and differences between organ doses in conventional and digital mammography examinations.

Material and methods

Establishment of average absorbed glandular doses of mammography patients was realized in a group of 7 934 women from two films screen and two digital mammography units in the period of March 2009 to February 2010 (Table 1). For collection of all parameters (x-ray tube output, exposure conditions, relevant patient parameters, etc.), used for the mean glandular tissue doses estimation a special software has been prepared in the case of film-screen mammography and for digital mammography units the required data has been collected from Central Hospital Information Systems.

Table 1. Number of monitored patients, monitoring period and useded mammography units.

Mammography units	Number of monitored patients	Monitoring period
Sophie Planmed F1 (Film-Screen)	2000	March 2009 – October 2009
Siemens Mammomat 3000 F2 (Film-Screen)	2000	June 2009 – November 2009
Hologic Selenia D1 (Digital)	2000	October 2009 – January 2010
Hologic Selenia D2 (Digital)	1934	October 2009 – February 2010

Results

Due to the wide variation in radiation risks with age at exposure, for the group of monitored patient we have determined the age distribution. The data were divided in to five years age bands (Table 2). For conversion into effective dose two expositions in both breasts were taken into account. The new tissue equivalent weighting factor for the breast (0,12) allowed to convert the mean glandular dose to both breasts into effective dose (ICRP 103). On the Fig 1 is presented the age distribution in all monitored mammography units and the average of the all units shows that the highest number of mammography examinations was observed in age interval form 50 to 55 years.

Table 2. Age distribution of monitored patients.

Age (years)	Number of monitored patients			
	D1	D2	F1	F2
under 40	9	7	7	5
40 – 45	269	313	394	296
45 - 50	350	318	408	457
50 - 55	392	460	400	439
55 - 60	421	380	343	320
60 - 65	273	235	199	237
65 - 70	177	146	143	144
70 - 75	61	47	73	69
75 - 80	29	19	25	29
over 80	19	9	8	4

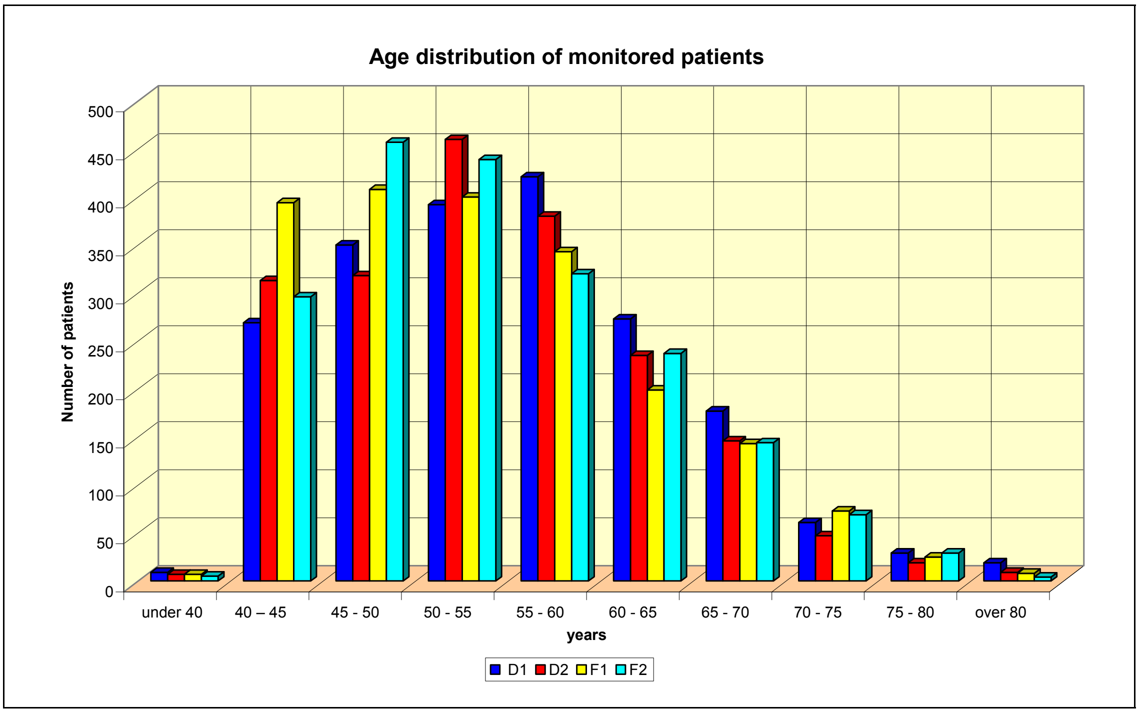

Fig. 1. Age distribution of monitored patients.

One of the most important factors that can significantly influence the absorbed dose to the breast is its volume and composition. As can be seen from the Table 3 there are significant differences in breast thicknesses, showing that in Slovakia the majority of examined patients has the thickness between (50 – 60) mm.

Table 3. Thickness distribution of monitored patients.

Thickneess of breast (mm)	Number of monitored patients			
	D1	F1	D2	F2
under 20	33	0	2	40
20 - 30	176	10	29	161
30 - 40	453	58	136	349
40 - 50	636	245	357	633
50 - 60	476	590	564	574
60 - 70	193	712	562	217
70 - 80	33	326	231	26
over 80	0	59	53	0

Average thickness distribution from all monitored mammography units are presented in Fig.2

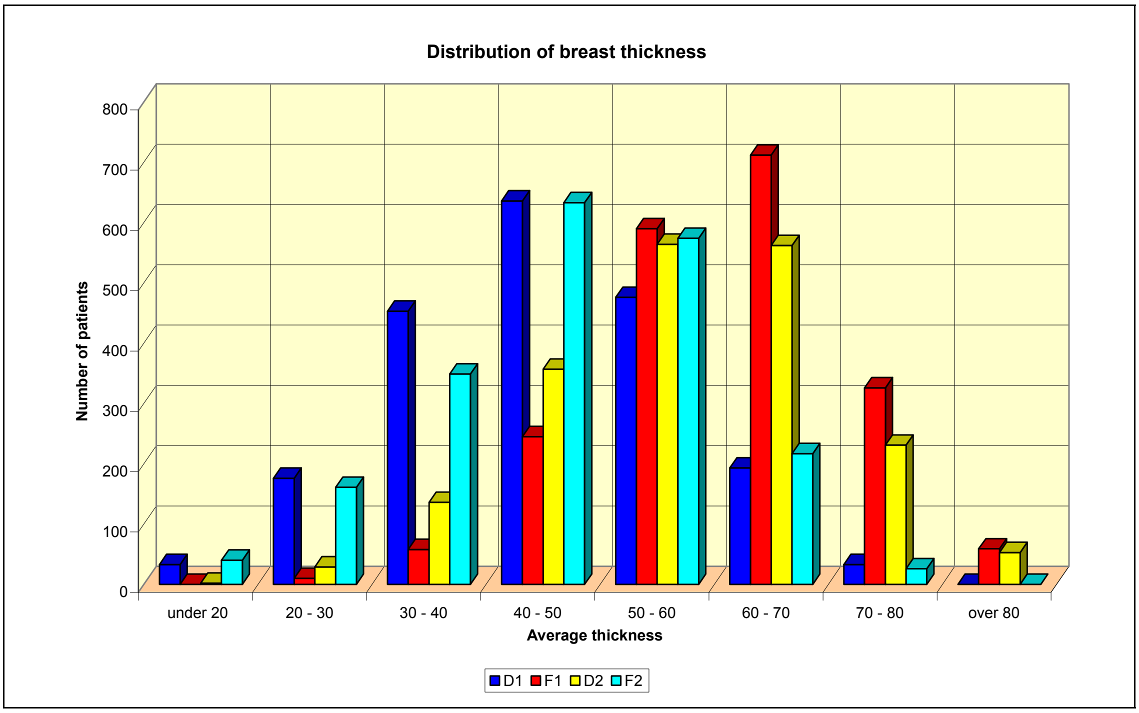

Fig. 2. Distribution of breast thickness.

Table 4. Absorbed doses in the breast for various thicknesses.

Thickneess of breast (mm)	Absorbed dose in glandular tissue (mGy)			
	D1	**F1**	**D2**	**F2**
under 20	1,188	-	0,611	0,465
20 - 30	1,509	1,264	0,88	0,530
30 - 40	1,66	1,415	1,099	0,621
40 - 50	1,816	1,604	1,245	0,718
50 - 60	2,05	1,693	1,405	0,883
60 - 70	2,25	1,669	1,496	1,188
70 - 80	2,08	1,80	1,57	1,646
over 80	-	1,95	1,777	-

Absorbed doses in glandular tissue were measured for thicknesses of breast from 20 mm to 80 mm with 10 mm step (Table 4). In the Fig. 3 are given the average values of glandular tissue doses for monitored mammography units.

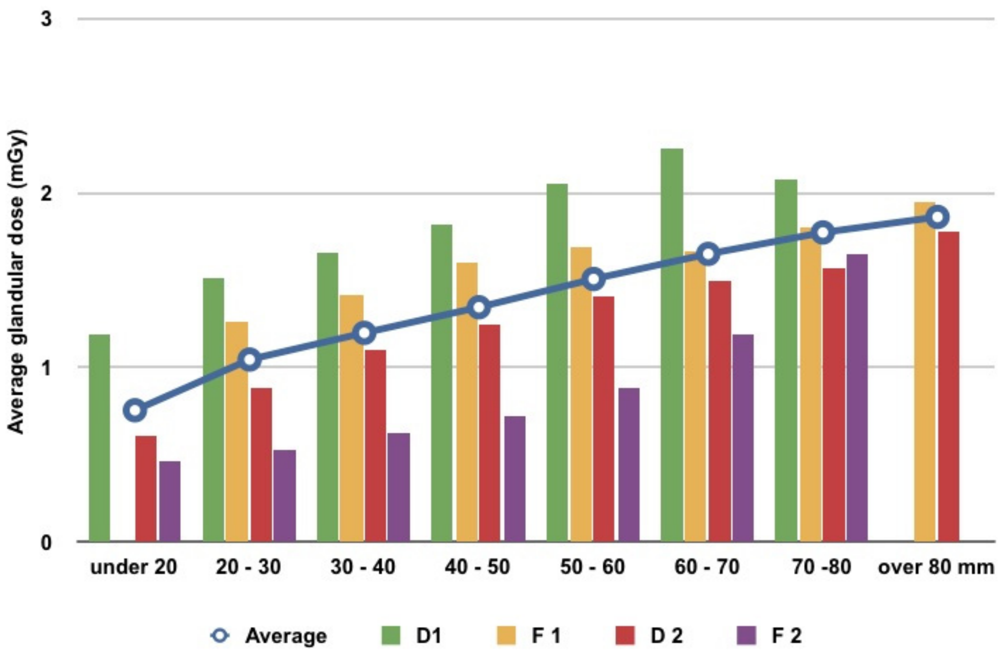

Fig. 3. Comparison of breast thicknesses and average glandular doses.

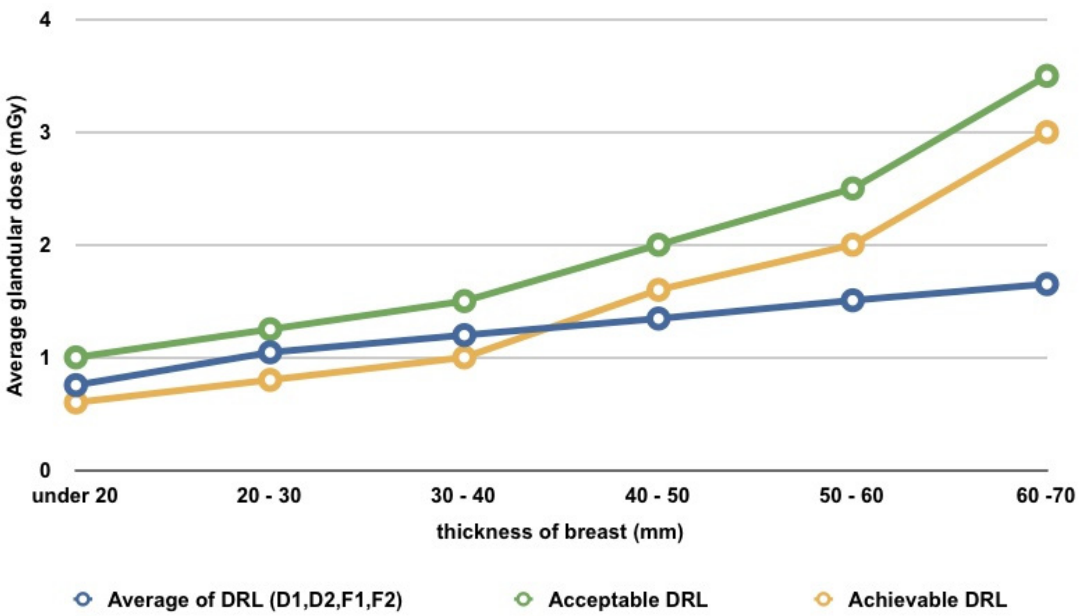

Fig. 4. Comparison of average glandular tissue doses with international diagnostic reference level.

Evaluated mean glandular tissue doses as a function of breath thickness has been compared with internationally accepted diagnostic reference level and with achievable level (Fig.4). On the basis of these results it can be proposed reducing of diagnostic reference levels in the monitored mammography units.

Discussion and conclusions

In our presentation we tried to compare the survey of patient doses on digital and conventional mammography units. In spite of the fact that we monitored two equal digital mammography units the differences in the average mean glandular doses reached up to 45% taking into account the breast thickness. The differences are higher for small thicknesses of breast.

The advantage of digital mammography is shorter time of whole examination by reducing time for developing the film, better storing of imaged, reduction of repeating examinations, it means, great potential in better cancer detection. On the other site, the radiation dose of the patients can be potentially increasing.

Due to the new tissue weighting factors recommended in the ICRP 103 the significant increase of effective doses calculated for the breast examinations compared with the past are observed. In the Table 5 are given the calculated effective dose ranges for two digital mammography units.

Table 5. Comparison of effective doses of two digital mammography units Hologic Selenia.

Mammography units	Effective dose (mSv)
Hologic Selenia D1	0,252 – 1,152
Hologic Selenia D2	0,504 – 0,957

In similar way we continue in the collecting of the information about population doses from medical exposure in the selected types of the X-ray examinations recommended in the European guidance No. 154.

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Utility of a web based data survey for a national MDCT radiology practice survey

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Abstract

Purpose: To develop a robust and simple Internet based survey form for national MDCT dosimetry surveys. To provide participating practices with a site specific practice reference level (PRL) report for submitted protocols and to use submitted and collated PRL data to construct national diagnostic reference levels (DRLs) for MDCT.

Methods and Materials: As part of a national MDCT dose survey project, a web based MDCT protocol survey form is being developed to record appropriate parameters for dosimetry calculations. The initial draft Excel®[1] version requested responders to provide practice contact and registration information. They were also asked to submit 20 sets of patient scan data for the 7 generic adult protocols chosen. These included integral protocol DLP (mGy.cm), patient weight, patient height and patient slice diameter at a specified anatomical landmark. Fifteen practices were invited to participate of which 11 responded. With version 2 of the survey form, more instructional detail was provided and the slice diameter data point was removed. Data sets are to be kept in an SQL database and transferred to a commercial spreadsheet program for additional analysis.

Results: Initially 15 radiology practices were invited to participate in the survey of which 11 responded. Practices were given 4 weeks to submit data and the survey response was poor. No practice managed to complete the survey with 20 patients per protocol (140 data points). Of the 77 protocols submitted (11 practices x 7 protocols) only 6 protocols were submitted with all 20 patient data fields satisfactorily populated. These were spread across 4 of the 11 practices. However, sufficient data was submitted, in most cases, for a generic dose report to be generated and a survey DRL to be calculated.

Conclusion: This pilot MDCT data survey has shown that the application of a web based MDCT protocol survey should provide sufficient data for accurate dosimetry calculations and comparative intersite surveys to be performed. When released nationally, it should serve as a useful data source for the development of national MDCT DRLs.

[1] Microsoft Corporation, USA.

Introduction

It is well recognised that the greatest source of patient dose from diagnostic imaging is from multi-detector computed tomography (MDCT) (NCRP 2009). The rapidly increasing growth of applications of MDCT scanning has the unwanted outcome of a significant increase in population cumulative effective dose (Brenner and Hall 2007). The Australian Radiation Protection & Nuclear Safety Agency (ARPANSA) estimates that growth in MDCT scans, based on Medicare Benefits Schedule data, is increasing at approximately 9% per annum with over 2 million MDCT scans being performed in 2009 (Fig.1) (Hayton, Wallace et al. 2009).

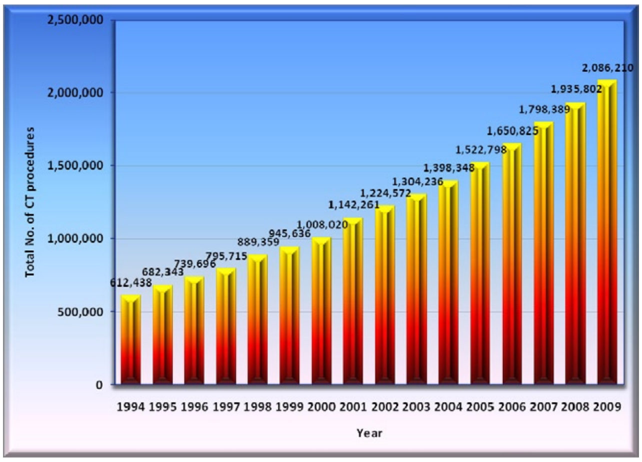

Fig. 1. MDCT procedures in Australia 1994 – 2009.

While the application of MDCT to diagnosis and therapy should always be applied in the sound knowledge of benefit outweighing risk, its application does increase the probability of stochastic detriment to the population e.g. expression of cancerous disease (Brenner, Elliston et al. 2001). Radiobiology research also points to the increased risk of stochastic effects in the paediatric population compared with adults (Hall and Brenner 2008). MDCT has many advantages to offer radiological investigations and its uses and applications are increasing. The development of the technology in terms of its power, flexibility, and utility, ease of use and diagnostic image quality has resulted in an exponential increase in its application in virtually all fields of clinical practice. While the dose to the individual and the consequent individual risk is relatively low, the population risk is compounded by the increasing number of imaging and therapeutic applications undertaken in current medical practice. This may result in an increasing public health concern due to the escalating risks of the expression of stochastic effects. As the expression of stochastic detriment takes many decades to appear, we may only be at the threshold of an increasing MDCT induced cancer rate (Larson, Rader et al. 2007). To address this increasing population health risk ARPANSA has developed a Code of Practice and Safety Guide for the application of ionising radiation in medicine (ARPANSA 2008; ARPANSA 2008). The Code of Practice has been taken up by state regulators and its provisions are now a necessary compliance requirement for statutory radiation safety acts, regulations and conditions of license.

To obtain a clearer assessment of this potential risk, it is imperative that a review of the doses delivered from common radiology procedures is undertaken both at the site of practice and consolidated into a regional or national analysis. DRLs can then be constructed and used as a comparative indicator of radiation efficiency at the practice, regional and national levels. Once site DRLs are established, they can be regularly reviewed and used as a baseline for the implementation of an optimisation program to maximise the efficient use of radiation while maintaining diagnostic image quality. Regional and national DRLs can be obtained and reviewed over a longer period and used as a dose benchmark that, by definition, 75% of practices will be able to achieve. As multiple iterations of the DRL review are undertaken it is expected that the spread of doses per procedure should decrease and the resultant 75th percentile value would diminish.

To assist with the efficient application of ionising radiation in medicine it has become common practice for regional dosimetry surveys to be undertaken to measure the spread of doses that are used for similar radiological investigations across various institutions. This spread of dose is statistically analysed and a 75th percentile (3rd quartile) value is calculated and used as an indicative measure of reasonable practice. The agreed percentile value is termed the Diagnostic Reference Level (DRL) and is conditional upon its dose delivered to the patient providing a diagnostic image of sufficient quality for diagnosis.

The development of dose reference levels for common radiology procedures has been ongoing for the past 2 decades. They provide a simple, comparative dose metric of the indicative, 75th percentile, dose delivered from common radiological procedures. Their application is through the comparison of individual mean or median site dosimetry with the 75th percentile value of the spread of doses for similar radiology procedures from multiple practices, regional or national estimates. Individual site dosimetry is established by recording and calculating, for a range of similar procedures, the mean or median value of the recorded doses for that procedure from a reference or general population of patients. The DRLs provide practices with a measure against which to compare their own practice doses with those of their peers and, if consistently exceeded without clinical justification, will provide an indication for the need of some modification of acquisition protocol to reduce dose. This process of individual site and regional/national comparison is undertaken on a regular basis.

It is important to understand that **DRLs are not dose limits**; they are simply indicators of common practice and are expected to vary over time depending upon changes in technology, acquisition protocols and clinical application. If a practice, after due consideration and optimisation, can justify a local DRL that is higher than what is commonly accepted then they have met the requirements of the DRL philosophy.

The UK introduced dose reference levels in 1990 as an indicator of ‘abnormally high doses’. They used the 3rd quartile value of the mean dose distributions taken from a national dose survey in the mid-1980’. If the practices’ mean dose consistently exceeded the reference dose then an investigation should have been undertaken to establish the cause and take corrective action if the doses are found to be clinically unjustified. The UK data have been updated periodically, every 5th year, for the past 20 years and each review has led to a reduction in the previously published MDCT DRL (Wall and Shrimpton 1998). Their next review (2009/10) will be their 5th iteration of the process. Unfortunately, the development of DRL data in diagnostic radiology within Australia is still at an early stage, as no national surveys have been carried out for any radiological examinations for the express purpose of establishing national DRLs. However, various organizations, regulatory authorities and individual practices have carried out limited CT dose surveys (Boal, Hedt et al. 1999).

ARPANSA had not undertaken a national MDCT dose survey since 1997 (Thomson and Tingey 1997; Wise and Thomson 2004). To estimate contemporary caput effective dose (mSv), Australian MDCT scanning episodes were combined with European MDCT dose data from the recent DOSE DATAMED surveys (EC 2008). The following graph shows the 3 estimates of extrapolated caput effective dose using the DOSE DATAMED cohorts of their low, medium and high ranges of survey doses. Unfortunately this estimate indicates that the Australian extrapolated doses closely fit the highest European cohort. The data estimate is that in 2009 the caput effective dose contribution from MDCT scanning is approximately 1.2 mSv.

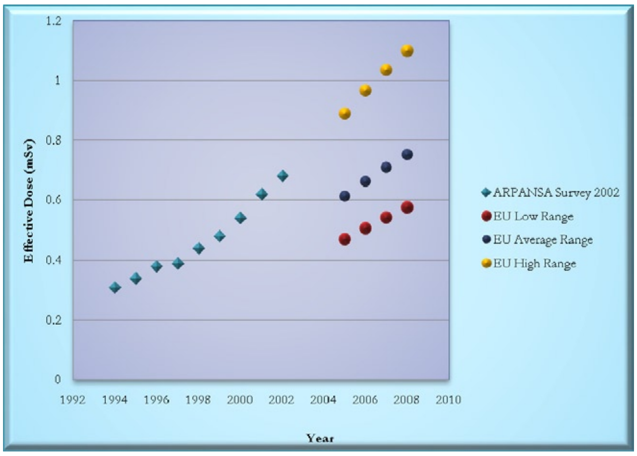

Fig. 2. Estimated caput effective dose (mSv) for Australian MDCT scanning 1994 – 2008.

Material and methods

The ARPANSA MDCT survey was based on the metrics of dose length product (DLP - mGy.cm) available from all MDCT consoles, plus patient weight, height and width, defined at a specific anatomical landmark. Patient weight and height, was to be obtained from questioning the patient and slice diameter was taken from review of the acquired image set. These parameters were recorded for 20 patients, for 7 common protocols. Fifteen radiology practices were invited to initially participate. The 7 common MDCT protocols are (i) head, (ii) cervical spine, (iii) chest, (iv) chest-abdomen, (v) abdomen-pelvis, (vi) lumbar spine and (vii) chest-abdomen-pelvis. Further protocols may be added to the survey as required. The survey forms will be accessible from the web via the ARPANSA website [www.arpansa.gov.au] and are expected to be posted towards the end of 2010.

To test the utility of the proposed web based program a draft survey was developed using Microsoft Excel (Microsoft Office 2003) and sent to selected radiology practices for completion and comment. Excel was chosen, as it is a readily recognisable, user-friendly program with which most scientifically educated users would be expected to have some experience. To minimise potential problems with data and worksheet corruption, due to participant 'editing', each worksheet was password protected.

The survey workbook was made up of a number of information and input worksheets that required participant response. These were a (i) general information page, (ii) radiologist, radiographer and practice registration page, (iii) protocol management page, (iv) data entry page per protocol, and a (v) data report page per protocol. Participants had full flexibility to choose the protocols that they wished to populate. For the initial draft survey it was hoped that all protocols would be fully completed. They were expected to provide a full data sheet for each protocol response. For each protocol it was requested that 20 sets of patient data points were logged. Each data set comprised of the integral dose length product (DLP, mGy.cm) recorded and displayed at the end of the investigation acquisition. This was usually to be found on the 'protocol page' that most systems generate at the end of a patient scan. Many practices 'screen grab' this page and send it to PACS as a formal dosimetry record of the investigation. As well as the integral DLP they were also requested to record the patients weight and height. It was not thought necessary to physically measure these parameters as a high degree of accuracy was not thought important. Participants were informed that a verbal questioning of the patient would be sufficient and would supply a sufficiently accurate indicative measure. It was also requested that the width of the patient at a defined anatomical landmark should also be recorded. This required the review of reconstructed images and has since been deleted as a survey data point. As each data set was added, the report page graph, which provided a histogram of DLP spread, the median DLP of the spread of input DLP and the maximum and minimum input DLP was updated. Participants were strongly requested to complete the protocol data with 20 patients to improve statistical accuracy.

To clearly distinguish between the all practice DRL and the individual practice or, local DRL, it was decided to use the term 'practice reference level' (PRL) which is defined as the median of the spread of all submitted DLPs per protocol.

At the completion of the survey period of 4 weeks each practice was also provided with a written report which included a letter of appreciation, including a table of European DRLs (Tsapaki, Aldrich et al. 2006) and graphs of all practice data for each protocol showing the individual practice PRL, the all practice DRL and the 95% confidence interval for the all practice DRL. This graph was also presented in a slightly different manner with all practice DRL, individual practice PRL and the maximum and minimum spread of individual practice DLP. A histogram of the spread of the practice DLPs per protocol separated into bins of 100 mGy.cm were also included with the survey DRL shown.

All submitted practice data was treated as strictly commercial-in-confidence. Participants only received their own practice data or anonymised all practice data for comparison.

Results

Of the 15 practices approached for participation in the survey only 11 responded, of these only 2 practices managed to submit 20 heads and 2 practices managed to submit 20 abdomen-pelvis scans. One participant provided information on 2 MDCT scanners but used retrospective data that did not include all patient weight, height or any slice diameters. The following table shows the author defined acceptable data responses from the participating practices per protocol.

Table 1. Acceptable Data Response per Practice per Protocol for a 4 week Survey Period.

Practice	Head	Neck	Chest	Chest and Abdo	Abdo and Pelvis	Lumbar Spine	Chest, Abdo and Pelvis	All (140)
A	6	3	5	2	13	1	17	47
B	1	2	3	0	2	4	18	30
C	1	8	14	0	20	0	0	43
D	20	6	8	0	15	3	6	58
E	12	4	8	0	11	5	9	49
F	16	10	10	0	1	10	13	60
G	20	5	13	0	20	12	8	78
H	20	0	14	1	20	0	6	61
I	0	2	0	0	0	4	0	6
J	0	0	0	0	0	4	0	4
K	4	6	8	2	11	0	12	43

However, even with the restricted data input it was still thought necessary to calculate an all practice DRL for the purpose of practice comparison. Recognising the limitation in the statistical accuracy of this calculation it still provides an indicative DRL value against which all practices could compare. The following table shows the European DRL, all practice survey DRL and practice reference level (PRL) spread (min – max).

Table 2. European DRL, All Practice Survey DRL and PRL spread per Protocol.

Protocol	European DRL (mGy.cm)	Survey DRL (mGy.cm)	PRL Spread (mGy.cm)
Brain	1050	1022	392 – 1692
Cervical Spine		815	190 – 1280
Chest	650	784	30 – 1488
Chest-Abdomen		659	324 - 723
Abdomen-Pelvis	780 (abdomen only)	837	143 – 1555
Lumbar Spine		1031	267 – 1879
Chest-Abdomen-Pelvis		1338	265 - 3028

Embedded within the survey forms was a report page where a scattergram was shown and updated after each data input for each protocol. It displayed a calculated PRL with noted minimum and maximum DLP values. As previously discussed, the participant was encouraged to input all 20 data points to increase the statistical accuracy of the displayed PRL. This PRL protocol report page could be sent to a printer as required for a hardcopy record.

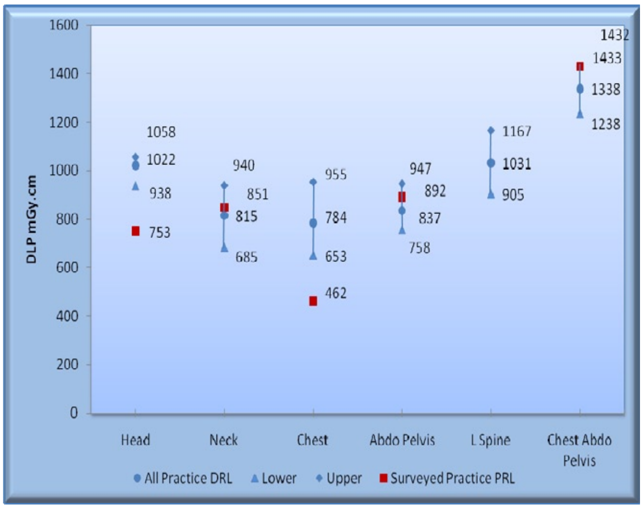

Fig. 3. All Practice DRL with 95% confidence limits and Surveyed Practice Local PRL.

As well as the embedded survey PRL protocol reports ARPANSA also generated a more detailed individual practice report that was sent post-survey. In the covering letter of the report was a table of European DRLs for reference (Tsapaki, Aldrich et al. 2006). This report provided the practice with additional information including DRL values per protocol with 95% confidence interval and their own practice PRL (Fig. 3).

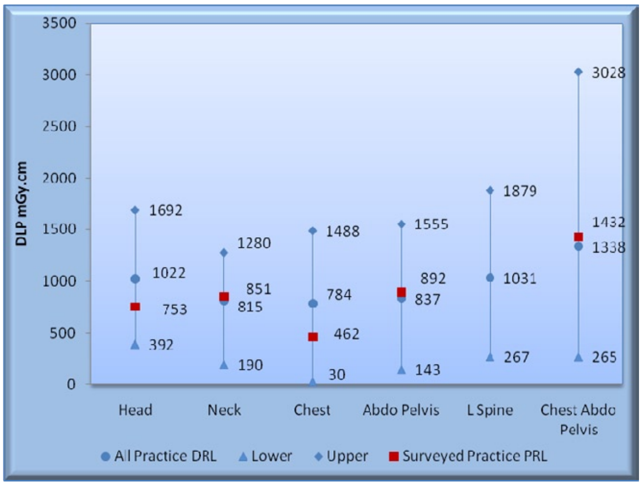

Fig. 4. All Practice DRL with max - min PRL spread values and Surveyed Practice Local PRL.

To provide the practice with information on where their protocol PRL sits in comparison to the spread of submitted PRL values a graphical presentation was attached (Fig. 4). These two graphs enabled each practice to clearly see where their practice stood in terms of the overall dosimetry for the generic protocol data submitted. If the practice did not perform the protocol or chose not to provide any data then a surveyed practice PRL data point could not be displayed.

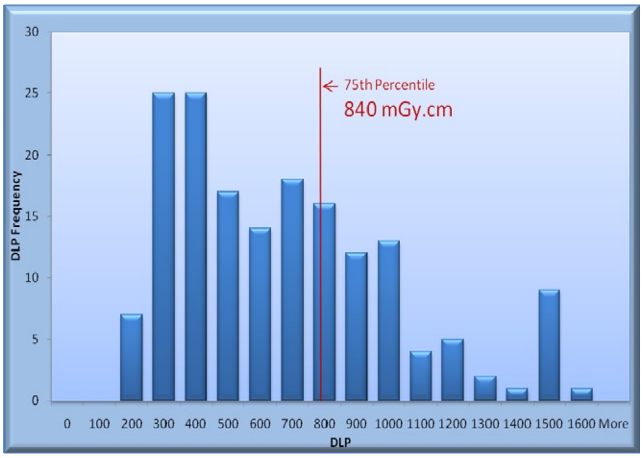

Fig. 5. All Practice Abdo-Pelvis DLP Spread.

Each practice was then provided with two graphs per protocol. The first histogram presented the all practice spread of DLP separated into 100 mGy.cm bins. The DRL was also shown (Fig. 5).

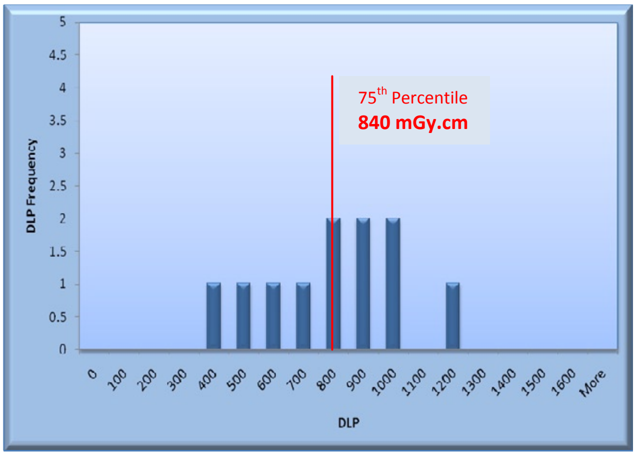

Fig. 6. Example Surveyed Practice Abdo-Pelvis DLP Spread.

The second histogram (Fig. 6) presented the spread of the practice DLP with the DRL. These graphs were displayed one over the other on the same page making it easier to visually compare the practice spread with the overall spread of all practice DLP. Any practice protocol with a significant spread of DLP values to the right of the DRL should indicate to that practice that an optimisation strategy should be undertaken.

Discussion

The initial survey period was for approximately 4 weeks. There was a variety of responses ranging from a complete 20 patient data point submission for some protocols to quite poor data submission compliance. Many practices were unable to submit the 20 patients for many of the protocols. Some practices did not submit all DLP, weight, height and slice diameter data sets. The paucity of complete responses necessitated a review of the survey forms and participant instructions.

Participant feedback from the first draft survey has led to the following modifications for future surveys. To remove any confusion concerning the scan range for each protocol a phantom graphic with red scan box defining the acceptable acquisition region has been included. Any patient volume scanned within the red box and meeting the appropriate protocol indications can be submitted. There was some confusion with the assumed accuracy required for the weight and height data. Indicative data was all that was required and it was not expected that patient measurement should be undertaken but a simple questioning of the patient would suffice. Weight within ±2 kg and height within ±3 cm was required. Provision of a pdf print paper ruler for door entrance and imperial to metric height convertor for older patients has been supplied for ease of use. The ruler can be taped to the entrance door at the required height and patients quickly 'eye-balled' for height estimation as they walk through. Participant response clearly indicated that the chest-abdomen protocol was not used as a common acquisition and due to its poor utility has been removed from the set of standard protocols for DRL assessment. Withdrawal of the slice diameter requirement, deemed to be an impediment for survey completion was removed to improve the ease of response for participants. This removes the requirement of the radiographer to retrospectively seek the appropriate anatomical plane to measure the patient diameter from the acquired image data set. Consequently, all data can be entered at the time of acquisition with no retrospective data mining required. As mentioned above a clearer definition of reference level terms with the inclusion of the defined 'practice reference level' (PRL) as the median of the spread of protocol DLPs was included. To provide information for appropriate feedback to participating practices concerning any excessive dose submissions, all protocol data pages now request the submission of their generic acquisition parameters at the top of the protocol data entry page. This should enable the survey team to communicate potential dose saving strategies to any practice that is clearly giving inappropriate doses. The feedback to the practices will be based on their need for the implementation of an optimisation strategy concentrating on the relevant acquisition parameters.

ARPANSA will use the accumulated practice data to calculate and publish, in consultation with stakeholders; Australian national DRLs. It is hoped that around one third of MDCT practices will respond in the 1st year of the survey. This should equate to approximately 300 practices from which DRLs could be published. Initially, practices will be provided with common European DRL data for comparison (Tsapaki, Aldrich et al. 2006)

It is expected that as the survey will be posted on the web and always available, it will be used as an optimisation tool for dose reduction when required. While this project does not profess to be an MDCT optimisation program, it does provide a robust metric tool for MDCT dose assessment and adjustment. As the optimisation of acquisition protocols is an iterative process, this survey provides a useful and easy to use tool for dose comparison. The acquisition of data for national DRL assessment is a passive outcome of a radiology practice investigating their own patient dosimetry. It is expected that other protocols will be added to the survey as required, e.g. CTCA, CT colonography, etc.

As an additional driver for survey participation it has recently been recognised by state and territory regulators as an approved method to provide compliance with the ARPANSA Code of Practice RPS 14 which requires;

'*3.1.8 The Responsible Person must establish a program to ensure that radiation doses administered to a patient for diagnostic purposes are:*

(a) periodically compared with ***diagnostic reference levels (DRLs) for diagnostic procedures for which DRLs have been*** *established in Australia; and*

(b) if DRLs are consistently exceeded, reviewed to determine whether radiation protection has been optimised.' (ARPANSA 2008).

When the MDCT survey is satisfactorily rolled out ARPANSA will then develop survey documentation to establish DRLs for interventional fluoroscopy, PET-CT, mammography, general radiography and fluoroscopy.

Conclusions

The future provision of this web based survey will provide MDCT radiology practices with a tool to assist them in recording their median dose metrics for MDCT procedures in units of DLP (mGy.cm). It will also generate a protocol report which may be used as a comparative metric for any optimisation program undertaken in response to dosimetry concerns. The survey report will also be taken as meeting the compliance requirements of ARPANSA RPS 14 which has been taken up as a regulatory requirement for all state and territory jurisdictions within Australia. ARPANSA will additionally provide the radiology practice with a written report comparing their protocol dosimetry with, initially European, and eventually, Australian generated, national DRLs.

The multi-disciplinary support of the survey by the various radiology professional societies, the ready availability of the survey via the internet and its acceptance by the regulators as a compliance tool should ensure its take up by radiology practices across the country. It is expected that after approximately 300 practices have responded ARPANSA will be able to generate an accurately determined set of DRLs for MDCT practice in Australia.

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Application of the European DOSE DATAMED methodology and reference doses for the estimate of Australian MDCT effective dose (mSv)

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Abstract

Purpose: To calculate the indicative effective dose (mSv) per individual per annum to the Australian population from current MDCT procedures using DOSE DATAMED MDCT dosimetry as a reference.

Methods and Materials: DOSE DATAMED was a multinational radiology dosimetry project with contributions from ten European countries with extensive experience in undertaking national dosimetry surveys (http:/ec.europa.eu/energy/nuclear/radioprotection/publication_en-htm). The lack of contemporary Australian MDCT dosimetry survey data post-2002, required the application of a recognised European dosimetry data set for current generic caput effective dose calculation. Utilising Medicare Benefits Scheme[1] (MBS), CT procedure data, plus an additional 31% gross-up factor for data not captured by MBS, an indicative caput dose calculation was made based on the current European DOSE DATAMED dosimetry data.

Results: DOSE DATAMED grouped all MDCT procedures into 7 generic scans with the following doses; Head (2.4 mSv), Neck (2.8 mSv), Chest (8.2 mSv), Spine (6 mSv), Abdomen (13.5 mSv), Pelvis (8.8 mSv) and Trunk (24.4 mSv). MBS MDCT categories were rendered down to fit into these broad categories and a resultant effective dose of approximately 1.2 mSv per individual per annum was calculated.

Conclusion: Previous ARPANSA CT surveys have shown the per caput dose has grown from 0.37 mSv (1994) to 0.9 mSv (2002) The current per caput dose contribution is approximately 1.2 mSv (2008). As the number of MDCT procedures in Australia is growing at approximately 9% per annum, the MDCT dose contribution to the Australian population background radiation dose is also steadily increasing.

Introduction

A 2009 report by the National Council for Radiation Protection (NCRP) showed that in the United States, Medical sources contribute 48% of the total dose received by the public every year. Of the total dose from Medical sources it was found that CT

[1] Australian Government National Health Insurance Scheme

contributed 50% of the dose while only making up 17% of the procedures (NCRP 2009). In Australia in 2009, Computed Tomography accounted for approximately 12% of all diagnostic imaging procedures recorded by Medicare Australia (Medicare 2010).

Material and methods

The numbers of CT procedures recorded by Medicare Australia are available via their website (Medicare 2010). Data is grouped according to 10 discrete categories. Category 5 – Diagnostic Imaging Services covers Ultrasound, Computed Tomography, Diagnostic Radiology, Nuclear Medicine Imaging and Magnetic Resonance Imaging. Data from category 5 was analysed as a whole and the data specifically for Computed Tomography was analysed for gender, age group, state and category break downs. Data on the age and sex structure of the Australian population was also obtained from the Australian Bureau of Statistics website (ABS 2010).

Results

Growth in CT

Over the 16 year period between 1994 and 2009 the number of MDCT procedures recorded by Medicare Australia has increased 240% from 610 thousand to 2.1 million (Medicare 2010). The Australian population increased only 23% in the same period (ABS 2010). The average per annum growth in MDCT procedures is 8.53% which is 6 times larger than the annual growth in the population which is 1.36% (Medicare 2010) (ABS 2010). Figure 1 shows the annual percentage growth rate of MDCT procedures and the Australian population.

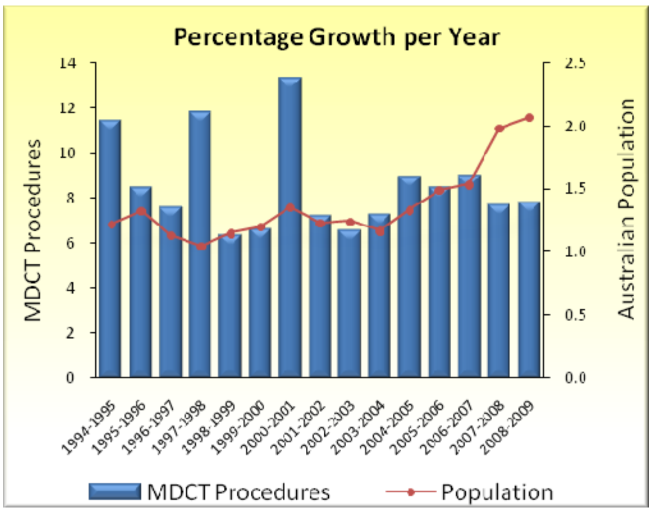

Fig. 1. Percentage growth in MDCT procedures and the Australian population 1994-2009.

Data Gap

Medicare Australia does not capture all MDCT procedures performed. A survey conducted by ARPANSA in the mid nineties estimated that Medicare Australia only captured 76% of all procedures actually carried out (Wise and Thomson 2004). This number is thought to be much lower for paediatric procedures (to be published by Brady).

Procedures that are not captured by Medicare Australia include – Public Patient Radiology (funded directly from state and territory governments), WorkCover referrals, Department of Veterans' Affairs referrals and Traffic Accident Commission referrals.

If this estimated data gap is accurate the number of MDCT procedures performed in Australia in 2009 would increase from 2.1 million to 2.7 million.

Age group breakdown

Medicare Australia breaks down procedure data into 10 separate age groups. Figure 2 shows this data for 2009 normalised to the population structure. The age group with the greatest number of MDCT procedures per person is the 75-84 year olds, closely followed by the 65-74 year olds.

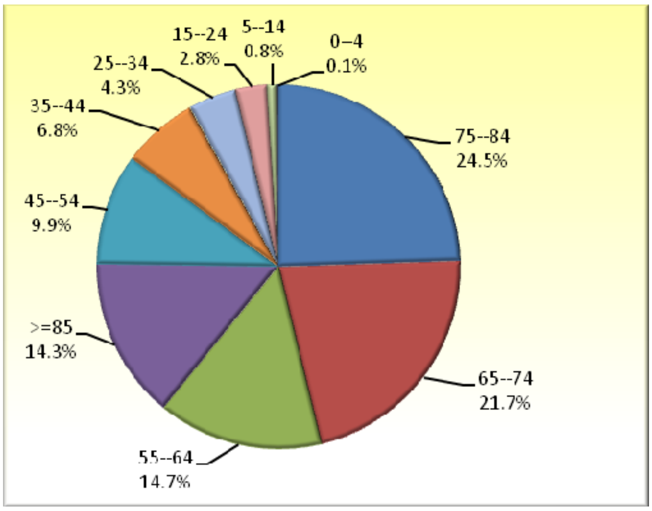

Fig. 2. Age break down of CT procedures for 2009.

Gender breakdown

The data from Medicare Australia shows that the majority of MDCT procedures performed each year in Australia are on females. MDCT procedures performed on females account for 54% of all procedures consistently over the past 16 years. This discrepancy is not accounted for by normalising to the gender structure of the population. However, females do not have the majority across all ten age groups. Both paediatric age groups are dominated by procedures on males. Figure 3 shows the gender breakdown for all age groups.

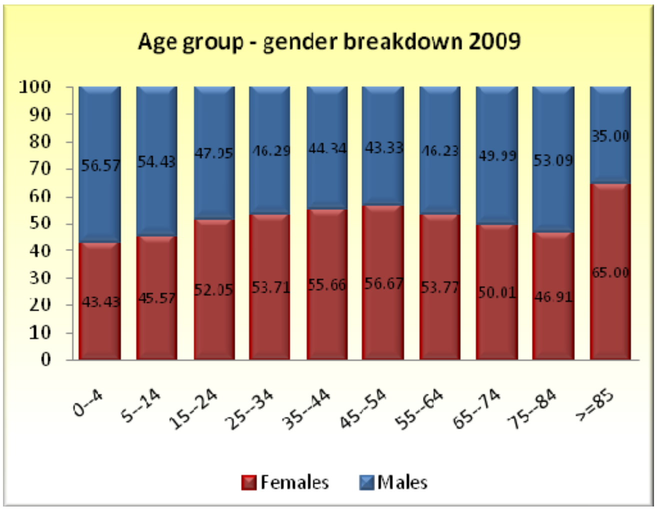

Fig. 3. Age group - gender break down of CT procedures for 2009.

Category break down

Medicare Australia records data on MDCT procedures according to the item number the procedure was billed under. Each item number indicates which anatomical region of the body was scanned and whether a contrast agent was used. These item numbers can be grouped according to general anatomical region. Currently the item numbers can be grouped into sixteen separate categories. Figure 4 shows the percentage of scans in each category for 2009. The five most common categories in 2009 were – head, abdomen and pelvis, lumbar spine, chest/thorax and neck, chest, abdomen and pelvis.

The rate of growth is not constant across each category. In the five years from 2004 the category with the highest percentage growth is interventional. Both the pelvimetry and abdomen category have negative growth. The total percentage growth of all 15 categories over the past five years is shown in figure 5.

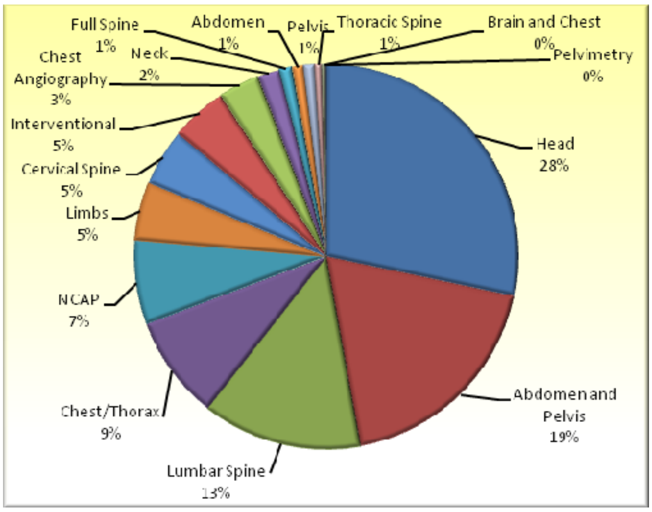

Fig. 4. Category break down of all CT procedures for 2009.

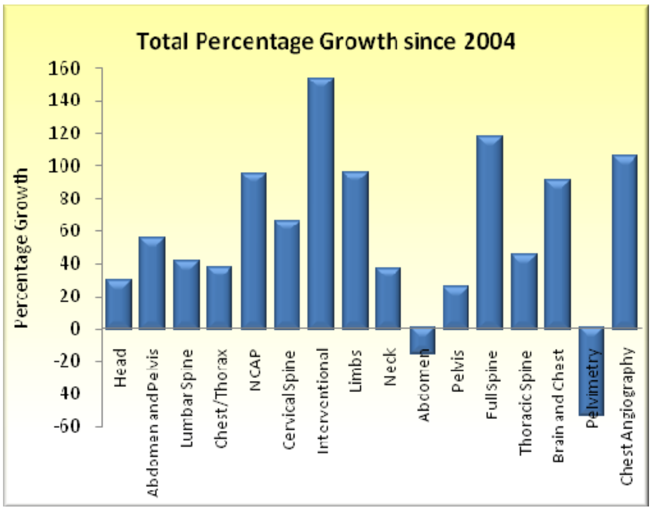

Fig. 5. Total percentage growth 2004-2009.

DOSE DATAMED: European Dose Survey

The European Commission has given advice on estimating the population dose from medical X-ray procedures (EC 2008). They use a list of "Top 20" most common procedures involving x-ray procedures. MDCT procedures make up 7 of this list of 20, they are – CT Head, CT Neck, CT Chest, CT Spine, CT Abdomen, CT Pelvis and CT Trunk. Low, average and high doses per procedure for each of these seven categories have been established via a survey of ten European countries. Thirteen of the sixteen current Medicare Australia Item number categories were grouped to form the seven CT categories as defined by DOSE DATAMED. The number of procedures performed in Australia in each of these categories from 2003 onwards was calculated. This data was then multiplied by each of the low, medium and high doses per procedure as obtained from the DOSE DATAMED Survey. Figure 6 shows this data combined with procedure number and dose level data from a previous 2002 ARPANSA survey. The data indicate that Australia would be considered a 'High' Dose level country. The population dose for Australia for 2009 based on the DOSE DATAMED method is represented in Table 1.

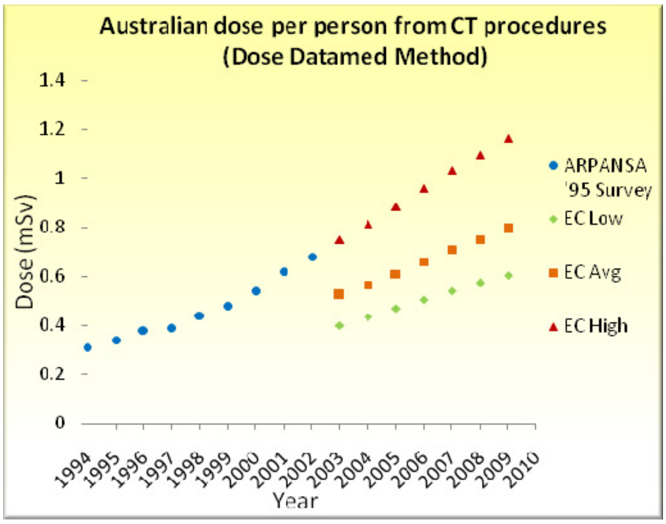

Fig. 6. Dose Per Capita calculated using the DOSE DATAMED method.

Table 1. CT procedure numbers and doses for Australia.

CT Procedure	Dose Level			Procedures 2009	Dose (mSv) (calculated using 'High' Dose Level)
	Low	Avg	High		
CT Head	1.6	2.0	2.4	776525	1863659
CT Neck	2.4	2.5	2.8	145247	406691
CT Chest	6.6	8.0	8.2	349025	2862002
CT Spine	3.6	5.3	6.0	399467	2396801
CT Abdomen	10.2	12.0	13.5	26577	358796
CT Pelvis	8.7	8.7	8.8	27168	239078
CT Trunk	10.4	14.0	24.4	710013	17324322
				Total Dose (mSv)	**25451349**
				Population	**21874920**
				Dose per Person (mSv)	**1.163**

In the Future

If current trends in MDCT procedure and population growth continue we can estimate that the annual dose from MDCT per person will reach 1.53 mSv in 2012. This will overtake the annual estimated natural background radiation dose of 1.5 mSv (ARPANSA 2010).

Discussion

The "Gross Up" Factor

In using the DOSE DATAMED method to calculate a dose per person from MDCT procedures in Australia we have used a single "gross up" factor to account for the procedures that are not captured by Medicare Australia. This "gross up" factor was obtained from a survey that was conducted in the mid nineties. The range of applications for CT scanning as a diagnostic imaging tool has dramatically increased over the past 16 years. This can, in part, be related to technological advances that have enabled faster, more complex, longer and interventional MDCT scanning to be undertaken. We expect the true “gross up” factor to also have changed in the past 16 years.

Our application of the DOSE DATAMED doses to our Medicare Australia numbers provides a calculated estimate of dose to the Australian population, it is not definitive. In aligning the Medicare Item numbers with the seven DOSE DATAMED categories there were a number of item numbers that were left out completely. These item numbers related to interventional CT, brain and chest CT, limbs CT and some item numbers relating to neck and cervical spine CT. Overall, the percentage of procedures recorded by Medicare Australia that have not been included in the seven DOSE DATAMED categories has been steadily increasing from 4% in 1994 to nearly 12% in 2009.

Despite Medicare Australia not providing definitive data, our generic calculation does provide a reasonable estimate of the population dose from MDCT procedures for Australia. The estimate of 1.2 mSv per person for 2009, and the current rate at which procedure numbers provide strong support for the implementation of a national survey to establish the true dose to the Australian population.

Conclusions

MDCT scans in Australia are increasing at approximately 9% per annum, which is more than six times greater than the growth in the population. On average more women are undergoing MDCT procedures than men. In 2009 more than 75% of all MDCT procedures were carried out on persons over the age of 55, with the most MDCT procedures per person occurring in the 75-84 year age group. MDCT scans of the head account for the largest portion of procedures but also has the second lowest positive percentage growth per year. The highest percentage growth per year was seen in the ‘interventional’ category of 31.7% which in 2009 only accounted for 3.4% of all procedures. Applying DOSE DATAMED dosimetry data to the calculation of population dose from MDCT procedures, Australia falls in to the “High Dose” category, which in 2009, resulted in an average dose to the population of 1.16 mSv per person.

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Comparison of spectroscopic investigation and computer modelling of lanthanide(III) and actinide(III) speciation in human biological fluids

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Abstract

Radionuclides represent a serious health risk to humans in case of incorporation. To get a first insight into the transport and metabolism in the human organism, we compared the speciation of Eu(III) and Cm(III) in human biofluids calculated by thermodynamic modelling with spectroscopic results obtained by time-resolved laser-induced fluorescence spectroscopy (TRLFS) with regard to the lanthanide/actinide analogy.

For TRLFS measurements, fresh human whole saliva and urine samples have been spiked in vitro with europium or curium. To identify the dominant species the measured spectroscopic data were compared to reference spectra with single organic and inorganic constituents of the biological fluids. The TRLFS spectra with saliva were all similar in the pH range between 6.8 and 7.4. The TRLFS spectra with urine showed that all samples with pH below and all samples with pH above 5.8 each exhibit strikingly similar spectra. Comparing the measured spectra with the reference data, we found that in urine at lower pH citrate complex species dominate the speciation of both metals while at higher pH and in saliva the spectra were identical to those of a mixture of inorganic and organic components.

The speciation calculation for saliva results for both metals in a predominant phosphate coordination with a small amount of coordination with citrate which reflects the experimental results not exactly.

The speciation calculation for urine shows up to pH 6.5 a dominating citrate speciation for both metals and only for higher pH values additionally phosphate coordination. However, spectroscopic investigations signed of a more complex speciation behaviour which cannot be described with single inorganic complexes.

To conclude, the comparison of experimental speciation investigation and computer modelling shows that because of their simplifications models cannot always image the complex natural processes correctly.

Introduction

In case of incorporation, radionuclides represent a serious health risk to humans due to their (radio-)toxicity. To understand their transport, metabolism, deposition and elimination in the human organisms it is crucial to determine their speciation and transport on a molecular level. Unfortunately, only little is known about the metabolism of trivalent actinides (An(III)). To address this lack of knowledge, in this study the speciation of curium (Cm(III)) and europium (Eu(III)) in saliva as the first contact medium at oral incorporation and urine as the main elimination medium was investigated spectroscopically with time-resolved laser-induced fluorescence spectroscopy (TRLFS). Additionally the speciation of the metal ions in the biofluids was calculated by thermodynamic modelling. Speciation calculation can be a helpful tool to simulate biokinetic processes when it is too risky to obtain experimental information (Webb et al. 1998). Another option is using non-radioactive analogues like trivalent lanthanides (Ln(III)) instead of An(III) to obtain experimental data. The aim of our study is to compare the results of both alternative methods, the speciation modelling and experiments with Eu(III) as a non-radioactive Ln(III) analogue with the experimental results of the speciation of the trivalent radioactive actinide Cm(III) in saliva and urine.

TRLFS is a useful tool to investigate the chemical environment of luminescence emitting ions like Cm(III) and Eu(III) at trace metal concentrations. The luminescence emission bands and the luminescence lifetimes provide information about the first coordination shell of the ions. For Eu(III), the emission bands $^5D_0 \rightarrow {}^7F_0$ (at 580 nm), $^5D_0 \rightarrow {}^7F_1$ (at 590 nm), and $^5D_0 \rightarrow {}^7F_2$ (at 615 nm) are of special interest. The first transition is formally forbidden and appears only under deformation of the first coordination shell of Eu(III). Whereas the magnetic dipole transition $^5D_0 \rightarrow {}^7F_1$ is hardly influenced by complexation, the relative intensity and the splitting pattern of the hypersensitive electric dipole transition $^5D_0 \rightarrow {}^7F_2$ show changes in the local environment of the Eu(III) species. The luminescence signal of the $^6D_{7/2} \rightarrow {}^8S_{7/2}$ transition of Cm(III) shows significant shifts with changes of the coordinative environment (Edelstein et al. 2006; Kim et al. 1991). From the lifetimes of both luminescent species, Eu(III) and Cm(III), the number of coordinating water molecules can be determined (Kimura and Choppin 1994).

Materials and methods

Sample collection

Unstimulated whole human saliva was obtained in absence of any gustatory or masticatory stimulation from healthy adult volunteers by spitting in a dry plastic tube. Samples (about 25 mL each) were collected at the same time of day (between 10:00 and 11:30 a.m.), at least one hour after the last intake of food or drink. Before using, the saliva samples were centrifuged (10 min, 4000 g).

Human urine samples were obtained from healthy adult volunteers as 24-hours-collecting-samples to eliminate fluctuations of the composition due to eating and drinking and used without further treatment.

A total of 5 saliva and 9 urine samples were investigated. All investigations were performed immediately (4-5 h), or otherwise saliva and urine samples were cooled stored at 4 °C (up to 24 h) or -20 °C (more than one day).

Analysis of the saliva and urine composition

The inorganic composition of all samples was determined using mass spectrometry with inductive coupled plasma (ICP-MS) and ion chromatography (IC). The content of cations was measured using the ELAN 9000 system from Perkin Elmer and the content of anions was determined using the Metrosep A Supp 4 column and a HPLC system from Metrohm. The total organic carbon (TOC) was measured with a multi N/C 2100 S from Analytik Jena AG. The pH values were determined with a BlueLine 16 pH electrode (Schott).

Speciation calculation

The calculation of the Eu(III) and Cm(III) speciation in saliva and urine was determined with the computer program MEDUSA (Make Equilibrium Diagrams Using Sophisticated Algorithms, version 2004), combined with the associated Hydrochemical Equilibrium-Constant Database HYDRA (version 2004). The stability constants for Eu(III) complexes were compared with the NAGRA database (Hummel et al. 2002) and, if necessary, readjusted. For the analogue Cm(III) complexes, the stability constants were taken from the NEA database (Guillaumont et al. 2003). If no stability constants for Cm(III) complexes are available (phosphate species), the values of Eu(III) complexes were applied.

Time-resolved laser-induced fluorescence spectroscopic measurements

For TRLFS measurements, the saliva and urine samples (3 mL each) were spiked with $3x10^{-5}$ M Eu^{3+} (Eu_2O_3, Aldrich) and $3x10^{-7}$ M Cm^{3+} (from a stock solution of the long-lived ^{248}Cm isotope in 1 M $HClO_4$; isotope ratio ^{248}Cm 97.3 %, ^{246}Cm 2.6 %, ^{245}Cm 0.04 %, ^{247}Cm 0.02 % and ^{244}Cm 0.009 %), respectively. All treatments with Cm(III) were carried out in a glove box.

The time-resolved laser-induced fluorescence spectra were recorded at 25 °C using a pulsed flash lamp pumped Nd:YAG-OPO laser system (Powerlite Precision II 9020 laser equipped with a Green PANTHER EX OPO from Continuum, Santa Clara, CA, USA). The optical parametrical oscillator (OPO) used to tune the wavelength of the emitted laser beam was pumped by the second harmonic oscillation of the Nd:YAG laser (532 nm). The doubled signal output wavelength can be varied between 330 nm and 500 nm. The laser pulse energy, which was between 1 and 2 mJ, was monitored using a photodiode. The fluorescence emission spectra were detected using an optical multi-channel analyzer-system, consisting of an Oriel MS 257 monochromator and spectrograph with a 300 or 1200 line mm^{-1} grating and an Andor iStar ICCD camera (Lot-Oriel Group, Darmstadt, Germany). The europium(III) and curium(III) emission spectra were recorded in the 440-780 nm (300 line mm^{-1} grating) and 570-650 nm (1200 line mm^{-1} grating) ranges. A constant time window of 1 ms was applied, and two excitation wavelengths, 395 nm (Eu) and 398 nm (Cm), were used. For time-dependent emission decay measurements, the delay time between laser pulse and camera grating was scanned with time intervals between 10 µs and 50 µs.

The TRLFS spectra were analyzed using origin7.5G. The single spectra were normalized to the same peak area (Cm) and of the $^5D_0$-$^7F_1$ transition peak (Eu), respectively. The lifetimes of luminescent species were determined according to the equation of exponential decay:

$$E(t) = \sum_i E_i \cdot \exp(-t/\tau_i) \qquad (1)$$

E is the total luminescence intensity at the time t, E_i the luminescence intensity of the species i at t, and τ_i the corresponding lifetime.

With the luminescence lifetimes the numbers of water molecules in the first coordination spheres of Eu(III) and Cm(III) were then calculated applying Kimura´s equations (Kimura and Choppin 1994):

$$nH_2O \pm 0.5 = 1.07 \cdot \frac{1}{\tau} - 0.62 \qquad \text{for Eu(III)} \qquad (2)$$

$$nH_2O \pm 0.5 = 0.65 \cdot \frac{1}{\tau} - 0.88 \qquad \text{for Cm(III)} \qquad (3).$$

Results

Analysis of selected components in saliva and urine

The composition of the main components of saliva and urine are summarized in Tables 1 and 2. We concentrated on the main cations sodium, potassium, magnesium (only urine) and calcium and the anions phosphate, sulphate (only urine), and carbonate as relevant potential complexation agents. The determination of chloride concentration was omitted because Eu- and Cm-chloride complexes are very weak and are not expected to play any role in the speciation.

The determined values are in good agreement with other studies on the composition of saliva (Edgar 1992; Rehak et al. 2000) and urine (Kirchmann and Pettersson 1995; Putnam and Thomas 1969).

Table 1. Concentrations of selected components of whole saliva.

Analyt	Range	Mean
Sodium (mmol/L)	2.98 – 6.57	4.00
Potassium (mmol/L)	3.35 – 21.46	9.39
Calcium (mmol/L)	0.99 – 1.14	1.03
Phosphate (mmol/L)	0.55 – 5.52	4.11
Carbonate (mmol/L)	3.10 – 8.00	4.87
TOC (g/L)	0.85 – 0.98	0.91
pH	6.78 – 7.43	7.05

Table 2. Concentrations of selected components of urine.

Analyt	Range	Mean
Sodium (mmol/L)	9.13 – 305.22	133.43
Potassium (mmol/L)	8.29 – 107.93	45.86
Calcium (mmol/L)	0.57 – 4.29	2.16
Magnesium (mmol/L)	0.70 – 6.54	2.42
Phosphate (mmol/L)	0.12 – 28.80	7.71
Sulfate (mmol/L)	0.08 – 16.03	4.97
Carbonate (mmol/L)	0.12 – 26.58	4.41
TOC (g/L)	2.24 – 7.86	4.71
pH	5.25 – 6.74	6.00

Additionally the total organic carbon was determined. It is well-known from the literature that the main organic substances in human saliva are proteins (e.g. mucin or enzymes). The major digestive enzyme of saliva, α-amylase, is present in concentrations of about 0.6-1.2 g/L (Edgar 1992). Citrate as the typical organic substance of many biofluids plays with a maximum of about 0.05 mmol/L (Zipkin and McClure 1949) only a minor role in saliva. In human urine, the main organic component is urea with about 0.2 mol/L (Putnam and Thomas 1969). An important organic substance for heavy metal complexation is citrate with an amount of about 3.2 mmol/L in urine (Putnam and Thomas 1969).

Speciation calculation of Eu(III) and Cm(III) in saliva and urine

For the computer modelling of the speciation of Eu(III) and Cm(III) in simulated saliva and urine, the mean values for inorganic components from Table 1 and 2 were used. Additionally mean values for citrate were included (0.05 mmol/L for saliva (Zipkin and McClure 1949) and 3.20 mmol/L for urine (Putnam and Thomas 1969)). The complexation of Eu(III) and Cm(III) with urea, the matrix substance of urine, was determined previously (Heller et al. 2009) and showed that the Eu(III) and Cm(III) urea complexes are very weak. Normally they do not influence the speciation of heavy metal ions in urine, therefore they were omitted. Also the complexes of Eu(III) and Cm(III) with various amino acids, which are present in saliva and urine, were investigated at neutral and slight acidic pH (Heller et al. 2010) and can likewise be omitted because they are also very weak.

The calculated speciation for Eu(III) and Cm(III) in simulated saliva shows at pH 7.0 a predominant coordination with phosphate (more than 80 %) with a small amount of citrate coordination. Figure 1 (left) shows the speciation diagram for Eu(III), the Cm(III) speciation looks similar. It is worth to mention that because for Cm(III) phosphate complexes no stability constants are known those of the analogue Eu(III) complexes were used for calculation instead. Therefore the Cm(III) speciation is doubtfully.

The speciation for Eu(III) and Cm(III) in simulated urine is pH dependent. In the pH range between 5.0 and 6.5 we observe nearly 100 % citrate coordination. At higher pH additionally phosphate coordination appears. Figure 1 (right) shows the speciation of Eu(III) in urine at pH 7.0. Here we observe still more than 50 % citrate coordination and about 40 % phosphate coordination. To Cm(III) speciation calculation in urine the same constriction does apply like to Cm(III) speciation calculation in saliva considering the stability constants of the phosphate species. The speciation calculation of Cm(III) in urine is therefore also doubtfully.

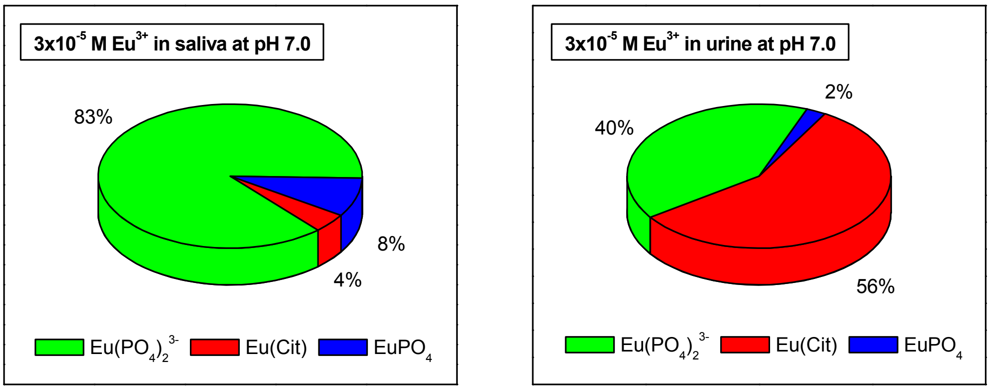

Fig. 1. Calculated speciation of europium in simulated saliva (left) and urine (right) at pH 7.0 with component composition of Table 1 (inorganic components) and additional citrate (0.05 mmol/L for saliva, 3.2 mmol/L for urine).

TRLFS measurements of Eu(III) and Cm(III) in saliva

A total of 5 measurements each of Eu(III) and Cm(III) in natural saliva samples with pH values in the range of 6.8 to 7.4 have been carried out. Figure 2 (left Eu, right Cm) shows that they are all similar. The luminescence decay of these mixtures is always mono-exponential for both, Eu(III) and Cm(III).

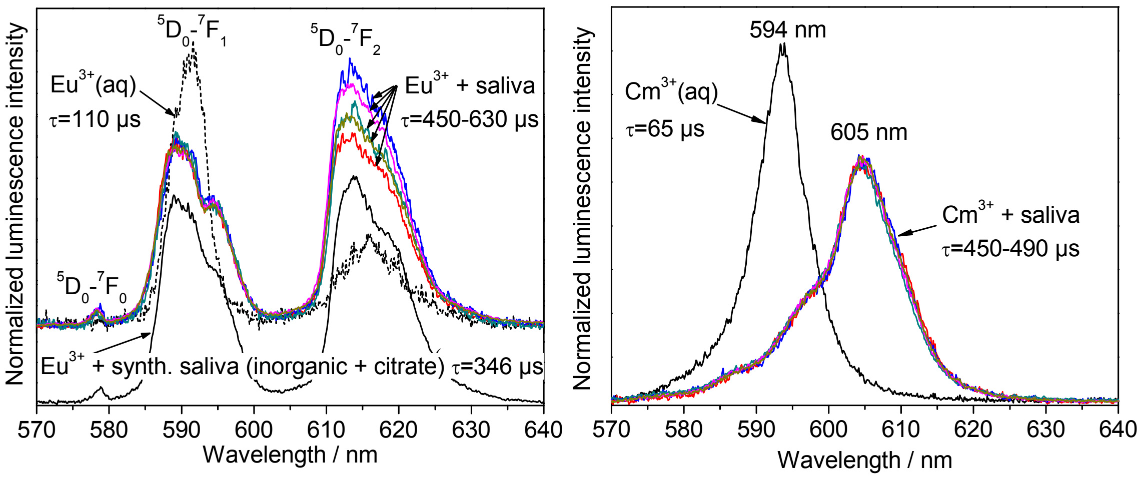

Fig. 2. Luminescence spectra of Eu(III) (left) and Cm(III) (right) in saliva.

In case of Eu(III) (Figure 2 left), the relative intensity of the $^5D_0 \rightarrow {}^7F_2$ transition is higher in comparison to the aquatic Eu^{3+} ion, both emission peaks, $^5D_0 \rightarrow {}^7F_1$ and $^5D_0 \rightarrow {}^7F_2$, are split and additionally the $^5D_0 \rightarrow {}^7F_0$ emission peak appears due to complexation with the saliva matrix. The luminescence lifetimes were determined to be between 450 and 630 µs (mean: 530 ± 75 µs) which corresponds to 1 - 2 remaining water molecules (applying Eq. (2)). For comparison, the Eu^{3+}(aq) ion has 9 water molecules in the first hydration shell. Consequently, 7 - 8 water molecules are exchanged through coordination sites of the saliva matrix.

For comparison, several synthetic saliva model solutions spiked with Eu^{3+} were investigated: with only the inorganic components according to Table 1 and with additionally the organic ingredients citrate, lactate and α-amylase, and also all single inorganic and organic components. Only the synthetic saliva with inorganic components plus citrate fits the static spectra correctly (see Figure 2 left) and fits additionally nearly the luminescence lifetime. All other tested models (spectra not shown here) show either different static spectra or considerably shorter lifetimes or even both.

The luminescence spectra of Cm(III) in human whole saliva are depicted in Figure 2 (right). The complexation of the Cm^{3+}(aq) ion with saliva causes a strong red shift of the luminescence signal of about 12 nm from 594 nm (Cm^{3+}(aq)) to 605 nm. The luminescence lifetime increases from 65 µs (Cm^{3+}(aq), according to 9 water molecules) to 450 – 490 µs (mean: 474 ± 16 µs), which corresponds to 1 remaining water molecule. Similar to Eu(III), about 8 water molecules of the first hydration shell of Cm(III) are exchanged through ligand coordination sites of the human whole saliva matrix.

Just like for Eu(III), for comparison several synthetic saliva models with only inorganic components and additionally with organic components and also the single inorganic and organic components, spiked with Cm^{3+}, were determined. In contrast to Eu(III), the synthetic saliva model with inorganic components plus citrate does not fit the spectra of human whole saliva, neither the peak maximum (600 nm) nor the luminescence lifetime (90 – 120 µs). Only with additionally α-amylase the peak maximum (603 nm) and luminescence lifetime (300 – 610 µs) fit nearly the values of human whole saliva.

TRLFS measurements of Eu(III) and Cm(III) in urine

Figure 3 depicts the luminescence spectra of Eu(III) samples in human urine at different pH values and also in several model solutions for comparison. In both cases, the spectra are different above and below pH 5.8. Like for saliva, also for the urine samples synthetic urine models as well as several single inorganic and organic components were investigated for comparison. Below pH 5.8 the static spectra and luminescence lifetimes are similar to those of the citrate complexes of Eu(III) (see Figure 3) and Cm(III) (peak maximum: 600 nm; lifetime: 121 µs). Above pH 5.8 the Eu(III) urine spectra corresponds to synthetic urine with only inorganic components (Figure 3). For Cm(III), neither the inorganic synthetic urine (peak maximum: 596 nm, lifetime: 70 – 80 µs) nor the model of inorganic synthetic urine with citrate (peak maximum: 600 nm, lifetime: 90 – 120 µs) fits the spectral data for human urine (see Figure 4). Therefore, no assured assignment is possible at the moment.

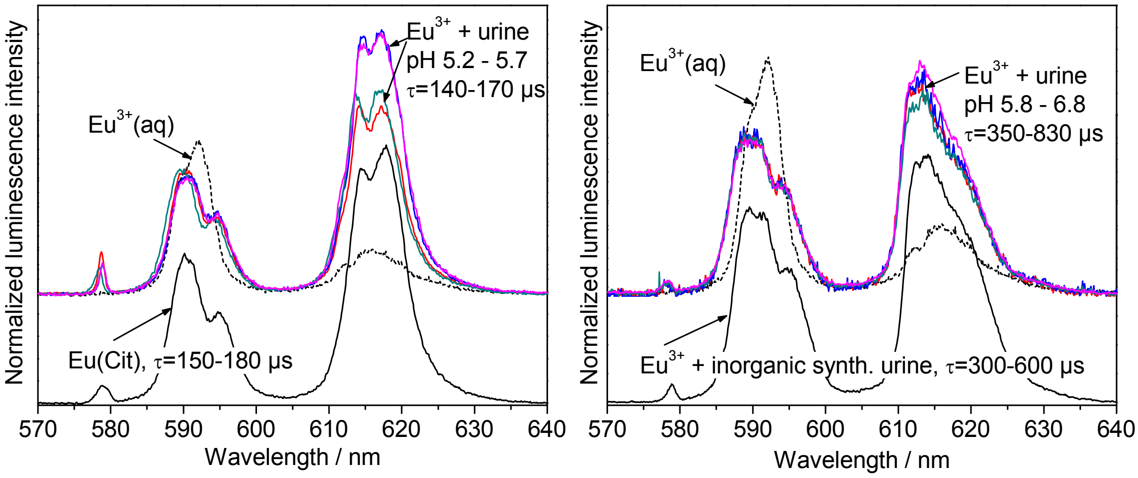

Fig. 3. Luminescence spectra of Eu(III) in urine and related model spectra. Left: Urine samples with pH values 5.2-5.7 and for comparison Eu-citrate. Right: Urine samples with pH values 5.8-6.8 and for comparison inorganic synthetic urine.

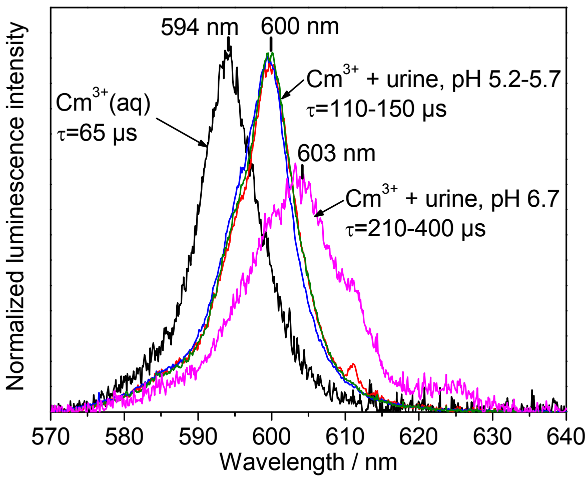

Fig. 4. Luminescence spectra of Cm(III) in urine.

Discussion

The comparison of speciation calculation of Eu(III) and Cm(III) in human whole saliva and human urine with the spectral data of Eu(III) and Cm(III) in original human whole saliva and human urine samples shows some analogies but also some discrepancies.

For saliva, the speciation calculation predicts predominant phosphate coordination with a small citrate fraction. The spectral analysis of the complexation of Eu(III) in human saliva in comparison to several synthetic saliva models and single saliva components results in a more complicated speciation. Beside a small citrate fraction not only phosphate coordinates the Eu^{3+} ion, but a mixture of inorganic anions. For Cm(III), even more organic components like α-amylase seems to be involved in complexation.

For urine, in the pH range between 5.2 and 5.7 the spectral results support the speciation calculation. Both trivalent metal ions are completely complexed with citrate. But in the pH range between 5.8 and 6.8, the results from spectral analysis differ strongly from the calculated speciation. The speciation calculation predicts even at pH 7 predominant citrate coordination and less than half phosphate coordination. In contrast, spectral data show for Eu(III) above pH 5.8 a mixed coordination with only inorganic anions present in urine. For Cm(III) a mixed coordination with inorganic and organic anions seems to be present.

The discrepancies between theoretical and practical results are due to a big lack in data for complex stability constants of single components, especially for Cm(III). Furthermore, no data about the stability constants of mixed ternary or higher complexes with two or more inorganic and/or organic anions are available for Eu(III) and Cm(III). One can assume that such ternary or higher mixed complexes are present in biofluids with a high variety of different inorganic and organic anions for metal ion complexation.

Conclusions

Time-resolved laser-induced fluorescence spectroscopy (TRLFS) is a useful tool to investigate the complexation behaviour of metal ions in biofluids. The spectroscopically determined speciation of Eu(III) and Cm(III) in human saliva and urine showed discrepancies to the computer modelled speciation due to a lack of complex stability constants, especially for Cm(III), and for ternary or higher mixed complexes for both metal ions. For better speciation calculation further investigations to determine complex stability constants are necessary.

The comparison of the complexation of Eu(III) as a selected trivalent lanthanide Ln(III) with Cm(III) as an example for a trivalent actinide An(III) shows less analogies than expected. Especially at nearly neutral pH the speciation in saliva and urine each differ between Eu(III) and Cm(III). This can possibly be due to the different metal ion concentrations used. However, this comparison shows that a prediction of the complexation behaviour of An(III) on the basis of results with Ln(III) should only be used as rough estimation. For precise conclusions, selected experiments with An(III) are necessary.

Acknowledgements

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Natural radioactivity of building materials: Radiation protection concepts, measurement methods and regulatory implementation

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Abstract

Most building materials, especially of mineralogical origin, contain natural radionuclides. Generally radioactivity in building material originates from radionuclides of the natural decay chains U-238, Th-232, U-235 plus progenies as well as the primordial radionuclide K-40. External radiation exposure by gamma-radiation and beta particles emission plus internal exposure due to inhalation of radon cause chronic exposure of the public. Due to increasing indoor habitation of persons – in average about 80% of time persons stay indoor – external and internal exposure caused by natural radionuclides in building materials are of increasing importance. Additional the rising production of building materials using industrial by-products and residual materials from NORM industry necessitate the consideration and regulation of this radiation protection issue in regard to chronically public exposure. In this paper historical and recent basic concepts of dose estimation, radioactivity measurement methods including metrology and calibration of instruments are given. Examples of building material evaluations are discusses with regard to approaches of regulatory standards and experiences with implemented legal guidelines and provisions. The result of this research could act as effective fundamental basis for harmonised guidelines in Europe to limit chronically public exposure due to natural radionuclides in building materials.

Mapping of aerosol releases from Forsmark nuclear power plant

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Abstract

Forsmark NPP has set a goal to reduce discharges of radioactive aerosols, mainly airborne particles of corrosion products such as cobalt-60, ^{60}Co, by 50% by 2011. To reach this goal we have run a project to map when and where aerosol discharges take place. We started by correlating aerosol discharges, measured in the stack, with events, such as reactor pool cleaning, in the plant. This has given us indications regarding which jobs release more aerosols. To further clarify the relationships of from which events and locations aerosol discharges emanate we have developed an instrument that measure aerosols in ventilation ducts, using the same millipore/coal filter cassettes that are used in the stack monitoring system. Our approach has been to confirm findings from the event-mapping project and further clarify which sections of the ventilation system that transport most radioactive aerosols.

By analyzing data from the event- release correlation study and the duct-measurement study we conclude that pool cleaning is the single major source of aerosol discharge and by introducing new work methods and air filtering systems during identified jobs aerosol discharge should be significantly reduced. We have also identified that the majority of aerosols comes from the reactor building, especially when small leaks occur - more optimized use of the filter banks might reduce the discharge of radioactive aerosols. In the end this will help lower the discharges of radioactive substances from Forsmark NPP making clean energy even cleaner.

Introduction

Even though nuclear power is a clean energy source, small discharges of radioactive substances to air and water occur. These discharges contribute only negligibly to the normal background dose. At Forsmark NPP the dose from the background radiation is estimated to be 1 mSv/y and Forsmark contribute to this dose with approximately 0.0002 mSv/y.

95-99% of the dose from the discharge of radioactive substances comes from ^{14}C produced in the cooling media during operation. We are currently unable to reduce this contribution to the dose since there are no suitable carbon-capture options available. The remaining 1-5% of the dose is made up of radioactive noble gasses, iodines and aerosol particles, mainly made up of radioactive corrosion products such as ^{60}Co. All discharges to air are measured in the stack monitoring system. At unit 1 and 2 (F1 and F2), all discharges go through the main stack, at unit 3 (F3) there are additional separate stacks for the waste department and the active workshop.

Forsmark NPP, striving to reduce our environmental impact, has a goal to reduce discharges of radioactive aerosols by 50% by 2011. To reach this goal we have performed a study to map when and where aerosol releases take place, this to find out how to avoid and reduce discharges.

Material and methods

To map aerosol discharges from Forsmark NPP we have used two tracks:

Our first approach has been to correlate aerosol discharges, measured in the stack, with events in the plant. We have looked at the years 2006-2009. From the stack monitoring system we have gathered monthly and weekly readings on the discharge of aerosols, which we have compared to listed events. To know what to look for in log-books and listings we compiled a list of possible events, which we then compared with discharges on a monthly basis. If a closer look was needed the weekly readings were used. This approach has given us indications regarding which jobs release more aerosols. We have chosen to concentrate on the release of the corrosion product ^{60}Co since this is the most prevalent nuclide and also the most dominant contributor to the dose from aerosol discharge.

Our second approach has been to develop an instrument that measures aerosol content in ventilation ducts. Ventilation air is sampled through a nozzle inserted in the duct and led through a filter cassette using an isokinetic flow. Aerosols are collected on a Millipore filter, the same type of Millipore/coal filter cassettes that we use in the stack monitoring system. The cassette has been connected to the airflow in the duct for approximately one week after which the Millipore filter was analysed for aerosol content using gamma spectrometry. This approach was developed to be able to verify the findings in the correlation study. We have only recently started the measurement program so more detailed data should come later.

By analyzing data from the event- discharge correlation study and the duct-measurement study we should be able to optimize where new aerosol filters could be introduced and point out which jobs can benefit from actions to help reduce the release of radioactive aerosols.

Results and discussion

Correlation aerosol discharge – events in the plant

Below are figures of the monthly discharges of ^{60}Co through the stacks at each unit. Red rectangles symbolizes months when planned outages have taken place. For F2 and F3 this correlates with high aerosol discharges (figure 1 B,C), for F1 the discharge occurs more uniformly over the year (figure 1A). For F2, major remodelling work occurred late 2009 where major components in both the reactor and turbine plant were replaced. This rendered, not unexpectedly, quite a large release of aerosols.

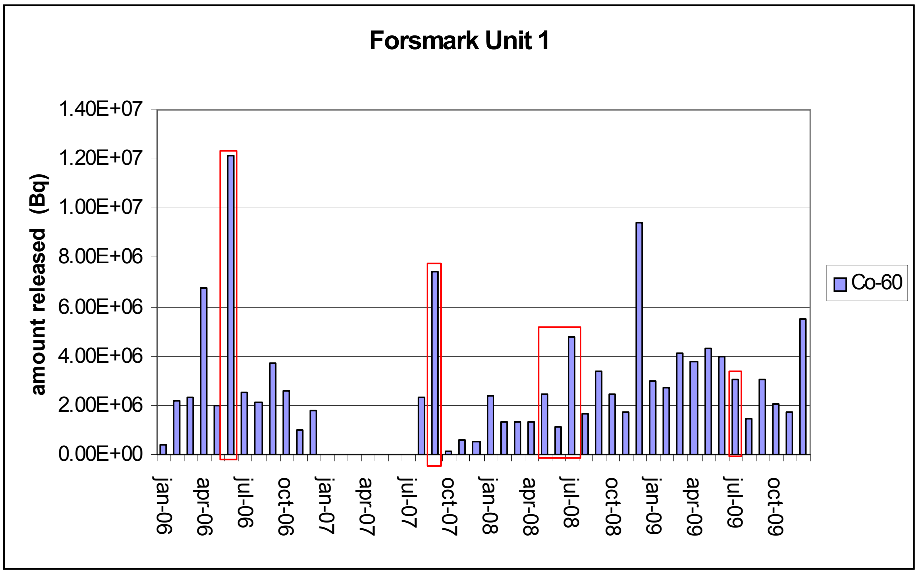

Figure 1A. Discharges of ^{60}Co from Forsmark 1.

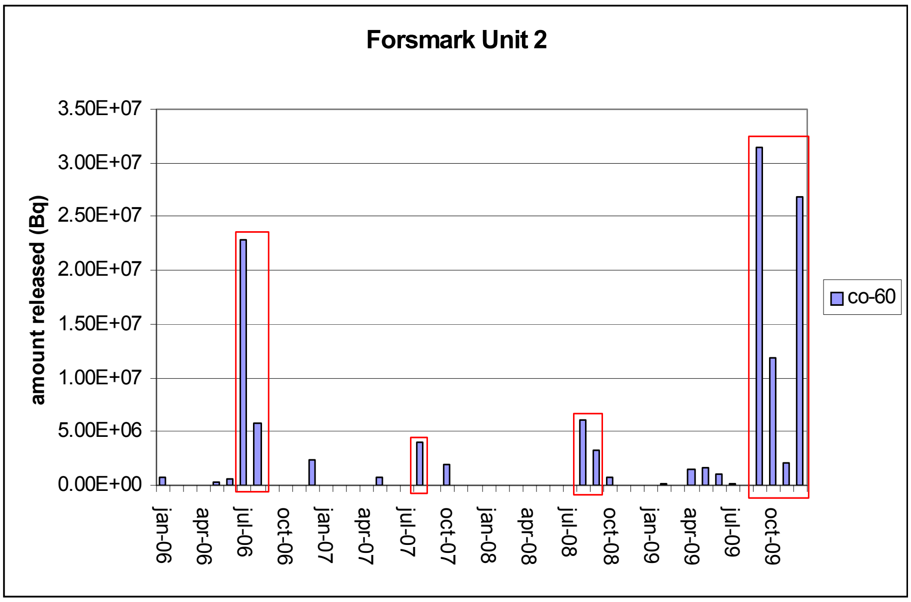

Figure 1B. Discharges of ^{60}Co from Forsmark 2.

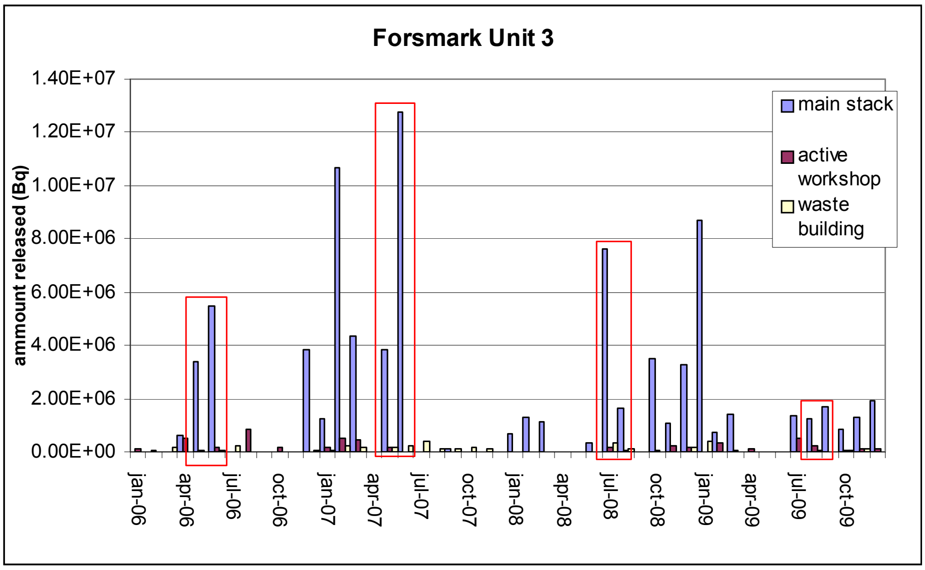

Figure 1C. Discharges of ^{60}Co from Forsmark 3.

Pool cleaning

During each outage the reactor-pool is emptied and cleaned before opening the reactor and before closing. These actions have been assumed to result in aerosol release. Cleaning of pools, some times using high-pressure washers, result in particles collected on the walls being released into the air of the hall. No filtration of the ventilation of the hall takes place before discharge through the main stack. In the correlation study it is shown that aerosols are discharged during outages, pool cleaning and opening of the reactor probably contributes to a large extent.

It is not only the reactor-pool that might give rise to aerosol release during cleaning. The fuel-flask pool is sometimes used as a waste pool where contamination ends up after work is done on control-rods, fuel elements and internals. Cleaning of the flask pool is shown to give rise to aerosol discharge if there have been aerosol generating work going on in the pool. The high peak at F3 in januari 2009 is an example of that. It probably comes from cleaning of the flask pool after control-rod segmentation during the latter part of 2008.

Ion exchange columns - flushing

Flushing of ion exchange columns was prior to more detailed study thought to be a source of aerosol release. No correlation has however been found and this might be attributed to the fact that the ventilation from the flushed tank is going through particle filters before being discharged into the air.

Waste handling

The more continuous discharge of aerosols from F1 was earlier thought to be attributed to the fact that ventilation from the waste department for solid waste went through the F1 stack but the results from our measurements in the ventilation canals (se below) show that no significant aerosol discharge emanates from the waste department. The ventilated air is filtered before the ventilation canal and it appears that the filtration works as planned.

Leakages

One other significant source for aerosol release is when leaks in the primary system occur. Leaks in the steam system might also contribute to aerosol discharge but only to a lesser extent since carry-over of aerosols to the vapour phase is very low. If a leak appears in a system carrying warm pressurized water in contact with the reactor, the possibility to steer the ventilation from that room to filter banks exist. This is not very often done with small leaks mainly because of the difficulty to identify when and where small leaks occur. The new measurement tool described above might help improve this work; it has recently been used to identify a leak in F1. Until the leak can be fixed the ventilation is now steered to the filter banks and thus the discharge is reduced.

Aerosol measurements

We have used the aerosol measurement instrument to determine from which part of the ventilation system most aerosols come. So far we have tested ventilation from the waste department at F1, the unfiltered ventilation from the reactor building at all units, and the waste department at F3. One instrument has also been used to trace a leak to a certain room at F1.

Waste department at F1

Contrary to our previous beliefs it seems as though no aerosols emanate from the waste department. All air is filtered before entering the ventilation canals and this filtration appears to work satisfactory.

Reactor building F1

The nuclides ^{60}Co, ^{58}Co and ^{51}Cr were detected on the filters sampling unfiltered air from the reactor building (Figure 2A), and the ratio between the nuclides correlates very well with the ratio between the same nuclides discharged in the stack (figure 2B). Since the nuclide distribution is slightly atypical, we can conclude that the majority of the aerosol discharge during the period at F1 has come from the reactor building, probably from the small leak recently identified.

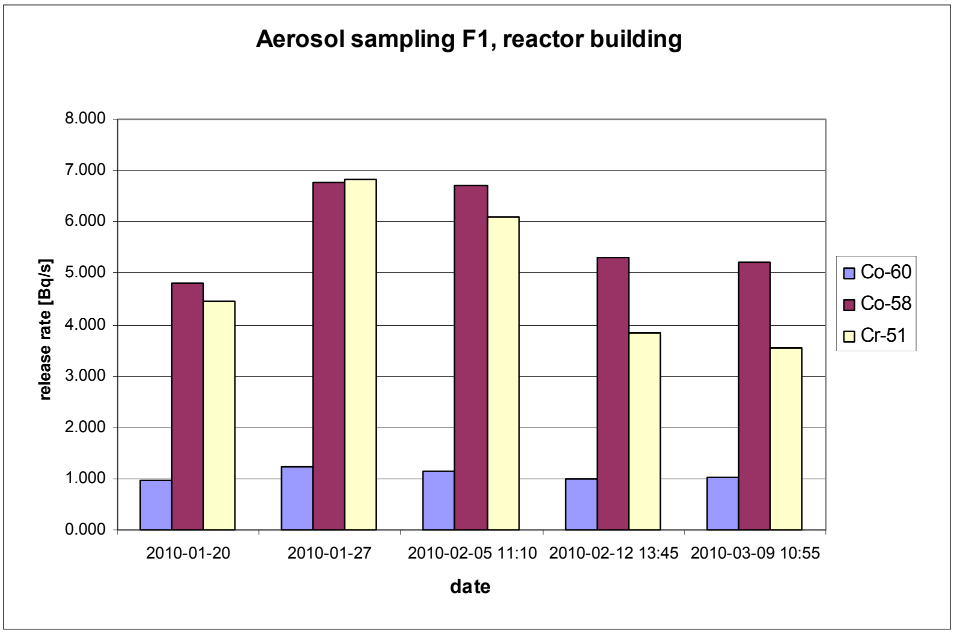

Figure 2A. Aerosol Sampling at the ventilation ducts from the reactor building at Forsmark 1.

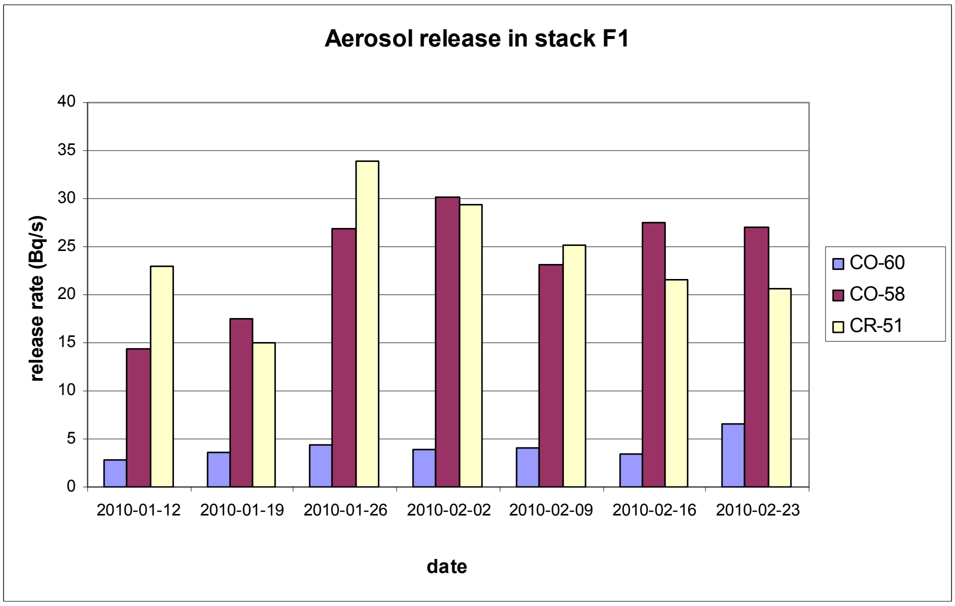

Figure 2B. Aerosol discharge from Forsmark 1.

Reactor building F2

No aerosols have been detected on the filters, which is expected since there have been no detectable aerosol discharge in the stack during the period.

Reactor building F3

Only very small amounts of ^{60}Co have been detected, which correlates with low discharges in the stack during the period.

Waste building F3

No aerosols were detected.

It has been a slight disappointment that we only detected aerosols on two instruments, but we can attribute this to the fact that there have been very low aerosol discharges from F2 and F3 during the period, which of course is positive

Conclusions

We have so far identified that pool-cleaning is a major contributor to aerosol discharge and will change the way we perform this task to reduce the release of aerosol-particles to the ventilation system. This will probably include the use of local air filters and some sort of "vacuum cleaning" of the pool walls.

It is also important that small system leaks are identified and fixed promptly, if this is not possible then ventilation should be steered to filter banks to make sure that no more elevated release of aerosols than necessary reach the environment.

It is good to note that no aerosol discharge seems to come from the waste department at F1. This is proof that the filter-banks work as intended. It is of utter importance that it is ensured that filter banks in the ventilation systems have a good filtering capacity.

We have now finished the initial study and have identified some actions that will be implemented to help lowering aerosol discharges but the work to map the ventilation system with regards to aerosols will continue to further improve our knowledge in how aerosols are produced and discharged.

A nation wide survey on drinking water radioactivity in Estonia: facts, risk assessment and remedial actions

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Abstract

Within the frame of a Twinning Light Project sponsored by European Union, a cooperative study between Estonia and Italy has been carried out to assess the actual radiological exposure of Estonian population due to the natural radioactivity of drinking water. Actually, about 22% of people is exposed to doses higher than 0.1 mSv/y (reference value for the Total Indicative Dose according to EU Directive 98/83) due to high concentrations of Ra-228 and Ra-226.

Estonian authorities were facing the problem of finding a feasible solution, but both a comprehensive overview of the situation and a deep cost benefit analysis were still missing.

Available radiometric data (nearly 1,000), as provided by Estonian institutions, were fully analysed in order to assess their representativeness and to identify the most critical situations. Though highest radioactivity values were found in the North of Estonia, where Cambrian Vendian waters are used, the few available data for the remaining part of the country cannot definitely assure compliance of drinking water, as previously supposed. On this basis, a new monitoring program has been proposed and the most suitable analytical tools have been suggested.

The effectiveness in radium removal of presently operating Fe/Mn treatment plants has been assessed. Since the resulting removal efficiency (about 15 %) was inadequate, a thorough cost benefit analysis of commercially available treatment plants for radium removal has been carried out, the main constraint being the small size of Estonian aqueducts (59 % serving less than 500 people).

Production and management of radioactive wastes and effluents in water treatment plants was examined on the basis of the technical documents issued by EU and IAEA (RP 122 and 135, IAEA RS-19).

A review of existing international guidelines was provided, and regulatory proposals were given in the light of proper risk assessments.

Regularities of long-term changes in artificial radionuclides content in the Barents Sea ecosystem

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Abstract

The comparative analysis is performed for the radiation contamination of the Barents Sea ecosystem in the 1980s and 1990s and in the first decade of the 21st century. Natural purification processes in the marine environment are the main factors of the decrease in the intensity level of artificial radioactive isotopes. These processes include repeated dilution, nuclear decay, sorption by sediments and suspended solid material, and accumulation by aquatic inhabitants. A stable decreasing trend is observed for the intensity level of artificial radioactive isotopes in the Barents Sea.

Introduction

The phenomenon of artificial radioactivity has been observed for the natural environment since the first nuclear tests in 1940s and continues at present. The disposal and dumping of both solid and liquid nuclear waste were typical for the Arctic and the northern seas. Radiological hazard sites can be observed in the coastal area, which are concentrated in Kola and Motovsky bays; the city of Murmansk, as well as towns: Severomorsk, Polyarny, and Gadzhievo; and the following fjords: Saida, Olenya, Pala, Zapadnaya Litsa, Ura, and Ara. Gaseous, liquid, and solid nuclear wastes are being stored there. The European reprocessing plants are the most significant producers of nuclear waste in the Arctic Ocean. The Sellafield factory (Great Britain) is the biggest one amongst them. The total activity of radionuclides discharged by these plants was 160×10^{15} Bq. Around 10–20% of ^{137}Cs discharged in Sellafield and around 30% of ^{90}Sr enter the Barents Sea. The plume of west European discharges crosses the shelf area of the Barents Sea and reaches the Central Polar Basin in six years (Nikitin et al., 1991; Nies, Nielsen, 1996; Matishov, Matishov, 2004). The input of Chernobyl accident provided from 10 to 20% of the radioactive pollution of the Kara and Barents seas in the 1990s.

The concentration of radioactivity has decreased an order of magnitude in the Arctic marine ecosystems from the 1960s to the 1990s as a result of water self-purification, natural decay of ^{90}Sr and ^{137}Cs, and the reduction of nuclear waste dumping. At the same time, with the time passed after the Chernobyl accident and the moment nuclear tests on the Novaya Zemlya and adjacent shelf were ceased, it is reasonable to follow the evolution of anthropogenic radioisotopes in the first decade of the 21st century. Of vital importance are also the problems of radiation safety, ecological standardization and forecast of the long-term effect of low doses on the marine biota. Not less important is the forecast of ecological security in case of nuclear energy installations' application while oil and gas development and extraction on shelf, in particular, at the Stockman gas condensate deposit.

Murmansk Marine Biological Institute of the Kola science center of the Russian academy of sciences (MMBI KSC RAS) has been carrying out regular monitoring of artificial radionuclides in the environment and organisms of the West Arctic seas since 1990 (Matishov 1994, 1995, 1996; Smith et al. 1995; Matishov, Matishov, 2004). In 1992-1999 radiation monitoring was carried out by MMBI in co-operation with Radiation and Nuclear Safety Authority (STUK, Finland). This co-operation got new development in 2006-2007.

Material and methods

Materials for radioecological investigations were collected during both coastal and marine multi-disciplinary expeditions of MMBI KSC RAS.

Preliminary concentration of cesium isotopes from water samples (100 l volume each) were performed by the cellulose-inorganic sorbent *ANFEZH*. The measurements of ^{137}Cs, ^{40}K, and ^{226}Ra activity were made by the *Canberra* gamma-spectrometer. To determine the specific activity of ^{90}Sr oxalate-radiochemical preparation was applied with subsequent measurement of counting sample at the *Beckman LS-6500* alpha-beta-scintillation counter by Cherenkov radiation of daughter radionuclide ^{90}Y. The anion exchange method was applied to analyze the Pu isotopes, when ^{242}Pu was used as a tracer agent. To determine Polonium samples were treated by tracer ^{209}Po and isotope ^{210}Po was deposited on a silver disk (Vesterbacka, Ikäheimonen, 2005). The activity of ^{238}Pu, ^{239,240}Pu, and ^{210}Po isotopes was measured by the alpha-spectrometer *Canberra Alpha Analyst.* The standard *Genie-2000* software was used for spectrum analysis.

The gamma-emitting isotopes and ^{90}Sr activity were measured at the certified radioecological laboratory of MMBI KSC RAS. The ^{238}Pu, ^{239,240}Pu isotope activity was measured at the laboratory of STUK. The inter-calibration, which gave good concordance of results obtained at two laboratories, was carried out during the joint activities.

Results and Discussion

The radioecological pattern of the Barents Sea at the end of the twentieth century was quite patchy. The ^{137}Cs activity concentration in the waters was relatively high at the end of the 1980s, from 10 to 40-90 Bq/m^3. The concentration of this isotope decreased in the middle of the 1990s an order of magnitude and did not reach more than 2-15 Bq/m^3. The content of ^{137}Cs in the waters of the Zapadnaya Litsa Fjord was 4.9 Bq/m^3, of the Ara Fjord – 4.7 Bq/m^3, in the entrance of the Kola Bay to the Barents Sea – 8 Bq/m^3. The volumetric activity of ^{90}Sr varied from 1 to 7 Bq/m^3 in the 1990s. The ^{239,240}Pu activity decreased from 13-18 to 6-11 Bq/m^3 from the 1980s to the 1990s. The waters close to the radioactive waste treatment plant RTP «*Atomflot*» were enriched by ^{239,240}Pu of 18-70 Bq/m^3 at the end of the 1990s (Matishov et al. 1996, 1997; Matishov, Matishov, 2004).

The specific activity of ^{137}Cs in the bottom sediments of the Barents Sea varied from 10 to 30 Bq/kg dry weight. The coastal areas of Spitsbergen, the Kola Peninsula, Novaya Zemlya, Franz Josef Land, and the shallow waters of the Pechora Sea were characterized by lower activity of ^{137}Cs, from 0.6-3.0 to 6.0 Bq/kg. Relatively higher concentrations of ^{137}Cs (5-11 Bq/kg) are typical for bottom sediments in all troughs and trenches in the open areas of the shelf. Clayish sediments covering the bottom of the Central Trench (300-380 m deep) contained 5-9 Bq/kg of ^{137}Cs and 0.2-0.8 Bq/kg of ^{90}Sr (Matishov et al. 1994, 1995, 1996; Matishov, Matishov, 2004). ^{239,240}Pu was found everywhere in the sediments of the Barents Sea shelf, from 0.1 to 20 Bq/kg. The distribution of this isotope depends on the sediment type and geographical distance to the nuclear weapons range. The silts of the Central Trench contain from 0.9 to 3.2 Bq/kg of ^{239,240}Pu. The concentration of this isotope is quite low in sands and aleurites of the shallow waters (0.1-1.0 Bq/kg); it may increase up to 3.2–4.3 Bq/kg in some trenches and downwarpings, and it is as high as 13 Bq/kg in the South-Novozemelskaya Trench to the south of Chernaya Guba (Matishov, Matishov, 2004). The bottom sediments of the Zapadnaya Litsa, Saida, Pala fjords (Kola and Motovskii bays) are characterized by a concentration of 2 to 9 Bq/kg of ^{239,240}Pu, and higher concentrations of ^{137}Cs (30-120 Bq/kg) and ^{60}Co (from 4-10 to 80 Bq/kg) (Matishov et al. 1996; Matishov, Matishov, 2004). These fjords are the nuclear-power submarine depot complexes.

Macroalgae, mostly *Fucus* spp. and *Laminaria* spp., accumulate the isotopes in the coastal waters. The maximal activity of ^{137}Cs was found in the early 1980s (up to 10 Bq/kg) and after the Chernobyl event (up to 7 Bq/kg) (Matishov 1995; Matishov, Matishov, 2004). In the 1980s and 1990s, the macrophytes contained up to 0.4-5.0 Bq/kg of ^{137}Cs, 0.3-3.0 Bq/kg of ^{90}Sr, and 0.02-0.3 Bq/kg of ^{239,240}Pu. The local contamination was focused on the water area close to the vessel «Lepse» belonging to RTP «Atomflot». This vessel is used as a spent-fuel storage; the concentration of isotopes in the macrophytes here was as high as 20–46 Bq/kg of ^{137}Cs, 1.2 Bq/kg of ^{134}Cs, 1.6 Bq/kg of ^{60}Co, and 4.6 Bq/kg of ^{152}Eu (Matishov, Matishov, 2004). However, this contamination is several orders lower than that observed in other areas after the Chernobyl event.

To ascertain the contemporary degradation regularities of anthropogenic radioactive substances, the Barents Sea water, bottom sediments, and algae – bio-indicators of radioactive contamination – were analyzed in 2005-2009. It is quite obvious that a natural decrease of artificial radionuclides' content is typical of all elements of marine ecosystem for the last decade (Matishov et al. 2004, 2005, 2007). Obviously, all the ecosystem elements are characterized by decrease in the anthropogenic isotope concentrations for the last decade.

The volumetric activity of the radionuclide ^{137}Cs in the Barents Sea in 2005-2007 is quite equable for all the investigated areas. The isotope activity is low and varies from 1-2 to 3.6 Bq/m^3, decreasing an order of magnitude compared to the 1970s. The maximal ^{137}Cs content of 6.5 Bq/m^3 was registered in Kola Bay waters near the area of RTP «Atomflot» outfall. Relatively high concentrations were found in the coastal area on the way of the Murmansk costal current. The activity of ^{137}Cs in Dalnezelenetskaya and Teriberskaya bays varied about 3.4-3.6 Bq/m^3. The activity of the same isotope constituted about 2 Bq/m^3 in the waters off the Novaya Zemlya western coast and, thus, was close to average for the Barents Sea. The ^{90}Sr comes from the Norwegian Sea by the warm currents, according to the data of 2005-2009. The Atlantic warm waters are characterized by increased concentration of ^{90}Sr, reaching up to 6-8 Bq/m^3. The concentration of ^{90}Sr is several times lower and is about 1-4 Bq/m^3 in the south-eastern sector of the Barents Sea, where the waters are already transformed. The activity of this isotope is 1.2-3.9 Bq/m^3 along the Novaya Zemlya coast.

The present-day contamination level of the bottom sediments in the Barents Sea is quite low. The average specific activity of ^{137}Cs is 1-3 Bq/kg and ^{90}Sr is 0.2-2.0 Bq/kg in the bottom sediments for the period of 2005-2009 (Fig 1a). The predictable activity of the isotopes is usual for all the bottom sediments of the Barents Sea with some exceptions for the coastal areas. The minimal activities of ^{137}Cs (0.5-2.9 Bq/kg) and ^{90}Sr (0.1-2.0 Bq/kg) were found in the open areas of the sea, where quartz is the main rock-forming mineral. Quartz is characterized by a low sorption capacity. The local increase of ^{137}Cs activity (5-8 Bq/kg) was observed in troughs near the Franz Josef Land. The bottom sediments there are mostly formed by silts. The bottom sediments off the Novaya Zemlya coast contained up to 4.7 Bq/kg of ^{137}Cs. The specific activity of ^{238}Pu in bottom sediment along the standard section No. 6 was 0.02-0.04 Bq/kg. ^{239}Pu activity was 0.4-1.3 Bq/kg. The analyzed ratio ^{238}Pu/^{239}Pu in the sediment samples showed that the plutonium is from global fallout from the atmospheric nuclear weapons testing.

The main part of system radioecological studies is the establishment of balance mathematical models. The obtained data on radionuclide distribition in waters was applied to calculate the current balance of artificial radionuclides in the Barents Sea. The calculation results show that the input of ^{137}Cs and ^{90}Sr is largely determined by the radionuclides exchange on the western boundaries of the Sea, where the resulting transfer is always directed from the Norwegian Sea to the Barents Sea. The contribution of atmospheric fallout and river runoff into the total radionuclides input is negligible at present as compared to radionuclide input with the Atlantic waters which is about 64 TBq/year of ^{137}Cs and 271 TBq/year ^{90}Sr.

The bottom sediments sampled from the station on the north-western slope of the Murmansk Bank, which is affected by the Norwegian Current, were characterized by the presence of ^{125}Sb (half-life period 2.77 years). This anthropogenic radionuclide, together with ^{99}Tc, is usually associated with liquid discharges of the nuclear industry plants. Its presence in the Barents Sea indicates that a small amount of radioactive wastes of the Sellafield Plant comes with Atlantic waters.

Some of the Barents Sea coastal areas, including Kola Bay, are characterized by increased concentration of the anthropogenic radionuclide in the bottom sediments. These areas belong to the local sources of pollution, such as numerous depot complexes of atomic submarines and icebreakers, the places of the moorings and waste reclamation of these vessels, and radioactive waste storage sites. Bottom sediments within the territory of RTP «Atomflot» are characterized by the highest levels of ^{137}Cs up to 20 Bq/kg (Fig 1b). Besides ^{137}Cs and ^{90}Sr some other anthropogenic radionuclides, such as ^{152}Eu, ^{60}Co, ^{134}Cs, were found in this sample. The presence of these isotopes is not typical for the open parts of the Barents Sea.

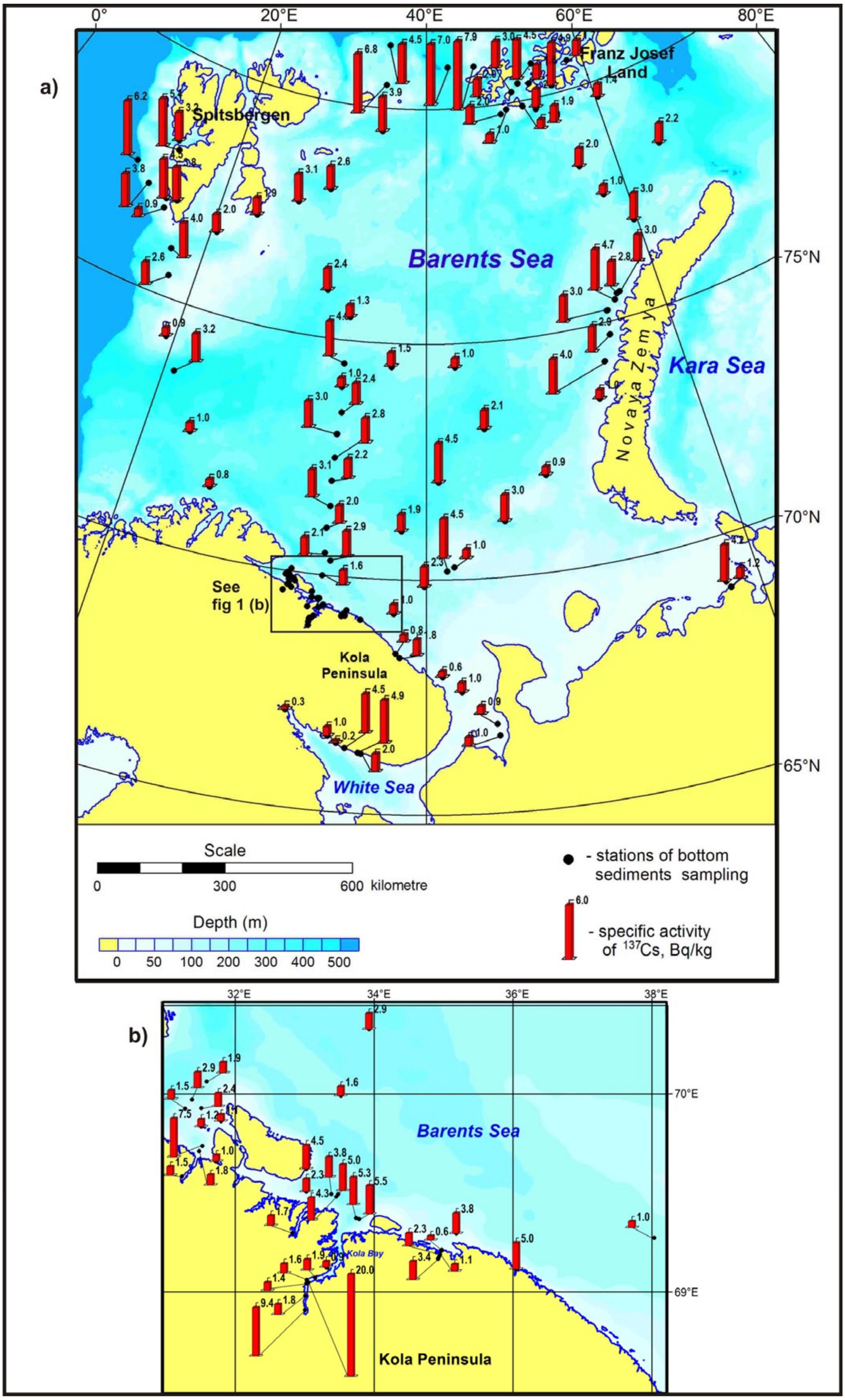

Fig. 1. Specific activity of ^{137}Cs in bottom sediments of the Barents, Greenland and White Seas, 2005-2009.

The activity of ^{137}Cs in contemporary macrophytes (Fucus vesiculosus. being mainly studied) varied from 0-1.2 to 8.4 Bq/kg dry weight and 90Sr, from 0.6 up to 4.0 Bq/kg (Table 1, predominantly *Fucus vesiculosus*). The maximal concentration of ^{137}Cs of 8.4 Bq/kg was found for the littoral macrophytes near the settlement of Mishukovo close to «Atomflot» Enterprise.

Table 1. Specific activity of artificial and natural radionuclides in *Fucus vesuculosus* of the Barents Sea, 2007.

Sampling area	Coordinates		Specific activityof of radionuclide, Bq/kg dry weight		
	N	E	^{137}Cs	^{90}Sr	^{40}K
Kola Bay, Abram-Mys settlement	68° 58' 53"	33° 01' 41"	<1,2	0,8±0,3	958±51
Kola Bay, Retinskoe settlement	69° 06' 48"	33° 22' 00"	4,7±1,0	2,3±0,9	591±70
Kola Bay, Min'kino settlement	69° 59' 36"	33° 01' 36"	<1,27	—	840±100
Kola Bay, Belokamenka settlement	69° 04' 34"	33° 10' 13"	<1,86	1,6±0,5	661±80
Upper part of Kola Bay	68° 54' 27"	33° 01' 32"	4,7±1,3	1,8±0,4	930±109
Teriberskaya Bay	69° 10' 48"	35° 10' 57"	<1,35	—	717±83
Dalnezelenetskaya Bay	69° 07' 01"	36° 04' 11"	<1,56	0,8±0,3	837±37
Kola Bay, Roslyakovo settlement	69° 03' 39"	33° 12' 30"	<1,07	2,4±0,8	722±28
Kola Bay, town of Severomorsk	69° 05' 00"	33° 25' 42"	<2,1	0,6±0,3	748±9
Kola Bay, Mishukovo settlement	69° 02' 39"	33° 02' 42"	8,4±1,7	1,8±0,7	707±9
Ura-Guba	69° 18' 08"	32° 50' 43"	<1,66	4,0±0,5	558±68

Radioecological studies of the Barents Sea commercial fish showed that the present-day activity of artificial radionuclides is extremely low. All investigated species (Atlantic cod, haddock, long rough dab, spotted wolfish) contained ^{137}Cs of less than 0.2 Bq/kg raw weight. Thus, the radioactivity of the Barents Sea ichthyofauna nowadays is primarily determined by the presence of natural radionuclide ^{40}K.

Temporal changes in the ^{137}Cs content have been best studied in Atlantic cod from the Barents Sea (Fig. 2). The average ^{137}Cs concentrations are known for the period beginning from 1979. The maximum ^{137}Cs content in cod (2.2 Bq/kg raw weight) was observed in 1982, which may have been accounted for discharges from radiochemical enterprises of the Western Europe. The maximum ^{137}Cs content in cod from the Barents Sea was delayed by four to five years with respect to the maximum amount of this nuclide in water (Matishov et al. 2001). The ^{137}Cs content in cod of the Barents Sea after 1982 was diminished exponentially with a seven-year half-life period. In 2005-2009, the ^{137}Cs concentration in cod did not exceed 0.2 Bq/kg raw weight. Taking into account the current composition of food consumed by human beings, the dose of radioactive cesium received through eating fish falls into the «radioactively safe zone».

In summary, the present-day concentrations of the anthropogenic isotopes in the ecosystem elements are evidence of the absence of high-energy sources of artificial radioactivity after the cessation of nuclear weapons tests on Novaya Zemlya and the adjacent shelf in 1950-1960s and the Chernobyl global emission in 1986.

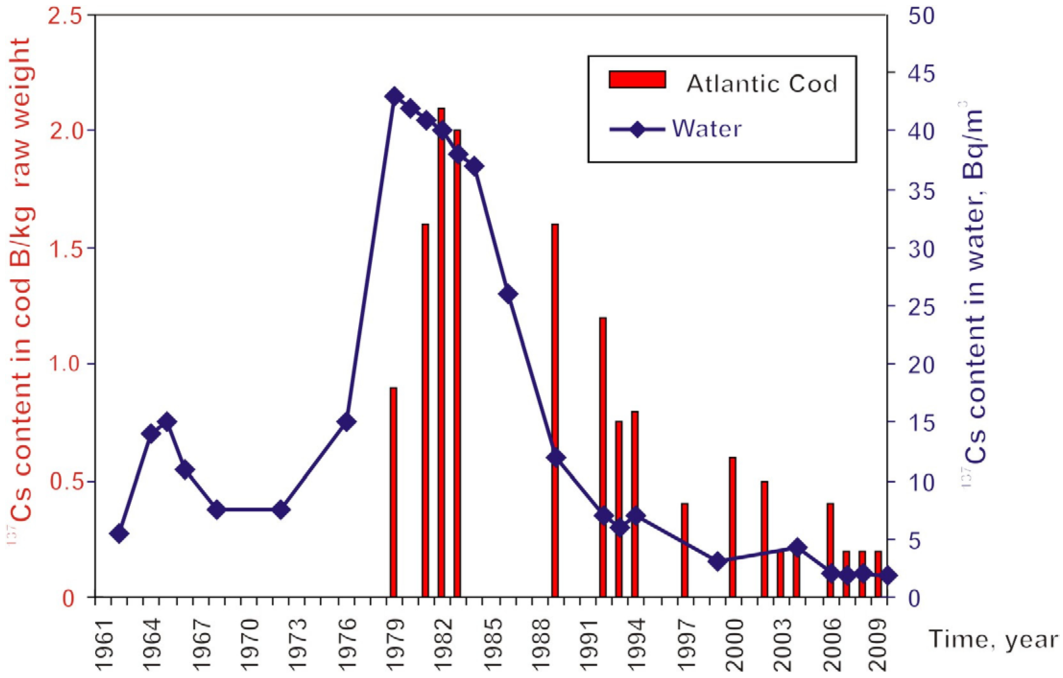

Fig. 2. Average ^{137}Cs concentration in water and in Atlantic cod from the Barents Sea (1979-2009).

Conclusions

The major source of the present-day radioactive pollution in the Russian Western Arctic are still the long-living isotopes of ^{137}Cs, ^{90}Sr, and ^{239,240}Pu, emitted during nuclear weapons tests in the middle of the twentieth century. Some artificial radionuclides originating from the Sellafield factory are coming with the Atlantic waters. The impact of local atomic power fleet and transport complexes is registered in sites 10-20 km apart.

Natural purification processes in the marine environment, such as repeated dilution, nuclear decay, sorption by sediments and suspended solid material, and accumulation by aquatic inhabitants, are the main factors of the decrease in the intensity level of artificial radioactive isotopes. Against the background of radioactive pollution decrease, studies on other natural previously unexamined radionuclides (^{40}K, ^{226}Ra, ^{228}Th, ^{210}Po and others) may become a question of interest.

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Human dose pathways from forests contaminated by atmospheric radionuclide deposition

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Abstract

After radionuclide contamination of forests, the potential health risk of both the early and late phase radiation exposure of people using forests needs to be assessed. Radionuclides vary in their exposure characteristics. Short-lived ^{131}I can be a significant source of internal and external radiation in the first weeks. Long-lived ^{137}Cs may call for long-lasting surveillance of forest products, and ^{134}Cs contributes to human doses for a few years. ^{90}Sr is seldom a significant contributor to the radiation doses from forests unless there is a high surface contamination of berry plants and mushrooms. Radiation doses depend on the ways people use forests and forest products. Consumption rates of wild foods and the time spent in forest vary by countries and population groups. Dose pathways that need attention are the ingestion of wild foods, the handling and use of wood ash, and the exposure to external radiation in forests particularly in the first weeks after radionuclide deposition.

A hypothetical contamination scenario was used to demonstrate the importance of the different pathways to the human doses. The forest food chain and dose model of RODOS, a European emergency response system, was used for scenario assessment. The assessments show that the highest annual doses arise from the ingestion of wild foods from the first harvest after deposition of ^{131}I, ^{134}Cs, ^{137}Cs and ^{90}Sr in early July. Exposure of adults to external radiation from ^{131}I in forest vegetation and ground is important in the first weeks. Small scale users of fuel wood are exposed via an additional dose pathway through the use of contaminated ash for enrichment of garden soil. Regular ash fertilization accumulates ^{137}Cs in soil year by year. A maximum annual dose of 60 $\mu Sv\ a^{-1}$ was estimated to residents due to stay in ash-fertilized garden, as late as seventy years after the hypothetical contamination event (100 $kBq\ m^{-2}$). Of course, actual doses will depend on the details of exposure situation. Nevertheless, the presented assessment may help in understanding the relevant pathways. Such an understanding is important when giving advice on the safe use of contaminated forests as are regionally adapted assessment models and knowledge of the local forestry practices.

Introduction

Atmospheric release and the consequent distribution and deposition of radioactive material in rural areas may present a threat to radiation safety of people using forests. The deposition pattern can vary substantially due to the characteristics of the release, weather conditions during the passage of the plume, and the distance from the source of release. Environmental and ecosystem related processes change the initial distribution of radionuclides in forests. In addition, radioactive decay reduces the overall activity in forest.

Radionuclide contamination of forests has been a concern of experts on radioecology, radiation protection and forestry since the Chernobyl accident in 1986. In recent decades long retention times of radioactive caesium in the nutrient cycle of forest became evident. Only insignificant losses of radionuclides through surface runoff or fixation in soil minerals were detected in boreal forests on podzolic soil horizons (Tikhomirov and Shcheglov 1994; Bergman 1994, Dahlgaard et al. 1994). Relatively high uptake of radioactive caesium from soil to forest vegetation and further to wild foods was also confirmed (Aarkrog 1994; Rantavaara 1990).

The aim of the present study was to clarify human dose pathways after radionuclide contamination of forests. Comparative model predictions of the potential doses to different users of forests were assessed and factors modifying the doses were referred to.

Dose pathways from contaminated forest

Radionuclides distributed in trees, understory, ground layer vegetation and soil are sources of external exposure to people spending time in contaminated forests. Sawn timber and logs used as building material can expose people to external radiation in residential and working environments. Living year-round (7000 h a^{-1}) in a wooden house made of 15-20 cm thick logs requires that the activity concentration of ^{137}Cs is less than 1700-2000 Bq kg^{-1}(80% dry matter) in order that the dose from building material to residents remains below 1 mSv a^{-1}. For sawn timber used in a timber-framed detached house the respective ^{137}Cs concentration level would be 2400 Bq kg^{-1} (80% dry matter). The dose assessment model MATERIA (Markkanen 1995) was used for this estimation. Various other wood products in residential environments, for instance furniture, are causing considerably lower doses than wood used as building material (IAEA 2003).

Wood ash concentrates those radionuclides that are not released to the atmosphere during combustion, e.g. isotopes of Cs and Sr. The concentration factor for the combustion of wood ranges from 50 to 200, and corresponds to mass ratio of dry fuel and ash. Workers involved in disposal and recycling of ash may receive the highest individual doses amongst the workers in the production chain of wood energy (Vetikko et al. 2004).

Ingestion of radionuclides in wild berries, mushrooms and game meat is often the most significant dose pathway from forest, mostly due to the consumption of large amounts of wild foods. Mostly minor doses are received from the inhalation of airborne radionuclides during the passage of a radioactive cloud or from resuspended particles, soil or ash dust, if the instructions for protection of workers and the public are followed.

About assessment data

The radiological significance of a contaminating event depends on the characteristics of deposited radionuclides, their decay rates, type and energy of radiation and their chemical behaviour in natural ecosystems and human metabolism. Site conditions and forest management practices govern the radionuclide dynamics in forests.

The dose that is received within the first year depends crucially on the season during which the radionuclide deposition occurred. Furthermore, the time from deposition to the actual radiation exposure needs to be taken into account. The longer the delay, the lower is the dose contribution from short-lived nuclides and from foliar contamination of plants. The ways people are using forests and forest products varies and requires surveying. For instance, berry and mushroom pickers und hunters have distinctive consumption patterns. Forestry workers were assumed to be exposed to radiation during working in forest only; they may still belong to other subgroups of users of forests.

It is challenging to quantify radionuclide flow in forests, a continuously changing ecosystem with a rotation period of about 80 years in Central Europe and even longer in sub-arctic conditions. Of great value to radioecology has been the experience gained in forest research and the collaboration with organizations that have comprehensive knowledge of forest. State-of-the-art sampling methods are essential for obtaining reliable field data on radionuclides in forest (Aro et al. 2009). There exist several assessment models for forests. The validity of some of them was compared in the BIOMASS programme of IAEA and reviewed by Shaw et al. (2005).

Two nuclear accidents in the past caused widespread contamination and had also an impact on forests in boreal and temperate Eurasia and America. These were the accident at the Chernobyl nuclear power plant in Ukraine in 1986 and the chemical explosion in the nuclear fuel processing plant Mayak near Kyshtym, Cheliabinsk province, Siberia in 1957 (Trabalka et al. 1980; Jones 2008). These accidents were widely studied and provided important radioecological field data on radionuclide behaviour in forests. Another valuable data source were studies related to the global fallout from atmospheric nuclear testing during 1945-1980. Also some smaller events have contaminated forests, albeit on a limited scale, and were subject to radioecological studies (Auerbach 1987). Radionuclide transfer in forests was one of the topics in a literature review published by IAEA (2009) and in a handbook on transfer parameters (IAEA 2010), which, regarding forests, was summarized by Calmon et al. (2009).

Dose assessment model

The forest food chain and dose assessment model FDMF, which is part of the European emergency response system RODOS, was used for the dose assessments (Rantavaara et al. 2001). The parameters of FDMF were derived from Finnish and other European data from experimental field studies, survey data on people's outdoor activities and consumption of wild foods. The compartment structure and transfer processes considered in FDMF allow assessment of the radionuclide flow in forests (Fig. 1).

Using the RODOS system, predictions for large affected areas can be provided. Through regional parameters the assessment results represent the conditions in calculation area. During dry weather the time-integrated air concentrations are used in the calculation of the initial activity densities of radionuclides on different surfaces within the forest. These surfaces are the forest soil, understory vegetation, and crown and trunk of trees. The deposition onto these surfaces during rain depends on the development stage of the canopy, the amount of rain and the total deposited activity. The result of a prognosis calculation depends on the choices made during the setup of a model run. The choices can be, e.g. the type of forest, the population group, the species of plants and animals providing wild foods, the type of fuel wood and the details of the use of wood ash. The model takes into account height and biomass density of the forest type chosen, regional consumption rates of wild foods and losses of radionuclides during cooking.

FDMF calculates the time-dependent radionuclide contents in forest compartments, activity concentrations in forest products and radiation doses to various population groups (hunters, pickers, public). The need for intervention and availability of acceptable wood and wild food products can be further assessed with another RODOS model, LCMforest (Rantavaara, Ammann 2005).

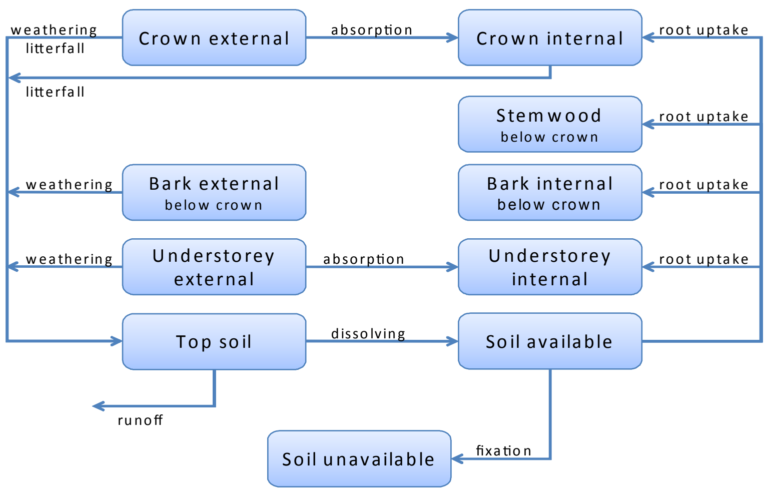

Fig. 1. Compartments of the dynamic module of the forest model FDMF and the processes changing their radionuclide content with time.

Calculation scenario

A hypothetical contamination of two forest types was assumed to have happened in July. The activity inventories of the forests contained the following four radionuclides: ^{90}Sr, ^{131}I, ^{134}Cs and ^{137}Cs (Table 1). The initial contamination exceeded manyfold the average activity densities of these radionuclides in most countries after the Chernobyl accident. The activity ratios of the radionuclides were roughly those found in Fennoscandia in early May 1986. Dry deposition was assumed to happen on the 8th of July. An advanced Scots pine *(Pinus sylvestris L.)* forest was chosen to represent the Northern Europe and a coniferous forest type the Central Europe.

Wild food species were blueberry *(Vaccinium myrtillus L.)*, moose *(Alces alces)* and mushrooms corresponding to an aggregated transfer coefficient of 0.05 m^2 kg^{-1} for isotopes of Cs. For Central Europe this uptake category included *Cantharellus cibarius, Boletus edulis, Collybia sp.*, several *Lactarius* sp., *Leccinum* sp. etc. and for Northern Europe *Cantharellus tubaeformis Cratellus cornucopioides*, some of the *Russula sp.* etc.

Table 1. Radionuclides included in the assessment, their half lives and activity inventories in forest at the end of radionuclide deposition.

Radionuclide	Half life[1)]	Inventory (Bq m^{-2})
^{90}Sr	28.79 y	10^3
^{131}I	8.02 d	5×10^5
^{134}Cs	2.065 y	5×10^4
^{137}Cs	30.07 y	10^5

[1)] Chu SYF, Ekström LP, Firestone RB. The Lund/LBNL Nuclear Data Search, Version 2, February 1999.

Potential doses of adults from external radiation exposure were calculated from air kerma in boreal Scots pine forest. In FDMF the conversion from kerma to effective dose of an adult takes into account the source compartments in forest and a correction for the size of a subject to obtain effective dose for a five years old child. Doses from ingestion of wild foods were estimated for Central Europe (CE) and Finland (FI).

Activity concentrations in timber for building (logs, sawn timber) and fuel wood (split firewood) were assessed. The dose pathway of using wood ash for soil enrichment was assessed with FDMF. Doses to residents were derived from ^{134}Cs and ^{137}Cs accumulating in soil due to a regular distribution of ash from fireplace to kitchen garden. The fuel wood from an affected area was used. Other assumptions were:

- delay since felling of trees until distribution of ash: 1 year
- ash application rate: 0.2 kg m^{-2} in a year
- surface soil was managed after each ash application, denoting a radiation attenuation factor of 0.3
- exposure time of adult residents to radiation from ash in garden: 300 h/year.

Results of scenario calculations

In the first week after radionuclide deposition ^{131}I in crown layer and in the entire forest contributed to the dose rate more than ^{134}Cs and ^{137}Cs together (Figs. 2 and 3). After the first week the ground layer contributed most to the dose rates in forest.

The activity concentrations in logs and split firewood (Fig. 4), the activity density accumulating in garden soil via regular ash application (Fig. 5) and the annual effective dose from additional ^{134}Cs and ^{137}Cs in garden (Fig. 6) due to ash mixed in soil indicate a gradual increase of the annual doses to residents during decades. The dose from the external exposure to radionuclides in ash distributed in the garden soil was approaching 0.04 mSv a^{-1} forty years after the deposition. The contribution of ^{134}Cs to the dose was negligible. The maximum annual dose to adult residents was estimated to be 0.06 mSv a^{-1} about seventy years after deposition.

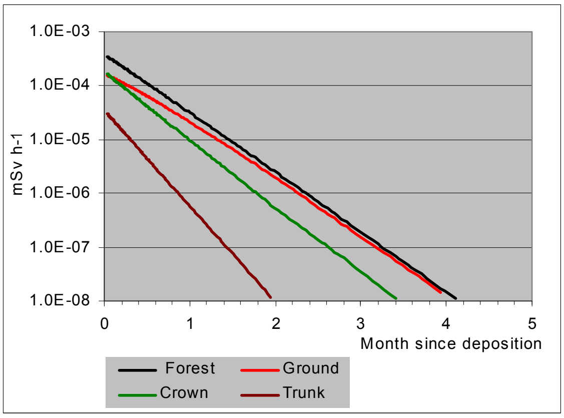

Fig. 2. Effective dose rate (mSv h^{-1}) due to radiation from ^{131}I in ground, trunk and crown layers, and the total dose rate in forest. Region FI.

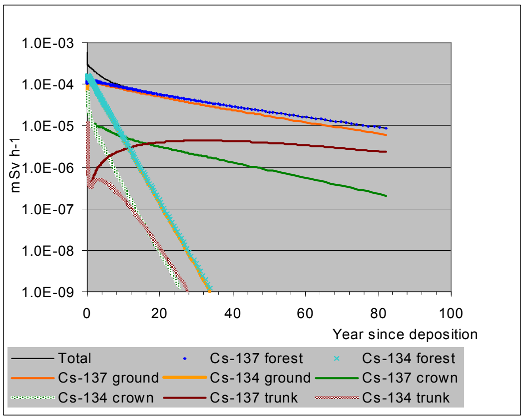

Fig. 3. Effective dose rate (mSv h^{-1}) due to radiation from ^{134}Cs and ^{137}Cs in ground, trunk and crown layers, and the total dose rate in forest. Region FI.

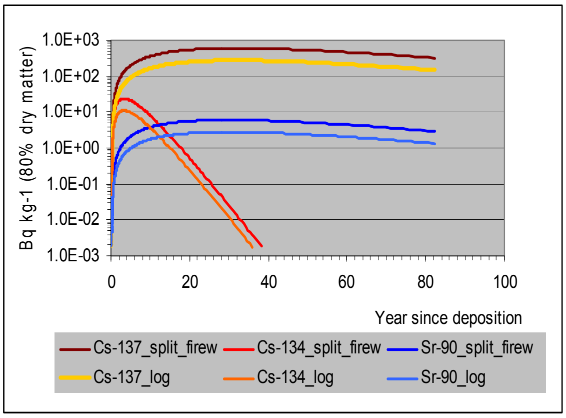

Fig. 4. Activity concentration in timber(Bq kg^{-1}, 80% dry matter) for building (logs) and in split firewood.

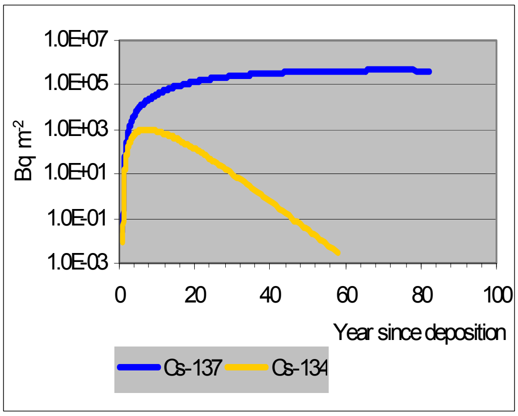

Fig.5. Activity density of ^{134}Cs and ^{137}Cs in kitchen garden due to soil enrichment with wood ash.

The doses to adults from external exposure in forests (Fig. 6) were used to give an idea of the differences between population groups. Adults in Central Europe received 33% and hunters and pickers 44% of the dose that adults in Finland would receive. Within the Finnish population the doses to different population groups were, again compared to adults, 33% for children (5 years of age), 140% for hunters and 110% for pickers. Forest workers were assumed to work 150 hours per month in forest without shielding from a harvester or other machines. Annual doses during 11 months per year were 15-fold compared to the dose of an adult (Fig. 6).

Wild foods contributed to the dietary dose significantly more in Northern (Fig. 7) than in Central Europe (Fig. 8). The doses differed mainly because of lower consumption rates in Central Europe (Fig. 9). Children (5 y) were not assumed to consume wild foods or go to the forest in CE. Radionuclides deposited on plant surfaces close to harvest or hunting seasons contributed significantly to the first year doses from blueberry and moose meat.

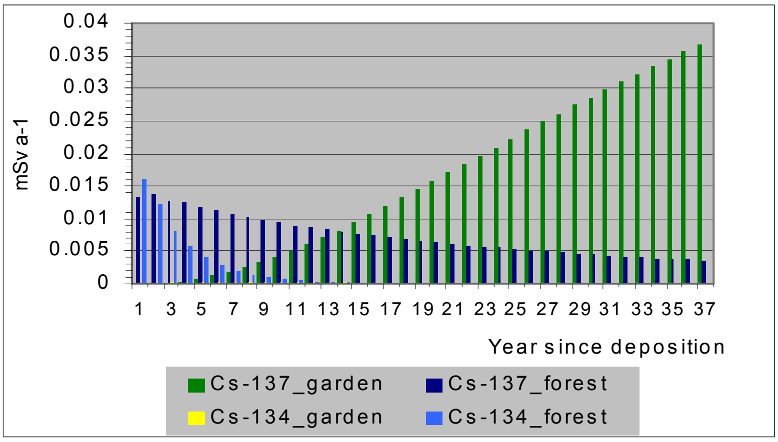

Fig. 6. Annual effective dose to an adult due to radiation from ash distributed in kitchen garden, compared to dose received in forest. Region FI.

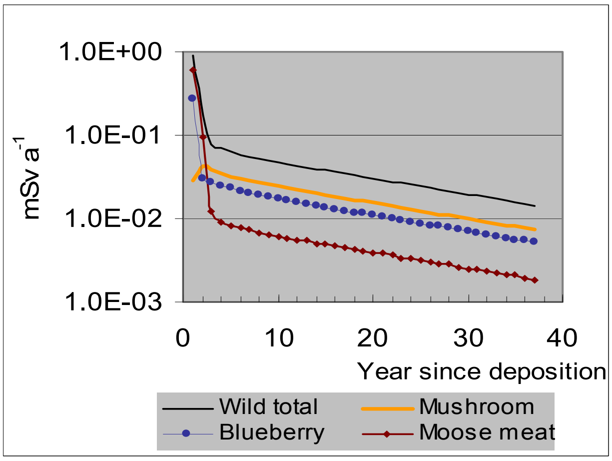

Fig. 7. Effective dose commitment to adults from annual ingestion of radionuclides in wild foods, region FI.

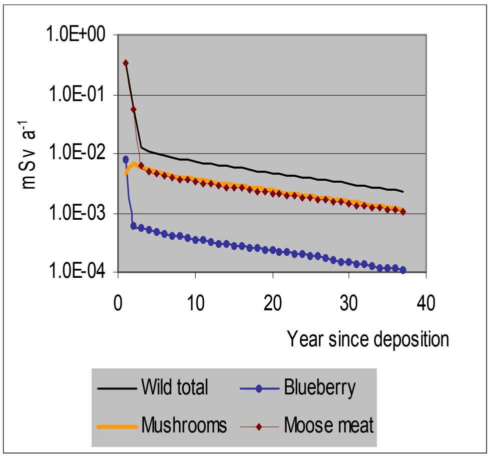

Fig. 8. Effective dose commitment to adults from annual ingestion of radionuclides in wild foods, region CE.

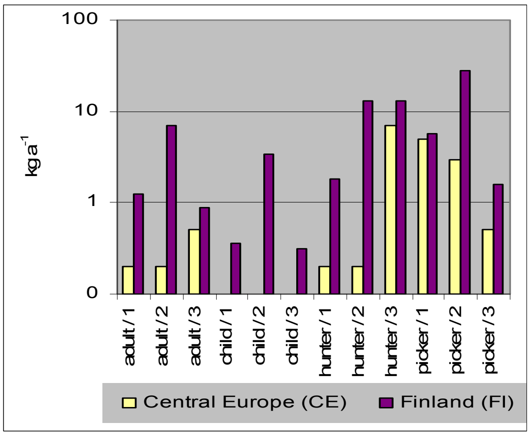

Fig.9. Consumption rate of mushrooms (1), berries (2) and game meat (3) by population group in two regions.

Discussion

The various dose pathways contribute to the human dose from forests in relation to radionuclide, region, population group, and the time since forest contamination. The ways people use forests and forest products are changing in time. Consumption rates of wild foods and the time people spend in forests should be surveyed regionally at intervals of a decade or so. A suitable survey methodology should be used, the types and species of wild foods specified and cultural and other sources of regional variation considered. All parameters determining the effective dose from external radiation or dose commitment from radionuclides ingested in wild foods are important independent of the nature of data.

Natural processes and forest management are changing the radionuclide distribution in forest vegetation and soil. Together with consumption rates they are contributing to temporal variation in radiation doses received from wild foods. The dominant dietary pathway to the public is seldom via forest during the first years after a contaminating event. Later this may change, and the exposure to slowly declining ^{137}Cs concentrations in wild foods will prevail while other foodstuffs are practically clean. An example on such changes was obtained via assessment of systematic foodstuff surveillance data collected in Finland in 1960-2005. Percentage contributions of terrestrial wild foods to dietary ^{137}Cs increased in time during periods of global nuclear fallout and after the Chernobyl accident (Rantavaara 2008). Essential was the simultaneous decrease in the total dietary intake, caused mostly by environmental processes in agricultural production systems. Substantial dietary changes did not hide the effect of environmental processes on doses from ingestion of ^{90}Sr and ^{137}Cs during these periods. The doses received in the first year after radionuclide deposition were the higher the closer to consumption of foodstuffs of the season the deposition occurred.

Timber used for building, actually stemwood, is the least contaminated part of a harvested tree, whereas fuel wood can contain also more contaminated parts of a tree. Particularly wood ash needs attention of experts on radiation protection, energy industry and small scale use of firewood. Bioenergy production based on forest biomass as fuel can be a pathway of concern in the future. Intensive harvesting of felling residues has been supposed to involve detrimental removal of mineral nutrients from forest. If carried out, a condition would develop where increased root uptake of radioactive caesium and strontium occurs, whereas returning nutrients to forest floor through wood ash fertilization could have a reducing effect on ^{137}Cs contamination of forest vegetation (Rantavaara and Aro 2009).

Radiation safety guidance is needed in energy industry for protection of workers handling with wood ash. The wood ash surveys in the 1990's (Rantavaara and Moring 1999) and early 2000's (Vetikko et al. 2004) clarified the need for site-specific measurements of industrial ash in Finland. The studies probably facilitated safe handling of ash and following the related radiation safety guide (STUK 2003).

Conclusions

Radionuclide contamination of a forested area should be assessed for the potential health risk. Providing local people and other concerned population groups with information and advice can help avoiding unnecessary radiation doses. Due to the complexities in radionuclide contamination of forests assessment models are needed to prepare the preliminary advice. Nevertheless, decisions on remedial measures and restrictions for the use of forests need to be based on qualified measurements because of their socio-economic consequences. For correct interpretation of the measurement results scientific design of measurement and sampling strategies are fundamental. Experts on forestry should be involved in the planning and implementation of intervention programmes.

Stem wood will be contaminated with delay, slowly accumulating long-lived isotopes ^{137}Cs and ^{90}Sr, and is the least contaminated part of a tree. Timber that is used for buildings and originates in other than the most contaminated areas of Europe is not supposed to cause to residents an effective dose higher than 1 mSv a^{-1}.

Doses from the treatment, transport and use of wood ash can occasionally exceed 1 mSv a^{-1} to e.g. long-range truck drivers. The small scale users of contaminated fuel wood and ash may need guidance to avoid unnecessary radiation exposure.

Acknowledgements

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Occurrence of plutonium in the terrestrial environment at Thule, Greenland

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Abstract
Samples of air, resuspended particles, water, soil and precipitation were collected in an area 10 km south of the Thule 1968 impact point and analysed for their content of ^{241}Am and plutonium. The results from the soil sampling show a very inhomogeneous distribution with hot spots ranging up to several hundred kBq per m^2 of Pu. Although concentrations in surface soil can be very high the concentration in analysed air filter samples and passive aerosol collectors are very low. Exposure to plutonium due to inhalation of airborne plutonium particles in the area is of little importance according to this study. To further assess the risk of inhaling resuspended material particles isolated from the different hot areas have been subject to investigation on stability and leaching behaviour.

Introduction
On the 21 of January 1968 an American B52 bomber carrying nuclear weapons crashed on the sea ice in Bylot Sound, about 15 km west of Thule Air Base, northwest Greenland, Fig.1. On impact the plane caught fire and the jet fuel as well as the weapon's chemical high explosives detonated causing a local dispersion of radioactive material. Part of the weapons plutonium was distributed over some square kilometres of the ice in the explosive fire that followed. Plutonium-contaminated ice was recovered and shipped back to the US, as was the plutonium-contaminated wreck. The underlying sea sediments received a fraction of the weapons plutonium when the sea ice melted the following summer and probably also during the accident as the impact caused the ice to break up (U.S. Air Force, 1970).

Flames from the fire reached heights of around 850 m and the smoke pillar even higher. Based on observations from eyewitnesses, metrological data and radar observations it was concluded that the contaminated cloud from the fire drifted south to southwest. At the time of the accident an inversion layer existed at around 830 m and the atmosphere was thermally stable up to around 2200 m. At a height of 1000 m the winds came from the north, higher up, at 3500 m winds were from the west. Wind speed was around 3 m s^{-1} at all heights. It was anticipated that from these observations, finer particles from the fire could have been transported far away from the crash site and reached land south of Bylot Sound, although at low concentrations. It was consequently expected that measurable concentrations of contamination could be found in the direction of the Inuit settlement site Narsaarsuk on the coast some 8 km south of the crash site and further in on land. Contaminated material on the crash site was also anticipated to have been resuspended and dispersed westwards towards Sounders Island due to the storm which followed on the 24 and 29th of January (U.S. Air Force, 1970).

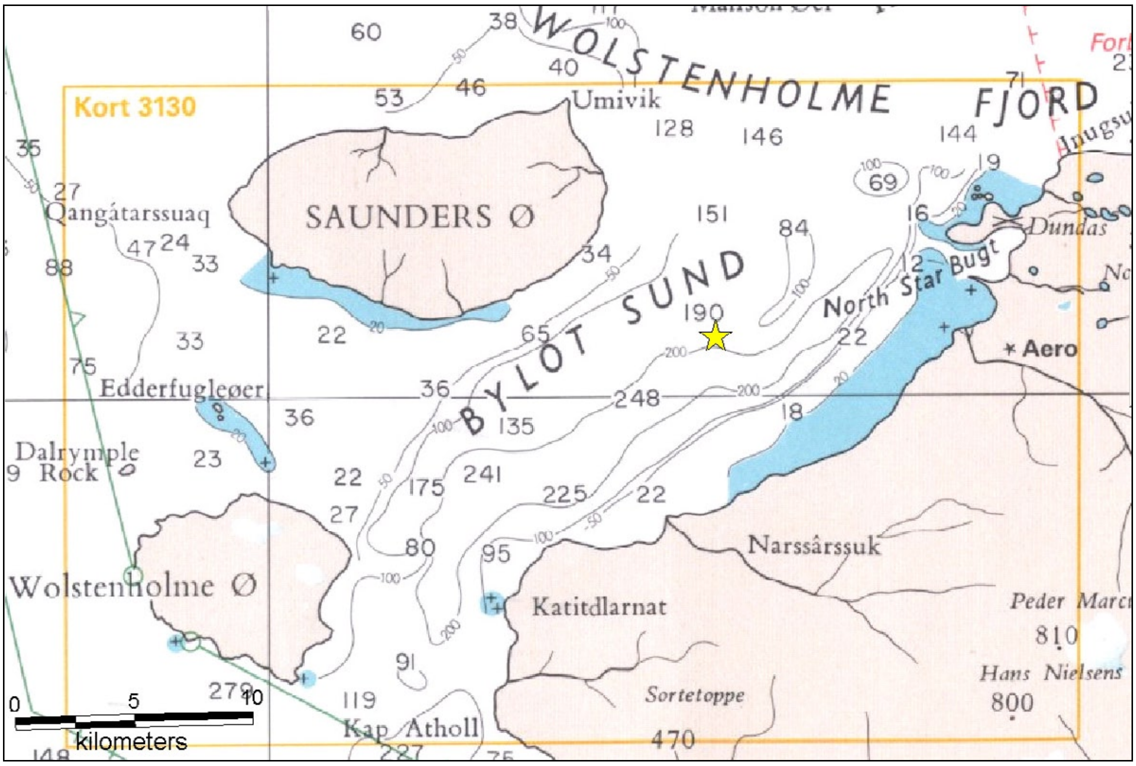

Fig. 1. Map of Bylot Sound showing the site of the aircraft accident (star).

Due to the frequent storms, melting snow and running surface water the contaminated material initially residing on the snow surface may to some extent have been redistributed in the months and even years following the accident. This redistribution may be anticipated to have taken place locally.

The distribution of plutonium on land was initially investigated only weeks following the accident by the analysis of snow samples. In general two zones with contaminated material was identified, one in direction towards south of the crash site, the primary fallout zone, and one towards west which most likely was due to redistributed resuspended material from the crash site in connection with the violent winds following the accident. The contaminated areas on land south of Bylot Sound showed maximum levels of around 9 kBq m^{-2} close to Narsaarsuk.

In connection with the Thule-2003 project concerning investigation of plutonium in the environment around Thule (Nielsen og Roos, 2006) soil samples were collected from 8 localities at Narsaarsuk. At all 8 sites plutonium from the accident could be quantified at different levels in the top soil layers. The spatial variation in plutonium concentrations is very un-even due to the presence of small particles. Detailed analysis of the soil samples collected in 2003 have shown particles with a ^{239,240}Pu content up to 150 Bq. In connection with a pilot study carried out in august 2006 at Narsaarsuk, soil samples were collected which further confirmed the presence of weapons plutonium as well as a large spatial variation of the contamination.

Occupancy at Narsaarsuk, where elevated plutonium levels have been found on the ground, may include a risk of inhaling radioactive particles if these are resuspended from the ground. This resuspension may be caused either by wind or by some mechanical action of the soil. Depending on the level of exposure to these radioactive particles there is a potential risk to human health.

Material and methods

During 2006-2008 a comprehensive field expedition was conducted in the Kap Atholl area at Thule. Soil samples were collected using a 10 cm diameter PVC-tube. At each sampling location 5 individual samples were taken in order to assess deposition variability. A total of about 500 soil-samples were collected over an area of about 20 km2. Overview of the ^{241}Am deposition in the area was obtained using a portable NaI-detector system. Air sampling was done using a Staplex air sampler with a flow rate of about 1 m^3/min. Passive collectors ('sticky vinyl') were placed on wooden frames or on to walls of the shacks in Narssarssuk in order to further assesse the presence of plutonium in air. Rain was collected using a 0.25m^2 funnel located close to the air sampler. Assessment of resuspension of plutonium from soil was done using a simple vacuum-cleaner on areas of about 1 m^2. In the laboratory soil samples were screened in 5-15g aliquots to quantify the total amount of ^{241}Am in each soil sample. The presence of particles and thus inhomogeneous soil samples made this procedure necessary. Passive collectors were analysed first using a digital autoradiography system and later analysed by radiochemical methods and solid state alpha spectrometry for Pu-isotopes. Air filters and rain samples were first analysed by gamma spectrometry and later by solid state alpha spectrometry for Pu-isotopes.

Results

The results from analysed soil samples shows that the Pu contamination is very unevenly distributed over the Kap Atholl area. Several so called 'hot spots' ranging from less than a meter in size to several tens of meter exist in the area around Narssarssuk. Deposition density of Pu in these areas are in the order of 10-100 kBq m^{-2} or even more but is difficult to determine due to the areal Pu density in many cases being determined by a few larger particles. Outside the high deposition areas concentration of Pu is more likely to be found in the 0.1-1 kBq m^{-2} range. A figure showing the frequency distribution of Pu from analysed soil samples are shown in figure 2.

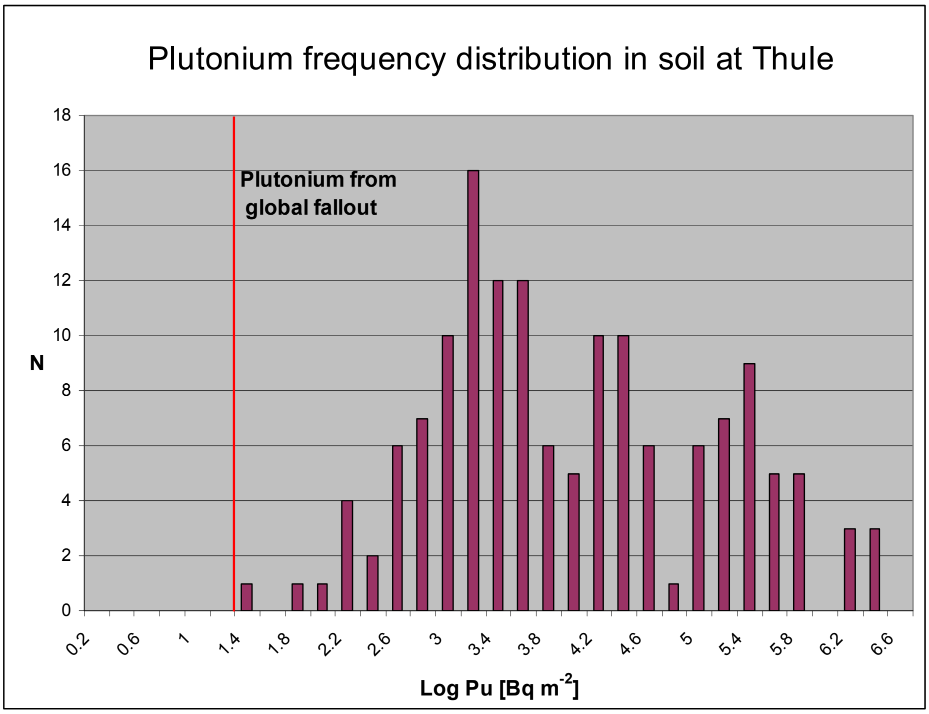

Fig. 2. Frequency distribution of Pu from analysed soil samples.

Results from the resuspension experiments using a simple vacuum-cleaner shows that Pu can be found in nearly all samples (figure 3). In spite of the relatively large amount of resuspended material even during a single exposure to vacuum cleaning the levels of plutonium in air was found to be very low during this study. From the air filters analysed no concentration higher than 5 nBq m^{-3} was observed. Similarly the passive particle collectors showed no sign on digital autoradiography of particles attached during the exposure period. Radiochemical analysis of plutonium on the passive collectors resulted in concentrations in the range 0.1-3 mBq. The conversion of these numbers to corresponding air concentrations is not straightforward and has been omitted.

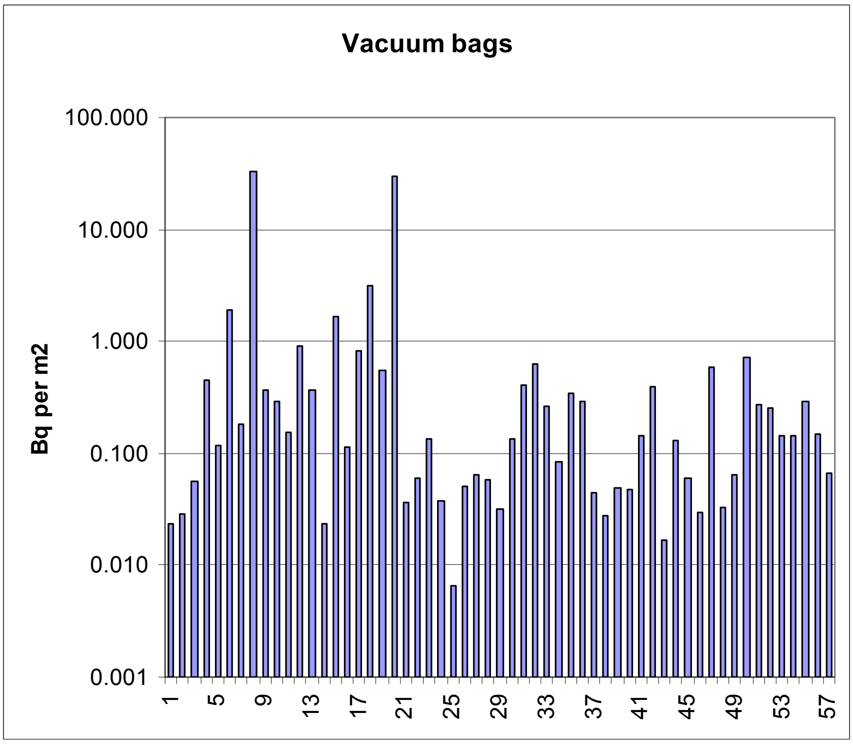

Fig. 3. Results of ^{241}Am from analysed vacuum cleaning bags. Corresponding ^{239}Pu levels can be obtained by multiplying the numbers by six (^{239}Pu/^{241}Am ratio in the Thule material is about 6).

Discussion

The deposition density of plutonium found in the terrestrial environment south of the impact point is of the same order of magnitude as the most contaminated sediments in Bylot Sound. While relocation of contaminated sediments to accumulation bottoms is an important process explaining the spatial distribution of plutonium in Bylot Sound similar mechanisms probably can explain the spatial variability of plutonium on land in the Narssarssuk area. The Thule area is classified as an arctic desert and large areas in the Kap Atholl peninsula is stony, rocky or sand terrain. The annual amount of precipitation is very low and in combination with violent storms it is evident that areas not sheltered from wind will be heavily eroded. Any material initially deposited on such areas will thus likely reside there only for a short time before being relocated successively to areas with wind shelter and/or areas having a high water content or vegetation. Since most areas with vegetation receive their water through nearby deposited snow melting during the summer a view over the area showing pockets of snow in summertime also reveal the potential areas where redistributed plutonium has finally ended up. Probably most of the initially deposited material on snow was relocated to their final destinations during the storm that took place a few days after the accident. All the hot spots identified in the Narssarssuk area are located close to snow patches but not all snow patches are associated to elevated plutonium areas.

The risk of resuspending the deposited material is depending on surface properties such as presence of vegetation, water and how exposed the area is for wind. Although the landscape has been exposed to more than 40 y of strong winds it is clear that the risk of resuspending deposited plutonium is significant according to data in figure 3. Air filter data however showed very low concentrations and the maximum observation of 5 nBq m^{-3} corresponds to doses due to the inhalation pathways which is in the order of 1 nSv y^{-1}. Possibly ingestion may be the dominating pathway of the terrestrial Thule plutonium to man in the area.

Conclusions

Although locally deposition densities of plutonium in the Narssarssuk area may reach close to 1 MBq m^{-3} concentrations of plutonium in air is very low and does not constitute any severe health problems according to this study.

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Environmental radioactivity assessment at nuclear legacy sites in the Republic of Tajikistan

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Abstract

The impressive number of extant nuclear and radiological sources in the Central Asia is a major environmental concern as they pose a potential threat of radioactive contamination to the whole region. Key sources of concern are technologically enhanced levels of naturally occurring radionuclides (TENORM) due to uranium mining and milling. Uranium ore mining and processing in the former Soviet Republic of Tajikistan resulted in origination of huge amounts of uranium tailing materials and waste rock deposits, often dumped near inhabited areas. Given the absence of a proper waste management in most of those areas, there is a considerable potential for the spread of contamination beyond existing contaminated sites. A joint field mission to Tajikistan was conducted in 2008 as part of the project: Environmental impact assessment of radionuclide contamination of selected sites in Kazakhstan, Kyrgyzstan and Tajikistan, coordinated by the Norwegian University of Life Sciences. The present work focuses on the assessment of radiation exposure and radioactivity levels in the former uranium mining and tailing sites at Taboshar, and Digmai, both located in northern part of Tajikistan. Gamma-dose rate surveys were carried out on each site. Generally, the dose rates measured at the uranium mining site in Taboshar varied between 0.70 and 4.4 µGy/h. However, this excludes some hot spots where dose-rates as high as 20 µGy/h were measured. On the tailing site which contains 1.2 mln tons of radioactive wastes the measured dose rates varied 0.7 – 1,9 µGy/h. It is noteworthy that the tailing site lies just 1 km far from the local school in Taboshar where we also did measurements. The highest level of radioactivity was exhibited by Digmai – 18.8 µGy/h. From each site soil samples were taken for subsequent lab analyses for radioactivity and track detectors placed on-site for determination of Rn concentration. The analyses are in progress and the results will be available by the end of 2009.

Improved model for estimation of fallout from atmospheric nuclear testing

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Abstract

Estimation of global fallout is still important when establishing background values for many radionuclides. Even where deposition measurements have been made, there is a limit for how comprehensive they can be or can have been. Using a model makes it possible to put the available data in a better framework. Two types of models for global fallout have often been used, one assuming that the variation in deposition is a function of latitude, the other assuming that the deposition can be described as the product of radionuclide concentration in precipitation and the amount of precipitation. The former has a meteorological basis for dividing global data into latitude bands and the quantitative estimate is based on the UNSCEAR compilation of deposition data, even though the original sources clearly state that the compilation is for the latitude band as such and should not be used as a model for individual sites. The latter has been used successfully for individual countries and regions, but the same parameters cannot be expected to hold for all conditions. The global model presented here uses the concentration function as a basis, but includes also latitude dependency and contribution of dry deposition through making the average annual deposition one of the parameters used in the model. The parameters of this new model have been determined using the data from the comprehensive global EML network and the model was validated with good results using data from other networks and data from the Nordic countries.

Reduction of radioxenon emissions from radiopharmaceutical facilities – A pilot study

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Abstract

The Comprehensive Nuclear-Test-Ban Treaty Organization (CTBTO) is building an International Monitoring System (IMS) in order to verify that the state signatories of the treaty fulfil their commitments of not performing any kind of nuclear explosion. The atmospheric noble gas monitoring is a part of this verification system and focuses on the measurement of short-lived radioxenon isotopes in the atmosphere. In order to improve the sensitivity of the IMS noble gas network, the radioxenon background should be decreased. Previous research has shown that a very limited number of radiopharmaceutical facilities are responsible for the major part of the radioxenon background [Saey 2009]. In addition xenon routine releases from such facilities can hide the release of radiological more important nuclides like iodine during on- or off-site gas monitoring. Reduction of radioxenon release will consequently decrease the global background and enhance the basic safety of such nuclear facilities.

This study reports on several techniques that could be installed in a radioisotope production facility to reduce the discharge of radioxenon in the atmosphere. This pilot study was centred on the Institute for Radioelements (IRE) facility in Fleurus, Belgium – the worldwide third largest Mo-99 producer. Each production step was analysed to determine the amount and the isotopic composition of a possible xenon release. Many techniques (e.g. adsorption on solid materials, cryogenic processes…) will be discussed in terms of performance and practical aspects for the filtration and/or delay of xenon fission gas emissions.

It has been demonstrated in this work that significant reductions of radioxenon atmospheric discharges from Mo-99 production facilities are theoretically possible. Nevertheless it requires a complete inventory of each pathway of xenon release during separation and purification steps of medical radionuclides.

Introduction

The Comprehensive Nuclear-Test-Ban-Treaty Organisation (CTBTO) is operating for verification purposes an International Monitoring System (IMS) based on four different techniques (seismic, hydroaccoustic, infrasound and radionuclide (particulate and noble gas) measurements) to detect if an explosion in nature is nuclear or not. The objective of the IMS is, according to the CTBT: "…At least 90% detection capability within 14 days after a nuclear explosion in the atmosphere, underwater or underground for a 1 kton nuclear explosion". The global monitoring of noble gases measures the activity concentration in air of four radioxenon isotopes (Xe-133, Xe-133m, Xe-135 and Xe-131m) with a limit of detection below 1mBq/m^3 for Xe-133 [Saey 2007]. However the global radioxenon background which has mainly an anthropogenic origin (nuclear power plants, hospitals and radiopharmaceutical facilities) is in certain areas up to 2 orders of magnitude above the detection limit of the IMS network. It has been demonstrated that the largest contribution to the radioxenon global background by far is coming from the four most important medical radioisotopes production plants that are producing 95% of the Mo-99 world demand by neutron irradiation of highly enriched uranium [Saey 2009]. Mo-99 is the precursor of Tc-99m which is used in 80% of the medical applications with radioisotopes involved world-wide. By consequence, reduction of the radioxenon discharges from these radiopharmaceutical production plants will decrease the global radioxenon background and thus increase considerably the sensibility of the noble gas monitoring network.

On the other hand reducing the radioxenon routine discharges of such kind of nuclear facilities can have another interesting impact regarding the nuclear safety of these installations. Even if the radioxenon nuclides are not important from the radiological point of view, the huge noble gas activity released can mask other more dangerous nuclides such as iodine during on- or off-site gas monitoring. Reduction of radioxenon emissions will consequently decrease the global background and enhance the basic safety of such nuclear facilities in case of an accident but also during the routine monitoring.

This pilot study of the reduction of radioxenon discharges coming from a radiopharmaceutical facility was focused on the case of the Institute for Radioelements at Fleurus in Belgium which is one of the largest Mo-99 producers.

Review of the noble gases retention-scrubber techniques

Xenon is part of the group of noble gases - its interactions with other elements or matter are, therefore, very limited. Except a few uncommon chemical reactions with strong reagents, xenon is chemically "inert". Xenon interactions with matter or other elements are limited to the van der Waals forces (more specific dispersion or London forces). Consequently only physical separation processes have been proposed and investigated for the recovery of radioxenon from gaseous effluents like adsorption on solid material, cryogenic distillation or diffusion through membranes. The cryogenic distillation process has already been studied extensively and tested for the treatment of krypton and xenon present in the off-gases coming from a nuclear waste reprocessing plant [Collard et al. 1982]. This technique is said to be promising, however it has been rejected because of the higher operational costs and the potential fire hazard caused by ozone accumulation [Geens et al. 1985]. Gas diffusion through thin polymer membranes technology has been already studied in the nuclear sciences field for the purification of a reactor building contaminated air with radioactive noble gases [Stern et al. 1980]. Despite the high efficiency of a multi-stage permeation system, this technique hasn't been selected because of its lower throughput and the sensibility of the thin membranes to the chemically reactive substances that could be present in the off-gas (e.g. iodine, volatile acids, nitrogen oxides…).

The adsorption process on activated carbons (A.C.) or zeolites is an easy-to-install and reliable process with limited expected operational costs. Several authors have already studied and tested this technique for the treatment of radioactive noble gases effluents coming from different nuclear installations [Moeller et al. 1981, Mondino et al. 2002]. Despite the fire hazard in presence of NO_x, activated carbons are a very effective type of adsorbents for the retention of xenon until the radioactive decay has reduced sufficiently the radioxenon activity at the output of the system. The off-gas from a radiopharmaceutical facility contains several radioactive gases (Xe, Kr, I and Rb) as well as non-radioactive gases (Xe, Kr, N_2, O_2, NO_x, CO_2, iodine and water vapour) that requires a multi component adsorption approach [Munakata 1999]. Moreover, a pre-treatment of the carrier gas will be necessary in order to remove the most important impurities (water vapour, iodine and carbon dioxide) that could interfere with xenon for the adsorption. Noble gas adsorption depends mainly on several parameters as nature and flow rate of carrier gas, temperature and pressure that can be optimized to obtain the maximal retention time for radioxenon. Finally, xenon breakthrough curve models taking into account the radioactive decay [Lee et al. 1971, Madey et al. 1981] or with an advanced description of the dynamic process [Munakata et al. 2001] can be used on a set of experimental data to extract some important parameters like effective adsorption coefficient and/or diffusion coefficients. In our study these models have been used as a prediction tool to determine the xenon retention time and the decontamination factor of a given adsorption system with the help from experimental data already published in the literature.

Pilot study → The Institute for Radioelements

The main radionuclides currently produced from irradiated uranium targets (up to 93% of U-235) at the IRE institute are Mo-99, I-131, Xe-133 and Y-90. About 15-20% of the worldwide production of Tc-99m is performed in the Fleurus radiopharmaceutical plant [Kidd 2008].

In order to understand the xenon discharges into the atmosphere by such a facility in detail, a complete analysis of the chemical processes of separation and purification of radionuclides is required. Furthermore, the quantitative simulation of fission products with the ORIGEN code based on target and irradiation specifications can allow us to estimate the amount of radioxenon released for each discharge pathway during the production.

Investigation of the xenon emissions during the radiochemical process of separation and purification of radionuclides

The chemical processes for the recovery and the purification of the medical radioisotopes play a major role in the dynamics of the radioxenon emissions in the atmosphere from a radionuclides production plant. For the clarity of the following discussion, the radioxenon released has been split in two categories:

- Xenon from dissolution is the xenon accumulated in the targets (by direct fission production or decay) at the time of dissolution and released during the targets dissolution step.

- Xenon from decay is coming from the radioactive decay of the other uranium fission product like iodine or tellurium after the dissolution step.

The isotopic ratio of these two categories is different with a important contribution to the xenon total activity of shorter half-life radionuclides (Xe-135 and Xe-135m) for the xenon from dissolution while Xe-133, Xe-133m and Xe-131m account for the major part of the activity released by radioactive decay of the other fission products. The general process taking place in the IRE institute for the separation and the purification of the medical radioisotopes is based on [Salacz 1985]. The flow sheet is shown in Fig.1 and the critical steps of the process in terms of potential radioxenon emissions are presented in yellow.

The target dissolution by a sodium hydroxide solution releases all fission gases that have been produced during the irradiation and the cooling time over a short period of time. There are about 30h between the end of irradiation and the beginning of the dissolution. Radioactive noble gases such as xenon and krypton account for the major part of the gaseous inventory and are carried out of the dissolver by a helium flow to a cryogenic trap at liquid nitrogen temperature for the recovery of Xe-133. An additional charcoal trap also working at low temperature is installed after the first one to reduce the radioxenon discharges in the ventilation system. The trapped xenon is transferred to MDS Nordion for purification and concentration by gas chromatography. The efficiency of the xenon recovery line and the purification step has been estimated by monitoring at 99% [Verboomen et al. 2009].

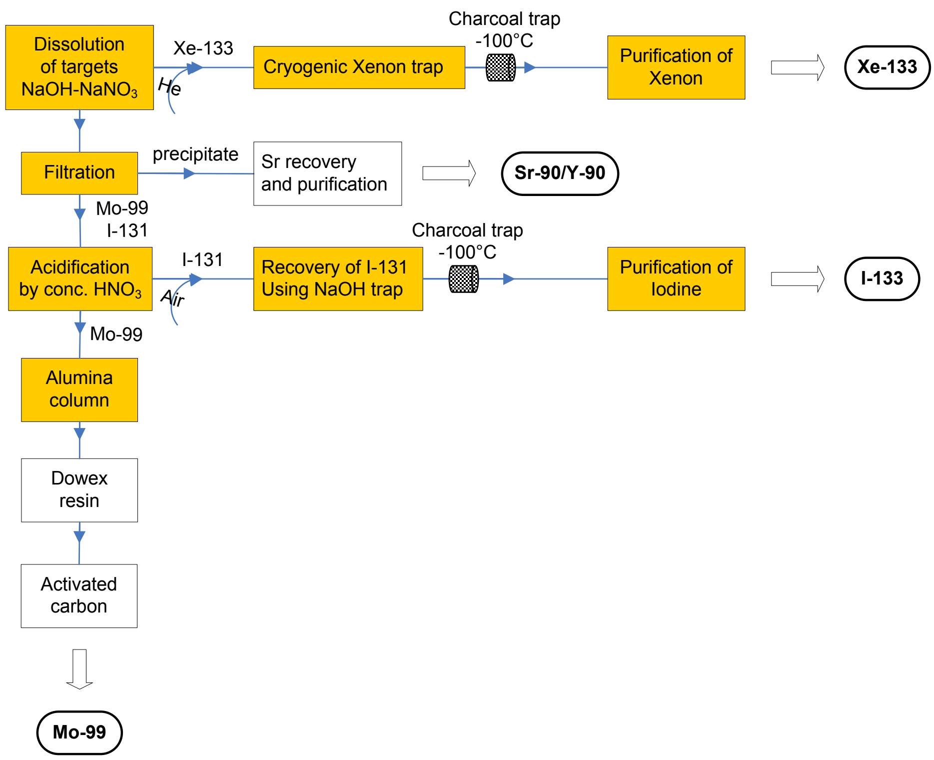

Fig.1. The process flow sheet for the recovery and the purification of Y-90, Mo-99, I-131 and Xe-133 medical radioisotopes at the Institute for Radioelements.

The precipitate containing the unfissioned uranium and some insoluble fission products like strontium is removed from the liquid phase by filtration. Radioxenon that is still present in the dissolver after the dissolution is released during this step. The next phase of the recovery and the purification of Sr-90 from the solid residue can be considered as free of radioxenon because of the time interval of years between the filtration and the beginning of the strontium production. However the precipitate continues to release radioxenon that is still contained in the solid material but also from the radioactive decay of iodine and tellurium traces.

The oxidation of I^- into I_2 by nitric acid induces the release of iodine in the gas phase. Iodine is trapped on a sodium hydroxide trap and stored until its purification. To reduce the discharge of xenon, a small charcoal trap has been installed after the iodine trap. The efficiency of the iodine recovery is maximum 90% [Salacz 1985]. About 10% of the total amount of iodine stays in the liquid phase and will be transferred on the first chromatography column for the continuation of the Mo-99 purification. The time between the end of the acidification and the beginning of the iodine purification is about 7 days for reaching the good isotopic ratio between I-131 and I-133. During this 7-days interval, the radioactive decay of iodine is producing continuously radioxenon (mainly Xe-133 and Xe-131m). The first alumina chromatography column of the Mo-99 purification step is also a potential source of radioxenon emission because iodine and tellurium traces are removed from the bulk solution and sent in the liquid waste tanks.

Finally, the liquid waste tanks and charcoal filters of the ventilation system that are containing iodine which has not been trapped or treated before, continuously produce some radioxenon isotopes by radioactive decay.

The profile of the radioxenon emissions from a radionuclide production facility is very complex because of the diversity of the emission sources in the process with a xenon isotopic ratio changing over the time. Contrary to a nuclear fuel reprocessing plant where the Kr-85 is mostly discharged all at once during the cutting and the dissolution of the fuel rods [Winger et al. 2005], the treatment of off-gases released only during the dissolution won't be good enough for obtaining a significant radioxenon decontamination factor. All the other critical steps in terms of potential radioxenon emissions in the process also have to be taken into account.

Estimation of the xenon source term using Origen code simulation

The quantification of the xenon contained in the uranium targets after irradiation has been done by using the Origen code simulation tool [Bowman et al. 1999]. The irradiation of the high enriched uranium targets used by the IRE is actually performed in three different research reactors: in the BR-2 reactor at Mol (Belgium), in the HFR reactor at Petten (the Netherlands) and in the OSIRIS Reactor at Saclay (France). The average irradiation conditions of the uranium targets in these three facilities are shown in Table 1.

Table 1. Input data for Origen calculations.

Target type	U-Al alloy (93% U-235)
Amount of U-235	4 g per target
Irradiation time	150h
Thermal Neutron flux	1.5 10^{14} n/cm^2.s
Cooling time	30h

Actually, 12 targets (48g of U-235) are irradiated and treated in one batch per production and there are 3 productions per week [Verboomen et al. 2009]. The time dependence of the activity for the main five radioxenon isotopes after the end of the irradiation is shown in Fig.2. The total xenon activity that is released from the targets during the dissolution step has been estimated to 438 TBq. If we consider an efficiency of 99% for the Xe-133 recovery and purification lines, a discharge of 4.4 TBq of "fresh" radioxenon (xenon from dissolution) per dissolution can be expected. The main contributions to this activity are coming from the Xe-133 and the short half-live xenon isotopes (Xe-135 and Xe-135m). This graph shows clearly that the xenon isotopic ratio of a discharge happening during the dissolution of the targets can be very different of one coming during the iodine purification step (7 days later). All the radioxenon discharges (puffs and constant releases) that are coming from different places in the facility at different times since the end of the irradiation merge together in the ventilation and reach the atmosphere by the same stack. The complete attribution of each signal detected by a single monitoring system at the level of the stack is virtually impossible because of the overlapping of all kind of radioxenon emissions.

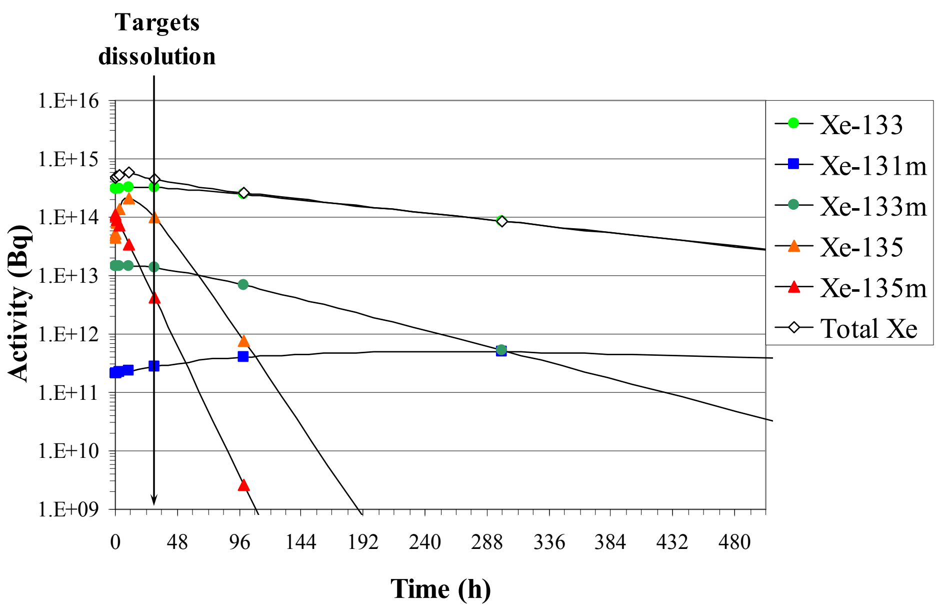

Fig. 2. Activity (Bq) versus time (h) for the main xenon radioisotopes produced during the irradiation period.

Another interesting point is to compare the contribution of the xenon activity released during the dissolution to the xenon activity produces by radioactive decay of the other fission products. The evolution of the xenon activity in function of time is shown in Fig.3. in terms of the sum of both types of xenon emissions. The xenon activity that can be released from fission products decaying is about 32 TBq after 100 hours. This contribution accounts for 12.5% of the total activity if all xenon is discharged at that moment. Although this part of radioxenon is generated over several days and probably not released during a short period of time, the amount produced from these decay chains is split during the acidification step between the first Mo-99 purification column (10%) and the purification of iodine (90%) (See Fig.1.). At the end about 10% of the xenon activity produced by radioactive decay of the fission products (3.2 TBq over 4 days) is sent to the liquid wastes storage facility. It illustrates clearly that an untreated xenon discharge can rapidly account for several percents of the total radioxenon emissions and thus can mask the effort put in treating other release pathways.

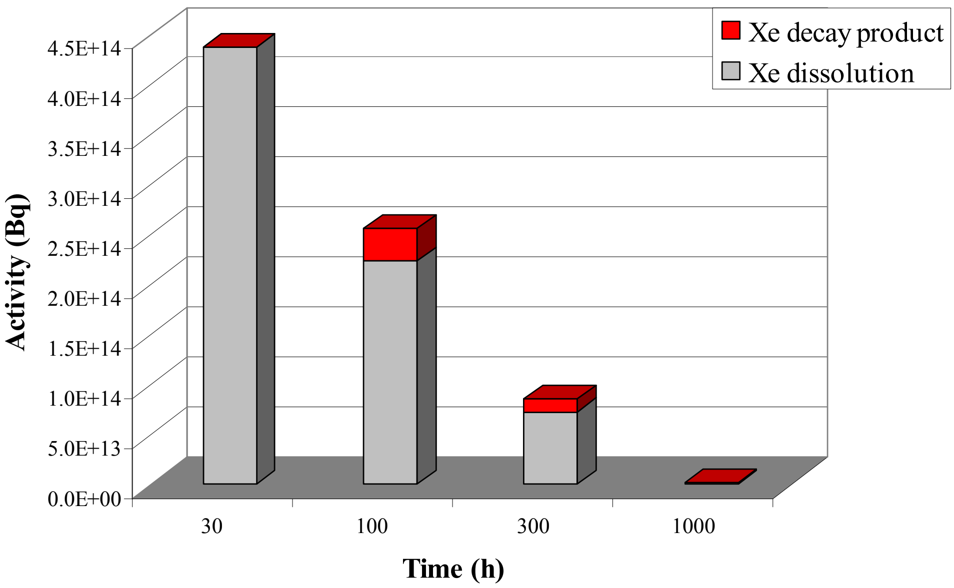

Fig. 3. Evolution of xenon activity (Bq) in function of time (h) (grey: Xenon released during dissolution; red: Xenon produced by radioactive decay).

Example of a xenon adsorption delay system

The scheme of the example presented below is shown in Fig.4. This system has been designed for the treatment of gaseous effluents with a relative small flow rate (up to 4l/min) and helium is used as the carrier gas to minimize its influence. Two preconditioning steps have been planned to remove most of the impurities before contact with the adsorbent bed. The first one is relative to the removing of nitrogen oxide by catalytic reduction with ammonia. The second is a pressure swing adsorption (PSA) system for the elimination of water vapour and the carbon dioxide. Four activated carbon packed bed columns (length=100cm, radius=10cm) installed in parallel are the adsorption part for the retention of the xenon.

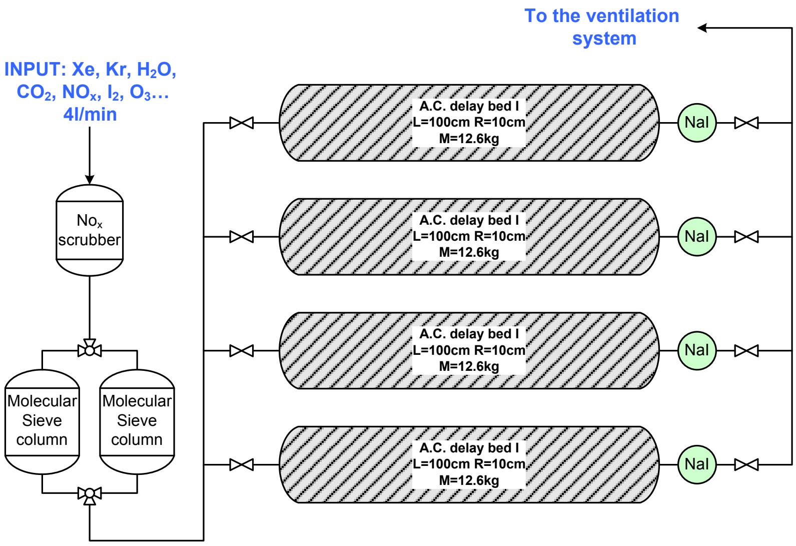

Fig. 4. Example of a xenon delay system for the treatment of contaminated gaseous effluents up to 4l/min.

A parallel multi-column adsorption system has two advantages:

- To reduce the linear velocity of the carrier gas in each column, which is an important parameter that can seriously affect the retention time;
- Increase the basic safety of the installation with the principle of redundancy.

The monitoring of the radioxenon activity is done by a sodium iodine detector on top of each column and each column can be isolated for decay if needed. The input and output parameters of the xenon breakthrough curve simulation, based on the experimental data of [Munakata et al. 2001], are listed in table 2.

Table 2. Input and output parameters for the breakthrough curve simulation of the example.

Parameters	Value
Flow rate	4 liters/min.
Temperature	195K
Column length	100cm
Column radius	10cm
Linear velocity	0.53 mm/s in each column
Bed density	0.40g/cm^3
Amount of A.C.	12.6kg/column → 50.4kg
Xenon partial pressure	100 Pa
Effective adsorption coef.	~6500 cm^3/g
Retention time	56.7 days
Decontamination factor (Xe-133)	> 1000

This example has shown that a small system with only 50kg of A.C. is able to treat a helium flow contaminated with radioxenon up to 4liters/min. We can imagine that each critical step, pointed out from Fig.1, can be treated with this system or that all the contaminated gaseous phases will be sent into a single tank connected to the delay system for a global treatment to reduce the operational costs.

From the radioprotection point of view, this kind of retention system can raise some problems for the protection of workers because of the high specific activities of short-life radioxenon isotopes and the important amount of radioxenon which is going to accumulate inside the adsorption beds.

Conclusions

The possible reduction of the atmospheric radioxenon discharges from a radioisotope production facility like the Institute for Radioelements at Fleurus, Belgium has been investigated. The review of the noble gases filtration/mitigation processes has pointed out that the adsorption on activated carbon is the most suitable technique for this purpose. The analysis of separation and purification steps of the radioisotopes, combined with quantitative estimation by using the ORIGEN code have shown that the dynamics of the xenon emissions are very complex and can not be attributed to only one process but to the entire process flow taking place in the facility.

A small adsorption system working at low temperature has been designed with four parallel activated carbon columns for the retention of radioxenon. Breakthrough curve models have shown that a decontamination factor of more than 1000 for Xe-133 can be expected. However the installation of such a system for the treatment of all possible sources of radioxenon from inside the facility could raise some problems of radioprotection due to the high radioxenon activity present.

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Public exposure by natural radionuclides in drinking water – Models for effective dose assessment and implications to guidelines

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Abstract

In Austria the legal framework to "Exposure by natural radionuclides in drinking water" is the Austrian Drinking Water Ordinance (Trinkwasserverordnung BGBl. II 304/2001) which implements exactly the European Drinking Water Directive 98/83/EC. The minimum requirements on the quality of drinking water and water intended for human consumption are appointed in it. For radioactivity two indicative standard parameter limits are established – tritium activity concentration of 100 Bq/l and total indicative dose TID (effective dose from radionuclides in drinking water except ^{3}H, ^{40}K, radon and radon progenies) of 0.1 mSv/a. The appointment and the evaluation of the TID are specified in the Austrian Standard OENORM S 5251:2005. Generally only the radionuclides ^{226}Ra and ^{228}Ra, dose conversion factors for adults and a yearly water consumption of 730 l are taken into account for dose calculation.

In the paper the estimation of the TID according to the drinking water directive and the OENORM standard is compared to dose estimations for other age groups and other nuclides based on measurements carried out in Upper Austria. The dose contributions of ^{210}Po and ^{210}Pb clearly preponderate the dose contributions of the radium isotopes. An alternative model for dose estimations has been developed, which takes into account a daily water intake and a continuous excretion of activity from the body. The presented dose assessment clearly yields lower annual effective doses for the population. Present regulations and guidelines for drinking water monitoring and surveillance are discussed and evaluated with regard to the results of this study.

Disagreement persists on methods and applied parameters for estimating total doses caused by natural radionuclides in drinking water within Europe, its individual countries and experts. This paper contributes supporting facts and feasibilities to yield a good basis for future guidelines.

Introduction

Drinking water is the most important food. Therefore its availability, quality and regulation are delicate and important topics. For this purpose it is fundamental to have an overview and hence reasonable regulations about natural radioactivity in drinking water. In the European Drinking Water Directive 98/83/EC (European Commission, 1998) a minimum requirement on the quality of drinking water and water intended for human consumption is appointed. For radioactivity two indicative standard parameter limits are established – Tritium activity concentration of 100 Bq/l and total indicative dose TID (effective dose from radionuclides in drinking water except ^{3}H, ^{40}K, radon and radon progenies) of 0.1 mSv/a.

Radon and the radon progenies are excluded from the European directive (European Commission, 1998). The European commission (European Commission, 2001) recommends for ^{222}Rn, that a reference level should be appointed above an activity concentration of 100 Bq/l, and with radon activity concentrations above 1000 Bq/l measures are justified. For the radon progenies ^{210}Pb and ^{210}Po the Commission recommends (European Commission, 2001) that above an activity concentration of 0.2 Bq/l and 0.1 Bq/l respectively, it should be tested whether any measures are necessary.

^{238}U were not taken into account in the European commission recommendations, but the World Health Organisation (WHO) recommends a guidance level of 15 µg/l natural uranium, which corresponds to a 0.19 Bq/l ^{238}U activity concentration in the drinking water guidelines (WHO, 2004). These guidance levels are applied to chemo-toxic effects of uranium, not on the radioactive exposure. The WHO defines guidance levels for several radionuclides in drinking water (artificial and natural) and says that no deleterious radiological health effects are expected from consumption of drinking water if the concentrations of radionuclides are below the guidance levels (equivalent to a committed effective dose below 0.1 mSv/a). This corresponds with the European directive 98/83/EC (European Commission, 1998).

For radon the WHO guidelines recommend that controls should be implemented if the radon concentration of drinking-water for public water supplies exceeds 100 Bq/l, which corresponds basically with the European Commission recommendation (European Commission, 2001) but is stricter.

All these recommendations afford high responsibilities of the countries to establish their individual and detailed limitations and regulations.

In Austria the legal framework for exposure from natural radionuclides in drinking water is the Drinking Water Regulation – TWV (Republic of Austria, 2001) which implements exactly the European Drinking Water Directive 98/83/EC (European Comission, 1998). The appointment and the evaluation of the TID are specified in the Austrian Standard OENORM S 5251 (OENORM, 2005). The required measurement techniques (e.g. detection limit), sampling site and the evaluation methods including examples are specified there. Generally only the radionuclides ^{226}Ra and ^{228}Ra with dose conversion factors for adults and a consumption rate of 730 l/a are taken into account for dose calculation of drinking water in Austria. Beside this standard there is a lack of regulation concerning other radionuclides e.g. ^{222}Rn, ^{210}Po and ^{210}Pb.

For taking of an inventory of radionuclides in drinking water a drinking water pilot study was carried out in Austria. Based on these measurements dose calculations were implemented according to the Austrian Standards and compared with other dose models and evaluated. The paper yields to provide a basis for further discussions because of disagreement on methods and applied parameters for estimating total doses caused by natural radionuclides in drinking water within Europe, its individual countries and experts.

Material and methods

A drinking water research project was carried out between 2004 and 2006 in Upper Austria (area: 12 000 km^2, population: 1.4 million) funded by the Government of Upper Austria in which 350 water samples were taken in water supplies, wells and at consumers' houses. All water samples were analyzed for different radionuclides (^{222}Rn, ^{226}Ra, ^{3}H, ^{238}U) and gross-alpha-beta-activity concentration by LSC and ICP-MS. A collection of the water samples were additionally analyzed for ^{228}Ra, ^{210}Po and ^{210}Pb by LSC. The detailed sampling methods and the project in whole is discussed in (Gruber et al., 2007, Gruber, 2009) and the detailed measuring procedures are also given in Landstetter & Katzlberger (2005).

For the analyzed radionuclides activity concentrations in the water samples (c_i) dose calculations were carried out. The total indicative dose (TID) was calculated according to the Austrian Standard OENORM S 5251 (OENORM, 2005). The total dose is the sum of the dose contributions of the single radionuclides (GD_i) (according to OENORM S 5251 basically only ^{226}Ra and ^{228}Ra), which are calculated from the activity concentrations (c_i) with the legal valid dose conversion factors ($h(g)_i$) for adults (age >17a) respectively and an annual consumption (KM) of 730 l/a (according to OENORM S 5251) (Formula 1). Activity concentrations below detection limit are set to zero. Uncertainties are calculated by error propagation without taking into account an uncertainty contribution of consumption and dose conversion factors.

$$GD = \sum_i GD_i = \sum_i h(g)_i \cdot c_i \cdot KM \qquad (1)$$

For the compliance of a reference value for the total indicative dose (RGD – e.g. the total indicative dose (TID) according to the drinking water directive (Republic of Austria, 2001)) the following requirement has to be proved, whereas ΔGD is the uncertainty of the total dose (Formula 2).

$$RGD \geq GD - \Delta GD \qquad (2)$$

The total dose (Formula 1) was additionally calculated regarding other measured nuclides (^{238}U) and also for nuclides excluded in the drinking water directive (European Commission, 1998) like ^{210}Pb, ^{210}Po. Besides the dose contribution of the different radionuclides and the total dose were estimated with dose conversion factors for other age groups (babies, children), because they are applied for dose assessments in some countries and this topic is still under discussion within the experts. The dose conversion factors are the values of the committed effective dose per unit intake via ingestion for members of the public at different age groups according to the Basic Safety Standards (IAEA, 1996, see Table 1). The annual water intake values were chosen by the Article 31 working party on radioactivity in drinking water after overview of different intake values by different organisations, including the WHO. The age group 12–17 a was not considered for dose calculation because of no defined water intake for that age group (Risica & Grande, 2000).

Table 1. Dose conversion factors for different age groups and nuclides (IAEA, 1996) and annual water intake (European Commission (1998), Risica & Grande (2000)).

Age group	Dose conversion factors (Sv/Bq)					Annual water intake (l)
	^{228}Ra	^{226}Ra	^{238}U	^{210}Pb	^{210}Po	
<1	3.0 E-5	4.7 E-6	3.4 E-7	8.4 E-6	2.6 E-5	250
1–2	5.7 E-6	9.6 E-7	1.2 E-7	3.6 E-6	8.8 E-6	350
2–7	3.4 E-6	6.2 E-7	8.0 E-8	2.2 E-6	4.4 E-6	350
7–12	3.9 E-6	8.0 E-7	6.8 E-8	1.9 E-6	2.6 E-6	350
(12–17	5.3 E-6	1.5 E-6	6.7 E-8	1.9 E-6	1.6 E-6)	
>17	6.9 E-7	2.8 E-7	4.5 E-8	6.9 E-7	1.2 E-6	730

Additional dose assessments were carried out based on a model by Bronzovic et al. (2006) taking into account a continuous daily water intake and a continuous excretion of the radionuclides from the body. Therefore the m(t) value according to IAEA (2004) was applied, which describes the fraction of a unit intake retained in the whole body at time t after intake. The major part of the radionuclide is excreted from the body within a few days after the intake (e.g. for Ra, Figure 1). A daily total radionuclide activity in the body was calculated for different radionuclides by a daily 2 litre drinking water intake with a constant activity concentration and a daily excretion specified by the m(t) value. The effective dose for the total body (without distinguishing between tissues and organs, related to a 70 kg reference person) was estimated taking into account the absorbed fraction, the energy emitted in the body by the radionuclides and the radiation weighting factors.

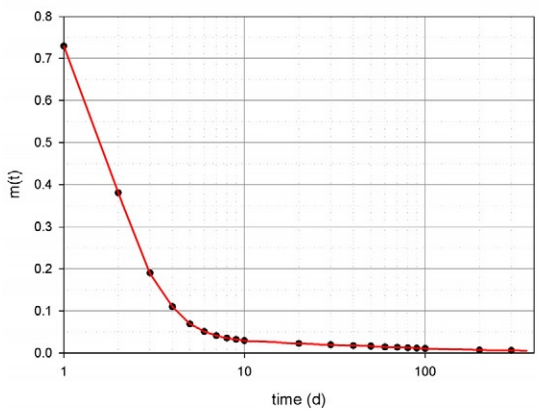

Fig. 1. The fraction of a unit intake retained in the whole body at time t after intake (m(t) value, for one year) for the radionuclides ^{226}Ra and ^{228}Ra.

Results and discussion

The calculated total dose according to formula (1) according to the Austrian standard OENORM S 5251 (OENORM, 2005) – considering only ^{226}Ra and ^{228}Ra is below the parametric value of total indicative dose (TID) of 0.1 mSv/a for all analyzed drinking waters in Upper Austria (Fig.2). Corresponding to the cumulative frequency distribution in Figure 2 for 90 % of the Upper Austrian drinking waters a total dose below 0.01 mSv/a is expected.

The dose calculations considering other nuclides show, that the dose contribution of ^{238}U is in general insignificant, but preponderate the dose contribution of ^{226}Ra and ^{228}Ra in some particular samples. The total dose caused by the radon progenies ^{210}Pb and ^{210}Po is clearly higher and 10 % of the water samples exceed the parametric value of the total indicative dose of 0.1 mSv/a (Figure 3). These radionuclides should therefore not be disregarded in guidelines and regulations for radiation protection purposes.

Only samples with both activity concentrations above detection limit (DL) are displayed in the figures.

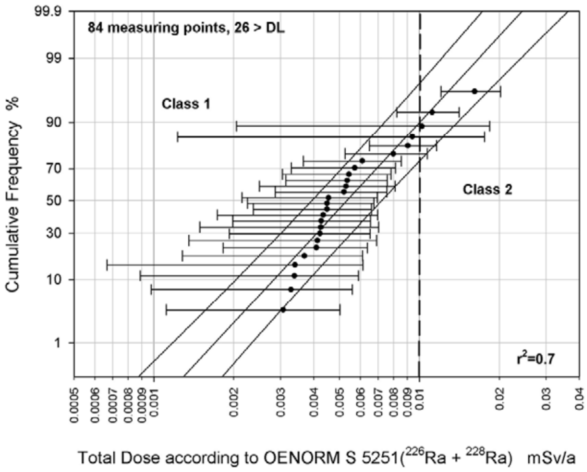

Fig. 2. Cumulative frequency of the total dose according to OENORM S 5251 (OENORM, 2005).

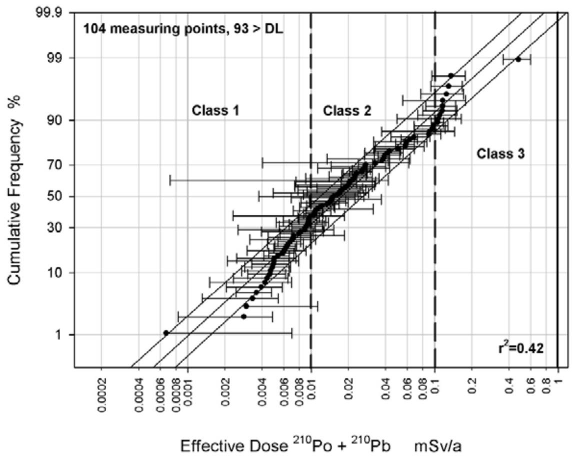

Fig. 3. Cumulative frequency of the total effective dose by ^{210}Po and ^{210}Pb calculated for adults.

The total doses for ^{226}Ra and ^{228}Ra are also clearly below the parametric value of 0.1 mSv/a for the other age groups (children). For about 10 % of the Upper Austrian drinking waters a dose above 0.1 mSv/a is expected for babies (< 1a) (Fig.4). No analyzed water sample has a dose caused by ^{238}U higher than 0.1 mSv/a for all age groups. The dose caused by ^{210}Pb and ^{210}Po for babies (< 1 a) exceeds 0.1 mSv/a for about 50 % of the analyzed Upper Austrian drinking waters. Also for the other age groups doses above 0.1 mSv/a occur for these nuclides (Figure 5).

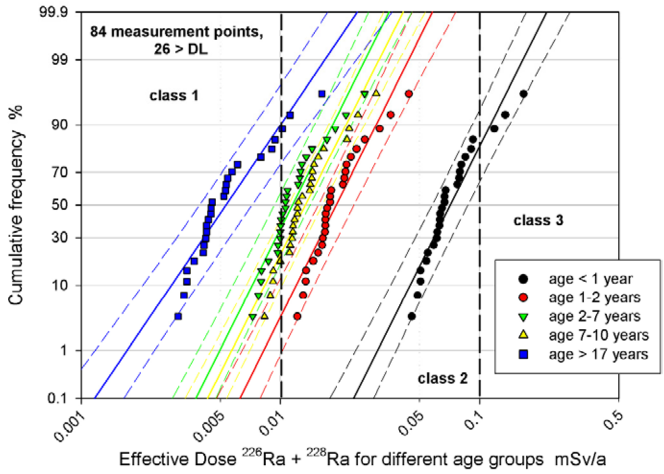

Fig. 4. Cumulative frequency distribution of the sum of the effective doses in drinking water caused by ^{226}Ra and ^{228}Ra for different age groups.

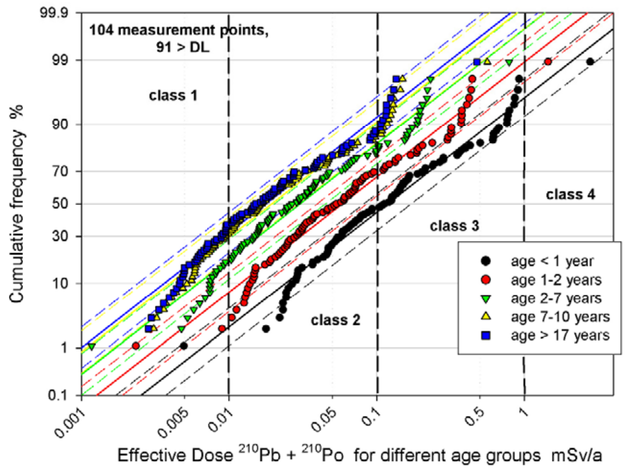

Fig. 5. Cumulative frequency distribution of the sum of the effective doses in drinking water caused by ^{210}Pb and ^{210}Po for different age groups.

Figure 6 illustrates the dose contribution of each nuclide to the annual total dose for adults (> 17a) estimated for each analyzed drinking water sample. For most of the samples the dose contribution of ^{210}Po dominates (up to 80-100 %) because of its high dose conversion factor (Table 1). The activity concentration of ^{228}Ra is below detection limit for most of the samples, but if the activity concentration is above detection limit, it contributes clearly to the total dose because of its high dose conversion factor (Table 1).

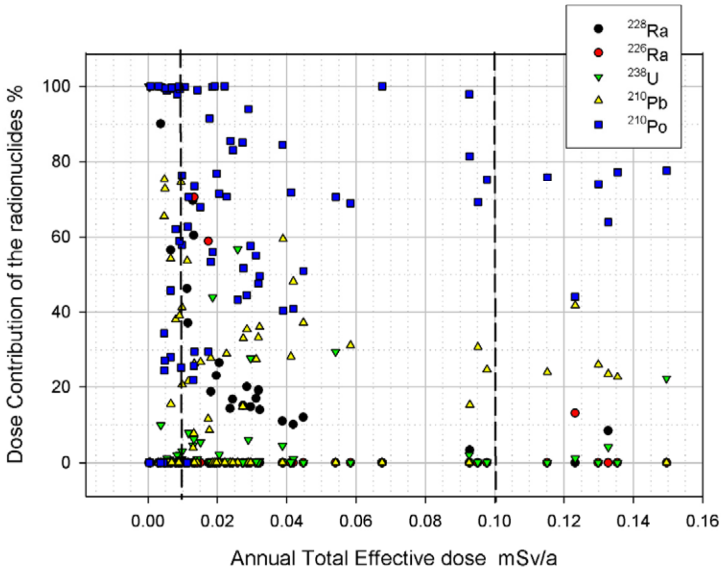

Fig. 6. Dose contributions of the nuclides to the annual total effective dose for adults (>17a).

The effective doses calculated with the above discussed alternative method, taking into account a daily water intake and a continuous excretion of the radionuclides from the body (m(t) value) are at least 2 magnitudes lower for adults than calculated with the conventional method (formular (1), Austrian Standards Institute, 2005) with the same activity concentration in drinking water. So it has to be proofed, if dose calculations according to formula (1) overestimate the radiation exposure of the public caused by radionuclides in drinking water and if the parametric value of the total indicative dose of 0.1 mSv/a is too conservative.

Conclusions

Dose assessments of radionuclides in drinking water are an issue of steady discussion within countries of the European Union and others and also within experts of one country. There are discussions about the legislative regulation and implementation of dose parameters or of reference activity concentrations. As discussed and explained in this paper, the European drinking water directive (European Commission, 1998) only states a total indicative dose and a tritium activity concentration, which was one-to-one adopted in various national laws (like Austria and Germany for example). For other radionuclides like the radon progenies ^{210}Po and ^{210}Pb only (activity concentration related) recommendations exists. As shown above the dose caused by ^{210}Po and ^{210}Pb is not negligible compared to the dose contribution of the radium isotopes. So these nuclides should be taken into account in applied regulations and dose calculations.

The survey in this paper should illustrate that in Upper Austria risk for the population caused by natural radioactivity in drinking waters only occurs in individual cases and for most of the Upper Austrian drinking waters no hazards for the population exist. Nevertheless drinking water should be controlled and surveyed regarding radioactivity, but drinking waters with little enhanced radioactivity concentrations according to different standards, recommendations and guidelines should not be set disabled for drinking water purposes without further surveys and dose assessments. For this purpose guideline values were developed in the framework of this project to simplify and standardize experts' activities in drinking water affairs. These recommendations are already adopted and published in the "Austrian food and drinking water codex", Codex alimentarius Austriacus, Chapter B, drinking water (BMGFJ, 2008) and are based on a differentiation between a monitoring level (indicative parameter of TID 0.1 mSv/a) and an intervention level of a total indicative dose of 1 mSv/a. If this intervention level is exceeded appropriate remedial actions should be recommended. In the codex also intervention levels for the activity concentration of ^{222}Rn (1000 Bq/l) and its progenies ^{210}Pb (2 Bq/l) and ^{210}Po (1 Bq/l) are included. The codex is not a regulation, but an important implement which proclaims terms and definitions, technical names and research in Austria. The worked out recommendation could also yield an extension of the Austrian drinking water regulations.

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Impact of facilities under the nuclear fuel cycle on the public health: SUE “Hydro Metallurgical Plant” (LPO “ALMAZ”) case study

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Abstract

Over the full period of the nuclear power engineering and industry development, comprehensive radio-ecological inspections are carried out in the Russian Federation to reveal and evaluate potential negative consequences of past activity of facilities under the Nuclear Fuel Cycle (NFC).

The Project No 3003 of the International Science and Technological Centre (ISTC) «Radiation Impact of the Facilities under the Russian Nuclear Energy Complex on the Environment and Human. Development of the Scientific Basis for Radiation Protection of the Environment» integrated efforts of several key international organizations involved in the radiation safety examinations. The Burnasyan Federal Medical Biophysical Centre under the FMBA of Russia (ex-Institute of Biophysics) is one of the participants of this project. The Centre accumulated the great experience in evaluation of the public doses originated from radioactive discharges and effluents; it also developed some methods for assessment of radiation impact on the human health. Following this work, the available results of the radiation situation monitoring have been summarized and analyzed; doses resulted from man-made radiation exposure to the public living nearby the Krasnoyarsk Mining Chemical Combine (MCC), LPO “Almaz” and Novovoronezh NPP have been assessed, and their contribution has been evaluated into dose due to the natural background at such areas.

At the same time, epidemiological inspection was being performed to identify potential connection between the radiation situation at the area of the facility impact and the public health. Data on incidence of malignant neoplasms have been collected over the period from 1991 to 2006 in respect to the residents of the areas studied. The special attention was paid to leukemia, which is the first response to over-exposure. The development took into account indexes of patients with cancer of lymphoid, blood-forming (hematoplastic) and their nested tissues, as well as cancers of other critical human organs. Finally, the database on epidemiological data has been arranged and treated.

Analysis of morbidity with malignant neoplasms helped to reveal significant differences in the morbidity statistics of male and female residents of Lermontov city

situated at the area of the LPO "Almaz" impact, in comparison with the control area in respect of common morbidity and incidence of the trachea, bronchus and lung cancers.

Material and methods

Three methods are used in epidemiology to answer the question about potential association between radiation situation at the area inspected and malignant neoplasms of the residents at this area. These are: *cohort examination*, when the morbidity risk of persons under radiation exposure is being evaluated in comparison with the control group; *case-control study* – risk connected with exposure is calculated by means of comparison of the diseased person group ("cases") with that of healthy individuals ("control") by their radiation exposure index; *territorial comparison*, when areas are chosen with different radiation exposure level.

Generally, case-control studies are less reliable than randomized control examinations or cohort ones.

The main requirement for application of the cohort or "case-control" study is availability of the personal dose data for each resident studied. Today, it is impossible to provide such kind of information. Therefore, we applied the method of territorial comparison in this work.

Obninsk city was selected as the control territory and its population was under examination. The town-forming facility is located in Obninsk – the Institute of Physics and Power Engineering (IPPE) equipped with the first Russian nuclear power plant – nevertheless, inspection aimed at risk assessment of leukemia incidence in the IPPE experts being carried out by V K Ivanov ea. [1], did not find any excessive risk, so the conclusion can be made on its absence for the rest residents of the city as well. In addition, «…long-term experience of nuclear facility operation in Obninsk shows that man-made radioactivity is low and does not impact significantly on doses to the public and environmental species» [2].

Having in mind that leukemia is generally the first response to over-exposure, our development included firstly indexes of the number of persons diseased with cancers of lymphoid, blood-forming and their nested tissues, as well as cancers of other critical human organs which are more frequent at the areas inspected.

The database arranged includes the areas being inspected over the period from 1991 to 2006. To improve the information validity, data have been treated by two-year cycles of study. For this purpose, all absolute indexes were being summed each two years.

Intensive indexes of morbidity were calculated according to the equation:

$$P=\frac{a \cdot k}{n}, \qquad (1)$$

where P – intensive index,

a – number of persons diseased,

n – number of the relevant serviced contingent,

k –for malignant neoplasms = 100 000.

The standard error is calculated for the intensive indexes (m) according to the equation:

$$m = \pm\sqrt{\frac{P \cdot Q}{n}}, \quad (2)$$

where m – standard error,
P – intensive index,
$Q = k - P$,
n – number of the relevant serviced contingent.

The standard error is included into the lists of the intensive indexes. Then, tables of morbidity indexes are being arranged for different nosologies and cycles of study.

To evaluate validity of difference between two indexes (the t – Student criterion), the mean error of this difference has been calculated, equal the square root from the sum of squares of the mean errors for these indexes

$$s.e.difference = \sqrt{(s.e.C_1)^2 + (s.e.C_2)^2} \quad (3)$$

If difference (C_1-C_2) is at least twice higher than its mean error, then some to the some extent we can say that the difference in the indexes is valid (non-random) and depends on some certain reason. If otherwise, this difference is less than twice higher than its mean error, then this difference is not valid (random).

The confidence probability specifies result reliability of the sampling study. We use 95% confidence probability in this work.

Results and Discussion

In the course of operation of the hydro-metallurgical plant (HMP), maximum doses originated from the current effluents took place near Ostrogorka village, subsidiary plots of which located near the border of the health protection zone. Table 1 shows annual effective dose to the residents of Ostrogorka village via each pathway of effluent-induced exposure over 1979 - 1991, being calculated according to [3].

Table 1. annual effective dose to the residents of Ostrogorka village originated from the current effluents, µSv.

Yar													
1978	1979	1980	1981	1982	1983	1984	1985	1986	1987	1988	1989	1990	1991
2,8	1,9	0,8	2,6	2,6	2,6	2,6	4,0	4,0	4,1	3,3	2,4	2,7	0,12

Data from Table 1 confirm that radionuclide release into atmosphere during the uranium ore milling at the HMP did not impact significantly on the public living to the leeward relating to the effluent sources. Internal exposure due to ingestion radionuclide intake via the local foods made the main contribution into annual effective dose to the public - 98 %. ^{210}Pb and ^{210}Po made the highest contribution into internal dose due to ingestion.

Effective doses of potential internal and external exposure to the public because of contamination of water in the surface ponds with the mine water

Self-leaked mine water from Gallery 32 at Beshtau hill passes to Zolotushka River and left tributary of Podkumok River through discharge 5. In summer, the residents use water from discharge 5 without purification to irrigate their gardens and vegetable gardens located close to the gallery. In addition, contamination of water reservoirs with mine water causes the external exposure to the public in the course of swimming and staying at the parts of contaminated floodplain.

The internal public doses have been evaluated hypothetically assuming that food intake received from the personal gardens was 211 kg/year for 1 person. In addition, in some cases small cattle is ranched at the subsidiary plots (goats, sheep, pigs, nutrias, rabbits, birds etc.), and fishing is undertaken in the ponds. Taking into account these factors of potential radionuclide intake via foods, the calculations included milk and meat intake at the 10% level of the value averaged over the Stavropol Territory, i.e. 20.3 l/year and 5.3 kg/year, respectively, while fish intake- at 5% level of the averaged value – 0.56 kg/year.

Table 2 includes assessment of effective internal doses to the critical population group due to food intake issued from the subsidiary plots, while Table 3 includes effective external doses originated from staying at the contaminated soils (plots) and swimming in the ponds.

Table 2. Internal public doses, mSv/year.

Radionucide	Fish	Potato and other agricultural products	Meat	Milk	Gross dose
^{238}U	1,6E-04	2,3E-03	6,1E-05	3,5E-04	2,9E-03
^{234}Th	1,1E-04	4,4E-06	1,2E-06	1,1E-07	1,1E-04
^{230}Th	1,5E-02	3,3E-03	2,1E-04	2,1E-05	1,9E-02
^{226}Ra	2,8E-03	7,3E-04	6,4E-03	6,0E-03	1,6E-02
Gross dose over all radionuclides	1,8E-02	6,3E-03	6,7E-03	6,4E-03	3,8E-02

For the part of residents who provide themselves fully with meat and milk from personal domestic animals, internal doses can reach 0,16 mSv/year. Internal doses to the critical group - «fishmen», who intake fish from the local ponds in 100 % volume, will be 0,39 mSv/year.

Table 3. External doses, mSv/year.

Radonuclide	Swimming	Staying at the area of the floodplain	Gross dose due to both exposure pathways
^{234}Th	1,5.10-6	2,5.10-9	1,5.10-6
^{230}Th	0	2,2.10-5	2,2.10-5
^{226}Ra	1,0.10-5	3,2.10-3	3,2.10-3
Gross dose over all radionuclides	1,2.10-5	3,2.10-3	3,2.10-3

In case of permanent living at this territory and full provision with the agricultural products, meat and milk originated from such personal plots, effective internal doses will be 1,7-2,5 mSv/year. At that, only 7-10 % of this dose are resulted from irrigation of the personal gardens with the mine water. External dose from radionuclides containing in soil will be 0,84 mSv/year.

The area of the village sitting is considered as radon hazardous territory. Additional exposure to the public can be resulted from inhalation intake of radon fission products, airborne in dwellings. Direct measurements of the radon EEC in the village are not being carried out. However, there are data on the neighbor Lermontov city. The more representative data [4, 7] are from the FMBC (ex-SRC Institute of Biophysics) experts under the FMBA of Russia and CSEN under MSCh-101. They carried out their measurements in 213 dwellings over the period since April 1995 till March 1996. The findings of inspections showed high EEC levels in the cottage-type houses, and revealed the season dependency of the radon EEC in dwellings. The EEC levels over the autumn-winter period were 2 and more times higher than those in summer. The evaluated annual effective doses to the residents of individual design dwellings were about 28 mSv. At that maximum doses about three times differed from the mean values. According to measurements of 2006, the mean radon EEC value was 230 Bq/m^3 in the cottage-type houses (85 measurements); this corresponds to about 17 mSv annual effective doses to the public.

Having in mind that the lung cancer is generally the first response to over-exposure of radon and its daughter products, our development included firstly the persons diseased with this nosology, as well as cancers of other human organs, which are more frequent at the areas inspected (Table 4)

Table 4. Distribution of persons diseased with malignant neoplasms over Lermontov and Obninsk cities by different nosologies and sex.

Sitting of malignant neoplasm	Sex	Totl number of diseases registered	
		Lermontov	Obninsk
1	2		
Malignant neoplasms – total including:	M	838	2528
	F	823	2955
Ventricle	M	75	343
	F	60	254
Trachea, bronchus, lung	M	240	458
	F	38	59
Breast	F	143	615
Lymphoid, кроветворнойblood-forming and their nested tissues	M	38	169
	F	30	162
Leukemia	M	21	70
	F	20	57
Leukemia except for CLL	M	12	51
	F	14	36

Figure 1 demonstrates dynamics of total morbidity with malignant neoplasms of the male residents of the cities by the examination periods

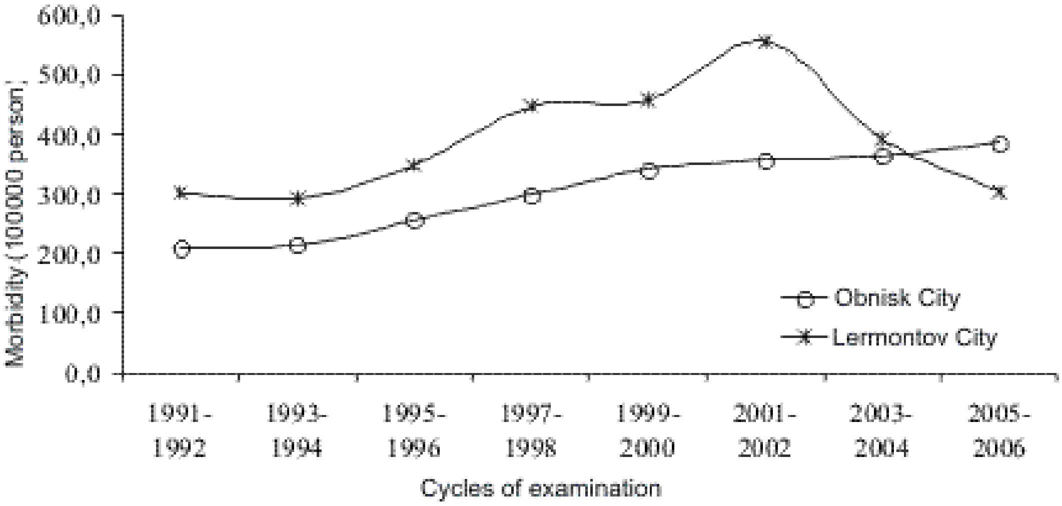

Figure 1. Dynamics of malignant neoplasm morbidity of male residents of Lermontov and Obninsk cities, by the examination cycles.

Such data demonstrate increasing index of common morbidity with malignant neoplasms of males with years in Obninsk city from 208,99 to 385,29 per 100000 males, i.e., 84 %. The indexes for the boundary periods for Lermontov city, such increasing has not been registered, but, for some cycles of examination, they reached rather high values – up to 554,59 per 100000 males over 2001-2002.

Over all periods of surveillance except for 2005-2006, male morbidity indexes for Lermontov are higher than the similar indexes for Obninsk city, and valid differences have been found for 6 cycles of examination of 8: 1991-1992, 1993-1994, 1995-1996, 1997-1998, 1999-2000, 2001-2002.

The most interesting is morbidity with cancer of trachea, bronchus and lung of males over Obninsk city increased with years a bit, while over Lermontov city it has increased up to 2001-2002 inclusively, reaching 153,13 per 100000, but over two last cycles of examination it decreased to 85,5 and 72,6 respectively.

Over each cycle of examination, the male morbidity indexes over Lermontov city were higher in comparison with those over Obninsk, and differences in the indexes were valid, except for two last cycles of examination (2003-2004 and 2005-2006).

The morbidity trend with cancer of lymphoid, blood-forming and their nested tissues was unstable both for Obninsk and for Lermontov. It has increasing tendency for Obninsk data, while for Lermontov it firstly increased, then decreased up to 4,27 per 100000 males over the last cycle of examination (2005-2006).

Comparison of indexes by cities showed that generally there were not valid differences by the cycles of examination.

Male morbidity with leukemia in Obninsk was wavelike. As for Lermontov, increasing morbidity is registered with years of inspection; rather high indexes of morbidity have ben registered over 2003-2004: it reached 27,01 per 100000 male residents of the city, nevertheless over the next period, 2005-2006, there were not registered leukemia events among them.

The most interest can be connected with data on leukemia morbidity except for the CLL, because they confirm unhealthy radiation impact of the environmental factors on the population. Over males in Lermontov, such kind of pathology has not been registered in 3 cycles of examination.

In two cycles (2001-2002, 2003-2004 гг.), the morbidity indexes over Lermontov city were higher than the similar values over Obninsk, but number of diseased persons was small, so differences in the morbidity indexes were invalid.

Thus, of all examined indexes of male morbidity with malignant neoplasms over Lermontov city in comparison with the Obninsk population, higher total cancer morbidity has been found as well as morbidity with the trachea, bronchus and lung cancers.

The female cancer morbidity in all malignant neoplasms both over Lermontov and over Obninsk generally trends for increasing with years of examination. It became 68 % higher for Lermontov, and 78 % for Obninsk.

Figure 2 demonstrates the comparative assessment of malignant neoplasm morbidity for females of Lermontov and Obninsk cities by the cycle of examination (per 100000 females).

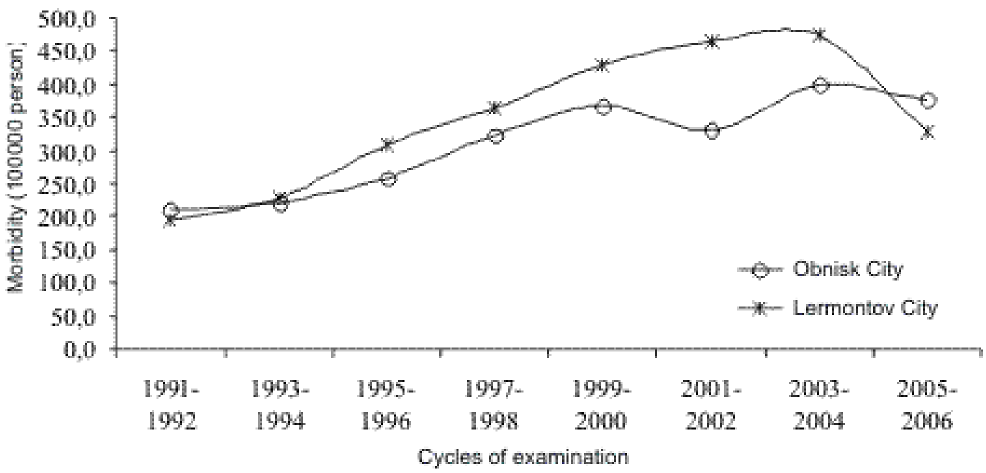

Figure 2. Dynamics of the female malignant neoplasm morbidity over Lermontov and Obninsk cities, by the cycles of examination.

Over some cycles of examination, the morbidity over Lermontov is a bit higher than over Obninsk, except for the last period. Differences in common malignant neoplasm morbidity is invalid, except for 2001-2002 period, when such difference is significant. Recently, the morbidity in Lermontov became lower in comparison with the Obninsk one.

Morbidity with the trachea, bronchus and lung cancers in both cities varies over the cycles of examination; this is more evident for Lermontov.

Over all cycles of examination, indexes for Lermontov are higher that for Obninsk. Differences in indexes are valid in two cycles – 1995-1996 and 2003-2004.

Comparative assessment of the breast cancer of females in Lermontov and Obninsk by the cycles of examination demonstrates their unstable difference by the cycles of examination.

For Obninsk, the morbidity increased slowly from 45,2 to 98,1 per 100000 females, while for Lermontov it reached maximum of 95,7 (period 1999-2000) and decreased significantly over the following cycles of examination up to 25,3 (2005-

2006). Regardless the fact that over two cycles (1997-1998 and 1999-2000) the breast cancer morbidity over Lermontov was higher than the similar index over Obninsk, this difference is invalid.

The female morbidity with the lymphoid, blood-forming and their nested tissues cancers is instable by the cycles of examination in both cities.

There were no valid differences of such kind of pathology over cities.

The excessive leukemia morbidity has been registered of females in Lermontov in comparison with that of Obninsk over 2001-2004, but because of small number of female diseased with leukemia this difference in indexes is invalid.

The female morbidity with leukemia except for the CLL has the same trends as the leukemia morbidity, but at the lower digital level, and over three cycles of examination in Lermontov such kind of disease has not been registered at all.

Thus, comparative analysis of female morbidity with all forms of malignant neoplasms in Lermontov demonstrated exceeding of this index in comparison with the similar index for Obninsk. Valid difference between them has been found only for the cycle of examination 2001-2002. The trachea, bronchus and lung cancer morbidity is notable in Lermontov, where in two cycles of examination (1995-1996, 2003-2004) in validly differs from that for Obninsk.

The above mentioned enables to make the following conclusions:

Morbidity with malignant neoplasms (by all analyzed forms) of residents in Lermontov city, especially the trachea, bronchus and lung cancer is higher than the similar indexes over the control area.

Excessive morbidity with malignant neoplasms (common by the analyzed forms) of the Lermontov population, especially the trachea, bronchus and lung cancer in comparison with the similar indexes over the Obninsk population can be explained by radiation exposure originated both from excessive contents of airborne radon and its daughter decay products during work of males in mines, and from living and staying in dwellings and public buildings being constructed over 1950-70s using the constructive materials with excessive contents of natural radionuclides.

Conclusions

Dose evaluation of man-made radiation exposure to the public living at the LPO "Almaz" impact area demonstrated that maximum effective dose via the meat and milk chain is 2,5 mSv/year. ^{226}Ra is the main dose-forming radionuclide in both cases; its contribution for both exposure pathways exceeds 90%. Maximum potential internal public dose due to intake of foods being produced at the subsidiary plots and gardens varies from 37 μSv/year to 470 μSv/year. On another hand, results of radiation survey in the city demonstrated the presence of high levels of radon and its progenies contents in dwellings of the city, especially in the cottage-type buildings. The evaluated annual effective doses to the public living in the individual design houses were 28 mSv over 1995-1996; according to measurements of 2006 - 17 mSv. Comparative analysis of doses demonstrates prevailing of the natural component above the man-made.

The above mentioned permits to conclude that inhalation intake of radon and its decay products is the main radiation factor affecting the public health in Lermontov city.

The analysis of malignant neoplasm morbidity helps to find some statistical domination over male and female residents of Lermontov in comparison with the similar index over Obninsk city in respect to common cancer morbidity and morbidity with the trachea, bronchus and lung cancer.

We think that such domination of the morbidity with malignant neoplasms of residents in Lermontov is induced by ionizing radiation due to radon release from the earth surface and due to residence and staying in dwellings and public buildings being constructed over 1950-70s using the constructive materials with excessive contents of natural radionuclides, that is confirmed by the higher level of the female morbidity with the trachea, bronchus and lung cancer in Lermontov [2, 5].

In addition, in past, males have been subjected to direct radiation exposure when mining the uranium ore under underground conditions, especially over the first after-war years, when work conditions of miners were being specified by excessive level of the quartz containing dust, radon and its progeny [6]. Taking into account long latent period in case of the lung cancer incidence, these factors could result in the increased morbidity of the population with such pathology.

Nevertheless, in data of the state medical report, it is impossible to identify miners in common population of Lermontov under service, therefore in the course of continuing study, it is reasonable to develop the register with personal data.

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Modelling with a CFD code the near-range dispersion of particles unexpectedly released from a nuclear power plant

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Abstract

An event in November 2007 in Ascó-1 nuclear power plant (Spain) originated the release of a significant amount of hot metallic particles through the discharge stack. Particles were dispersed and deposited in roofs and neighbouring areas within the NPP controlled area. However, the event was not detected until March 2008. More than 1,300 hot points with radioactive particles were found, 94% located inside the double fenced controlled area and 6% within the exclusion area; 5 particles were out of the exclusion area, across the river.

To provide additional insights on the potential consequences of the release, a computational fluid dynamics (CFD) code, Ansys-CFX-11, has been used to simulate the near-range atmospheric dispersion and deposition of the particles. The purpose of the analysis was to assess the distance travelled by particles of different sizes. A very detailed model of the site was built, taking into account the buildings and the terrain features including the river valley and the surrounding hills. The modelled domain was 3.2 x 5.2 km, with the atmospheric layer up to 4 km height. The atmospheric conditions recorded during different periods of time were classified into 37 representative categories.

In general, the distribution of the particles found was adequately reproduced. Particles larger than 100 microns could not travel beyond the double fence. Particles between 50 and 100 microns could have been deposited mainly within the exclusion area, with a small probability of travelling farther. Smaller particles could have travelled beyond, but also should have been deposited in the nearby area, while the majority of particles found are larger, thus indicating that the size of the released particles should be above 50 microns.

The detailed CFD simulation allowed answering relevant questions concerning the possibility of having an impacted region larger than the exclusion area.

Introduction

An incident classified as level 2 on the INES scale happened at the Ascó I nuclear power plant in Spain, consisting of the release of radioactive particles with activated corrosion product isotopes. This occurred due to the contamination of the fuel building ventilation system with water originating from the cleaning of the fuel transfer canal at the end of the refueling outage of the reactor, as a result of a combination of incorrect practices and noncompliance with the operating standards (CSN, 2009a).

The detection of the release and its subsequent notification took place over four months after the occurrence of the event, since it became evident not because of the available automatic radiological control systems but through a site radiological surveillance walkthrough. This was due mainly to the fact that these systems are designed to detect homogeneous radioactive emissions and not discrete particles such as those involved in the event. On March 14th 2008, hot particles were first detected in the containment hatch area. A further increase in radiological surveillance activities in the following days lead to discover several hot points on the roofs of the buildings adjacent to the NPP stack (see figure 1). On April 4th a report was released to the regulatory authority, the Nuclear Safety Council (CSN), which was followed by press releases and official statements to the public, as well as a wide campaign to check more than 2,700 persons through the whole body radiological counter, including workers and visitors. No person was found contaminated. A team of experts from the European Commission's General Directorate of Energy and Transport visited Ascó on April 29th and verified the radiological protection control methodology which confirmed the non-radiological significance of the event and endorsed the technology employed to guarantee the control measures from the operative, administrative and quality points of view.

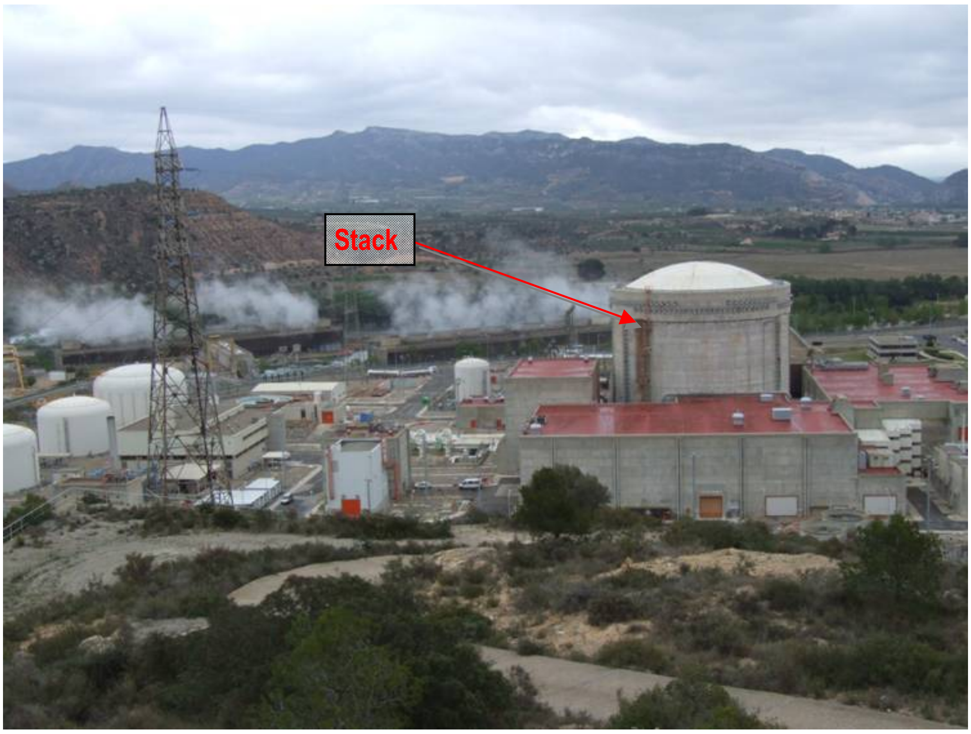

Fig. 1. Photograph of the Ascó I reactor building and the adjacent buildings of the nuclear power plant. The release of particles took place through the stack.

The event was investigated and the conclusion was that the release of hot particles to the atmosphere started on November 29th 2007, when the ventilation system was switched from filtered mode to normal mode (without filtration). As a consequence, particles were dragged out through the stack and then dispersed via the stack to the roofs of Unit I buildings.

An exhaustive active particle location programme was soon accomplished on the plant site, by the licensee in the area under its control and by the CSN in off-site areas, with more than 1,300 particles collected with a total activity of 409 MBq, subsequently calculated on November 26th 2007 (CSN, 2009b). As a comparison, the cleaning of the ventilation system allowed to recover a total of 37,6 GBq. 94% of the particles were located inside the double fenced controlled area and 6% within the exclusion area; 5 particles were out of the exclusion area, across the river (figure 2).

To provide additional insights on the potential consequences of the release, a computational fluid dynamics (CFD) code, Ansys-CFX-11, has been used to simulate the near-range atmospheric dispersion and deposition of the particles. The purpose of the analysis was to assess the distance travelled by particles of different sizes (and activities) and the probability that they have been deposited at a given location.

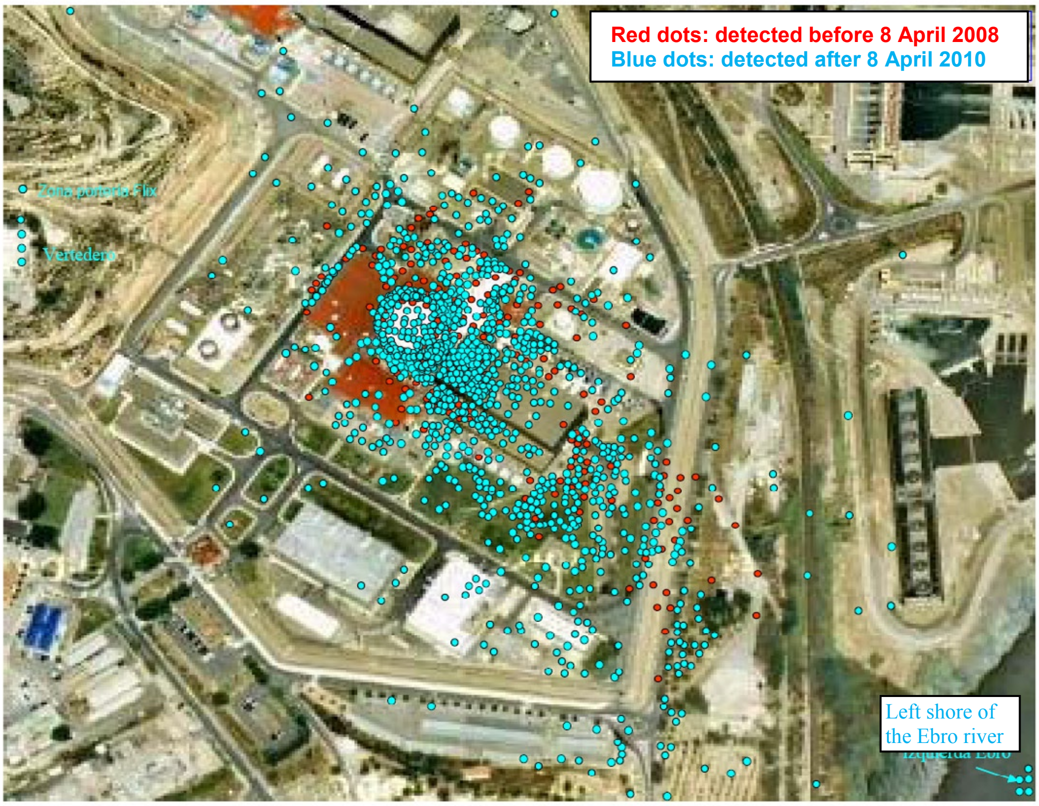

Fig. 2. Aerial view of Ascó I site showing the locations where hot particles were collected.

Material and methods

The modelled fluid flow presents two well differenced phases: a continuous phase of air mixed with steam and a dispersed phase, constituted by solid particles. Consideration of steam was necessary due to the presence of the forced and natural flow cooling towers.

Turbulent phenomena in the air flow have been included in the simulation by means of the SST model (Shear Stress Transport), widely used for industrial applications where limit layer effects in contact with surfaces are relevant. Therefore, the flow characterization is performed by solving the three equations of momentum for gases in x, y and z; the continuity or mass conservation equation; the energy conservation equation; the two equations of the turbulence model: for turbulent kinetic energy and frequency of turbulent structures; and the transport equations for steam. Also, the equations relative to particle transport which are solved by means of a lagrangian model coupled to the fluid flow model (one-way coupling).

A very detailed numerical model of the site was built, taking into account the buildings and the terrain features including the river valley and the surrounding hills. The modelled domain was 3.2 x 5.2 km, with the atmospheric layer up to 4 km height. The atmospheric conditions recorded during different periods of time were classified into 37 representative categories.

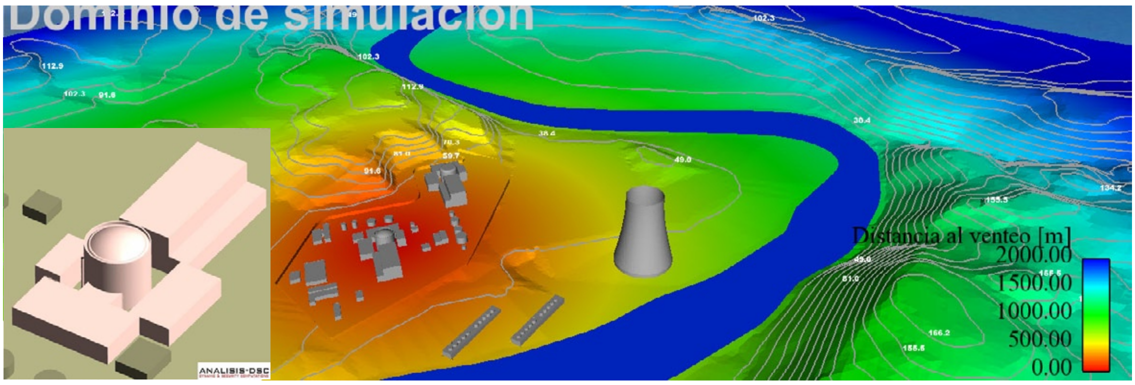

Fig. 3. Overall view of the geometric model of Ascó I site. On the left, a detail of the main buildings of the plant.

The steps followed in the study were the following:

1. Generation and adjustment of a 3-D detailed model for Ansys-CFX based on the terrain elevation (digital map of the area) and the dimensions of the buildings. The building geometries are less detailed for those that are farther from the release point. The general view of the geometric model can be seen in figure 3. The nodalization and mesh structure of the model is based in tetrahedrical finite volumes complemented by prismatic volumes near the surfaces. It is shown with some details in figure 4. Near the surfaces, the size of the cells is small enough as to adequately capture the behaviour in the interface.
2. Analysis of the meteorological data recorded at the site between 29 November 2007 and 31 January 2008. Data have been recorded at 10, 24.5 and 60 m above ground at 15 minutes intervals. The analysis of these data has lead to classify the different atmospheric conditions in a total of 37 categories as reasonably representative of the local meteorology during the period under study. There was a compromise between the need for realistic calculations and the computational resources needed for each simulation. In practical terms, the combination of these 37 categories, with adjusted frequencies, allowed to reasonably represent the atmospheric conditions in the following periods: 29/Nov/2007; from 29/Nov/2007

to 31/Dec/2007; from 29/Nov/2007 to 31/Jan/2008 (one day, one month and two months from the change in the ventilation system to non-filtered mode).

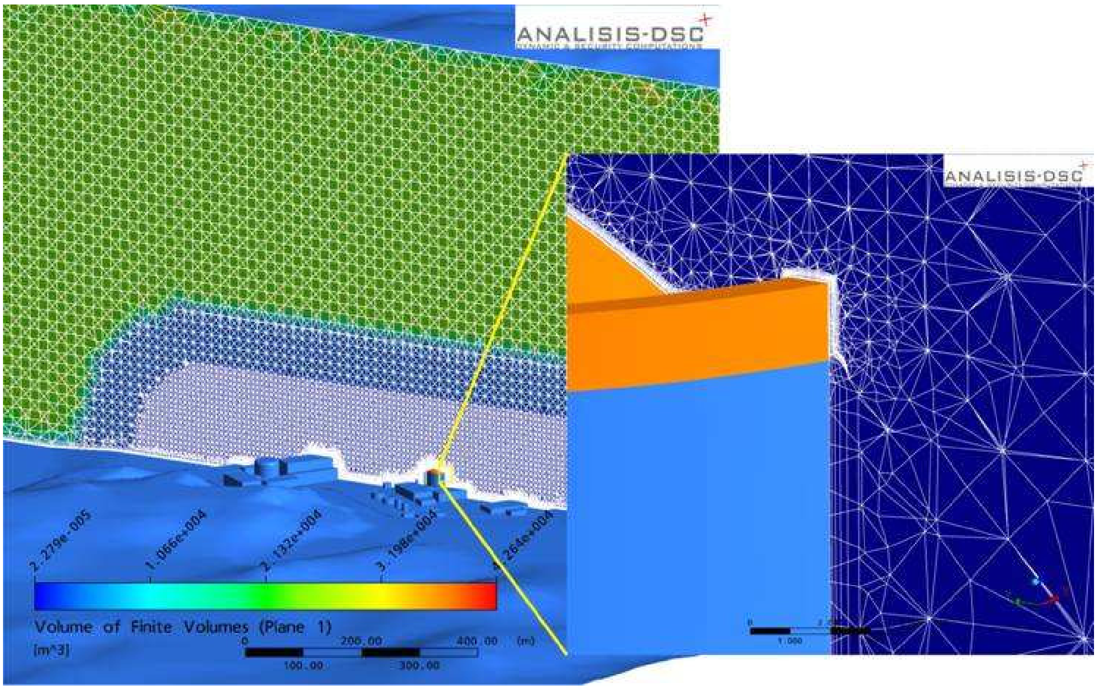

Fig. 4. Details of the nodalization of the atmosphere around the reactor building and in the mid-range distances along the site and up in the atmosphere.

3. Simulation of the 37 categories, with specific conditions of temperature gradient and atmospheric stability, relative humidity, wind speed and direction. The temperature gradient chosen was the typical for the atmospheric stability (Snell, 1994) and the most frequent temperature was chosen as representative for each category. Wind speed profiles with height were adjusted by considering a potential law dependent on the atmospheric stability (Hanna, 1982). The air flow released from the stack is 37 m^3/s with a temperature 20ºC. 3-D effects in the air flow around the site were very relevant, due to the irregular terrain features and the influence of the cooling towers.
4. Seven classes of particles were taken into account in the simulations: 1–25 μm; 25–50 μm; 50–75 μm; 75–100 μm; 100–150 μm; 150–250 μm; 1–250 μm. For bigger sizes, their behaviour is dominated by inertial forces and their distribution is similar. Larger particles could not leave the stack, as demonstrated with a preliminary study of balance of forces in the released flow. The particles density was taken as 7 g/cm^3, as it corresponds to metallic compounds from corrosion of the reactor primary cooling circuit.
5. Parametric study to get a probability of deposition of particles of different size in a given zone, taking into account the atmospheric dispersion. Combination of results for each of the 37 categories with their frequencies during each time period.

Results

The particle deposition map (fig. 2) is the result of the atmospheric dispersion and deposition of the particles released through the stack plus the later processes of resuspension and deposition by wind, transport by rain water runoff and other weathering factors which cannot be simulated in the model. It is therefore reasonable to see some differences with respect to the calculated deposition patterns. It is also necessary to remind that the collection of particles started in April 2008, while the release took place from 29 November 2007.

In order to give a useful representation of the deposition pattern, the number of simulated particles was 10,000 for each size range of 25 microns. To represent it, we have multiplied it by 100 so we have a number of 10^6 particles and the graphical representation displays the number of particles deposited (per m^2) per million particles released. The minimum representation limit is 1 particle per m^2. For the global case comprising 1–250 μm the number of particles assumed was 10^7.

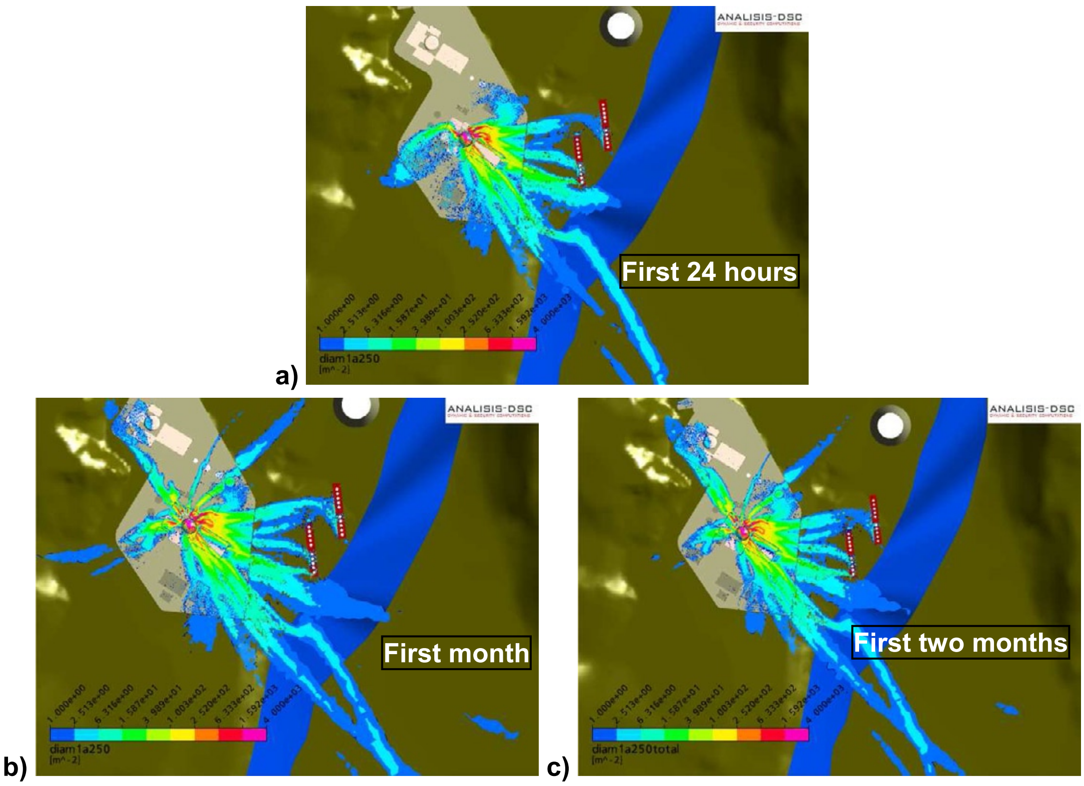

Fig. 5. Results of the simulated deposition of particles of diameters 1–250 μm considering the atmospheric conditions in three different time periods. The patterns represent the particle density per m^2 assuming a release of 10^7 particles.

There is a preferential deposition in the South East area, partly due to the higher frequency of winds in that direction, combined with the effect of the forced flow cooling towers which force humid air to go upwards and which cause some entrainment of particles in that direction. By comparing the three periods considered in the calculations, the conclusion is that a release during the first 24 hours (fig. 5-a) cannot explain alone the pattern of particles found. However, the deposition reached in the period from 29 November 2007 to 31 December 2007 (fig. 5-b) is very similar to the one if the period extended up to 31 January 2008 is considered (fig. 5-c). This result suggests that the release could have taken place most likely in the first month after the change of the ventilation system to normal mode.

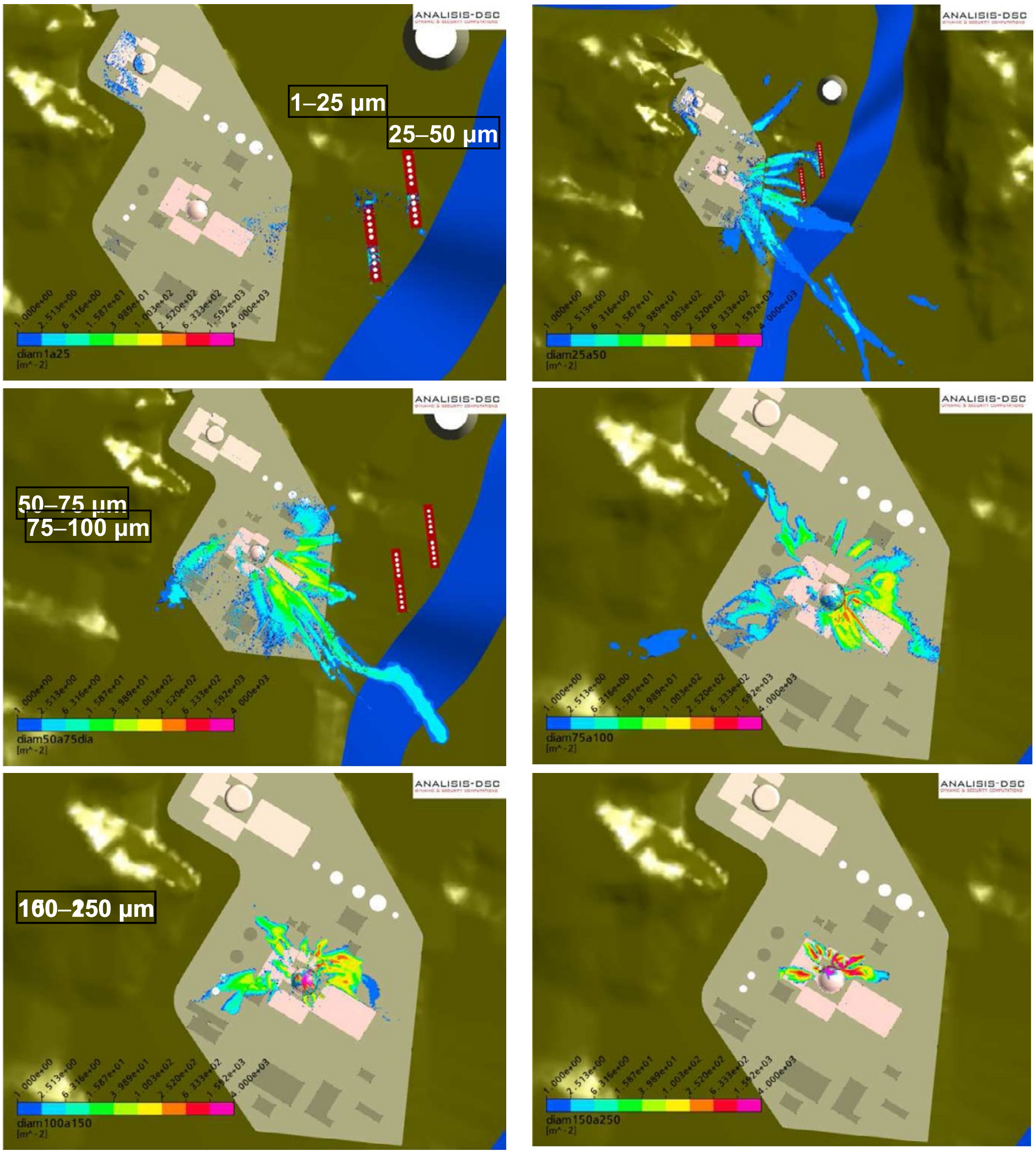

Fig. 6. Results of the simulated deposition of particles of different diameter ranges considering the weather conditions during the first month after the change of the ventilation system to normal mode (29/Nov/2007 to 31/Dec/2007). The patterns represent the particle density per m^2 assuming a release of 10^6 particles.

Fig. 6 displays the deposition patterns of particles of different size. Small particles (1–25 µm), of very low activity compared to the large ones, would be able to travel out of the calculation domain. They could also deposit, in small concentrations, around the stack and in other points of the site, with influence of the forced flow cooling towers.

However, in the radiological survey such small particles were not found, and therefore this result suggests that, if present, their fraction in the total should have been extremely low.

Particles of larger size (25–50 μm) show a distribution through a very large area, and their deposition density should be very small, with a likely deposition within the site, but also off-site. A significant fraction of such particles is leaving the calculation domain.

For particles with size between 50–75 μm, the larger fraction should be deposited within the site, although some of them could travel a bit out and cross the river. This could explain why 5 particles were found in that zone after a careful radiological survey. The radial-like deposition pattern seen in the figure is a reflection of the way in which the calculation has considered wind directions, with fixed angle in each category and a weighted combination of the results obtained.

When larger particles are considered, with sizes between 75–100 μm, they hardly seem able to leave the site, and they are deposited mainly along the dominant wind directions, towards E and SE as commented above.

Particles with size between 100–150 μm would be totally deposited within the plant fence, dominantly towards E and NE, because of the influence of the buildings in the local wind flow patterns. The majority of particles found did have sizes of this range, which could have been altered with time due to the particle "life" in the environment. In fact, many particles show a composition which is not purely metallic but associated with carbonates or silicates.

The biggest particles, with diameter larger than 150 μm, relatively very heavy and active, would deposit totally in short distances around the stack, many of them in the roofs of the reactor building and the surrounding buildings: auxiliary equipment building; fuel management and storage building; turbine building. Some of them would be trapped by the main wind flow and be able to go beyond the buildings, but certainly not far from the plant due to their high inertia. Later, resuspension or runoff phenomena could have transported them farther.

Discussion

This study had as starting point the distribution of the deposited particles, after about four months of weathering in the site. However, it has been affected by some uncertainties impossible to exclude; two of them were fundamental:

- Knowledge of the precise moments at which the release of particles took place.
- The particle size.

The study has tried to overcome these uncertainties by undertaking a wide parametric study covering a full range of particle sizes, from very small to the largest particles able to exit through the stack at the existing flow dynamic conditions, together with a variety of atmospheric conditions, 37 in total, which covered a high percentage of those existing during the likely emission periods.

Based on those considerations, the study has given valuable information with regard to the likely deposition pattern of particles of different sizes in different release periods, concluding that particles of all sizes could have been found within the inspected area where the hot particles were found. Particles larger than 100 μm could have not travel beyond the fenced area of the plant. A small fraction of particles sized between 50–75 μm could have leave the fenced area travelling towards the SE direction, where some particles were effectively found across the river. Small particles could in principle have travel far from the site, but they should also have deposited on site, and this was not the actual finding. Given their origin and the spread of particles found it is not very likely that these small particles were abundant in the release.

Conclusions

The main conclusion is about the usefulness of this study in order to better assess the radiological importance of the event. The great capability of CFD models to simulate very local effects –like the flow perturbation by the forced cooling towers, the buildings of the plant or the surrounding hills–, has proven essential to accurately interpret the behaviour of particles of different sizes. In summary, the detailed CFD simulations allowed answering relevant questions concerning the possibility of having an impacted region larger than the nuclear power plant exclusion area.

However, to give realistic results, these models need a significant effort in terms of modelling the site features, terrain elevation and geometry of the buildings and installations which could alter the overall wind flow. Also, current computational capacities in general do not allow simulating dynamic sequences with changing weather; therefore, we have chosen a representative set of "static" sequences and have weighted the results in order to obtain a realistic pattern of the likely deposition of the leaked particles.

In general these methods would be recommended only when the geometry and dispersion conditions of the site are very complex as well as in the case of particles whose dispersion is better simulated with lagrangian models.

Acknowledgements

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Inspection Plan for the detection of contamination at a Nuclear Fuel Cycle facility

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Introduction

A number of incidents related to the finding of contamination spots outside main operation buildings were detected at different nuclear sites in Spain over 2008. In response to these incidents, the CSN (Spanish Nuclear Regulatory Body) issued requirements demanding all the operators the execution of the site's comprehensive inspection plans in order to ensure that no contamination spots would be found out of control.

The ENUSA's Fuel Fabrication Plant at Juzbado is a nuclear fuel cycle facility whose specific features make it different to the rest of Spanish nuclear installations, provided that only Low Enriched Uranium (5% maximum) is handled as process material. Therefore, the inventory of isotopes consists only of U-isotopes and their daughters. These isotopes can be found in Nature in different amounts, depending on the geologic characteristics of the area, amongst others. Therefore, the natural background itself can contain spots of the contaminant, and must be taken into account for the preparation of the inspection plan adding complexity to its implementation. The criteria to distinguish background from non background values must be set (impacted and non impacted areas).

To implement the inspection plan at the Juzbado Plant, the MARSSIM methodology was chosen for the Scope & Characterization Surveys, following the Data Quality Objective (DQO) process. Derived Concentration Guideline Levels (DCGL) were not used: comparison between background values versus field values was used instead.

MARSSIM (Multi-Agency Radiation Survey and Site Investigation Manual) is a tool to conduct radiation surveys and investigation of contaminated sites. This method is used by different agencies and it is a reference for NRC, amongst others, being described on NUREG 1575.

Material and methods

Following the MARSSIM approach, there are six steps in the Radiation Survey Process:

a) Site Identification.
b) Historical Site Assessment.
c) Scoping Survey.
d) Characterization Survey.
e) Remedial Action Support Survey.

f) Final Status Survey.

As this is a facility that is not going to be decommissioned, only steps a) to d) will be performed.

To prepare the inspection plan, a review of all the activities that took place on site has been done. This, History Site Assessment (HSA using MARSSIM vocabulary), give us a map where contaminated material has been handled. The contaminated material has only been handled in the main building and radioactive liquid treatment facilities.

Fig 1. Map of Juzbado facility. Red spots indicate areas where contamination could be found.

After the review of the activities has been done, the other main conclusion at this point is that the only radioactive material handled was low enriched uranium, in the form of oxides (no UF_6).

With this information, we have a clear idea of what and where we have to found in order to prepare the characterization plan. No scoping survey will be done, we will try to perform only one characterization survey, using the regular periodic inspection that have been done in the facility as an scope.

The first part in the characterization survey is to classify the zones first in land areas or structures areas, because the types of surveys are different. After this, a risk based classification will be done. The areas with higher risk of contamination will be classified as class 1, and the areas with lower risk will be classified as class 3. To act in a conservative way, all the areas were considered as impacted areas, so all the areas must be surveyed.

The land areas we have in the facility according to the classification are included in the following map. The criteria used to classify each area is that areas where contaminated material was handled are classified as class 1, areas surrounding areas class 1 are class 2 and the rest is class 3.

Fig 2. Land areas. Green area is class 3 and blue areas area class 2.

The type of survey necessary to inspect each area was selected using the DQO (Data Quality Objective) process. These areas must be inspected by a scanning process and a simple measurement. For the scanning part, gamma measurement will be done, using portable equipment (INa monitor). For the single measurement a soil sample will be taken, and later alpha spectroscopy of the soil will be done, where ^{234}U and ^{238}U will be compared. For material with enriched uranium, the amount of ^{234}U is higher than what we can found in nature. To conclude if an area is contaminated or not, we compare the ratio ^{238}U:^{234}U (sample) vs ^{238}U:^{234}U (background). If the mean ^{238}U:^{234}U concentration in samples collected from a survey area is greater than the mean ratio from an equivalent number of samples randomly selected from the available background sample population + a substantial difference (S) of 2 times the full background sample population standard deviation determined, then decide enriched uranium contamination is present. Otherwise decide the uranium present is natural background. A summary of the process for class 1 areas is included on this table:

Table 1. Example of measures for the land areas, class 1.

Land Area	Scanning	Field Screening	Measurement/ Sampling
Class 1	Perform medium density (50%) gamma surface scans of all accessible surfaces in each Class 1 survey area.	Pause and investigate any locations with gamma radiation levels distinguishable from background. Mark investigative locations for potential judgmental sampling.	Use an statistics application to determine the sample size for each survey unit The input parameters, using the Wilcoxon Rank Sum (WRS) planning mode are: False Rejection Rate: 0.05% False Acceptance Rate: 0.05 or 0.10% Lower Bound of the Gray Region: 1.033 Specified Difference of True Means or Medians: 1.205 Estimated Standard Deviation: 0.086 NOTE: The WRS test mode is used for determining sample size. The non-parametric statistical test will be the Wilcoxon Mann-Whitney test.

Later, to analyze the data, Wilcoxon Mann-Whitney Test will be used, taking both sample and background data.

For the structures area, a map of the facility was taken, and every building or road was assumed to be a significant area. The criteria used to classify areas as class 1, 2 or 3 is the same as before. If contaminated material was handled, then the area is class 1. All areas surrounding class 1 are class 2. Once the area is classified, it splits on different cells, each one with a maximum surface, depending on the classification.

Table 2. Maximum surface allowed on every area.

Type of area	Maximum surface of the cell
Class 1	100 m2
Class 2	1000 m2
Class 3	No limit.

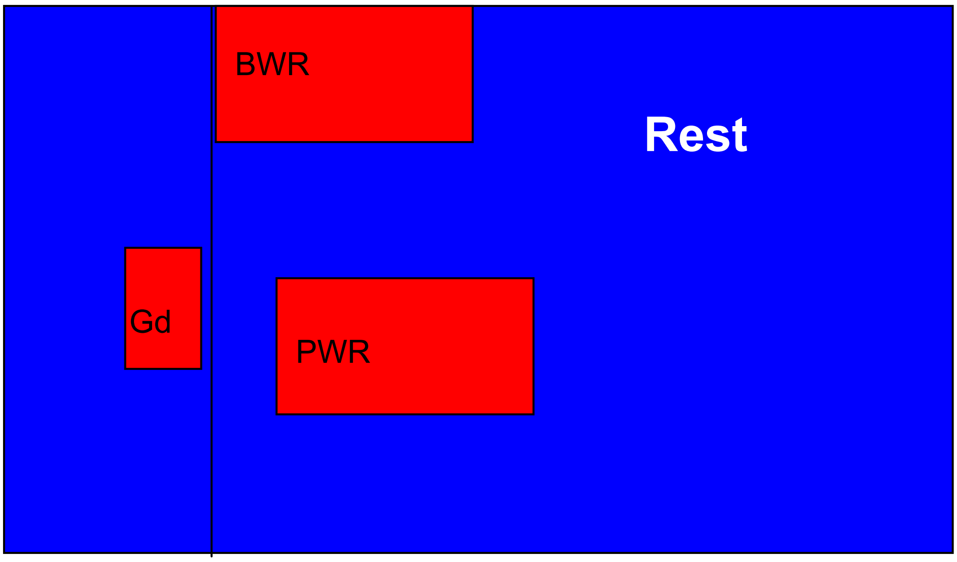

Fig 3. Example of survey plan on the roof. Areas around the stacks are red (class 1). The rest of areas are class 2 because they are near areas class 1.

The type of survey that will be performed on every area is also a scan or an static measurement, both with portable monitor able to measure alpha radiation. The measure will be done without any background subtraction, and will be compared to different background of different materials. If the mean cps (counts per second) in the survey area is higher than the mean of the cps from any equivalent number of background simple randomly selected from the available population + the equivalent value in cps of 0.04 Bq /cm^2, then it is determined there is contamination due to uranium. Otherwise, decide that area do not have contamination. In this step Wilcoxon Rank Sum Test will be used.

Fig 3. Device used to scan surfaces in order to find contamination spots.

Table 3. Example of measures for structures areas, class 1.

Structure Areas	Scanning	Field Screening	Measurement/ Sampling
Class 1	Perform high density scan (100%) in all the accesible surfaces	Pause and investigate any locations with radiation higher than background. Mark investigative locations for potencial judgmental sampling.	Use an statistic application to determine the sample size for each survey unit. The input parameters using WRS will be, for the surfaces used in background (concrete and asphalt): False Rejection Rate: 0.05% False Acceptance Rate: 0.05% Lower Bound of the Gray Region: (0.35 ó 0.44) Specified Difference of True Means or Medians: (1.41 y 1.50) Estimated Standard Deviation: (0.076 y 0.08) NOTA: The WRS will be used.

Conclusions

The MARSSIM methodology it is a powerful tool. It can be used for different projects, mainly decommission projects, but also for characterization purposes. For this reason the steps related to final status survey (that uses the data that support the decision for release of an area) is very well detailed in the NUREG 1575. On the other hand, characterization survey (or scoping survey) is not detailed as good as final status survey.

This plan intends to use all the steps of MARSSIM approach, for a characterization survey, using some of the analysis methods proposed for the final status survey.

With this plan, all the areas inside the fence of the facility will be scanned in order to prove that there is no contamination risk. The requisites asked by the regulator will also be fulfilled.

If contamination is detected over the limits specified in the regulations, will be cleared, explaining all the actions taken in the report that is going to be sent to the regulator.

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Establishment of a special radiological surveillance programme at the “El Cabril” solid radioactive waste disposal facility

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Abstract

Due to a requirement by the Nuclear Safety Council, the Spanish regulatory authority, a special radiological surveillance plan (SRSP) has been developed at the low and intermediate level Radioactive Waste Disposal Facility (El Cabril centre) in Spain. The objective of this plan, which covers outdoor areas on the site, is to identify and remediate possible contaminated areas, including exterior ground, walls, roofs or terraces, the walls of buildings, drains, channels, paths and roads.

The RSP has been developed in accordance with the MARSSIM methodology and includes an historical analysis of the facility and an initial radiological characterisation of the protected area.

Objective and scope

In response to a requirement by the Nuclear Safety Council, the Spanish Regulatory Authority, a special radiological surveillance programme (SRSP) has been developed at the Low and Intermediate Level Solid Radioactive Waste Disposal Facility (El Cabril), located in the province of Córdoba (Spain), covering outdoor areas of the protected zone of the Site, the aim being to identify and eliminate any possible contamination.

In accordance with the instructions given by the Nuclear Safety Council, the programme has addressed the following requirements:

– Analysis of practices that may have given rise to the presence of contamination at the Site.
– Special attention to the existence of points at which sludges are accumulated or concentrated.
– The entire surface area of the Site will be covered, with a more detailed and precise systematic approach being adopted for areas identified as presenting the highest risk of contamination.

The SRSP has been applied to the entire outdoor area of the protected zone of the Site. The study areas have included outdoor paved areas, walls, the roofing and/or terraces and walls of buildings, gardens, forested areas and external structures exposed to the weather, such as slabs, drains, channels, roads and paths, etc.

As regards the radiological surveillance of walls, it was decided that this would be carried out only if contamination were detected in areas in which the presence of contamination might be suspected.

The SRSP has been carried out in accordance with the MARSSIM methodology, including a historical analysis of the Facility and the initial radiological characterisation of its protected zone.

Development of the special radiological surveillance programme (SRSP)

The SRSP has been carried out by the Radiological Protection and Environmental Service of the El Cabril, with support from the ENRESA Radiological Protection Technical Unit and Safety Department. In addition, there has been support from an expert in spectrometry and instrumentation. Performance of the measures was undertaken by an external company, with a Senior Technician and two field measurement systems Operators.

The SRSP was performed between September 16th and October 8th 2009. The first task consisted of analysing the initial situation of the outdoor areas, as a basis for planning of the programme. As a result of this analysis, the protected zone was divided into Impacted and Non-Impacted Areas, these being defined as follows:

Non-impacted areas: those in which there has been no relation with radioactive materials during the operating lifetime of the Facility.

Impacted area: those in which there may have been a relation with radioactive materials as a result of the operating lifetime of the Facility. These are divided into three classes:

- Class 1: areas in which the existence of residual radioactivity is highly probable.
- Class 2: areas in which the existence of residual activity is a possibility.
- Class 3: areas in which the probability of residual activity is low.

The SRSP identified only the existence of class 2 and 3 Impacted Areas, no class 1 Impacted Areas were detected.

Definition of surveillance units (SU)

So–called Surveillance Units (SU) were defined for all the areas classified as being Impacted. These consisted of known surfaces of a specific area with a similar history and homogeneous radiological characteristics; in other words, the unit is constituted of the same material (concrete, natural terrain, gravel, etc.) and the spatial distribution of the contamination should be approximately homogeneous.

As regards size, the areas defined were larger or smaller depending on their radiological classification:

- SU's defined for outdoor areas (asphalt and concrete): 10 SU's between250 m^2 and 24,114 m^2.
- SU's defined for the vertical outdoor walls of buildings (concrete): 3 SU's of 500 m^2.
- SU's defined for the roofing of buildings (concrete, metallic sheeting and gravel): 6 SU's between 660 m^2 and 1,471 m^2.

Assessment of the radioactive background and identification of reference areas

In order to determine the radioactive background of the Site, Reference Background Areas (RBA) have been identified in non-impacted areas on the site. Given that the reference areas should have physical, chemical, biological and radiological characteristics similar to those of the areas in which the radiological surveillance is to be performed, a reference background area has been selected for each type of material in the areas to be characterised (asphalt, concrete, natural terrain and roofing gravel).

Thus, each SU has associated with it a non-impacted area known as the RBA that serves to establish the levels of investigation of the initial surveillance phase, on the basis of the radioactive background of the Site.

The measurements for characterisation of the RBA's were carried out using the same procedures and equipment as used for the Surveillance Units (SU's). In each RBA several scans were performed, each with a minimum of 30 data, and 30 stationary measurements were carried in different locations. In each RBA the average and typical deviation has been determined, this being used to determine the Background Estimator and Minimum Detectable Concentration for each measuring technique in each type of material.

Reference levels

In order to determine whether any point at the Facility presented a level of residual activity, different reference levels were established, these being classified as action levels (AL) and investigation levels (IL).

Action Levels: The action levels used are those established in the CRL for surface concentration on building surfaces (63FR222) and those for mass concentration in soils (64FR234). The levels of action for the key radionuclides Co-60 and Cs-137 are show in table 1.

Table 1. Action levels.

Type of surface	Action Level	
	Co-60	Cs-137
Permeable soil: earth, gravel, gardens	0.14 Bq/g	0.41 Bq/cm^2
Low permeability surfaces: concrete, asphalt	1.18 Bq/g	4.67 Bq/cm^2

Investigation Levels: For each SU the applicable investigation levels are determined taking into account: the applicable action level, the background in the RBA associated with the SU, the classification of the SU / (class 1, 2, 3) and the measurement method (scanning, static measurement or sample). The levels of investigation established are in table 2.

Table 2. Investigation levels.

SU Class	Scan	Static measurement or sampling
Class 1	> 10 NA	> 10 NA
Class 2	> NA	> NA
Class 3	> NA	> 0.5 NA

Once the investigation levels were defined, these were determined in the equipment measurement units. For the UCRM-II equipment, the investigation levels were determined in cpm, since this is the unit in which the results of the measurement are obtained.

Measuring methods used

The wastes arising from the operation and dismantling of Nuclear Power Plants constitute the largest part of the total waste stored at the El Cabril, in terms of volume and activity, this being represented to the largest extent by Co-60 and Cs-137. These are, therefore, the radionuclides to be characterised. However, if the spectrometric measurements provide evidence of the existence of other significant radionuclides, these will also be evaluated.

The measurement methods used have been as follows:

– Scanning with the UCRM-II equipment.
– Static measurement with the UCRM-II equipment.
– Static measurement with the ISOCS equipment.
– Sampling and laboratory analysis.

The scanning measurements with the UCRM-II equipment have been performed to detect potential areas of activity. Two types of scanning were performed: systematic scanning in accordance with a determined itinerary and specific scanning carried out in areas with the highest probability of presenting residual activity. At those points static measurements were performed for confirmation.

The Inspector 1000 has been used to locate areas with higher count rates in those areas in which investigation levels have been recorded in scanning.

The static measurements with ISOCS and the sampling have been performed to determine the average activity of the SU.

The number of measurements performed has been as follows:

– Scanning: 10% of the surface of SU’s of class 3, 50% for SU’s of class 2 and 100% for SU’s of class 1.
– Static measurements with UCMR-II: Whenever an investigation level has been detected in scanning.
– Static measurement with ISOCS: 5 measurements per SU (an initial point selected at random and the remainder in accordance with the pattern of the apexes of a W). Static measurements have also been performed using ISOCS at those points where a stationary investigation level has been confirmed with UCMR-II.
– Specific sampling in rainwater collection boxes and at rainwater pond discharge points. Samples have also been taken where the measuring equipment was inaccessible or in areas with high background levels.

Selection of measuring equipment

The detection and measuring equipment used in performance of the Special Surveillance Plan was as follows:

a) Radiological measurement field unit (UCMR-II)
The UCMR-II is a system that has been developed to cover radiometric requirements in the field, in scenarios with difficult access. It is a 2"x2" Ina detector connected to a CPU, both being incorporated on a transport carriage.
The equipment is made up of: a central processing unit; keyboard and LCD screen; data acquisition system; GPS unit; three-wheel transport carriage with metallic structure and collimating shielding.

b) ISOCS equipment (In Situ Object Counting System)
Is a system for the measurement of objects by spectrometry, made up of: coaxial germanium detector characterised by means of the Montecarlo modelling code; shielding assembly; Inspector 2000 MCA; ISOCS software and Genie-2000 data acquisition programme and three-wheel metallic structure transport carriage.

c) INSPECTOR 1000 equipment
Is a multi-channel analyser that allows for the measurement of dose and dose rate and for the acquisition and analysis of gamma spectra.
The equipment incorporates an internal Geiger-Müller detector for high dose rate measurement and a probe with a INa (Tl) scintillation detector. The equipment has been used in scan mode for the location of hot spots.

All the equipment used was calibrated "in situ" or had calibration certificates in accordance with the respective calibration procedures approved by ENRESA.

Analysis and evaluation of results

This section analyses and evaluates the results obtained in performing the SRSP for each of the Surveillance Units. In table 3 the main characteristics of the SU and the measurements carried out are shown.

Outdoor areas of the site. Paved areas

a) Building area yard (Zone 11)
<u>Surveillance Unit UV-A-VIG-11</u>
Asphalted section located between the Transitory Reception, General Services, Conditioning, Active Laboratory and Auxiliary Conditioning Buildings.

Scans were performed in one scan an Investigation Level was detected. This was verified by way of a one-off measurement and a sample was taken for analysis in the laboratory. The results of this analysis did not reveal any activity above the detection thresholds.

In the static measurements performed at random locations no activity was detected.

Specific sample of sediments from the rainwater collection box was taken. The results did not reveal any activity.

This area, which was initially classified as class 3, was reclassified as non-impacted.

Surveillance Unit UV-A-VIG-33

This belongs to unit UV-A-VIG-11 and was created as an intermediate unit between the class 3 UV-A-VIG-11 and UV-A-VIG-12, reclassified as class 1 after the initial measurements.

No investigation levels were recorded in gamma scans performed.

No activity was detected above the detection thresholds in statics measurements.

This Surveillance Unit, initially classified as class 2, was reclassified as class 3 since no activity was detected above the detection thresholds.

Table 3. Characteristics of SU and measurements performed.

Surveillance unit	Area (m^2)	Initial classification	Gamma scans			Number static measurements	Number samples
			Number	Area (m^2)	%		
UV-A-VIG-11	8,560	Class 3	21	874	10.2	5	2
UV-A-VIG-33	577	Class 2	6	305	53	5	-
UV-A-CON-12	945	Class 2	14	945	100	5	1
UV-A-VIA-17	24,114	Class 3	37	2,647	11	15 + 5	1
UV-A-PLA-18	1,930	Class 3	7	280	11	5	-
UV-A-PLA-19	1,930	Class 3	8	218	11	5	-
UV-K-VIG-26	1,114	Class 2	23	617	55	15 + 5	1
UV-K-VIG-27	1,965	Class 3	5	213	11	5	-
UV-K-VIG-28	513	Class 3	2	108	21	5	3
UV-K-VIA-31	2,852	Class 3	16	1,158	39	5	-
UV-K-VIA-32	250	Class 3	1	75	30	2 + 5	-
UV-C-01	1,270	Class 3	5	152	12	5	2
UV-C-02	660	Class 2	21	376	57	-	5
UV-C-03	1,100	Class 2				15 + 5	3 + 15
UV-C-04	1,471	Class 2	32	786	53	5	6 + 4

b) Parking area (Zone 12)

Surveillance Unit UV-A-CON-12

This is an asphalted area for the transitory parking of trucks carrying radioactive wastes.

In specific gamma scans 7 points above the Investigation Level were detected. 6 points above the Investigation Level were verified by static measurement. One point with a high Cs-137 count rate was confirmed by means of a specific measurement. In view of the results, this Surveillance Unit was reclassified as class 1, 100% of its surface area was investigated by gamma scanning.

In the static measurements at random locations, no activity was detected above the detection thresholds.

A sample was taken at the location with the high count rate. The result gave a value 10 times higher than the Action Level. This area was cleaned and a new soil sample was taken, the result was below the Action Level.

This Surveillance Unit, initially classified as class 3, was reclassified as a non-impacted area.

c) Access road to North Platform and South Platform (Zone 17)
Surveillance Unit UV-A-VIA-17
Asphalted area corresponding to the access road to the radioactive waste Disposal Platforms zone.

In gamma scans performed 5 Investigation Levels were obtained. Static measurements were performed at these 5 locations, no activity was detected above the detection threshold.

No activity was detected above the detection thresholds in statics measurements at random locations.

Specific samples of sediments from the rainwater collection box were taken, no activity was detected.

This Surveillance Unit, initially classified as class 3, was reclassified as a non-impacted area.

d) Temporary ISO container disposal facility to the south of the South Platform (Zone 18)
Surveillance Unit UV-A-PLA-18
This is an asphalted area located to the south of the South Platform where ISO containers with radioactive wastes from different incidents (Acerinox and others) are kept.

Any point above the Investigation Level was detected in gamma scans.

In static measurements at random locations, no activity above the detection thresholds was detected.

This Surveillance Unit, initially classified as class 3, was reclassified as a non-impacted area.

e) Temporary ISO container disposal facility to the north of the South Platform (Zone 19)
Surveillance Unit UV-A-PLA-19
This is an asphalted area located to the north of the South Platform where ISO containers with radioactive wastes from different incidents (Acerinox and others) are kept.

No points above the Investigation Level were detected in gamma scans.

In static measurements at random locations, no activity above the detection thresholds was detected.

This Surveillance Unit, initially classified as class 3, was reclassified as a non-impacted area.

f) Outdoor area to the west of the Modules and Technology Building (Zone 26)

Surveillance Unit UV-K-VIG-26

This is a concrete area located to the west of the Technology Building and the temporary radioactive waste storage Modules.

In gamma scans performed 15 Investigation Levels were detected. Static measurements were performed at these locations; no activity was detected above the detection threshold.

In static measurements no activity was detected above the detection thresholds.

Samples of sediments from the rainwater collection box were taken. The results for the sample showed background values.

This Surveillance Unit, initially classified as class 2, was reclassified as class 3.

g) Outdoor area to the east of the Modules and Technology Building (Zone 27)

Surveillance Unit UV-K-VIG-27

This is an outdoor concrete area located to the east of the Technology Building and the temporary radioactive waste storage Modules. It also includes the areas between the Modules.

In gamma scans performed, 21 Investigation Levels were detected. A checking was made and it was considered that these values were due to the influence of a high background resulting from the radioactive wastes contained in these Modules.

The static measurements showed that no activity was detected above the detection thresholds.

This Surveillance Unit, initially classified as class 3, was reclassified as a non-impacted area.

h) Outdoor area to the east of the Technology Building and the east and south of the Modules (Zone 28)

Surveillance Unit UV-K-VIG-28

This is an area of earth located to the east and south of the temporary radioactive waste storage Modules.

In gamma scans performed 3 areas exceeding the Investigation Level were detected. Samples were taken and no activity was detected.

No activity was detected above the detection thresholds in static measurements.

This Surveillance Unit, initially classified as class 3, was reclassified as a non-impacted area.

i) Access road to Cell 29 (Zone 30) and road between the Technology Building and Cell 29 (Zone 31)

Surveillance Unit UV-K-VIA-31

This is an asphalted area providing access to Cell 29.

A large number of alarms above the Investigation Level were registered, in gamma scans, due to natural radionuclides in the ground surrounding the road.

No activity was detected above the detection thresholds in static measurements.

This Surveillance Unit, initially classified as class 3, was reclassified as a non-impacted area.

j) Band of ground between zones 11 and 13 (Zone 32)
Surveillance Unit UV-K-VIA-32
This is an area of land bounding Surveillance Units UV-A-CON-12, UV-A-VIG-11 and UV-A-VIA-17.

A single gamma scan was performed along the entire Surveillance Unit, 2 Investigation Levels were detected and confirmed by static measurements. Two samples were taken from these locations, no activity above the Detection Threshold was detected.

The measurement of the 5 random points was performed by sampling. The values obtained were those habitually found in the soils in the area.

This Surveillance Unit, initially classified as class 3, was reclassified as a non-impacted area.

k) Outdoor areas in the zone surrounding the rainwater collection pools
The radiological surveillance of these areas was performed by taking samples and analysing sediments at the outlet from the buildings and platforms rainwater pools. The two samples gave values of Cs-137 similar to those of the soils in the area due to fallout.

Roofs of buildings

a) Roof of the Auxiliary Conditioning Building
Surveillance Unit UV-C-01
This is a concrete roof with a weatherproofing sheet and a final layer of gravel. It is a surface not considered to be affected by the discharge of gaseous effluents, for this reason was classified as the area in which it is located.

During scanning 2 investigation levels were registered, both confirmed by specific measurements.

The measurements performed at the 5 random points showed two points with values of CS-137 activity above the AMD/2 detection threshold, but below the actions levels. The specific samples taken at the two points did not show Cs-137 or Co-60 activity in either the coarse gravel fraction or the fraction of fine materials above the detection thresholds.

This Surveillance Unit, initially classified as class 3, was reclassified as a non-impacted area.

b) Roof of the Active Laboratory
Surveillance Unit UV-C-02
This is a concrete roof with a weatherproofing sheet and a final layer of gravel. It is a surface considered to be affected by the discharge of gaseous effluents.

During performance of the scans, 3 investigation levels were registered. The specific measurements performed did not confirm the alarms.

The measurements at the 5 random sampling points were performed by sampling due to difficulties in accessing the roof with the equipments. The results pointed to the no existence of residual activity above the detection thresholds.

This Surveillance Unit, initially classified as class 2, was reclassified as a class 3 area. However, it might be reclassified as a non-impacted area since no residual activity above the detection thresholds was detected in the random samples performed.

c) Roof of the Conditioning Building

This is considered to be affected by the discharge of gaseous effluents, for which reason it was initially classified as class 2.

Surveillance Unit UV-C-03

This includes the part of the rood of the Conditioning Building located to the west of the general ventilation duct. This is a concrete roof with a weatherproofing sheet and a final layer of gravel.

Systematic scanning was performed and 51 investigation levels were registered, both in the window for Cs-137 and Co-60, although expected due to the foreseeable influence of the contents of the Conditioning Building. A triangular mesh of 15 points, for specific samples, was designed additionally to the 5 points already established corresponding to the random ones.

In three of the statics measurements, activity above the action level was registered. These results show a more intense influence of the background which corresponds to the location of the containers shed. Samples were taken at these three points for laboratory analysis.

The analysis of the 15 specific samples and the three samples from random points reveals that in none of the 18 samples measured were Cs-137 or Co-60 activity values registered in excess of the detection thresholds.

This Surveillance Unit, initially classified as class 2, was reclassified as a class 3 area.

Surveillance Unit UV-C-04

This includes the part of the roof of the Conditioning Building located to the east of the general ventilation duct. This is a concrete roof with a weatherproofing sheet and a final layer of gravel.

The scans were performed in front of the stack emission points and the general ventilation duct, 8 investigation levels were detected. Of these 8 investigation levels, 6 were confirmed by specific measurements due to the confirmed influence of the interior of the building. Samples were taken at these six points for analysis.

The measurements at the random sampling points were performed using the ISOCS equipment but in four cases samples were taken due to the equipment was not operative.

The analysis of the results of the samples indicates that no Cs-137 or Co-60 activity was detected above the detection thresholds.

This Surveillance Unit, initially classified as class 2, was reclassified as a class 3 area.

d) Roofs of Modules and Technology Building
<u>Surveillance Units UV-C-05 and UV-C-06</u>
The radiological surveillance was not performed in these surveillance units, belonging to the roofs of the Temporary Storage Modules and Technology Building, respectively. In accordance with the initial analysis, the fact that these roofs are affected by the discharge of effluents constitutes a risk factor. However, since the Technology Building had not initiated its operation as of the date of performance of the programme, they were classified in the same class as the outdoor area in which they are located (class 3).
Consequently, in view of the results of the surveillance corresponding to the outdoor area, their reclassification as non-impacted might be assessed.

Vertical outer walls of buildings

The radiological surveillance of the surveillance units on walls (UV-P-01, UV-P-02 and UV-P-03) has not been carried out since no contamination was detected in the outdoor areas in which they are located.

Conclusions

– The methodology used in this SRSP is adequate for the purpose and will serve for the performance of later routine surveillance at the site of the El Cabril disposal facility.
– All the outdoor areas may be classified as non-impacted areas, with the exception of UV-A-VIG-33, which was reclassified as class 3.
– All the roofs of buildings were reclassified as class 3 areas. Nevertheless, they might be reclassified as non-impacted, since no residual activity above the detection thresholds was detected in the specific or random samples performed.
– All the vertical exterior walls of buildings might be reclassified as non-impacted or class 3 areas, since no contamination was detected in the outdoor areas in which they are located.

Monitoring of radionuclides in the vicinity of Czech nuclear power plants

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Abstract

The monitoring in the surrounding environment of Czech nuclear power plants (NPPs) Dukovany and Temelin consists from routine determinations, performed by NPP's staff, and also extended, supervisory, sampling performed by research institutions. In our contribution several results of this monitoring are briefly discussed and summarized.

Introduction

At present, the most significant artificial sources of radiocarbon in the environment are effluents from nuclear power facilities, even though it is a minor contribution in comparison with its natural production. Nevertheless, radiocarbon makes a major contribution to the collective effective dose from all radionuclides released by nuclear power plants (NPP) with light-water pressurized reactors (LWPR) during normal operation (UNSCEAR, 2000). Although ^{3}H is not considered to be a particularly radiotoxic radionuclide and releases by commercial nuclear power plants have traditionally been well below regulatory limits, control and monitoring are important because of the sensitivity of the public to issues of radioactive material release and the potential of radiation exposure (Harris et al. 2008; Andersen 1995; Liu et al. 2003). The radionuclides ^{3}H and ^{14}C require special consideration because of their high mobility in the environment and the fundamental nature of hydrogen and carbon cycles in the biosphere (UNSCEAR 2000). Tritium produced by NPPs with LWPR can be released into the environment through waste discharge in either airborne or liquid forms (Kim & Han 1999; Harris et al. 2008). In general, normalized ^{3}H gaseous and liquid releases from LWPR are about 1.5 and 20 TBq GWyear^{-1}, respectively (Luykx & Frazel 1986). Tritium released to the atmosphere occurs in two forms: tritiated hydrogen (HT) and tritiated water vapour (HTO). HTO is subject to the same wet and dry deposition processes as other nuclides, but it can also diffuse into the soil pore space and the leaf stomates (Belot et al. 1979; Garland 1979). Tritium which is not returned to the atmosphere by evaporation moves through the soil primarily by the mass flow of liquid water (UNSCEAR 2000). The Czech Republic has two LWPR equipped NPPs, Temelin and Dukovany, with the installed power output 2x1000 MW and 4x440 MW, respectively. The monitoring in the surrounding environment of Czech nuclear power plants (NPPs) Dukovany and Temelin consists from routine determinations, performed by NPP's staff, and also extended sampling which is performed by several research institutions. Systematical monitoring of HTO in the air of NPPs surrounding was launched there by the National Institute of Radiation Protection in 2008. Systematic sampling of biota and agricultural products for ^{14}C activity monitoring was started in 2002 by Nuclear Physics Institute in the cooperation with National Institute of Radiation Protection (Svetlik et al. 2007, 2008). Environmental monitoring, performed by staff of NPPs, includes these determinations: gamma spectrometry, ^{90}Sr, ^{3}H (in surface and underground water), gross alpha and beta, and dose rate (ETE 2010, EDU 2010). Still other institutes are involved in the specialized monitoring of Czech NPPs surrounding, namely: Water Research Institute T.G.M. in Prague – HTO and ^{137}Cs in connected surface water (Hanslik et al. 2009a, b); Czech Technical University, Faculty of Nuclear Sciences and Physical Engineering, Department of Dosimetry – gamma spectrometry and dose rate in the NPPs vicinity (Thinova & Trojek 2009; Thinova & Kluson 2008).

Material and methods

As a part of project SÚJB 5/2008 "Monitoring and evaluation of nuclear power plants discharges containing tritium" tritium activity concentration in the air humidity was also determined. Air humidity samples were collected in surroundings of the two Czech nuclear power plants (Malatova & Hulka 2008). A sampling method using sorption of the air humidity on dried silica gel was developed. The method operates in two different time regimes utilizing both static (sampling period of about four weeks) and dynamic sampling applying Dwarf aerosol sampler for point sampling with duration of about 10 minutes (Fejgl et al. 2009). Tritiated water was recovered from silica gel by means of freeze-drying. Tritium activity concentrations were measured by low-background liquid scintillation spectrometers. Four outdoor HTO sampling campaigns at the vicinity of the Dukovany power plant and one outdoor sampling expedition to the vicinity of the Temelin power plant were performed during the years 2008 and 2009. During these sampling campaigns 112 air humidity samples were collected. Sampling around power plant Dukovany was emphasised because of elevated tritium activity concentration in cooling water (hundreds Bq per liter), therefore tritium could be discharged from the power plant not only via ventilation stack but also via cooling towers (Fejgl et al. 2009a). 10 sampling points for dynamic sampling and 17 sampling points for static samplings were laid out in distance between 100 and 900 meters from the cooling towers, see Fig. 1.

Monitoring of ^{14}C in surrounding of the Czech NPPs and in reference localities is performed utilizing biota samples, see fig 2. This type of monitoring is restricted on the part of vegetation period. Time interval of ^{14}C activity record depends on period of biomass cumulation in a given plant. ^{14}C activity in the sampling points can be locally influenced (lowered) by anthropogenic CO_2 emissions from fossil fuel combustion (traffic, heating of buildings). Fossil carbon (without significant ^{14}C amount) causes dilution of ^{14}C in carbon isotopic mixture (Suess effect). Local component of Suess effect can be observed namely in the vicinity of greater roads, cities, and other greater fossil CO_2 sources (Suess, 1955; Kuc & Zimoch 1998).

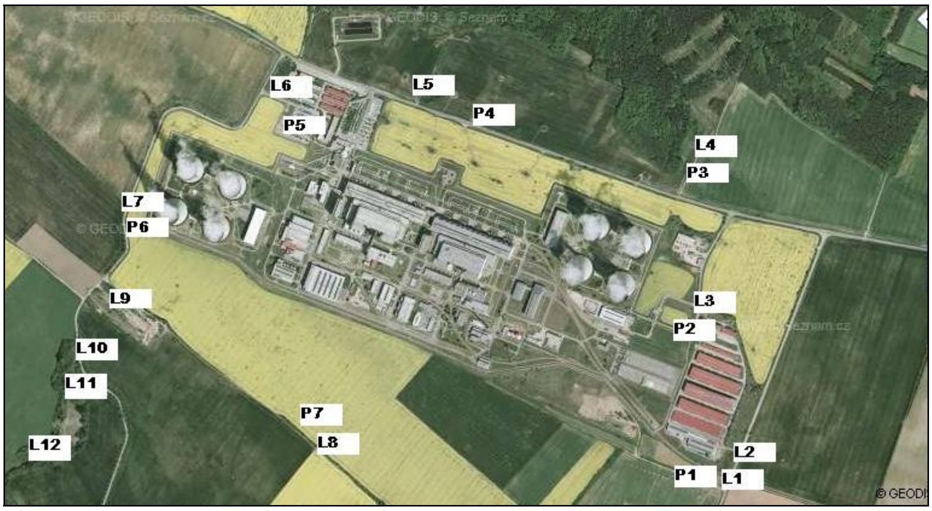

Fig. 1. Nearest sampling points around NPP Dukovany for HTO activity monitoring in air moisture. P – prompt sampling, L – long term period of sample cumulation.

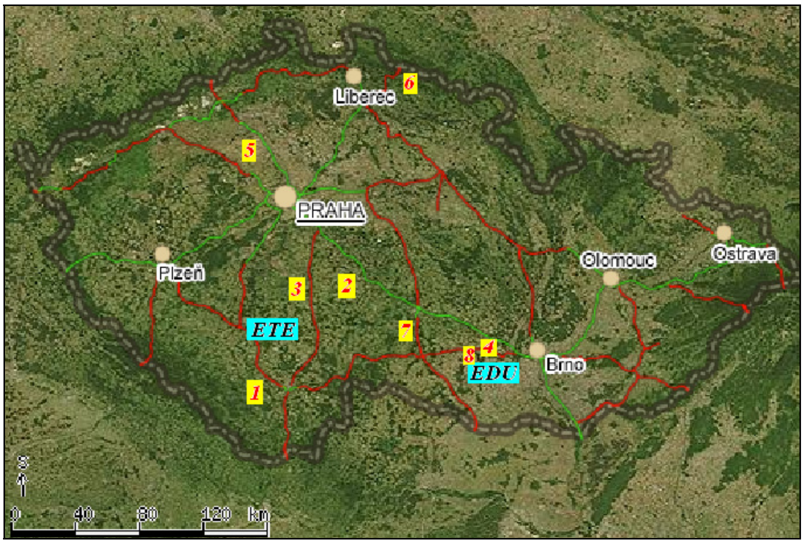

Fig. 2. ^{14}C sampling points in the area of the Czech Republic: NPP Dukovany – EDU; NPP Temelin – ETE; 1-8 reference sampling points in localities with smaller load of fossil fuel combustion.

Prevailing part of biota samples were leaves of deciduous trees, analogically to published studies performed in the vicinity of other NPPs. Leaves of *Sambucus nigra L.* (pipe tree) were preferred, because this tree is widespread in the Czech Republic, and it can be easy identified. Smaller part of samples collected were agricultural plants (spikes of wheat and barley). Dusty biota samples were washed by 10% HCl and distilled water, dried (105 ^{o}C) and homogenized. Washing by diluted HCl was suspended if presence of dust on sample surface was not evident. Dried samples were combusted and resulting CO_2 was purified. In the NPI AS CR a routine of sample processing based on benzene synthesis was followed (Gupta & Polach, 1985). ^{14}C activity was measured by liquid scintillation spectrometer Quantulus 1220 in 3 ml low-background Teflon vials. Total counting time was about 2000 minutes per sample. Benzene distributed by Sigma-Aldrich (spectrophotometric grade) was used as a blank sample. Calibration was performed using oxalic acid NIST (NBS) HOX II, SRM 4990-C (Schneider et al., 1995). Combined uncertainties include the individual uncertainties of measured sample, blank sample, calibration, quenching corrections, and uncertainty of the $\delta^{13}C$ value (Curie 1995). In the vicinity of NPPs Temelín and Dukovany several roads are situated and there also are some smaller cities and villages. The influence of local Suess effect could be estimated/quantified with difficulties, as data about local fuel combustion and density of traffic are absent. To compare ^{14}C activities in the NPPs surrounding with actual ^{14}C activity level in the environment two types of areas with different load of Suess effect were selected. It can be supposed that actual size of Suess effect influencing in NPPs surrounding will be in the interval demarked by the two types of reference areas (Svetlik et al. 2007). (A) Localities, in greater distances from fossil carbon sources, where only small local influence of Suess effect was supposed (Košetice, Kleť, Sudoměřice u Bechyně, Krokočín). (B) Localities where medium local Suess effect influence can be expected (bordering parts of Prague).

Staff of nuclear power plants Temelin and Dukovany is monitoring radionuclides in the NPPs vicinity following internal documents (programs of Environmental Monitoring in the Vicinity of NPPs Temelin and Dukovany). All gamma, alpha and liquid scintillation spectrometry measurements (^{90}Sr, HTO) of radionuclides were performed applying metrologically validated and calibrated instruments (Czech Metrology Institute - Inspectorate for Ionizing Radiation). Most of the utilized analytical methods are a subject of accreditation (Czech Accreditation Institute). The environmental monitoring of radionuclides includes these sample types: (1) Aerosols (13 sampling points in the distances from 2 to 26 km, flow rate about 40 m^3 per hour, one week exposition – determinations: gamma emitting radionuclides and ^{90}Sr), monitoring of ^{131}I for each NPP is also performed at distance from NPP Temelin 5 km (one point) and Dukovany, 6 points at distances 3-11 km (flow rate 4 m^3 per hour, one week of iodine cartridge exposition); (2) bulk atmospheric deposition (8 sampling points in distances from 2 to 11 km to NPP, sampling area 0.5 m^2, one month sampling period – determinations: gamma emitting radionuclides); (3) atmospheric precipitation (7 points, sampling period is one month, in distances from NPP Temelin and Dukovany from 1 to 11 km, respectively – determination HTO); (4) surface water in surrounding lakes (7 sampling points) and corresponding river profiles 10 of Vltava and Jihlava (one month sampling period – determinations: gross alpha and beta, gamma spectrometry, HTO), from each profile and sampling point also determination of ^{90}Sr is annually performed; (5) underground water, 25 boreholes in distances from NPPs Temelin and Dukovany from 1 to 3 km - determinations: gamma spectrometry, HTO); (6) drinking water (at least samples from 11 sampling points of tap and well water - determinations: gross alpha and beta, gamma spectrometry, HTO, ^{90}Sr); (7) milk (3 pieces of 3 litre samples in one month period (minimally) for gamma spectrometry measurement, subsequently are samples cumulated and ^{90}Sr determination is performed annually); (8) sediments and soils, annual quantity is about 16 sampling points (surface layers till 10 cm, fraction below 2 mm, gamma spectrometry measurement); (9) 12 agricultural and forest products (gamma spectrometry measurement) are processed annually.

Results

Basic statistical parameters of HTO monitoring in air humidity in the vicinity of Czech NPPs and background areas are reported in Table 1.

Table 1. Mean values of observed HTO activity concentration in air moisture and supporting data (prompt sampling/long term sample cumulation).

	NPP Dukovany	NPP Temelin	Background samples [c]
Median, Bq L^{-1} [a]	2.60 / 2.53	2.19 / 1.82	2.81
Number of values below significance level [b]	5 / 17	0 / 1	6
Number of outlying values	2 / 4	0 / 0	0
Number of observations	32 / 50	7 / 5	23

[a] Excluding outlying values in the case of NPP Dukovany.
[b] Decision threshold was calculated for 5% probability of the first kind error and was in the range 0.8 to 1.3 Bq L^{-1}.
[c] Background sampling was performed only for long term sample collection.

During period 2002 – 2005, 77 biota samples for ^{14}C analyses were collected in the vicinity of NPPs Dukovany (EDU) and Temelín (ETE). Likewise, 30 samples were collected in reference areas influenced with slight (A) and extended (B) local Suess effect. Basic statistical parameters of results (EDU, ETE, A, B) are reported in Table 2. Standard deviations of couples EDU-B, ETA-A, and ETE-B are equal on the base of F-tests performed (Fischer-Snedecor test). In the next step results from each type of area were compared utilizing t-test (student test, unpaired, probability of first kind of observation error 1%), see Table 3.

Table 2. Basic statistical parameters of biota samples collected in the vicinity of NPPs Dukovany (EDU), Temelín (ETE) and in reference localities with slight (A) and extended (B) local Suess effect. Sampling period 2002 – 2008. Activities are reported in ‰ of $\Delta^{14}C$ (Stuiver & Polach, 1977).

	EDU	ETE	Ref. localities A	Ref. localities B
Average	60.1	61.0	56.2	47.4
Median	58.3	60.4	56.2	45.7
Standard deviation (σ)	13.2	9.0	6.5	7.3
Variation	173	81	42.1	53.5
Number of observations	27	50	21	9
Observed maximum	95.9	84.4	67.9	58.7
Observed minimum	39.8	41.7	44.0	38.0

Table 3. Comparisons of activities of observed results from each type area (group of the data), values of T reported in table: T_o (observed) and T_c (critical); t-test, unpaired, probability of first kind of observation error 1%.

Couple compared	T_o	T_c	*t-test enclosures*
A-B	2.621	3.106	$T_o < T_c \Rightarrow$ *difference is not significant*
EDU-A	1.479	2.712	$T_o < T_c \Rightarrow$ *difference is not significant*
EDU-B	2.507	2.037	$\boldsymbol{T_o > T_c \Rightarrow}$ ***difference is significant***
ETE-A	2.336	2.651	$T_o < T_c \Rightarrow$ *difference is not significant*
ETE-B	3.913	2.668	$\boldsymbol{T_o > T_c \Rightarrow}$ ***difference is significant***
EDU-ETE	0.272	2,712	$T_o < T_c \Rightarrow$ *difference is not significant*

Discussion

HTO activity concentrations in the air humidity from background reference sites fall in the range from <1.2 up to 3.5 Bq/L. All the samples from NPP Temelín vicinity and most of the samples from the NPP Dukovany vicinity do not exceed the same range. In some of the samples significantly increased tritium activity concentrations were found. An increase was found in the samples from the sampling point L5 (10.57 ± 0.53 Bq L^{-1}, September 2008) for long-term sampling and P4 for instant sampling (11.37 ± 0.57 Bq L^{-1}, September 2008). Both these sampling points are placed near the wastewater pipe outfall; the increase can be explained by a local evaporation of the wastewater. Another increased activity concentration was found in the sample from September 2008 from the sampling point P7 for instant sampling (7.90 ± 0.62 Bq L^{-1}). This point is placed 100 meters to the south from the grounds of NPP Dukovany. It was raining during the sampling and the wind blew southwards. This increase is the only one which is explainable as a sorption of tritiated water evaporated from the cooling tower (Fejgl et al. 2009). Other increases were found in the samples from the dam Mohelno vicinity from September 2008 and September 2009 i.e. 13.28 ± 0.51 and 7.03 ± 1.08 Bq L^{-1}, respectively. Probable reason is the evaporation from the dam reservoir (tritium volume activity concentration ranges between tens and hundreds Bq per litre), see Fig. 3. Related increase was also observed in the sample collected near the Jihlava River (8.77 ± 0.91 Bq L^{-1}, September 2008).

Comparison of two air humidity sampling methods sorption on silica gel versus cold trap method (air flow-rate 30 L $hour^{-1}$, temperature -20 ^{o}C, sampling period a month) at the same place shows systematically augmented tritium activity concentrations in samples collected by sorption on silica gel. With respect to the fact that chemical laboratories of NRPI process highly tritiated water samples (up to 100 kBq L^{-1}), the cross-contamination seems to be the most probable reason of tritium activity concentration increase. An experiment with very low tritium activity water was performed off the premises. The water vapour was sorbed into silica gel, this was lyophilized and subsequently the tritium activity concentration in the sample was determined. The found value was 1.22 ± 0.59 Bq/l. This result confirms possibility of cross contamination during sample processing in the lab. Contamination of the dried silikagel just after it has been dried an before it is filled in the storage vessel seems most likely. Some precautions were applied in order to reduce cross-contamination risk. Processing time was shortened and the drying temperature was enhanced from 105 to 130 °C, the drying time was extended to 8 hours, what ensures extrusion of lasting humidity. For final evaluation of precautions it is necassary to get more data, therefore the monitoring of the samples from the background reference sites will continue.

Results of statistical comparison confirmed significantly greater ^{14}C activity level in surrounding biota from both NPPs in the comparison with reference area B with greater load from fossil fuel combustion (bordering parts of Prague). Nevertheless, if these tests were performed for probability of first kind of error 5% (a), significant differences were found also between reference localities A and B. The difference of ^{14}C activity level between biota from NPP Temelín vicinity and biota from reference area A (small local fossil fuel combustion) was found to be statistically significant also, see Table 2. In the point of view of local loads from fossil fuel combustion, it can be supposed that relevant reference ^{14}C activity level for NPPs surroundings with relatively frequented roads is allocated in the interval between reference areas A and B. Observed ^{14}C activity values for each type of locality are charged by relatively greater variations, namely in samples from NPPs surroundings, probably caused by local Suess effect from surrounding roads especially. Other reason of fluctuation can be given also by relatively short time interval of biomass cumulation in leaves of deciduous trees (about the first third of vegetation period). At this part of year activity of atmospheric $^{14}CO_2$ changes rather quickly (Levin & Kromer 2004; Levin et al. 1995, 2008, 2010; Molnár et al. 2007) and biomass of plants is cumulated in the dependence of the local microclimatic parameters (atmospheric precipitation, soil moisture, sunlight exposure). Local microclimatic differences can cause small time shift of atmospheric $^{14}CO_2$ intake period. Additional reason of ^{14}C activity variations in NPPs surroundings is given by relatively greater distances from NPPs stacks (below 9 km) in certain sample collection localities. Potentially influenced zones around NPPs can be probably found in the distance till 2 km, on the basis of ^{14}C dissipation model, (Rousel-Debet et al., 2006). Also the direct results of ^{14}C in biota monitoring performed in the vicinity of NPPs with boiling water reactors (BWR)[1] in Sweden confirm similar distances from stacks for maximal surplus of ^{14}C activity in biota (Stenström et al. 1996; Stenström et al. 1998).

On the basis of environmental monitoring performed by staff of NPPs Temelin and Dukovany during year 2009, there were observed activities of HTO enhanced above background level in surface water, see fig. 3. These activities are given by liquid releases of ^{3}H, which did not exceed authorized limits given by State Office for Nuclear Safety. In the case of NPP Dukovany surrounding, there were observed also several increased HTO activities in underground water, which did not exceed 60 Bq L^{-1}. Origin of such HTO is probably connected with cooling towers, which are evaporating water from Mohelno reservoir, see fig. 3. Analogically to previous years, during 2009, no activities exceeding reference activity values were observed in the environmental

[1] In the comparison with LWPR contain releases from BWR considerably greater percentage of $^{14}CO_2$, above 90% (Kunz 1995). This chemical form of released ^{14}C can be assimilated by plant photosynthesis and hence greater ^{14}C activity excess can be observed in the surrounding biota of NPPs with BWR.

surrounding of both NPPs (ETE 2010, EDU 2010). By direct field measurements or on the basis of laboratory determinations performed in laboratories of NPPs, the only radionuclides of artificial (possible artificial) origin measurable in the surrounding environment are ^{3}H and ^{137}Cs. Dominant source of ^{137}Cs in the environment of the Czech Republic was accident of Chernobyl NPP in 1986 and no increased activity level of this radionuclide (in the comparison with background values) was observed in the vicinity of both NPPs.

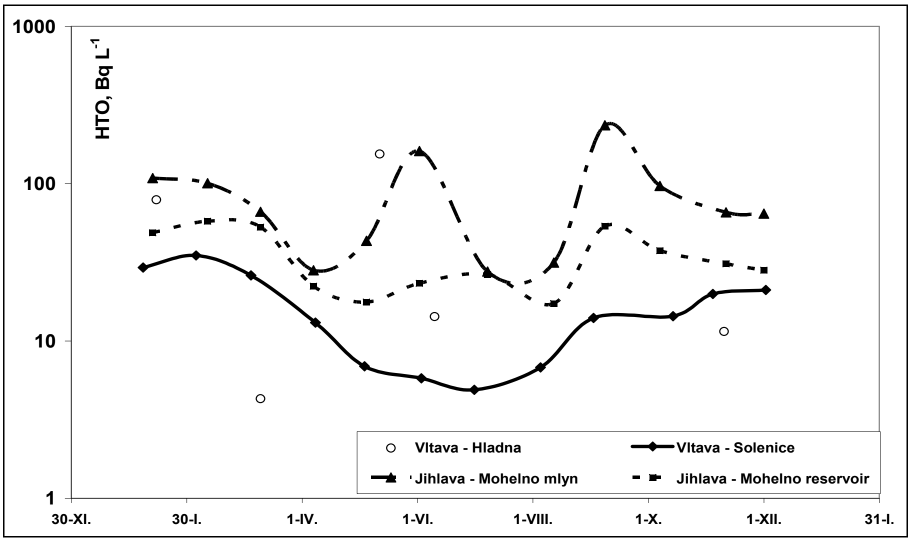

Fig. 3. Year 2009. Time course of HTO activity concentration in profiles Hladna (about 5 km downstairs from releasing channel of NPP Temelin in Korensko) and Solenice (below barrier of the great river reservoir Orlik, about 50 km downstairs from the releasing channel). Incrreased HTO activities in the profile Hladna can be observed only several days after discontinuous discharges from the NPP Temelin, in other cases are activities below 3 Bq L^{-1}. Liquid releases from NPP Dukovany are exhausted into Mohelno reservoir. River profile Jihlava-mlyn is about 1500 m downstairs from Mohelno reservoir barrier.

Conclusions

A brief summary of the monitoring performed by several institutions in the vicinity of NPPs Temelin and Dukovany was assembled. In the point of view of HTO monitoring in air moisture in the vicinity of NPP Dukovany, only several samples with significantly increased HTO activities were collected. Nevertheless, our sampling method based on silica gel sorption did not allow fine resolution and will be modified and validated. Results of ^{14}C activity monitoring, utilizing leaves of deciduous trees and agricultural products, in the surrounding of both NPPs highlights small probable surplus in the surrounding of NPP Temelin (about 5‰ Δ^{14}C). Enhancement of ^{14}C activity was not statistically significant in the vicinity of NPP Dukovany.

Enhanced HTO activities in rivers Vltava and Jihlava, where liquid releases are exhausted, have been observed by staff of NPPs within the frame of routine environmental monitoring. In the case of NPP Dukovany, HTO activities in the connected underground and well water have also been observed above actual environmental background, but deeply below limit of 100 Bq L^{-1}, given by Czech legislative. Significant, measurable, presence of other radionuclides originated from

NPPs has not been observed by monitoring performed by staff of nuclear power plants in the surrounding, although susceptible measuring methods were applied.

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^{14}C in biological samples from the vicinity of NPP Krško

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Abstract
Monitoring ^{14}C activity in the atmospheric CO_2 and in biological samples (fruits and vegetables) in the close vicinity of the NPPK was performed regularly since 2006 to estimate the possible influence of the plant on environmental ^{14}C levels and the possible contribution to the effective dose of local population through food chain.

Mean values of ^{14}C activity in atmospheric CO_2 in the years when the refuelling was not performed represent a barely visible increase in relation to the activities of ^{14}C in the atmosphere at the control point. Immediately after refuelling the atmospheric ^{14}C activity was increased for a short period of time. Spatial distributions of ^{14}C activities in biological samples show the dependence on distance and orientation. In all campaigns the highest activities were obtained at the same locations in the SW-NE direction that coincided with the most pronounced wind directions. Estimated contribution of ^{14}C released by NPPK to the natural dose is less than 1 μSv.

Introduction
^{14}C is formed in upper layers of the atmosphere in a reaction of neutrons from the cosmic rays and nitrogen atoms. All carbon isotopes take part in chemical reactions, hence ^{14}C is also oxidized into $^{14}CO_2$ and distributed uniformly throughout the atmosphere. Plants assimilate $^{14}CO_2$ through the photosynthesis process, and animals are fed by plants, thus the ^{14}C isotope is a part of the natural carbon cycle. In the past, the equilibrium between radioactive decay of ^{14}C and its production rate has been established, and the natural ^{14}C concentration, or better to say, specific activity of ^{14}C, in the Earth's atmosphere and biosphere is approximately constant and it equals 226 Bq/kg of carbon. Assuming atmospheric CO_2 concentration of 370 ppm, this equilibrium specific activity of ^{14}C in the atmosphere corresponds to 0.037 Bq/m^3 of air.

In the second half of the 20th century the natural ^{14}C distribution was disturbed by atmospheric bomb tests and in 1963 the atmospheric ^{14}C specific activity reached maximum of twice the natural activity. After the ban of atmospheric bomb tests the present-day atmospheric ^{14}C activity approaches the natural (non-disturbed) values.

^{14}C is produced also in nuclear power plants. The emitted $^{14}CO_2$ enters the carbon cycle, and through food chain it may contribute to the dose of the local population. The aim of this study was to determine distribution of ^{14}C in a close vicinity of the Nuclear Power Plant Krško (NPPK) in Slovenia close to the border with Croatia, and to estimate possible contribution of NPPK to the effective dose of local population through food chain.

Sampling and measurements

Atmospheric CO_2 was collected at two locations inside the NPPK area, marked as **A** and **B** in Figure 1, close to the release point of the plant ventilation system. Integral samples were collected since 2006 on a regular bimonthly basis by absorption of CO_2 in saturated NaOH solution. Na_2CO_3 thus produced was in the laboratory dissolved in HCl, and the obtained CO_2 was used for benzene synthesis. Benzene samples were measured by a liquid scintillation counter *Quantulus* (Horvatinčić et al. 2004).

Specific ^{14}C activity of biological samples (apples, corn, cereals, borecole, grass) was measured in immediate vicinity (locations **C**, **D**, **E**, **I**, **J**, **L** and **R**, 200 – 400 m from the NPPK release point), and in a wider environment (locations **F**, **G**, **H**, **K**, **M**, **N**, **O**, **P** and **Q**, about 1000 m) of NPPK (Fig. 1). As a control site, where no influence of the plant is expected, we have chosen village Dobova, about 10 km SE from NPPK. Two sampling campaigns have been performed each year since 2006, in June/July and in September/October. Samples were dried, carbonized (pyrolysis at 600°C), and combusted in a stream of oxygen. The obtained CO_2 was absorbed in a mixture of Carbosorb®E and Permafluor®E. ^{14}C activity was measured by the LSC *Quantulus 1220* (Horvatinčić et al, 2004; Krajcar Bronić et al, 2009).

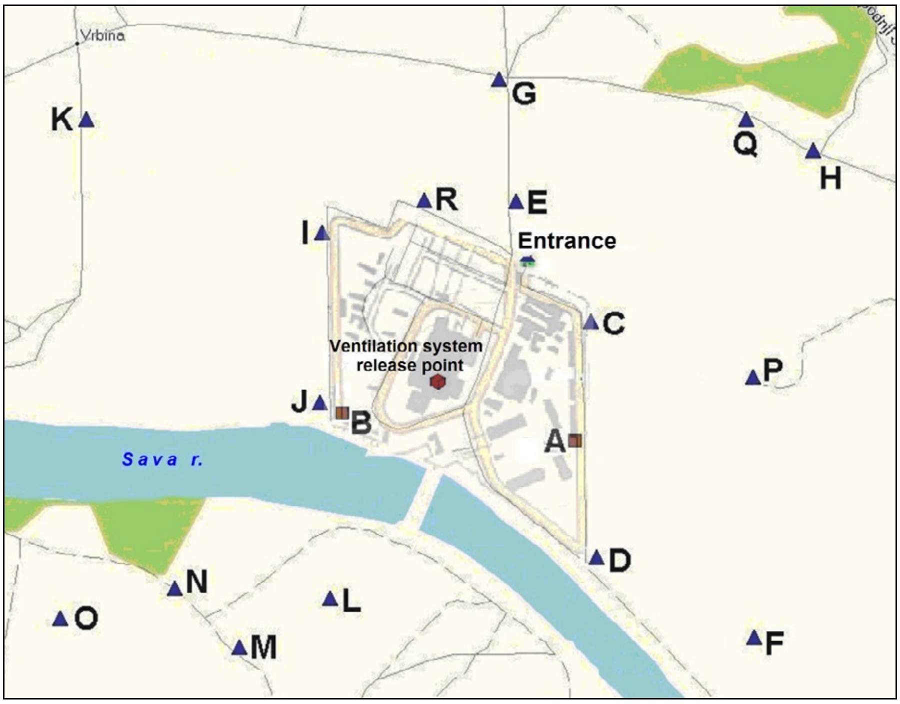

Fig. 1. Sampling locations in the vicinity of Nuclear Power Plant Krško. Locations A and B for atmospheric CO_2. Locations C – R for biological samples.

Results

The results are usually expressed as the relative specific activity $a^{14}C$, defined as the ratio of the specific activity of the sample and that of the atmosphere undisturbed by anthropogenic influence, in "percent of Modern Carbon", or pMC. Consequently, relative specific activity of 100 pMC corresponds to specific activity of 226 Bq/kgC or to the $^{14}CO_2$ activity concentration of 0.037 Bq/m^3 of air.

Activity in atmospheric $^{14}CO_2$

^{14}C activity concentrations in the atmosphere at two locations (**A**, **B**) inside the NPPK are shown in Figure 2. For comparison, a part of long-term atmospheric ^{14}C data in the city of Zagreb is shown (Krajcar Bronić et al, 2009), as well as the integrated monthly ^{14}C activities released through the plant ventilation system. Atmospheric ^{14}C activity at the location **B** is always slightly higher than that at the location **A**. This difference can be explained by the position of location **B** relative to the release point and the local prevailing wind direction SW – NE (Fig. 3), and is particularly evident in the refuelling outage periods and immediately thereafter. The half-life of the released ^{14}C in the atmosphere is estimated to app. 1.5 months. During the refuelling in October 2007 higher $a^{14}C$ values at the site **B** were observed (102 mBq/m^3) than during the refuelling in April 2009 (69 mBq/m^3) probably due to the of shorter sampling period (ten days in 2007 in comparison to three weeks in 2008), which coincided better to the highest ^{14}C release to the atmosphere.

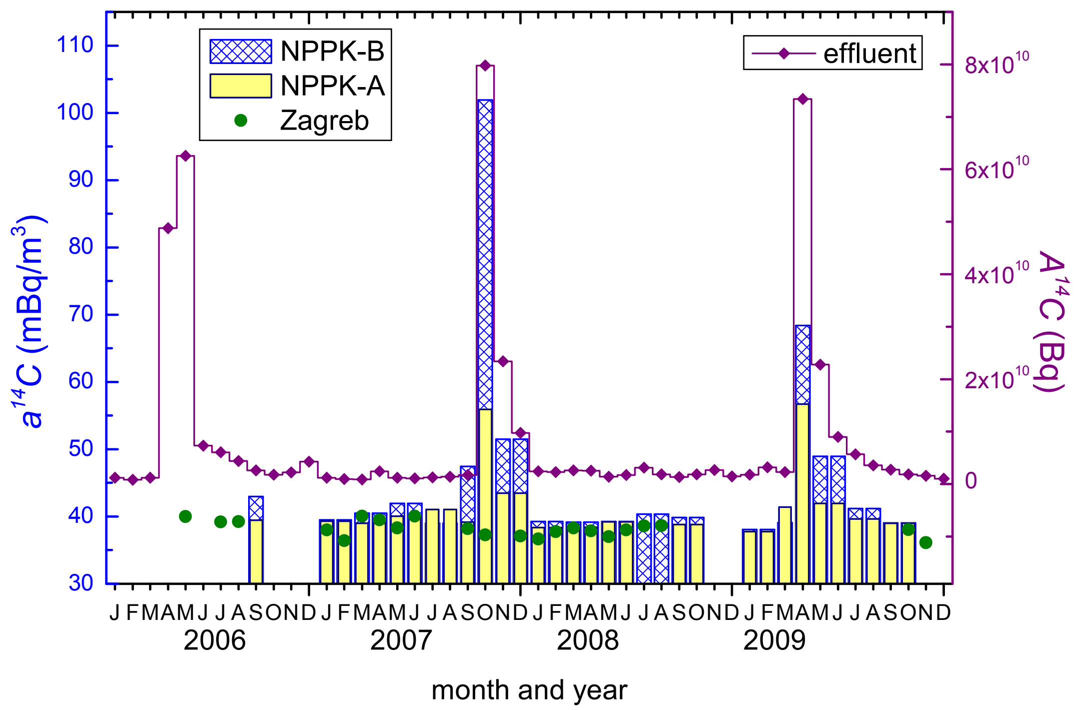

Fig. 2. ^{14}C activity concentration in atmospheric CO_2 at locations A and B and comparison with the results obtained in Zagreb (40 km E) (left ordinate). Monthly ^{14}C activity in effluent released through the plant ventilation system (right ordinate).

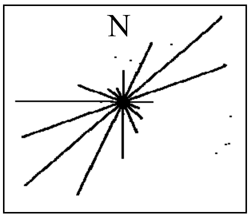

Fig. 3. Wind rose in the vicinity of NPPK, measured on a 10 m high column (according to the Environmental Agency of the Republic of Slovenia)

Mean values of $a^{14}C$ in atmospheric CO_2 in the years when the refuelling was performed (2007 and 2009) are 41.7 and 41.5 mBq/m^3 at the location **A**, and 47.9 and 44.2 mBq/m^3 at the location **B**. In 2008, when no refuelling was performed these values were 38.7 (location **A**) and 39.6 mBq/m^3 (location **B**). They represent a barely visible increase in relation to the activities of ^{14}C in the atmosphere measured in Zagreb (38.3 mBq/m^3 in 2007 and 37.8 mBq/m^3 in 2008), where the effect of reducing ^{14}C atmospheric activity is possible due to combustion of fossil fuels.

Activity in biological samples

In Figure 4. we present the spatial distributions of measured ^{14}C activities (expressed as specific activities in pMC) for all eight sampling campaigns, as well as the corresponding polar diagrams that show the dependence on distance and orientation. In polar diagrams we have separated closer locations **C**, **D**, **E**, **I**, **J**, **L** and **R** (inner circle) and farther locations **F**, **G**, **H**, **K**, **M**, **N**, **O**, **P** and **Q** (outer circle), and have also shown $a^{14}C$ values at the control point Dobova. Increased activities were observed in all sampling campaigns in SW – NE direction of most pronounced winds. As expected, higher activities of ^{14}C were measured in biological material which used CO_2 in the refuelling outage period which was performed during and immediately after the vegetation period. In 2006 and 2009 the power plant was refuelled in spring period, so the apples, having their vegetation period immediately afterwards, collected more active CO_2. Therefore, at all locations the highest ^{14}C activity was in July 2006 and June 2009, since both campaigns were performed after the refuelling in April of the corresponding year. At the other hand, in 2007 the refuelling was in autumn after the apples were harvested, and the mean ^{14}C activities in both sampling periods in 2007 (July and September) were similar to the activity measured at the control point Dobova (Table 1).

Based on predetermined spatial distribution of ^{14}C activities in biological samples around the plant, it is clear that important results could be obtained only within a few hundred meters distance from the plant (Breznik et al, 2008). This close distance is not very convenient for testing the dispersion model but enabled to detect higher activities in a shorter time interval.

Table 1. Mean values of $a^{14}C$ (pMC) in biological samples in all eight sampling campaigns. Inner circle – locations C, D, E, I, J, L and R. Outer circle – locations F, G, H, K, M, N, O, P and Q. * Measurements of samples from 9/2009 have not yet been completed.

Collection time	$a^{14}C$ (pMC) Inner circle	$a^{14}C$ (pMC) Outer circle	$a^{14}C$ (pMC) Control point
07 / 2006	120.6 ± 11.0	108.3 ± 3.0	103.2 ± 1.5
10 / 2006	112.3 ± 12.0	105.1 ± 2.0	104.0 ± 1.5
07 / 2007	103.7 ± 3.9	103.7 ± 2.8	105.6 ± 1.9
09 / 2007	106.8 ± 1.7	105.7 ± 2.6	103.8 ± 1.8
07 / 2008	109.6 ± 3.5	107.3 ± 1.7	104.1 ± 2.3
10 / 2008	109.1 ± 3.3	109.1 ± 3.0	104.4 ± 2.7
06 / 2009	117.0 ± 11.2	110.5 ± 2.0	105.4 ± 1.4
09 / 2009	115.9 ± 11.4*	103.2 ± 2.3*	102.0 ± 1.4

Discussion

Equivalent annual dose E received by an average adult person by ingestion of food of specific ^{14}C activity $a^{14}C$ (Bq/kgC) can be estimated as

$$E = e \times a^{14}C \times m \times t$$

where t is a 1-year period (365 days), m is a mass of carbon ingested daily by food (0.3 kg, ICRP, 1996), $a^{14}C$ is measured ^{14}C specific activity (Bq/kg C), and e is the ingestion dose coefficient for ^{14}C, i.e., expected effective dose per unit ^{14}C activity, 5.8×10^{-10} Sv/Bq (ICRP, 1996).

It is difficult to realistically estimate a possible increase of the annual dose in case of ingestion of fruit grown in the vicinity of NPPK. As we have shown here, there are spatial and temporal variations in $a^{14}C$. It is also difficult to estimate the fraction of total ingested food that originates from the close vicinity of NPPK. It is reasonable to assume that the local products are dispersed in the general food supply system, and not all consumed by local population. The simplest and the most conservative model of ingestion takes "the most exposed adult person" who would consume only apples from the close environment of NPPK throughout the year. In that case, and for the year 2006 (highest mean ^{14}C activity) the increase of the total annual dose is about 0.1%. Since this the most conservative estimate is unrealistic and not probable, we propose the following more realistic although still rather conservative model. We suppose that: i) in the daily consume of 0.3 kg of carbon, one half comes from the fruits; ii) 6 months the fruits from the environment of NPPK are not available, and during this period one can ingest only the food which has $a^{14}C$ as it is at the control site; iii) during the other 6 months, ⅓ of the total ingested carbon comes from fruits from any point of the monitored area around the NPPK, and the rest ⅔ come from the control site. Therefore, this model is equivalent to the assumption of ingestion of the apples from the vicinity of the power plant during 1 month in a year, while during the rest of the year carbon comes from the wider area. Taking into account these assumptions, we can calculate the increase of the equivalent ^{14}C dose relative to the natural ^{14}C dose measured at the control point, as well as relative to the total natural dose in our country (1.22 mSv). The results for each year since 2006 are shown in Table 2.

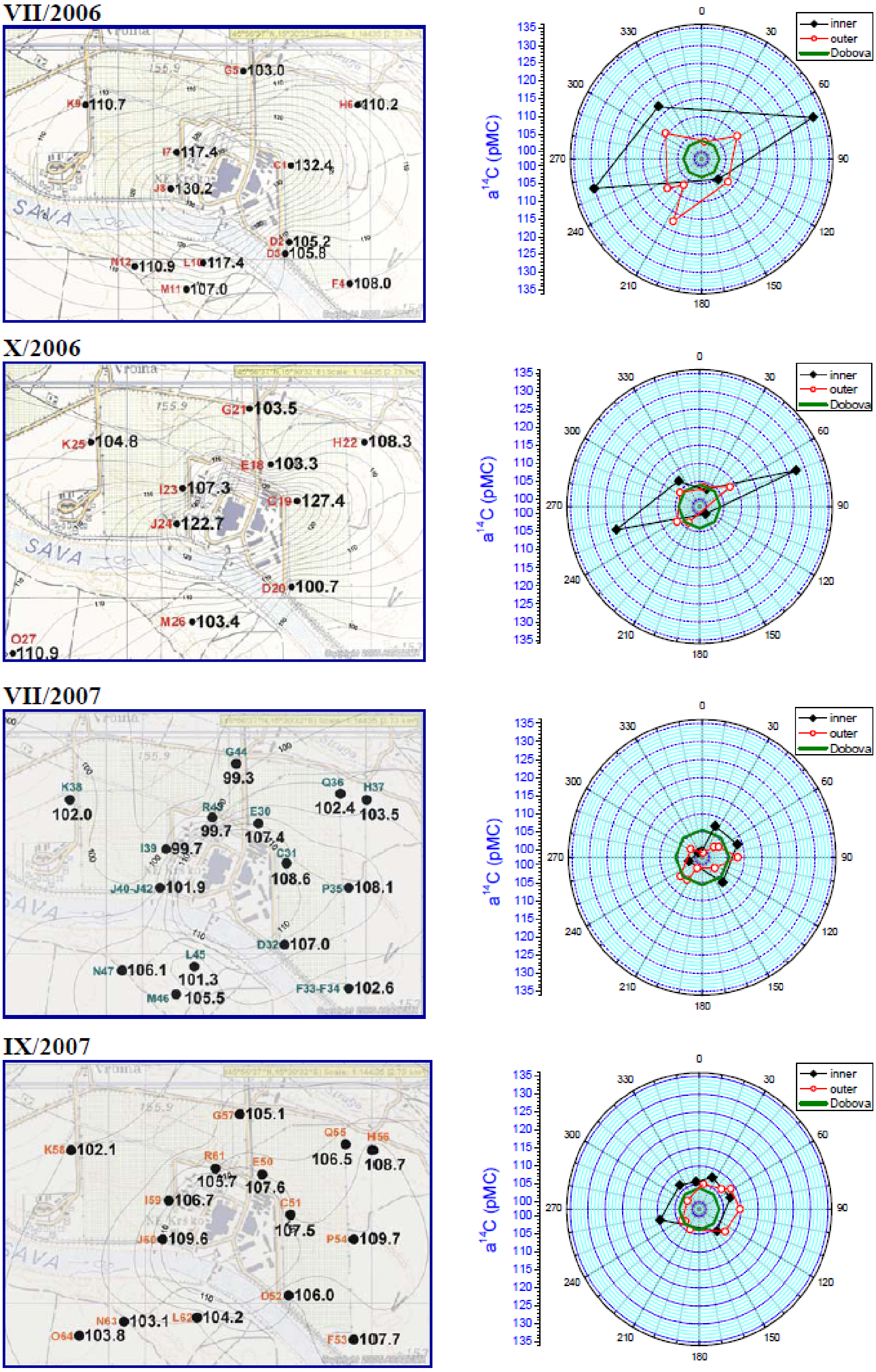

Figure 4. Spatial distribution and polar diagrams of relative specific ^{14}C activity (expressed in pMC).

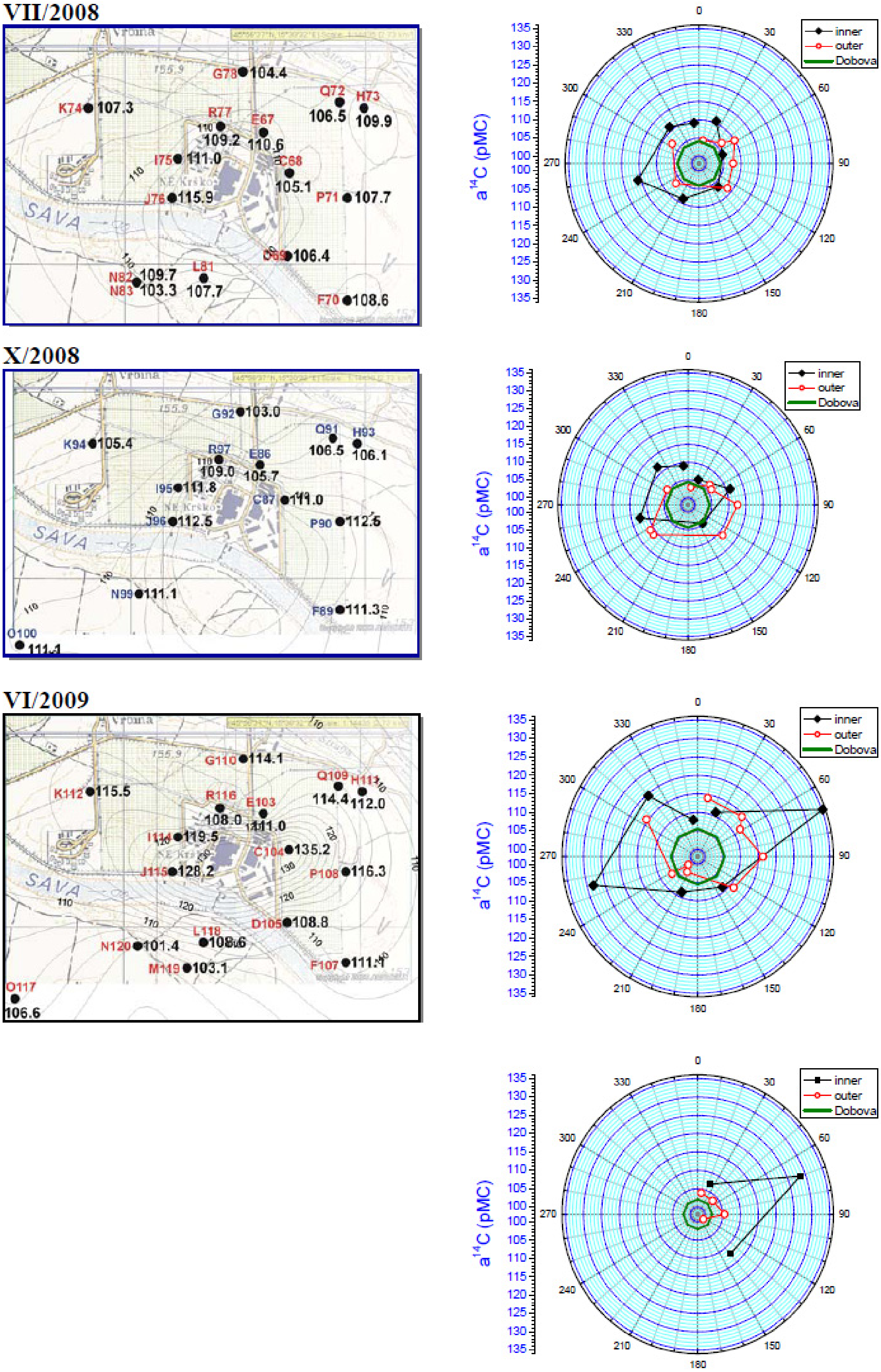

Figure 4 continued. Spatial distribution and polar diagrams of relative specific ^{14}C activity (expressed in pMC). Data for the sampling campaign in September 2009 not yet completed.

Table 2. Increase of equivalent ^{14}C dose relative to the natural ^{14}C dose measured at control point, as well as well as the increase relative to the total natural dose in our country (1.22 mSv).

Year	$a^{14}C$ [pMC] NPPK *	$a^{14}C$ [pMC] Dobova	^{14}C dose [μSv] NPPK	^{14}C dose [μSv] Dobova	Increase of ^{14}C dose (%)	Increase of total dose (%)
2006	111.6	103.5	14.95	14.86	0.65	0.0079
2007	105.0	104.7	15.03	15.03	0.02	0.0003
2008	108.8	104.3	15.02	14.97	0.35	0.0044
2009	110**	103.8	14.97	14.89	0.50	0.0061

* mean value of ^{14}C activity in inner and outer circle of NPPK
** data for September 2009 not yet complete

Conclusion

Monitoring of $a^{14}C$ in biological samples in immediate vicinity and wider environment of the Nuclear Power Plant Krško in Slovenia resulted in spatial distribution of $a^{14}C$ that is clearly determined by the dominant wind directions (SW – NE) and influenced by the outflow of ^{14}C from the release point of the plant ventilation system. The difference to the natural present-day $a^{14}C$ is determined relative to the control site Dobova, about 10 km SE from NPPK. At most sites the difference ranges from 0 to several pMC, being the highest 29 pMC at site **J** in July 2006. However, these differences do not affect significantly the total annual dose due to natural radiation sources, according to our model of ingestion.

Acknowledgments

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Uranium and long-lived decay products in water of the Mulde River

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Abstract

The Mulde River is a left side tributary of the Elbe River and mainly situated in Saxony. The river system consists of the Freiberger Mulde River and Zwickauer Mulde River, which merge to form the Vereinigte Mulde River. The Zwickauer Mulde River drains the former uranium mining and milling areas in Saxony. This research project was established to quantify the long-term effect of the former uranium mining and milling activities by investigating the content of uranium and polonium of the water of the Mulde River. The specific uranium activity in samples from the Zwickauer Mulde River is still high compared with the natural background. The values measured in the water of the Vereinigte Mulde River are also elevated, but to a lesser extent due to the dilution effect caused by the merging with the Freiberger Mulde River. Furthermore, the level of contamination of the river water decreased by at least a factor of three as compared to the early 1990's. The specific activity of polonium shows no correlation with that of uranium and is generally much smaller.

Introduction

At the time of the Warsaw Pact, the former German Democratic Republic (GDR), was the third largest producer of uranium in the world and the most important supplier of uranium for the USSR. The Zwickauer Mulde River in Saxony and the Weiße Elster River in Thuringia are the most important river systems draining the uranium mining and milling dominated areas of the western Ore Mountains and its foreland, resulting in accordingly high heavy metal loads. Thus they are of particular interest for radiation protection and radioecology. Today this area is subject to a remediation project which, due to its large scale, can be characterized as pioneer work.

The Mulde River is a left side tributary of the Elbe River and mainly situated in Saxony. The river system consists of three main rivers: Zwickauer Mulde, Freiberger Mulde and Vereinigte Mulde. The Muldenberg Reservoir in the western Ore Mountains is deemed to be the source of the Zwickauer Mulde River. Four kilometers downstream of the dam, which is used as a drinking water reservoir, the river already receives contaminated water from the mine Schneckenstein. On its way through the Ore Mountains the river flows through the uranium mining area Aue and Bad Schlema. Downriver of Zwickau the Zwickauer Mulde River passes Crossen and its uranium ore processing plants and merges with the Freiberger Mulde River at river kilometer 162. The source of the Freiberger Mulde River is located in the Czech Republic, about 5 km from the German border on the ridge of the eastern Ore Mountains. It drains mining areas, however there is no uranium mining in its catchments area. The Vereinigte Mulde River, which is often simply called Mulde River, originates from the merging of the two frontal flows Freiberger Mulde and Zwickauer Mulde and disembogues into the Elbe River north of Dessau at river kilometer 124. At high water-levels the Mulde River is deemed to be the fastest flowing river in Central Europe. Due to its origin in a high-mineral and high-ore affected area it is the most significant reason for heavy metal entering into the Elbe River and thus into the North Sea.

This research project was established to quantify the long-term effect of the former uranium mining and milling activities by investigating the content of uranium and polonium of the water of the Mulde River. It is part of a work package dealing with transport and availability of uranium and its decay products in the Mulde floodplains, which in turn is part of a joint project on radionuclides in the environment and their transport to man via food chains, supported by the German Federal Ministry for Education and Research (BMBF).

Sampling and analysis

A total of 26 water-samples were collected in April, May and October 2008. 19 samples were collected along the course of the Zwickauer Mulde River and the Vereinigte Mulde River. Water from the Freiberger Mulde River, the headwaters of the Zwickauer Mulde River, a small influent of the Zwickauer Mulde River, and from the Leine River near Hanover, respectively, were used as reference samples. Figure 1 shows the Mulde system and the sampling locations.

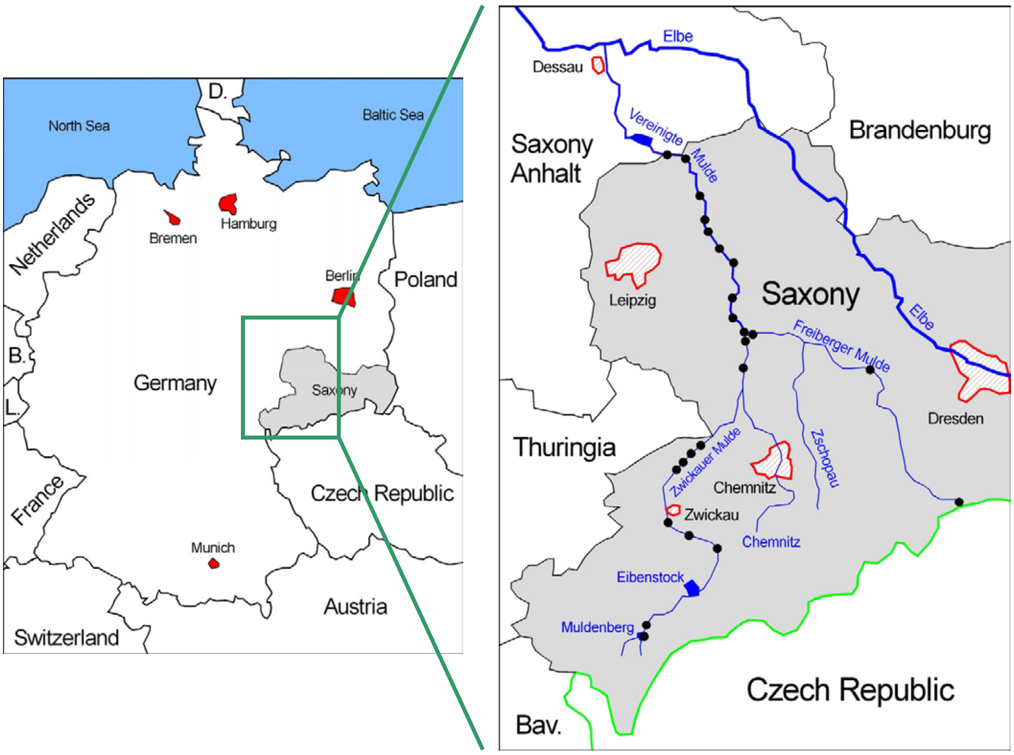

Fig. 1. The Mulde River system with the sampling locations.

Generally, four litres of water were collected at each sampling location. In four cases only two litres per sample were taken. The water was filtered on-site using folded filters of a pore size of 7 µm. Subsequently the samples were acidulated with nitric acid to a pH of 1.5 to 2 and stored on ice.

In the laboratory the samples were filtered through cellulose nitrate filters of a pore size of 0.45 µm. As a central step an enrichment of radionuclides via iron hydroxide-coprecipitation was performed. Following this polonium and uranium were separated by solid-phase chromatography using Pb-Resin and UTEVA from Eichrom, respectively. The polonium sample was obtained by means of autodeposition on nickel. The corresponding uranium sample was prepared using electrical deposition on stainless steel. Both samples were measured alpha-spectrometrically by a surface barrier detector. Figure 2 gives an overview of the analytical procedure.

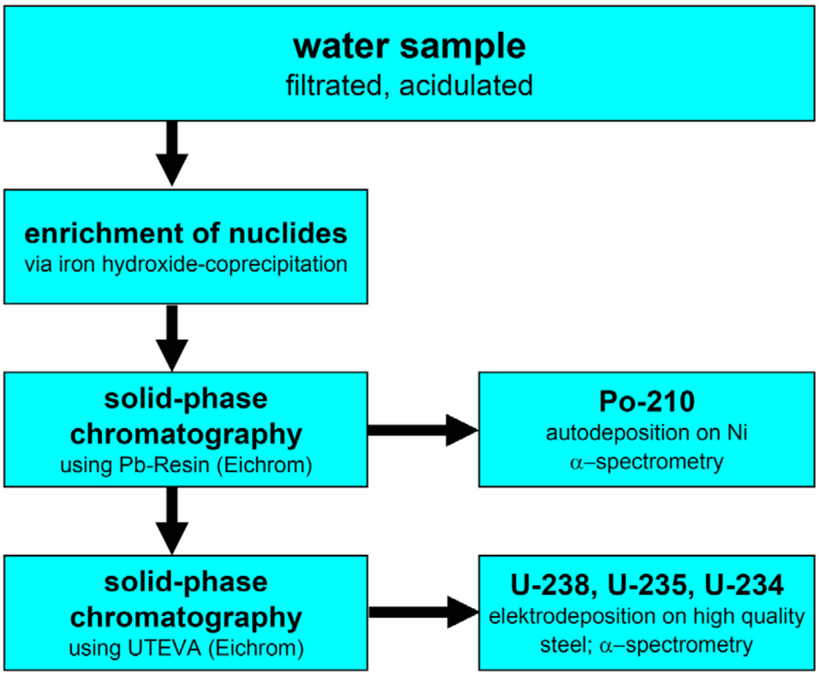

Fig. 2. Overview of the analytical procedure.

Results

Table 1 summarises the measured specific activities of uranium and polonium. The samples are numbered consecutively in flow direction, in which the two letters denote the origin of the sample ("FM“ = Freiberger Mulde River, "ZM“ = Zwickauer Mulde River, "VM“ = Vereinigte Mulde River).

Figure 3 shows the variation of the specific activities of uranium and polonium along the course of the Mulde River. Additionally, the arithmetic mean of five polonium blanks and the upper 1σ-interval are plotted. The blank value determination was performed for the overall procedure using ultra-pure water (18.2 MΩ*cm). Determination of five blanks results in an arithmetic mean of 1.75 mBq/kg with a 1σ-interval ranging from 0.26 mBq/kg to 3.24 mBq/kg. The determination of uranium blanks results in a value close to the detection limit.

Table 1. Specific activities (*a*) and related uncertainties (u(*a*)) for uranium and polonium in the water of the Mulde River; red values marked with "<“ lie below detection limit.

		U-238 (mBq/kg)		U-234 (mBq/kg)		U-235 (mBq/kg)		Po-210 (mBq/kg)	
		a	u (*a*)	*a*	u (*a*)	*a*	u (*a*)	*a*	u (*a*)
1	Leine	13.4	1.3	26.3	2.4	0.6	0.2	4.3	0.9
2	Bach	8.4	1.0	11.8	1.3	< 0.7	---	3.6	0.5
3	FM1	2.6	0.3	2.7	0.3	0.2	0.1	3.0	0.3
4	FM2	2.0	0.2	2.2	0.3	< 0.1	---	2.8	0.6
5	FM3	12.2	1.2	16.6	1.6	0.8	0.2	1.1	0.3
6	ZM1	2.0	0.2	3.4	0.4	0.5	0.1	< 5.0	---
7	ZM2	3.0	0.4	3.5	0.5	< 0.3	---	3.7	1.9
8	ZM3	73.9	7.5	74.0	7.5	2.0	0.7	10.7	2.5
9	ZM4	69.3	7.1	64.1	6.6	< 1.9	---	13.2	3.1
10	ZM5	63.5	6.4	68.5	6.8	1.9	0.6	3.8	0.8
11	ZM6	79.0	6.8	84.2	7.2	2.9	0.4	10.9	1.1
12	ZM7	85.9	8.1	86.6	8.2	3.5	0.7	2.8	0.3
13	ZM8	83.7	7.7	95.2	8.6	4.4	0.7	4.2	0.4
14	ZM9	82.4	7.8	87.7	8.3	3.5	0.6	4.9	0.9
15	ZM10	69.5	5.8	80.0	6.7	4.9	0.4	4.2	0.4
16	ZM11	74.0	6.6	78.9	7.0	3.4	0.5	2.6	0.4
17	VM1	74.4	6.9	81.4	7.6	3.0	0.5	3.5	0.5
18	VM2	25.2	2.6	31.6	3.2	0.6	0.3	2.6	0.5
19	VM3	38.6	3.3	46.0	3.9	1.8	0.2	2.1	0.4
20	VM4	30.9	3.1	40.0	3.9	1.0	0.3	3.5	0.4
21	VM5	30.8	1.4	38.6	1.7	1.3	0.3	7.3	0.6
22	VM6	33.4	1.5	34.1	1.6	1.0	0.2	3.8	0.5
23	VM7	30.8	2.6	36.9	3.1	1.7	0.2	11.0	0.8
24	VM8	32.4	3.1	36.8	3.4	1.3	0.3	4.8	1.1
25	VM9	30.9	2.9	36.4	3.4	1.0	0.2	2.3	0.4
26	VM10	30.1	3.2	34.2	3.6	0.9	0.3	3.1	0.4

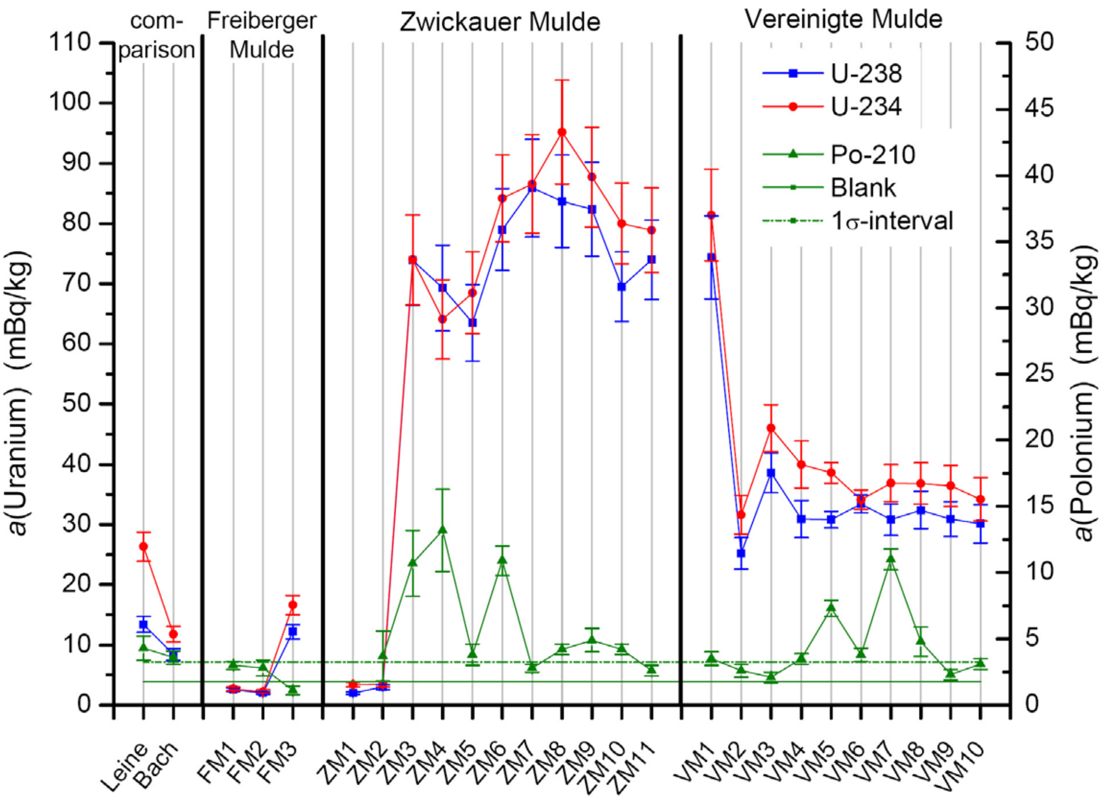

Fig. 3. Variation of the specific activities of uranium-238, uranium-234 and polonium-210, respectively, along course of the the Mulde River; additionally, the arithmetic mean of five polonium blanks and the upper 1σ-interval is plotted.

Discussion

Uranium

The measured values of the uranium isotopes (U-238, U-234 and U-235) exhibit the composition of natural uranium in almost all of the samples. This is in accordance with the expectation, as isotopic enrichment of uranium was no common practice in Germany.

The values can be divided roughly into three groups (see Fig. 3). The first group represents the reference values, which belong to non-affected areas with regard to uranium mining and milling activities. This includes the samples from the Freiberger Mulde River (FM1, FM2), from the small influent of the Zwickauer Mulde River (Bach) and the Leine River (Leine), as well as the samples ZM1 and ZM2 from the headwaters of the Zwickauer Mulde River. The first contaminated water from the mine Schneckenstein receives the River about 4 km downriver of the dam of the Muldenberg reservoir, which is regarded as the source of the river, and a few hundred meters downriver of the sampling location of ZM2 (see Fig. 4), respectively. The mean specific activity of uranium-238 for the reference samples is 6.2 mBq/kg, which represents a typical background value for unaffected surface water. The second group of values can be attributed to the Zwickauer Mulde River (ZM3 to VM1) – with a mean value of 75.5 mBq/kg for uranium-238. The third group consists of samples from the Vereinigte Mulde River (VM2 to VM10), showing a mean value of 31.5 mBq/kg. The VM1 sample was collected about 400 meters downriver of the rivers' confluence, more precisely, at the side, where the Zwickauer Mulde River flows in. It can be assumed that at this particular point no considerable mixing of the rivers has taken place.

Along the Zwickauer Mulde River significant variations of the uranium activities can be detected. Hence, Fig. 4 shows the variation of the specific activities of uranium-238 plotted against the river kilometers of the Zwickauer Mulde River extending to a point 15 km downstream of the confluence with the Freiberger Mulde River. Additionally, values from the years 1991 to 1993, determined by Beuge et al. [1], are depicted in order to allow a comparison. The numerical values are listed in Table 2. The determination of the uranium concentration by Beuge et al. [1] was performed employing total-reflection X-ray fluorescence analysis (TXRFA). The results are given as mass concentration in µg/L. The conversion to specific activity of uranium-238 in mBq/kg was performed assuming a density of 1 g/mL. Measurements of samples collected along the Freiberger Mulde River and the Vereinigte Mulde River yielded values, which rarely exceeded the detection limit of TXRFA at ~ 60 mBq/kg (5 µg/L). Thus, they are not considered here.

Table 2. Comparative values determined by Beuge et al. [1]; samling locations are specified by the corresponding river kilometers; uranium concentration is given as mass concentration (*β*) and as specific activity (*a*) of uranium-238, respectively.

sampling location	river kilometer	*β* (U) (µg/L)	*a* (U-238) (mBq/kg)
weir Hartenstein	55	27	335
short before Crossen	80	16	198
near Glauchau	94	23	285
near Sermuth	161	14	174

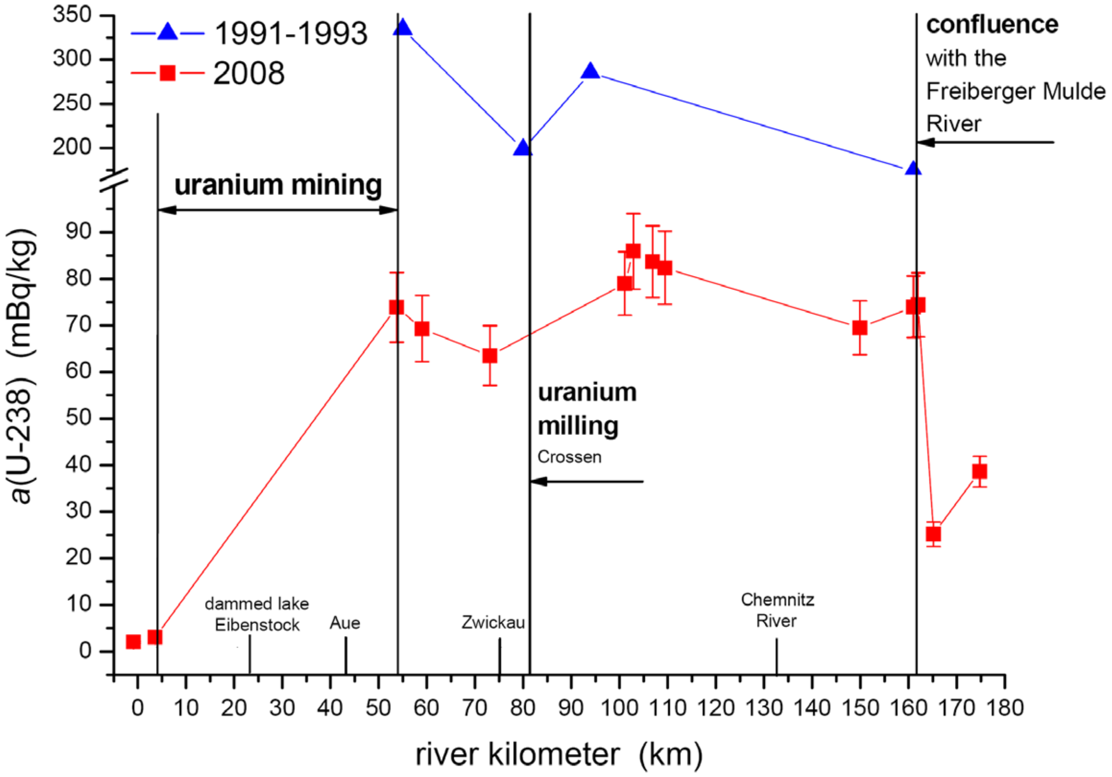

Fig. 4. Variation of the specific activities of uranium-238 in water of the Mulde River, plotted against the river kilometers; comparative values for the years 1991 to 1993 adapted from Beuge et al. [1]

As mentioned before, the first two samples taken from the source of the Zwickauer Mulde River represent the natural background of uranium. The next sample (ZM3) was collected at Hartenstein, where the river has already passed the uranium mining area around Aue and Bad Schlema. According to this position, the sample shows a high uranium activity. Following the course of the river to Zwickau (ZM5), the specific activities seem to decrease slightly. This effect can be explained by dilution with less contaminated water from tributaries. Downstream of Zwickau the specific activity of uranium in the water increases again. This increase results from the effluents of the uranium milling industry near Crossen. In the further course of the Zwickauer Mulde River, the activity remains approximately constant until the confluence with the Freiberger Mulde River. Due to measurement uncertainties, the values are in agreement with a slight decrease of the specific activity corresponding to dilution effects caused by tributaries, especially the Chemnitz River, largest tributary of the Zwickauer Mulde River. In contrast, the confluence of the Zwickauer Mulde River with the uncontaminated Freiberger Mulde River reveals a clear dilution effect, and correspondingly a significant decrease of the specific uranium activity.

The measurements from the early 1990's show the same characteristics, but at a much higher (activity) level. During the last twenty years, the contamination of the Mulde River decreased considerably, on average by a factor of 3.4. This effect decreases in the course of the Zwickauer Mulde River. At the upper reaches of the river, at Hartenstein, the difference between the values measured in 2008 (73.9 mBq/kg) and in the 1990's (335 mBq/kg) is considerably high, resulting in a decrease by a factor of 4.5. At Sermuth, which is located close to the confluence of the Freiberger Mulde River, the difference between the current value (74 mBq/kg) and the one from the 1990's (174 mBq/kg), yield a decrease by a factor of 2.3. This indicates that the decrease of the contamination mainly results from the decreasing emission of the uranium mining and milling industry and accordingly can be attributed to the remediation activities.

The specific uranium activity remains constant along the Vereingte Mulde River, showing an arithmetic mean of 31.5 mBq/kg. The variations of the first three measured values show a correlation between the side of the river on which these samples were taken and the confluence of the Zwickauer and Freiberger Mulde River. This has already been explained in the case of sample VM1. For the samples VM2 and VM3 an incomplete mixing of the water of the two rivers can be assumed as well. The sample VM2 was taken on the right-hand side in flow direction, and thus a stronger influence of the Freiberger Mulde River can be assumed. In contrast, the samples VM3 and VM1 were taken on the left-hand side, which causes a stronger influence of the Zwickauer Mulde River. At the same time the convergence towards the mean value shows the progressive mixing of the water. A decrease of the uranium activity along the Vereinigte Mulde River due to the dilution by smaller tributaries cannot be observed, because the influence of the tributaries decreases with enlarging of the river.

Polonium

The specific activities of polonium-210 are much lower than those of uranium. A large number of the measured values are well below 5 mBq/kg, and thus those data are relatively difficult to distinguish from the background. There is no apparent correlation with the specific activities of uranium. The low specific activities of polonium in the water can be explained by its very strong affinity to adsorption. Due to the strong adsorption to sediments the variations of the polonium activities are expected to change within a short distance. Only five samples exhibit specific activities above 5 mBq/kg. The samples ZM3 and ZM4 are the first two samples after the Zwickauer Mulde River passed the uranium mining region of the Ore Mountains. Consequently, the river water shows higher values for the polonium activities than the uncontaminated headwaters. The high activities of the other samples cannot be explained yet.

Radioecological relevance

The uranium and polonium concentrations found in river water are insignificant from the radioecologically point of view. The recommended value for uranium in drinking water given by the Federal Environment Agency (UBA) is < 10 µg/L [2]. This corresponds to a specific uranium-238 activity of 123.5 mBq/kg. None of the samples shows a uranium activity exceeding this value. In its recommendation from 2001 the EURATOM states 0.1 Bq/kg as maximum specific activity for polonium in drinking water [3]. Again, none of the analysed samples reaches such a high polonium activity.

Conclusions

During the last 20 years, the contamination of the Mulde River decreased by at least a factor of three. This effect mainly results from the decreasing emission of the uranium mining and milling industry and accordingly can be attributed to the remediation activities. On the other hand, an influence of the former uranium mining and milling activities can still be detected. The specific uranium activity in samples from the Zwickauer Mulde River is still high compared with the natural background. The values measured in the water of the Vereinigte Mulde River are also elevated, but to a lesser extent due to the dilution effect caused by the merging with the non-affected Freiberger Mulde River. The specific activity of polonium shows no correlation with that of uranium and is generally much smaller.

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^{129}I in Finnish waters

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Abstract

^{129}I is a long-lived beta-emitting (E_{max} 154,4 keV) radioisotope of iodine. Its half-life is 15,7 million years. ^{129}I is produced mainly by human nuclear activities and especially it has been released to the environment from the spent nuclear fuel reprocessing plants. In the pre-nuclear era ^{129}I/^{127}I ratios in the environment were approximately 10^{-12}. Nowadays ^{129}I/^{127}I ratios have reached values from 10^{-10} to 10^{-4}.

In this study, activity concentrations of ^{129}I and its distribution into various chemical species (iodide I-, iodate IO3- and bound in organics) were analyzed from four lakes in Finland and from four different sea locations on the Gulf of Finland, the Bothnian Sea and the Bothnian Bay. ^{129}I was also analyzed from four rainwater samples.

After filtering the 0.3 l water samples, separation of various iodine species was done by anion exchange chromatography: $^{129}IO_3^-$ passes through an anion exchange resin bed in NO_3^- form while $^{129}I^-$ absorbs into the bed. $^{129}I^-$ is eluted from resin with NaClO. Finally samples were precipitated by $AgNO_3$ to form AgI and ^{129}I was measured by accelerator mass spectrometry (AMS). Stable iodine (^{127}I) was analyzed by inductively coupled plasma mass spectrometry (ICP-MS).

First results from a lake in the southern Finland and from sea water taken from the Finnish Bay in front of Helsinki show that levels of ^{129}I in lake water are around 1×10^9 atoms per litre while in sea water the levels are 4 – 5 times higher. ^{129}I occurs both in lake and sea water mainly in iodide form and the fraction of iodate form is only about 5%. The ^{129}I/^{127}I ratio is clearly elevated compared to natural levels, and are approximately the same in sea and in lake, 14×10^{-8} and 8×10^{-8}, respectively. These results are only preliminary and a better picture of the situation will be obtained after finalizing the project. The results obtained so far are, however, at the same level as obtained in Swedish studies at the same latitudes.

Monitoring and assessment of radioactivity in the Baltic Sea coordinated by HELCOM

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Abstract

The Baltic Marine Environment Protection Commission (HELCOM) works to protect the marine environment of the Baltic Sea from all sources of pollution through intergovernmental co-operation between Denmark, Estonia, the European Community, Finland, Germany, Latvia, Lithuania, Poland, Russia and Sweden. Investigations of radioactivity in the Baltic Sea has been part of the HELCOM work since 1986 and carried out by the MORS project (Monitoring of Radioactive Substances) with participation from laboratories and institutes in the countries and organizations mentioned above including the IAEA.

The objective of the MORS project is to implement the Helsinki Convention on matters related to monitoring and assessment of radioactivity in the Baltic Sea. The Contracting Parties to the Convention carry out basic monitoring programmes and transfer the data to the HELCOM database. The monitoring is carried out according to guidelines which are revised annually. The guidelines give details on sampling locations, sample types covering seawater, sediment, fish, aquatic plants and benthic animals, and radionuclides to be determined in the samples. The guidelines furthermore specify the format for reporting the data to the database. High quality of the data is ensured through participation of the MORS laboratories in frequent group intercomparisons of analytical data from laboratory analyses as well as from evaluation

of the reported data at the annual meetings of the MORS group. Intercomparisons cover laboratory analyses of radioactivity in seawater, sediment and biota.

The data collected is summarized in indicator fact sheets that present results of radioactivity in seawater and fish from the Baltic Sea. More comprehensive assessments of radioactivity in the Baltic Sea are made by the group at larger intervals. The assessments include summaries of sources and inputs of radioactivity into the Baltic Sea, environmental levels and model-assisted estimates of radiation doses to man.

Results obtained in monitoring programmes and environmental studies carried out during more than 40 years in the sea areas surrounding the Finnish NPPs

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Abstract

Environmental effects of thermal and radioactive discharges from the Loviisa and Olkiluoto NPPs in the recipient sea areas were assessed. The effects of cooling water on the temperatures in the sea were most obvious in winter. The formation of a permanent ice cover has been delayed, and the break-up of the ice has been advanced. The prolonging of the growing season has been the most significant biological effects of thermal pollution. At Loviisa, the thermal discharges have increased the production of organic matter in the discharge areas, which has led to more organic bottom deposits. The depletion of oxygen has caused remobilization of phosphorus from the bottom sediments, and contributed to deterioration of benthic macrofauna. Phytoplankton primary production has doubled in the area, and the thermal discharge has contributed to a stronger increase in the discharge area than in the intake area. The eutrophication of littoral vegetation has been the most obvious and unambiguous biological effect of the heated water in both areas.

Small amounts of local discharge nuclides were regularly detected in environmental samples taken from the discharge areas: tritium in seawater, and activation products in suspended particulate matter, bottom sediments and in indicator organisms. Discharge nuclides from the local nuclear power plants were almost exclusively detected at the lower trophic levels of the ecosystem. The concentrations of local discharge nuclides in the environmental samples have noticeably decreased in both areas in recent years. The radiation doses caused by the radioactive discharges to the population and to the biota were very low, practically insignificant. The effects of the thermal discharges were more significant, at least to the wildlife in the discharge areas of the cooling water, although the area of impact has been relatively small. The results show that the nutrient level and the exchange of water in the discharge area of a nuclear power plant are of crucial importance.

Introduction

During recent decades, thermal and radioactive discharges from nuclear power plants into the aquatic environment have become the subject of lively debate as an ecological concern. Recently, an increasing demand for facts has appeared in context with the Environmental Impact Assessment procedures that are being in progress for planned new nuclear power units in Finland. This paper is based on a thesis examined in University of Helsinki in September 2009 (Ilus 2009). The target of the thesis was to summarize the large quantity of results obtained in extensive monitoring programmes and studies carried out in recipient sea areas off the Finnish nuclear power plants at Loviisa and Olkiluoto during more than four decades.

Especially in the conditions specific for the northern Baltic Sea, where biota is poor and adapted to relatively low temperatures and to seasonal variation with a cold ice winter and a temperate summer, an increase in temperature may cause increased environmental stress to the organisms. Furthermore, owing to the brackish-water character of the Baltic Sea, many organisms live there near the limit of their physiological tolerance. On the other hand, the low salinity increases the uptake of certain radionuclides by many organisms in comparison with oceanic conditions.

The sea areas surrounding the Finnish nuclear power plants differ from each other in many respects (efficiency of water exchange, levels of nutrients and other water quality parameters, water salinity and consequent differences in species composition, abundance and vitality of biota). In addition, there are differences in the discharge quantities and discharge design of the power plants. In the thesis the environmental effects of the two power plants on the water recipients are compared and their relative significance is assessed.

There are four nuclear power plant units in Finland: two 488 MW_e units at Loviisa and two 840 MW_e units at Olkiluoto. The units at Loviisa were commissioned in 1977 and 1980, and those at Olkiluoto in 1978 and 1980. Environmental studies were initiated at Loviisa about ten years and at Olkiluoto six years before the start of operation of the power plants. Thus, 40-year-long time-series of results are available from the hydrographical, biological and radioecological studies carried out for monitoring the environmental effects of the thermal and radioactive discharges from the power plants in the recipient sea areas.

Brackish water from the Baltic Sea is used for cooling in the Finnish nuclear power plants, and both the thermal and liquid radioactive discharges are let out into the sea. Each of the power plants use cooling water at a rate of about 40–60 m^3s^{-1}, and the temperature rises in the condensers by about 10–13ºC. Loviisa NPP is located on the coast of the Gulf of Finland and Olkiluoto NPP on that of the Bothnian Sea. The state of the Gulf of Finland is clearly more eutrophic: the nutrient (total phosphorus and total nutrient) concentrations in the seawater are about 1½–2 times higher at Loviisa than at Olkiluoto, but the total phosphorus concentrations have still increased in both areas, even doubling at Loviisa between the early 1970s and 2000 (Fig. 1). The salinity is generally low in the brackish-water conditions of the northern Baltic Sea. However, the salinity of surface water is about 1‰ higher at Olkiluoto than at Loviisa (varying at the latter from near to 0‰ in early spring to 4–6‰ in late autumn). Thus, many marine and fresh-water organisms live in the Loviisa area close to their limit of existence, which makes the biota sensitive to any additional stress. The characteristics of the discharge areas of the two sites differ essentially from each other in many respects: the discharge area at Loviisa is a semi-enclosed bay in the inner archipelago, where the exchange of water is limited, whereas the discharge area at Olkiluoto is more open, and the exchange of water with the open Bothnian Sea is more effective.

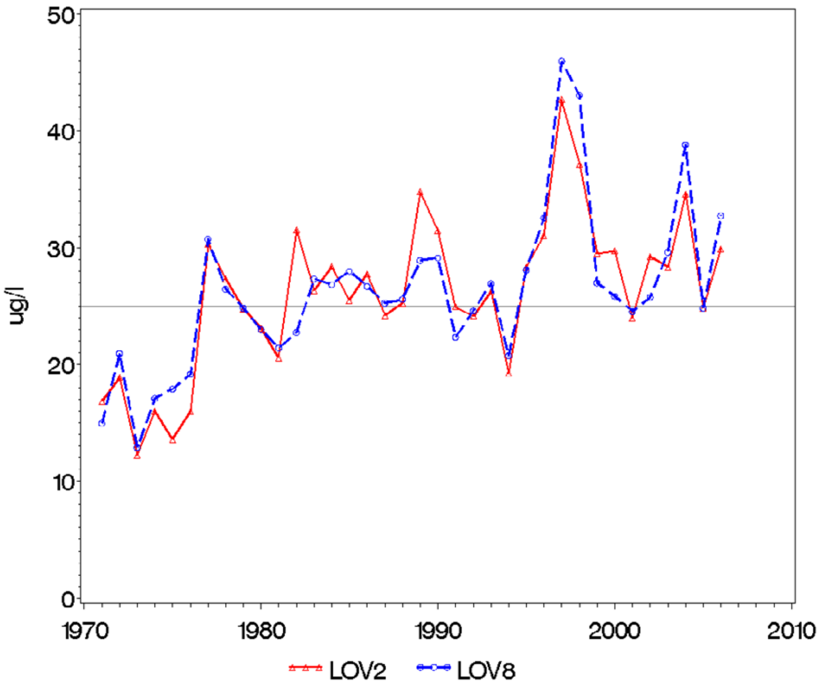

Fig. 1. Average total phosphorus concentrations (µgP l^{-1}) of surface water during the growing seasons (May–October) in the intake (Station 8) and discharge area of cooling water (Station 2) at Loviisa in 1971–2006.

Results and discussion

Environmental effects of cooling water

The effects of the cooling water on the temperatures in the sea were most obvious in winter, when the conditions also most fundamentally differed from those of the natural state. Thermal discharges have significantly affected the ice conditions in the vicinity of the power plants. The formation of a permanent ice cover in the discharge areas has been delayed in early winter. On the other hand, the break-up of the ice occurs earlier in springs so that the growing season has been prolonged at both ends. From the biological point of view, the prolonging of the growing season and the disturbance of the overwintering time, in conditions where the biota has adjusted to a distinct rest period in winter, have been the most significant biological effects of the thermal pollution. Aquatic organisms in the northern Baltic Sea are acclimatized to a distinct annual winter period. The shortening of the ice winter or a total lack of ice cover has led to a blurring of the limits between the growing season and the winter season.

Table 1. Change in mean surface water temperature of the growing season at eight sampling stations off Loviisa during 1976–2006. Station 8 is located in the intake area of cooling water and considered as a reference station.

Area	Hästholmsfjärden			Våd-holms-fjärden	Klobb-fjärden	Orren-grunds-fjärden	Hudöfjärden	
Station	5	2	3	4	1	7	8	10
Waterway distance from outlet [km] and direction	0.4 E	1.0 NE	1.7 E	2.8 SE	3.4 NE	5.4 SE	3.0 SW	3.4 SW
Change in mean surface water tem-peratureof growing season (May-Octo-ber) in 1976-2006:								
Probability p_F	<.0001	0.0002	0.0005	0.0017	0.0034	0.0527	0.0628	0.1343
Average difference to Station 8 in 1997-2001 [°C]	4.3	2.6	3.1	1.3	2.4	-0.3	0	-0.1
Average difference to Station 8 in 2002-2006 [°C]	4.9	2.6	3.3	1.2	2.4	-0.3	0	-0.3

During the growing season, the cooling water has raised the mean surface water temperatures at Loviisa by 4–5ºC at a distance of 0.4 km from the outlet and by 2.5–3ºC at a distance of 1–2 km in the discharge area, Hästholmsfjärden (Table 1). The rise in temperature has also been statistically significant at Station 4 in Vådholmsfjärden (distance 2.8 km) and at Station 1 in Klobbfjärden (distance 3.4 km), but not at Station 7 in Orrengrundsfjärden (distance 5.4 km).

A temperature rise generally increases the metabolic activity and growth rate of aquatic organisms. This means an increased production of organic matter, and thus, thermal pollution promotes the eutrophication process in eutrophied environments. The raised temperature also increases the rate of decomposition of organic matter in the receiving water bodies and leads to depletion of oxygen in deep water layers.

The hydrographical and biological results in the Loviisa area indicated a clearly higher level of eutrophy, which was based on the state of the whole Gulf of Finland. Thus, it was a challenge to distinguish the local effects of thermal discharges from the general eutrophication process of the Gulf of Finland.

During the past 40 years the soft-bottom macrofauna has steeply deteriorated at many sampling stations, at some to the point of almost complete extinction. A similar decline of the macrozoobenthos has been reported over the whole eastern Gulf of Finland. However, the local eutrophication process seems to have contributed into the decline of the bottom fauna in Hästholmsfjärden at Loviisa (Fig. 2). Thermal discharges have increased the production of organic matter, which again has led to more organic bottom deposits. Furthermore, these have increased the affinity of the isolated deeps for a depletion of oxygen, which has in turn caused a strong remobilization of phosphorus from the bottom sediments to the water phase.

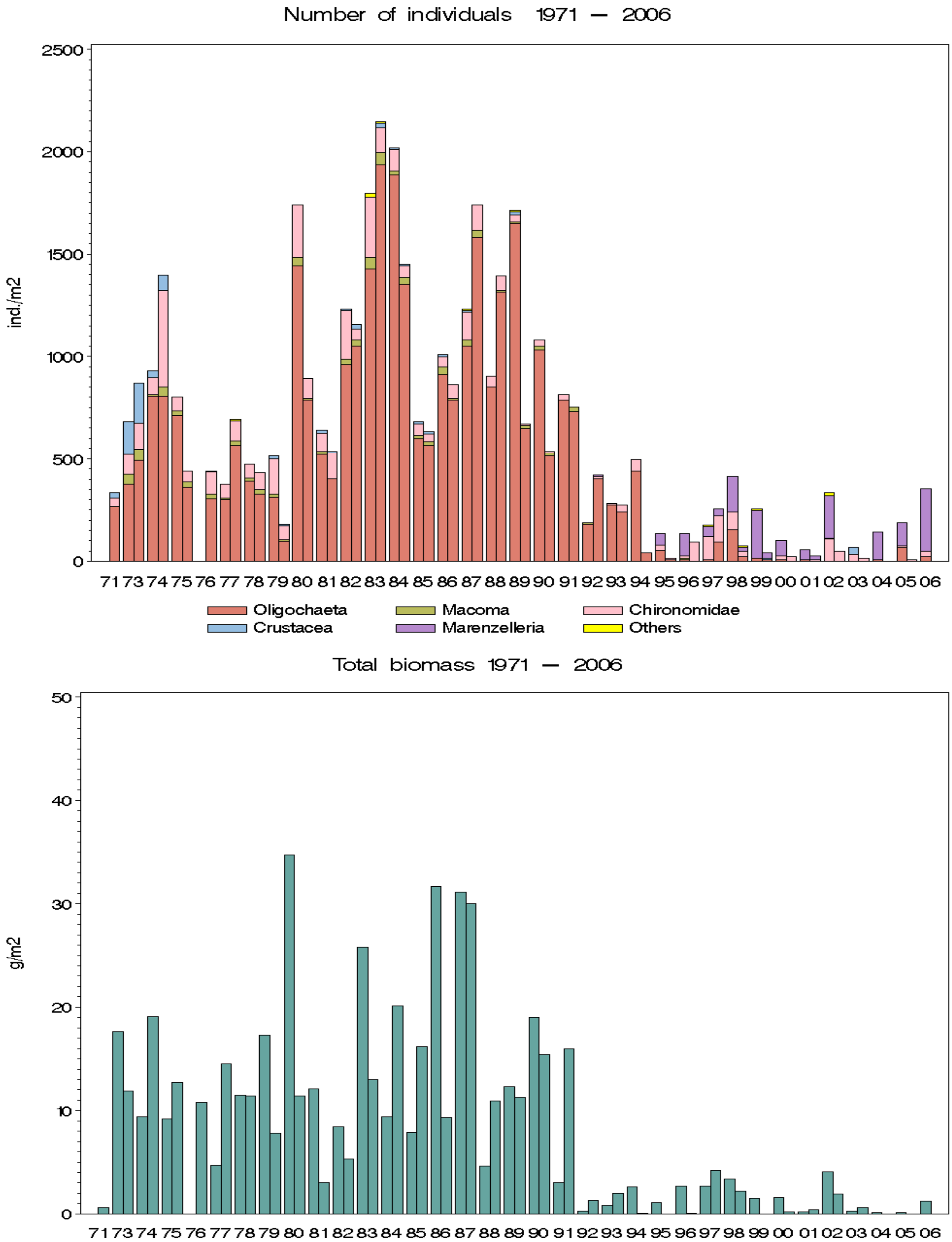

Fig. 2. Abundance of main species or taxonomic groups (ind. m^{-2}) and total biomass of macrozoobenthos (g m^{-2}) at a sampling station situated 0.4 km off the cooling water outlet at Loviisa.

Phytoplankton primary production and primary production capacity doubled in the whole area between the late 1960s and the late 1990s, but started to decrease a little at the beginning of this century. The focus of the production shifted from spring to mid-and late summer. The general rise in the level of primary production was mainly due to the increase in nutrient concentrations over the whole Gulf of Finland, but the thermal discharge contributed to a stronger increase of production in the discharge area compared to that in the intake area of the cooling water (Fig. 3).

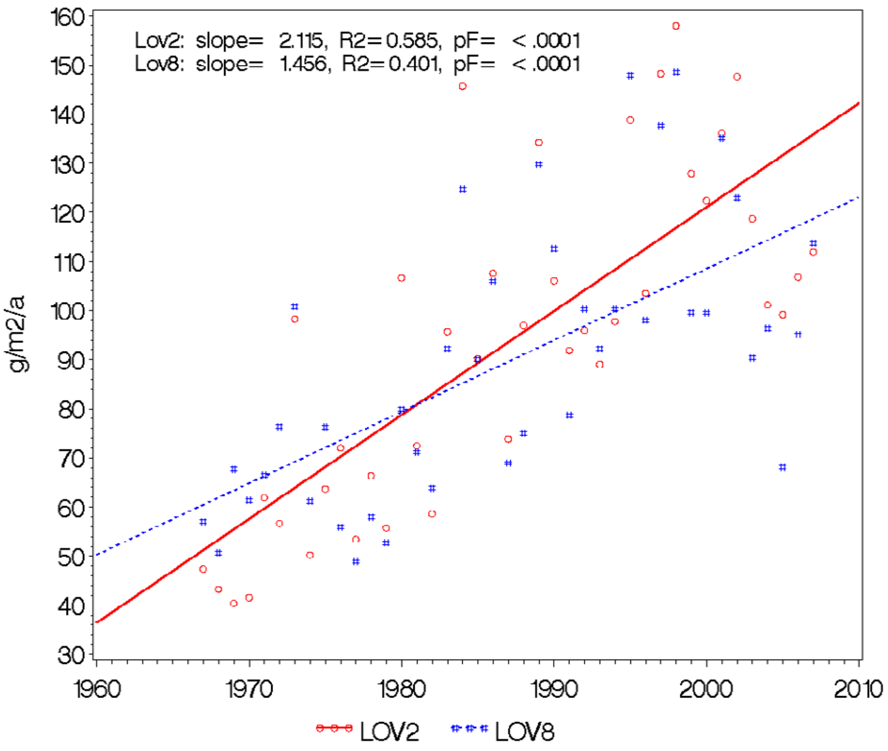

Fig. 3. The regression lines of annual primary production at the Loviisa 2 (discharge area) and Loviisa 8 (intake area) stations between 1967 and 2006.

The eutrophication of littoral vegetation in the discharge area has been the most obvious, unambiguous and significant biological effect of the heated water. Spiked water milfoil *Myriophyllum spicatum,* the pondweeds *Potamogeton perfoliatus* and *Potamogeton pectinatus* and the vigorous growths of numerous filamentous algae as their epiphytes have strongly increased in the vicinity of the cooling water outlet, where they have formed dense populations in the littoral zone in late summer (Fig. 4). However, the strongest increase of phytobenthos has extended only to a distance of about 1 km from the outlet, i.e., the changes in vegetation have been largest in those areas that remain ice-free in winter. A weaker eutrophication of littoral vegetation appeared in the whole area of Hästholmsfjärden Bay, but outside this area the phenomenon was slighter and observed only locally.

Fig. 4. Dense vegetation on the south coast of Hästholmsfjärden at Loviisa. (Photo by Anna Weckman).

At Olkiluoto, the studies focusing on the effects of warm water discharge were more concise. Similar trends to those noticed in the Loviisa area regarding to increasing eutrophication were also discernible at Olkiluoto, but to a clearly smaller extent; this was due to the clearly weaker level of background eutrophy and nutrient concentrations in the Bothnian Sea, and to the local hydrographical and biological factors prevailing in the Olkiluoto area. The level of primary production has also increased at Olkiluoto, but has remained at a clearly lower level than at Loviisa. In spite of the analogous changes observed in the macrozoobenthos, the benthic fauna has remained strong and diversified in the Olkiluoto area.

Environmental radioactivity

Radioactive discharges into the sea from the Finnish NPPs have been on average below 10% of the statutory limits. The discharged amounts of tritium were the most abundant, but those of other discharge nuclides were only a few percent of the limits, and still significantly decreased during recent years (during the last ten years to below 0.5%). Small amounts of local discharge nuclides were regularly detected in environmental samples taken from the discharge areas: tritium in seawater samples, and activation products, such as ^{60}Co, ^{58}Co, ^{54}Mn, ^{110m}Ag, ^{51}Cr, among others, in suspended particulate matter, bottom sediments and in several indicator organisms (e.g., periphyton and the bladder-wrack *Fucus vesiculosus*) that effectively accumulate radioactive substances from the medium.

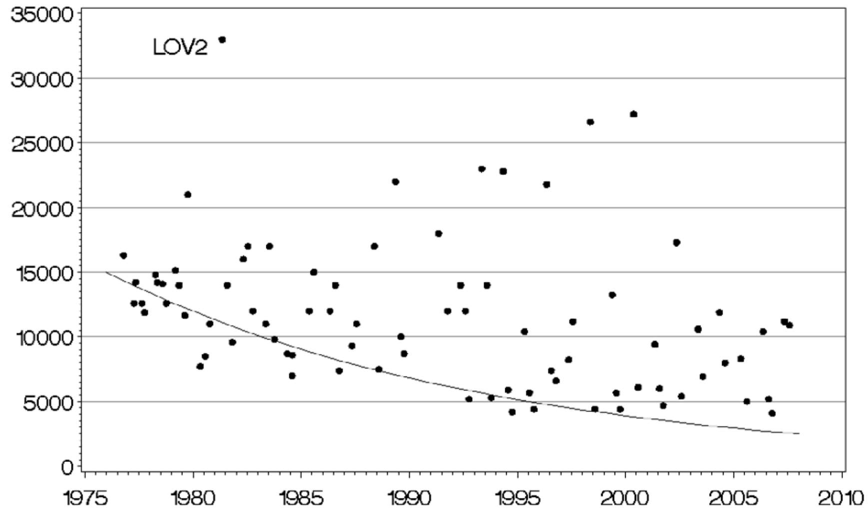

Fig. 5. Tritium concentrations in surface seawater (Bq m^{-3}) in Hästholmsfjärden (discharge area) at Loviisa in 1976–2007. The curve in the graph indicates the decay of weapons-tests tritium in Finnish coastal waters during the monitoring period.

The tritium discharges and the consequent detection frequency and concentrations of tritium in seawater were higher at Loviisa (Fig. 5), but the concentrations of the activation products were higher at Olkiluoto, where traces of local discharge nuclides were also observed over a clearly wider area, due to the better exchange of water than at Loviisa, where local discharge nuclides were detected outside the Hästholmsfjärden Bay only quite rarely and in small amounts. At the farthest, an insignificant trace amount (0.2 Bq kg^{-1} d.w.) of ^{60}Co originating from Olkiluoto was detected in *Fucus* at a distance of 137 km from the power plant (Fig. 6).

Discharge nuclides from the local nuclear power plants were almost exclusively detected at the lower trophic levels of the ecosystems (Fig. 7). Traces of local discharge nuclides were very seldom detected in fish, and even then only in very low quantities, but not at all in birds nor in the inner organs and reproductive products of fish and birds.

The best indicators for ^{60}Co were periphyton, the spiked water milfoil *Myriophyllum spicatum* and the bladder-wrack *Fucus vesiculosus*, whereas the intake of, e.g., Chernobyl-originated ^{137}Cs was highest in predatory fish, perch and pike. As a consequence of the reduced discharges, the concentrations of local discharge nuclides in the environment have decreased noticeably in recent years at both Loviisa and Olkiluoto (Fig. 8). Radioactive substances that originated from the Chernobyl accident and weapons-tests fallout (e.g. ^{137}Cs, ^{90}Sr, ^{239,240}Pu) were still being detected in the environmental samples; the concentrations of ^{137}Cs and natural radionuclides (e.g., ^{40}K, ^{7}Be) were in general higher than those of the local discharge nuclides.

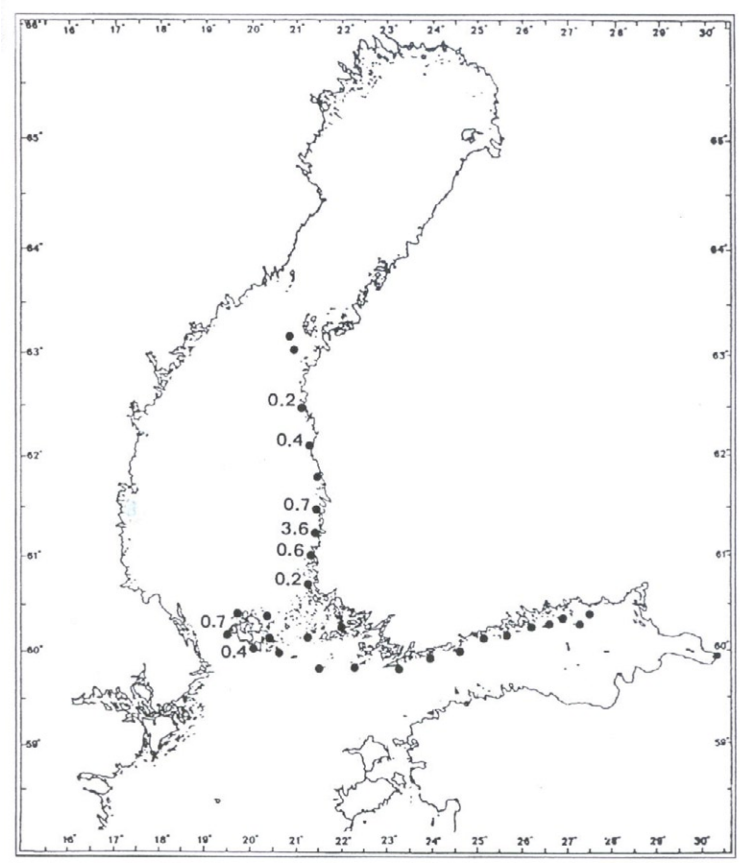

Fig. 6. Activity concentrations of ^{60}Co (Bq kg^{-1} d.w.) in *Fucus vesiculosus* collected from 29 sampling sites along the Finnish coast in 1995. A dot without a number means that the concentration was below the detection limit of 0.1 Bq kg^{-1} d.w.

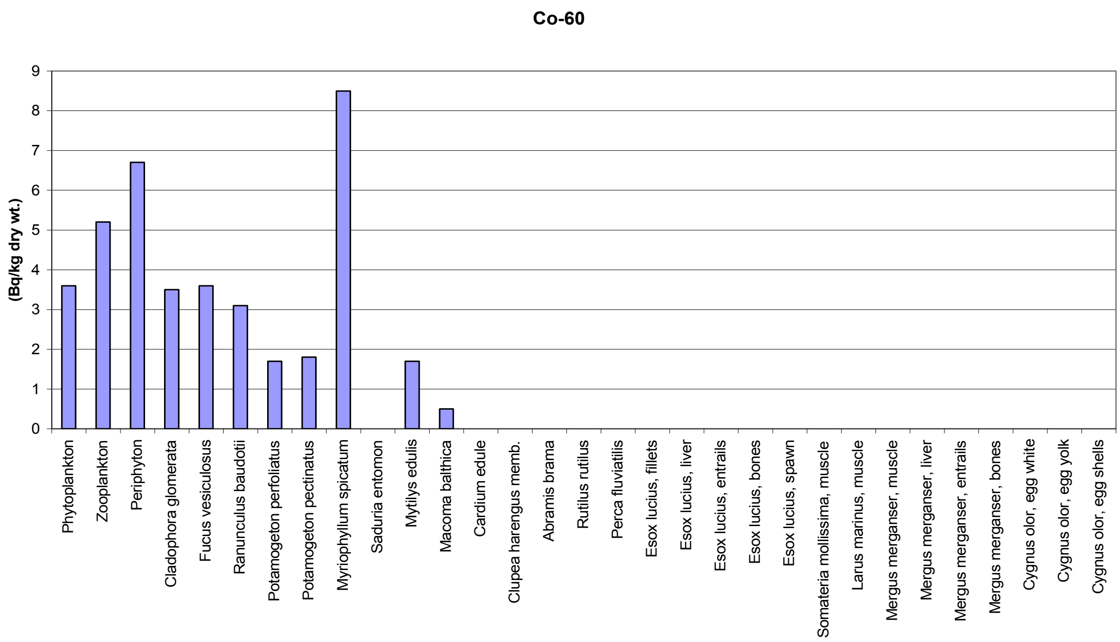

Fig. 7. ^{60}Co mean values (Bq kg^{-1} d.w.) in some indicator samples in the sea area off Olkiluoto in 2001.

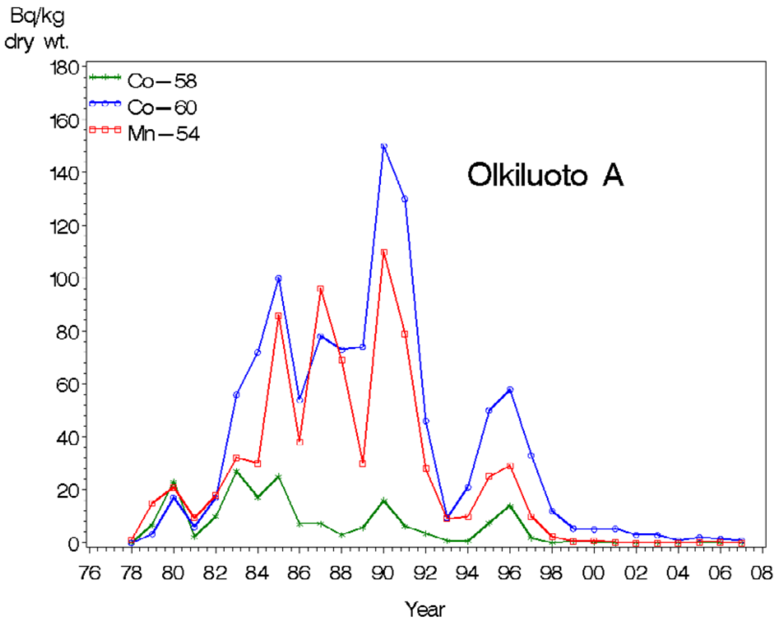

Fig. 8. Annual mean concentrations of ^{58}Co, ^{60}Co and ^{54}Mn in *Fucus vesiculosus* (Bq kg^{-1} d.w.) at a sampling site nearest to the cooling water outlet at Olkiluoto in 1987–2007.

The radiation doses to the public caused by discharges of radioactive substances from the Finnish nuclear power plants were small. The dose limit set for members of the public from the normal operation of Finnish nuclear power plants is 0.1 mSv a^{-1}. This is approximately 1/40 of the average radiation dose received by Finns from different sources during one year. During the whole operational history of the power plants, the effective dose commitments of the critical groups have been at their highest less than 4%, and during more recent years clearly below 1% of the set limit. In general, the minor doses of local origin to the critical groups have been due to liquid discharges of ^{60}Co and people's shore occupancy. The environmental risk caused by the ionizing contaminants discharged from the Loviisa and Olkiluoto power plants was negligible: the doses to organisms were far below the conservative screening level of 10 µGy h^{-1}.

Conclusions

Although the concentrations in environmental samples, and above all, the discharge data, are presented as seemingly large numbers, the radiation doses caused by them to the population and to biota are very low, practically insignificant. The effects of the thermal discharges have been more significant, at least to the wildlife in the discharge areas of the cooling water, although the area of impact has been relatively small. The results show that the nutrient level and the exchange of water in the discharge area of a nuclear power plant are of crucial importance.

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Tritium level along Romanian Danube river sector

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Abstract

Danube River is the eleventh longest river in the world, and is one of the important rivers from Europe. Tritium activity concentration from 12 locations distributed along Romanian Lower Danube River was determined by means of low level liquid scintillation counting. The three sampling campaigns were conducted between 2006 – 2007, on 975 km. The average concentration value registered for tritium activity for the studied area was around 15.4 TU, with higher values for Bechet and Seimeni locations. Special attention was accorded to Turnu Severin location, where tritium activity concentration from monthly Danube water was compared with monthly tritium concentration in precipitation. The behaviour of tritium concentration in the Danube water, at the begging of studied area, doesn't follow the behaviour of tritium concentration in precipitation, but the established values are far from any concern from radiological point of view.

Introduction

The Danube River Basin is the second largest river basin in Europe covering 801463 km^2 [ICPDR, 2005]. From its source in the Black Forest at height of 1000 m above sea level, to its three-armed delta on the Black Sea, the Danube flows through eights countries and drains a total of eighteen. Due to its geologic and geographic conditions the Danube River Basin can be divided into 3 main parts: Upper Danube Basin, Middle Danube Basin and Lower Danube Basin. 65% of Lower Danube Basin is the natural border between Romania and Bulgaria.

During the 80's the contribution of nuclear power to total primary energy supply as a percentage of fuel mix has increased significantly, from 9% in 1985 to 15% in 1995 [EC, 2000]. Nuclear power generates between 10% and 45% of electricity in the Danube Space countries. Nuclear electricity has been, on a variable cost basis, cheaper than other options, and consequently used in preference to coal, oil and gas. Two important nuclear objectives for both countries are situated in this region: Kozloduy NPP and Cernavoda NPP. Nuclear energy is considered by the two countries an option for the future development, so Romania already built a new CANDU type unit in Cernavoda, which has been operating since September 2007, and Bulgaria started negotiations for a new nuclear power plant in Belene.

The knowledge of tritium in the Romanian Lower Danube offers two important information: first in the environmental radioactivity monitoring program (Cernavoda is a CANDU type reactor, where tritium is the most important radioisotope evacuated in the environment) and second a primary estimation of river discharge signature for the end part of Danube by comparing the tritium level from precipitation and river water.

Material and methods

The Danube typology represents a harmonized system developed by the countries sharing the Danube River [ICPDR, 2005]. The most important factors used for this system are mean water slope, substratum composition, geomorphology and water temperature. Lower Danube Basin contains four ~~types of~~ sections: Iron Gates Danube, Western Pontic Danube, Eastern Wallachian Danube and Danube Delta.

Iron Gates Danube section (A, fig. 1) has an average width of about 750 m and runs in a canyon or through valley form. Dominant main channel has numerous rocks situated directly under the surface water. Sampling point chosen was Ieselnita (1, fig.1), location with easy access to the left side of the Danube. Next sampling point was from Cerna, Danube tributary, at Toplet location (2, fig. 1). Cerna is a small mountain river without industrial activity. The end of this section was Turnu Severin (3, fig.1) another sampling point.

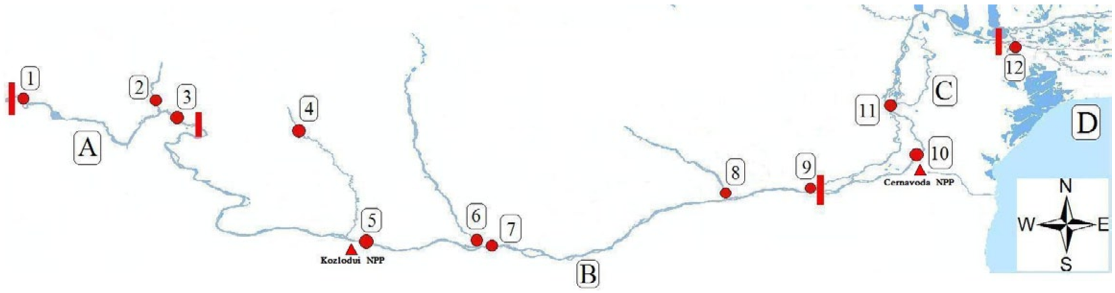

Fig. 1 Section type and sampling locations from Lower Danube Basin; A - Iron Gates Danube; B - Western Pontic Danube; C – Eastern Wallachian Danube; D – Danube Delta.

Western Pontic Danube section (B, fig.1) contain floodplain landscape with higher plains of terraced accumulation in a meander and plain floodplain valley. The Romanian bank is low and terraced with wide floodplains. The main channel has an average width of 830 m and mean depth of 8.5 m. The diversity of water bodies in this area close to the stream is large, but we focused on the main course and on few tributaries with important mining industry, Jiu river (4, fig.1), and chemical industry Olt river (6, fig. 1) and Arges river (8, fig.1). Bechet location (5, fig.1) from main course is important due to its location downstream ~~the~~ Kozloduy NPP. Another sampling point was Turnu Magurele (7, fig.1) which is located downstream to jonction to Olt river, approximately in the middle of this section. The end of this section is Chiciu-Silistra (9, fig.1) another important sampling point, before Danube splits in two branches: Old Danube and Borcea.

The Danube changes its watercourse northward in Eastern Wallachian Danube section (C, figure 1). There are two large isles, Balta Ialomitei and Balta Brailei, and many natural lakes. The main channel has an average width of 650 m, and a mean depth of 10.5m. The section is characterized by slow current velocities of 0.8 m/s. The first sampling point of this section was established in Seimeni (10, fig.1) downstream of

NPP Cernavoda discharge channel and the second in Giurgeni (11, fig.1), a point between the two isles were Danube has one main branch for a short distance.

The Danube Delta section (D, figure 1) is the Danube's youngest teritory. There are three main water channels: Chilia, Sulina and Sf. Gheorghe, with numerous canals and floating islands. This section is characterized by an average current velocity of 0.7 m/s. The shape of the delta is triangular, and at mean water levels, 60% of this area is covered by waters. The point from this section was only Tulcea (12, fig.1), due to difficult access to this part of the Danube.

Precipitation sampling was made on a monthly basis in a typical rain water collector [IAEA, 1989] in Drobeta Turnu Severin location and our Institute, starting with January 2007, and ending with October 2007. At the end of the month the container was shaken thoroughly and a sample of 1 liter was filled for shipment to the analytical laboratory. Special attention was given for collection and preservation of water sample [APHA-AWWA-WEF, 2005]. Type of section, water body, and sampling location are in the table 1.Water samples for tritium measurement were collected one meter from the left bank of Danube, 10 cm from the water surface, in glass bottle. pH, conductivity and temperature for water sample were measured in sampling location using portable apparatus WTV pH/cond 340i.

Table 1. Geographical coordinates of sampling locations.

Location number	Location name	Section type	Water body	Latitude	Longitude
1.	Ieselnita	A	Danube	44^{0}48'31''	21^{0}24'11''
2.	Toplet	A	Cerna	44^{0}47'58''	22^{0}23'26''
3.	Drobeta Turnu Severin	A	Danube	44^{0}40'2''	22^{0}32'51''
4.	Filiasi	B	Jiu	44^{0}34'8''	23^{0}27'19''
5.	Bechet	B	Danube	44^{0}44'56''	23^{0}57'14''
6.	Islaz	B	Olt	43^{0}44'38''	24^{0}46'46''
7.	Turnu Magurele	B	Danube	43^{0}42'45''	24^{0}53'25''
8.	Oltenita	B	Arges	44^{0}05'51''	26^{0}38'12''
9.	Chiciu/Silistra	B	Danube	44^{0}07'56''	27^{0}16'19''
10.	Seimeni	C	Danube	44^{0}23'31''	28^{0}03'28''
11.	Giurgeni	C	Danube	44^{0}45'2''	27^{0}52'17''
12.	Tulcea	D	Danube	45^{0}10'58''	28^{0}48'25''

As tritium is a soft beta emitter (5.72 keV mean energy), liquid scintillation is the most appropriate technique for its measurement. In this work, the low-background liquid scintillation spectrometer Quantulus 1220 (Wallac) has been used to determine tritium in water samples. The analytical method used to determine tritium in water samples was, briefly, the following: samples were filtered through slow depth filters; 250 ml of filtrate was distilled using ISO method [ISO, 1989]; 8 ml of distillate was mixed with 12 ml of scintillation cocktail OptiPhase Hisafe 3 in polyethylene vials; three background samples and tritium standards were simultaneously prepared. Samples, backgrounds and tritium standards were counted using Quantulus 1220 during

1000 min/samples. The tritium standards were internal standard capsules containing a tritium-labeled organic compound [fructose-1-^{3}H] provided by PerkinElmer. Tritium-free water (blank water) was deuterium-depleted water with a D/(D+H) value of 15 ppm (Varlam et al. 2009). The counting efficiency at the best factor of merit was between 24.84% and 25.02%, and the background between 0.714 ± 0.016 CPM (counts per minute) and 0.748 ± 0.017 CPM, following a minimum detectable activity of around 4 TU (confidence level of 2). The uncertainty due to the statistical nature of radioactive decay and background radiation was reported at 1σ.

Results and Discussion

The average fallout during the year 2007 in Romania was ~~with~~ 13.33% higher than the normal climatologic average (NRMA, 2007): for months of April, May, June and July it has been recorded a deficit in precipitation monthly average with an important negative deviation for April (75%), whereas for the month of October there was a significant excess in monthly average (124.9%).

Tritium concentration during the year 2007 at Turnu Severin location had a mean of 10.2 ± 2.1 TU (one Tritium Unit, TU, one tritium atom corresponds to 10^{18} hydrogen atom). The same value was recorded in our Institute, 10.7 ± 2.1 TU. We recorded a minimum tritium concentration of 4.6 ± 2.1 TU in January 2007 for Turnu Severin location (table 2) and a minimum tritium concentration of 6.1 ± 2.1 TU in February 2007 for our Institute. The maximum tritium concentration of 16.1 ± 2.1 TU was measured in July for Turnu Severin location, but we recorded for our Institute two months with the higher values of 17.2 ± 2.2 TU and 16.8 ± 2.2, May and August. Comparing published values for tritium in precipitation in Austria [Kralic et al., 2005] with measured values for the monitored locations we can conclude that annual tritium concentration average has had the same trend for the past years: 10.4 TU for 2000, 10.5 TU for 2001 and 10.46 TU for 2002. There are no other influences and tritium behaviour in precipitation has the same tendency to decrease in the environment.

Table 2. Tritium concentration during observation period of 2007 in Turnu Severin location and our Institute.

Period of sampling	Tritium concentration [TU] Turnu Severin	Tritium concentration [TU] Our Institute
January 2007	4.6+/-2.1	7+/-2.1
February 2007	7.3+/-2.1	6.1+/-2.1
March 2007	4.8+/-2.1	8.1+/-2.1
April 2007	8.9+/-2.1	12.7+/-2.2
May 2007	15.3+/-2.2	17.2+/-2.2
June 2007	12.7+/-2.2	8.8+/-2.1
July 2007	16.8+/-2.2	14.7+/-2.2
August 2007	15.9+/-2.2	16.8+/-2.2
September 2007	8.8+/-2.1	9.5+/-2.1
October 2007	7.2+/-2.1	6.8+/-2.1

A special behaviour regarding tritium concentration was recorded in the Danube water at Turnu Severin location, table 3. The average tritium concentration in the Danube water for this location was 13.7 ± 2.2 TU, higher than that of precipitation, without seasonal variation normally expected for surface water. The measured average for tritium concentration is far from any radioprotection concern (aprox. 1.6 Bq/l comparing with 100 Bq/l recommended by Council Directive 98/83/EC for drinking water), but is important for Danube River discharge signature at the beginning of Lower Danube Basin.

Table 3. Tritium concentration in Danube water and precipitation at Turnu Severin location.

Period of sampling	Tritium concentration [TU] Danube	Tritium concentration [TU] Precipitation
January 2007	11.5 +/- 2.2	4.6 +/- 2.1
February 2007	17 +/- 2.2	7.3 +/- 2.1
March 2007	13.7 +/- 2.2	4.8 +/- 2.1
April 2007	15.3 +/- 2.2	8.9 +/- 2.1
May 2007	14.8 +/- 2.2	15.3 +/- 2.2
June 2007	11.2 +/- 2.2	12.7 +/- 2.2
July 2007	16.3 +/- 2.2	16.8 +/- 2.2
August 2007	14.9 +/- 2.2	15.9 +/- 2.2
September 2007	10.6 +/- 2.2	8.8 +/- 2.1
October 2007	11.4 +/- 2.2	7.2 +/- 2.1

Tritium sampling campaigns were performed in August 2006, March 2007 and October 2007, table 4.

Table 4. Tritium concentration along Romanian Danube Sector for sampling campaign of August 2006, March 2007, and October 2007.

Location number	Location name	Water body	Tritium concentration August 2006 [TU]	Tritium concentration March 2007 [TU]	Tritium concentration October 2007 [TU]	Mean tritium concentration [TU]
1.	Ieselnita	Danube	17.6 +/- 2.2	7 +/- 2.1	15.7 +/- 2.2	13.4 +/- 2.2
2.	Toplet	Cerna	11.9 +/- 2.2	8.3 +/- 2.1	10.7 +/- 2.2	10.3 +/- 2,2
3.	Drobeta Turnu Severin	Danube	14.8 +/- 2.2	6.2 +/- 2.1	16.7 +/- 2.2	12.6 +/- 2,2
4.	Filiasi	Jiu	12.7+/- 2.2	9.3 +/- 2.1	10.5 +/- 2.2	10.8 +/- 2.2
5.	Bechet	Danube	27.9 +/- 2.3	29.2 +/- 2.3	24.7 +/- 2.3	27.3 +/- 2.3
6.	Islaz	Olt	11.6 +/- 2.2	6.8 +/- 2.1	13.2 +/- 2.2	10.5 +/- 2.2
7.	Turnu Magurele	Danube	12.8 +/- 2.2	8.9 +/-2.1	13.2 +/- 2.2	11.6 +/- 2.2
8.	Oltenita	Arges	12.2 +/- 2.2	11.6 +/- 2.2	11.8 +/- 2.2	11.9 +/- 2.2
9.	Chiciu/Silistra	Danube	9.7+/- 2.1	12.2 +/- 2.2	8.3 +/- 2.1	10.1 +/- 2.2
10.	Seimeni	Danube	32.4 +/- 2.3	33.5 +/- 2.3	28.7 +/- 2.3	31.5 +/- 2.3
11.	Giurgeni	Danube	15.9 +/- 2.2	7.4 +/- 2.1	7.2 +/- 2.1	10.2 +/- 2.2
12.	Tulcea	Danube	11.6 +/- 2.2	10.8 +/- 2.1	16.5 +/- 2.2	13 +/- 2.2

Two higher values than the tritium concentration average of 16.2 ± 2.2 TU were recorded for Bechet and Seimeni, 27.3 ± 2.2 TU and 31.5 ± 2.2 TU respectively. The two locations have nearby two different nuclear power plants, Kozloduy and Cernavoda, with important tritiated effluents discharged in the Danube. Tributaries had lower mean tritium concentration (between 10.3 ± 2.2 TU for Cerna River and 11.9 ± 2.2 TU for Arges River) than the tritium concentration average of the Danube. Tritium level in the Danube water is continuously decreasing from 20-25 TU in 1995 [Rank et al., 1999] to precipitation level, even if ~~the~~ nuclear objectives are in the monitored areas.

Conclusions

The tritium level in precipitation during the monitoring period was 10.2 ± 2.1 TU for Turnu Severin location. Tritium concentration in the Danube water in the same location had higher values than that of precipitation, the mean value calculated during the monitoring period being 13.7 ± 2.2 TU. This value is far from any concern from radiological point of view, but the behavior of tritium concentration during the cold months of the year (January, February) with values higher than 10 TU proves the influence of nuclear activity developed along the Danube. Tritium concentration average in the Danube water along the Lower Danube Basin was 16.2 ± 2.2 TU. Two higher values than the tritium concentration average were recorded for Bechet and Seimeni. The two locations have nearby important nuclear power plants. The highest mean tritium concentration measured during the three sampling campaigns for the Danube water was recorded in August 2006 with a value of 16.6 ± 2.2 TU. The lowest mean tritium concentration measured during the three sampling campaigns for the Danube water was recorded in March 2007 with a value of 13.8 ± 2.2 TU. Tributaries had lower values than tritium concentration average due to their different basins with strong groundwater components. Tritium level in the Danube water is continuously decreasing, from 596 TU in September 1966, to 12.6 TU in December 2005 (GNIR, Vienne Station). Environmental values recorded confirmed that precipitation is the primary source of tritium in the Danube water, in spite of the existence of the two important nuclear power plants in the monitored area, as previously mentioned.

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Radioactivity monitoring of sediments in rivers in Serbia during the period 2005–2009

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Abstract

In this paper we present radioactivity monitoring results of the sediments in Serbian rivers from 2005 through 2009. Sediment samples have been collected at eight sites on six Serbian rivers, usually during spring and autumn.

We have applied gamma spectrometry, gross alpha beta activity and radiochemical method of 90Sr separation as measurement methods.

Our results show that activity concentration of natural radionuclide in sediments (mean annual values in Bq/kg) is within the average values limits for the territory of Serbia. Activities of long living radionuclide of artificial origin have tendency of mild decrease as a result of dissolution, leaching and changes in river flow.

Introduction

Sediment is an essential dynamic component of aquatic systems due to the strong tendency towards bonding the tank of toxic and persistent compounds of anthropogenic origin. Sediment is composed of organic and mineral components. Organic component of the plant and animal material from the river and the watershed area and discharge of industrial waste water utilities. Mineral component contains erosive material from the surface of the entire river basin.

Sediment monitoring and assessment of its quality is usually conducted to determine the extent to which sediment tank and a secondary source contaminenata in surface waters. The aim of this monitoring was conducted in determining the status of the quality of sediment and its impact on the environment and human health through the study of various interactions in the sediment-water system.

Material and methods

Radioactivity control of sediment in the territory of Serbia is done in the following rivers:

– Danube river (sampling is done on Bezdan near the border with Hungary, in Belgrade and in Prahovo near Romania border)
– Sava (sampling is done at the Belgrade)
– Tisa (sampling is done at Kanjiza)

– Nisava (sampling is done near Pirot)
– Timok (sampling is done at Zaječar)
– Drina (sampling is done at Loznica)

Sediments of the Danube and Sava rivers were taken four times durnig the year, while sediments from other rivers were taken in the spring when water level is the highest and in the fall when water level is the lowest.

One by one kilogram of river sediment was taken by sampler and placed in to plastic containers. Sediment was dried at 105^0C to constant weight, sieved trough sieve and the fraction less than 250 µm is taken.

Gammaspectrometry is carried out on pure germanium detector manufactured by EG&G ″ORTEC″, which is connected with multichannel analyzer (8192 channels) produced by the same manufacturer and with adequate computer facilities. Energetic calibration, as well as calibration of detector efficiency is performed by means of Amersham radioactive standard (Debertin K. et al.1988, Pantelic G. et al. 1996)

The measurement of gross beta activity is carried out by alpha-beta proportional gas counter PIC-WPC-9550. The level of basic radiation is 0.4 imp/min. The size of planchet is 5cm. The performance of counter is 47% and is determined by ^{90}Sr standard.

Radiochemical method of ^{90}Sr separation is based on oxalate isolation of Ca and Sr, ignition to oxides and usage of aluminum as ^{90}Y carrier. The equilibrium is achieved in 18 days, and after that time ^{90}Y is isolated on Al $(OH)_3$ carrier (Brnovic 1972), which is then ignited to oxide that is subsequently measured by alpha-beta proportional gas counter

Results and discussion

Table 1 shows gross beta activity results with standard deviation. The scope of a monitoring program included gross beta activity measurements of river Danube at sites Bezdan and Belgrade as well as river Sava near Belgrade.

During the observation period 2005-2009 there was no significant variations of sediment measurement results of river Danube on both locations.

During the observation period 2005-2009 there was no significant variations of sediment measurement results of river Danube and river Sava obtained on a location near Belgrade.

Table 1. Gross alpha beta activity in sediments Danube and Sava river (mean annual values).

Year		2005	2006	2007	2008	2009
		Activity (Bq/kg)				
	Bezdan	708 ± 65	470 ± 140	658 ± 170	473 ± 92	573 ± 70
Danube	Belgrade	520 ± 140	508 ± 100	520 ± 180	503 ± 86	460 ± 220
Sava	Belgrade	480 ± 180	580 ± 140	590 ± 220	530 ± 120	487 ± 95

Table 2 illustrate the results of activity concentration of natural radionuclide in river sediments in Danube and Sava (mean annual values in Bq/kg with standard deviation).

Table 2. Activity concentration of natural radionuclide in river sediments Danube (Zemun) and Sava (Belgrade).

Year	River	^{40}K (Bq/kg)	^{232}Th (Bq/kg)	^{226}Ra (Bq/kg)	^{238}U (Bq/kg)	^{235}U (Bq/kg)	^{7}Be (Bq/kg)
2005	Danube	480 ± 22	34.5 ± 0.4	40.4 ± 2.2	50.0 ± 8.4	2.3 ± 0.3	< 7.2
2006		464 ± 14	30.8 ± 5.2	40 ± 13	41 ± 21	1.8 ± 0.8	< 8.7
2007		411 ± 84	26.4 ± 8.7	30.3 ± 6.3	28 ± 13	1.5 ± 0.6	< 7.2
2008		589 ± 60	36.5 ± 5.0	51.2 ± 4.5	56.5 ± 5.2	2.7 ± 0.6	< 11
2009		480 ± 160	35 ± 16	34 ± 15	43 ± 15	1.8 ± 0.5	37 ± 16
2005	Sava	465 ±31	32.3 ± 2.6	35.9 ± 5.8	39.5 ± 4.9	1.7 ± 0.3	< 7.0
2006		477 ± 46	28.7 ± 6.5	40 ± 14	35.3 ± 7.6	1.6 ± 0.4	< 8.9
2007		400 ± 120	25 ± 11	26 ± 11	19.8 ± 1.2	0.9 ± 0.1	< 11
2008		600 ± 100	39.8 ± 4.8	51.4 ± 5.6	48.1 ± 1.9	2.2 ± 0.1	< 8.2
2009		460 ± 120	14.9 ± 5.0	37± 12	40 ± 17	1.9 ± 0.8	< 20

The average natural radionuclide concentration with standard deviation for sediments in other Serbian rivers (Nisava, Timok, Tisa and Drina) in table 3 is shown.

Activities of natural radionuclides measured in all sediment samples are within the average values for the territory of Serbia. Activity concentration of cosmogenic radionuclide ^{7}Be is below the limit of detection.

Table 3. Activity concentration of natural radionuclide in river sediments Danube (Zemun) and Sava (Belgrade).

Year	River	^{40}K (Bq/kg)	^{232}Th (Bq/kg)	^{226}Ra (Bq/kg)	^{238}U (Bq/kg)	^{235}U (Bq/kg)	^{7}Be (Bq/kg)
2005	Timok	320 ± 11	21.1 ± 1.2	23.2 ± 4.0	25.2 ± 5.2	1.1 ± 0.2	24 ± 4
2006		397 ± 79	24 ± 11	25 ± 11	26 ± 12	1.2 ± 0.5	< 25
2007		450 ± 140	22 ± 13	27 ± 14	45 ± 10	1.7 ± 0.3	< 6.1
2008		471 ± 56	23.7 ± 5.7	32.2 ± 3.9	32.1 ± 0.9	1.4 ± 0.3	< 13
2009		384 ± 13	17.4 ± 2.7	23.3 ± 1.8	22.8 ± 4.8	1.1 ± 0.1	16 ± 9
2005	Nisava	390 ± 12	23.3 ± 0.9	22.7 ± 2.6	25.1 ± 2.6	1.0 ± 0.1	8 ± 2
2006		362 ± 42	29.1 ± 1.8	35.1 ± 3.5	30 ± 10	1.5 ± 0.3	30 ± 6
2007		364 ± 87	20.1 ± 0.4	20.8 ± 2.0	13.8 ± 3.1	< 1.2	< 3.9
2008		423 ± 7	27.9 ± 6.4	31.9 ± 8.6	29.3 ± 8.1	1.2 ± 0.2	< 7.2
2009		482 ± 91	28.1 ± 8.9	32.3 ± 13.1	45 ± 23	2.0 ± 1.2	< 5.5
2005	Tisa	492 ± 16	35.6 ± 1.9	36.4 ± 5.5	54.1 ± 5.2	2.4 ± 0.3	< 12
2006		390 ± 2	19.9 ± 0.4	24.2 ± 1.4	20.9 ± 5.6	1.1 ± 0.1	< 12
2007		557 ± 24	39.5 ± 0.4	36.9 ± 1.6	30.4 ± 6.6	< 2.0	< 9.8
2008		555 ± 52	32.8 ± 1.3	36.2 ± 3.2	41.1 ± 1.4	1.9 ± 0.1	8 ± 2
2009		443 ± 16	34.5 ± 7.9	39 ± 10	42 ± 14	2.1 ± 0.8	< 6.1
2006	Drina	406 ± 19	28.0 ± 1.8	27.5 ± 2.9	28.8 ± 4.7	1.4 ± 0.2	24 ± 16
2007		340 ± 170	22 ± 15	26 ± 23	21 ± 12	< 1.8	9 ± 5
2008		450 ± 200	28 ± 16	31 ± 14	31 ± 13	1.5 ± 0.6	6 ± 1
2009		288 ± 15	18.4 ± 5.4	19.9 ± 3.7	23 ± 12	1.0 ± 0.2	< 8.9

The most important source of artificial radionuclides in the environment in Serbia is nuclear accident in Chernobyl Nuclear Power Plant. Results of gamma spectrometric measurement ^{137}Cs in sediments of Danube river in tree sites Bezdan, Belgrade and Prahovo shows descreasing during the period 2005-2009 (figure 1). Maximum activity 137Cs was mesured in 2009 in site Prahovo (figure 1).

The sediment was sampled in high water level season. At the actual sampling time, the river flooded the area outside banks, so the sampling of nearby soil has been performed as well.

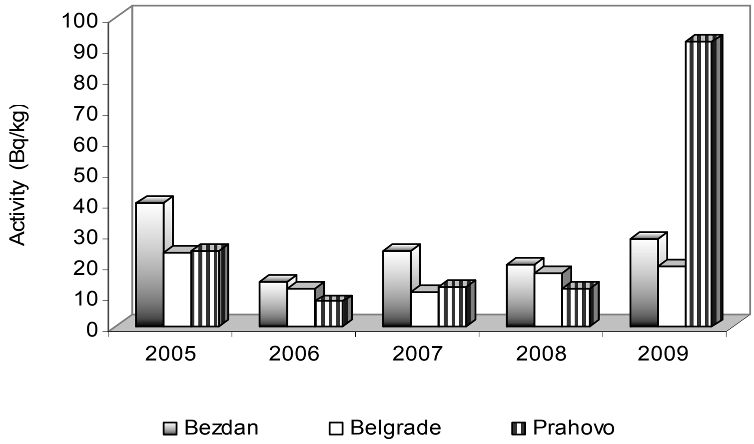

Fig. 1. ^{137}Cs activity concentration in sediment of Danube River (mean annual values).

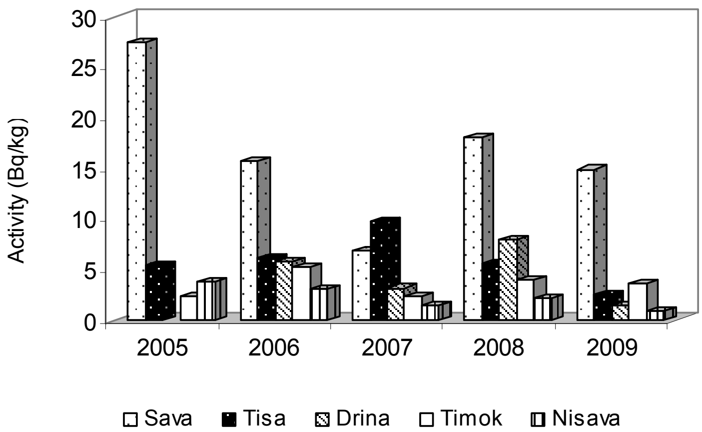

Fig. 2. ^{137}Cs activity concentration in sediments Sava, Drina, Tisa, Timok and Nisava (mean annual values).

Maximum activity of ^{137}Cs was in 2005. in Sava river sediment (figure 2).

Figure 3 shows 90Sr activity in Danube sediment during the period 2005-2009 at sites Bezdan and Belgrade. Average annual values in 2005 was 0.41Bq/kg at site Belgrade. Average annual values in 2007 was 0.12Bq/kg at site Bezdan. In 2009 90Sr activity in Danube sediment at site Prahovo was not measured.

During the period 2005-2009 90Sr activity was less then 1Bq/kg on all rivers at all sites and tends to decrease, as figure 4 illustrates.

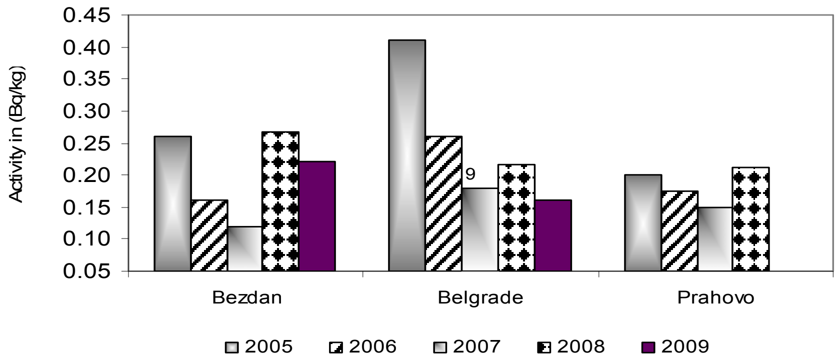

Fig. 3. ^{90}Sr activity concentration in sediments Danube river (mean annual values).

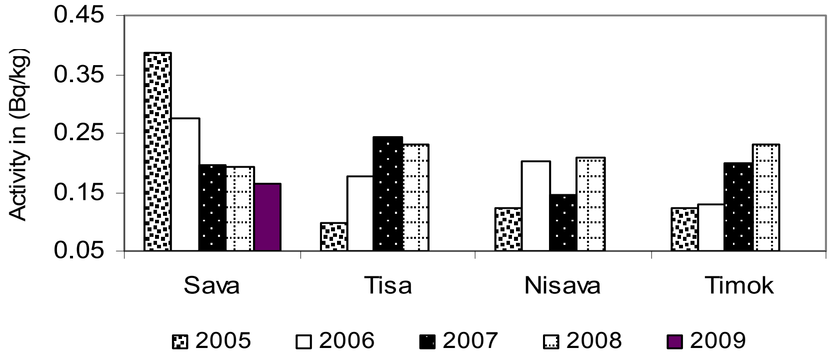

Fig.4. ^{90}Sr activity concentration in sediments Danube river (mean annual values).

Conclusions

By means of gamma spectrometry measurements of natural and artificial radionuclide as well as gross beta and ^{90}Sr activity in sediments of rivers in Serbia it is possible to assess the sediment quality and its impact on the environment. The results of the measurements conducted during the period 2005-2009 are within normal limits. Artificial radionuclide activity tends to decrease as a result of dissolution, leaching and river flow changes.

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Radiation and tritium activity in the environment of the National Park Plitvice Lakes

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Abstract

The disturbance of natural distribution of tritium (^{3}H) and radiocarbon (^{14}C) caused by thermonuclear weapon tests in the sixties of the last century made these isotopes very important tracers in environment. Their concentration can be also locally affected by fossil fuel combustion (decrease of ^{14}C) or by various nuclear facilities (increase of ^{14}C and ^{3}H). In our comprehensive study of the environment in the Plitvice Lakes National Park we measured ^{14}C in monthly samples of atmospheric CO_2 and ^{3}H in monthly precipitation in period 2003 – 2006. ^{14}C activity was also measured in the beech leaves and needles of spruce and abies collected as 1-year composite samples in the woods of the Plitvice Lakes area in 2005 and 2006. ^{14}C activity of monthly CO_2 samples vary between 101.5 and 109.6 percent of modern carbon (pMC) with slightly lower ^{14}C activities in winter months due to the influence of CO_2 from fossil fuel combustion. The values are similar to those measured in the city of Zagreb, which is a densely populated industrial center. The ^{14}C activities of tree leaves and needles are slightly higher than mean yearly atmospheric ^{14}C activities: the highest values have needles of abies (111 pMC, 110 pMC) and spruce (110 pMC, 109 pMC), followed by beech leaves (106 pMC, 105 pMC) and atmospheric CO_2 (104 pMC, 105 pMC). Higher ^{14}C activities of coniferous wood indicate that the collected needles represent an average of several years period while the ^{14}C of leaves of deciduous wood represent the mean ^{14}C from the growing period (spring-summer). ^{3}H activities of monthly precipitation showed seasonal fluctuations with maximum in summer (13 – 18 TU) and minimum in winter (0 – 5 TU), and are in good correlation with ^{3}H in the Zagreb precipitation. In conclusion, ^{14}C and ^{3}H in the atmosphere of the Plitvice Lakes area reflect the global trend of these isotopes in the atmosphere, and are at the same level as in highly populated area of Zagreb.

^{137}Cs concentrations in Saimaa ringed seals during 2003–2009

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Abstract

The Saimaa ringed seal (*Phoca hispida saimensis*) became isolated in Lake Saimaa at the end of the last ice age ca. 8000 years ago when the connection between Lake Saimaa and the Baltic Sea was broken. The Saimaa ringed seal is one of the very few seal species that live in inland waters. Due to climate change, however, the lack of an ice cover on the lake threatens to drastically decrease the seal population. The Saimaa ringed seal is an endangered species, with an estimated population of 260 seals, and it is protected by law.

As top predators in the aquatic food chain, fish-eating seals are vulnerable to the accumulation of contaminants. Seals are also important indicator species, which provide information about the current state of the environment. The aim of this study was to provide baseline data on radionuclide ^{137}Cs concentrations in Saimaa ringed seals. Altogether 54 seals were collected in Lake Saimaa by the Metsähallitus Natural Heritage Services between the years 2003 and 2009. The seals had died of natural causes or accidentally by drowning. The seals were sampled under the supervision of the Finnish Food Safety Authority EVIRA. Concentrations of ^{137}Cs were analysed in muscle, liver, kidney, bone, spleen and pancreas by the Radiation and Nuclear Safety Authority -STUK in Finland.

The mean radionuclide ^{137}Cs concentrations in muscle, liver and kidney, bone, spleen and pancreas were 76.2, 65.3, 63.5, 75.4, 90.0 and 83.3 Bq/kg f.w., respectively. The highest ^{137}Cs concentration, 180 Bq/kg f.w., was measured in 2004 and the lowest, 28.8 Bq/kg f.w., in 2009 in muscle tissue. The results indicate that the ^{137}Cs concentrations in Saimaa ringed seals decreased consistently not only in muscle, but also in liver and kidney, during the study period. A similar trend was found in the correlation between seal weight and the ^{137}Cs concentrations, which indicates high accumulation of ^{137}Cs in this species.

Introduction

The seals (*Phocida*) belong to high trophic level feeders that bioaccumulate many contaminants to a considerable extent. As top predators in the aquatic food chain, fish-eating seals are exposed to the accumulation of contaminants. Seals are also important indicator species, which provide information about the current state of the environment.

From the radiation protection point of view, ^{137}Cs is the most significant anthropogenic radionuclide because of its 30-year half life and mobility in the environment and subsequent accumulation in food chains. The Chernobyl accident in Russia in 1986 deposited ^{137}Cs fallout over large areas of Europe (AMAP 1998; Hamilton, 2004), including Lake Saimaa (Ylipieti et al.2008). The content of ^{137}Cs varies clearly in different parts of Lake Saimaa, the amount of fallout in 1986 ranging between 3-6 kBq/m^2 (Saxen, 2001).

In this study the concentration of ^{137}Cs in Saimaa ringed seals were analysed. The seal samples were collected in Lake Saimaa, which is a 4 400 square kilometer large freshwater lake situated in South East Finland (Fig. 1.). Lake Saimaa is the largest lake in Finland. All 54 seals were collected by Metsähallitus from different sites in Lake Saimaa: Haukivesi, Orivesi, Puruvesi, Pihlajavesi and Saimaa between the years 2003 and 2009 (Fig. 2.). All the seals had died of natural causes or accidentally by drowning.

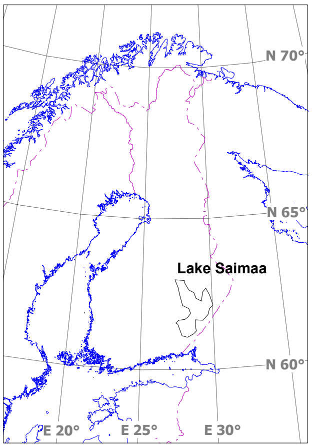

Fig.1. Location of Lake Saimaa.

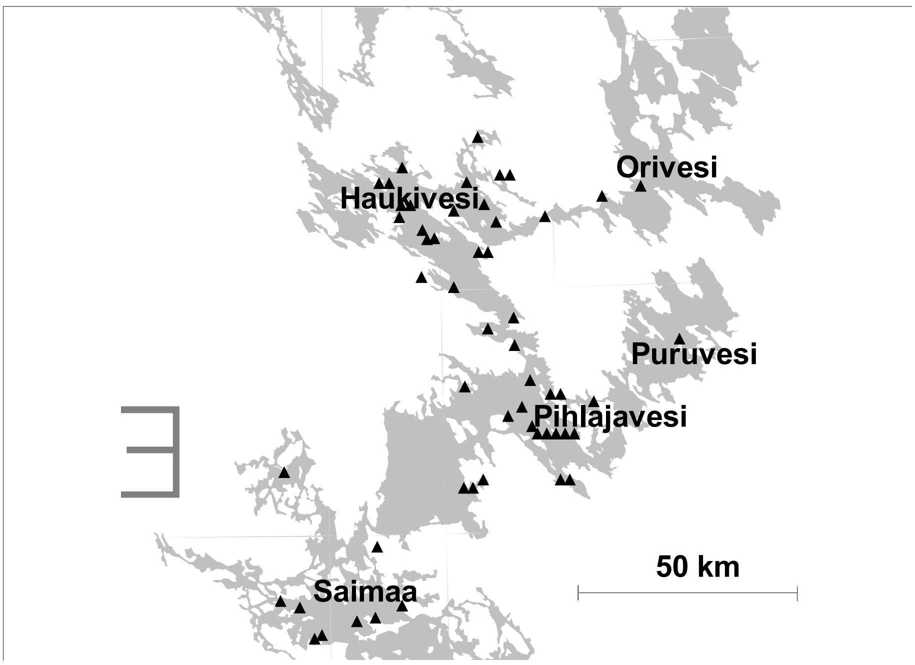

Fig. 2. Locations of the sampled seals.

Material and methods

The seals were divided into two groups according to their age. Adult seals (n = 35) were over one year old and pups (n = 19) under one year, most of the pups being only a few months old (Kokkonen, 2006). Preparation of the seals was carried out under the supervision of personnel from the Finnish Food Service Authority EVIRA. ^{137}Cs were analysed in muscle, liver, kidney, bone, spleen and pancreas by the STUK's Regional Laboratory in Northern Finland. A High-Purity Germanium (HPGe) detector with 30% relative effective was used in the analysis. The radioactivity concentration was determined using Gamma-99 software developed by STUK (Rantavaara et al., 1994).

Results

Muscle

^{137}Cs activity concentrations in muscle in adult and pup seals are presented in Table 1 and in the box plot in Fig. 3. Excluding 4 outlier points (2, 10, 11 and 23), the results show that the pups had almost similar ^{137}Cs concentrations in muscle than adult seals. In general, ^{137}Cs activity concentrations were below 100 Bq/kg f.w. Chernobyl-specific, short-lived ^{134}Cs was still detectable in small amounts in two muscle samples (0.23 and 0.38 Bg/kg f.w.). The correlations between ^{137}Cs concentrations and seal weight in adult seals are shown in Fig. 4.

Table 1. ^{137}Cs activity in muscle in adult and pups seals.

Descriptive Statistics

	N	Minimum	Maximum	Mean	Std. Deviation
adult_muscle	35	31,0	180	78,1	36,0
pup_muscle	19	28,8	154	72,7	30,4

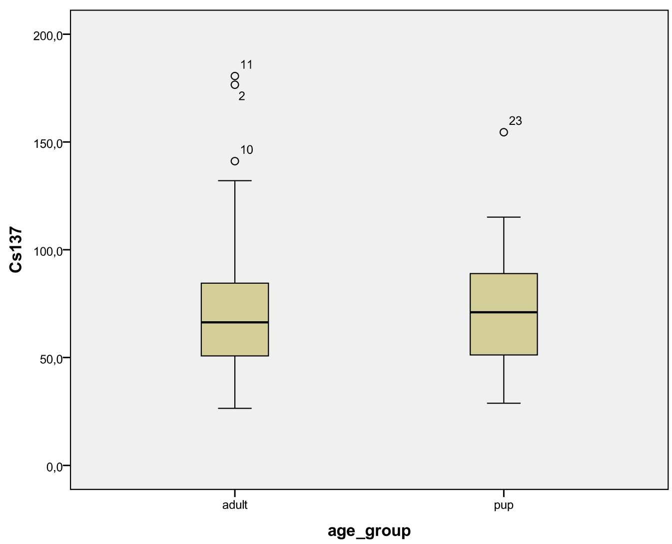

Fig. 3. ^{137}Cs concentrations in seal muscle in the two different age groups.

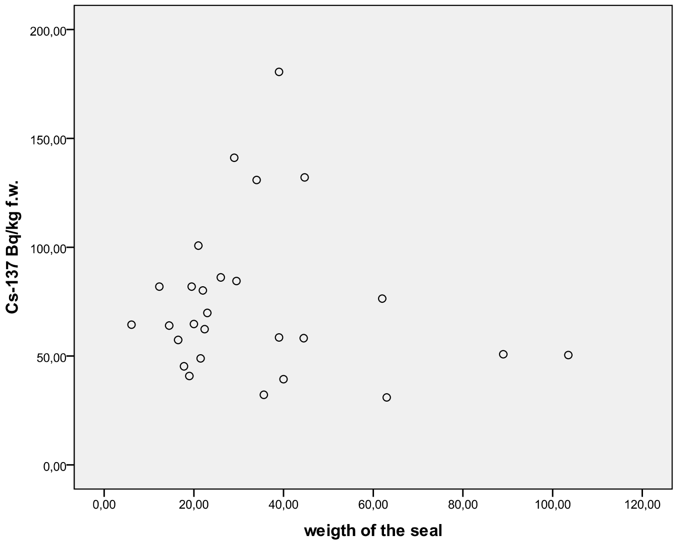

Fig. 4. Correlations between ^{137}Cs concentrations and seal weight in adult seals.

Liver

^{137}Cs activity concentrations in the liver samples were analysed in the same seals except for four adult seals and one pup seal. The results were at the same level and the same trend was observed as in the muscle samples Table 2. Pups were found to have almost similar ^{137}Cs activity concentrations than adults, as seen in Fig. 5.

Table 2. ^{137}Cs activity in seal liver in adult and pup seals.

Descriptive Statistics

	N	Minimum	Maximum	Mean	Std. Deviation
adult_liver	31	23,4	179	66,1	34,2
pup_liver	18	17,4	123	64,0	24,5

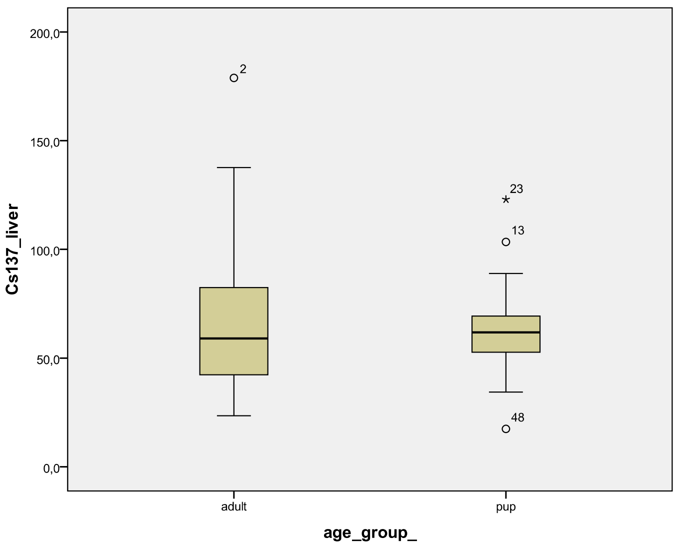

Fig. 5. ^{137}Cs concentrations in seal muscle in the two different age groups.

Kidney

^{137}Cs activity concentrations in the kidney samples are presented in Table 3. A small amount of Chernobyl-specific radionuclide ^{134}Cs was still detected in one kidney sample (4.6 Bq/kg f.w.).

Table 3. ^{137}Cs activity in adult and pup seal kidney.

Descriptive Statistics

	N	Minimum	Maximum	Mean	Std. Deviation
adult_kidney	31	21,0	167	60,1	31,9
pup_kidney	18	30,2	147	69,2	27,7

Bone, spleen and pancreas

Some bone, spleen and pancreas were also analysed. The results are presented in Table 4. ^{137}Cs concentrations in the spleen were at the same level in both adult and pups seals.

Table 4. ^{137}Cs activity in bone, spleen and pancreas in adult and pup seals.

Descriptive Statistics

	N	Minimum	Maximum	Mean	Std. Deviation
adult_bone	4	25.9	92.1	66.2	29.3
pup_bone	1	112	112	112	
adult_spleen	14	33.8	159	89.6	40.5
pup_spleen	16	34.1	159	90.4	33.9
adult_pancreas	1	117	117	117	
pup_pancreas	1	50.0	50.0	50.0	

Geographical distribution of the ^{137}Cs in seal muscle in Lake Saimaa

The geographical distribution of ^{137}Cs in seals muscle is presented in Fig. 6. Red bars indicate the mean concentrations in adult seals and yellow the mean concentrations in pups during 2003-2009. The fallout situation in 1986 and the ^{137}Cs activity in seal muscle are combined in the same geographical context. Intensity of the gray shading indicates the magnitute of the fallout. (Arvela et al., 1990).

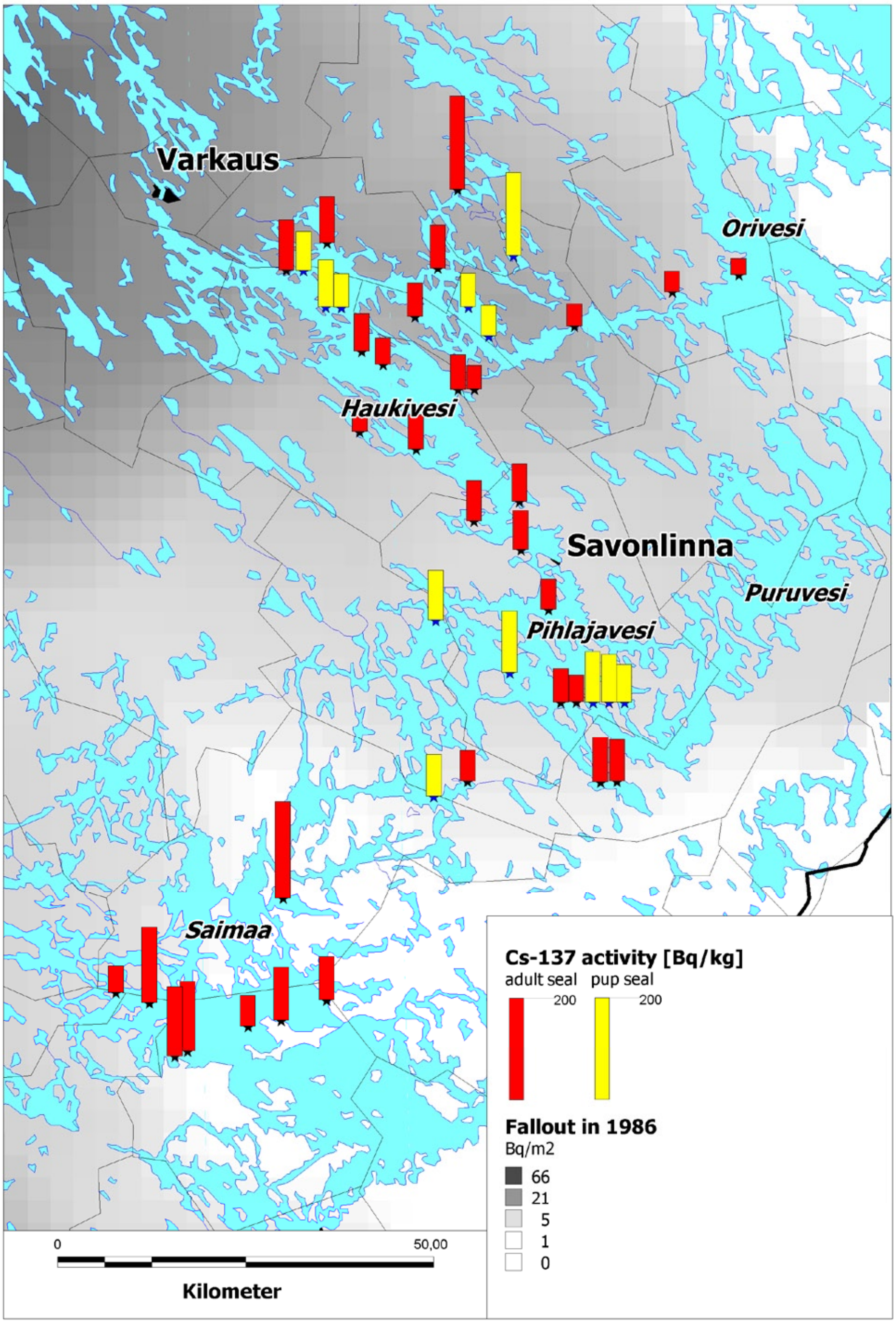

Fig. 6. ^{137}Cs activity in seal muscle and the level of Chernobyl fallout in 1986.

Changes in ^{137}Cs concentrations over time

Changes in the ^{137}Cs concentrations in time were observed in three tissues: muscle, liver and kidney (Fig. 7). One bar symbolizes the average minimum and maximum concentration per year. The ^{137}Cs concentrations in adults and pups were combined.

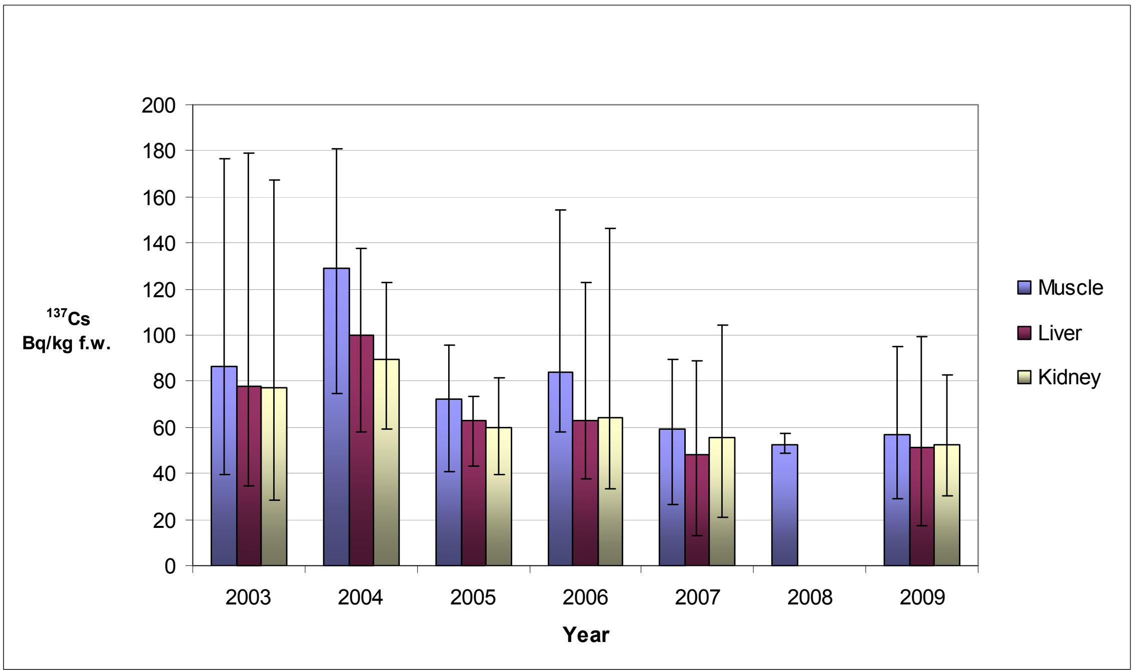

Fig. 7. ^{137}Cs concentrations in muscle, liver and kidney during 2003-2009.

Conclusions

Pups were found to have almost similar ^{137}Cs activity concentrations in their tissues (muscle, liver and kidney) than adult seals. No clear correlation was found between seal weight and ^{137}Cs activity in adults seals. This may be because the seals had been lying dead in the field for days or weeks before collection, with subsequent losses in their body weight. ^{137}Cs activity concentrations were found to decrease regularly over time in the tissues.

The highest ^{137}Cs concentration, 180 Bq/kg f.w., was found in muscle in adults seal. The mean concentration in muscle samples was 76.2 Bq/kg f.w, which is much higher than that in ringed seals from the Arctic Ocean, 0,21 Bq/kg f.w. (Carroll et al., 2002). This is due to the higher deposition from Chernobyl into Lake Saimaa than in the Arctic. The large variation in the ^{137}Cs contents is due to the uneven fallout from Chernobyl. A small amount of Chernobyl-specific radionuclide ^{134}Cs was detected in the same area where the fallout in 1986 was the highest, but the current ^{137}Cs activity concentration does not seem to correlate with the geographical distribution of the fallout in the southern part of the lake.

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Radioactivity of ^{210}Po in oysters collected in Taiwan

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Abstract

The Uranium-238 series decay product, polonium-210(^{210}Po), exists widely in the environment and is well known to be enriched in the tobacco, animal's viscera and suspension particles in surface sea-water. Oysters are cultivated in great quantity along the coast of Taiwan, and it is one of the most favorite foods for the Taiwanese. The oysters intake the suspension particles in surface sea-water, and the ^{210}Po is accumulated indirectly in them. It is necessary to evaluate the internal effective dose of ^{210}Po coming from the intake of oysters by Taiwanese. In this study, oysters around the coast of Taiwan were collected, dried, spiked with a ^{209}Po tracer for yield, and digested with concentrated HNO_3 and H_2O_2, and finally dissolved in 0.5 N HCl. The polonium was then spontaneously deposited onto a silver disc, and the activity of ^{210}Po was measured using an alpha spectrum analyzer equipped with a silicon barrier detector. Meanwhile, the internal effective dose of ^{210}Po coming from the intake of oysters by Taiwanese was evaluated.

Introduction

Natural environmental radioactivity arises mainly from primordial radionuclides, such as ^{40}K, and the radionuclides from the ^{232}Th and ^{238}U series. The major contribution to the radiation exposure received by mankind comes from natural sources. These include external sources such as cosmic rays and radiation from primordial radionuclides (^{238}U and ^{232}Th) and their decay products in the environment. Information on the levels of naturally occurring radionuclides is important as they also contribute a substantial fraction of the radiation dose to the natural ecosystems (Holtzam, 1966).

A large contribution to the radiation dose received by humans comes from the naturally occurring uranium series radionuclides accumulated in the body, namely alpha-emitting ^{210}Po (of physical half-life 138.4 days) and ^{210}Pb (precursor of ^{210}Po with physical half-life 22.2 years) (UNSCEAR, 1993). ^{210}Po is reported to account for up to 75% of the alpha-radiation dose in marine organisms (McDonald, Fowler, Heyraud, & Baxter, 1986) and up to 50% of the internal alpha-radiation dose in people (Holtzman, 1966). In 1988, a report of the United Nations Scientific Committee on the Effects of Atomic Radiation indicated that ^{210}Po is estimated to contribute about 7% of the total effective dose to man from ingested natural internal radiation (UNSCEAR, 1988). The polonium isotopes are amongst the most radiotoxic nuclides to human beings

(McDonald et al., 1986). The maximum permissible human body-burden for ingested ^{210}Po is only 1.1×10^3 Bq (CRC, 1982).

The main source of ^{210} Po in the environment is the gaseous ^{222} Rn, which escapes from the earth's crust into the atmosphere. The daughter products of ^{222} Rn are removed from the atmosphere with aerosol particles wet and dry deposition on the land surface and oceans (Skwarzec et al., 2001; Ladinskaya et al., 1973).

The main sources of ^{210}Po in the human body are food and cigarette smoke (Parfenov, 1974; Holtzman, 1978; UNSCEAR, 1982). Intake of ^{210}Po by food strongly depends on dietary habits. Polonium has an affinity for protein, enabling it to pass in significant quantities up the food chain. Thus, increased ^{210}Po burdens are found in people who consume high-protein diets of meat or fish and other seafood(Watson, 1985; Carvalho, 1995; Skwarzec, 1995). UNSCEAR (2000) quotes a worldwide average annual intake of 58 Bq of ^{210}Po in the diet. The value for annual ^{210}Po intake in the typical European diet is 40 Bq; however, this is based on data from only five countries (Italy, Poland, Romania, Russia and the UK).

The radioactivity of ^{210}Po had been measured in drinking and mineral water (Desideri et al., 2007a,b; Forte et al., 2007; Skwarzec et al., 2003, 2004; Vesterbacka, 2007)and cigarette but there are not much data for seafood , like oysters. The oyster is cultivated in great quantity along the coast of Taiwan, and it is one of the most favorite foods for the Taiwanese. The oysters intake the suspension particles in surface sea-water, and the ^{210}Po is accumulated indirectly in them.

The aim of the present study was to provide information on the levels of natural radionuclides ^{210}Po in samples of oysters products in different regions of Taiwan. The purpose of this work is to calculate the internal effective dose from ^{210}Po for the individual local public through oysters ingestion.

Experiment

Sampling and pretreatment

The samples of oysters sampling from 6 Taiwanese regions, as shown in Fig. 1, were purchased from traditional consumer markets.

The fresh oysters were dried at 110 ℃ for 24 hours in the oven. A weighted dry oyster sample was transferred into a beaker, then a known activity of ^{209}Po was spiked into the sample as a yield tracer. The oyster sample was digested with conc. HNO_3 and H_2O_2 solution. After the digested solution was evaporated to near dryness, the residue was then dissolved in 6 M HCl. The solution was filtered and the filtrate solution was then evaporated to near dryness. Finally, the residue was dissolved in 0.5 M HCl for deposited spontaneously. The process for the ^{210}Po analysis is shown in Fig. 2.

Spontaneous deposition

Polonium was spontaneously deposited onto a silver disk, from the 0.5 M hydrochloric solution in the presence of 1 g ascorbic acid (reduction of Fe^{3+}), at 90–95 ℃ in continuous for 4 hours. The spontaneous deposition plating apparatus for Po is shown in Fig3. The activities of ^{209}Po and ^{210}Po on the silver disk were measured by an alpha spectrometer equipped with semiconductor silicon detectors of surface barrier type. The counting efficiency of the silicon detectors for ^{209}Po and ^{210}Po was about 40%. The polonium yields from the analyzed oysters samples ranged from 50% to 65% for 80000 s counting time.

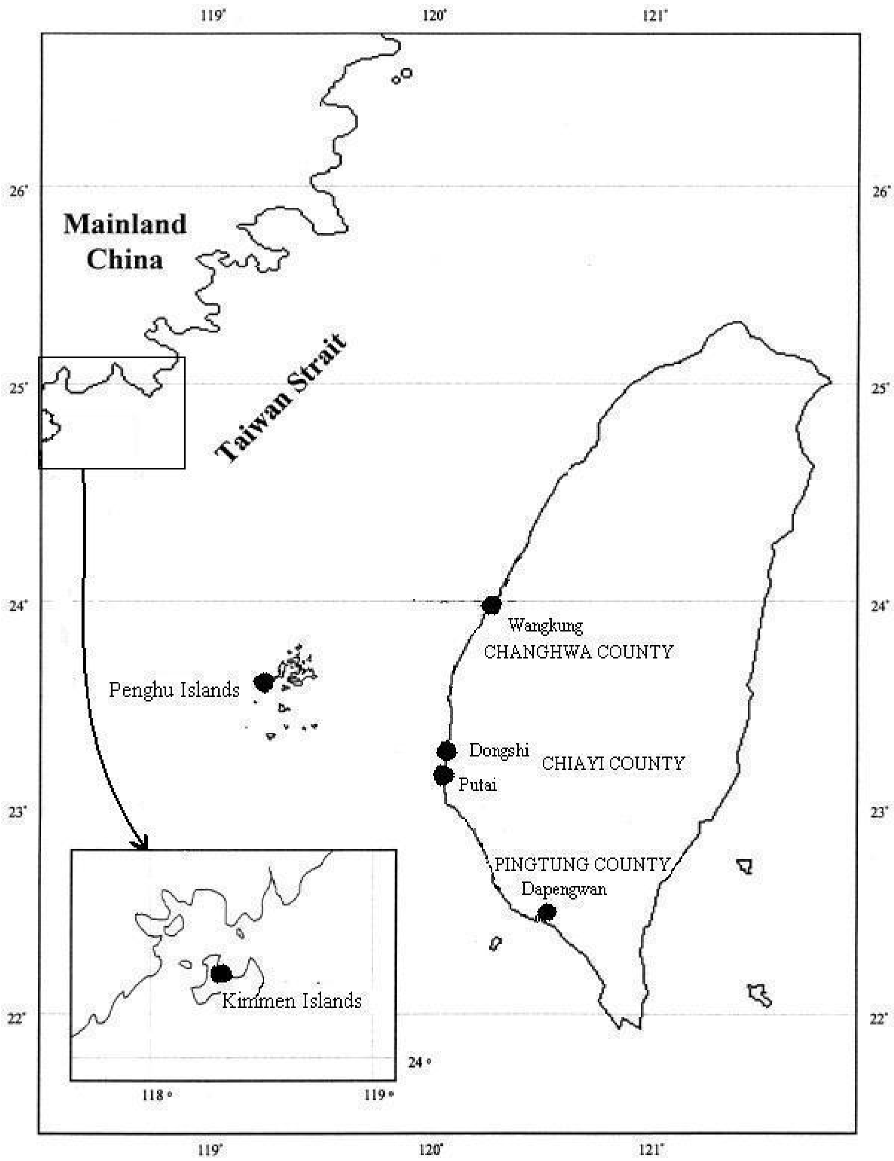

Fig. 1. Sampling locations(•) of oysters collected from different coastal areas in Taiwan.

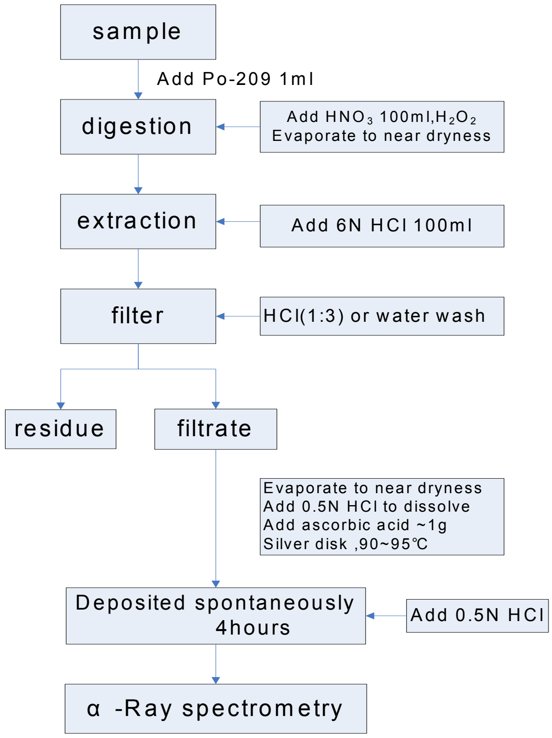

Fig. 2. process for Po-210 Analysis.

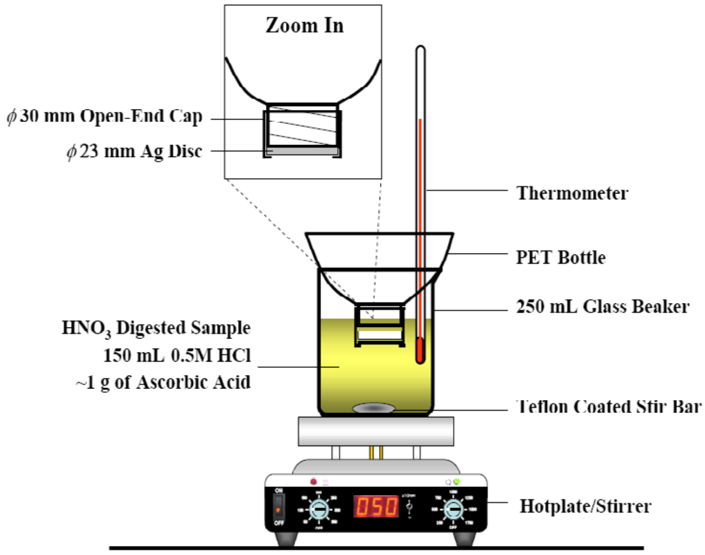

Fig. 3. Po spontaneous deposition plating apparatus.

Results and discussion

Radionuclide activity concentration

The ^{210}Po activity concentrations for oysters sampling from 6 regions of Taiwan in our study are listed in Tables 1. The mean value of ^{210}Po activity concentrations for all oyster samples studied is 50.1 Bq kg^{-1} .The minimum activity concentration of ^{210}Po (16.9 Bq kg^{-1}) was found for sample in Dongshi region. And, the maximum activity concentration of ^{210}Po(193 Bq kg^{-1}) was found for sample in Penghu island region.

The annual production of oysters in Taiwan was about 30420 metric ton(Fisheries Agency,R.O.C. 2006,2007,2008), among which the production of oysters in Kinmen and Penghu islands regions were only 1.3%. The main 98.7% of oysters are produced around the Taiwan coastal areas. However, according to the study results in Table 1, the oysters coming from the Penghu islands and the Kimmen islands have the higher ^{210}Po activity concentrations.

Table 1. ^{210}Po activity Concentration(BgKg^{-1})in oysters.

Sample	^{210}Po Concentration(BgKg^{-1})
Wangkung	23.3 – 24.3
Dongshi	16.9 – 18.8
Pudai	28.8 – 31.5
Dangpenwan	27.6 – 29.5
Penghu islands	59.6 – 193.1
Kimmen islands	69.5 – 78.4
Mean	50.1

Dose assessment

The internal effective dose of Taiwanese via ingestion of ^{210}Po present in the oyster was calculated. Annual individual effective doses were calculated using the dose per unit ingestion conversion factors(ICRP-72, 1996) of 1.2 μSv Bq^{-1}for ^{210}Po in the body. Based on the ^{210}Po daily intake from oysters ingestion. The oysters consumption quantity was calculated by the average product of oysters in Taiwan from 2006 to 2008. The average product for three years of oysters in Taiwan was 30420 metric ton.

The internal dose of the ingestion for ^{210}Po in oyster for Taiwanese can be calculated by Eq. (1) :

$$D = K^{\cdot} \quad G^{\cdot} \quad C \tag{1}$$

where D is the effect dose via ingestion(Sv); K is the ingesting dose conversion factor of the ^{210}Po radionuclide(1.2 μSv Bq^{-1}) (ICRP-72, 1996); G is the oysters consumption per year; here is 1.32 kg y^{-1}, which is calculated from the annual production of oyster in Taiwan divided by population of Taiwanese; and C is the mean activity concentration of the ^{210}Po radionuclide(50.1 Bq kg^{-1}).

The annual ingestion (Bq y^{-1}) and the effective doses(mSv y^{-1}) of ^{210}Po in oyster for Taiwanese were estimated and listed in Table 2. The annual ingestion and effective dose of Taiwanese due to the ^{210}Po in oysters were found to be in the range from 22.3 to 254.9 Bq y^{-1} and 2.68×10^{-2} to 3.06×10^{-1} mSv y^{-1}, and the mean values of the annual ingestion and the effective dose for Taiwanese were 66.1 Bq y^{-1} and 7.94×10^{-2} mSv y^{-1}, respectively.

Taiwan Radiation Monitoring Center indicated that the annual effective dose for adults from natural background radiation in Taiwan is 1.62 mSv y^{-1}, including 0.90 mSv from external radiation exposure and 0.72 mSv y^{-1} from internal radiation exposure. The natural radiation dose level is only about 2/3 of the global average(2.4 mSv y^{-1}) quoted in the UNSCEAR 1993 Report. The mean annual effective dose of Taiwanese due to the ingestion of ^{210}Po in oysters was 7.94×10^{-2} mSv y^{-1}; which is 11.0 % of the ingestion dose(0.72 mSv y^{-1}) and 4.9 % of the total effective dose(1.62mSv y^{-1}) of the Taiwanese.

Table 2. Annual effective dose(mSv y^{-1}) of Taiwanese due to the ingestion of ^{210}Po in oyster.

Locations	Annual ingestion of ^{210}Po in oyster (Bg y^{-1})	Annual dose (mSv y^{-1})
Wangkung	30.8 – 32.1	3.69×10^{-2} – 3.85×10^{-2}
Dongshi	22.3 – 24.8	2.68×10^{-2} – 2.98×10^{-2}
Pudai	38.0 – 41.6	4.56×10^{-2} – 4.99×10^{-2}
Dangpenwan	36.4 – 38.9	4.37×10^{-2} – 4.67×10^{-2}
Penghu	78.7 – 254.9	9.44×10^{-2} – 3.06×10^{-1}
Kimmen	91.7 – 103.5	1.10×10^{-1} – 1.24×10^{-1}
Mean	66.1	7.94×10^{-2}

Conclusion

The ^{210}Po activity in the different Taiwan regions were determined. The results indicated that the oysters coming from Penghu island and Kinmen island regions contain higher concentrations of ^{210}Po in comparison with oysters from other regions, in spite of the oyster production of these two regions were just about 1.3 % of the total amount of Taiwanese oysters.

The mean annual ingestion and effective dose of Taiwanese due to the ingestion of ^{210}Po in oysters were 66.1 Bq y^{-1} and 7.94×10^{-2} mSv y^{-1}; which is 11.0 % of the ingestion dose(0.72 mSv y^{-1}) and 4.9 % of the total effective dose(1.6mSv y^{-1}) of the Taiwanese.

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Natural alpha emitting radionuclides in bottled drinking water, mineral water and tap water

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Abstract

Quantitative information about the activity concentrations of critical alpha- emitting radionuclides in food and drink is important in the study of cumulative radiation effects on human health. In most countries there is an increasing tendency to replace tap water by consumption of commercial bottled natural and mineral water. Furthermore, various beverages as well as dietary supplements are also prepared from mineral water, not ordinary tap water. In this work tap water and bottled drinking and mineral water were collected in Slovenia and analysed in order to assess the radiation doses from ^{238}U, ^{234}U, ^{226}Ra and ^{210}Po. On the basis of radionuclide activity concentrations the internal radiation doses to individuals were assessed and are discussed together with the contribution of each particular radionuclide to the dose.

Introduction

In most European countries, there is an increasing tendency in population to replace tap water of satisfactory quality for human consumption with commercial bottled natural and mineral water. Moreover, various beverages are also prepared from mineral water, not ordinary tap water. Systematic studies on radiological characterisation of drinking water started after 1993, when the recommendations of the Guidelines for drinking water quality, issued by the World Health Organisation were published (WHO, 1993). These guidelines state that drinking water is safe from the radiological point of view if within the range of normal consumption (2 L per day), the annual dose rate originating from the presence of radioactive nuclides does not exceed 0.1 mSv. UNSCEAR reports (UNSCEAR, 1998, 2000) estimated that exposure to natural sources contributes more than 98 % of the radiation dose to the population (medical treatment is not taken into account). The main contribution to dose is largely due to the presence of naturally occurring radionuclides of both the uranium and thorium decay series. Due to their high radiotoxicity, the contributions of ^{210}Po and ^{228}Ra to the dose are more pronounced. The dose contributions of the radionuclides are in the order: ^{210}Po > ^{228}Ra > ^{210}Pb > ^{226}Ra > ^{234}U > ^{238}U > ^{224}Ra > ^{235}U. Increased concerns concerning the radiological quality of drinking water has led to an increased demand for real data assessment. The old drinking water regulation 980/778/EEC from 1980 (EC, 1980) in which neither radioactivity nor uranium were mentioned, were replaced by the European Directive 98/83/EC in 1998 (EC,1998). In this Directive, the reference dose level of committed annual effective dose due to drinking water consumption is 0.1 mSv. The Directive points out that the total indicative dose must be evaluated excluding tritium, ^{40}K, ^{14}C, radon and its decay products, but including all other radionuclides of the natural decay chains. The maximum values for radon and long-lived radon decay products such as ^{210}Pb and ^{210}Po are proposed in the European Commission Recommendation 2001/928/Euratom (EC, 2001). Uranium is covered by the Directive, although its contribution to the dose is minor due its small dose conversion factor. However, uranium is a toxic heavy metal and therefore has to be regulated and controlled. The WHO set the most stringent limitation of 2 µg/L in its 1998 report (WHO, 1998), but later (WHO, 2004) changed this limit to 15 µg/L; the USA (EPA, 2000) set the limit to 20 µg/L. Recently, Germany set the uranium limit of 2 µg/L for mineral water considered suitable for infants (Bundesgesetzblatt, 2006). Considering the importance of water for human consumption, its quality has to be assured and regularly controlled. The assessment of the radiological quality of natural or bottled drinking and mineral waters is also important in view of assessment and reduction of the radiation exposure of the population. For practical purposes, the recommended screening levels for drinking water below which no further actions are required are 0.1 Bq/L for gross alpha activity and 1 Bq/L for gross beta activity. If these values are exceeded, determination of particular radionuclides dissolved in drinking water needs to be performed. In 2004, the WHO published the third edition of its guidelines for drinking water (WHO, 2004) in which the recommended screening level for gross alpha activity was increased from 0.1 to 0.5 Bq/L. Due to the increasing tendency in the consumption behaviour of the population to replace surface tap water of sufficient quality for human consumption with commercial bottled drinking natural and mineral water, several studies to assess the radioactivity levels in bottled drinking and mineral water were performed around Europe. The report of Weknow (Weknow, 2005) gave a good overview of radioactivity levels found in drinking water across Europe, while data on drinking water quality in Slovenia are very scarce.

The experimental design of our study was to determine the activity concentrations of the alpha emitters ^{238}U, ^{234}U, ^{226}Ra and ^{210}Po in Slovenian bottled drinking and mineral water, as well as in tap water. The studied samples included bottled drinking and mineral water purchased in the period from October 2009 to March 2010, on the market in Ljubljana. All water samples originated from Slovenia. There are many companies in Slovenia producing natural drinking and mineral water from bedrock aquifers of different depths. Tap water was collected from chosen cities in Slovenia.

Materials and methods

For this study we analysed three of the most frequently sold mineral waters and eight bottled drinking waters “from the shelf”. We also analysed six tap waters (Fig.1). All reagents used in the analysis were of analytical grade. The tracer solutions (^{232}U, ^{209}Po, ^{133}Ba) used in the study were prepared from calibrated solutions purchased from Analytics, Inc. (Atlanta, GA, USA). The producer maintains traceability to NIST standards.

An alpha spectrometer (EG&G ORTEC) with a passivated implanted planar silicon (PIPS) semiconductor detector with an active area of 450 mm^2 and 28% efficiency for a 25 mm diameter disc was used for alpha-particle spectrometric measurements. The calibration of the detector was made with a standard radionuclide source, containing ^{238}U, ^{234}U, ^{239}Pu and ^{241}Am (code: 67978-121), obtained from Analytics, Inc. A coaxial HP Ge detector was used for measurements of the gamma emitting nuclide ^{133}Ba.

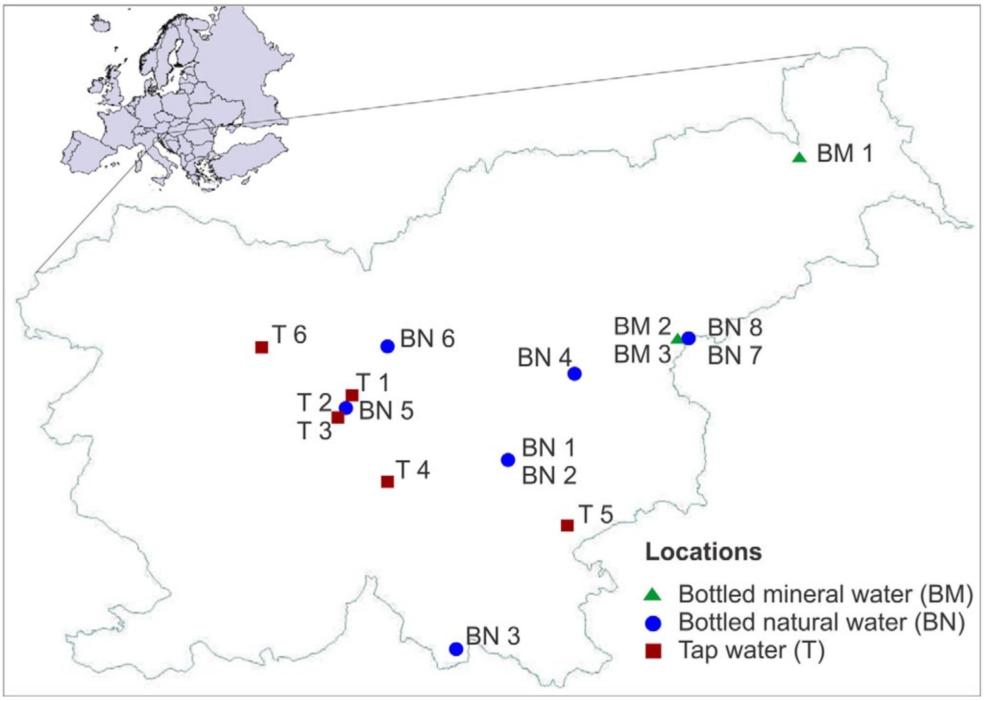

Fig. 1. Locations of tap water and bottling facilities of selected natural and mineral waters from Slovenia.

For determination of uranium radioisotopes a known amount of ^{232}U tracer (~ 0.5 Bq) was added to the water sample which was acidified with concentrated HNO_3 (3mL of acid per 1L of sample). Uranium was preconcentrated from the water samples by coprecipitation with iron(III) hydroxide at pH 9-10 using an ammonia solution. The precipitate was separated by centrifugation, washed with distilled water and dissolved in concentrated nitric acid. The solution was adjusted with distilled water to 3M HNO_3 and loaded onto a UTEVA column (Eichrom Industries Inc.) (Horwitz et al., 1993) pre-conditioned with 5 mL 3M HNO_3. The column was then washed with 3M HNO_3. Thorium radioisotopes were stripped from the column with 9M and 5M HCl. Uranium radioisotopes were eluted with 15 mL 1M HCl. The microcoprecipitation method with neodymium fluoride was used for thin source preparation in the alpha spectrometric determination (Hindman, 1983, Sill and Williams, 1981). The neodymium fluoride suspension was filtered through a 25 mm diameter 0.1 µm polypropylene filter. The dry filter was mounted on a stainless steel disc.

The analytical scheme for determination of ^{226}Ra was adapted from Lozano et al. (Lozano et al., 1997). The procedure is based on coprecipitation of $Pb(Ra)(Ba)SO_4$. The water sample was transferred to a glass beaker and acidified with concentrated H_2SO_4 (10 mL of sulphuric acid per 1L of sample). After addition of ^{133}Ba tracer together with Ba-carrier, the sample was stirred for approximately 30 min. With stirring, 30 mg Pb^{2+} was added in portions to allow good coprecipitation of radium and barium. After settling, the suspension was centrifuged and washed with distilled water. The $PbSO_4$ precipitate containing radium and barium was dissolved in 4 mL 0.1M EDTA, prepared in 0.5M NaOH. For ^{226}Ra determination 250 µg of 0.3 mg/mL Ba^{2+} solution was added together with 4 mL of saturated Na_2SO_4 solution. With stirring, 1:1

acetic acid solution was added until pH 4–5 was reached, thus precipitating $BaSO_4$, while Pb^{2+} ions remained in solution. Immediately after, 0.2 mL of a 0.125 mg/mL $BaSO_4$ suspension was added, acting as a seeding precipitate to obtain small particles. The suspension was allowed to settle for 30 min and filtered through a 25 mm 0.1μm polypropylene filter. The filter with $BaSO_4$ deposit was dried and mounted on a stainless steel disc and measured by γ–ray spectrometry for ^{133}Ba yield determination and by α–particle spectrometry for determination of ^{226}Ra.

For determination of ^{210}Po in water, ^{209}Po (~ 0.3 Bq) tracer was added to 9 L of water. After sample acidification with concentrated HCl (2 mL of acid per 1 L of sample), the radionuclides were coprecipitated with MnO_2. Precipitation of MnO_2 was achieved by adding $KMnO_4$ and $MnCl_2$ and adjusting the pH to 9 with ammonia solution. The precipitate was then dissolved with a mixture of HCl and H_2O_2, and adjusted with distilled water to pH 1. To prevent co-plating of other potentially interfering ions (Fe^{3+}, Mn^{6+}), 0.5 g of ascorbic acid was added. The spontaneous deposition of polonium on a 19 mm diameter silver disc was carried out at 90 °C for 4 hours. The Ag disc, covered on one side, was fixed in a holder and immersed in the solution (Benedik and Vreček, 2001). Polonium radioisotopes were then measured by alpha spectrometry.

Based on the results of activity concentrations of the four alpha emitters in drinking water presented in Table 1, the internal doses (committed effective dose) for an adult were estimated using an annual consumption rate of 730 L/year accoording to the WHO Guidelines for Drinking Water Quality (2004) and the dose coefficients of the relevant radionuclides from the “International Basic Safety Standards for Protection against Ionizing Radiation and for Safety of Radiation Sources” (IAEA, 1996).

Results and discussion

In Table 1 the results of the activity levels of ^{238}U, ^{234}U, ^{226}Ra and ^{210}Po in three different groups of drinking water, natural and mineral bottled water and tap water, are given. As seen the activity concentrations of uranium in the water samples analysed ranged from 1.1–57 mBq/L and 2.8–173 mBq/L for ^{238}U and ^{234}U, respectively. These values, except for the mineral water BM2 (57 and 173 mBq/L for ^{238}U and ^{234}U, respectively) were relatively low, comparable with some literature data from Italy (Jia and Torri, 2007) and well below the limit values (98/83/EC, 2004, WHO,2004). The lowest absolute values were found in tap waters; however, somewhat elevated levels were measured in three bottled natural waters originating from the east (BN1, BN2) and south (BN3) of the country.

^{226}Ra (Table 1) was in the range between 0.14–31.7 mBq/L and like uranium, can be regarded as low and comparable with data reported for Europe (Weknow, 2005). The highest absolute level of 32 mBq/L was found in a bottled natural water (BN1) which also had an elevated uranium level. Elevated activity concentrations higher than 10 mBq/L were found in two mineral (BM2, BM3) and two bottled natural waters (BN7, BN8) coming from the same region in the eastern part of the country (Fig.1), known for its thermal and mineral springs and spas. With the exception of one sample (T4), tap water had low levels of ^{226}Ra in a narrow range between 0.3-and 1.4 mBq/L.

Table 1. Radionuclide activity concentrations (mBq/L) in natural and mineral bottled water and tap water collected in Slovenia.

Sample	Type of sample	U-238	U-234	Ra-226	Po-210
BN1	**Natural water**	**28 ± 3**	**71± 8**	**32 ± 4**	1,1 ± 0,4
BN2		18 ± 1	57 ± 6	11 ± 1	2,1 ± 0,3
BN3		13 ± 2	15 ± 2	1,7 ± 0,2	0,9 ± 0,2
BN4		2,2 ± 0,5	3,5 ± 0,7	2,3 ± 0,3	0,43 ± 0,12
BN5		8,3 ± 1,1	12 ± 1	0,14 ± 0,05	2,0 ± 0,4
BN6		2,9 ± 0,5	13 ± 2	6,8 ± 0,8	1,2 ±0,3
BN7		5,1 ± 0,6	14 ± 2	16 ± 2	0,24 ± 0,07
BN8		4,2 ± 0,6	8,7 ± 1,2	15 ± 2	0,6 ± 0,2
BM1	Mineral water	1,1 ± 0,2	2,8 ± 0,5	2,4 ± 0,3	0,39 ± 0,11
BM2		**57 ± 9**	**173 ± 28**	12 ± 2	1,0 ± 0,3
BM3		5,2 ± 1,8	12 ± 2	17 ± 2	0,6 ± 0,1
T1	Tap water	7,2 ± 0,7	8,5 ± 0,8	1,0 ± 0,2	0,25 ± 0,06
T2		4,8 ± 0,9	6,9 ± 1,1	0,47 ± 0,10	1,0 ± 0,2
T3		6,7 ± 1,1	11 ± 2	1,3 ± 0,1	0,77 ± 0,16
T4		8,2 ± 2,8	8,8 ± 2,9	15 ± 2	1,1 ± 0,2
T5		7,0 ± 0,9	11 ± 1	1,4 ± 0,1	1,8 ± 0,4
T6		1,1 ± 0,3	3,0 ± 0,5	0,30 ± 0,03	0,67 ± 0,19

Among the alpha emitters analysed ^{210}Po has the lowest activity concentrations in the range between 0.24-2.1 mBq/L in all three groups of drinking water.

In Fig.2 the results of the calculated total effective doses (μSv/a) for adults drinking different types of water are presented. It is seen that doses are on average very low, in the range of 1-11 (μSv/a) and only by drinking for a whole year one mineral (BM2) or one natural bottled water (BN1) would the person obtain a dose higher than 10 μSv/a, which still represents only one tenth of the recommended reference dose level (RDL) of 0.1 mSv from 1 year´s consumption of drinking water (EC 1998).

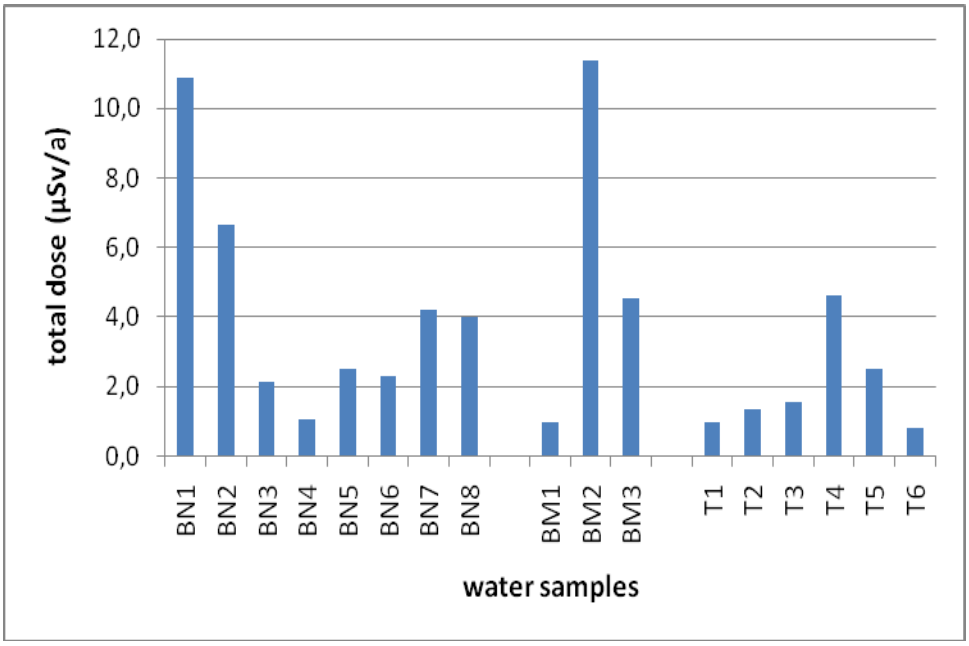

Fig. 2. Total internal dose (µSv/a) to adult member of the public due to drinking different types of water.

The contribution of each analysed radionuclide to the annual total internal dose varies among different types of water samples, but on average in natural bottled waters the contributions were in the order ^{226}Ra (mean: 46 ± 28%) > ^{210}Po (27 ± 22%) > ^{234}U (19 ±8%) >^{238}U (8 ± 6%) (Fig.3). In mineral water the order was similar (Fig.3). Only in the mineral water BM2 did ^{234}U contribute more than 54%, followed by ^{226}Ra (22 %), ^{238}U (16%) and ^{210}Po (8%). In tap water ^{210}Po (mean: 48 ± 23%) was the main radionuclide contributing to total dose due to its high dose conversion factor.

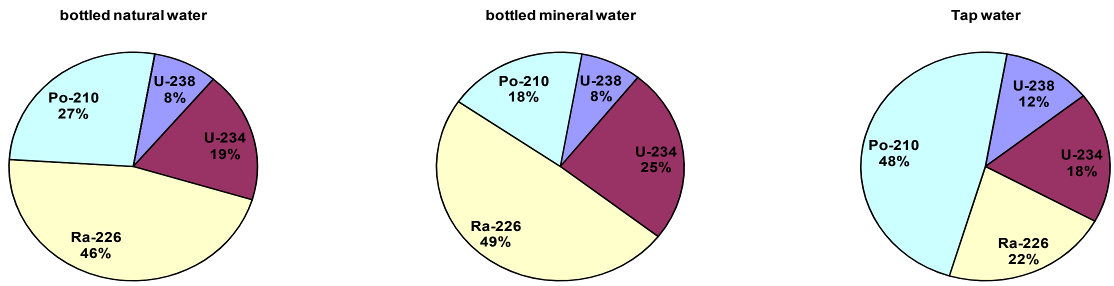

Fig. 3. The average contribution (%) of alpha radionuclides to internal commited effective doses to adult members of the public drinking different types of water.

Conclusions

The present study was a pilot study where only four selected alpha emitters were analysed in different drinking waters from Slovenia. From the survey it is evident that the activity concentrations, with the exception of one mineral and two natural drinking waters, were very low also leading to low calculated internal doses which constitute only a few percent of the recommended reference dose level (RDL) of 0.1 mSv. However, beside the analysed radionuclides there are also some other radio isotopes, namely the long- lived radon decay products ^{210}Pb and ^{228}Ra, which both have high dose conversion factors and should be determined and included in dose estimations in the future.

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Radiation protection of the public and the environment: long-term, large-scale radioecological monitoring by spruce needles

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Abstract

In a two years radioecological study spruce needle samples of the Austrian Bioindicator Grid collected between 1984 and 2008 were analysed retrospectively by low-level gamma-ray spectrometry to investigate the geographical and temporal distribution of radionuclides in spruce needles of the last 25 years. Main focus was the development of the radioactive contamination before and after the Chernobyl fallout 1986. Overall more than 750 spruce needle samples of selected locations – well distributed among the area of Upper Austria – were analysed for different natural and anthropogenic radionuclides: Cs-137, K-40, Pb-210, Ra-226, Ra-228, U-238. Additionally soil samples were taken at selected sites to estimate transfer factors to describe the transfer of radionuclides from soil to spruce needles. On the basis of the measured Cs-137 activity concentrations in the spruce needles and soil samples estimations are carried out, how to use the Bioindicator spruce needles for environmental radioactivity monitoring. Hence the detection limits for additional Cs-137 deposits (Bq/m^2) in spruce needles samples are estimated at various locations. Furthermore the results have been integrated into an existing environmental surveillance programme in Upper Austria.

Slovenian experience with inconsistencies in the global contamination monitoring results

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Abstract

Introduction Monitoring of the global radioactive contamination due to atmospheric nuclear bomb tests (1951-1980) and the Chernobyl accident (1986) has been carried out in Slovenia since the early sixties. The primary purpose of the monitoring programme is to provide a basis for calculating the exposure of the population due to radioactive contaminants. Secondarily, due to the very long measuring period, it is possible to observe and understand the trends of radionuclide concentrations in different media. Above all, two long-lived fission radionuclides, ^{137}Cs and ^{90}Sr, have been followed in the atmosphere, water, soil and in drinking water as well as in foodstuffs and feeding stuffs. In addition to that, river water contamination with ^{131}I due to medical use was also monitored. In all samples, other natural gamma emitters are also measured, as well as ^{3}H in surface water, drinking water and precipitation.

The measurement programme was for years roughly equally divided between two technical support organisations, each organisation always measuring the same type of samples, with few samples duplicated for checking purposes. With the changes in legislation and public procurements, this system had to be changed. As a consequence, the continuity is broken since 2005. The monitoring programme is still divided into 2 roughly equivalent parts, with organisations switching between those parts practically every year.

Both organisations are authorised for performing monitoring by the Slovenian Nuclear Safety Administration. The Slovenian legislation has set the accreditation according to ISO/IEC 17025 standard, used by testing and calibration laboratories, as one of the conditions for this. If we take this into account, the change of monitoring operator should not present a problem and all results should be consistent. Nevertheless, the SNSA have noticed some irregularities in long term trends.

Material and methods

Programmes for monitoring of levels of radioactivity in the environment as a consequence of global contamination due to nuclear bomb tests and the Chernobyl accident have been continuously carried out since the 1960s. In addition to the data on levels of radioactive contamination of the environment, the SNSA has also gathered data from the Krško NPP, the Žirovski Vrh Mine, the Low and Intermediate Level

Waste Depository and the Reactor Centre at Brinje. Beside regular monitoring programmes, the SNSA orders individual studies of particularly interesting aspects of contamination in the environment. All data from regular programmes and most of the data gathered in individual studies carried out since 1961 have been entered into a single database at the SNSA, named ROKO (the name derives from Slovenian Radioactivity in the Environment).

ROKO is a relational database with a simple user interface, intended for use of the SNSA staff and the whole community (http://www.radioaktivnost.si/ROKO/roko.php). We used data from the database to examine trends and to find inconsistencies in monitoring results that have arisen from changes of sampling locations, different calibrations, possible different sampling procedures or simply unexplained discrepancies.

In Slovenia, there are few organization that are authorised by the SNSA to perform environmental monitoring. The two most important ones are "Jožef Stefan" Institute (IJS) and Institute for occupational safety (ZVD). These two are accredited for measuring radioactivity in all environmental samples and perform yearly monitoring programmes for the SNSA and all nuclear facilities.

Results and discussion

Concentrations in air

Radionuclide concentration in air was historically measured by ZVD at Golovec, a small hill in the suburb of Ljubljana. The sampling point was situated in the forest environment, with higher ^{137}Cs content in upper layer of soil and consequently resuspension of particles with higher activity concentration of ^{137}Cs. In 2006, as well as in 2008 and 2009, IJS measured air in Podgorica, on flat alluvial ground with lower content of ^{137}Cs. The 2007 measurements were made by ZVD at Polje, situated also on the flat area of the Ljubljana basin. ^{137}Cs was measured in monthly samples and averaged over the whole year (
Figure 1). Values measured in Podgorica are systematically lower than the other measurements.

Although these 3 locations are geographically close (few kilometres around Ljubljana), their properties regarding environmental radioactivity are very different, making them unsuitable for following long term concentration trends.

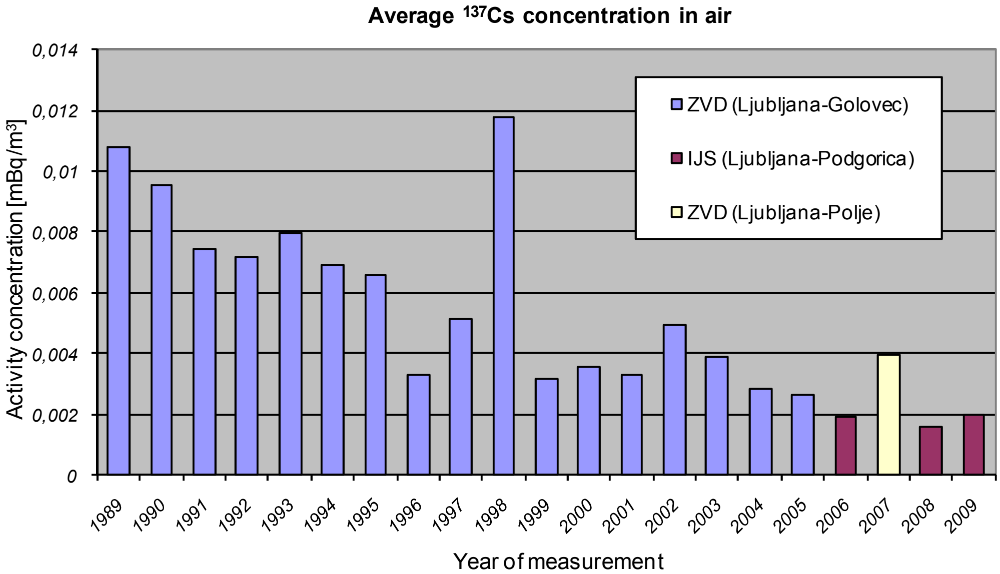

Figure 1. Average concentration of ^{137}Cs in air. Elevated values in 1998 are due to the accidental smelt of a ^{137}Cs source in Algeciras, Spain.

The activity concentration of ^{7}Be in air is also interesting. Values, measured by IJS, are obviously higher. Possible reasons for that could be in the difference between locations. Golovec is windswept and elevated, situated in a forest, Ljubljana-Podgorica is flat for several kilometres with no trees so there is no filtration, as shown on Figure 2. The measurements in Polje can not yet be evaluated, since there are not enough data till now.

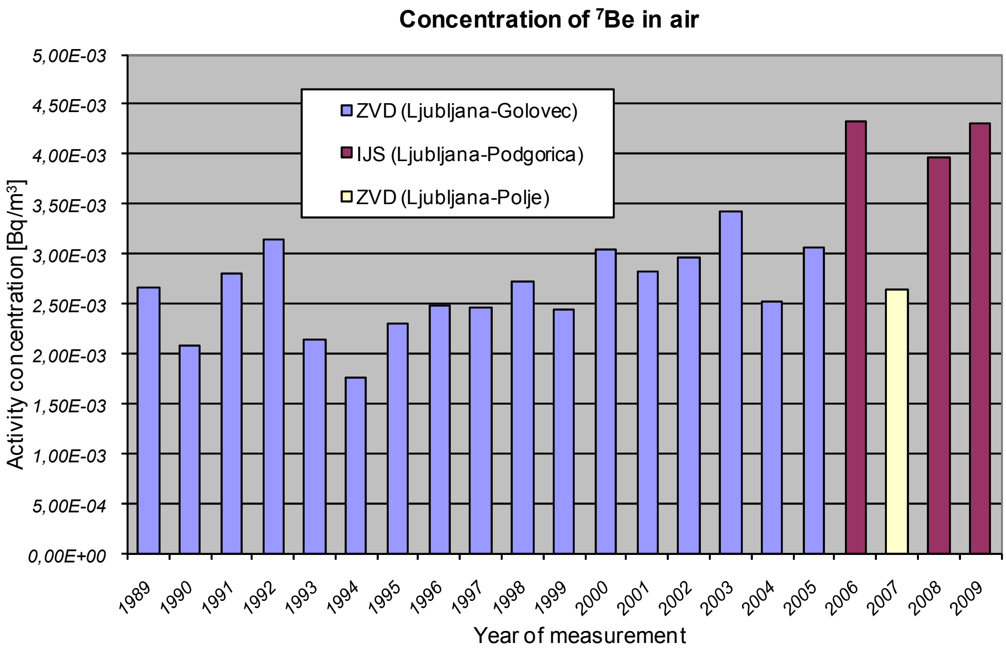

Figure 2. Average concentration of ^{7}Be in air. Measurements in Ljubljana (Podgorica) yield higher values.

Concentrations in soil

The long-term sampling place for radionuclide concentration measurement in soil was chosen in the 1960s at a big co-operative farm in Barje near Ljubljana (location name Ljubljana), with the intention to follow ^{137}Cs and ^{90}Sr along the food-chain. The laboratory that performed soil sampling in 2006 changed the sampling point to Podgorica. As a consequence, clayey soil sampled for four decades was substituted by washed-off alluvial soil with poor ^{137}Cs content. The measurements (Figure 3) show lower values for both years when soils was sampled in Podgorica (2006 and 2008), and the value for 2007, sampled again in Barje, is consistent with past measurements. In 2009, the measurements were again done in Barje, but this time by the laboratory that previously used the Podgorica location. The results are consistent with measurements done at the same location in the past, so we can assume that both laboratories have comparable sampling procedures and the past discrepancies were only due to different soil characteristics.

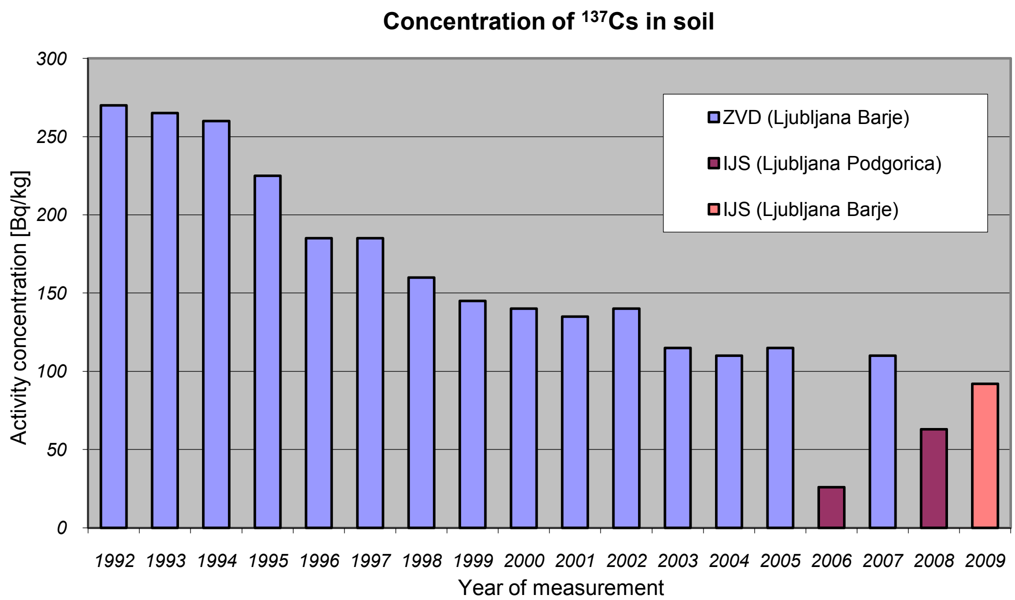

Figure 3. Activity concentration of ^{137}Cs in soil. Measurements done on different soil show different ^{137}Cs content.

However, such discontinuities in sampling and measurement may also lead to faulty analyses and evaluations. In 2006, the concentration of ^{137}Cs in soil was generalized for the whole country, and the evaluator omitted any comment on discontinuity of sampling history. As a consequence, the dose to the population calculated from this result was vastly underestimated.

Concentrations in river water

In the last decade, the content of ^{3}H in the Sava river showed average yearly concentrations between 1,3 and 1,6 kBq/m^3, while in the last 2 years the averages dropped to around 0,8 kBq/m^3. Up till the year 2008, the measurements were performed by IJS, department for environmental sciences, the first 9 months of 2008 were analysed by Austrian Research Centre Seibersdorf, and the last 3 months as well as 2009 by IJS again. (Figure 4).

The elevated values in the years before 2008 are explained by the laboratory to be a consequence of faulty calibration of the equipment (roughly 20%), which was later rectified, yet the results since are still not within the expected values. As a comparison, we used data collected by ARC Seibersdorf in rivers in Austria as well as data from Slovenia measured by IJS. The concentrations of ^{3}H in all these examples are consistently in the 1,1-1,4 kBq/m^3 range. ^{3}H concentrations measured in precipitation in these cases are slightly higher but also of the same range, so the low values measured in the Sava river at Krško are not expected and can not be easily explained.

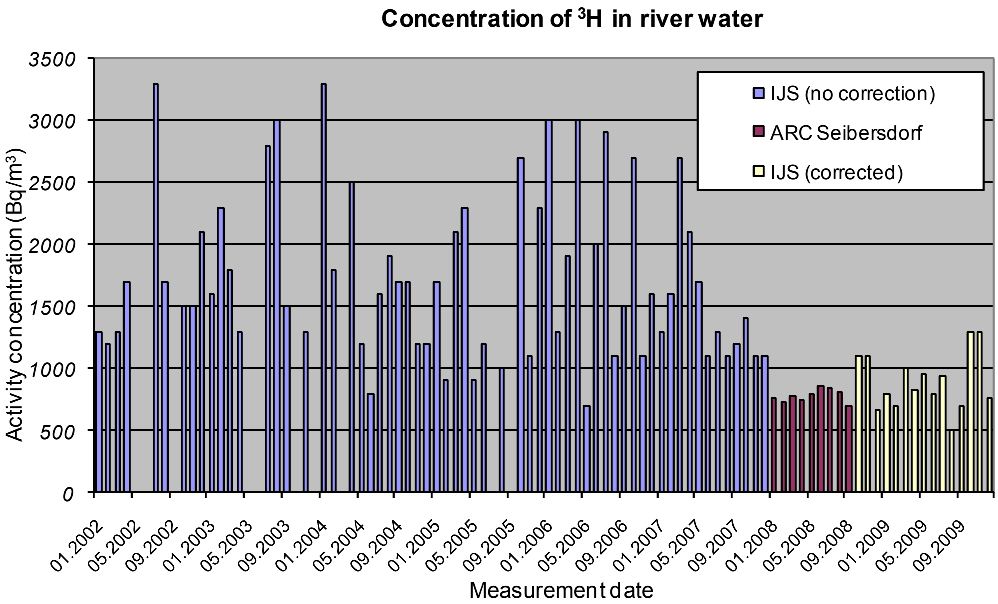

Figure 4. Activity concentration of ^{3}H in river water. Recent measurements show significantly lower values.

In 2008, the SNSA has started a campaign of independent monitoring programs designed to confirm results obtained thrugh the operational monitoring of emissions and immissionsthat is performed by nuclear facilities. Each year, a number of selected samples are simultaneously measured by an authorised organisation that is not involved in the operational monitoring of those samples. The sampling process is allways witnessed by the SNSA staff.

Among the samples checked in 2009 were grab samples of unfiltered Sava river water. It was measured by IJS in the scope of the operational monitoring program and checked independently by the ZVD laboratory. Measurements of concentrations of some radionuclides in samples simultaneously taken at Krško and Brežice are shown in Table 1.

Table 1. Measurements of grab Sava river water samples.

	^{210}Pb (Bq/m^3)	^{131}I (Bq/m^3)	^{40}K (Bq/m^3)
IJS	<2 to <6	8	120
ZVD	14 to 35	26	43

The values differ by up to a decade in worst cases, although the sampling was done simultaneously. Through the analysis of the SNSA photo archive, we were able to pinpoint the possible reason for these discrepancies in different sampling metodology. It is worth mentioning that results from simultaneously taken sediment samples show acceptable agreement. Since sediments are much easier to uniformly sample then unfiltered river water, it only strengthens the premise that sampling procedures must be re-examined and possibly prescribed by the authority in order to obtain relevant and comparable results.

Concentrations in deposition

In the scope of the yearly environmental monitoring programmes, the SNSA has organized simple intercomparison measurements for radioactive deposition due to precipitation at the sampling point Ljubljana. The measurements show that ZVD has consistently measured order of magnitude higher concentrations for ^{137}Cs then IJS (Figure 5). Even though the measured values are very low, thus higher variability is expected, the sampling and measuring procedures should be re-evaluated by both organisation to find the cause of this discrepancy.

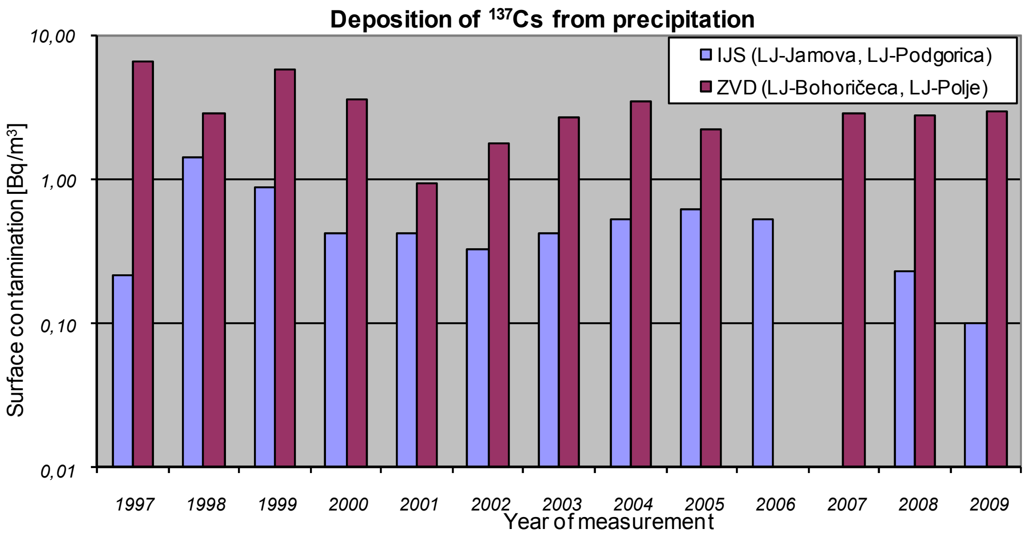

Figure 5. Activity concentration of ^{137}Cs in precipitation. Measurements done by ZVD show consistently higher values then IJS. There were no intercomparisons in 2006 and 2007.

Conclusions

The examples that are presented in this paper are only a fraction of the overall scope of the results of environmental monitoring, not to mention operational monitorings of different nuclear facilities. It is important to stress that there was no intention to qualify measurements as “right” or “wrong”. In most described cases it would even be impossible, since measurements are done at different locations. The intention was to discuss different influences on monitoring results.

Measurements of concentration of nuclides in air and soil show that location change leads to difficulties in data evaluation. Measurements of concentrations in water and precipitation, on the other hand, show discrepancies that may be due to transport, different sampling and preparation procedures or some, for now, unexplained reason. It is important to observe that the differences between measurements (50% and more) are much larger than the uncertainties (with the exception of precipitation, in the 10% range).

It is very important, that the contractors continuously analyse and evaluate measured results and compare them to expected values and historical trends. Whenever they find an inconsistency, it is imperative to try to find the reasons for it before reporting the measurements to the consignee.

It would be advantageous for the community if there would be more communication and cooperation between different organizations carrying out monitoring programmes. Due to the nature of the tendering process, it is possible that different organizations switch monitoring programmes from year to year. Especially in that case, it is important that the organizations share their experience and information so the continuity of measurement is preserved.

One of the more important facts that the contractors need to keep in mind is that the Slovenian data can not differ significantly from data of other, geographically similar regions or countries, so this can be used as one of the filters for data when it is needed.

ROKO database, maintained by the SNSA, can be one of the tools for all support organizations for evaluation and comparison of data. It is very important to carefully check all data before entering them into the database, since it is much harder to find mistakes later. Having in mind the number of measurements performed each year through and the number of values (roughly 10000) that are yearly entered into the database, it is a very demanding task.

All organizations performing environmental monitoring of radioactivity in Slovenia are accredited according to ISO17025 standard. This is an important step towards ensuring the uniformity of measurements, but it turns out that there are even more elements that influence results that are not included in the accreditation process.

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Problems connected to measuring a valid Peak-to-valley ratio in field gamma spectrometry

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Abstract

During in-situ gamma spectrometry, the ratio between count rates of the full-energy peak and the count rates of an energy region just below the peak, can be used as an indicator of depth of the source below ground level. The peak-to-valley ratio decreases with increasing depth of the source due to forward scattered Compton photons producing counts in the valley. The concept Peak-to-valley ratio used for environmental radioactivity measurements, is connected to a lot of difficulties regarding equipment and measurement geometry. There is great need to keep the ratio as free from disturbance as possible, this because the ratio is always suffering from bad statistics. Tests how changes in measurement geometry changes the ratio in laboratory environment was performed to better understand what to be accounted for in a normal measurement situation. Tests of how dead time might influence the ptv-ratio was conducted with good results for two specific preamplifier types, the resistor feedback and the transistor reset type. The transistor reset preamplifier was showing stability of the ratio for the different settings of the electronics and preserving the ratio to a constant value.

Interception of wet deposition of radiocaesium and radiostrontium by *Brássica napus*

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Abstract
In this study we examined the effects of ^{134}Cs and ^{85}Sr in the form of wet deposition on spring oilseed rape (*Brássica napus* L. 'Joplin'). Simulated rainfall containing the radioisotopes ^{134}Cs and ^{85}Sr was applied to the crop on six different occasions during the season. The interception fraction was found to be positively correlated to the leaf area index, with an *r*-value of 0.75 for ^{134}Cs and 0.77 for ^{85}Sr and to crop biomass, with an *r*-value of 0.80 for ^{134}Cs and 0.77 for ^{85}Sr.

Introduction
It is important to investigate the direct uptake of ^{134}Cs and ^{85}Sr during the first year after wet deposition, when the rate of translocation of these radioisotopes to edible plant parts is high compared with the rate of transfer from soil to plant (Madoz-Escande *et al.*, 2004).

The level of radionuclide capture, or interception, by plant parts is dependent on climate conditions, physico-chemical form of the radionuclides, plant morphology and biomass density (Vandecasteele *et al.*, 2001; Rosén & Eriksson, 2008). The proportion of precipitation that can be hold by the plant declines over time, but after the maximum water retention capacity is reached, the concentration of radioactive particles may continue to increase due to accumulation depending on their physico-chemical properties (Kinnersley *et al.*, 1997). The time from deposition to harvest also has an effect on the total uptake of radio isotopes in plants. This effect, referred to as 'field losses' has been described by Chadwick and Chamberlain (1970).

In the present study, wet deposition of ^{134}Cs and ^{85}Sr were applied to spring oilseed rape at different growth stages in order to study interception of ^{134}Cs and ^{85}Sr of different development stages of the crop and of plant growth. The crop used was spring oilseed rape (*Brássica napus* L.) and the contaminating isotopes that were used, ^{134}Cs and ^{85}Sr, were deposited using simulated rainfall. The study was conducted in the field, on agricultural land under normal cultivation treatments. Biomass samples were dried and activity measured by HPGe detector.

Materials and methods

Cultivation of rape

Spring oilseed rape (*Brássica napus* L. 'Joplin') was cultivated on a clay soil (situated at Vipängen, Uppsala, Sweden), with fertilizer supplied at sowing. The field trial consisted of 81 plots (1 m × 1 m) and radionuclides were applied of six plant growth steps.

Rain simulation

A rain simulator was constructed at the Department of Soil and Environment based on a drip infiltrometer described by Joel and Messing (2001). The water drops hit the crop at a 90° angle and an intensity of 41.6 mm h^{-1}. The target amount of precipitation was 1 mm m^{-2}. The precipitation consisted of water that was purified and deionised before each deposition event. The appropriate volume of stock solution was added to obtain the desired concentration of radionuclides (20 kBq m^{-2}) for both isotopes. The actual content of the radionuclides in the precipitation was measured using an HPGe detector (GMX-13200) after application to the plants.

Radioisotopes

The radioisotopes used were supplied by Canberra Solution AB on 20 May 2009: ^{85}Sr in the form of strontium chlorine in a 0.5 M HCl solution at a volume of 2 mL and an concentration of 1.84 MBq g^{-1}; and ^{134}Cs in a 0.1 M HCl solution at a volume of 1 mL and an concentration of 4.991 MBq g^{-1}.

Wet deposition and measurements

The deposition time was equivalent to application of 1 mm of precipitation to the crops. Each plot had three replicates for each treatment and these plots were randomised and plant materials were taken on six different plant growth stages. The plant materials were weighed, dried, re-weighed and milled before measurement of radioactivity. Measurements were made using three HPGe detectors that had an efficiency of 14.2% (GMX-13200), 31.3% (GMX-33210) and 20.5% (GEM-20200) at 1.33 MeV ^{60}Co. The leaf area index (LAI) was measured on the day of sampling using a LAI 2000 device (LI-COR, Nebraska, USA). Statistical analyses were performed using the statistical computer programme Minitab 15.1.30.0 (Minitab Inc.).

Interception fraction of ^{134}Cs and ^{85}Sr

The percentage of radionuclides intercepted after a wet deposition event was determined by calculating the interception fraction (*f*) as the ratio between the activity in the dry weight biomass of plants directly after deposition had occurred (A_i, Bq m^{-2}) and the total amount of activity deposited (A_t, Bq m^{-2}) and multiplying by 100 to get percentage. Modified after Proehl (2009):

$$f = \frac{A_i}{A_t} \times 100 \qquad (1)$$

Results

Leaf area index and biomass

The leaf area index increased with growth of the spring oilseed rape up to a maximum at flowering stage and then decreased until harvest. Plant biomass of the spring rape reached its maximum at the stage at development of fruit and then declined until the senescence stage.

Interception fraction of ^{134}Cs and ^{85}Sr

Interception fraction of ^{134}Cs and ^{85}Sr was calculated for samples taken on the next day after deposition, or two days after in the case of growth stage leaf development (first leaf unfold) due to weather conditions (Table 1). The interception fractions of ^{134}Cs and ^{85}Sr were not statistically different (paired *t*-test *p*-value of 0.132). The interception fraction of ^{134}Cs and ^{85}Sr increased over time in line with growth of the spring oilseed rape. The interception fraction was positively correlated to the leaf area index of the spring oilseed rape (Fig. 1) and as well to crop biomass (Fig. 2).

Table 1. Interception fraction of ^{134}Cs and ^{85}Sr deposition for spring rape (mean ± S.D.)

Growth stage	Biomass (g dry wt. m^{-2})	LAI*	Total amount deposited (A_t)		Interception on spring rape *f* ** (A_i)			
			^{134}Cs (kBq m^{-2})	^{85}Sr (kBq m^{-2})	^{134}Cs (kBq m^{-2})	^{85}Sr (kBq m^{-2})	^{134}Cs (%)	^{85}Sr (%)
Leaf development (first leaf unfold)	5	0.0	16.2 ± 0.13	14.8 ± 0.1	0.1 ± 0.02	0.1 ± 0.008	0.3	0.5
Leaf development (5 leaves unfold)	42	1.8	19.2 ± 4.29	17.7 ± 3.9	0.4 ± 0.07	0.4 ± 0.099	2.0	2.5
Full flowering	354	4.1	17.0 ± 0.24	16.7 ± 0.2	2.8 ± 1.93	2.7 ± 1.821	16.3	16.2
Development of fruit	667	4.1	17.3 ± 0.78	16.1 ± 0.7	4.5 ± 0.22	4.7 ± 0.531	17.6	29.4
Beginning of ripening	892	2.8	16.9 ± 0.32	15.3 ± 0.3	4.5 ± 0.18	4.7 ± 0.521	26.6	30.5
Senescence	648	1.7	16.5 ± 0.06	15.1 ± 0.1	1.7 ± 0.20	2.1 ± 0.140	10.4	13.8

***leaf area index and **interception fraction**

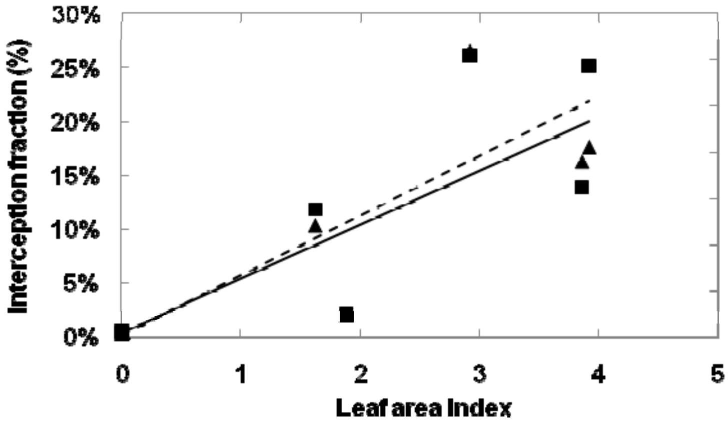

Fig. 1. Relationship between interception fraction and leaf area index of spring oilseed rape plants during the growing season. Mean values shown for ^{134}Cs (▲), ^{85}Sr (■), linear function of interception fraction for ^{134}Cs (—) and linear function of interception fraction for ^{85}Sr (– –).

The interception fraction was positively correlated to the leaf area index of the spring oilseed rape, with high interception of radioisotopes at high values of leaf area index. The data were fitted with a correlation coefficient had an *r*-value of 0.75 for ^{134}Cs and 0.77 for ^{85}Sr (Fig. 1). Plotting the interception fraction against biomass density (expressed as kg m^{-2}) showed that interception values reached highest at high density of the biomass. The correlation coefficient had an *r*-value of 0.77 for ^{134}Cs and 0.80 for ^{85}Sr (Fig. 2).

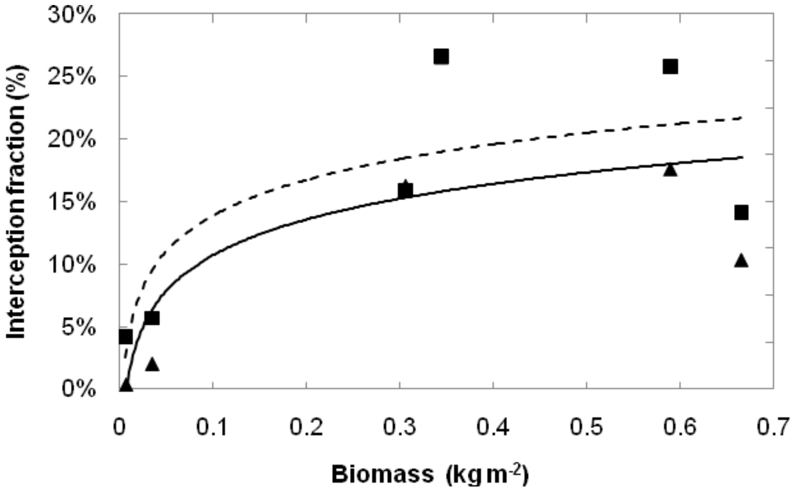

Fig. 2. Relationship between the interception fraction and dry biomass density (kg m^{-2}) of spring oilseed rape during the growing season. Mean values shown for ^{134}Cs (▲),^{85}Sr (■), logarithmic function of interception fraction for ^{134}Cs (–) and logarithmic function of interception fraction for ^{85}Sr (−−).

Discussion

The interception fraction, which was positively correlated to the leaf area index, followed the same pattern, increasing to a maximum value and then decreasing during ripening of the crop (Table 1, cf. columns LAI and *f*). A similar relationship between interception fraction and LAI, and between interception fraction and biomass, has also been observed for wet deposition on spring wheat (Vandecasteele *et al.*, 2001).

A higher value of interception fraction was observed for ^{85}Sr in the stage of development of fruit than for ^{134}Cs and as well in beginning of ripening. This can probably be explained by different physico-chemical form of ^{134}Cs and ^{85}Sr or the physiology of the epidermis. The interception fraction by spring oilseed rape was lower when compared to a study performed by Vandecasteele *et al.* (2001) of interception fraction by spring wheat. In the study the interception fraction was up to 84 % for ^{137}Cs and 88 % for ^{90}Sr.

Conclusions

This paper presents the results from the first year of a three-year study. The overall aim is to examine the effects of wet deposition of radioisotopes in ‘real’ conditions on oilseed crops and the levels of concentration of caesium and strontium in the crop during the season.

These preliminary results showed a positive correlation between the interception fraction for caesium and strontium radioisotopes and the biomass and leaf area index of the spring oilseed rape. There was also found that ^{134}Cs intercepted to a maximum level of 26.6 % and ^{85}Sr of 30.5 % both at growing stage beginning of ripening.

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Variation of dietary intake of radioactive cesium after the Chernobyl fallout in Finland

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Abstract

The deposition of radiocesium after the Chernobyl accident in 1986 was unevenly distributed in Finland. The variation of ingestion doses was studied in three areas, in central Finland with the ^{137}Cs deposition 50–80 kBq m^{-2} and in southern and northern parts of the country where the deposition was 1–5 kBq m^{-2}.

Estimates of dietary intake of radiocesium were made for years 1987, 1997 and 2007 by using the foodstuffs monitoring data and statistics of food consumption. The annual internal radiation doses in the three areas ranged in 1987 from 0.04 mSv a^{-1} to 0.38 mSv a^{-1} and decreased then ranging in 2007 from 0.006 to 0.09 mSv a^{-1}. The peak values of radiocesium in agricultural products were found in 1986–87 and the contents of ^{137}Cs decreased rapidly thereafter. The contribution of wild products to ^{137}Cs intake in 1987 varied from 32 to 56 percent. Due to the faster decline of ^{137}Cs in agricultural products the contribution of wild products to the intake increased, and in 2007 their contribution was 77–97 percent of the ingestion dose. Currently the variation in consumption and origin of the wild foodstuffs are the main contributors to the dose variation in Finland.

The ingestion doses calculated via data on foodstuffs were comparable to those received from the whole-body measurements or using data on mixed-diet measurements.

Introduction

The accident at the Chernobyl nuclear power plant in 1986 gave rise to an unevenly distributed ^{137}Cs deposition in Finland. In consequence, contamination of foodstuffs varied considerably in various parts of the country. In northern Finland, elevated ^{137}Cs contents in foodstuffs were found in reindeer meat and a slight increase also in freshwater fish and wild berries and mushrooms, but hardly any change in agricultural products. In central Finland with high ^{137}Cs deposition, higher levels of radiocesium were measured in all foodstuffs, especially in freshwater fish after the deposition. The variation of radiocesium levels in agricultural products produced in different parts of Finland was largest during the first years after the accident but later on the ^{137}Cs contents of agricultural products have been rather low and do not differ significantly in various parts of the country. In this work variation of ingestion dose due to radiocesium was assessed in 1987 and after that in ten year intervals, 1997 and 2007. The ingestion dose depends not only on the contamination level in the area but also on the consumption rates of different types of foodstuffs and the proportion of local foodstuffs in the diet.

Ingestion dose can be estimated by calculating the intakes of radiocesium through various dietary components, when information on the consumption rates and activity concentrations is available. Another way is to analyse whole representative mixed-diet samples instead of the main components of diet. Analysing mixed-diets gives the intake where the food consumption and processing are already included. Disadvantage of this method is that it gives no information on the contributions of different components to the intake. The third way is to estimate internal radiation dose by measuring the body content of ^{137}Cs in people with whole-body counters. In this work the results of these three different methods are compared.

Material and methods

The three studied areas, Helsinki, Tampere and Rovaniemi are situated in southern, central and northern parts of Finland (Fig. 1). The ^{137}Cs deposition in Tampere region after the Chernobyl accident varied from 50 to 80 kBq m^{-2}, and in Helsinki and Rovaniemi regions it was less than 5 kBq m^{-2}. The map (Fig.1), based on mobile radioactivity measurements of STUK, shows the distribution of ^{137}Cs deposition in Finland (Arvela et al. 1990). The foodstuffs data originates from the nationwide monitoring data of years 1987, 1997 and 2007 (Rantavaara 1991, Mustonen 1999, 2008, Rissanen 1997, Kostiainen 2007, 2008, Ylipieti 2007, 2008).

The ingredients of the diet included in the study were: vegetables, potato, fruit and berries, cereals, egg, meat (pork, beef, poultry, mutton), milk, fish, reindeer and game meat, wild berries and mushrooms. For the years 1997 and 2007 also data of the preceding and following years were used, if sufficient data for the years in question was not available. The production of vegetables and cereal grains is low in northern Finland, and therefore in our calculations we used ^{137}Cs concentrations of these products originating from the main production areas. The main local products of diet in Finland are freshwater fish and forest products: various game, moose, wild berries and mushrooms. For game the activity contents of moose were used, because moose meat covers over 70 percent of the game meat annually consumed. All the reindeer meat consumed in Finland originates from the reindeer herding area in northern Finland.

The food consumption data for 1987 is mostly based on the balance sheets for food commodities (Rantavaara 1987, 1991).The consumption data for 1997 and 2007 originates from national Findiet surveys (Finravinto 1997, Paturi et al 2008) and the information on consumption of domestic vegetables and cereals (Penttilä 2000). The national Findiet surveys are based on the results of the 48-hour dietary recall data. The same food consumption rates were used for all the areas. Dietary habits and food choices vary in different parts of the country, and also by gender and age. The consumption rates used in this study are averages for the Finnish adult population, 25- to 64-year-old men and women. The same consumption rates of wild berries and mushrooms were used for all the years in spite of possible variation during the years (Markkula 1997). The statistics of game bags were used for reindeer and game consumption (Saarni et al 2005, Annual Game Bag Statistics 2008). The fish consumption data for 1997 and 2007 originates from statistics of Finnish Game and Fisheries Research Institute and data for 1987 is based on statistics of fish catches.

In calculations of effective doses the average ^{137}Cs concentrations of foodstuffs for each year were used, and the delays in consumption were not taken into account although cereals, potato, root vegetables, cabbages, fruit, berries and mushrooms are harvested in autumn, and consumed until the next harvest. The delays in cereal consumption are even longer. Also the effect of food processing on the intake was neglected. It was assumed that all the foodstuffs consumed were domestic. The consumption of e.g. domestic fruit comprises only a minor part of fruit consumption, and the consumption rate for fruit includes only the domestic fruit.

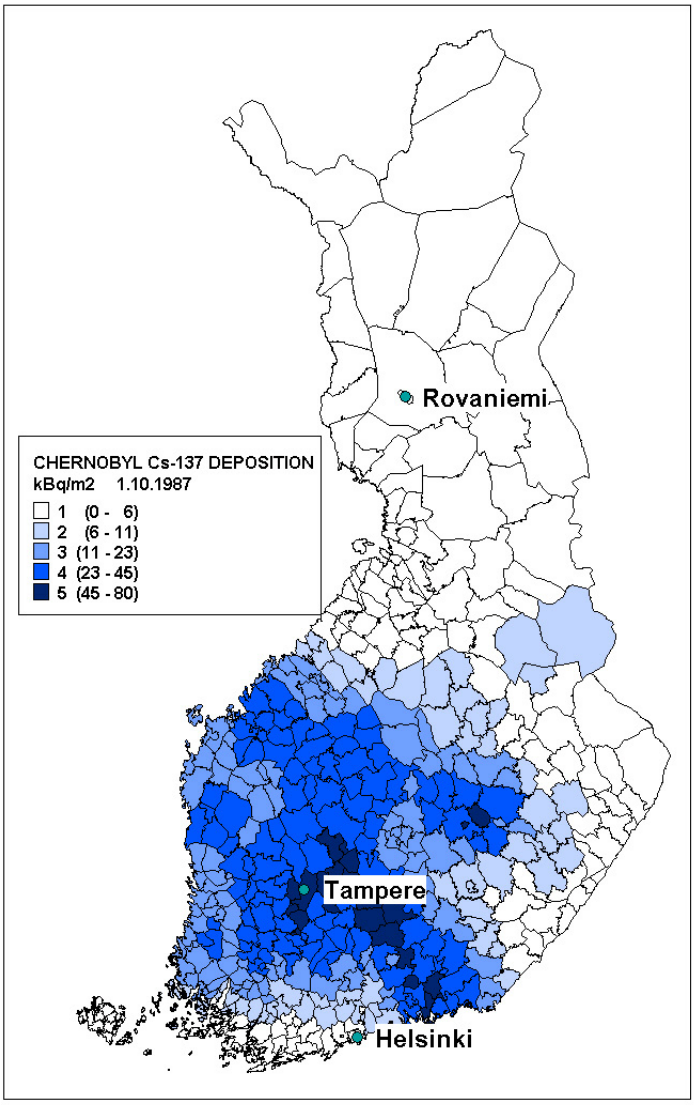

Fig. 1. Distribution of ^{137}Cs deposition kBq m^{-2} in Finland, reference date October 1, 1987. Study sites: Helsinki, Rovaniemi and Tampere.

Results and discussion

The consumption rates of foodstuffs are given in Table 1. The consumption rates for 1987 are higher than those for the years 1997 and 2007 due to differences of the methods used in collecting the dietary data. In the consumption of wild foodstuffs, especially of wild berries and mushrooms, there is a lot of variation from season to season caused by the availability of the natural products, which depends on the weather conditions. The mean consumption of mushrooms varies between 0.5 to 2 kg a^{-1}.

Table 1. Annual mean consumption of foodstuffs (kg/year) for Finnish adult population in 1987, 1997 and 2007.

Foodstuff	1987	1997	2007
Milk & milk products	262	157	145
Eggs	11.0	6.9	5.8
Beef	16.4	8.4	7.8
Pork	19.0	12.0	9.5
Poultry	4.0	6.0	10.0
Mutton	0.2	0.3	0.3
Wheat	44.2	33.8	29.6
Rye	19.0	22.3	19.0
Oats	3.0	9.0	3.0
Barley	1.6	0.4	0.4
Potato	65.7	40.9	31.0
Leafy vegetables	14.6	13.0	5.2
Fruit vegetables	16.1	20.0	23.3
Root vegetables	18.3	8.8	11.8
Garden berries	15.7	12.2	12.1
Domestic fruit	5.1	0.5	0.7
Baltic herring	1.7	0.8	0.4
Freshwater fish	4.1	3.7	3.5
Game	1.7	1.1	2.5
Reindeer	0.8	0.4	0.4
Wild berries	8.3	8.3	8.3
Wild mushrooms	1.3	1.5	1.5
Drinking water	730	730	730

The average ^{137}Cs concentrations for the areas used in dose calculations are given in Table 2. The sampling density and the sampling sites remained not the same during the years for all foodstuffs. This increases variation in ^{137}Cs concentrations, and the data in Table 2 shows the level of ^{137}Cs contents of foodstuffs in each area based on sample measurements.

The concentrations of ^{137}Cs in agricultural products were highest in 1986-87 and have decreased since then rapidly. Since 1997 the average ^{137}Cs contents of the agricultural products in all the areas were less than 5 Bq kg-1, whereas those of natural products were remarkably higher, ranging up to some hundreds of becquerels per kilogram. The dose conversion factors used were those reported by ICRP (1996).

Table 2. Average ^{137}Cs concentrations of foodstuffs (Bq kg^{-1}) in Helsinki, Rovaniemi and Tampere areas in 1987, 1997 and 2007.

Foodstuff	Helsinki			Rovaniemi			Tampere		
	1987	1997	2007	1987	1997	2007	1987	1997	2007
Milk	13	0.5	0.44	2.7	0.6	0.39	43	1.9	0.9
Eggs	0.5	0.13	0.02	0.5	0.13	0.02	0.5	0.16	0.02
Beef	55	3.1	2	12	2.82	2	150	3.1	2
Pork	12	1.5	0.9	9.37	1.5	0.9	13	1.5	0.9
Poultry	5.2	0.5	0.5	5.2	0.5	0.5	5.2	0.5	0.3
Mutton	65	5	5	65	5	5	251	30	15
Wheat	0.15	0.11	0.14	0.15	0.11	0.14	0.63	0.32	0.14
Rye	2.17	0.54	0.42	4.88	0.2	0.42	5.29	0.65	0.42
Oats	0.43	0.43	0.56	5.21	0.69	0.56	2.76	2.18	0.56
Barley	1.08	0.16	0.21	0.86	0.51	0.21	1.13	0.4	0.21
Potato	1.1	0.1	0.02	0.3	0.1	0.02	2.8	0.38	0.2
Leafy vegetables	0.55	0.43	0.1	0.55	0.43	0.1	2.0	0.45	0.2
Fruit vegetables	1	0.02	0.1	1	0.02	0.1	16	0.47	0.1
Root vegetables	1	0.23	0.1	1	0.23	0.1	0.21	0.35	0.1
Garden berries	0.89	0.36	0.02	0.89	0.07	0.02	5.22	1.11	0.8
Domestic fruit	0.51	0.023	0.01	0.506	0.0227	0.01	2.15	0.27	0.01
Baltic herring	34.0	13.7	6.8	34.0	13.7	6.8	34.0	13.7	6.8
Freshwater fish	214	101	42.0	95.0	38.0	20.0	2359	763	964
Moose	156	74	71	45	36	8	228	352	249
Reindeer	600	200	100	600	200	100	600	200	100
Wild berries	70	36	16	40	15	10	232	264	220
Wild mushrooms	215	107	91	125	100	50	1347	828	554
Drinking water	0.66	0.038	0.021	0.0005	0.0005	0.0003	0.190	0.011	0.005

The annual internal radiation doses for the areas ranged in 1987 from 0.04 to 0.38 mSv a^{-1} and decreased then ranging in 2007 from 0.006 to 0.090 mSv a^{-1} (Fig. 2.).

The contribution of wild products (reindeer, game, fish, berries, mushrooms) to the total internal dose in 1987 was 32 percent in Helsinki area, 56 percent in Rovaniemi and 49 percent in Tampere. By 2007, the contribution of natural products to the dose increased to 77 percent in Rovaniemi, 84 in Helsinki and 97 percent in Tampere. Natural products in the diet only account about 5 percent of the total consumption but their relative contribution to the dose is larger depending on the time after the deposition. In 1987, the contribution of drinking water to the dose in Helsinki area was 6 percent, and decreased by 2007 to 2 percent. In Rovaniemi and Tampere the doses received via drinking water were less than 0.5 percent of the ingestion doses. The raw water in Helsinki is surface water, in Tampere both surface and ground water, and in Rovaniemi all the raw water is ground water which is well protected against fallout radioactivity (Mustonen 2000).

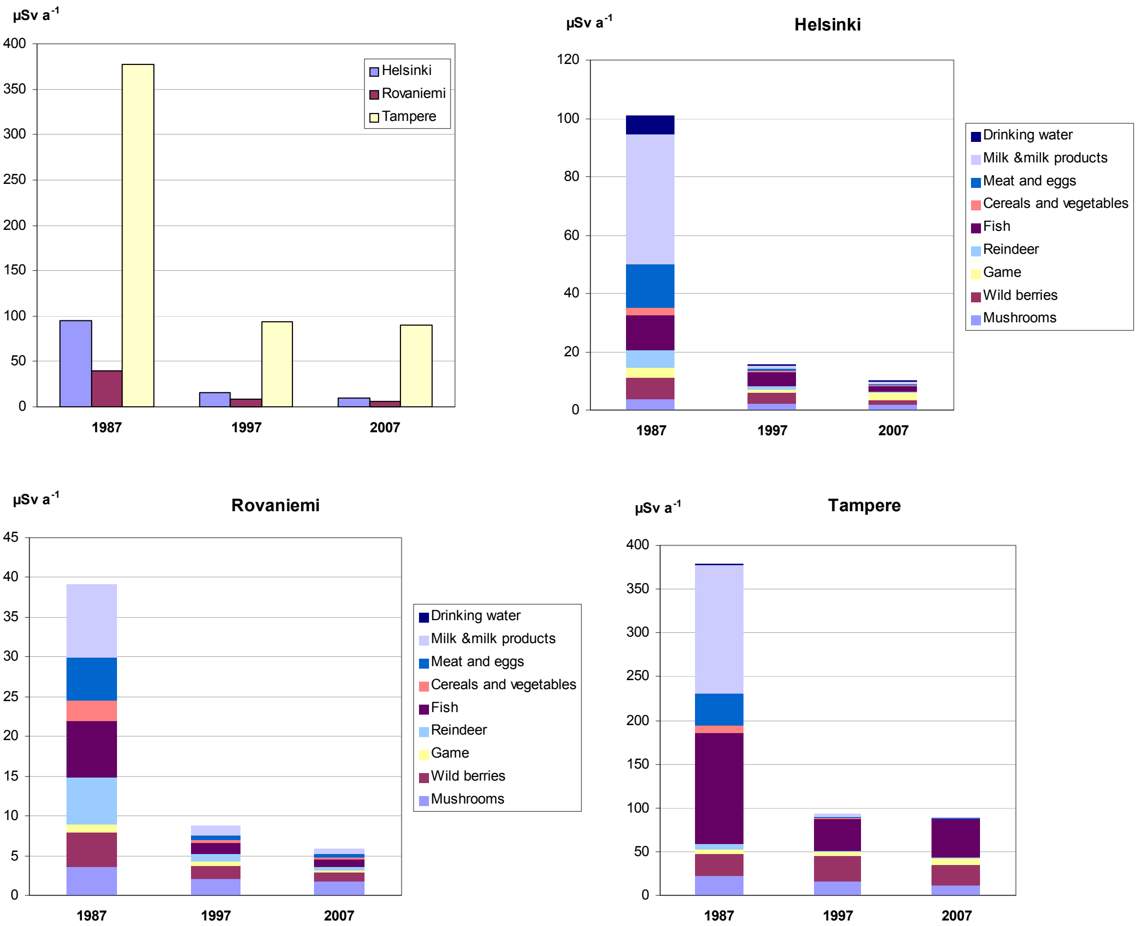

Fig. 2. Annual internal doses μSv a^{-1} in Helsinki, Rovaniemi and Tampere regions and the distribution of the doses by foodstuff groups in 1987, 1997 and 2007.

Discussion

The differences in the radiation doses in the three studied areas reflect the ^{137}Cs deposition levels in the areas. The ^{137}Cs deposition in Tampere region was more than ten times higher than in Helsinki or Rovaniemi, and this same ratio was seen in the internal doses calculated by using the foodstuffs monitoring data from these areas. The same consumption rates were used in all the areas although it is known there are some differences in consumption rates of northern and southern parts of the country. For example, the consumption of vegetables, fruit and poultry is lower in the northern parts of Finland compared to the central or southern parts of the country (Helakorpi 2002). Also the consumption of cereals and wild berries is highest in northern Finland. The total consumption of foodstuffs for 1987 was about 50 percent higher than that for 2007 due to different origin of consumption data in different years. If consumption data of 2007 were to be used for all the years, the dose due to the agricultural products in 1987 would decrease about 50 percent, but the dose from natural products would remain almost the same.

The average annual doses from ingestion of ^{137}Cs in Finland have been assessed to be about 0.15 mSv a^{-1} in 1987 and about 0.03 mSv a^{-1} in 1997 (Rantavaara 2008). The results of this study indicate that the average doses received by people living in the most contaminated area were about three times higher in 1987.

The surveillance program on environmental radioactivity run by STUK since 1999 includes regular monitoring of mixed diet samples in Helsinki, Tampere and Rovaniemi (Mustonen 2000, 2008). This monitoring program typifies the food prepared in institutional kitchens, and the level of radioactivity it contains. The deficiencies in this program are minor amounts of natural products included in the diets of institutional kitchens and the contingency of the diet as it is based only on two sampling days. The annual ingestion doses calculated via these mixed diet samples were 6.8–7.6 µSv a^{-1} in 1999 and 3.1–6.7 µSv a^{-1} in 2007. In 1996 the mixed-diet sample meals were studied by analysing the diets of six weeks, in which case elevated ^{137}Cs concentrations were seen in the meals that included wild products. (Rantavaara 2002). The annual dose calculated from the results of this six week study was 8.7 µSv a^{-1} in 1996 in Helsinki area.

The internal annual doses calculated via the results of whole-body measurements were 0.074 mSv in 1987, 0.011 in 1997 and 0.008 in 2007 in Helsinki area. The doses calculated via mixed-diet in Helsinki were lower, about 60 percent of the doses received via the whole-body measurements. This difference may reflect the absence of wild products in the meals of institutional kitchens. The whole-body measurements for a group of people living in central Finland near Tampere area were 0.42, 0.12 and 0.09 mSv a^{-1} in 1987, 1997 and 2007, respectively. The people in this group consume a lot of natural products like freshwater fish, wild berries, mushrooms and moose meat. The doses calculated from the whole-body measurements are at the same level as the doses calculated via foodstuffs for Tampere area with average consumption rates of natural foodstuffs. We have assumed in our calculations that all the food consumed is produced in the area, although in practice food products are distributed all over the country and only natural products are consumed locally.

Table 3. Internal doses received by the three different methods.

Site	Internal dose, mSv a^{-1}, in 1987; 1997; 2007		
	Foodstuffs data	**Mixed diet data**	**Whole-body measurements data**
Helsinki	0.10; 0.016;0.010	–[1]; 0.009[2]; 0.005	0.074; 0.011; 0.008
Rovaniemi	0.039; 0.009; 0.0060	–; –; 0.003	-
Tampere	0.38; 0.093; 0.090	–; –; 0.007	0.42; 0.20; 0.09 [3]

[1] No data; [2] Data of 1996; [3] Data from group of people that consume natural products in large quantities

The results of this study were calculated assuming average consumption habits. However, consumption of natural products that contributes most to the ingestion dose in later years after the deposition varies greatly from one individual to another. Among the critical group that consume natural products in large quantities, the doses can be more than tenfold compared with the average consumer.

The internal doses assessed here do not include the dose increase due to ^{134}Cs, which was in 1987 about 40 percent. By 1997, the dose from ^{134}Cs decreased being less than two percent compared to the dose from ^{137}Cs.

Conclusions

The different ways to assess internal doses due to ^{137}Cs in foodstuffs give similar results. When calculating the doses via foodstuffs, the consumption data used affects greatly the results. However, it gives information on the dose distribution by different foodstuff groups which is not extracted from diet measurements or whole-body measurements.

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Investigation of ^{137}Cs redistribution within urban ecosystem

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Abstract

Contamination of puddle sediments by ^{137}Cs within the urban ecosystem is studied. The investigation is made on the example of Ekaterinburg city, Middle Ural region of Russia. Contamination density of ^{137}Cs in the region due to fallouts after atmospheric testing of nuclear weapons and nuclear accidents is assessed about 5.1 kBq/m^2, it associates with maximum activity concentration of the upper 15 cm layer bellow 30 Bq/kg. Results of the survey indicate a mean value of ^{137}Cs concentration in puddle sediments of 70 Bq/kg, with a maximum value of 540 Bq/kg. It is estimated that horizontal migration has led to about fourfold concentration of ^{137}Cs in puddle sediment.

Introduction

During the period of atmospheric testing of nuclear weapons and accidents at nuclear facilities the primary attention of researchers was drawn to study radionuclide redistribution patterns in natural ecosystems. The urban ecosystem was not considered as the main object of the research. The aim of the study is to obtain data on levels of accumulation and redistribution of ^{137}Cs in the city.

The basic hypothesis for the study was the significant redistribution of ^{137}Cs in the urban environment. Local surface depressed zones where rain water and sediment accumulate in puddles are considered as such deposits. ^{137}Cs precipitates at the puddle catchments area, which include surrounding soil and ground and roofs of buildings. Radiocaesium is supposed to reach the puddles along with horizontal migration, in soluble (precipitations) and insoluble form (particles of soil and other material such as silt, peat, decomposed plants, domestic and construction wastes on which ^{137}Cs has adsorbed). In the urban environment routine measures such as cleaning the territory, grass mowing, garbage disposal, land fill, planting and uprooting, adjusting the channels and drainage channels contribute to ^{137}Cs redistribution as well.

Measurements of activity concentration of ^{137}Cs at puddle sediments were conducted in Ekaterinburg city, Russia. ^{137}Cs contaminated the urban ecosystem under study due to the global fallout after atmospheric nuclear weapon's testing and the Chernobyl accident. The study includes the analysis of archive data on ^{137}Cs contamination in Ural region, sampling of puddle sediments in Ekaterinburg city, direct measurements of ^{137}Cs activity concentrations in the samples and data analysis.

Material and methods

Appropriate depressed zones in the city were selected by visual inspection. Samples were taken at the depressed zones that seemed as the eldest and relatively undisturbed for past years. The samples were taken from the upper 5-cm layer, with a sample mass of about 1.5 kg of dry weight. During the sampling process, domestic wastes and grass were removed. Information on location and site-specific description and photo documentation of puddle and environment were performed. The position of the sampling areas was assigned according to its address and was marked on the city map. The sampling of puddle sediments was carried out during the field study seasons 2007-2009 in 240 locations in 15 districts of the city.

The samples were air-dried under the indoor temperature in the laboratory during one month at least. Dried samples were milled and homogenized. ^{137}Cs activity concentrations were measured using stationary gamma-spectrometer equipped with NaI(Tl) scintillation detector (crystal size 63 x 63 mm.). Counting time was 3 hours providing detection limit 5 Bq/kg.

Results and Discussion

Available archive data on ^{137}Cs contamination were considered as follows:

- results of direct measurements of ^{137}Cs concentration in fallout by The Federal Service for Hydrometeorology and Environmental Monitoring over the period 1994-2006 presented in the annual reports,
- results of grass contamination measurements over the period 1966-1987 obtained from reports of local agricultural service.

In addition levels of annual deposition of ^{137}Cs produced in atmospheric nuclear testing over the period 1945-1985 in the northern hemisphere summarized by UNSCEAR (UNSCEAR, 2000) were used.

Analysis of these data allowed reconstructing the ^{137}Cs contamination in Middle Ural region retrospectively. According to this reconstruction:

- total deposition density of ^{137}Cs in soils reaches 5.1 kBq/m^2 in 2007 with regard to decay;
- over the 20 years after the Chernobyl accident, the ^{137}Cs activity has increased approximately by 1 kBq/m^2;
- the values of the ^{137}Cs concentration in the 15-cm undisturbed ground in 2007 in dependence on the year of surface formation were modelled under conditions of absence of horizontal migration and suggested median rate of vertical migration <0.5 cm/year (Schuller, 1997; Arapis, Karandinos, 2004);
- the Chernobyl accident contributed up to 30% to the ^{137}Cs activity concentration in the 15-cm undisturbed ground of Middle Ural region. While the ^{137}Cs vertical migration rate <0.5 cm/year, the 15-cm layer accumulates more than 70% of total nuclide activity in 2007. The ground surface, which has been formed in the years just before the Chernobyl accident and undisturbed later, are contaminated by ^{137}Cs in the range of 20-30 Bq/kg, the landscape formed in the period of intensive nuclear weapon testing (1950s and 1960s) possesses higher ^{137}Cs concentrations,

up to 50-60 Bq/kg. The estimates of contamination obtained by modelling are in agreement with the results of numerous measurements of the upper ground horizon in Ekaterinburg city. ^{137}Cs surface contamination level of soils below 30 Bq/kg is typical for the city. That level corresponds to the ground formation in period 1970-2000, which is most likely according to direct observations as well.

Fig. 1 presents distribution of ^{137}Cs activity concentrations in samples of puddle sediments obtained for Ekaterinburg city. The arithmetic mean of ^{137}Cs concentration is 70 Bq/kg, the maximum value of the ^{137}Cs concentration is 540 Bq/kg and the minimum value is below the detection limit. The geometric mean of the ^{137}Cs concentration is 57 Bq/kg. The concentration of ^{137}Cs exceeds 100 Bq/kg in about 15 % of samples. According to chi-square test, ^{137}Cs concentrations distribution deviates from lognormal ($\chi 2=34.80$, df=7, $p<0.01$).

The heterogeneity of data on ^{137}Cs concentrations was supposed to be associated with features of spatial distribution of contamination. Results of gamma spectrometry measurements of ^{137}Cs concentration in puddle sediments by the city districts are presented in Table 1.

Process of puddle formation is considered to be similar to forming large water reservoir. Under gravitation transport of water the solid particles are scavenged and transferred to surface depressed zone forming the puddle sediments. ^{137}Cs concentration in the puddle sediment is supposed to consist of the concentration transferred with solid particles from the surface ground (on which the adsorption in the catchments area surface during the water accumulation in puddle took place) and concentration of water-soluble fraction of ^{137}Cs in atmospheric precipitation.

The typical activity concentration in the ground's upper layer, C_{soil}, for the case under consideration is about 20 Bq/kg, it corresponds to the forming of the catchment area in 1987. The ^{137}Cs activity concentration of puddle sediments, $C_{sed.}$, is the sum of water-soluble activity concentration of atmospheric precipitation, C_{water}, and the activity concentration transferring into the puddle with solid particles from the ground surface, $C_{soil.}$ C_{water} is calculated as

$$C_{water} = I \cdot k \cdot \frac{S}{m}, \qquad (1)$$

where I is the ^{137}Cs total activity of atmospheric precipitation estimated as $I \approx 1000$ Bq/m^2 for period 20 years after the Chernobyl accident; k is the factor of collection of ^{137}Cs activity within puddle; S is the catchment area, m^2; m is the puddle sediment mass, kg. m/S is estimated as 2 kg/m^2 for the typical puddle existing for 20 years, i.e. a typical puddle catchment area is about 200 m^2 and puddle sediment mass is about 400 kg. Then $C_{sed.}$ is calculated as follows:

$$C_{sed.} = I \cdot k \cdot \frac{S}{m} + C_{soil}, \qquad (2)$$

Accepting $C_{sed.}=70$ Bq/kg and solving equation (2) relative to k we obtain a value $k=0.1$. Thus about 10% of the ^{137}Cs activity of precipitations are transferred to the puddle in water soluble phase. Water transfer forms about 70% of activity concentration in puddle sediments.

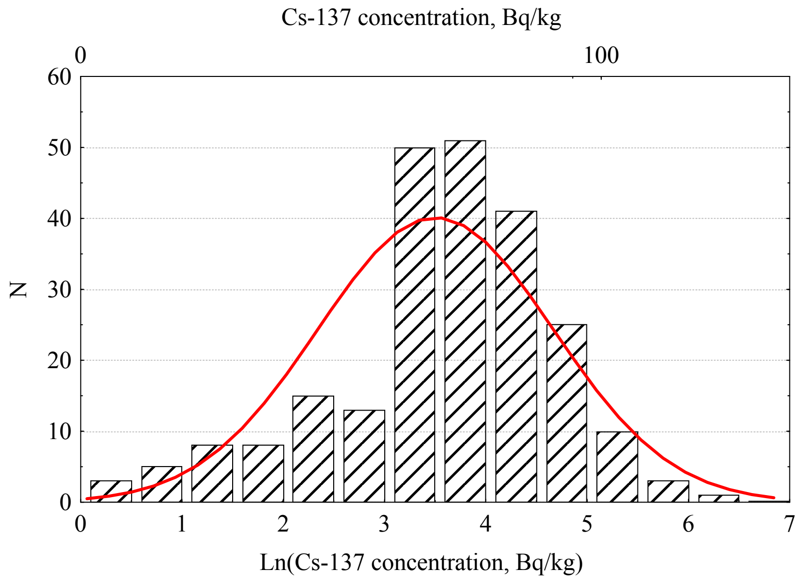

Fig. 1. Distribution of ^{137}Cs activity concentrations in samples of puddle sediments obtained for Ekaterinburg city.

Table 1. Arithmetic, geometric mean and portion of samples with ^{137}Cs concentration >100 Bq/kg by districts of Ekaterinburg city. Results based on gamma spectrometry measurements in samples of puddle sediments.

District	Arithmetic mean of ^{137}Cs concentration, Bq/kg	Geometric mean ^{137}Cs, concentration, Bq/kg	Portion of samples with ^{137}Cs concentration >100Bq/kg
West	63	30	0.16
Centre	59	41	0.19
North	58	40	0.14
North East Suburb	67	42	0.19
East Suburb	29	16	0.05
South	44	35	0.05
South West Suburb	64	36	0.22

Conclusions

The hypothesis that puddle sediments in local depressed zones of the urban landscape are the final deposits of ^{137}Cs has been proved. ^{137}Cs contamination levels in puddle sediments are relatively high in comparison with contamination of surrounding soils. The activity concentration of ^{137}Cs in 14 % of the samples exceeds 100 Bq/kg, with a maximum activity concentration of 540 Bq/kg. Such activity concentrations are comparable with the activity found in bottom sediments of water reservoirs of nuclear facilities in the region and lakes of the East Ural Radioactive Trace (Trapeznikov, 2007).

Characteristics of the urban environment and ground management can provide suitable conditions for local migration and concentration of radioactive contamination. The observed local surface migration of ^{137}Cs has led to an approximately fourfold concentration in puddle sediments in comparison with typical activity concentrations in surrounding soils. The study of ^{137}Cs migration in the urban environment should take into account the influence of anthropogenic processes on radionuclide redistribution.

Puddle sediments are supposed to be prospective object of environmental monitoring in the urban ecosystem. During such monitoring ^{137}Cs can be applied as a marker of puddle's age. Obtained characteristics of ^{137}Cs redistribution in the urban environment can be used in investigation of environmental pollution by other metals similar by their properties.

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Radioactivity in the environmental samples around the Cernavoda NPP

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Abstract

The Cernavoda Nuclear Power Plant is dedicated to produce electrical & thermal power in a safe and efficient manner for at least 30 years, from nuclear power using CANDU technology.

The factors presented below ensure that the public health and environment are adequately protected:

- source control;
- effluent control;
- effluent monitoring;
- environmental monitoring.

The Environmental Radioactivity Monitoring Program around Cernavoda NPP is designed to meet the following objectives under normal NPP operating conditions:

- to measure the radionuclide concentrations in environmental media;
- to provide an independent assessment of the effectiveness of the source control, effluent control and monitoring based on measurements in environment;
- to validate the models and parameters used in the calculation of the derived emission limits;
- to provide data to aid in the development and evaluation of models and methodologies which adequately describe the movement of the radionuclides through the environment.

The Environmental Control Laboratory of the Cernavoda Nuclear Power Plant, located in Cernavoda town, is equipped with modern analyzing systems to determine the natural and artificial radionuclide content in the following environmental samples from an area with 30 km radius around the NPP:

- airborne particles, iodine, aqueous vapours and deposition from air;
- surface water, deep ground water, infiltration water, drinking water;
- soil and grass;
- sediment and fish;
- milk, eggs and meat (chicken, pork and beef);
- vegetables, grain, corn and fruits.

The results of the monitoring program are annually compared with the results of The Preoperational Environmental Monitoring Program performed between 1984 and 1994. There were no environmental radioactivity modifications around the Cernavoda NPP comparing to the period before the NPP operation.

Introduction

The site of the first Romanian Nuclear Power Plant comprising five 700 MWe units was chosen in the Dobrogea area, near the Cernavoda - a town of approximately 20,000 inhabitants. The choice was based on the following main reasons: the Danube river-Black Sea Channel that represents an important source of cooling water, as it provides an easy access route for the heavy and beyond standard equipment, and the low seismicity of the region as compared to other regions of the country.

Cernavoda Nuclear Power Plant is dedicated to produce electrical and thermal energy in a safe and efficient manner for at least 30 years, using CANDU (Canadian Deuterium Uranium) nuclear technology.

Among the activities to be performed during the operational phase of Cernavoda NPP, the following factors are identified which ensure that the public health and the environment are adequately protected: a) source control; b) effluent control; c) effluent monitoring; d) environmental monitoring.

The purpose of the environmental monitoring program is to provide reliable and accurate data, which comprise statistically valid data sets per nuclide/ environmental media combination on an annual basis.

The monitoring program is designed to meet the following objectives under normal nuclear power plant operating conditions:

- to measure the radionuclide concentrations in environmental media;
- to provide an independent assessment of the effectiveness of the source control, effluent control and monitoring based on measurements in environment;
- to validate the models and parameters used in the calculation of the derived emission limits;
- to provide data to aid in the development and evaluation of models and methodologies which adequately describe the movement of the radionuclides through the environment.

Material and methods

The Environmental Control Laboratory of the Cernavoda Nuclear Power Plant, located in Cernavoda town, at 1.8 km far from the plant, is performing the Environmental Radioactivity Monitoring Program for Cernavoda NPP.

Indicator locations are outside the plant perimeter, and are established depending on emission type, critical groups and pathways used for DEL (Derived Emission Limit) calculation. In addition to these locations, a network of TLD locations was established around the plant beyond the exclusion zone.

Measurements of ambient background are conducted beyond the influence of station emissions. For emissions to air and direct exposure pathway (external irradiation and inhalation) one background location is provided. For water and sediment samples, one background location each is fixed.

The background levels were determined before of first operation of Unit 1 of the Cernavoda NPP, through the Preoperational Environmental Monitoring Program in period 1984 – 1996; all present results are compared to these levels to assess the influence of power plant operation on environment and public health.

The following environmental samples are included in the actual monitoring program, from an area with 30 km radius around the NPP:

- airborne particles, iodine, aqueous vapours and deposition from air;
- surface water, deep ground water, infiltration water, drinking water;
- soil and grass;
- sediment and fish;
- milk, eggs and meat (chicken, pork and beef);
- vegetables, grain, corn and fruits.

The frequency of monitoring or sampling is related to the mean lifetime of the nuclide in a pathway. In the same time for air monitoring, the frequency was established as a function of plant emissions (percentage of DEL/weeks).

Table 1. Determination of monitoring frequency.

Nuclide	Half life ($T_{1/2}$)	Environmental media	Mean residence time	Mean life time	Monitoring frequency
Tritium	12.3 years	air	minutes	minutes	continuous
		vegetables-fruits	3 months	3 months	quarterly
		milk	5 days	5 days	weekly
		water	1 day	1 day	daily

Airborne water vapours are collected by drawing air through molecular sieve. Sampling is continuous with an integration period of one month, during which 4 – 6 m^3 of air are passed through the collector. The absorbed water is removed from the sieve by heating at 350ºC. Water samples are prepared by distillation, free water from food samples (milk, fish, vegetables, fruits) is extracted by azeotropic distillation in toluene.

Tritium activity concentration is determined by liquid scintillation counting on sample prepared by mixing extracted water with a scintillation cocktail.

Table 2. Sample measurements: measurement device used and counting time for tritium.

Nr. Crt.	Sample type	Measurement device	Radionuclide assessed	Counting time
1.	Aqueous vapours in air	20 ml PE vial	Tritium	400 min
2.	Water – individual and bulked sample			
3.	Milk – individual			
4.	Deposition – bulked sample			
5.	Fish, meat, vegetable, fruit, grain			
6.	Soil and sediment			

The frequency of analysis is determined by the following elements:

(1) the minimum required sensitivity;
(2) the analytical sensitivity of the method;
(3) the number of results per nuclide/pathway combination per year required to generate a statistical valid data set;
(4) significance of plant radioactive effluent emissions.

The minimum required detectable specific activity should be such as to detect radionuclides present in the environment as a result of Cernavoda NPP's operations. The Minimum required specific activity was calculated for each specific radionuclide and environmental media with *the following relation:*

$$MDA = \frac{D_a}{C \times DCF}$$

where: MDA = Minimum Detectable Activity; D_a = Maximum additional annual whole body dose above background due to Cernavoda NPP' operations; C = Annual Intake Rate for analyzed media; DCF = Dose Conversion Factor.

Table 3. Measurement system for tritium.

Measurement system	Description	
Liquid scintillation analyzer	**Manufacturer**	• Wallac
	Type	• 1220 Quantulus Ultra Low Level
	Calibration procedure	• Internal departmental calibration procedures
	Maintenance procedure	• Maintenance and service contract
	Standards used	• Unquenched H-3, C-14 and background standards for energy calibration and weekly QA verification
		• Quenched H-3 and C-14 sets of standards for annually efficiency calibration

Results

Forteen years of experience in CANDU operation at Cernavoda NPP have shown that tritium is the most significant radionuclide released in gaseous and liquid effluents, mostly as tritiated water, which represents more than 80% of the total radioactivity released. For this reason the environmental monitoring program is heavily weighted toward measurement of tritium.

Air from the radiological area is conducted, after filtration, to the evacuation stack where it is measured for radioactive particulates, gases and tritium water vapours. The total radioactive gaseous effluent emissions are weekly compared with the administrative limits, which are below the approved DELs. An indicator air monitoring station for critical group (Cernavoda inhabitants) is located at 1.8 km (NV) far from the Cernavoda NPP, in the Cernavoda town (ADI-08) and the background station is located at about 30 km (N) far from the NPP (straight line) in Topalu village (ADB-01). Results of tritium measurements (Bq/m^3) in the water vapour samples from the two locations, in period 2006 – 2008, are represented in the Figure 1.

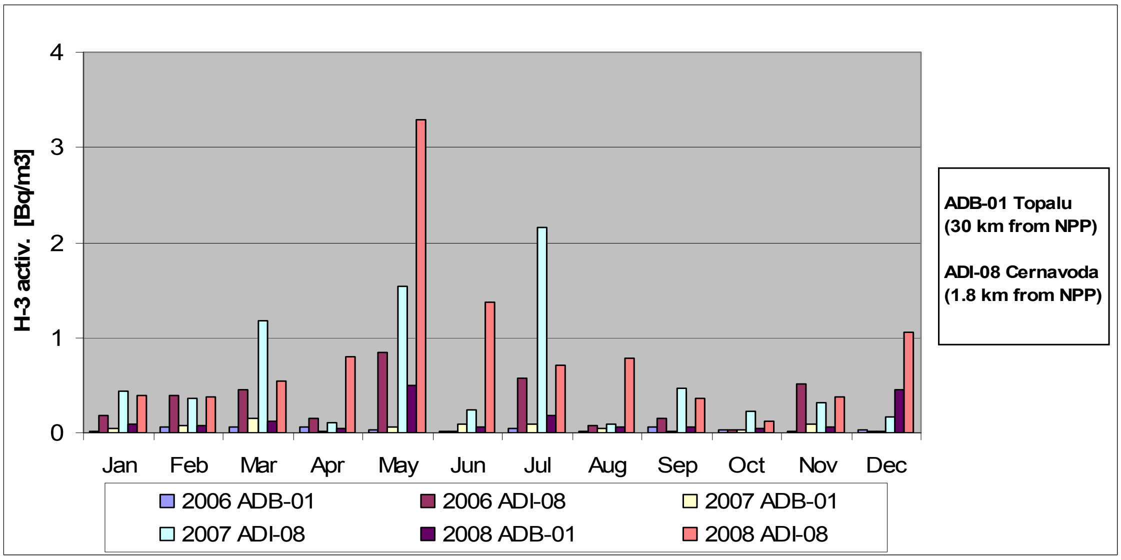

Figure 1. Tritium concentration activities in air, between 2006 and 2008, around Cernavoda NPP (2 monitoring stations situated at about 1.8 km and 30 km far from NPP).

Drinking water for the Cernavoda town is coming from the Danube River and after the specific treatments is distributed through the drinking water system. Samples are from the laboratory tap (AII-03, 1.8 km NV). Drinking water for Saligny is supply from underground water and samples are from a drilled well (SSS-03, 5 km SE). Results of tritium measurements (Bq/l) in the drinking water samples, in period 2007 – 2009, are represented in the Figure 2; the legal limit (100 Bq/l) is also represented.

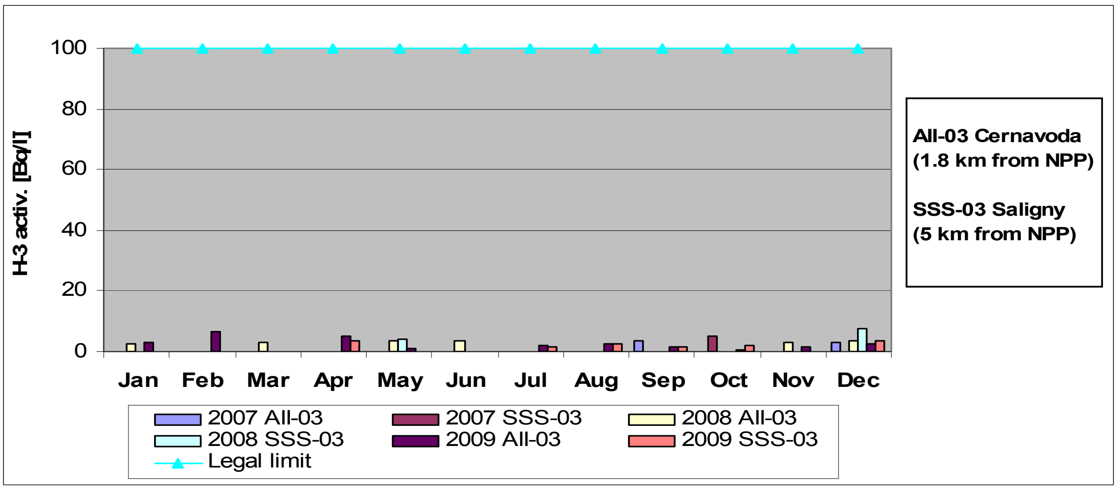

Figure 2. Tritium concentration activities in drinking water, between 2007 and 2009, around Cernavoda NPP (2 locations situated at about 1.8 km and 5 km far from NPP).

All food samples are cultivated, grown or produced in the monitoring area and representative for the local diet. Milk samples are provided by a dairy farm located in Seimeni village (AII-02), at about 8 km (NNE) far from the plant. Results of tritium measurements (Bq/l) in the milk samples, in period 2006 – 2008, and the mean value over this period are represented in the Figure 3.

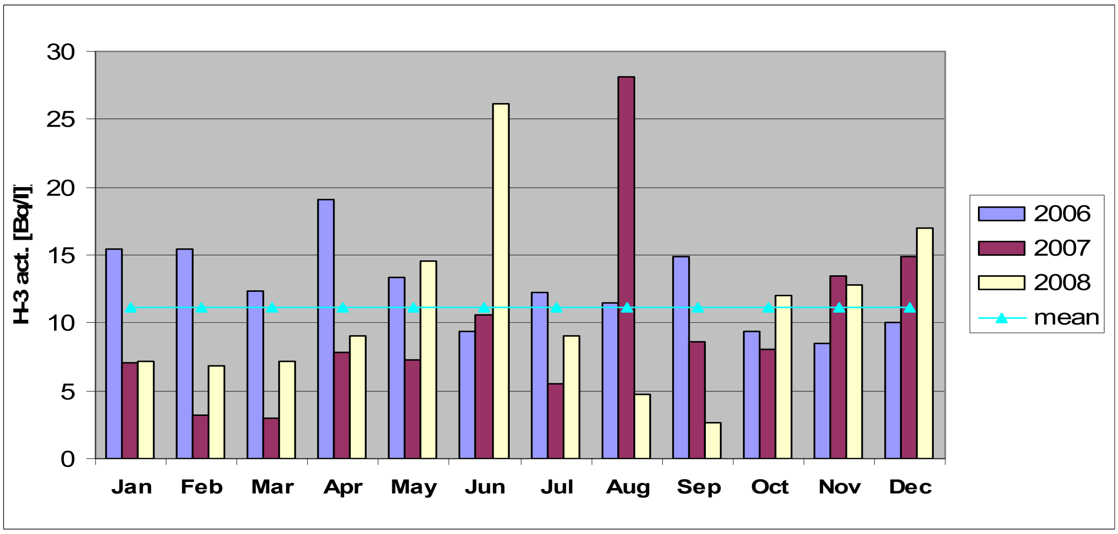

Figure 3. Tritium concentration activities in milk, between 2006 and 2008, around Cernavoda NPP.

Fish samples are analysed from Danube River – Cernavoda Harbour (indicator location LII-09, downstream of confluence of evacuation canal with Danube River) and Danube-Black Sea Channel – Cernavoda Harbour Lock (reference location LII-05, upstream of evacuation way) – for liquid emissions in the environment, and 3 pounds: Domneasca (AII-01, 8 km NNE), Baciu (SSS-14, 17 km SV) and Faclia (SSS-15, 8 km SE) – for gaseous emissions in the environment; the last two locations have been introduced in the monitoring program since 2008. Results of tritium measurements (Bq/kg) in the fish samples, in period 2006 – 2008, are represented in the Figure 4.

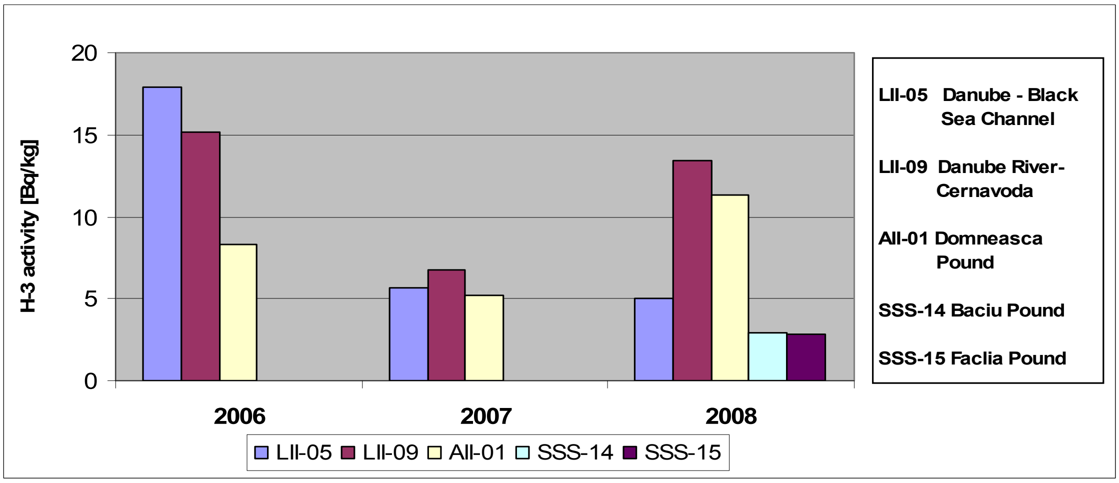

Figure 4. Tritium activity concentration in fish samples from the Cernavoda NPP monitoring area, between 2006 and 2008.

Discussion

Results of tritium measurements in water vapour samples, in period 2006 – 2008, for 2 locations (represented in the Figure 1), are relatively low. For the indicator location – Cernavoda town (ADI-08) results are situated between 0.010 and 3.290 Bq/m^3 (mean value: 0.580 Bq/m^3) and for background station – Topalu village (ADB-01) results are from 0.010 to 0.050 Bq/m^3 (mean value: 0.080 Bq/m^3).

Tritium activity concentrations in drinking water sample, represented in Figure 2, are very low comparing with legal limit of 100 Bq/m^3. Most of the results are situated below MDA (2.4 Bq/l).

In milk samples, tritium activity concentrations (monthly mean values of weekly sample measurements) in period 2006 – 2008 (represented in Figure 3) are situated between 2.7 and 28.2 Bq/l. Annual mean values of the monthly averages are the following: for 2006 – 12.6 Bq/l, for 2007 – 9.8 Bq/l, for 2008 – 10.8 Bq/l. The mean value for entire studied period is 11.1 Bq/l.

For fish samples, it can be notice that tritium activity concentrations in free water, in the studied period (represented in Figure 4), are lower in the pound water than in the flowing water. The mean value for period 2006 – 2008, for flowing water is 10.7 Bq/kg and the mean value for the same period in pond water is 6.1 Bq/kg.

Conclusions

The results of the monitoring program are annually compared with the results of The Preoperational Environmental Monitoring Program performed between 1984 and 1996. There were no environmental radioactivity modifications around the Cernavoda NPP comparing to the period before the NPP operation.

In generally, measurements shown that the radiation levels in the external area of a nuclear power plant are around 0.01 mSv per year, comparing with the annual effective dose due to natural radiation sources of 2.4 mSv – the major component coming from the radon (Rn-222, $T_{1/2}$ = 3.8 days) – about 1.3 mSv.

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Radionuclides activity concentration in soil in Serbia

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Abstract
The soil is the basic environment of migration of radionuclides into plants, where from they reach the people and animals through food. The type of soil affects the distribution of radionuclides in the soil itself and their transfer into plants, respectively. Systematic testing of radioactivity of soils in the Serbia is performed after the Chernobyl accident. The periods of sampling were spring and autumn every year. According to collected data, the activity of natural radionuclides was very equal on all locations every year as it was expected because of long half lives of natural radionuclides. High standard deviation and big difference between minimal and maximal ^{137}Cs activity concentrations suggested typical artificial pollutant. The activity of long-living radionuclide ^{137}Cs is constantly decreasing in the upper layers of uncultivated soil, but because of its long half-life it will remain in ecosystem for a long time.

Introduction
The primary reason for being concerned about radioactive contamination of the environment is that it results in exposure of humans. External irradiation from radionuclides naturally present in the environment or released from man-made practice or events is usually an important component of the exposure. Most of the food consumed by human beings is grown on land. Contamination of land can occur either from deposition of material originally introduced to atmosphere, or from waste products discharged directly placed in or on the ground, from which they are eventually mobilized by groundwater or erosion.

Soil consists of mineral and organic matter, water, and air arranged in a complicated physicochemical system that provides the mechanical foothold for plants in addition to supplying their nutritive requirements. The type of soil affects the distribution of radionuclides in the soil itself and their transfer into plants, respectively.

Radionuclides deposited on land may enter human food chain in the following ways:

– by direct deposition onto the leaves or exposed parts of plants that are eaten by humans or other animals,

– by persistence in layers of soil from which they are taken up into growing plants through their roots,
– by resuspension as dust from the soil or from other exposed surfaces,
– by being washed of the surface or from deeper ground layers into water sources that are used ultimately for drinking or for irrigation, or as media from which fish or other food are drawn.

The global source of radionuclide contamination in our country is the fallout due to previous nuclear testing and Chernobyl accident, and, in the recent time, after the NATO aggression, depleted uranium content in the environmental samples.

The three naturally occuring radioactive series and ^{40}K contribute for most of the naural terrestrial radoactivity. Aditionally ^{137}Cs is the most important radionuclides which contribute to human exposure after nuclear testing and Chernobyl accident.

Material and methods

Systemic testing of radioactivity of soils is performed on 10 locations in the Serbia (Belgrade- 3 sites, Lazrevac, Obrenovac, Novi Sad, Subotica, Niš, Zaječar, Zlatibor), twice a year during the 20 years period after the Chernobyl accident according to methods defined by regulations (Pantelić et al., 2006).

Additionally to that program undisturbed grassland soils were sampled in 33 different sites in Serbia during 2006-2008. Several sites are located in a mountainous area in the west and central of Serbia (Zlatibor, Kopaonik), several sites in a flat northern Serbia (Vojvodina) and several sites in the south Serbia. In each site 5 cm thick sample layers were separately collected. The periods of sampling were spring and autumn.

The soil samples were purified from plants and rocks. Each sample was dried in an oven at 105°C-110°C to constant weight during 24-48 h. The dry soil was crushed and sieved (0.5 mm). The resulting sample was weighed and transferred into a Marinelli beaker. Natural and artificial radionuclides concentrations were measured using a high-resolution gamma ray spectrometer (HPGe detector - 30 % efficiency at 1332 keV). Time of measurement was at least 20000 s.

The radionuclide activity of uranium and thorium series and ^{40}K, as well as the artificial radionuclide ^{137}Cs was determined.

Results and discussion

The radionuclide activity of natural uranium and thorium series, and natural radionuclide ^{40}K was determined in all soil samples. Among fission products, only ^{137}Cs was measured, while other radionuclides were below detection limit.
The results of measurements are presented in tables 1-4.

Minimum, maximum and average radionuclide concentration measured in 2005 in different regions in Serbia in table 1 is showed. Natural radioactivity in Vojvodina and west and central part of Serbia is very similar, while this activity in south part of Serbia is little bit higher. These results are the same as previos obtained results (during 1987-2004) for the same sites in Serbia because of long half lives of natural radionuclides (Eremić-Savković et al., 2002; Pantelic et al., 2000; Pantelić et al., 2006).

The similar results for natural radionuclide concentration were obtained during 2006-2008 (table 2-4). Maximum activity concentratin for all natural radionuclide is twice time higher then minimum concentration for the same radionuclide at the same region, but calculated standard deviation is less then 30 % for almost all natural radionuclides.

Table 1. Radionuclide activity concentration in the soil in Serbia in 2005.

Region		^{40}K (Bq/kg)	^{137}Cs (Bq/kg)	^{232}Th (Bq/kg)	^{226}Ra (Bq/kg)	^{238}U (Bq/kg)	^{235}U (Bq/kg)
Vojvodina	Minimum	306	1.5	20	21	18	< 1
	Average	420 ± 80	7.5 ± 4.5	30 ± 9	33 ± 11	32 ± 13	1.5 ± 0.6
	maximum	523	17	48	54	65	2,9
West and central	Minimum	102	5.9	8.6	7.6	19	< 0.6
	Average	490 ± 200	57 ± 47	40 ± 16	42 ± 17	44 ± 19	2 ± 1
	maximum	687	172	75	76	103	4.2
South Serbia	Minimum	439	0.9	37	27	35	1.6
	Average	590 ± 130	36 ± 28	42.5 ± 4.3	38 ±11	75 ± 43	2.3 ± 0.5
	maximum	829	68	48	54	153	2.9

Table 2. Radionuclide activity concentration in the soil in the Vojvodina region.

	^{40}K (Bq/kg)	^{137}Cs (Bq/kg)	^{232}Th (Bq/kg)	^{226}Ra (Bq/kg)	^{238}U (Bq/kg)	^{235}U (Bq/kg)
Minimum	312	1.5	28	26	29	1.2
Average	460 ± 85	7.5 ± 4.5	38.9 ± 5.8	40.4 ± 6.6	50 ± 22	2.0 ± 0.4
maximum	612	17	48	49	88	2.8

Table 3. Radionuclide activity concentration in the soil in the west and central part of Serbia.

	^{40}K (Bq/kg)	^{137}Cs (Bq/kg)	^{232}Th (Bq/kg)	^{226}Ra (Bq/kg)	^{238}U (Bq/kg)	^{235}U (Bq/kg)
Minimum	162	6.3	7.4	11	9.4	0.5
Average	440 ± 170	69 ± 73	35 ± 19	36 ± 15	45 ±22	1.7 ± 0.7
maximum	586	197	60	55	117	2.6

Table 4. Radionuclide activity concentration in the south part of Serbia

	^{40}K (Bq/kg)	^{137}Cs (Bq/kg)	^{232}Th (Bq/kg)	^{226}Ra (Bq/kg)	^{238}U (Bq/kg)	^{235}U (Bq/kg)
Minimum	368	1.4	28	28	24	1.3
Average	512 ± 83	21 ±11	43 ± 16	46 ± 12	74 ± 44	2.5 ± 1.0
maximum	603	31	70	67	146	4.4

Radionuclide ^{137}Cs was present in all soil samples. High standard deviation and big difference between minimal and maximal activity concentrations of cesium suggested typical artificial pollutant. Given that its half-life is 30 years, it is distributed deeply into soil by washing out, and it will remain in ecosystem for a long time. Extremely high ^{137}Cs concentrations were recorded in the uncultivated soil in the west part of Serbia, that could be explained by first precipitations following immediately the Chernobyl accident. The activity of long-living radionuclide ^{137}Cs is constantly decreasing in the upper layers of uncultivated soil comparing with mesurement before 2006 (Pantelić et al., 2006).

Conclusions

Three years of monitoring the content of natural radionuclides as well as radionuclides of artificial origin in soil samples in the Republic of Serbia indicated that there was no deviations in the natural activity from the average values for the same types of soil samples obtained before 2006. High standard deviation and big difference between minimal and maximal ^{137}Cs activity concentrations suggested typical artificial pollutant. The activity of long-living radionuclide ^{137}Cs is constantly decreasing in the upper layers of uncultivated soil, which is the result of cesium breakdown and its transport into lower layers.

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Monte Carlo calculation of ambient dose equivalent and effective dose from natural radionuclides in the soil of Vojvodina district in Serbia

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Abstract
Presented results are the part of the project financed by Ministry of Science and Technological Development, Republic of Serbia. As the main influences come from uranium, thorium and potassium we measured their concentrations in soil by radiometric techniques. Further we calculate external doses as well as effective doses for particular organs and specified tissues using Monte Carlo code. In this paper only the results for external dose equivalent at 1 m from the surface are presented. Two depths of the soil are taken into consideration, infinite and 15 cm. The results are given for the territory of Bela Crkva and Vrsac in Vojvodina district, Serbia.

Introduction
The most widely spread radionuclides of natural origin come from uranium and thorium series as well as from potassium (UNSCEAR 2000). Those natural occurring radionuclides are internal or external sources of radiation depending on the position to the organisms. In this survey just radiation originating from the internal exposure is taken into consideration, which arises from intake of terrestrial radionuclides by inhalation and ingestion.

Recently, few researches were carried out on territory of Republic of Serbia in order to determine concentrations of natural occurring radionuclides in surface soil (Bikit et al. 2005; Dragovic et al. 2006). In this study, in addition to measuring concentrations of natural radionuclides in the territory of Bela Crkva and Vršac, the goal was the estimation of effective dose from inhalation and ingestion. The region of interest is located in a northern-east part of Serbia, in the province of Vojvodina. It is mainly oriented to the agriculture, but it is potentially interesting for tourism. Therefore it is important to determine the impact of internal exposure to natural radionuclides (uranium, thorium and potassium). In Figure 1 the location under investigation is shown on the map of Vojvodina.

Eighty soil samples were taken from the area of Bela Crkva and Vršac to evaluate concentration of these natural radioactive sources. The concentrations of radionuclides in the samples were measured by radiometric methods. Based on those measurements the activities of uranium, thorium and potassium were calculated. Using Monte Carlo simulation the conversion factors were determined in order to estimate the contribution of above-mentioned radionuclides to the effective dose from ingestion and inhalation.

This study was supported by Ministry of Science and Technological Development of Republic of Serbia. The purpose of the project was determining concentration of natural radionuclides in the soil samples and evaluating the radiation doses, received by people living in the area under investigation, resulting from the inhalation and ingestion. The outputs of the project were radiometric maps, which showed the level of the exposure from the radioactive sources.

Fig. 1. Map of Vojvodina district.

Material and methods

Sampling and radioactivity measurements

For the purpose of this research, eighty soil samples were taken from 13 uncultivated locations in the same geographic area in the vicinity of Bela Crkva and Vršac. They were collected at a depth of (0-5) cm, and each sample weighed 0.5 kg. Geographical coordinates of sampling positions were determined using a GPS tracker. All the samples were air dried, homogenized and sieved to grain size of less than 0.60 mm. They were prepared and placed in cylindrical gas-tight containers with the same geometry as to sample container used for efficiency calibration. The samples were kept for at least three weeks before the measurement to reach secular equilibrium between thorium and radium and their decay products.

Further, gamma-spectrometric measurements were performed with high purity, low energy germanium semiconductor detector (HPGe), manufactured by ORTEC, accompanying electronic equipment and ORTEC software for spectra evaluation. Relative efficiency of HPGe is 28 % and the energy resolution achieved in the measurements is 2 keV, both at the 1.33 MeV reference transition of ^{60}Co. Calculated minimum detectable concentration (MDC) is 1.1 Bq/kg for ^{238}U, 1.3 Bq/kg for ^{232}Th and 2.4 Bq/kg for ^{40}K. The counting geometry is identical for all radionuclides. The expanded uncertainty of determined radionuclide concentration is 15 % (k = 2) in each case. Detectable energy range of used instrument is up to 2 MeV. Instrument calibration was performed using reference sources:

1) NBL 103 (USAEC), content of U: 0,05 %
2) NBL 107, 0,1 % Th
3) K in form of potassium-chloride
4) Certified mix source, Amersham, 1988 (^{55}Fe, ^{60}Co, ^{137}Cs, ^{226}Ra and ^{241}Am).

Monte Carlo simulation

In dose estimation, it is assumed that natural uranium contains three isotopes: 99,284 % ^{238}U +0,711 % ^{235}U + 0,0058 % ^{234}U, while other isotopes of uranium neglected.

In order to determine the conversion factors which are needed for calculation of the effective dose due to exposure to radionuclides entered by inhalation or ingestion, Monte Carlo method is used. Firstly, the dose rate conversion factor (absorbed dose rate in air per unit activity per unit of soil mass, nGy/h per Bq/kg) was estimated. For simulation the inscattered and scattered radiation is taken into account.

The main assumption of the problem starts from the polynomial expansion matrix, which solves the transport problem of radionuclides in the soil and between soil and air. Given the gamma lines of radionuclides of interest, the problem of relatively narrow energy range from 1 keV to 2.75 MeV was resolving.

It is assumed that the ground is infinite medium for photon scattering and radionuclides are uniformly distributed over the surface from which the soil samples were taken. Characteristics of soil and cross-sections for photons are taken from existing database.

The code for simulation is written in FORTRAN 77. The simulation is done in 2π geometry, which was only reasonable solution due to the assumption of indefinite soil thickness. The same density of soil is taken for all the samples, ρ = 2.6 kg/m^3, which brought additional errors in calculation, but not greater than 2%. The “detector” is 1 m above the soil and is simulated by a square surface with side of 2 m, located in the center of the square slab which represents the soil.

The mechanisms of interaction of photons with matter taken into account were the photoelectric effect, Compton scattering and pair production. Keeping in mind the characteristics of natural radionuclides and its interaction with soil, a cut off the energy range from 50 keV to 2.6 MeV was introduced. This is the reasonable cut off, as the highest important gamma energy of natural radionuclide is 2614 keV and because photons even below 50 keV contribute a negligible amount to the dose rate. (Clouvas, A. et al 2000)

In order to determine effective dose, voxel phantom was used in Monte Carlo simulation. It enabled solving transport problem of radionuclides entering human body through the respiratory and digestive tract. The effective dose coefficients obtained by Monte Carlo method are expressed in units of Sv/Bq. Effective doses from inhalation and ingestion were determined by multiplying the activities and obtained conversion factors.

Results

The activity concentrations of ^{238}U and ^{232}Th varied from (3.24-57.02) Bq/kg and from (3.4–90.42) Bq/kg, respectively. ^{40}K was found in higher concentration, it ranges from (84.78–3114.8) Bq/kg. In some samples it was not possible to determine the radionuclide concentration, because it was lower than the minimum detectable concentration. The mean concentration is 25.23 Bq/kg, 39.03 Bq/kg and 873.69 Bq/kg for ^{238}U, ^{232}Th and ^{40}K, respectively.

The concentrations of ^{238}U, ^{232}Th and ^{40}K analyzed in this research and those reported by UNSCEAR (B) (2000) are presented in Table 1. It is shown that activity concentration of ^{238}U agrees with one in other countries, and concentration of ^{232}Th is a bit out of worldwide activity range. In the case of ^{40}K, the mean activity concentration is much higher that the worldwide average value (400 Bq/kg). It may be explained by either the use of fertilizers or the soil type. (Colmenero Sujo et al. 2004)

The mean concentrations in soil samples from Serbia and Montenegro are 597.96 Bq/kg, 35.697 Bq/kg and 42.12 Bq/kg for ^{40}K, ^{238}U and ^{232}Th, respectively (Dragovic et al. 2006). When the obtained results from are study are compared with these values, it is obvious that concentrations of ^{238}U and ^{232}Th are lower, but measured concentration of ^{40}K is higher in our study.

Table 1. The comparison of activity concentration ranges of natural radionuclides in soil samples from the present study with the values obtained in other studies conducted worldwide.

Concentration (Bq/kg)	^{40}K	^{238}U	^{232}Th	Reference
Croatia	140-710	83-180	12-65	UNSCEAR, 2000
Bulgaria	40-800	8-190	7-160	UNSCEAR, 2000
Romania	250-1100	8-60	11-75	Iacob, 1996
Hungary	79-570	12-66	12-45	UNSCEAR, 2000
Poland	110-970	5 -120	4 -77	Jagielak et al., 1992
Median value	140-850	16-110	11-64	UNSCEAR, 2000
Bela Crkva and Vršac	84.78-3114.8	3.24-57.02	3.4-90.42	Present study

Furthermore, the effective dose was calculated using the determined activity concentrations of ^{238}U, ^{232}Th and ^{40}K and the conversion factors, which the output of Monte Carlo simulation were. The annual effective dose from inhalation of ^{238}U ranges from (58 - 1100) µSv, the effective dose of ^{232}Th varies from (39 - 1040) mSv and the effective dose of ^{40}K varies from (0.53 – 19.5) µSv. On the other hand, the annual effective dose received by ingestion of ^{238}U ranges from (0.01 – 0.2) µSv, the effective dose of ^{232}Th varies from (1.25 – 33.3) µSv and the effective dose of ^{40}K varies from (0.21 – 7.8) µSv.

Using adequate software and based on the GPS coordinate of a taken soil sample, the distribution maps of the activity concentration were generated. The maps are shown in Figures 2-4. It is important to emphasize that distribution of effective dose due to ingestion and inhalation has the same distribution, just the corresponding values are different.

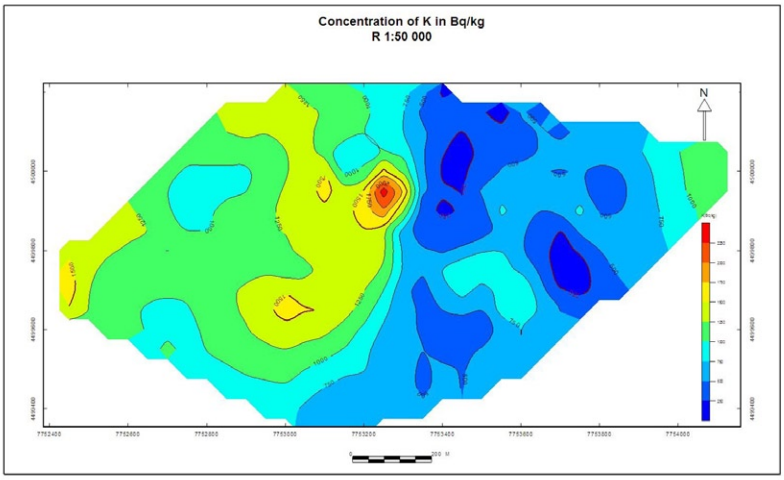

Figure 1. Activity concentration of ^{40}K in the area of Bela Crkva and Vršac.

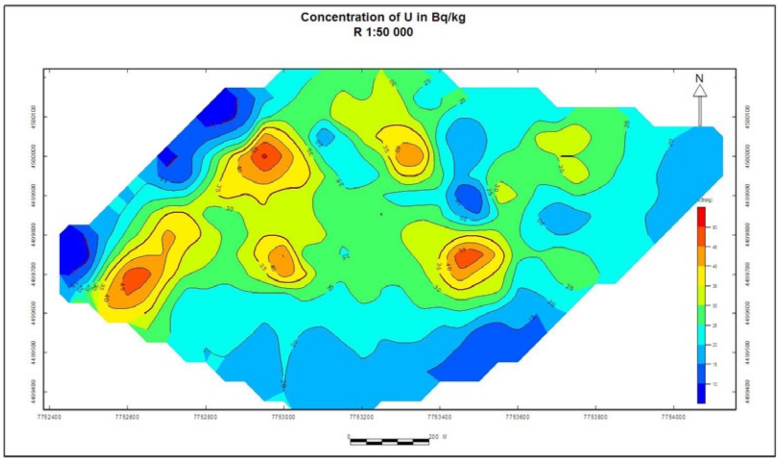

Figure 2. Activity concentration of ^{238}U in the area of Bela Crkva and Vršac.

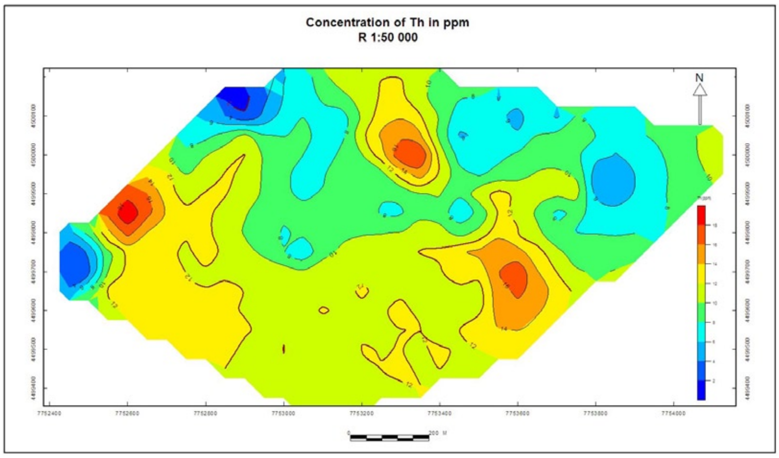

Figure 3. Concentration of ^{232}Th in the area of Bela Crkva and Vršac.

Discussion

There are two main processes that contribute to internal exposure, the general term used to describe exposures that involve the intake of radionuclides into the body. The two processes are inhalation of contaminated air and ingestion of contaminated foodstuffs. (UNSCEAR 2000) As it is already mentioned, the aim of this research was to determine annual effective dose from inhalation and ingestion of long-lived radionuclides.

Table 2. Annual effective dose from ingestion of terrestrial radionuclides.

Ingestion	Average (UNSCEAR 2000)	Bela Crkva and Vršac (Present study)
^{40}K	0.17	0.004
U and Th series	0.12	0.017
Total ingestion exposure	0.29	0.021

Since ^{238}U and ^{232}Th are alpha-emitters, when they are ingested or inhaled, they contribute significantly to the radiation dose that people receive. (Colmenero Sujo et al. 2004) Whereas potassium is beta-emitter, its impact is less significant. These facts agree with the output of this research. In Table 2 the obtained effective doses are compared with values reported by UNSCEAR. It is shown that effective dose from ingestion of ^{40}K (4 μSv) is lower than the average value (170 μSv). Obtained annual effective dose from ingestion of ^{238}U and ^{232}Th (17 μSv) is lower than the average value (120 μSv). The total effective dose from inhalation is higher than average value given in UNSCEAR report.

Conclusions

This paper presents the radioactivity concentrations of ^{238}U, ^{232}Th and ^{40}K for 80 soil samples collected in the area of Bela Crkva and Vršac. According to the obtained results, it is shown that activity concentration of ^{40}K exceeds worldwide average value. Further investigation should be done in order to give an explanation, but it can be assumed that using fertilizers could contribute to higher potassium concentration. Concentrations of ^{238}U and ^{232}Th are similar to the reference values provided by UNSCEAR.

Effective dose, an indicator of the stochastic effect of radiation, has been widely used in dose evaluation in the environment. In this research the effective dose from inhalation and ingestion of natural radionuclides is estimated. It is proven that effective dose from ingestion is pretty lower than average values. As regards inhalation, effective dose exceeds average value.

Further research will include using risk assessment.

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Interpretation of radionuclide concentrations near the detection limit for dose calculations

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Abstract
As with many other environmental pollutants, the concentration distribution of various radioactive substances in the environment can be measured. The influence of radioactive releases into the environment, based on these measurements, can then be used in assessments of the radiation doses, so determining the implied risks to the population.

The outcome of a measurement of the activity concentration can be described as a result that is below the decision threshold, a result that has an uncertainty too large to be reported or a definite measurement outcome. In the first case the analyte does not appear in the list of the results, in the second case it appears as an upper limit and in the third case as a measurement result including its uncertainty. A method for consistently treating these three kinds of measurement outcomes is presented.

When the measurement data are used for dose assessments, averaging may be introduced in order to reduce the amount of input data. It has been shown that in order to reduce the systematic influences occurring when calculating averages from sets of data, where measurement outcomes below the decision threshold or measurement results with an unacceptably high relative uncertainty are frequent, the maximal acceptable relative uncertainty should be set as high as possible. Because of the different characteristics of the analytes, however, it is advantageous to use analyte-dependent maximal acceptable relative uncertainties.

Introduction
Radiological environmental-monitoring programs are carried out in order to survey the radioactivity in the natural and living environment. The purpose of the survey is to assess the impact of the radioactivity on the population, which is performed by evaluating the doses. The radiological impact of nuclear installations in normal operations is commonly considered to have two aspects: the annual dose to a member of the critical group, and the collective dose. This annual dose is compared to the respective regulatory dose limit. To perform the comparison it is useful to know the uncertainty of the dose, which includes the uncertainties associated with the measurement results. Therefore, for a realistic evaluation all the relevant pathways have

to be controlled and the concentrations of the most radiotoxic isotopes have to be measured.

It is important to assess the doses in accordance with the directions of the regulatory bodies and the recommendations of international organisations. If possible, the doses are calculated from the concentrations of materials from the environment. However, doses from discharges in normal operation can accumulate over many years and at distances where the radionuclides are diluted below the detection limit. Therefore, mathematical models and statistical approaches have to be used to evaluate the doses, including the projected doses, correctly.

If the concentrations of the radionuclei of interest are below the detection limit of the measurement methods used, the measurement outcomes may be described as measurement results or concentrations below the decision threshold. Alternatively, some of the radionuclei may be identified in the sample, but the concentrations may be too low to be evaluated quantitatively. In this case the relative uncertainty of the measurement result may approach or even exceed 100%. To treat these results a maximum relative uncertainty is introduced for results reported. The measurement results having a larger relative uncertainty are quoted as a concentration below an upper limit, which is given as the measurement result, increased by its uncertainty, multiplied by a coverage factor, which is determined by the interval of confidence. To arrive at a reliable dose estimate these incomplete data have to be included in the evaluation. However, neither the decision thresholds nor the upper limits are properties of the sample and therefore do not describe the state of the nature but the properties of the measurement process. Therefore, the doses assessed must not depend on them.

It is the aim of this contribution to present a method for how to treat the experimental data in such a way that the assessed doses themselves depend on the sample properties and the uncertainties of the doses on the properties of the analytical process. How to carry out the analytical process in order to minimize the systematic influences arising from converting the measurement results into upper limits is described. This method is used in the evaluation of doses in the program of the off-site radiological monitoring of the NPP Krško (Glavič-Cindo 2006, Glavič-Cindo 2007, Črnič 2008, Črnič 2009, Zorko 2010).

Material and methods

The analytical process

The samples that are collected in the annual programs of the radiological survey are usually taken fortnightly, monthly or quarterly. The yearly averages, which are used in the dose assessment, are obtained by averaging the measurement results. Most of the samples, collected in the environment, are measured using gamma-ray spectrometry. For such samples, gamma-ray spectrometry is the method of choice, because of its sensitivity, selectivity and price. Since gamma-ray spectrometry is a multi-nuclide method, the result of a gamma-ray spectrometric measurement is a list of radionuclides identified in the sample and their concentrations.

Within the sample-analysis procedure several operations are made, where the peak count rates or activities are corrected for contributions that do not originate in the material sampled. These operations are a background subtraction, the correction for a blank and the interference corrections. If the signal from the sampled material is not present or much smaller than the uncertainty of the contribution, which is subtracted, a positive or a negative result is equally probable. In the first case a type-1 error occurs, which occurs in the report as a measurement result with a large relative uncertainty. In most cases this is quoted in the form of an upper limit. Such results are indistinguishable from the results corresponding to samples with a concentration above the decision threshold (critical limit) but having a relative uncertainty exceeding the maximal. It follows that type-1 errors may occur frequently in the measurement outcomes for radionuclei, which are present in the spectrometer background. These are long-lived, naturally occurring radionuclei with their decay products and Cs-137.

If the result of the subtraction is negative, the peak or the radionucleus is deleted from the list of peaks or the radionuclides present in the material sampled (Canberra Industries, 1998). It follows that by the repeated measurements of samples containing a negligible concentration of a radionucleus, one half of the measurements will result in a positive identification. This fraction may be even larger for multi-gamma emitters, if the criterion for the detection allows identification by the recognition of just one of several peaks. For a multi-gamma-ray emitter the activity is calculated as an average over the activities obtained from attributed peaks only, which may result in a considerable systematic influence. The peaks with a negative area are excluded from further analysis and the average calculated corresponds only to the peaks with positive net areas. In the extreme case it may happen that the uncertainty of the average shrinks to a value that results in the average having a relative uncertainty smaller than the maximal. In this case an erroneous result is reported. To decrease the probability of such errors the laboratories maintain maximal relative uncertainties that are much smaller than 100%.

Presentation of the measurement results

Often the measurement results referring to a specific medium are presented in tables. The measurement results of periodically collected samples of the same medium are presented as records, describing the concentrations of radionuclei at a given sampling time. The measurement outcome can appear in one of three forms:

- as a measurement result: $a \pm u(a)$, where $u(a)$ denotes the uncertainty of the activity concentration a,
- as an upper limit: $< q(a)$, where $q(a)$ denotes the decision threshold or the sum of the measurement result and its uncertainty multiplied by the coverage factor
- as an empty space (blank), which implies that the radionucleus was not detected in the sample.

To calculate the yearly averages the three types of entry content have to be converted to the measurement results and their uncertainties. It should be noted that the averaging must not be performed with weights, depending on the uncertainties, in order not to introduce systematic influences into the average.

The uncertainty of the average can be evaluated as the *a-priori* uncertainty, which is given by the uncertainties of the individual measurement results, or as the *a-posteriori* uncertainty, which is given by the spread of the measurement results around the average. Here, care should be taken not to interpret the expected time variations, e.g., seasonal variations, as deviations from an average, which is assumed to represent a concentration characteristic for the time interval of the averaging. Usually, as the uncertainty of the average, the larger of the *a-priori* and *a-posteriori* uncertainties is used.

Often, the uncertainties are quoted with a coverage factor of unity, defining the interval with a confidence level of 68%. On the other hand, the upper limits are normally calculated for a confidence interval of 95% (ISO 2005). For the one-tailed probability distribution this corresponds to a coverage factor of 1.65 (Hurtgen, 2000). Since the upper values, like the uncertainties, describe the analytical process, they should be treated on the same basis. Therefore, when upper limits are converted to measurement results they are transformed into uncertainties calculated with a coverage factor of unity.

The transformation, which is used for converting the three forms of data in a format that is appropriate for use in the averaging, is the following:

$$\begin{aligned} &a \pm u(a) \Rightarrow a \pm u(a) \\ &< q(a) \Rightarrow 0 \pm q(a)/1.65 \\ &[blank] \Rightarrow 0 \pm 0 \end{aligned} \qquad (1)$$

It can be observed from the Eqs. (1) that the upper limits are transformed to measurement uncertainties. Consequently, the upper limits do not influence the average itself, but its uncertainty.

When performing this transformation, a systematic influence is introduced since all the measurement results having a relative uncertainty exceeding the maximal are replaced by zeros. This effect partially compensates for the effect of false identifications.

If in the spectrum analysis just one peak is used for the calculation of an activity concentration of a radionuclide, and if this peak occurs in the spectrometer background, the probability density distribution for reporting the result $a \pm u(a)$ when the radionucleus is not present in the sample is

$$p(a) = \frac{1}{2u(a)\sqrt{2\pi}} e^{-\frac{a^2}{2u(a)^2}} \qquad (2)$$

giving an average reported activity concentration of $u(a)(\sqrt{2/\pi} \pm \sqrt{1-2/\pi})$ (Korun, 2010). It follows that the relative uncertainty of the average of the wrongly reported activity concentrations, is $\sqrt{\pi/2-1} = 0.76$. Then, in the measurement reports such results are quoted as $< u(a)\sqrt{2/\pi}(1+1.65\sqrt{\pi/2-1}) \approx 1.8u(a)$ and interpreted in the average as $0 \pm 1.8u(a)/1.65$. It follows then that type-1 errors arising from the background subtraction, interference corrections and corrections for blank for the radionuclides that are determined from one spectral peak are, in the averages, properly taken into account with a maximal relative uncertainty of 76%.

It should be noted that the approach described does not comply with the European Commission recommendation on the information on discharges from nuclear installations (Commission recommendation 2004/2/Euratom). Here, the measurement outcomes below the decision threshold should be substituted by one half of the decision threshold. Only in the case of repeated measurement outcomes below the decision threshold should zeros be assumed to be true values. In our approach the concentration of the radionuclides is reported correctly from the point of view of the analytical process, since the measurement uncertainty is used for determining the measurement outcome.

Results and discussion

In this section, examples are given in order to illustrate the influence of the maximal relative uncertainty for various types of gamma-ray emitters on the average activity concentration and consequently on the assessed doses.

Let us present the influence of the maximal relative uncertainty on the average over the measurement data, transformed to a form shown in Eqs. (1). In the second row in Table 1 are the hypothetical measurement results, evenly distributed over the interval from 0.7 to 2.7. Since the uncertainties near the decision threshold are dominated by the statistical uncertainty, the measurement uncertainty is a slowly varying function of the measurement result (Weise 2006). Therefore, uncertainties independent of the measurement result are assumed. In these conditions, the *a-priori* uncertainty exceeds the *a-posteriori* uncertainty, and therefore only the *a-priori* uncertainty of the average is presented. For each maximal relative uncertainty in the first row the measurement results, as presented in the measurement report, are given, and in the second row these data are transformed by Eqs. (1) into the form appropriate for averaging. It can be observed that the average value reproduces the average of the measurement results better if the upper limits are calculated for results having a high relative uncertainty, i.e., by using a high maximal relative uncertainty. As it was shown in the previous section, the upper limits for the results with a relative uncertainty exceeding 76% should be quoted. It should also be noted that the difference between the average over the measurement results and the average of the transformed data is covered by the uncertainty in all the maximal relative uncertainties used, but the systematic effect amounts to a factor of 4 when a maximum relative uncertainty of 80% instead of 40% is used.

Table 1. Comparison between the average over the measurement results (in arbitrary units) and the averages over the measurement data transformed to a form that is appropriate for averaging.

Meas. No.	1	2	3	4	5	6	7	8	9	10	Av.	Unc.
Results	0.8±1	1.0±1	1.2±1	1.4±1	1.6±1	1.8±1	2.0±1	2.2±1	2.4±1	2.6±1	1.7	0.32
Limit of quant.:0.8	< 2.5	< 2.7	< 2.9	1.4±1	1.6±1	1.8±1	2.0±1	2.2±1	2.4±1	2.6±1		
	0±1.5	0±1.6	0±1.7	1.4±1	1.6±1	1.8±1	2.0±1	2.2±1	2.4±1	2.6±1	1.2	1.2
Limit of quant.:0.6	< 2.5	< 2.7	< 2.9	< 3.1	< 3.3	1.8±1	2.0±1	2.2±1	2.4±1	2.6±1		
	0±1.5	0±1.6	0±1.7	0±1.9	0±2.0	1.8±1	2.0±1	2.2±1	2.4±1	2.6±1	1.0	1.4
Limit of quant.:0.4	< 2.5	< 2.7	< 2.9	< 3.1	< 3.3	< 3.5	< 3.7	< 3.9	< 4.1	2.6±1		
	0±1.5	0±1.6	0±1.7	0±1.9	0±2.0	0±2.1	0±2.2	0±2.3	0±2.5	2.6±1	0.3	1.9
Limit of quant.:0.2	< 2.5	< 2.7	< 2.9	< 3.1	< 3.3	< 3.5	< 3.7	< 3.9	< 4.1	< 4.3		
	0±1.5	0±1.6	0±1.7	0±1.9	0±2.0	0±2.1	0±2.2	0±2.3	0±2.5	0±2.6	0	2.1

For gamma-ray emitters radiating photons at a number of different energies in the identification process, the so-called "abundance limit" parameter is used (Canberra, 1998). This parameter defines for a gamma-ray emitter the minimum ratio of the sum over the emission probabilities for gamma rays identified in the spectrum versus the sum of all the probabilities for the emission of gamma rays for this emitter. It is advantageous to use as many gamma-ray peaks as possible in the activity calculation since interference corrections can then be performed and the average becomes less susceptible to a possible wrong-peak-area evaluation. If the spectrum of a multi-gamma-ray emitter appears in the spectrometer background, it is expected that 50% of the peaks will be identified, providing the concentration of the emitter is well below the decision threshold. With an abundance limit of 50% it is expected that such a radionucleus will be identified in approximately one half of the measurements.

To illustrate the influence of the maximal relative uncertainty on the average a hypothetical example is presented, describing the case of an emitter radiating at several energies with approximately equal probabilities. The case of a multi-gamma-ray emitter radiating at an energy with a much larger probability than at other energies is similar to the case of a radionucleus radiating at a single energy, which was considered in the previous paragraph. The emitter is present in the spectrometer background and radiates at three energies with such probabilities that the uncertainties of the activity concentrations calculated from the areas of the peaks at the three energies are equal. A nucleus, which resembles this example, is U-238, which is in equilibrium with its daughter Th-234 and with U-235 in the natural isotopic ratio. Then, the activity concentrations calculated from the 63-keV, 93-keV and 186-keV peaks have approximately equal uncertainties. The value of the abundance limit parameter of 50% implies that the emitter is identified if any two of the three peaks are identified. Thus, the probability for a false identification is 50%. The average activity concentrations are $u(a)[\sqrt{2/\pi} \pm \sqrt{(1-2/\pi)/2}]$ in the case of two identified peaks (38% of measurements) or $u(a)[\sqrt{2/\pi} \pm \sqrt{(1-2/\pi)/3}]$ in the case of three identified peaks (12% of measurements). The corresponding relative measurement uncertainties are 54% and 44%. To avoid reporting these false results for radionuclei resembling this example, the maximal relative uncertainty should be decreased to below 44%.

Alternatively, increasing the value of the abundance limit parameter can prevent the reporting of false results. A value of 0.7 decreases the probability for reporting false results to 12%, but this is accompanied by an increased probability of type-2 errors.

To present the influence of the abundance limit and maximal relative uncertainty on the frequency of various kinds of measurement outcomes if the activity concentration is near the decision threshold, let us consider again the case of a gamma-ray emitter radiating at three energies. It is assumed that the sample contains an activity concentration that equals its uncertainty if the activity concentration is determined from just one of the three gamma-ray peaks. Since the emitter radiates at three energies, in one measurement three determinations are made and the relative uncertainty of the average activity concentration is $1/\sqrt{3}$. From the probability-density distribution it follows that the probability of the occurrence of type-2 errors is 0.041, the probability for reporting an upper limit is 0.625 (with a maximal relative uncertainty of 80%) and the probability for reporting a positive measurement result is 0.334.

Table 2 presents the probabilities for the type-2 errors and the probabilities for reporting the upper limits if the abundance limits of 50% or 70% were used in the analysis. In the first case the emitter is detected if two of the three peaks in the spectrum are recognized. In the second case the emitter is recognized only in the case when all three peaks are recognized in the spectrum. The probabilities are given for a maximal relative uncertainty of 80% and 44%.

Table 2. Probabilities for type-2 errors and reporting the upper limits for the case of a gamma-ray emitter radiating at three energies with a concentration corresponding to the statistical uncertainty of one peak area.

Abundance limit	Probability of Type-2 errors	Maximal relative uncertainty	
		80%	44%
50%	0.025	0.15	0.64
70%	0.31	0.14	0.43

It can be observed that at a high abundance limit an unacceptably high probability for measurement outcomes resulting in type-2 errors occurs. These errors occur since the emitter is identified only if all three peaks are identified. At the abundance limit of 50%, at the maximal relative uncertainty of 80%, the minority of measurement outcomes are reported as an upper limit, whereas at the maximal relative uncertainty of 44% the majority is reported in this form. In Table 3 the average over the measurement outcomes resulting in reporting activity concentrations are presented as well as the averages over all the measurement outcomes. In this average activity the concentrations corresponding to type-2 errors and the upper limits are taken into account as zeros.

It can be observed that the systematic influences covered by the uncertainties are smaller at the abundance limit of 50%. At the maximal relative of 44% these influences are larger, since here most of the results are reported as upper limits. It can also be observed that the systematic influences are smaller when a higher maximal relative uncertainty is used.

Table 3. Average activity concentrations reported and average activity concentrations, calculated over measurement results (in arbitrary units). Here type-2 errors and upper limits have been taken onto account as zeros.

Abundance limit	Maximal relative uncertainty			
	80%		44%	
	Average reported activity concentration	Average over all results	Average reported activity concentration	Average over all results
50%	1.17 ± 0.59	0.97 ± 0.64	1.56 ± 0.59	0.52 ± 0.92
70%	1.23 ± 0.58	0.68 ± 0.47	1.54 ± 0.58	0.40 ± 0.58

Conclusions

It can be concluded that systematic influences originating from transforming the measurement outcomes to a form that is appropriate for averaging can introduce substantial systematic influences into the averages, if results near the detection limit are abundant. To minimize the systematic effects on the average the maximal relative uncertainty, quoted in the reports, has to be optimised. This optimisation has to take into account two conflicting requirements: minimizing the probability of reporting the wrong results and minimizing the systematic influences on the average. Whereas according to the first requirement the maximal relative uncertainty should be set low, the second requirement demands a high relative uncertainty. The smallest systematic influence on the average can be achieved by using a maximal relative uncertainty, which is nuclide dependent. When the measurement outcome is dominated by the information extracted from a single determination, a relative uncertainty of 76% should be used. On the other hand, when the measurement outcome information from more determinations contributes, the maximal relative uncertainty should be set to a smaller value.

If the averages are used in dose assessments, great care should be taken in order to minimize the systematic influences on the calculated doses. Since doses should be assessed realistically, they must be free of impacts originating in the analytical process as well as in the way the measurement results are presented.

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Environmental tritium monitoring techniques applied for a tritium removal facility

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Abstract

Nuclear and radiological installations are subjected to the regulatory control throughout their life. A preoperational monitoring program must be designed to assess the background radioactivity level in the area impacted by any future nuclear or radiological activity and shall cover at least one year before the installation will start to be operated. The paper presents the analytical techniques that were developed, optimized and applied during such an environmental monitoring program which addressed the impact area of a pilot tritium removal facility. Since the radionuclide of concern was tritium, a difficult to measure radionuclide, various methods were applied to extract and purify the water from air, vegetation and food products, which are presented in the paper. Due to the low background tritium measurement method applied, a seasonal variability was exhibited as concerns the tritium in air concentration, which confirms the natural origin of tritium background.

Radioecological studies in the Barents Sea (results of expedition in 2007–2009)

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Abstract
The results of radioecological investigations carried out within the framework of the Russian-Finnish high-latitude expeditions in 2007-2009 are presented.

Introduction
Since 1991, the Murmansk Marine Biological Institute of the Kola science center of RAS (MMBI KSC RAS) has carried out comprehensive investigations of radioactivity of Arctic and Subarctic marine ecosystems, which are based on extensive documentary, geographic, and taxonomic materials. Interest in this region stemmed from the availability of potential regional and local sources of radionuclide emission: atomic fleet bases, nuclear test sites on the Novaya Zemlya Archipelago, and radioactive waste burial sites on the shelf.

MMBI KSC RAS implements annual scientific expeditions to the Barents Sea. In 2007 and 2009, a scientist from Radiation and Nuclear Safety Authority (STUK) joined MMBI expeditions. The objective of collaboration was to study contemporary distribution of radioactive contaminants, anthropogenic and natural (NORMs) radionuclides in the Barents Sea ecosystem.

Material and methods
The expeditions were carried out with MMBI's research vessel RV Dalnie Zelentsy. The vessel is equipped to carry out extensive marine research for wide field of disciplines. The area and the route of the last three expeditions (2007-2009) covered parts of the Barents Sea which included standard section No. 6 (Kola Section) in the Central Barents Sea and the Franz Josef Land area. The focus of the studies in 2007-2008 was primarily on the Eastern Barents Sea, when sections along the northeastern border of the Sea, the area near the Novaya Zemlya western coast, and section via the trenches in the southeast of the Sea were investigated. In 2009, the focus was on the Western Barents Sea, when investigations along standard sections No. 3 and No. 19 and in the Svalbard/Spitsbergen area were carried out (Fig 1).

Sampling of seawater, bottom sediments, and living organisms was carried out during the expedition. Bottom sediments were sampled by Van-Veen grab where the top layer (0-3 cm) was collected. Cesium isotopes' pre-concentration from 100-litres of sampled water was performed by the cellulose-inorganic sorbent «Anfezh» on board the vessel. The samples' analysis was carried out jointly at the laboratories in Murmansk (Russia) and Rovaniemi (Finland). The samples were prepared before measurements. The activity concentrations of ^{137}Cs, ^{40}K, ^{226}Ra, ^{228}Th were measured with low-background gamma-spectrometer. The content of ^{238}Pu, ^{239,240}Pu, and ^{210}Po isotopes was determined by alpha-spectrometric method and the content of ^{90}Sr was determined by beta-radiometric method after the corresponding radiochemical preparation of the samples.

Results

Research results of the previous years show very low levels of anthropogenic radionuclides/radioactivity in all components of the Barents Sea ecosystem. The average ^{137}Cs content in seawater was about 2 Bq/m^3, and of ^{90}Sr − 5 Bq/m^3 (Fig. 2). The specific activity of ^{137}Cs and ^{90}Sr in bottom sediments varied within the range of 1−8 and 0.2−4 Bq/kg of dry weight respectively (Fig. 3). However, there are some regional differences in radioactivity concentrations. Higher concentrations of radioactivity were found, in bottom sediments and in sea water, in western parts of Barents Sea where Atlantic waters are more dominant. In proportion, areas where Arctic waters are more dominant show clear reduction in radioactivity concentrations. In addition, in bottom sediments increase in radioactivity concentrations were also observed in straights and deep trenches of the Barents Sea were excess radioactivity has accumulated in a sedimentation process (Fig. 2 and 3).The ^{238}Pu concentrations in bottom sediment along the standard section No. 6 were 0.02-0.04 Bq/kg and ^{239}Pu concentrations 0.4-1.3 Bq/kg, respectively. Thus the ^{238}Pu/^{239}Pu ratio in the sediment samples was 0.03-0.04 indicating that the Plutonium is of the global fallout origin relating to atmospheric nuclear weapon testing. In addition to water and sediment samples some samples of Barents Sea fauna were also measured. The ^{137}Cs concentrations in the main commercial fish species were about 0.2 Bq/kg of wet weight. This is very low compared to ^{137}Cs concentrations of tens of Bq/kg in fresh water fish in Northern Scandinavia (AMAP, 2009). The radioactivity in the Barents Sea nowadays is primarily determined by the presence of natural radionuclides (NORMs) (Table 1).

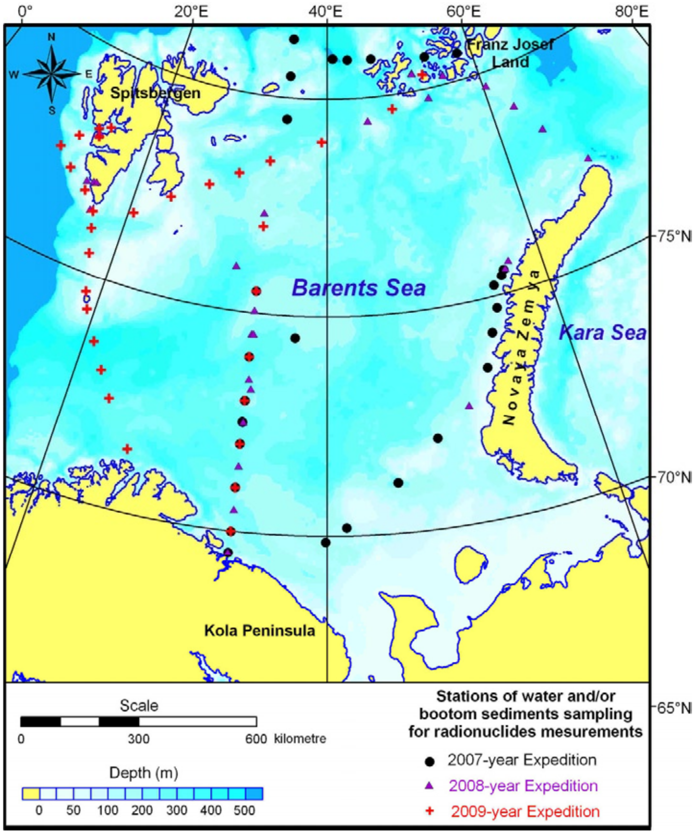

Fig. 1. Stations of water and bottom sediments sampling during MMBI's annual high-latitude expeditions, 2007-2009.

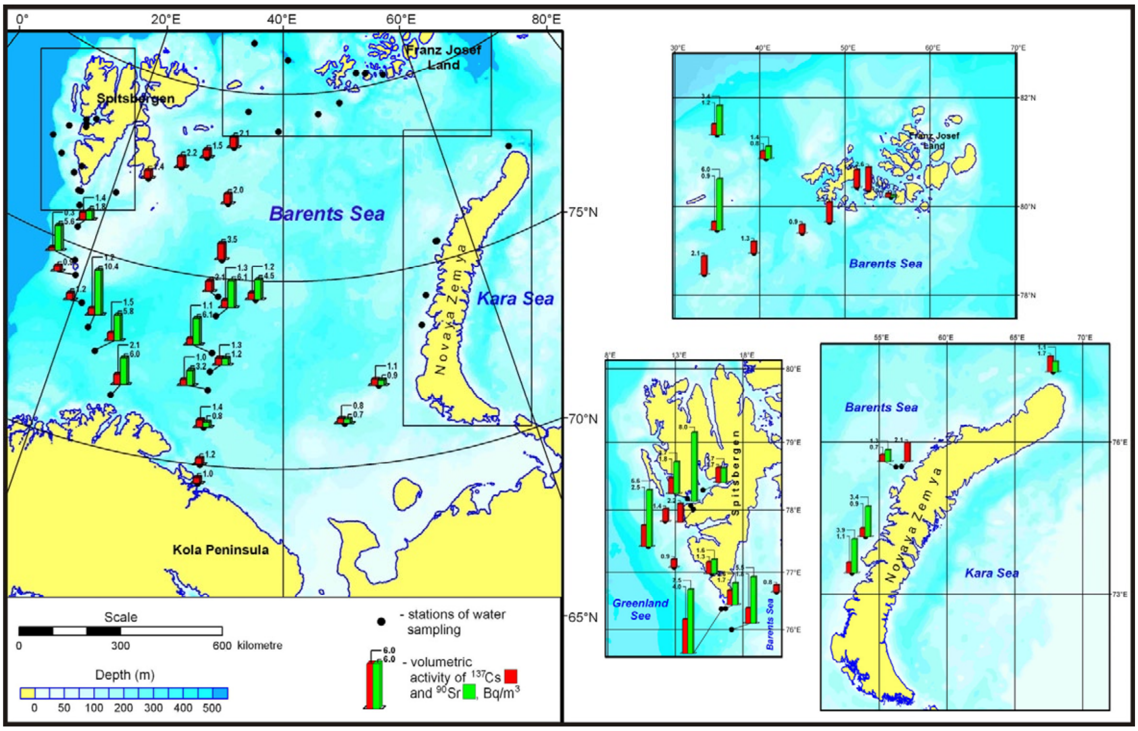

Fig. 2. Volumetric activity of ^{137}Cs and ^{90}Sr in the Barents Sea water masses, 2007-2009.

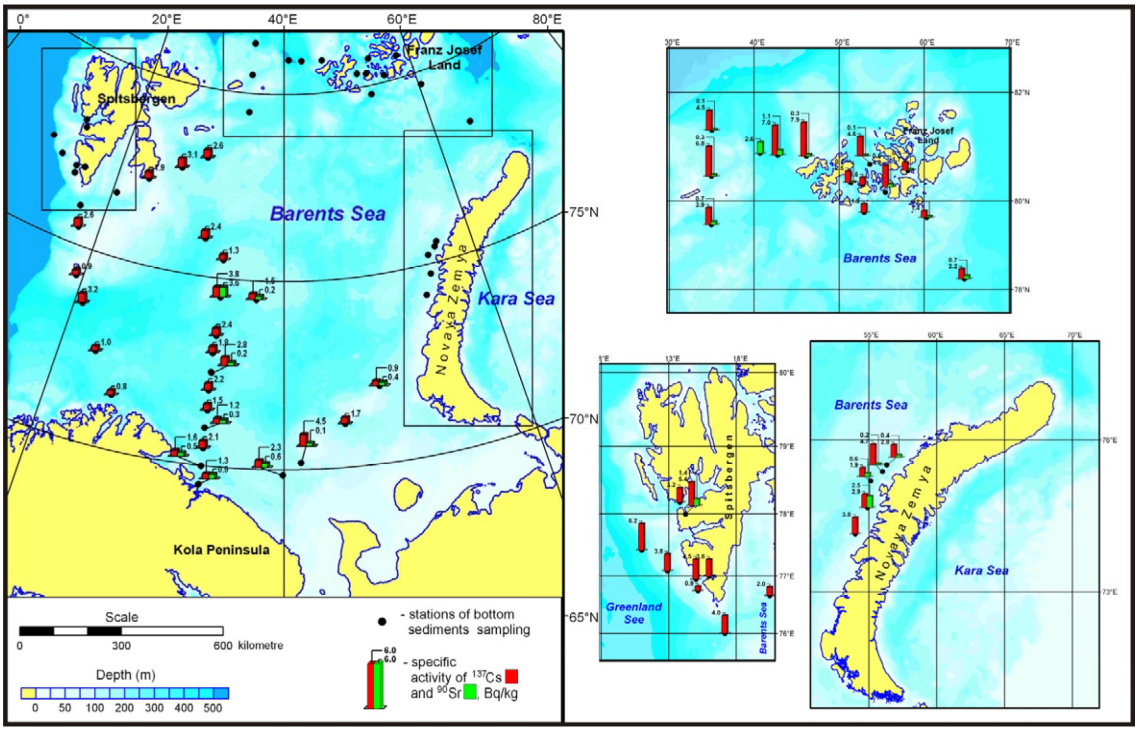

Fig. 3. Specific activity of ^{137}Cs and ^{90}Sr in the Barents Sea bottom sediments, 2007-2009.

Table 1. Gamma radionuclides in Barents Sea fish, 2007-2009, Bq/kg wet weight.

Species	^{137}Cs, Bq/kg wet weight	^{40}K, Bq/kg wet weight
Long rough dab *(Hippoglossoides platessoides)*	0.10±0.04	106±12
Atlantic Cod *(Gadus morhua)*	0.20±0.08	109±12
Atlantic Cod *(Gadus morhua)*	0.15±0.08	116±12
Atlantic Cod *(Gadus morhua)*	0.15±0.06	105±12
Haddock *(Melanogrammus aeglefinus)*	0.07±0.04	116±12
Spotted wolffish *(Anarhichas minor)*	0.20±0.05	115±13
Spotted wolffish *(Anarhichas minor)*	0.15±0.05	99±12
Haddock *(Melanogrammus aeglefinus)*	0.12±0.05	110±11
Haddock *(Melanogrammus aeglefinus)*	0.12±0.05	108±10
Haddock *(Melanogrammus aeglefinus)*	0.10±0.05	116±12
Haddock *(Melanogrammus aeglefinus)*	0.16±0.05	106±10

Conclusions

As a whole, according to the contemporary levels of radioactive contamination, almost the entire Barents Sea may be considered as an area very minor affects of human activity. The local influence of anthropogenic factor was observed mainly in water areas affected by economic activity in bays and inlets of the Barents Sea. Although the radioactivity of the marine environment is generally decreased, it is important to carry out regular monitoring especially in the regions where the potential risk of radioactive pollution is high.

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Experimental study of the radionuclides transport in soil and plants from waste dump

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Abstract

The transport of radionuclides through terrestrial environments is determined by a multiplicity of processes with time-scales ranging from a few minutes to many years. The importance of understanding and predicting the radionuclide migration from soils to vegetation arises from its potential radiological impact: slow migration has as result an increase availability of radionuclides for root uptake and gives rise to external doses for a long time.

The transfer factors (TF) for natural uranium isotopes (^{234}U and ^{238}U) and ^{226}Ra were obtained in vegetation samples growing in granitic soils around disused uranium mine located in the Ciudanovita region in the West of Romania. Affected and non-affected areas of the mine presented large differences in the activity concentrations of radionuclides of the uranium series. We also determined transfer factors for several stable elements (essential and non-essential). A set of statistical tests were applied to validate the data.

Introduction

It is important to understand the behaviour of natural radionuclides in the environment (distribution pathways, mobility, transfers, etc.) because the information can be used as a natural analogue for the long-term behaviour of materials and processes in developing and testing models, and in obtaining the associated parameter values appropriate for radiological performance assessments [4]. After the exploitation of the mineral, the uranium mine enters a phase of inactivity until its restoration. During both these phases, large amounts of materials, more or less rich in radionuclides, are exposed to environmental agents. In the phase of inactivity, the uranium mine can be regarded as a very appropriate natural laboratory to investigate the mobilization of natural radionuclides by means of their distribution in different compartments (soil, vegetation, water, sediment, etc.), as well as the transfer between them. The soil-to-plant transfer factor is one of the important parameters widely used to estimate the internal radiation dose from radionuclides through food ingestion. In general, transfer factors show a large degree of variation dependent upon several factors such as soil type, species of plants and other environmental conditions. Due to a predicted long-term transfer of radionuclides in the environment, an understanding knowledge of the geochemical and ecological cycles is also needed as they relate to the behaviour of radionuclides. In addition, the distribution of radionuclides in plant components is beneficial in understanding the dynamics of radionuclides in an agricultural field. Because non-edible parts of agricultural plants are returned to the soil, they may again be utilized in the soil-plant pathway. As a case study, in this work has been considered the uranium waste dump reservoir from the mining perimeter of Ciudanovița.

Material and methods

The environmental samples like soil and vegetation was collected from the dump of the waste rock of the mine EM Banat, Oravita, Caras-Severin county, in order to evaluate soil to plant radionuclide transfer factors (TF). The definition of TF proposed by the IUR [5] has been used:

$$TF = \frac{concentrationofradionuclideinplant(Bqkg^{-1}drycropmass)}{concentrationofradionuclideinsoil(Bqkg^{-1}drysoilmas\sin theupper20cm)}$$

The vegetation samples at each point were collected from the surface at which the soil sample had to be removed. In all cases, only the aerial fraction was sampled. About 100 g of a representative fraction was obtained by homogenization after drying and chopping. The samples were carefully washed in the laboratory in order to remove all the adhered soil particles. Materials for sampling and pre-treatment of the samples were always cleaned before initiating each procedure in order to avoid cross contamination. Twenty-five sampling campaigns soil and vegetation were performed during a one-year period (2007-2008) in order to take the variability in meteorological conditions into account. The total uranium concentration and the isotopic composition from vegetation and soils samples collected at the same sampling site, were determined by high resolution ICPMS Element 2 after separation and preconcentration of uranium. Content of ^{226}Ra, ^{234}U and ^{238}U from the solid waste and vegetation samples were determined by means of the spectrometric gamma measurement chain HPGe-Oxford, with 40% efficiency. Processing of experimental data was performed using STATISTICA 6.0 software which used methods Quadrat Surface and spleen, with a confidence coefficient set to introduce the experimental values of 95%.

Results

STATISTICA 6.0 has enabled the distribution of transfer factors for ^{226}Ra, ^{234}U and ^{238}U, depending on the concentrations of both radionuclides in soil and vegetation in 25 point of prelevation. These distributions are show in the graphs in Fig. 1-2-3. Radium is the last member of the alkaline earth metals, a group of metals whose lighter members (Ca and Mg) play a very important role in plant growth and nutrition. The TF values show a very wide variability for ^{226}Ra. Results have been obtained for radium in grass with a TF mean value of 4.705, and a range between 0.07-9.34. The statistical analysis of the results indicates that the TF mean values in grass corresponding to the ^{234}U are 0,17, and a range between 0.03-0,17, and the TF mean values for ^{238}U are 0,355 and a range between 0,05-0,66. It was established correlations: the concentration of ^{226}Ra (soil) - the concentration of ^{226}Ra (grass) -TF (relationship 1) and the concentration of ^{238}U (soil) - the concentration of ^{238}U (grass) -TF (relationship 2), which are plotted in Figures 1 and 3.

$$\text{TF } ^{226}\text{Ra} = 61,635+2,0345x-20,434y+0,0181x^2-0,3817xy+1,8685y^2 \quad (1)$$

where: x= the concentration of ^{226}Ra in soil [Bq/kg] and y= concentration of ^{226}Ra in grass [Bq/kg]

$$\text{TF } ^{238}\text{U}_{nat} = 36,9526+1,0011x-11,5731y+0,0088x^2-0,1876xy+1,0083y^2 \quad (2)$$

where: x= the concentration of ^{238}U in soil [Bq/kg] and y= concentration of ^{238}U in grass [Bq/kg].

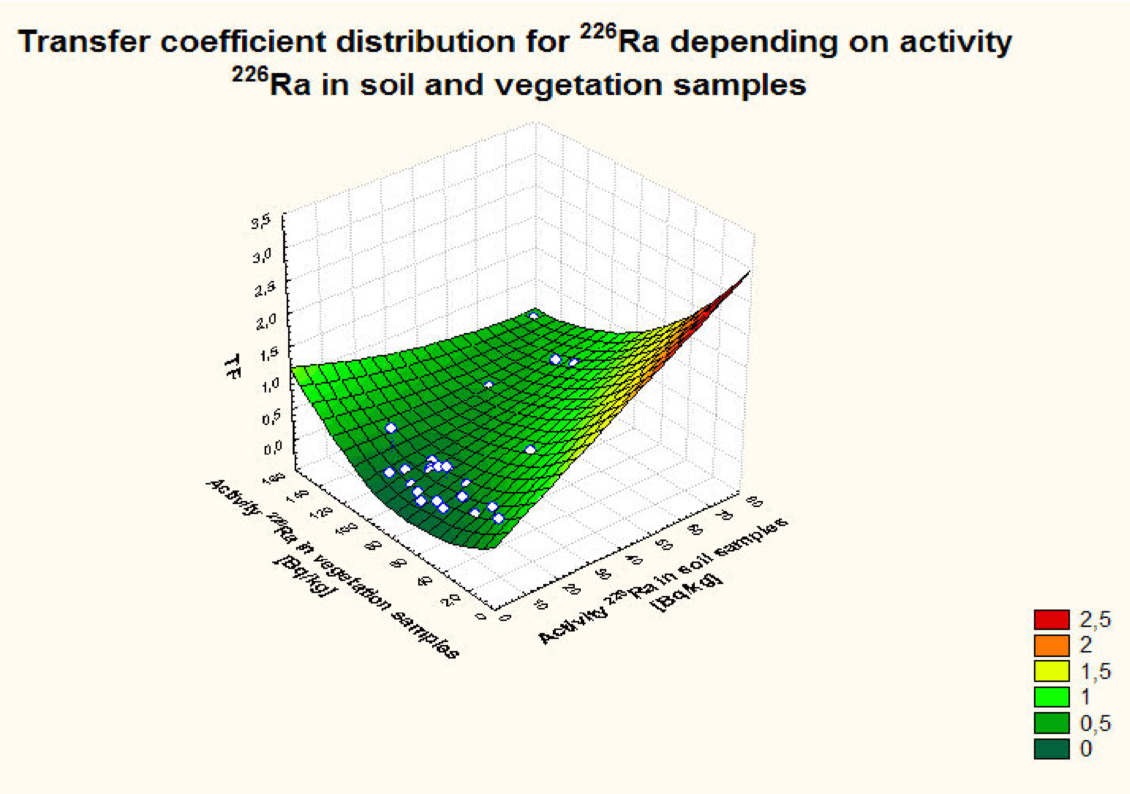

Fig. 1. Distribution of the TF values for ^{226}Ra corresponding of the activity concentrations of ^{226}Ra in soil and grass.

The TF corresponding to ^{226}Ra can be considered, at a 95% confidence level, higher (by two orders of magnitude) than uranium isotopes studied. The excess of ^{226}Ra in vegetation versus ^{234}U and ^{238}U must be explained by the higher absorption of radium. From the comparison between the TF values corresponding to the two elements studied (uranium, and radium), we conclude that the uptake for radium is higher than for the other once element. The fact that uranium is both actinides may explain their more analogous chemical behaviour, whereas this argument cannot be extended to radium which is an alkaline-earth.

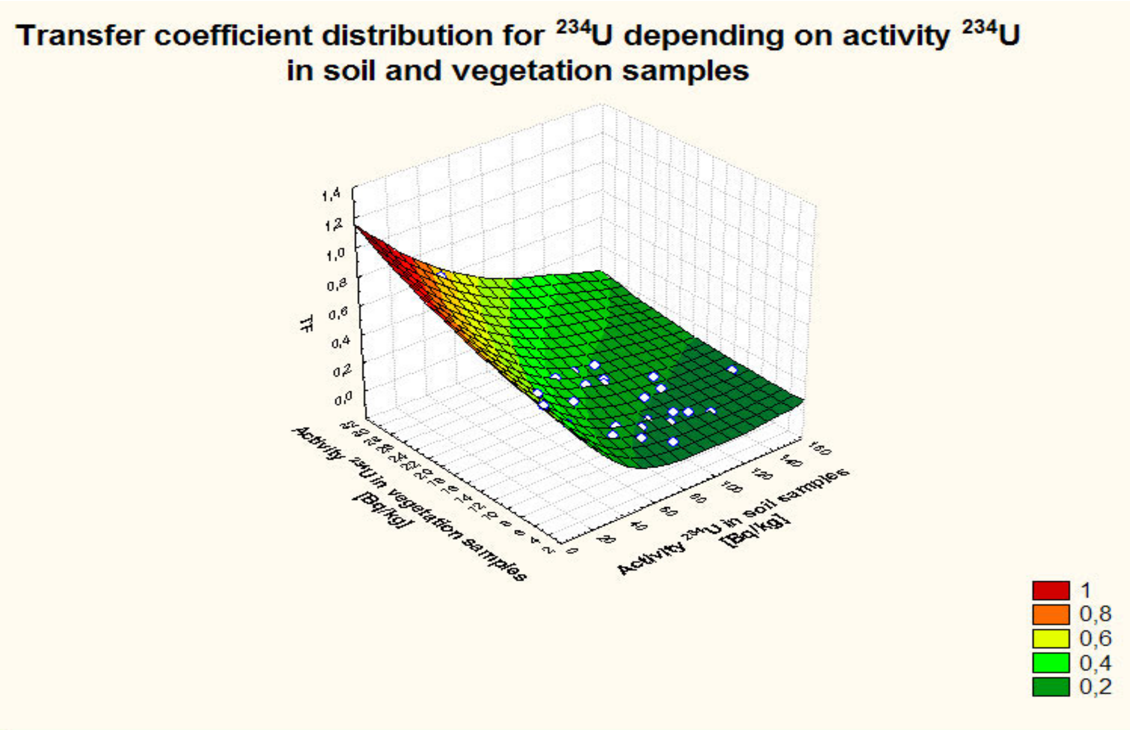

Fig. 2. Distribution of the TF values for ^{235}U corresponding of the activity concentrations of ^{235}U in soil and grass.

This general result is in agreement with other afirmation that TF values for elements in oxidation state +II are almost always greater than for those elements in oxidation state +IV. The results obtained for the soil samples show that the $^{234}U/^{238}U$ atom ratios in all soil samples, are clearly higher than the natural $^{234}U/^{238}U$ atom ratios, 5,54019E-05. Higher $^{234}U/^{238}U$ ratios in soil were observed at 100 and 200m from uranium dump, decreasing with further distance from the uranium tailing dump. When aquatic systems are in contact with minerals, selective leaching processes lead to preferential dissolution and transport of ^{234}U, resulting in enhancement of $^{234}U/^{238}U$ ratio. The reason underlying the enhancement of the ratio is attributed, as a major cause, to a "recoil induced vulnerability to leaching". The mechanism is view as a creation of defects in the crystal lattice when the parent nuclide (^{238}U) recoils during emission of an alpha particle, thus the daughter nuclide (^{234}U) is in a microenvironment that is more susceptible to chemical attack than the parent [1]. Electron stripping during the decay process such that ^{234}U is more likely to be in more soluble U(VI) state, facilitating the solution of this isotope by a surface etching process [2]. In the vegetation samples collected up to 200 m from the uranium tailing dump, half of the samples analysed presented an enhancement of the ratio $^{234}U/^{238}U$ ratio being always higher in grass than in Tussilago farfara. The mobility of uranium in plant tissues is limited, as it tends to adsorb on cell wall materials; therefore, concentrations are typically higher in tissues found lower on the plant and are highest on the root surfaces [3].

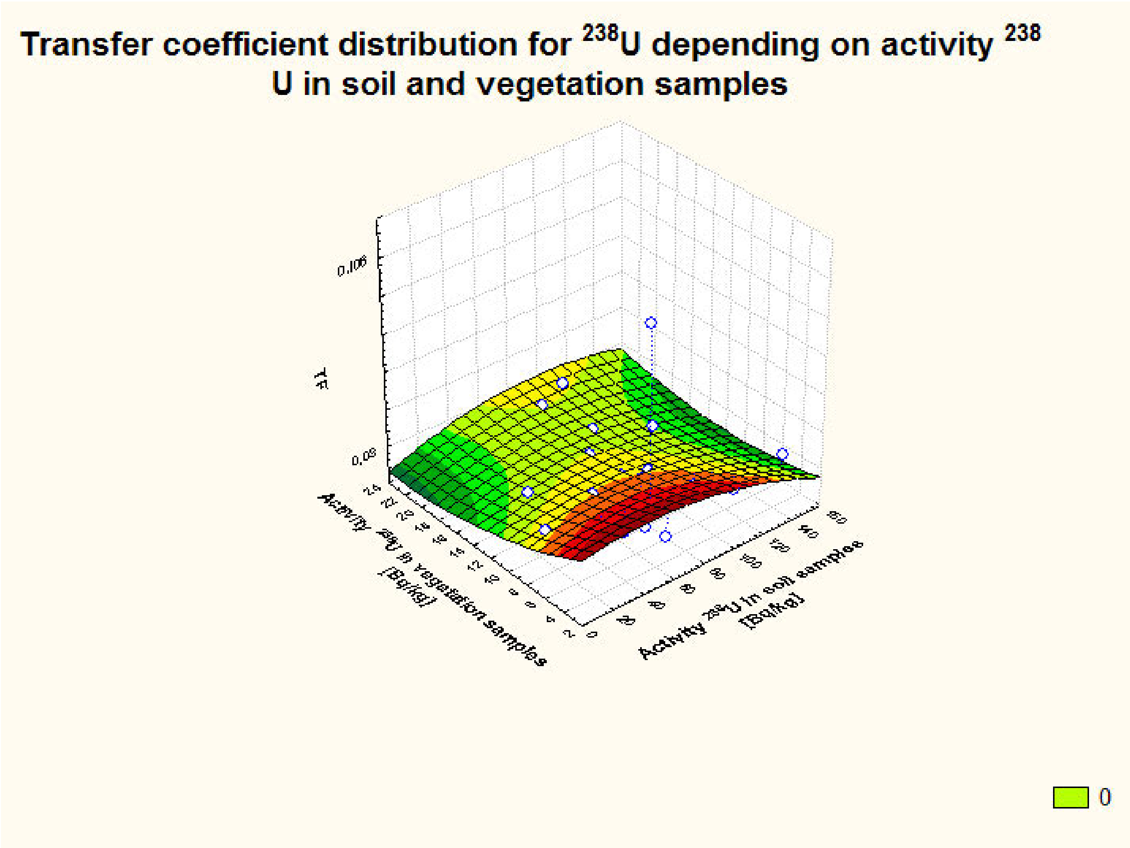

Fig. 3. Distribution of the TF values for ^{238}U corresponding of the activity concentrations of ^{238}U in soil and grass.

In fig.4 is shown correlations between ^{234}U/^{238}U ratios in grass and Tussilago farfara and in soil where the plants were grown.

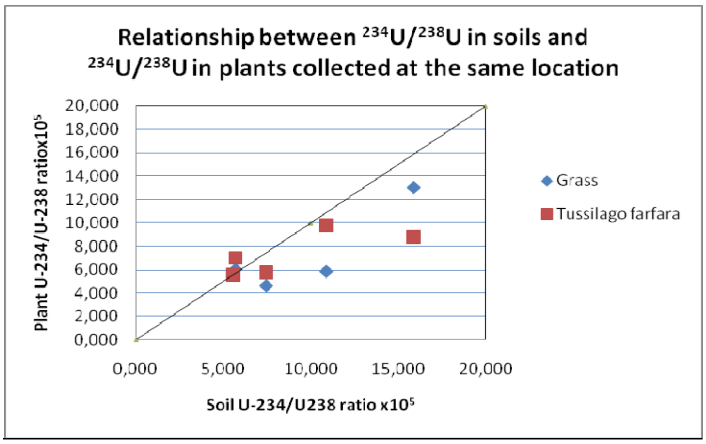

Fig.4. Relationship between ^{234}U/^{238}U ratios in soils and ^{234}U/^{238}U ratios in plants collected at the same location.

Although these plants were grown and harvested simultaneously, we can see that both ^{234}U and ^{238}U were more easily transferred from soil to roots of grass as compared as compared to transfer of these radionuclides from soil to roots Tussilago farfara.

Plant radionuclide concentrations are not so often linearly related to soil radionuclide concentrations. Nonlinearity can complicate the measurement of bioavailability, because each plant and soil combination may have a unique curvilinear relationship. The study of temporal variations of ^{234}U and ^{238}U in plant showed that short-term dynamics of radionuclide plant concentrations are rather significant and species-specific.

Conclusions

Transfer factors (TF) for different natural radionuclides (^{226}Ra, ^{234}U and ^{238}U) have been presented for grass samples in an area where a disused uranium mine is located.

The radium uptake is greater than for uranium by about some orders of magnitude. These differences have been attributed to the different solubilities of the elements with oxidation state +II. The Ra-transfer factor depends the plant part concerned, climate conditions, and the physicochemical form of radium. Radium has a high affinity for the regular exchange sites of the soil.

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HYDRUS-computer simulation of radionuclide migration in groundwater due to clearance of low-level waste from decommissioning

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Abstract
Bulk amounts of cleared building rubble mainly arise due to decommissioning of nuclear power plants. Depending on the type of clearance, weakly contaminated rubble can be released to be deposited in landfills. Leaching of radionuclides is caused by infiltrating rainwater and may lead to migration of radionuclides through the landfill and the vadose zone into the aquifer where the contaminated seepage is mixed with groundwater. If contaminated groundwater is used for irrigation or direct consumption, the food chain may be affected (water pathway). Clearance levels for radionuclides have to be calculated in such a way that the water pathway is considered and the effective dose for an individual of the public is at most of the order of 10 micro Sievert per year. In the wake of availability of inexpensive and fast computers, the leaching of radionuclides, their transport through the various zones of rubble and soil and eventually the contamination of groundwater can be computed by means of sophisticated mathematical models. The software package HYDRUS constitutes an internationally established standard in contaminant transport modeling. We apply HYDRUS to problems of nuclide transport and water dynamics in landfills, the vadose zone and the aquifer.

Introduction
Most of the building rubble from decommissioning of nuclear power plants has low radioactivity and can be released for disposal in landfills, provided the activity concentration (in Bq/g) of the material per radionuclide is below the respective clearance level. In Germany, clearance levels are part of the radiation protection ordinance. Following progress in science and technology and adapting to changing conditions concerning, among others, new requirements of new waste legislation, they are continuously updated.

An important radionuclide pathway in the case of low level material contained in landfills is the so-called water pathway. The assumption is that radionuclides contained e.g. in concrete rubble are carried away by rainwater. Radionuclides may migrate through the landfill and the unsaturated vadose zone beneath the landfill. The contamination may also reach the aquifer and eventually the drinking water and the food chain.

As new approach to estimate groundwater activities resulting from disposal of concrete rubble, the radionuclide transport in the landfill, the vadose zone and the aquifer was calculated in one dimension by means of the computer program HYDRUS (Simunek et al. 1998). With HYDRUS, water transport, leachability of radionuclids and radionuclide migration through the porous structures of rubble, soil and the matrix of the aquifer was simulated numerically. The main equations used within HYDRUS are the Richards equation (water transport) and the convection-diffusion equation (CDE, for the nuclide transport). The computer program HYDRUS represents an internationally established standard tool in the area of groundwater hydrology and contaminant modeling.

With the Richards equation considered here, wetting, drying and the water movement through porous material is adequately described also for non-saturated water contents and precipitation can be varied according to the annual seasons. Physical effects for radionuclide migration considered in our model include the advective transport of radionuclides, their sorption to the background matrix (e.g. rubble or soil), diffusion and dispersion. Also, the radioactive decay is considered as a sink term in the transport equation. Altogether, the HYDRUS code allows for calculation of groundwater activity values. The results are compared with values from the IAEA model of the water pathway suggested for exemption and clearance (IAEA Safety Reports Series No. 44, cf. (IAEA 2005)).

Material and methods

Water transport model

The fundamental physical basis used for the water transport is Darcy's equation. It desribes fluid flow through porous media. It is known that soil and the aquifer are best described as porous matrix. The same is assumed for rubble in a landfill.

Usually, Darcy's equation for the water velocity is written as (v: Darcy velocity, K: hydraulic conductivity, h: hydraulic head, l: spatial coordinate, ψ: pressure head, cf. the monograph by Freeze and Cherry (1979) for all details on groundwater hydrology)

$$v = -K(\psi)\frac{dh}{d\ell}$$

In Hubbert's analysis of the fluid potential, the relationship between different pressures within a system of fluid, air and porous medium is expressed in terms of manometer heads and written as $h = z + \psi$, where z (elevation head) measures the vertical distance to the reference point z=0 (datum) and ψ represents the suction effects of the porous material. It is thus the total gradient of both elevation and suction that drives the water flow through a porous medium.

The equation for v was established around 1856 by French engineer Henry Darcy following his engineering work related to water supply. Nowadays, it can be derived from the more general Navier-Stokes equations in case of water flowing through a porous matrix. This is the basic situation in a saturated aquifer. However, physics becomes more involved once the pore space is only partially saturated, which is the case in the vadose zone and within a landfill consisting of rubble. Besides ψ, the moisture content θ (0 ≤ θ ≤ n, with n: porosity) becomes an important characteristic of the system.

In unsaturated material, K and θ may be related to ψ. The most commonly used empirical relationship between these quantities for soil material is named after M. van Genuchten and Y. Mualem as van Genuchten-Mualem model (VGM). In principle, this model leads to S-shaped wetting and drying curves saying that conductivity is increasing as moisture content is approaching saturation. The curves have to be inserted into the respective water transport equations with the consequence that a considerable degree of complexity is added to the theory. Furthermore, the main experimental problem is to determine the curve parameters for a given soil type. For example, for concrete rubble, these parameters are usually unknown and hence the water dynamics cannot easily be predicted.

For unsaturated media, the Darcy equation can be coupled with the equation of continuity to yield what is known as the Richards equation (Lorenzo Richards, 1931):

$$\frac{\partial \theta}{\partial t} = \frac{\partial}{\partial x}[K(\frac{\partial \psi}{\partial x} + 1)]$$

In most of the models of the vadose zone, the Richards equation is the starting point for a mathematical analysis of the fluid flow. It can be assumed to hold for concrete rubble as well, at least as long as a sufficient amount of fine-grained material is present.

While hydraulic parameters of the VGM are tabulated for many soil types, available data for rubble is sparse. It is not clear, for example, whether data for coarse-grained sand is sufficient to describe wetting and drying dynamics in a landfill. Therefore, the hydraulic properties of concrete rubble are presently determined by Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) in collaboration with the Technical University Braunschweig as part of a consultant contract with the German Federal Office for Radiation Protection (BfS). A preliminary set of values was calculated from multistep-outflow experiments by means of inverse modeling.

Radionuclide transport model

Radionuclide transport in water is described by the convection diffusion equation (CDE). In the HYDRUS model, the CDE approach was adopted for radionuclide transport in water flowing through the porous media of concrete rubble, unsaturated soil (vadose zone) and saturated aquifer (Freeze and Cherry (1979). For undisturbed soils, the validity of the CDE approach was demonstrated by Bossew and Kirchner (2004)). The CDE is basically a differential equation expression of mass conservation,

$$\frac{\partial(nC + \rho_b \overline{C})}{\partial t} + \overline{v}\frac{\partial}{\partial x}nC = \frac{\partial}{\partial x}D\frac{\partial}{\partial x}nC - \lambda\rho_b\overline{C} - \lambda nC$$

(C: radionuclide concentration in water, $\overline{C}$: sorbed nuclide concentration, D: dispersion coefficient, λ: decay constant, ρ_b: solid material density).

The dispersion coefficient includes both the effects of diffusion and of hydrodynamic dispersion caused by the porous material structure. Initially sharp and narrow concentration profiles are thus smeared out after travelling through the porous material, resulting in lowered local concentrations.

Sorption is described by a linear relationship between C and $\overline{C}$, which is known as K_d model

$$\overline{C} = K_d C .$$

The CDE can be reduced to an equation for the dissolved fraction C only. However, migration of the dissolved fraction is retarded: high values of K_d therefore result in slowly progressing contamination fronts.

Modeling the water pathway with HYDRUS-1D

In the HYDRUS numerical code, all of the equations described above are implemented. These differential equations are coupled and usually cannot be solved analytically. HYDRUS attempts a numerical solution of both Richards equation and CDE by applying finite difference and finite elements methods. HYDRUS was validated by BfS against known solutions of the CDE under simplified conditions. The HYDRUS output proved to be in excellent agreement with the known solution. In addition, various test runs were performed before modeling started. For example, in a numerical scheme it can be important to control the (finite) timestep and the resolution of the numerical spatial grid.

We have applied the one-dimensional version of the code to a model situation as depicted in Fig. 1. The landfill is located on top of vadose zone and aquifer, and rain is assumed to infiltrate the landfill. Radionuclides migrate in vertical direction and enter the aquifer where the seepage is diluted by incoming groundwater.

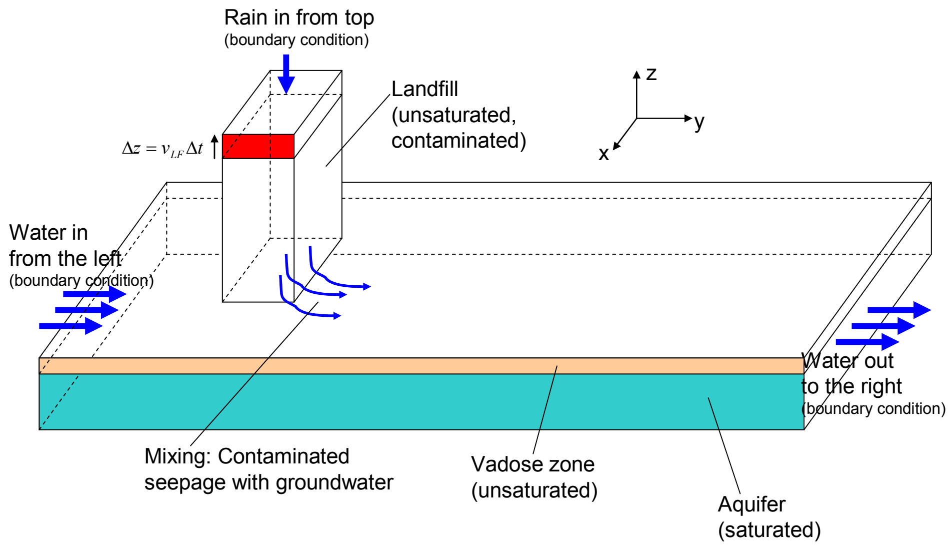

Fig. 1. 3D-Illustration of the model situation used. Modeling is performed by combining two 1D-Hydrus computer runs, one for the landfill and vadose zone and another one for the aquifer.

Two one-dimensional computer runs are necessary in the model situation described in Fig. 1, the first one comprising landfill and vadose zone. Dilution with groundwater is performed analytically and the result is inserted into a second simulation for the aquifer.

The IAEA (SR 44) water pathway model

The IAEA modeling approach is a straightforward analytical method to obtain generic activity levels of contaminated groundwater and was inspired by the RESRAD family of computer codes (Yu et al. 2007). In the IAEA Safety Reports Series No. 44 (IAEA 2005), the water pathway is part of a set of exposure scenarios. Altogether, the exposure scenarios serve to calculate so-called activity concentration (AC) values (in Bq/g). The IAEA suggests to use AC values for exemption and clearance.

The IAEA model is not designed to solve differential equations, rather, it is based on simple estimates and conservative assumptions. It applies the concept of K_d and retardation and directly results in radionuclide concentrations in the seepage and groundwater. Since it does not solve differential equations, the time evolution of the contamination is impossible to follow, which is certainly one of the major drawbacks of this model. Furthermore, dispersion remains unconsidered and the dynamics of the water flow is neglected.

Basically, the model starts with a contaminated zone containing a radionuclide of 1 Bq/g specific activity. It is assumed that the entire inventory of radionuclides is carried away by rainwater at a leach rate of

$$L = \frac{I}{\theta^{cz} z^{cz} R^{cz}} \quad .$$

In this equation, I stands for the infiltration rate, θ^{cz} stands for the volumetric water content of the contaminated zone, z^{cz} is the thickness of the contaminated zone and R^{cz} is the retardation factor. R^{cz} is largely determined by K_d . Note that the volumetric water content is taken to be a constant, since full water dynamics cannot be modelled with the IAEA model.

The radionuclide concentration in the seepage can be calculated according to the formula

$$C^s = \frac{McL}{IA^{cz}}$$

(M: total mass of contaminated material, c: specific activity of the radionuclide in the contaminated material, A^{cz} is the surface area of the contaminated zone). The explanation of these formulae is in the end straightforward application of proportionalities.

An unsaturated zone is assumed in the model. For this zone, a delay time t_i is inserted into the law for radioactive decay, which considers the time needed to traverse the unsaturated zone. There is no further impact from the assumption of an unsaturated zone. This is a considerable simplification and, in principle could only be justified retrospectively, after a detailed model of the vadose zone has been calculated. The delay time is calculated on the basis of proportionalities from infiltration rate, thickness of the unsaturated zone, retardation factor, saturation ratio and porosity of the unsaturated zone.

In the end, the radionuclide concentration in the well water is calculated from the radionuclide concentration in the seepage by

$$c^w = \frac{IA^{cz}}{U^{gw} + IA^{cz}} C^s e^{-\lambda t_i} .$$

Here, U^{gw} is the volume of groundwater per unit of time. The first term in the last equation represents dilution by groundwater. There is no detailed modeling of the aquifer and obviously, the well has to be assumed conservatively as being located very close to the landfill.

Results

In the HYDRUS-modeling approach, two one-dimensional HYDRUS computer runs are combined to calculate the groundwater contamination at a distance of 500 m from the landfill. The first run is for the downward directed water and radionuclide transport through the landfill and the vadose zone.

We have used hydrological data from standard hydrology literature to model in a preferably realistic rather than overly conservative manner (see, for example, the monographs by Freeze and Cherry (1979) and Heath (1988)). Hydraulic data for the vadose zone and the aquifer is provided by HYDRUS (Simunek et al. 1998) within the VGM framework. Respective VGM data for the landfill was determined experimentally by GRS. In the HYDRUS model, an annual precipitation typical for Northern European climate conditions was assumed. An initial activity of 1 Bq/g was taken for the radionuclides contained in the rubble.

Usually, we apply K_d values that are the conclusion of a literature survey within a study performed by a technical consultant of the German Ministry of Environment, Nature Conservation and Reactor Safety (BMU) and published by Deckert et al. (1993). However, note that BfS is also involved in both experimental and theoretical derivation of K_d values. We plan to analyze possible differences in a future investigation. Furthermore, for a preliminary numerical study such as the one sketched here, it is sufficient and easier to analyze if the same K_d is used in all segments of the model.

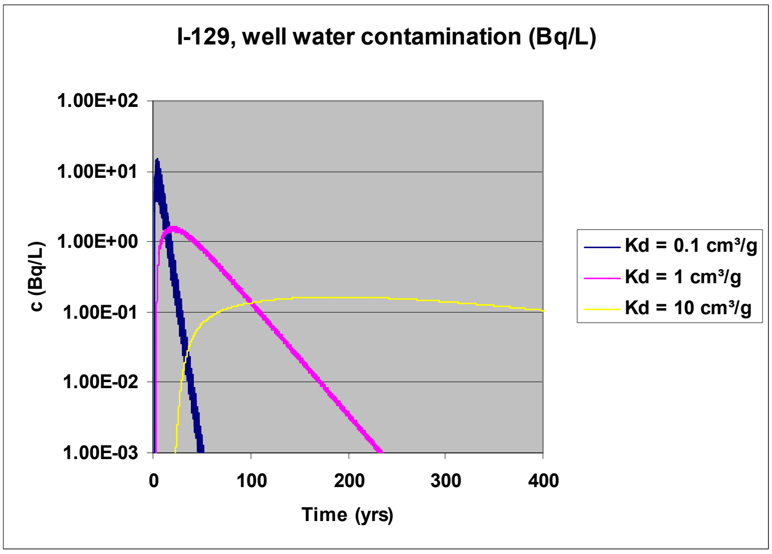

Fig. 2. I-129 well water contamination time scales (in years) for K_d = 0.1; 1; 10 cm³/g calculated with HYDRUS.

Fig. 2 shows HYDRUS-modeling results for I-129 (half-life 1.6×10^7 yrs). K_d was varied between 0.1 and 10 cm³/g, the baseline value being 1 cm³/g. The figure displays the evolution of I-129 groundwater contamination at a model well at a distance of 500 m from the landfill. Obviously, K_d controls the radionuclide transport dynamics to a large extent. Peak contamination levels span three orders of magnitude between around 0.1 and 10 Bq/l and the duration of contamination at the well varies considerably between about 50 yrs (K_d = 0.1 cm³/g) and more than 400 yrs (K_d = 10 cm³/g). Similarly, the arrival times at the well may vary depending on sorption. Although peak contamination levels can often be estimated from a given K_d , reliable contamination time spans are difficult to obtain without numerical analysis. Reasons for this are the dynamics of leaching and the dispersion effects along the path that tend to broaden the contamination peak structure.

In general, it was found that the IAEA modeling results are comparable to our HYDRUS results, provided free parameters are adjusted to ours. It is easier to study how the IAEA model *itself* performs if parameter values vary as little as possible between the models considered. Applying a simplified model can be useful whenever it is intended to get a rough idea of the course of data in a given context. The IAEA model can be programmed as a relatively short computer script and does not consume much computation time.

An example is shown in Fig. 3. Well-water contamination levels were calculated by means of our own computer scripts based on an IAEA model with parameters adjusted to the HYDRUS-model parameters.

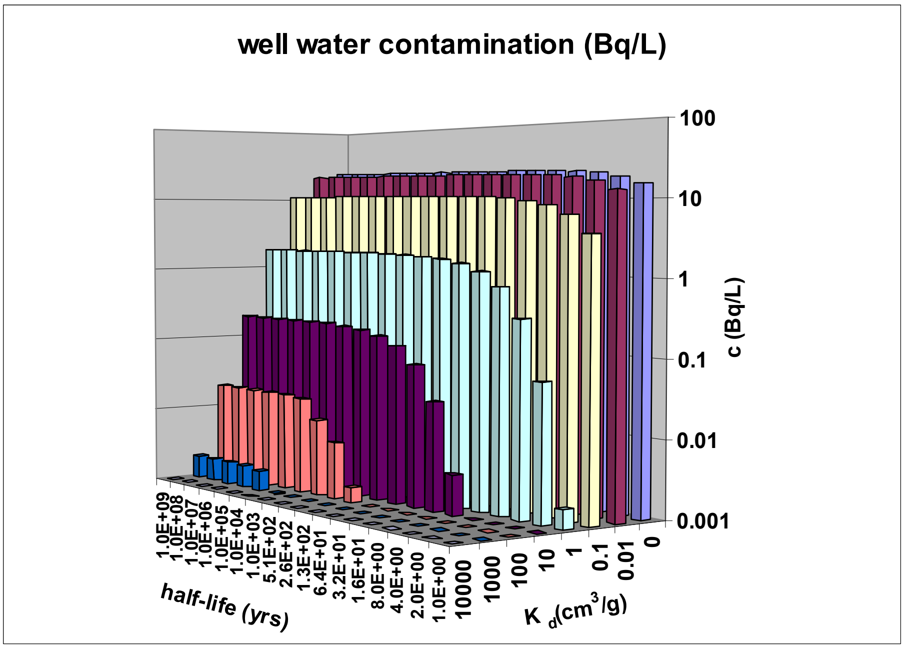

Fig. 3. The IAEA (SR 44)-modeling approach as applied to the parameter space spanned by K_d and half-life. The columns represent the calculated well water contamination values for a given pair of K_d and half-life. Other free parameters are chosen in accordance with the HYDRUS model (cf. text for details).

Due to computation time restrictions, general results are sometimes difficult to obtain by performing detailed HYDRUS simulations. For example, the plot depicted in Fig. 3 would require a total number of 272 HYDRUS-computer runs. On the other hand, if a site-specific study of great importance is needed, one should resort to a detailed computer simulation. The same holds if the transport process itself (e.g. the transport through a given segment of soil) is to be studied and compared with experimental data.

Conclusions

Well water activity levels (in Bq/l) resulting from leaching of radionuclides contained in concrete rubble were calculated by means of HYDRUS computer simulations. The computer program HYDRUS for one-dimensional water and nuclide transport through landfill, vadose zone and aquifer proved to be a robust and modern tool to assess groundwater activity values. Modeling results are similar to values obtained by applying the modeling strategy suggested by the IAEA in the publication (IAEA 2005). The simplified analytical IAEA model for the groundwater pathway appears to be well suited for general-purpose estimates, e.g. to derive generic clearance levels as indicators for a broad class of disposal situations.

In general, the calculation outcome largely depends on K_d . The determination of K_d values, however, was not part of our study so far. In the present approach, K_d is merely an input parameter and was not even varied along the radionuclide path. The same holds for the IAEA study. As the K_d concept appears to control the outcome of water pathway calculations to a large extent, the question may arise whether clearance level calculations based on a generic approach are trustworthy or not. If site-specific parameter values (e.g. K_d) are available by explicit measurement, detailed computer modeling is therefore to be preferred in site-specific clearance as far as the water pathway is concerned.

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Radioactivity in Trinitite – a review and new measurements

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Abstract

Samples of Trinitite and soil from Trinity site were studied in the radioactivity measurements laboratory at the University of Bremen and at the authors' facilities in Anaheim (JCR) and Edgewater (WMK). Gamma spectroscopy was used to identify and quantify radionuclides in Trinitite and to perform a radiometric characterization of soil at the Trinity site. Additionally, a similar material ("atomsite") formed during a soviet test at the Semipalatinsk nuclear test site was investigated. Fission products (^{137}Cs, ^{155}Eu) together with activation products (^{60}Co, ^{133}Ba, ^{152}Eu, ^{154}Eu, ^{241}Am) and ^{239}Pu were identified. A literature search including some publicly available archive sources was conducted and our data compared to previously published results. Obtained data on Trinitite were also compared to literature data on atomsite formed during atmospheric nuclear tests in Algeria. Variability of radioactivity in Trinitite and relationship of distance from the ground zero and activation were discussed.

Introduction

Trinity

The terms Trinitite or atomsite are used for a fused glass-like material formed during the first nuclear test in the desert in White Sands Missile Range, New Mexico, USA, on July 16, 1945. Its colour is usually greyish-green, the top surface (facing the explosion) is smooth and typically more active than the bottom side, which is rougher and contains sand and small stones from the desert surface. The thickness of the Trinitite layer varies usually between 0.5 and 1 cm and the material contains plenty of air inclusions. Trinitite was cleared from the area by the Atomic Energy Commission in 1952, but visitors of the Trinity site can still find small pieces on the ground. Although it is illegal to remove Trinitite from the site now, specimens can be found for sale on the internet and from mineral dealers. The Ground Zero (GZ) of the Trinity test, which is a National Monument since 1965, lies within the military area and can be visited by the public twice a year, always in April and in October (WSMR, 2010).

The physical properties of the glass formed during the first nuclear explosion are described in LA-1126 report (Staritzky 1950). The glass covered an area of about 610 m diameter with a total estimated mass of $17 \cdot 10^5$ kg. A health physics survey more than 20 years after the Trinity test was made to determine the radiological risk for the public visiting the site (Fey 1967). In this study, monitoring of samples of Trinitite carried away from the site for gamma-exposure rates, measurement of surface exposure rate from individual Trinitite pieces and calculation of deposition of radioactivity in the body from ingested Trinitite were performed. In a later report (Hansen, Rodgers 1985), radiological conditions were evaluated for the Trinity site and the associated fallout zone. Here also qualitative data for activity concentrations of selected radioisotopes in the soil at GZ were given.

As for scientific literature easily accessible to the public, Atkatz & Bragg published a Trinitite NaI gamma spectrum in 1995 and from ^{137}Cs activity they calculated the explosive yield of the device. In a response to the paper, Schlauf et al. (1997) showed advantages of HPGe spectroscopy and identified other gamma emitters in Trinitite. Sixty years after the Trinity test, a comprehensive study of Trinitite radioactivity by means of alpha-, beta- and gamma- spectroscopy was published (Parekh et al., 2006). Among other findings, the authors used ^{152}Eu as a slow neutron flux monitor and conducted isotopic analyses of Pu in Trinitite samples. In a subsequent study (Semkov et al 2006) based on isotopic data, calculations and modeling, parameters characterizing the Trinity test were determined. Additionally, based on non-radioactive Trinitite properties, Hermes and Strickfaden (2005) devised a new theory on its formation during the explosion. The majority of the Trinitite layer was formed not on the ground, but by a rain of molten glass. After falling to the ground, the surface of Trinitite was further heated by the fireball and developed a smooth surface.

Fig. 1. A specimen of Trinitite. Left: top side. Middle: edge. Right: bottom side. Mass 3.170 g.

Semipalatinsk test site

The USSR conducted its first nuclear explosion at the Semipalatinsk test site in Kazakhstan on August 29, 1949. The Soviets coded it RDS-1 (the acronym is not well understood), while the American intelligence designated it Joe-1, after Joseph Stalin. The tested weapon was a plutonium bomb design similar to that in Trinity, detonated on a tower. The yield was estimated to 20 kt TNT.

Although not much information has been published about radioactivity at the Semipalatinsk test site, there is evidence that the first Russian explosion also created pieces of fused rock at GZ (Kruglov 2002, Hodge and Weinberger 2008). They are called Kharitonchiki in honour of one of the leading Russian nuclear weapons scientists, Yuly Khariton, but are more broadly referred to as atomsite. The authors are not aware of any clean-up involving removal of atomsite, as in the case of Trinity site.

Information on activity concentrations of radionuclides in the upper 2-3 cm of soil at the site of the RDS-1 test at the Semipalatinsk test site has been published by Yamamoto et al. (1996). A mainly in-situ gamma spectroscopic study was performed by Shebell & Hutter (1998).

Fig. 2. A specimen of atomsite from Semipalatinsk test site. Left: top side. Middle: edge. Right: bottom side. Mass: 2,545 g.

Algeria

France performed 4 atmospheric nuclear tests in the Algerian part of the Sahara desert between 1960 and 1961 at the Saharan Military Test Centre near Reggane. The first of the tests, conducted on February 13, 1960, was called Gerboise Bleue. The fission device was detonated on a 100 m tower with an estimated test yield of 40-80 kt. In 1999 the IAEA conducted a field expedition with the goal of evaluating residual activity due to the atmospheric nuclear tests (IAEA, 2005, Danesi et al. 2008). The atomsite from Gerboise Bleue test is described as black, vitreous and porous material, typically 100-1000 times more active than unmelted sand.

Material and methods

Trinitite[1] samples analyzed in the Radioactivity Measurements Laboratory, University of Bremen (Landesmessstelle für Radiaktivität, LMS) included 6 individual larger pieces of 1.4-3.8 g each (one of them in Figure 1), 3 plastic cylindrical containers filled with many small pieces (68, 50 and 14 g each) and 2 samples of powdered Trinitite (1.9 and 1.1 g). Additionally, a 103 g bulk sample of soil from the upper 5 cm of soil collected outside the inner fence, in the distance of approx. 100 m from GZ, was measured. The samples were analyzed by low-level low-background gamma

[1] Most of the Trinitite in the present study is from the Derik Bower collection. Based on personal communications with Ralph Pray (April 2006 to September 2009) and Derik Bower (January 2003 to October 2008), it is almost certain this material was collected by Ralph Pray in the summer of 1951. It would have been located on the south road leading to GZ, possibly between 150 and 220 m from GZ.

spectroscopy using a coaxial HPGe detector (Canberra Industries) of 50% rel. efficiency housed in a 10 cm Pb shielding with Cu, Cd and plastic lining, operated under Canberra Genie 2000 software. Efficiency calibration was performed for each individual sample separately based on its individual size, density and geometry relative to the detector using the Monte-Carlo based LabSOCS Genie 2000 calibration tool (Bronson 2003). Since some of the analyzed gamma emitters have complicated decay schemes (^{133}Ba, ^{152}Eu), the probability of detecting two photons emitted by the same decaying nucleus as one (a phenomenon known as cascade summing), is not negligible (up to 25%), therefore a cascade summing correction has been applied.

Spectra of another piece of Trinitite (#26) purchased from United Nuclear and a sample of atomsite from the Semipalatinsk test site (Figure 2) were obtained using the author's (JCR) Ortec LO-AX-51370/20 coaxial HPGe detector shielded by approximately 7cm Pb, lined with Cu and Sn foil within PVC and acrylic layers. Samples collected by the author (WMK) in October 2006 at Trinity site with known locations around GZ were measured using the author's (WMK) 5" Bicron NaI(Tl) well detector housed in 2 cm of Pb shileding with Cu foil grading. For the two later mentioned setups the efficiency calibration as described above was not achievable, therefore the resulting spectra were used for qualitative and comparative purposes only.

Results

Gamma spectroscopy of Trinitite enabled identification of several natural and artificial radionuclides, formed as a result of fission and activation or found as remains of nuclear fuel. A gamma spectrum of one of the samples is shown in Figure 3. Radionuclides are listed in Table 1 together with remarks on their origin. Some of the radionuclides (^{60}Co and ^{155}Eu, which were not reported in previous studies) were initially present in very high concentrations; therefore it is possible to detect them, even after more than 10 half-lives. Table 2 gives the specific activity of individual isotopes in Trinitite compared to literature values (Atkatz & Bragg 1995, Schlauf et al. 1997 and Parekh et al 2006).

Table 1. List of detected artificial gamma emitters artificial in Trinitite and their origin.

Isotope	Half-live (yr)	Origin
^{60}Co	5.3	Activation of ^{59}Co – from test tower steel and from soil
^{133}Ba	10.5	Activation of ^{132}Ba. Baratol - Ba $(NO_3)_2$ - was part of explosive lens system of the Gadget
^{137}Cs	30.0	Fission product (beta decay of ^{137}Xe and ^{137}I and also independently)
$^{152,\ 154}$Eu	13.3 / 8.8	Activation of stable isotopes ^{151,153}Eu in soil by slow neutrons
^{155}Eu	4.8	Fission product
^{239}Pu	24110	Principle isotope of nuclear fuel
^{241}Am	433	Mostly present as daughter product of ^{241}Pu (beta emitter), produced mainly from ^{239}Pu during the explosion via double-neutron capture. Based on ^{241}Am ingrowth activity of ^{241}Pu is possible to determine.

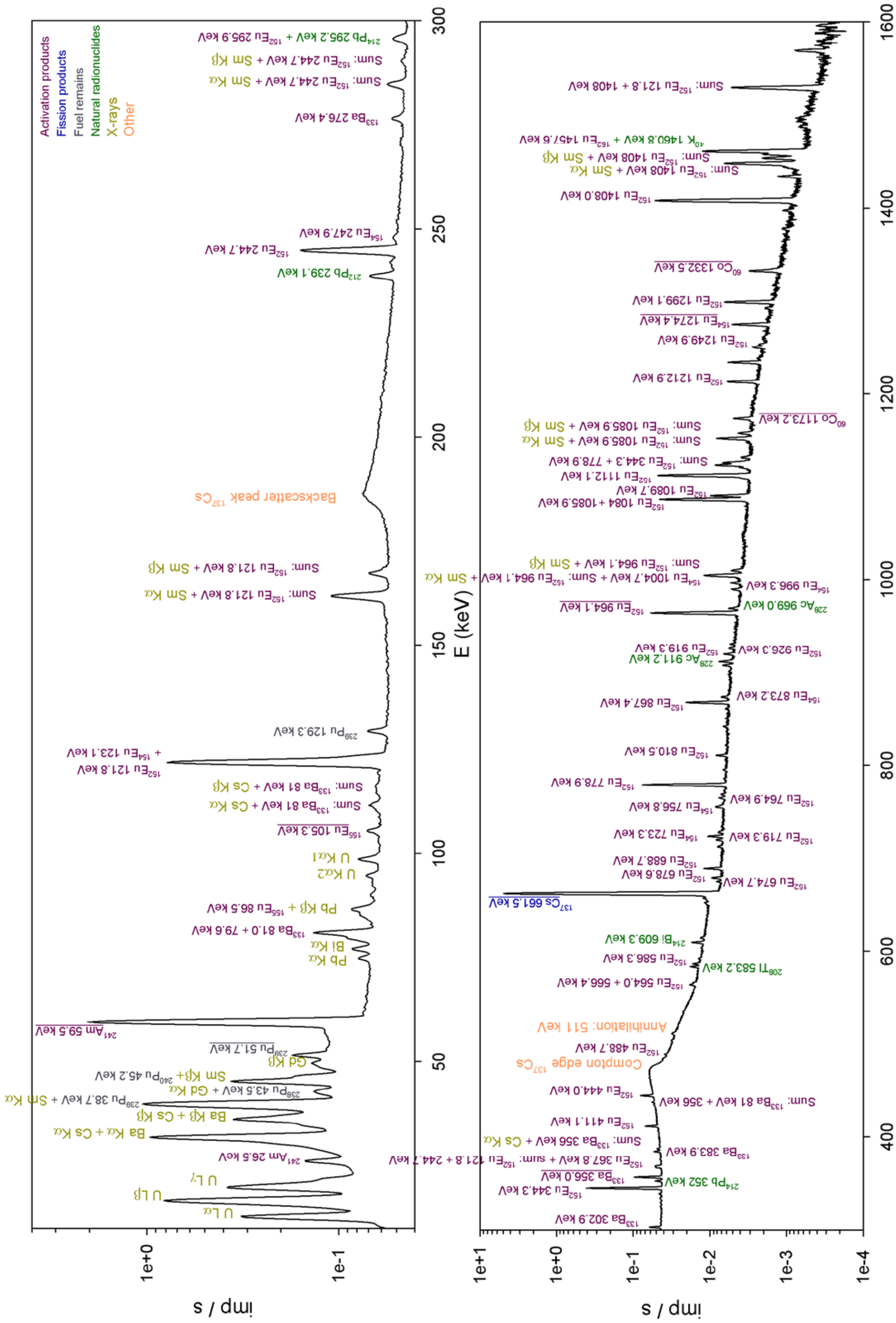

Fig. 3. A HP-Ge gamma spectrum of a sample of Trinitite measured in LMS (50.2 g, pieces of Trinitite in a plastic cylindrical container, height 20 mm, diameter 70 mm, live-time 423281 s).

Table 2. Table of comparison of activity concentrations (Bq/g, uncertainties 1σ) of artificial radionuclides detected in Trinitite and soil from nuclear test sites recalculated to the date of respective nuclear explosions (^{241}Am, which is ingrowing from ^{241}Pu, is reported to the date of analysis - in brackets).

Radio-isotope	LMS, 2010	Atkatz & Bragg, 1995	Schlauf et al., 1997	Parekh et al., 2006	IAEA, 2005	LMS, 2010	Hansen & Rodgers, 1985	Yamamoto et al., 1996
	Median, Min - Max			Min, Max	Min - Max	Min, Max	Min - Max	
	TRINITITE				ATOMSITE ALGERIA	SOIL TRINITY		SOIL SEMIPALATINSK
Bq/g	11 samples of Trinitite (see text)	Small piece	8,3 g piece	3 samples: several small pieces and 2 bigger individual pieces	Black fragments of fused sand	2 samples of soil (see text)	Samples at GZ area	2-3 upper mm of soil
^{60}Co	47.7 ± 4.6 <51.8[1] - (72.5 ± 4.8)	-	44 ± 4	44.4 ± 4.6 62.0 ± 4.9	168.9	2.9 ± 0.8 20.8 ± 1.3	64 - 320	1946 ± 43
^{133}Ba	9.10 ± 0.77 (5.04 ± 0.51) - (17.8 ± 1.3)	-	9.9 ± 0.6	7.55 ± 0.45 9.80 ± 0.26	48.9	0.104 ± 0.009 0.36 ± 0.03	0.73 - 1.73	-
^{137}Cs	48.3 ± 1.4 (16.26 ± 0.88) - (80.9 ± 2.4)	83.2	90 ± 9	27.33 ± 0.08 121.8 ± 0.1	78.1	0.142 ± 0.004 0.879 ± 0.026	0.38 - 1.87	236 ± 2
^{152}Eu	26.0 ± 1.1 (14.2 ± 1.3) - (55.5 ± 1.3)	-	27 ± 1	22.61 ± 0.38 78.89 ± 0.61	54.0	7.92 ± 0.19 33.42 ± 0.79	3.2 - 347	998 ± 10
^{154}Eu	7.08 ± 0.24 <6.03[2] - (12.76 ± 0.36)	-	4.8 ± 0.6	2.45 ± 0.60 16.1 ± 1.3	27.2	1.07 ± 0.08 4.80 ± 0.28	-	100.8 ± 4.5
^{155}Eu	274 ± 25 <361[3] - (461 ± 54)	-	-	-	241.2	<28.4 10.8 ± 2.8	-	-
^{239}Pu	73.6 ± 3.5 (25.7 ± 3.6) - (133 ± 25)	-	-	86.3 ± 2.7	<230	< MDA < MDA	-	-
^{241}Am	1.870 ± 0.080 (0.741 ± 0.048) - (4.47 ± 0.19) (2009)	-	2.9 ± 0.5 (1997)	1.841 ± 0.053 4.137 ± 0.058 (2006)	2.3 (1999)	0.0082 ± 0.0004 0.0369 ± 0.0018 (2009)	-	0.52 ± 0.01 (1994)
^{241}Pu[4]	63.6 ± 2.7 (25.2 ± 1.6) - (152.0 ± 6.5)	-	100 ± 17	62.9 ± 1.8 141.3 ± 2.0	85	1.254 ± 0.061 0.279 ± 0.014	-	18.6 ± 0.4

Several values measured by LMS were below detection limit, mainly due to low sample mass and/or shorter measurement times.
[1] 3 values under MDA, minimal value above MDA is 34.5 ± 5.6. **[2] 3 values under MDA, minimal value above MDA is 2.70 ± 0.07.**
[3] 5 values under MDA, minimal value above MDA is 115 ± 38. **[4] Calculated from ^{241}Am ingrowth**

The spectrum of radionuclides (with exception of ^{239}Pu) found in Trinity soil was similar to that of Trinitite. The absolute activities of $^{152,154}Eu$ and ^{60}Co, isotopes being formed by activation of elements present in soil itself, were present in the same order of magnitude as in Trinitite. The activity concentrations of ^{137}Cs, ^{133}Ba and ^{241}Am were 1-2 orders of magnitude lower in soil than in Trinitite (Table 2).

Discussion

Correlations

Correlation analysis was performed for the set of 11 Trinitite samples measured in LMS using Pearson test (normality was positively tested by Shapiro-Wilk test). Significant positive correlation was found for ^{152}Eu and ^{154}Eu (r=0.823;P=0.016). That is in agreement with similar origin of both isotopes (activation of stable Eu isotopes from soil). On the other hand, activation product ^{152}Eu and fission product ^{155}Eu are not correlated (r=0,305; P=0,557). ^{239}Pu and ^{241}Am, which is present as a decay product of ^{241}Pu (activation of fuel during the explosion), show strong positive correlation (r=0.967; P=0,000001). Activation product ^{133}Ba is positively correlated with ^{239}Pu (r=0,768; P=0,006) and ^{241}Am (r=0.785; P=0,004). Fission products ^{137}Cs and ^{155}Eu were significantly correlated with ^{239}Pu (r=0,827; P=0,0017 and r=0,836; P=0,038, respectively) and similarly with ^{241}Am. Positive correlation was also found for ^{137}Cs and ^{152}Eu (r=0,703; P=0,0159). ^{60}Co is not correlated to any other radioisotope.

Inhomogeneous distribution of radioactivity in Trinitite

The difference of activity at the top surface and the bottom surface of Trinitite is remarkable, mainly for beta activity. Measurements performed on 6 Trinitite specimens collected by WMK in 2006 using a 2" pancake tube inside a plastic bag to stop alpha radiation showed the top/bottom surface ratios ranging between 2.6 and 22.9 (with exception of one sample of unusual appearance showing ratio <1). Gamma measurements do not show such a pronounced difference due to higher penetrability of gamma radiation. A repeated measurement of one piece of Trinitite on a HPGe detector (LMS) top side up and top side down showed top/bottom ratios significantly increased for ^{137}Cs (1.24±0.05), ^{239}Pu (1.49±0.11) and ^{241}Am (1.25±0.08). Top/bottom ratios for ^{133}Ba and ^{152}Eu,however, were close to 1 (1.0±0.1 and 0.96±0.09, respectively).

Variability of radionuclides in Trinitite

While some variation in radionuclide activity (Table 2) can be attributed to sample size and normal variance, differences due to location are also expected and have been observed. Semkow et al (2006) found a power-law dependence between ^{152}Eu and slant range to the nuclear explosion center. Semkow calculated a least-squares exponent of -6.257 using 1946 data from 300 to 500 m (instruments closer to GZ were destroyed), whereas a spherical spread of neutrons would have resulted in an exponent of -2. Trinitite collected by the author (WMK) in October 2006 from 18 locations between 60 and 260 meters from GZ did, however, exhibit a power-law dependence with an exponent of -2.03 and least-squares correlation of R^2=0.777. The activity in counts-per-hour per gram at 1112 keV was found to be 68350 $s^{-2.03}$, where s is the slant range in meters to the top of the 30 m tower. The same specimens showed no significant relationship between ^{137}Cs activity (a fission product) and slant range using the 662 keV peak. It is not known how site clean-up activities may have affected the distances measured to GZ but evidence found during the 2006 survey suggests the specimens were probably within a meter of two of their original locations.

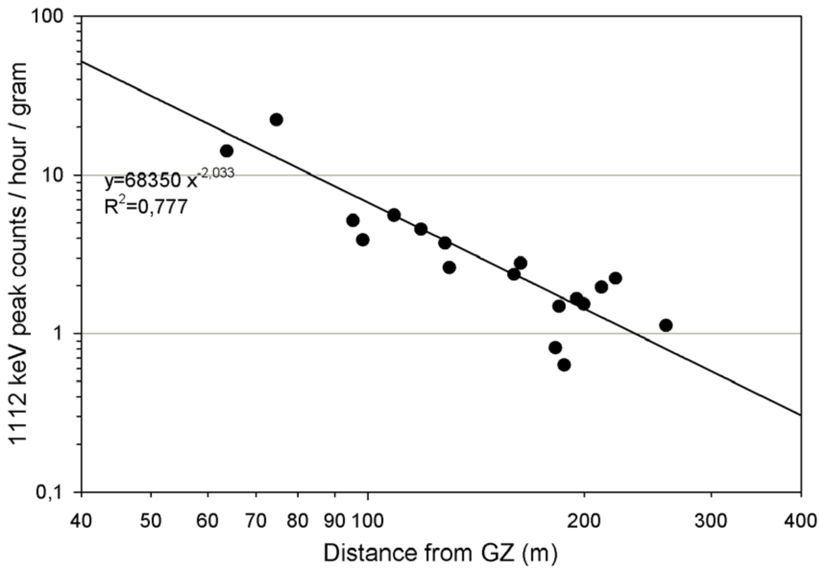

Fig. 4. ^{152}Eu activity vs distance from GZ in Trinitite samples collected in October 2006 by WMK measured with 5'' NaI(Tl) well detector.

Comparison of Trinitite to atomsite from Semipalatinsk

Gamma activity of a 2.545 g specimen of atomsite (Figure 2) from Semipalatinsk was compared to a typical 6.5 g specimen of Trinitite (gamma spectra in Figure 5). The net peak areas of radionuclides measured in both spectra were corrected by respective masses and decay corrected to the date of explosions (1945 and 1949, respectively). Atomsite from Semipalatinsk showed significantly higher activity than Trinitite. The activity ratios of Semipalatinsk/Trinity atomsite were: ^{137}Cs: 16.1±0.1, ^{152}Eu: 23.0±1.5, ^{241}Am (^{241}Pu): 3.4±0.1 and ^{239}Pu: 3.4±0.6. ^{154}Eu and ^{60}Co were under detection limits in Trinitite. ^{133}Ba was not detected in atomsite from Semipalatinsk. This is surprising due to the fact, that Baratol was also used in explosive lenses in the first Soviet nuclear test (Kruglov 2002).

Comparison of Trinitite to atomsite from Algeria

Literature values of a sample from Gerboise Bleue atomsite from Algeria (IAEA 2005 – sample No. ALG 4) described as black fragments of fused sand were compared to values of Trinitite measured by LMS (Table 2). Generally, the Algerian atomsite is more radioactive (activities recalculated to dates of explosions). ^{137}Cs value is about 1.5 times higher than median Trinitite values, ^{152}Eu 2 times higher, ^{154}Eu 4 times higher, ^{60}Co 3.5 times higher, ^{133}Ba 5 times higher. ^{155}Eu and ^{241}Am values are comparable for both Trinitite and atomsite from Algeria. Higher radioactivity in Algerian atomsite is likely due to the higher yield of the tested device.

Comparison of soil from Trinity to soil from Semipalatinsk test site

Soil data collected from the First Experimental site near the hypocenter where the first Soviet nuclear bomb was tested were published by Yamamoto et al (1996). They were compared to samples of soil from Trinity site. All activities were 1 or 2 orders of magnitude higher in soil from Semipalatinsk site than Trinity. The main difference was the absence of ^{133}Ba, as the activation product of stable Ba in baratol in Semipalatinsk site.

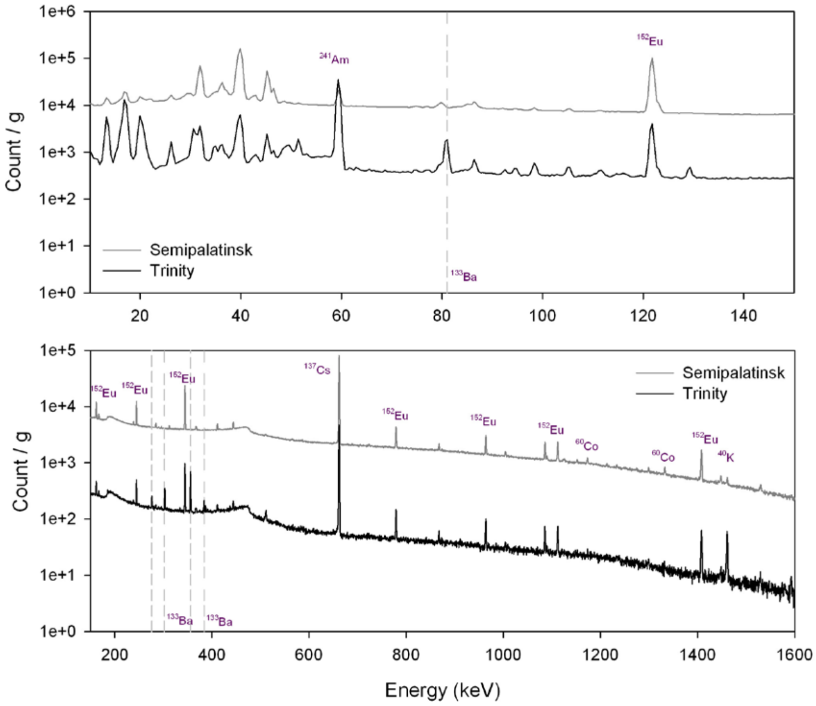

Fig. 5. 24 hour comparison spectra of atomsite from Semipalatinsk and Trinitite (sample #26) measured by Ortec LO-AX HPGe detector. Dashed lines indicate the main ^{133}Ba gamma lines in Trinitite and the lack of ^{133}Ba in the sample from Semipalatinsk.

Conclusions

Gamma spectroscopic studies of Trinitite, atomsite from Semipalatinsk and soil from Trinity was conducted. The experimental data were compared to literature data for Trinitite, atomsite from Algeria and soil from Semipalatinsk. ^{133}Ba (an activation product of Ba in explosive lenses) is present in Trinitite and atomsite from Algeria, but absent in atomsite and soil from Semipalatinsk. ^{155}Eu, a fission product, was found to be detectable in Trinitite after more than 13 half-lives. Correlations between radionuclides in Trinitite and the inhomogeneous distribution of radioactivity in Trinitite were discussed. Power-law dependence with an exponent close to -2 between ^{152}Eu and distance from GZ was found.

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External exposure of a representative individual at selected sites of the peaceful underground nuclear explosions in Russia

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Abstract

During the period 2001–2009, eight field expeditions have been conducted to seven selected sites of the peaceful underground nuclear explosions, which were performed in the European and Asian parts of Russia in the last century. Those are: "Crystal" and "Kraton-3" (Yakutia); "Dnepr" (Murmansk region); "Angara" and "Quartz" (Khanti-Mansiysk region); "Globus-1" (Ivanovo region); "Taiga" (Perm region). Evaluation of current doses from the man-made source to a representative individual was one of the main aims of the radiological investigations at the areas. This paper summarizes the key experimental data that are relevant to estimation of external doses.

External doses from the man-made γ-ray emitting radionuclides to a human were calculated using data of *in situ* measurements and results of laboratory radiometric analyses. Realistic estimations of the location factors were used for model calculations of the doses. The estimated effective doses included contribution from global fallout and Chernobyl debris. It is demonstrated that for some UNE sites the dose of interest may exceed a negligible limit of 10 μSv y^{-1}. At the same time, for all these sites the current doses are far below a value of 300 μSv y^{-1}, which is the threshold for application of countermeasures, accordingly to the Russian legislation.

Introduction

Practical implementation of underground nuclear explosion (UNE) technologies in a framework of the National Program "Nuclear Explosions for the National Economy" (Program №7), which had been carried out in the USSR throughout the period 1965–1988 (Ministry of the Russian... 1996), resulted in appearance of a number of sites contaminated by the long-lived man-made radionuclides (Logachev 2001, 2005; Norduke 2000; Yablokov 2003). The majority of the industrial UNE (81 from a total of 124) were conducted in the Russian Federation at the areas located beyond the boundaries of conventional nuclear test sites. The most pronounced contamination of the territory was reported for the UNE Kraton-3 (with ^{137}Cs, ^{90}Sr, plutonium) and Globus-1 (^{137}Cs, ^{90}Sr), where accidental releases of the radioactivity had occurred, and for the UNE Crystal (^{137}Cs, ^{90}Sr, plutonium, ^{60}Co) and Taiga (^{3}H, ^{137}Cs, ^{90}Sr, plutonium, ^{241}Am, ^{60}Co) polluted as a result of the planned technological conditions

(Gedeonov et al. 2002a, 2002b; Logachev 2001, 2005; Lurie 2002; Miretsky et al. 1997). Presently, general public has unlimited access to the sites, and therefore some additional exposure from these man-made sources of the ionizing radiation to a human might be expected.

This paper is devoted to evaluation of the current external γ-ray dose to a representative individual with respect of the four above mentioned sites of UNE. Three different sites (Dnepr, Quartz-3 and Angara), which did not have significant contamination by the long-lived γ-ray emitting radionuclides from the local sources, are taken for comparison.

Material and methods

Table 1 summarises locations and some technological characteristics of the explosions considered. Additional technical details on the UNEs, including geological conditions, may be found in Logachev (2001), Norduke (2000), and Ramzaev et al. (2007, 2009a, 2010b). The sites of selected UNEs are located at forested areas within the moderate or cool climatic zones. Examples are given in Fig. 1 and 2. All selected explosions were carried out in the closest proximity of rivers.

Table 1. Locations and some technological characteristics of selected UNEs (in chronological order). The geographic coordinates have been recorded during our expeditions to the UNE sites; other data are given accordingly to Logachev (2001).

Name of UNE	Year of detonation	Region, geographic coordinates	Depth, m	Power equivalent, kt of TNT	Purpuse
Taiga	1971	Perm region 61.3° N, 56.6° E	127	3×15 (45)	Constructing of a canal
Globus-1	1971	Ivanovo region 57.5° N, 42.6° E	610	2.3	Deep seismic sounding of the Earth's crust
Dnepr-1	1972	Murmansk region 67.8° N, 33.6° E	131	2.1	The breakage of ore
Crystal	1974	The Republic of Sakha (Yakutia) 66.5° N 112.5° E	98	1.7	Constructing of a reservuar dam
Kraton-3	1978	The Republic of Sakha (Yakutia) 65.9° N, 112.3° E	577	22	Deep seismic sounding of the Earth's crust
Angara	1980	Khanti-Mansiysk AO 61.7° N, 67.1° E	2485	15	Stimulation of oil production
Quartz-3	1984	Khanti-Mansiysk AO 61.9° N, 72.1° E	726	8.5	Deep seismic sounding of the Earth's crust
Dnepr-2	1984	Murmansk region 67.8° N, 33.6° E	175	2×1.8 (3.6)	The breakage of ore

Fig. 1. The mountain of Kuel'por (Khibiny, Kola Peninsula) is the site where nuclear explosion technologies were used for the experiments on breakage of the phosphate ore in 1972 and 1984 (project "Dnepr"). The group of tourists and researches is on the road which connects the tourist's camp (to the right) with the site of explosions. July 2008.

Fig. 2. A bed for this beautiful lake had been created by a simultaneous detonation of three thermonuclear devices in 1971. The "Taiga" experiment was conducted with the purpose to obtain some field data for the final elaboration of the project on Kama-Pechora canal. The lake "Taiga" has a length of ca. 700 m and a width of (350 to 380) m. The photo was taken from a helicopter in August 2009.

Fig. 3. Measurements of γ-ray dose rate in air using the gamma dosimeter EL-1101. The plot is located on the axis of the Kraton-3 radioactive trace at a distance of approximately 1.5 km from the borehole. 100% mortality in the larch tree (*Larix gmelinii*) population is observed. July 2002.

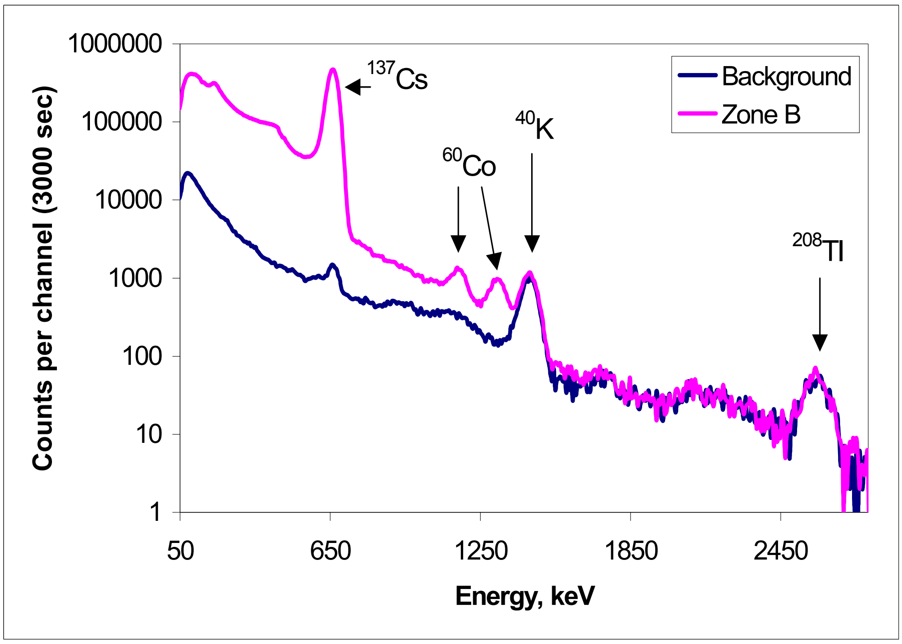

Fig. 4. The γ-ray spectra were recorded on a background plot in Yakutia and on a plot located within a boundary of the affected forest (zone B) at the Kraton-3 site in July 2002. Note a strong excess the ^{137}Cs peak on the spectrum from the Kraton-3 site above the background one (actually, 662 keV peak is related to the short-lived meta-stable daughter of ^{137}Cs – ^{137m}Ba). Two peaks from ^{60}Co (1173 keV and 1332 keV) can also be seen here. There is no difference between the background and *on site* spectrum with respect of peaks from the natural radionuclides of ^{40}K (1460 keV) and ^{208}Tl (2615 keV).

Although the experiments (excluding, perhaps, the Crystal UNE) were conducted at remote areas, the sites are frequently used by tourists and local population for collecting mushrooms and berries, as well as for angling and hunting. The investigators, who conduct field studies, form another group of the exposed individuals.

We suppose that a total duration of a person's staying at the area of an UNE is equal to 14 days in a year. A representative individual was assumed to have stayed within the contaminated site 8 hours each day, totally 112 hours in a year. During 224 hours in a year the person is staying at the background locations: in a camp, forest, on the bank of a river, etc. The scenario is based on our own experience and the results of interviewing other investigators, local citizens and tourists.

For the measurements of γ-ray dose rate (DR) in air at a height of 1 m above the ground, portable γ-ray dosimeters EL-1101, EL-1117 and AT-1121 (ATOMTEX, Belarus) were used (Fig. 3). Technical characteristics of the dosimeters are given in Ramzaev et al. (2006, 2010b). The registered values of DR (Table 2, column 5) include intrinsic noise of the device, cosmic radiation response, contribution from the natural radionuclides, and contribution from the man-made radionuclides. The routine measurements of dose rates in air were supplemented by *in situ* γ-ray spectrometry (Fig. 4) and GPS mapping (Ramzaev et al. 2010b). Soil samples were collected in the plots that might be contaminated as a result of the UNEs and in the "background" plots, which do not lie on sectors affected by the radioactive plume

Table 2. Mean±standard deviation values of the ground surface contamination with ^{137}Cs (kBq m^{-2}), total measured γ-ray dose rate in air (nSv h^{-1}), and calculated external γ-ray dose (µSv y^{-1}) to a representative individual. The dose is attributed solely to the man-made radionuclides: ^{137}Cs and some others, as indicated in column 4. The dose that has been calculated using formula (2) marked (*). Number of sampling plots is given in brackets. The experimental data are obtained from EKORANT (2007), Federal Scientific (2008, 2009), Ramzaev et al. (2007, 2009a) and St-Petersburg Research (2002). The reference date is the sampling date given in the first column.

Name of UNE, year	Location	^{137}Cs, kBq m^{-2}	Other man-made γ-ray emitters	Total dose rate in air, nSv h^{-1}	Dose, µSv y^{-1}
Taiga, 2009	site of UNE	452 ± 421 (6)	^{60}Co, ^{207}Bi, ^{241}Am	274 ± 274 (498)	22*
	background	1.4 (1)	Not detected	76 ± 20 (9)	0.2
Globus-1, 2008	site of UNE	17400 ± 19300 (3)	Not detected	174 ± 240 (189)	14*
	background	2.5 (2)	Not detected	46 ± 4 (5)	0.3
Dnepr, 2008	site of UNE	1.9 ± 0.9 (4)	Not detected	180 ± 202 (60)	0.1
	background	2.2 (1)	Not detected	108 ± 18 (25)	0.3
Crystal, 2001	site of UNE	21 ± 25 (5)	^{60}Co, ^{241}Am	104 ± 70 (6)	7.1*
	background	0.8 ± 0.1 (6)	Not detected	41 ± 8 (8)	0.1
Kraton-3, 2001-2002	site of UNE	955 ± 1870 (14)	^{60}Co, ^{241}Am	240 ± 230 (332)	22*
	background	0.8 ± 0.1 (6)	Not detected	41 ± 8 (8)	0.1
Angara, 2007	site of UNE	1.3 (2)	Not detected	62 ± 12 (34)	0.1
	background	1.3 (1)	Not detected	57 ± 4 (6)	0.2
Quartz-3, 2007	site of UNE	1.4 ± 0.2 (4)	Not detected	43 ± 7 (61)	0.1
	background	1.4 (1)	Not detected	34 ± 4 (3)	0.2

trajectories. Between three and seven cylindrical samples, each with a ground surface of 20 cm^2, were taken at a plot to determine the man-made radionuclides activity per unit area. The depth of such cores ranged from 10 cm to 40 cm, depending on tasks of sampling and presence of pebbles and stones in the underlying ground. Additional details on the soil sampling procedure may be found in Ramzaev et al. (2006, 2007). Content of the man-made γ-ray emitting radionuclides in soil samples was determined by direct gamma-spectroscopy using shielded high-purity Ge detectors and multi-channel analysers.

Results

^{137}Cs is the only one man-made γ-ray emitting radionuclide detected in the top-soil samples that had been collected at the background plots (Table 2). Levels of ^{137}Cs deposit in individual plots varied from circa 0.7 kBq m^{-2} in Yakutia to 3.1 kBq m^{-2} in Ivanovo region. Global fallout might be considered as the main source of ^{137}Cs for background plots in the Asian part of Russia (the UNE Angara, Kraton-3, Crystal and Quartz-3), while for the plots in the European part (the UNE Globus-1, Dnepr, and Taiga) some contribution from Chernobyl source should be anticipated.

^{137}Cs ground contamination at sites of the UNE Dnepr, Quartz-3 and Angara did not deviate from the respectful background levels; for the UNE Quartz-3 and Angara there were also no principal difference between the background and *on site* measured DR (Table 2). At the same time, the DR values registered at the Dnepr site, which is located at a mountainous area, were on average somewhat higher than the background values. The *in situ* gamma-spectroscopy indicated that this might be attributed to the local leakage of the underground air, which is enriched with the radon and its daughters. The DR measured near the mouth of one of the tunnels, which connect intramountain cavities with the Earth's surface, was about an order of magnitude higher than levels determined at background plots (Scientific Enterprise… 2008).

Levels of ^{137}Cs surface contamination at sites of the UNE Kraton-3, Crystal Globus-1 and Taiga exceeded the background values drastically (Table 2). The maximum contamination density (40000 kBq m^{-2}) was deduced for the Globus-1 site (Scientific Enterprise… 2008). Some other man-made radionuclides that were found at UNE sites are listed in Table 2, column 4. Elevated levels of DR have been measured at all four sites that had been significantly contaminated as a result of UNE (Table 2). At the Kraton-3 and Taiga sites, the contamination is observed at an area covering more than 1 km^2. A smaller spatial contamination may be deduced with respect of the Crystal and Globus-1 sites – about 6 ha and 2 ha, respectively (Ramzaev et al. 2007; Scientific Enterprise… 2008).

The effective external dose (E_{ext}) from ^{137}Cs to a human at a background location can be calculated according to:

$$E_{ext} = T_{loc} \cdot A^{137} \cdot K^{E} \cdot g^{137}, \qquad (1)$$

where
T_{loc} – is duration of staying at a location, h. T_{loc} is 224 h;
A^{137} – is the ground surface contamination with ^{137}Cs, kBq m^{-2};

K^E – is a coefficient converting the absorbed dose in air to effective dose for a human, Sv Gy^{-1}. The numeric value of K^E is 0.71 Sv Gy^{-1} (Golikov et al. 2007);
g^{137} – is a transfer factor from the ground surface contamination by ^{137}Cs to absorbed dose rate in air, (μGy h^{-1})/(kBq m^{-2}). The numeric value of g^{137} is $0.71 \cdot 10^{-3}$ (μGy h^{-1})/(kBq m^{-2}) (United Nations 2000).

Formula (1) has also been applied for calculating the effective dose for a human staying at sites of the UNE Dnepr, Quartz-3 and Angara where no excess of the man-made contamination was observed. Duration of staying at the site of an UNE, T_{loc} has been taken as 112 h. The results of estimation of the external γ-ray dose from the man-made sources are given in the last column of Table 2. As expected, negligible doses (much less than 1 μSv y^{-1}) have been calculated for sites of these three UNE and for all background locations.

For calculating the external doses (E_{ext}) at contaminated sites, the other approach based on a difference between DR measured *on site* and on background plots was adopted. The equation used for these four sites of UNE is:

$$E_{ext} = T_{site} \cdot (P_{site} - P_{bg}), \qquad (2)$$

where
P_{site} – is the total averaged DR at a site, μSv h^{-1};
P_{bg} – is the total averaged DR at a background location, μSv h^{-1};
T_{site} – is duration of staying at an UNE site, 112 h.

For the three sites of UNE (Taiga, Kraton-3 and Globus-1) the effective dose exceeded a negligible limit of 10 μSv y^{-1} (Table 2).

Discussion

The legislative status of the peaceful UNEs conducted at the territory of Russia is not yet established officially (Ramzaev et al. 2009b). Nonetheless, the situation of "existing exposure" is, perhaps, the most closely related definition that can be applied for describing general radiological status of the UNE sites nowadays. A short-term staying (during days or weeks in a year) of a very limited number of people on the contaminated areas is a specific feature of the exposure conditions at such sites.

Our estimations of external exposure to a human at the seven sites of peaceful UNEs indicates that for three cases the technogenic dose may exceed a negligible limit of 10 μSv y^{-1}. At the same time, for all these sites the current doses appeared to be far below a value of 300 μSv y^{-1}, which is the threshold for application of countermeasures, accordingly to the Russian legislation (Federal Service 2009).

The annual effective external dose, which has been calculated according to formula (2), includes contributions from a total suite of the man-made γ-ray emitting radionuclides and some bremsstrahlung radiation from interaction of beta-rays with nuclei in soil and air. The latter source may be of significance at the areas contaminated with ^{90}Sr/^{90}Y, for example at the Kraton-3 site (Ramzaev et al. 2009a). The direct beta exposure to the skin and eyes of a person in a radiation field at the UNE sites may somehow contribute to the total dose (see Ramzaev et al. 2009a and references therein). The issue related to the beta-ray exposure requires further studies because relevant quantitative estimations are not presented.

Additionally to the external exposure, which is an inevitable component of the total dose for the UNE situation, a human, in principle, may be exposed from the man-made radionuclides ingested with food products of natural origin and water. Inhalation of the radioactive aerosols may be another pathway of internal exposure at the UNE sites. Contribution of internal exposure to the total may vary significantly. Accordingly to preliminary conservative assessments, the input from internal sources is approximately 20 % at the Taiga, site and it is more than 90 % at the Dnepr site (Scientific Enterprise 2008, 2009).

Conclusions

The residual levels of the surface ground contamination with ^{137}Cs at the sites of UNE Kraton-3, Crystal, Globus-1 and Taiga are significantly higher than those detected for background areas.

Our estimations show that, presently, the UNE-relevant effective external dose to a representative individual may exceed a negligible limit of 10 μSv y^{-1} with respect of the Kraton-3, Globus-1 and Taiga sites.

For some sites of UNE, the external gamma radiation may constitute the major contribution to the total technogenic dose.

Periodical monitoring of radiation conditions at the sites of UNE is recommended (Ramzaev et al. 2010a).

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Radionuclides and heavy metals bioavailability in a Norwegian area rich in naturally occurring radioactive materials

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Abstract

Recent study on environmental samples from thorium rich Norwegian area showed high gamma dose rate and significant transport of radionuclides into environmental compartments giving multiple stressors action in natural conditions.

The main objective of this work was to investigate distribution and uptake of ^{232}Th, ^{238}U and heavy metals such as Cd, Cu, Zn and Pb from soil and rocks into biota samples in natural conditions.

Soil samples, plant samples of the most abundant species in area and earthworms were sampled on high gamma dose rate location in Fen (Norway). Radionuclides and heavy metals were analyzed by ICP-MS.

Soil analysis showed high level of radionuclides, 4040 and 111 Bq/kg, for ^{232}Th and ^{238}U, respectively. Pb and Zn were in moderate high levels, but comparable with other Norwegian soils, Cd and Cu were in range of non contaminated natural soil. Among the plants collected, the highest uptake of both ^{232}Th and ^{238}U was found in moss. Distribution of these radionuclides was positively correlated with levels found in soil taken below. Different earthworms species showed high levels of ^{232}Th, ^{238}U, Cd and Pb what could suggest bioconcentration from soil – worms' habitat and food. It was shown that these Fen invertebrates had significantly higher levels of several contaminants in comparison to those from chosen reference site in the vicinity.

These preliminary obtained results will form the basis for future work in aspect of:

- determination of sensitive plant, fish, earthworm and mice species,
- determination of transfer factors and possible bioconcentration,
- investigation of different plant parts ability to concentrate naturally occurring radionuclides and study on dependence between uptake and change of natural conditions,
- investigation of target organs for radionuclides and metals deposition and possible biological effects,

which will be presented in the conference.

Introduction

Data on the activity concentration of naturally occurring radionuclides in the environmental media (i.e., soil, water, sediment) and organisms of interests are of high importance when assessing radiation impact on non-human organisms. Parameters obtained in investigation of terrestrial and aquatic biota uptake of radionuclides in natural conditions are highly relevant when it comes to estimate consequences of nuclear pollution or accidents problems. However, the knowledge gap and lack of data considering many organisms must be considered (Jones et al., 2003; Beresford et al., 2008). Additionally, biota is rarely exposed to only radiation, but also very often to some heavy metals. Thus, it is significant to predict the most probable exposure scenario and to investigate both levels of radionuclides and heavy metals in living organisms, as well as to investigate possible diverse i.e., additive, synergistic or antagonistic biological effects (Salbu et al., 2005).

Fen central Complex in southern Norway is geologically specific area of magmatic, carbonatite rocks rich in naturally occurring radioactive materials (NORM), in first place thorium (Th) and uranium (U), but also in iron (Fe) and rare earth elements (REE). This part of the country is well known with its high exposure radiation doses to humans, mainly due to high levels of radon (Sundal and Strand, 2004). Several mining sites were exploited in this area for Fe, niobium (Nb) and scandium (Sc) during the past centuries. All mining activities were finished in 1960s, but technologically enhanced naturally occurring radioactive materials (TENORM) sites are still important sources of environmental pollution. Beside NORM sites, the TENORM sites also contribute to the high exposure dose to the Fen human population (Sundal and Strand, 2004; Mrdakovic Popic et al., 2010, in preparation).

The main objective of this study was to investigate the correlation of high soil concentrations of naturally occurring radionuclides ^{232}Th and ^{238}U and some heavy metals such as Pb, Cd, Zn and Cu and their uptake in the living organism in the NORM rich area. According to that, transfer factors were calculated using organisms concentration and soil concentration. Assessment of environmental impact considering radionuclides is done on the basis of obtained activity concentrations of radionuclides using the ERICA assessment tool (Brown et al., 2008). Further activities will be to investigate possible biological effects on organisms that were shown to be the most sensitive and under risk.

Material and methods

Fen Central Complex is a specific geological area positioned 120 km southeast of Oslo. It is an area rich in volcanic carbonatite rocks: Søvite, Rødberg, Rauhaugit and Fenite (Fig.1). Some of these rocks were first described by Brøgger (1921) and named worldwide after these geographical names. This part of Norway is known with its high thorium ore deposits (USGS, 2007, Thorium report, 2008) and high radon levels and consequently high annual radiation exposure doses (Stranden E., 1986; Solli et al., 1986; Sundal and Strand, 2004; Smerthust et al., 2006).

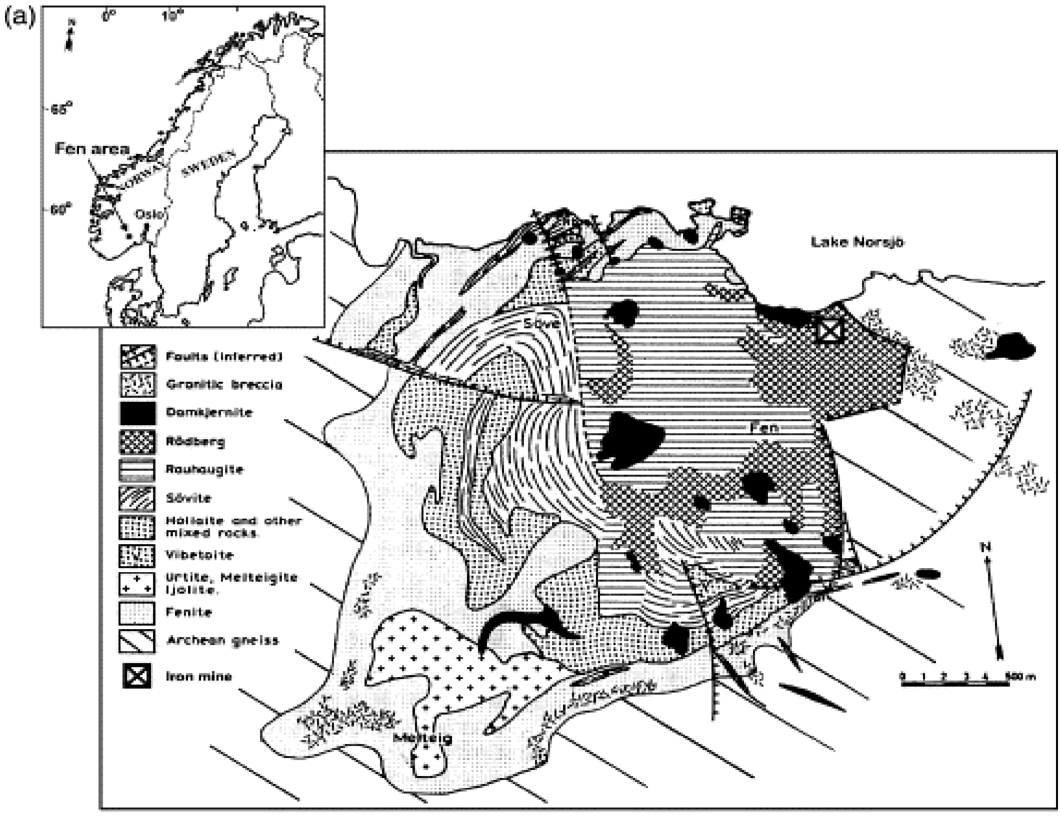

Fig. 1. Bedrock geology of Fen Central Complex (Ramberg and Barth, 1966).

Collection of soil, plants and earthworms were done during the fieldwork at spring 2008, in Fen, Telemark, Norway. Sampling location was chosen according to increased terrestrial gamma exposure dose rate, measured by hand detector type Automess AD4LF, type 6150. Soil samples were collected from depth 0-20 cm as bulk samples. Earthworms (40 adult organisms) were collected from several points at chosen location by hand sorting, and after identifying in the laboratory they were washed and starved for 2 days until their gut contents were completely egested. Earthworms were then frozen in liquid nitrogen, freeze dried and ground to a fine powder. Plants, that were found to be the most abundant in the area i.e., lichen, moss, birch, pine and spruce were collected in paper bags, at least 10 different plants from each species. All plants were, further in the laboratory, cleaned from visible soil, washed several times with distilled water and dry for several weeks at temperature 40°C. They were homogenized by milling. Soil samples were air dried for several weeks, crushed to a fine powder and finally homogenized by sieving through 2 mm sieve. Aliquots of soil were dried at 80°C to constant mass and analyzed for pH, water and organic matter content prior to decomposition for analysis.

Concentrations of radionuclides ^{232}Th, ^{238}U and heavy metals in all samples were measured with ICP-MS after extracting with microwave decomposition (Milestone Inc., Ultraclave High performance reactor, Shelton, CT), employing high-grade purity HNO_3 acid for soil and HNO_3/H_2O_2 for plant and earthworms samples. ICP-MS analysis was done using a Perkin ElmerSciex Elan 6000 (Norwalk CT, USA). Quality control assurance was performed by substantial use of internal standard, method and instrument blanks and by using proper certified reference materials.

Results and discussion

Results of soil analysis (Table 1) showed high concentrations of ^{232}Th and considerably lower concentrations of ^{238}U, 4040 and 111 Bq/kg, respectively. These values clearly exceed the world average values for soil 30 Bq/kg for ^{232}Th and 35 Bq/kg for ^{238}U given by UNSCEAR (2000). Average value for Pb was approximately 2-3 times higher than non-polluted soil values given by Norwegian Pollution Control Agency, SFT (2009), and values given by Arctic Monitoring and Assessment Programme, AMAP (1998). Local contamination of terrain by physical and chemical weathering of bedrock and rock-forming minerals is the most probable cause of obtained high concentrations. Concentrations of Cd and Cu found in Fen soil samples were in range for non-polluted soil and similar to results obtained for soil from other Norwegain parts (Salbu and Steinnes, 1995; Steinnes and Ruhling, 2002). Zn in our study was present in soil at 2.5 times higher concentrations that given by SFT (2009). These high levels in Fen area are, probably as in the case of Pb, results of local loading of Zn due to weathering of bedrocks and leaching upon different environmental conditions.

Table 1. Thorium and uranium concentrations in soil samples from Fen.

Sample	^{232}Th (mg/kg)	^{238}U (mg/kg)	^{232}Th (Bq/kg)	^{238}U (Bq/kg)
Fen site	995±284	9±3	4040±1153	111±37
UNSCEAR (2000)	-	-	30	35

Table 2. Heavy metals concentrations in soil samples from Fen.

Sample	Pb (mg/kg)	Cd (mg/kg)	Cu (mg/kg)	Zn (mg/kg)
Fen site	126±78	0.5±0.1	8.5±10.5	508±151
SFT (2009)	60	1.5	100	200

To obtain a kind of initial screening and an overall picture about radionuclides and heavy metals bioavailability for living organisms uptake, the typical and the most abundant species in the area were chosen for analysis. In addition to the five plant species we analyzed two different species of earthworms that have habitat in the inner layers of soil. Results obtained in Fen plants and earthworms analysis are given in Tables 3 and 4.

Among the plants, the highest uptake of radionuclides was noticed in moss, 90 and 6.4 Bq/kg for ^{232}Th and ^{238}U, respectively. Obtained values for ^{232}Th and ^{238}U in plant samples were higher than reference values for leafy plants given by UNSCEAR (2000), 0.016 Bq/kg and 0.062 Bq/kg for ^{232}Th and ^{238}U respectively,. Lower but comparable results for moss and grass samples were obtained in similar investigations of Ramli et al. (2005) and Vandenhove at al. (2006). Metals Pb and Cd were also in the highest concentrations in moss. Essential metals like Cu and Zn were in the highest concentration in birch leaves.

Radionulides concentrations of two different endogeic species of earthworms were significantly high, 316 ad 178 Bq/kg, for *Aporectodea caliginosa* and *Dendrodrilus rubidus*, respectively. These two species, living in the soil, were chosen with expectance that both the internal and external exposure to a cocktail of radionuclides and heavy metals could affect them.

Table 3. Radionuclides and heavy metals concentrations in plant samples from Fen.

Sample	^{232}Th (Bq/kg)	^{238}U (Bq/kg)	Pb (mg/kg)	Cd (mg/kg)	Cu (mg/kg)	Zn (mg/kg)
Pine	0.8	0.2	0.32	0.06	4.4	70
Spruce	0.2	0.04	0.06	0.05	6.1	67
Birch	2.2	0.2	0.27	0.15	8.3	95
Moss	90	6.4	10.2	0.21	4.1	63
Lichen	20	2.6	3.5	0.08	2.2	29

Table 4. Radionuclides and heavy metals concentrations in earthworms from Fen.

Earthworms	^{232}Th (Bq/kg)	^{238}U (Bq/kg)	Pb (mg/kg)	Cd (mg/kg)	Cu (mg/kg)	Zn (mg/kg)
A.caliginosa	316	15	19	4	7	-
D.rubidus	178	5	9	6	6	-

In order to obtain the information on uptake in biota, radionuclide transfer factors (TF) were calculated. TF were calculated as a ratio between radionuclide concentration in organism and radionuclide concentration in soil (Table 5).

Table 5. Transfer factors for ^{232}Th and ^{238}U in Fen plant and earthworms samples.

Sample	TF ^{232}Th	TF ^{238}U
Pine	2E-04	2E-03
Spruce	5E-05	4E-04
Birch	5E-04	2E-03
Moss	3E-02	6E-02
Lichen	5E-03	2E-02
Earthworms	6E-02	9E-02

Average values for plants transfer factors in this study are one order of magnitude lower or comparable with values in ranges published by IUR (1994) and ERICA (2008), for ^{232}Th: 0.001-0.11 and for ^{238}U: 0.002-0.23. The highest transfer factors for

^{232}Th and ^{238}U were obtained for moss and earthworms. Uptake of ^{238}U in the earthworms was higher than the average values given by ERICA data base (2008).

The total exposure rates and risk presented in Fig.2 were obtained by ERICA assessment tool (Brown et al., 2008) on the basis of media concentrations of selected radionuclides. It is also confirmed that bryophytes and earthworms are the most sensitive and potentially vulnerable organisms in this area.

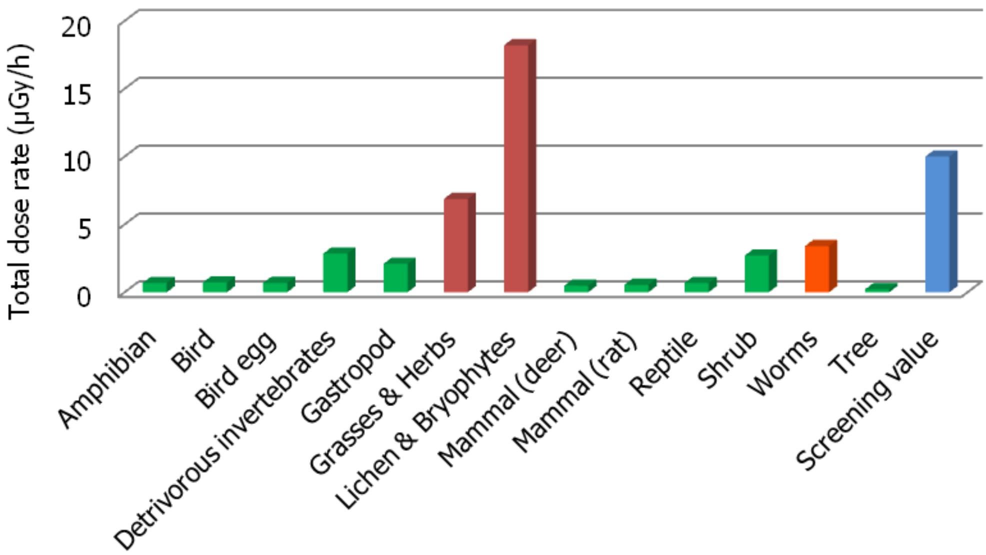

Fig. 2. Total exposure dose per terrestrial organism groups, calculated on the basis of internal exposure.

Conclusions

The main objective of this study was to investigate the correlation of high soil concentrations of naturally occurring radionuclides ^{232}Th and ^{238}U and some heavy metals such as Pb, Cd, Zn and Cu and their uptake in living organism in the area. Data on the activity concentration of naturally occurring radioactive materials in the environmental media (i.e., soil, water, sediment) and organisms of interests are of high importance when assessing radiation impact on non-human biota. Results from this study show some organisms groups that are under high risk and provide information on groups that should be more investigated in terms of early biological responses.

Analysis of Fen soil samples showed high concentrations of both ^{232}Th and ^{238}U, significantly higher than reference values given by UNSCEAR (2000). The found radionuclides concentrations in the soil suggested very inhomogeneous distribution and various influence on living organisms.

When it comes to assessing the radiation risk for terrestrial non-human species the knowledge gap and lack of data must be considered (Jones et al., 2003; Beresford et al., 2008). In our study the selected plant and earthworm species, representative samples from the Fen area, were analyzed for radionuclides and transfer factors were calculated. Levels of both ^{232}Th and ^{238}U in moss and lichen samples were higher than those found in similar plant species (Ramli et al.,2005; Vandenhove at al.,2006). However, no much data for comparison purposes is available on the species grown in natural conditions and on especially species that are not important when it comes to human consumption. Still, despite the fact that high levels of radionuclides were found in some plants, obtained transfer factors were approximately one order of magnitude lower that those published by IUR (1994) and ERICA (2008). Opposite, the concentrations and transfer factors obtained for earthworms are higher than those published by ERICA (2008) and suggest active biological uptake of these organisms. Considering only radionuclides, the organism groups that are under the highest risk in this area, are bryophytes and earthworms.

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Radionuclides and heavy metals levels in environmental samples from thorium rich Fen area in Norway

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Abstract

Fen area in southern Norway is documented to be among the highest natural world reservoirs of thorium ore. It is geologically well investigated area with ^{232}Th rich carbonatite rocks. Dose risk assessment and radon investigations, done in the past, showed total indoor effective dose up to 14 mSv/year. However, there are still huge knowledge gaps when considering environmental impact from the natural occurring radioactive material. This paper is a result of the first investigation phase on a multiple stressors study which comprises identification of contaminants, distribution in natural compartments, biota uptake and total environmental impact assessment.

Gamma dose rate measurements gave an average dose rate of 4.2 μGy/h, much higher than world's average dose rate of 0.059 μGy/h. Rock levels of radionuclides were consistent with previous findings and positively correlated with recorded gamma dose rates in the area. Total soil analysis showed concentrations in the range 69-6098 Bq/kg for ^{232}Th and 49-128 Bq/kg for ^{238}U. Concentrations of Cd in soil were higher than world average, but in range of Norwegian soil values. However, concentrations for Pb ranged from 47 to 287 ppm and for As from 4 to 20 ppm, probably due to mobilization and weathering from rocks. Despite high soil levels, total and size fractionated water samples showed neither radionuclides nor heavy metals contamination. Size fractionation showed that ^{238}U was more soluble and in the low molecular mass fraction, while ^{232}Th was in colloidal and particulate phase, confirming typical behavior in nature.

Considering both radionuclides and heavy metals found in rock and soil from this area, future work will focus on investigation of mobilization conditions, transfer to biota, biological effects and possible synergistic action of different radionuclides and heavy metals.

Introduction

Increasing world's energy demands and limited sources, as well as pollution problems, put the nuclear energy production recently in focus in many different countries worldwide. Considering thorium resources, the Fen area (county of Telemark, southern Norway) is assumed to be among the world's largest with estimated total of 170 000 tones (USGS, 2007).

Related to these high levels of naturally occurring radioactive material (NORM) and technologically enhanced naturally occurring radioactive material (TENORM), the Fen area is well known as area with high mean value of the annual total effective radiation dose to the population. Doses up till 4 times higher than the estimated average effective radiation dose to the Norwegian people are found (Stranden E., 1986; Stranden and Strand, 1986; Sundal and Strand 2004).

Previously, investigations in Fen complex showed unique geology formation with enhanced content of ^{232}Th, ^{238}U and rare earth elements (Brøgger 1921; Landreth 1979; Dahlgren S., 1983, Stranden E, 1982.) and increased gamma radiation dose rates (Sundal and Strand, 2004; Smerthust et al., 2006). Since the main contributor to the total effective radiation dose from naturally occurring radionuclides is exposure to radon, many studies have investigated the indoor radon concentrations in Fen area dwellings, relationship of indoor and outdoor radiation doses and their dependence on geology formation of terrain (Stranden E., 1982; Solli et al., 1986, Sundal and Strand, 2004; Smerthurst et al., 2006).

Typical Fen complex rock types, first described by Brøggger 1921., occupying the largest fraction of the surface, are carbonatite rocks of magmatic origin. Intensive mining activities in the area for iron and niobium production were performed from 17th to 20th century. The radiological and epidemiological impact of these former mining activities, evaluated by Stranden E., 1982, showed that workers received an annual dose equivalent of 150 mSv, obviously much higher than occupational dose limits of 20 mSv/y for radiation workers.

Similar investigations of radioactivity in high background natural radiation areas have been conducted worldwide (Malanca et al., 1993; Quindos et al., 1994; Vera Tome et al., 2002; Ramli et al., 2005; Vandenhove et al., 2006; Prasad et al., 2008). Some studies showed various pathways of radionuclides and heavy metals leaching from rocks and soil and their transportation and concentration into the water, plants and animals in different degrees (Bernhard et al., 1996; Apps et al., 1998; Vandenhove et al., 2006).

The main objective of this work was to determine the concentration of the radionuclides ^{232}Th and ^{238}U and heavy metals such as As, Pb, Cd and Cr in different Fen samples in order to obtain an overall picture on dispersion to different environmental compartments. Relation between radionuclides levels and outdoor radiation doses, together with preliminary estimation of total contamination degree due to former mining activities, were also performed.

Material and methods

The Fen Central Complex is located in Nome municipality, Telemark County in Norway. This complex has carbonatite rocks of volcanic origin, with the most abundant rock types: Søvite (calcite carbonatite), Rauhaugite (dolomite carbonatite), Rødberg (hematite-calcite-carbonatite) and Fenite (alkali-metasomatised granitic gneiss) (Barth and Ramberg, 1966). As mentioned above, intensive mining activities in these locations ended during the 1960s. After that, several recordings of elevated natural gamma radiation doses in this complex were seen, either in portable hand detectors measurements (Dahlgren, 1983; Stranden, 1982) or in helicopter measurements (NGU, 2007).

Field work was conducted in May, 2008 (Fig.1). Terrestrial gamma radiation dose rates were measured by gamma detector (Automess AD4LF type 6150), at approximately 1 m distance above the ground. External gamma dose rates for different locations were calculated as the mean values from separate measurements done at each location. Eleven sampling sites were chosen in NORM and TENORM areas and outside these areas but within Fen. Geographical coordinates of sampling locations were read and recorded with portable GPS (Germin, USA).

Soil samples were collected at seven separate locations, with different gamma radiation levels. At each sampling site, soil was collected as bulk samples from minimum five points, at depth 0-20 cm, and then mixed to combine composite samples for every location. Rock samples of different sizes (3-15 cm) were taken randomly and later identified as rock type Søvite and Rødberg, typical for Fen geology complex. At least ten separate rocks were taken at every sampling location. After identification, rocks were laboratory milled to fine powder, homogenized and further decomposed for ICP-MS analysis of ^{232}Th, ^{238}U and heavy metal contents. Water samples were taken at five different locations. Water of Nordsjø Lake was sampled at four sites within the Fen public area and with different distance to area with high background radiation and at one site in Fengruve area rich in thorium bearing rocks. Physical and chemical parameters were measured *in situ* with portable pH-meter (WTW Multi 340i), previously calibrated with acid and neutral standard solutions. To study size distribution of radionuclides and heavy metals, i*n situ* size filtration with 0.45 µm membrane filter and ultrafiltration with 10 kDa hollow-fiber (Milipore) filter were conducted on Nordsjø lake samples from area with high background radiation. This was done to study behavior, mobility and possible impact on environmental compartments (Salbu et al., 2000). Water on other sampling sites was taken as total samples. All samples were conserved by acidifying with concentrated HNO_3 to obtain pH approximately below 2.

Soil samples were air dried in the laboratory, at ambient temperature for several weeks. Then, they were crushed and homogenized by sieving through 2 mm sieve. Aliquots of each were oven dried (105ºC) to constant mass prior to HNO_3 acid and microwave decomposition (Milestone Inc., Ultraclave High performance reactor, Shelton, CT) and analysis of heavy metals and radionuclides.

Concentrations of radionuclides ^{232}Th, ^{238}U and heavy metals As, Pb, Cd and Cr were measured with ICP-MS analysis using a Perkin ElmerSciex Elan 6000 (Norwalk CT, USA). In order to ensure analysis accuracy and precision, internal standards were added systematically to each sample before decomposition and appropriate standard reference materials were analyzed alongside the samples. All rock and soil samples were 1 % acid solutions. Each sample was ICP-MS determined by 3 injections. Detection limits were calculated as three times of standard deviation of measured blank samples.

Fig.1. Study area with marked sampling locations in Fen Norway (Norwegian Mapping Authority, 2010), (◯TENORM sites, ▭ NORM sites, △ background sites).

Results and discussion

Measurements of gamma ray radiation dose rates (Table 1) showed values ranging from 0.15 to 9.20 µGy/h for different Fen locations. These values are significantly higher than world average terrestrial dose rate of 0.059 µGy/h (UNSCEAR, 2000) and higher than gamma radiation dose rates found in similar investigations of other high NORM areas (Morales et al., 1996; Ramli et al., 2005; Prasad et al., 2008). Maximum gamma dose rate was measured at site G13, in area called Gruveåsen. These readings are related to the significant concentrations of ^{232}Th and ^{238}U found in rødberg rocks and soil samples from the same locations. High readings were also obtained at sampling locations in Fengruve area (F2, F3, F4 and F5) and in Søve area (S1). Locations Gruveåsen, Fengruve and Søvegruve (Fig.1) are previous mining sites and high values for radiation dose rates are directly consequence of TENORM. However, similar values (up till 3.24 µGy/h) were obtained for the measurements done at NORM site Rullekol, away from past mining sites and in the vicinity of houses. Other locations in Fen had lower gamma radiation dose rates measured, i.e. 0.11-0.25 µGy/h. Still, these values are approximately 2-4 times higher than world average level of 0.059 µGy/h published by

UNSCEAR (2000). Values obtained in this study are part consistent with previous measurements done in this Norwegian area (Stranden, E., 1982; Dahlgren, S., 1983; NGU 2007).

Table 1. Terrestrial gamma radiation dose rates at Fen sites.

Sampling sites	Terrestrial gamma dose rates (µGy/h), range
S1	2.09 - 8.50
F2	1.90 - 4.40
F3	3.30 - 3.70
F4	0.80 - 3.80
T1	0.15 - 0.35
F5	2.30 - 4.45
L1	0.11 - 0.34
R1	1.09 - 3.24
G11	4.11 - 5.08
G13	7.20 - 9.20
G14	3.80 - 4.50

The average concentration of ^{232}Th for rødberg rocks collected at Gruveåsen and Fengruve locations (Table 2) was 1552 mg/kg. The average concentration for ^{232}Th for søvite rock samples was found to be considerably lower, 64 mg/kg.

Table 2. Concentrations of ^{232}Th and ^{238}U in rocks from Fen area.

Rock type	^{232}Th (mg/kg)	^{238}U (mg/kg)	^{232}Th/^{238}U ratio
Rødberg	1552	6	342
Søvite	64	6	11

Results of previously reported rock samples analysis (Dahlgren 1983; Sundal and Strand 2004; NGU 2007), showed comparable concentration ranges for ^{232}Th in these rocks. Values for rødberg are much higher than typical values of 0.1-87.5 mg/kg for Norwegian bedrock (Nordic, 2000), giving the highest ^{232}Th level in Norwegian bedrock (Sundal and Strand, 2004). Uranium in rocks was present in much lower concentrations, 6.47 mg/kg and 5.53 mg/kg, for rocks rødberg and søvite, respectively. As presented in Table 2. value for ^{232}Th/^{238}U ratio for søvite rock type was 11.11 and for rødberg rock type was 342. These clearly distinguished values are reflecting the high difference in ^{232}Th concentrations and similar ^{238}U concentrations in the two different rock types.

Results of soil analysis (Table 3) showed wide ranges in radionuclides activity concentrations, 69-6098 Bq/kg and 49-128 Bq/kg, for ^{232}Th and ^{238}U, respectively. These values are obviously higher than the world average values for soil 30 Bq/kg for ^{232}Th and 35 Bq/kg for ^{238}U, given by UNSCEAR (2000).

Table 3. Activity concentrations of radionuclides and concentrations of heavy metals in soil from Fen area.

Sampling sites	^{232}Th (Bq/kg)	^{238}U (Bq/kg)	As (mg/kg)	Pb (mg/kg)	Cd (mg/kg)	Cr (mg/kg)
S1	69 ± 8	49 ± 12	4±4	47 ± 5	0.26±0.01	26±23
F3	4007 ± 1108	111 ± 37	20±5	134 ± 95	0.60 ± 0.14	33±19
F4	4563 ±93	111 ±37	17±1	121 ± 23	0.73 ±0.07	40±8
T1	1059 ±28	128 ± 4	7.1±0.6	287± 10	0.48±0.02	30.0±0.3
G11	6098 ± 1429	74 ± 62	20±6	55±7	1.88±0.59	54±14
G13	5781 ± 848	68 ± 6	20.3±0.9	65± 10	0.68 ± 0.07	66±4
G14	4843 ± 126	85 ± 5	16.7±0.2	59 ± 2	0.54 ± 0.07	89±3

The highest ^{232}Th value was found in soil in the area Gruveåsen (G11), and the lowest in the area Søvegruve (S1). The pattern of ^{232}Th distribution in soil is correlated with the abundance of rock type rødberg (with high ^{232}Th level) in Gruveåsen and Fengruve areas and with the abundance of rock type søvite (considerable lower ^{232}Th level) in area of Søvegruve. In general, soil concentration results suggest that erosion and weathering of rocks containing minerals rich in radionuclide during the time led to formation of soil layers significantly enriched in radionuclides.

To check the possibility for a multiple radiological and metal contamination in this area, soil samples were also analyzed for metals such as As, Cr, Cd and Pb. Obtained concentrations are given in Table 3. Arsenic concentrations in soil samples were higher than norm value of 8 mg/kg for non-polluted Norwegian soil (SFT, 2009). The range of Cd concentrations in soil was from 0.26 mg/kg up till 1.88 mg/kg. These values are higher than the average natural abundance of Cd in the earth's crusts, but are in ranges for soil in the vicinity of sedimentary or Fe rich rocks, 0.1-5.0 mg/kg (Cook and Marrow, 1995; Salbu and Steinnes, 1995). The average Pb concentration in samples of soil taken in Fen area was 110 mg/kg. Non-disturbed Norwegian soil has the average Pb concentration of 15 mg/kg, and range 10-19 mg/kg (Berg et al., 2003), but some places in southern part of country are more exposed to Pb pollution due to long range transport of pollutants and have 10 times higher concentrations than in central part of the country (Steinnes and Ruhling, 2002). A map of Pb distribution in Norway, given by Arctic Monitoring Assessment Program (AMAP, 1998), gave an approximately level of 40-60 mg/kg for Pb in Fen, Telemark area. Cr concentrations exceeded SFT (2009) norm values in samples taken at sites G11, G13 and G14, in wooden area of Gruveåsen.

Radionuclide concentrations in water taken in the Fen area ranged from 0.038-0.187 µg/L and 0.136-4.414 µg/L for ^{232}Th and ^{238}U, respectively. These numbers exceed values given by UNSCEAR (2000) for world average concentration in water that contribute to exposure from natural sources (0.012 and 0.083 µg/L for ^{232}Th and ^{238}U, respectively). However, values for both radionuclides are still far below the significant contamination levels in the guideline for drinking water (WHO, 2003). Previously, Stranden (1982) also reported no radiation dose contribution for Fen people, regarding Nordsjø lake water. Comparison of ^{232}Th and ^{238}U levels showed several times lower levels of dissolved thorium species in all water samples. This is in agreement with theory and generally low solubility of thorium due to high affinity for sedimentation and particle formation. Results obtained for heavy metals As, Pb, Cd and Cr were below the limit levels for non-polluted water (SFT, 2009).

Size fractionation of ^{232}Th in Nordsjølake samples gave results (Fig.3) as follows: 43.75% were present in particulate fraction, 48.44% as colloids and 7.81% as low molecular mass species. Regarding typical chemical behavior of ^{232}Th, we expected considerable higher percentage in particulate phase. However, more than 50% of ^{232}Th was present in form of mobilizable colloidal and low molecular mass species. Thus, colloidal transportation is expected and might induce further distribution and bioaccumulation in living species of this water system.

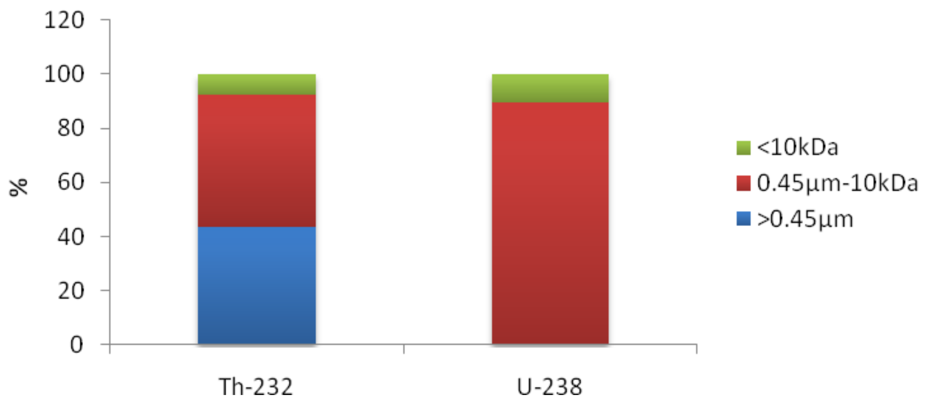

Fig. 2. Size fractionation of ^{232}Th and ^{238}U radionuclides in water samples from lake Nordsjø.

Analysis of uranium showed that ^{238}U was present mainly as colloids (90%) (Fig.2). On the basis of these results, ^{238}U is expected to be mobile, more soluble and could be subject of active uptake by aquatic biota in this water. Values for all measured metals were far below the levels that are expected to pose detectable effects on water living biota (SFT, 2009).

Conclusions

Measurements of gamma radiation dose rates due to NORM and TENORM and investigation of soil, rocks and water concentrations of radionuclides and metals were done in Fen area, Ulefoss, Norway. These activities were conducted as the first stage of environmental impact assessment of this specific area.

Obtained values for exposure dose rates ranged from 0.15 to 9.20 µGy/h. These values clearly exceed world average terrestrial dose rate of 0.059 µGy/h (UNSCEAR, 2000) and highly reflect the presence of radionuclides bearing minerals in volcanic rocks of this terrain. Two typical rock types rødberg and søvite analysis showed different levels of ^{232}Th and ^{238}U, but the results were consistent with previously reported studies (Dahlgren 1983., Sundal and Strand, 2004; NGU, 2007). The high concentrations of ^{232}Th in rødberg place this rock as the highest in ^{232}Th of Norwegian bedrocks.

Soil ^{232}Th and ^{238}U concentrations were found to be much higher than the Norwegian average soil values and obviously highly correlated with the presence of thorium and uranium rich rock types. Inhomogeneous distribution of radiation was found in the whole area. Radionuclides and heavy metals were particularly high in the vicinity of former iron mining sites. Analyzed water samples showed no contamination, neither with radionuclides nor with heavy metals, despite the high levels in surrounding media. It implicates that there are minor leaching or other transportation from these sites. Still, despite the low total levels, results of size fractionation, i.e., presence of all metals and high percentage of both radionuclides in the biologically most important - low molecular mass fraction, could be reasonable basis for investigation of possible accumulation in fish living in the lake.

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Thoron and its airborne progeny in Irish dwellings

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Abstract

During the period 2007-2009 long term indoor concentrations of thoron gas and airborne thoron progeny were measured using passive detectors in 205 dwellings in Ireland. Passive alpha track detectors of different types were used to make these measurements in the indoor air over a typical exposure period of at least three months. Thoron concentrations were measured using Raduet detectors supplied by Radosys, Budapest while thoron progeny concentrations were measured using passive detectors designed by the National Institute of Radiological Sciences (NIRS), Chiba, Japan. Radon concentrations were also measured in these dwellings using the standard alpha track detectors employed by the Radiological Protection Institute of Ireland (RPII). The main results obtained are given below. It should be noted that these results are based on the actual measurements made and are not seasonally adjusted. The estimated annual doses for thoron progeny (ThP) and radon (Rn) were calculated using dose conversion factors (DCF). The DCF_{ThP} used was based on the dosimetry models of Kendall and Phipps (2007) and of Ishikawa et al (2007). Exposure to indoor air containing thoron decay products at a concentration of 1 Bq/m^3 EETC (Equilibrium Equivalent Thoron Concentration) was estimated to result in an annual effective dose of 0.75 mSv. For radon a DCF_{Rn} of 1 mSv/year for indoor exposure under standard conditions to 40 Bq/m^3 was used.

Thoron gas activity concentrations ranged from < 1 to 174 Bq/m^3 with an arithmetic mean of 22 Bq/m^3. Radon gas concentrations ranged from 4 to 767 Bq/m^3 with an arithmetic mean of 75 Bq/m^3. The corresponding estimated annual doses are 0.1 (min), 19.2 (max) and 1.9 (arith.mean) mSv/year .Thoron progeny ranged from < 0.05 to 3.8 Bq/m^3 (EETC) with an arithmetic mean of 0.47 Bq/m^3 (EETC). The corresponding estimated annual doses are 2.9 (max) and 0.35 (mean) mSv/year.

It is of interest to note that in 14 of the 205 dwellings investigated the estimated annual dose from thoron progeny exceeded that from radon.

Introduction

There are over 30 known isotopic forms of radon. Most of these have very short half-lives in comparison to radon-222. Due to their very short half-lives they cannot migrate far from their origins and consequently are of no radiological health significance. One of them, however, is radon-220 (usually called thoron) and despite its short half-life (55.6 sec) may be present in indoor air at a sufficiently high concentration to be of some radiological significance. Thoron is a member of the thorium-232 decay series. A simplified version of the thoron decay scheme is shown in Figure 1. Henceforth in this paper, for convenience and in keeping with common practice, radon-222 will be referred to as "radon" and radon-220 will be referred to as "thoron". Unlike radon,which can enter a dwelling from the ground underneath it, indoor thoron due to its short half-life almost exclusively comes from the materials of the internal surfaces of rooms in dwellings.On a global basis ,in comparison with radon, information on the concentrations of thoron gas and in particular of its airborne progeny in dwellings is not very extensive (UNSCEAR 2000). Many of the values quoted in the literature are as a result of short-term or grab sampling and therefore are not suitable for estimating long term exposures in dwellings. For this and other reasons in most countries it has been difficult to estimate the long–term radiation dose and risk from indoor thoron and its progeny. In this paper exposure to indoor air containing thoron progeny at a concentration of 1 Bq/m^3 EETC (Equilibrium Equivalent Thoron Concentration) is estimated to result in an annual effective dose of 0.75 mSv. This dose conversion factor (DCF_{ThP}) is based on the dosimetry models of Kendall and Phipps (2007) and of Ishikawa et al (2007). For radon a DCF_{Rn} of 1 mSv/year for indoor exposure under standard conditions to 40 Bq/m^3 was used in this work. It should be noted that this DCF_{Rn} is presently under review and it is anticipated that a dose conversion factor of 1 mSv/year resulting from residential exposure to 30 Bq/m^3 will become the norm in many EU countries. This dose conversion factor is already being used by the World Health Organisation (WHO 2009).

Study outline

A pilot study of 40 Irish dwellings was carried out by University College Dublin in 2005 using passive CR-39 based detectors which give long term measurements of thoron and its progeny. This study found,for these dwellings, that the mean value of indoor thoron gas was 30 Bq/m^3 and the mean value of airborne thoron decay products was approx.1 Bq/m^3 (EETC) (Ní Choncubhair *et al*. 2008.) On the basis of the dosimetric modelling of Kendall and Phipps (2007) and Ishikawa et al (2007) the mean annual dose from thoron progeny in these dwellings is estimated to be 0.75 mSv/year. This is about 38 % of the estimated 2 mSv/year radiation dose that would be expected to be received by occupants of a typical Irish dwelling due to exposure to radon. Even though this pilot study cannot in any way be considered as representative of the national housing stock nevertheless it indicated that doses from thoron decay products are not always negligible and should not be ignored. It was therefore decided to carry out a larger survey of the concentrations of indoor thoron gas and its airborne decay products in Irish dwellings. This survey was carried out during the period 2007-2009 and was a collaboration between the Radiological Protection Institute of Ireland (RPII) and the National Institute of Radiological Sciences (NIRS), Chiba, Japan.

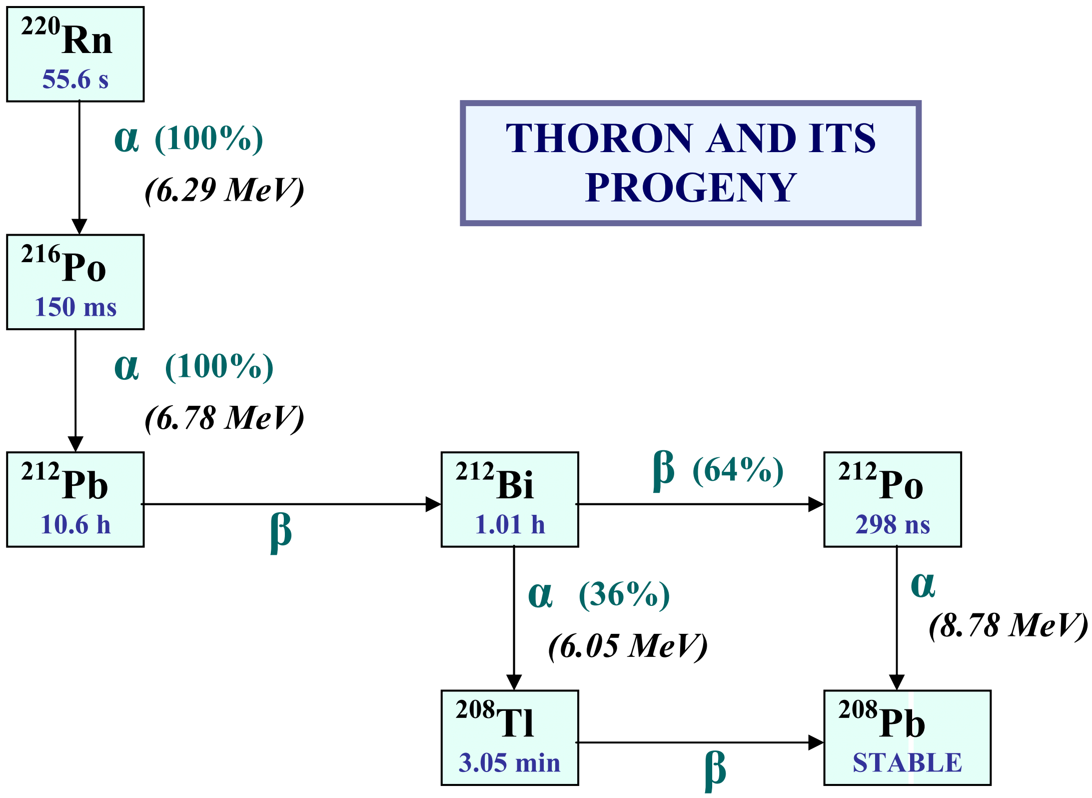

Figure 1. Thoron decay scheme.

The survey was conducted in two phases. Phase 1 comprised households which had participated in the past in the RPII national radon survey in the period 1992-1997 (Fennell et al.,2002) and which were in or around the major population centers of Dublin, Cork and Limerick. Phase 2 households were selected to increase the proportion of new houses in the survey and were comprised mainly of the houses of RPII staff and their families. Phase 2 participants were not part of the national radon survey but did have a previous radon measurement carried out in their home. As explained below the packages of detectors sent to the participating households gave a measurement of radon as well as of thoron and its progeny. Therefore, in addition to thoron/thoron decay product measurements the opportunity was also taken to make comparisons between the present radon concentrations in these dwellings and those in the past. Invitations to participate in the project were sent by post. In phase 1 a total of 365 invitations were issued of which 200 agreed to participate. Detectors for these households were issued in December 2007. During phase 2 detectors were issued to 40 households in June 2008. Of the 240 dwellings surveyed 82 were in or around Dublin, 82 in or around Cork and 69 in or around Limerick. Pre-paid post envelopes for return of detectors were supplied to participating households. A questionnaire dealing with house and household characteristics accompanied the detectors. Detectors were exposed in the dwellings for a minimum of three months in order to obtain long-term average measurements of thoron, thoron progeny and radon.

Satisfactory measurements of thoron and its progeny as well as of radon were completed in a total of 205 dwellings in Ireland. Due to the manner in which they were chosen the 205 dwellings investigated cannot be taken to be an unbiased or fully representative sample of the national housing stock. Nevertheless this number of dwellings,while small in absolute terms, should be seen in the context of the population of the Republic of Ireland which is approximately 4.5 million.

In this survey the long term measurements of thoron and its progeny and also of radon were made using three types of passive CR-39 alpha track detectors. Thoron gas concentrations were measured using Raduet detectors supplied and processed by Radosys (Budapest). The Raduet thoron detector,derived from an original design of NIRS, consists of two small diffusion chambers one of which is designed to admit both thoron gas and radon gas while the other admits only radon gas (Tokonami et al (2005)). The difference in the CR-39 alpha track density between the two diffusion chambers yields the mean thoron gas concentration over the exposure period at the measurement point in a dwelling. Airborne concentrations of the thoron decay product lead-212 were determined using a passive detector designed and developed by NIRS (Zhou and Iida,2000). In this detector CR-39 is covered with a Mylar film equivalent to an air thickness of about 7 cm so that only the 8.78 MeV alpha particles from polonium-212 can penetrate the Mylar film and are recorded by the CR-39. All other alpha particles from both the thoron and radon decay series , up to and including the 7.68 MeV alpha from polonium-214, have insufficient energy to penetrate the Mylar and are thus not recorded by the CR-39. Thus this detector type is sensitive only to polonium-212 even in the presence of radon and its progeny in the air. On the basis of laboratory calibrations at NIRS the concentration in the air of polonium-212 is determined .This in turn yields the concentration of lead-212 from which the airborne thoron progeny concentration can be expressed as Bq/m^3 EETC (Zhou and Iida (2000)). Radon gas concentration measurements were obtained by using the black SSI type diffusion detectors normally used by the RPII in its various radon surveys.In addition independent measurements of radon were obtained from the Raduet detectors.

As shown in Figure 2 detectors were mounted on a specially designed bar-coded plastic frame which could be easily hung on walls. For each dwelling two such detector frames were supplied. Householders were instructed to mount one of these in the livingroom and the other in the principal bedroom on a wall or other suitable surface. In Figure 2 the thoron/radon Raduet detectors are the two black detectors mounted at the bottom of the frame The NIRS thoron decay product detector is the square metal object mounted in the centre of the frame. The black detector at the top of the frame is the RPII detector.

Figure 2. Typical deployment of detectors in a dwelling.

Results and discussion

In this project thoron,thoron decay products and radon were measured in a total of 205 Irish dwellings. The results obtained are summarized in Table 1 and Table 2.

Table 1.

Thoron Concentration. Bq/m^3		Thoron Decay Products Concentration. Bq/m^3 (EETC)		Estimated Annual Effective Doses from Thoron Decay Products (mSv/year)	
Arith Mean	Maximum	Arith Mean	Maximum	Arith Mean	Maximum
22	174	0.47	3.8	0.35	2.9

Table 2.

Radon Concentration. Bq/m^3		Estimated Annual Effective Doses from Radon Decay Products (mSv/year)	
Arith Mean	Maximum	Arith Mean	Maximum
75	767	1.9	19.2

There are a number of observations that can be made regarding the estimated doses from thoron and radon progeny in the 205 investigated dwellings.

Tables1 and 2 clearly show that ,as not unexpected, in absolute terms the doses from radon and its progeny are much greater than those from thoron and its progeny. It is interesting to note,however, that in 14 or 7% of the dwellings the estimated doses from thoron progeny were greater than those from radon and its progeny.

The total per caput radiation dose in Ireland has most recently been estimated to be 3.95 mSv/year (Colgan *et al 2008)*. This includes an estimated contribution of 0.28 mSv/year from thoron decay products based on the 2005 pilot study of 40 dwellings (Ní Choncubhair *et al*. 2008.) If the measured data from the present study of 205 dwellings given in Table 1 were nationally representative the estimated mean dose of 0.35 mSv /year from thoron decay products would bring the total per caput annual dose to 4.02 mSv. The contribution from thoron progeny would then be about 9% of this total. It should be emphasized that this is a preliminary estimation and is subject to further review but does indicate that thoron derived doses are not always negligible.

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Radiation from geological samples in museums and showrooms

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Abstract

In the last two decades a considerable attention of the radiation safety community is paid to exposures due to natural radioactive materials based on fact that doses when handling such materials can be considerable. In general, the doses are mainly a result of inhalation of radon decay products. In order to identify all practices related to radiation sources, the Slovenian Nuclear Safety Administration (SNSA) as a regulatory authority preformed an extensive campaign of inspections in museums, research institutions, educational institutions and companies handling geological samples. Many radioactive items, mainly radioactive minerals and ores were identified. In some cases the dose rates at the contact of a specimen were few orders of magnitude higher than a natural background and elevated radon concentration was measured in rooms where specimens were collected. In general, users were not aware of risks associated with handling and storing such specimens. As a result of SNSA campaign radiation safety measures were implemented.

Countries use very different approaches regarding radioactive specimens in collections e.g. uranium and thorium minerals, radioactive fossils, petrified wood. No harmonisation on the international level exists regarding handling radioactive specimens in state and private collections resulting in difficulties in posing effective radiation protection control over such specimens.

Introduction

In the nature approximately about 340 nuclides can be found and among them around 70 are radioactive (Eisenbud 1997). Usually they are classified regarding their origin, i.e. primordial radioisotopes (e.g. ^{40}K, ^{87}Rb, ^{238}U, ^{232}Th), secondary radioisotopes and cosmogenic radioisotopes. Primordial radioisotopes have half-lives sufficiently long that they survived since their creation, i.e. comparable to the age of the universe. The secondary radioisotopes are the result of the decay of primordial radioisotopes while the cosmogenic radioisotopes are the result of the continuous bombardment of stable nuclei by cosmic rays (Holmes-Siedler 2006). In addition, the military activities and operation of nuclear installations affected the concentrations of some isotopes in the Earth (UNSCEAR 2000).

In the last two decades the international radiation safety communities e.g. IAEA, EC (IAEA 2003, EC 1999, EC 2002) paid a considerable attention to naturally occurring radioactive materials (NORM) as defined for example in ICRP publications (ICRP 2007) or to technologically processed materials and existence of technologically enhanced naturally occurring radioactive materials (TENORM). Exposures of workers handling such materials can be very high, in some cases even substantially higher than occupational exposures of workers dealing with the licensed radiation sources. Being aware of that, the radiation safety community has published first safety standards related to NORM e.g. given in (IAEA 2004). The natural radioactive materials are mainly used in industry with the highest radioactivity levels in uranium mining and lower levels in phosphate processing, oil and gas industry etc. The working conditions in industry are a subject to a control not only due to the exposure to the radiation but in many cases the good general safety practice to handle materials is enough also in order to provide sufficient radiation protection. For example, a use of a mask when handling dust materials can provide at specific circumstances a sufficient protection against internal exposure.

Natural radioactive materials are also quite frequently present in research laboratories and educational institutions. For example, the geological faculties usually have a collection of samples. Among more than 4000 minerals which are known today, some are highly radioactive e.g. uraninite, pitchblende, brannerite, coffinite or thorite. Even more, radioactive minerals are often present in museums and showrooms of minerals causing exposure of workers and the public. Private collectors of minerals very often handle natural radioactive materials. In addition, geological institutes usually keep geological samples during research projects which might be radioactive e.g. radioactive minerals or sands. Very often even after finishing research projects they keep such materials in their storages, which therefore can contain quite a lot of radioactive samples.

The radioactive samples are not only in a form of ores, minerals or sand, sometimes they are in form of petrified samples or even radioactive fossils. Such samples are for example handled outside geological departments of natural museums or faculties for natural sciences. Furthermore, radioactive samples can be a part of collections for a long period without noticing that they require special safety attention. They can be a part of private or institutional donations and they do not necessary have any connection to samples taken from local area, e.g. radioactive samples can be brought from very distant areas. The collectors from states with known areas of radioactive fossils or uranium mines could have higher probability to keep radioactive samples in museums, showrooms, storages of geological institutions and elsewhere where such samples were collected.

The persons handling such materials are very often not aware about the radiation risks they are exposed to when handling or storing such materials. Especially, when the above mentioned radioactive geological samples are not handled due to theirs characteristics related to radioactivity, users of such samples are not aware that they are actually handling radioactive materials and do not pay any attention to safety precautions.

Identification of radioactive samples

In order to establish the safety regime in institutions and other sites where radioactive geological samples are handled the identification of radioactive samples is necessary. A detailed analysis should follow regarding the main principles of radiation protection, namely justification, optimisation and dose limitation. According to the experiences radioactive samples are very often handled without basic knowledge of a justification principle and without any basic knowledge of safety measures. When the owners are faced radiation protection issues they often decide to reduce the number of exhibited samples. In such cases waste management issues also occur. Once the identification of samples is done the assessment of public and occupational doses when handling radioactive geological samples should be prepared and safety measures put in place.

The extensive campaign of identification conducted by the inspection of the Slovenian Nuclear Safety Administration in Slovenia, (SNSA, (http://www.ursjv.gov.si/en/), started a few years ago and the preliminary results of the campaign were published elsewhere (Janzekovic 2008). The SNSA is the responsible authority for control of radiation sources outside medicine and veterinary medicine. The campaign mainly focused on institutions which were also related to former uranium mine in Slovenia, the Zirovski vrh uranium mine. In a period 2007-2010 the SNSA identified many radioactive items in educational and research institutions, dealing with geological samples. The samples containing uranium and thorium can have different forms e.g. ores, minerals and petrified samples. The fossil of a mastodon tusk with elevated dose rate level at the contact was identified for instance and the analysis of its radioactivity is underway. Some samples inspected during the campaign are shown in Figure 1. The figures show a small portable radiation monitor named the radiation pager. The indication "9" shown on the pager indicates that the actual photon dose rate is above 38 μSv/h and the indication "4" indicates that the actual photon dose rate is above 12 μSv/h.

Fig. 1. Identification of elevated dose rates due to enhanced radioactivity of minerals (left, 2007) and a mastodon tusk (right, 2010).

The process of searching radioactive geological samples is similar to the process of detecting and finding other orphan sources described elsewhere (Janzekovic 2006). A well kept documentation regarding the origin of samples can facilitate the identification of radioactive material among usually a large number of non radioactive samples. The detection of photons originating from radioactive material was performed by detection of external radiation using instruments like hand held gamma spectrometry unit Fieldspec shown in Figure 2, already mentioned radiation pager and even contamination monitor as a very sensitive instrument. It should be noted that in many cases inspection identified higher dose rate at the lower part of the showcases where actually radioactive minerals were stored and were not exhibited. In addition, the staff of the institutions frequently stored radioactive items in their office, frequently in a very vicinity of their own working place.

Fig. 2. Measurements of the photon dose rate of the geological specimen using Fielspec and a use of a radiation pager for identification of elevated dose rates in 2007.

The detected photon dose rates at the contact with samples could be a few orders of magnitude above the natural background and no provisions to avoid internal exposure are usually in place. Besides the presence of uranium and thorium and theirs daughters, elevated levels of ^{40}K can be present in some samples e.g. in coal samples.

The geological samples with enhanced radioactivity were stored in collections, laboratories, offices, halls and storages. Very seldom only one radioactive sample is present at a particular site and usually few tens of them were collected or exhibited together. The shapes of specimens are very different as well as their homogeneity. The shielding of radioactive object by other objects can be a considerable problem in order to identify radioactive geological samples. Slightly elevated levels of radiation at some distances from the showcases can indicated the presence of radioactive samples with high radioactivity.

After the inspections the majority of samples were carefully investigated by qualified experts while the investigation of others is still underway. Because of a huge amount of items, namely few hundreds, the identification of radioactive specimens, e.g. uraninite minerals specimens from local uranium mine and the worldwide uranium mines, was very difficult. It was based on the exemption levels valid for ^{238}U and ^{232}Th in secular equilibrium given in EU directive (EC 1996) which are given in Table 1. The qualified expert developed simplified identification methodology by using data on measured dose rate, mass of an object and conversion factor between photon dose rate and activity concentration in order to compare the estimated activity concentration and appropriate exemption level. Details are given in (Stepisnik 2007). According to data from this reference less than 5 % of potentially radioactive samples identified at inspections in 2007 were later not identified as radioactive specimens.

Table 1. Exemption levels for ^{238}U an ^{232}Th in secular equilibrium taken from (EC 1996).

Series	Exemption levels	
$^{238}U_{sec}$	1 kBq/kg	1kBq
$^{232}Th_{sec}$	1 kBq/kg	1kBq

As reported in (UNSCEAR 2000) the global average values of ^{238}U and ^{232}Th in soil are about 30-50 Bq/kg. In specific soil and rocks the concentration can be some orders of magnitude higher. While processing natural radioactive material in secular equilibrium some elements are usually extracted and no secular equilibrium is present any more.

The enhanced external dose rate due to presence of radioactive geological specimen was used as an indicator for presence of enhanced radioactivity in specimens during inspections. From radiation safety point of view the presence of radon and corresponding dose to inhalation should be controlled. The staff, e.g. guardians of showrooms are present actually whole working day in showrooms. The preliminary results of measurement radon concentration in showrooms with mineral collections and in a storage with radioactive samples showed that the maximum ^{222}Rn concentration exceeded 1000 Bq/m^3 in a room with a mineral collection and reach about 1800 Bq/m^3 in a storage in a basement. Additional measurements are necessary in order to investigate the actually concentrations. The concentrations were measured in springtime and could be even higher in winter. The concentrations should be at least compared to concentrations recommended by the EC from (EC 1990), the IAEA from (IAEA 1996) and WHO levels published in (WHO 2009) and given in Table 2. It shows that the reference level for workplaces given by the IAEA is lower than measured concentrations. It should be taken into account that geological radioactive specimens are sometimes exhibited in numerous private collections mostly in private houses. As a result the reference levels for the members of the public should be applied in such cases.

Table 2. EC, IAEA and WHO recommendations related to reference levels for consideration of remedial actions, details are given in (EC 1990), (IAEA 1996) and (WHO 2009).

Exhibition site	Reference	Reference levels of radon indoor concentration
Public museums (workers)	Occupational exposure (IAEA 1996)	1000 Bq/m^3
Private collections (members of the public)	Existing buildings (EC 1990)	400 Bq/m^3
	Future constructions (EC 1990)	200 Bq/m^3
	Buildings (WHO 2009)	100 Bq/m^3
	Buildings in special circumstances (WHO 2009)	300 Bq/m^3

After the campaign of the SNSA remedial actions took place, e.g. around altogether 100 kg of radioactive materials were transported from an old storage of geological samples to the Storage for radioactive waste and a few hundreds of kilograms of radioactive waste originated from the educational institution were stored in the Storage for radioactive waste. The specimens which are still in use at other sites, e.g. geological samples used at a faculty, are subjects of regulatory control.

After the SNSA campaign the actual exposures of workers, students, collectors and visitors were efficiently lowered because the majority of sources were put in suitable storages and many safety precautions were implemented.

Radiation safety and radioactive geological specimens

Many radiation safety requirements during handling and storing the samples were ignored because they were not identified as radioactive materials or sources. The risk associated with radioactive geological samples originates manly from:

- external exposure due to decay of ^{238}U and ^{232}Th series
- internal exposure due to presence of long-lived radionuclides in a sample and radon with its short-lived decay products.

Both components should be actually assessed prior the handling, preparing and exhibition of geological samples. The assessment should be done for all lifetime phases of a use of radioactive samples i.e. from “its cradle to its grave” in order to avoid unjustified exposure of workers, including guardians of showrooms, researchers, students, visitors or private collectors. While prevention from external exposure can be achieved without huge expenses, the presence of accumulation of ^{222}Rn can be an important and even major issue. In order to safely handle radioactive geological samples many measures can be applied, either administrative or physical, e.g.:

- exhibit replica specimens in museum collections and limit handling radioactive specimens as much as possible
- ventilate a room before or during, as appropriate, when entering storages with radioactive materials
- use protective glass as for example lead glass to prevent exposure from beta and gamma radiation
- use gloves and other personal protective belongings when handling radioactive samples
- encapsulate specimens in showrooms in order to prevent contamination and radon dispersion
- control the contamination when handling specimens which crumble
- use separate and suitable storages for radioactive geological samples
- define the specific area in laboratory or showrooms where radioactive samples are handled, make the dose mapping, assess the doses and control the area
- put radioactive samples at remote area also if they are present in private collections
- control the donation of specimens from radiological point of view
- prepare the waste management plan
- establish written working procedures and maintain the records of samples
- inform students, researchers, guardians of showrooms and other stakeholders about working procedures and importance of following established rules.

Conclusions

Regarding the fact that external radiation and internal exposure due to radon are mostly neglected when handling radioactive geological samples, the need for action provided by a regulator is unavoidable in order to keep the system of radiation protection effective. The countries use very different approaches when handling radioactive geological and other specimens in museum and showrooms. No international guidelines are given in order to harmonise the approaches.

Due to a fact that radioactive samples can be easily transferred from one state to another, harmonisation of national legislations can be very useful. In general, four approaches are identified.

a. The authorisation of practice is required and radioactive minerals and other geological radioactive specimens are controlled as all radioactive sources, according to requirement of the (EC 1996). Such approach requires substantial work of qualified experts and regulatory authorities in order to implement the requirements.

b. The practice is registered under specific conditions e.g. handling of only ten radioactive geological specimens while authorisation is required, for example, for more than ten specimens.

c. The handling of radioactive specimens in a workshop is an authorised practice and a full scope of authorisation is required. The exhibition of a single specimen requires less stringent requirements e.g. a dose rate at a place of a visitor should be less than 20 µSv/h because the exposure of a visitor is a "transient one" e.g. a person will receive only a small dose due to staying a short time at that place.

d. The practice is exempted from the legislation.

Approaches b and c are based on a graded approach and try to balance the burden posed by the full scope authorisation and neglecting of radiation safety.

In general, no international harmonisation is achieved in the area of natural radioactive specimens. While some countries pose full scope control as for any other radioactive material, some countries use graded approach. In addition, in some countries no precaution principles are applied when handling radioactive samples e. g. fossils. In view of a basic radiation protection principles and taking into account that in general collecting such specimens does not require huge effort it would be beneficial to prepare specific international standards or guidelines in this area. In addition, controlling numerous private collectors is even more challenging issue.

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Natural radiation background time series from gamma detector stations in Iceland

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Abstract

We present here for the first time data from four continuously monitoring gamma detector stations in Iceland along with basic analysis. The results from all four stations show an average dose rate of 44-49 nSv/hr, and standard deviation around 5 nSv/hr (10 minute sampling cycle), with a distribution consistent with a Gaussian one.

Introduction

In 2005, the Icelandic Radiation Safety Authority (IRSA) began operating two gamma monitoring stations. In 2007, two more were added so now there is one station in each of the island's four corners, all near sea level (see Figure 1). The stations are maintained in cooperation with the Icelandic Met Office. Each station also collects pressure, temperature, precipitation and wind data for its location. The gamma dose rate and accumulated rainfall data is continuously relayed to a web server and can be monitored by visiting IRSA's web page here[1]. The data is monitored for extreme values and sudden changes.

Iceland's rock foundation contains relatively little uranium and its daughter products. Hence the environmental radioactivity there is rather low. Dust and radon gas carried by global weather systems from the continents on either side of the Atlantic and brought to the ground with precipitation can have a measurable effect on the background radiation signal.

This is the first publication of the collected data, but these stations are being added to the European Radiological Data Exchange Platform (EURDEP) real time gamma monitoring station network to ease access to the data.

[1] http://www.gr.is/gammastodvar/

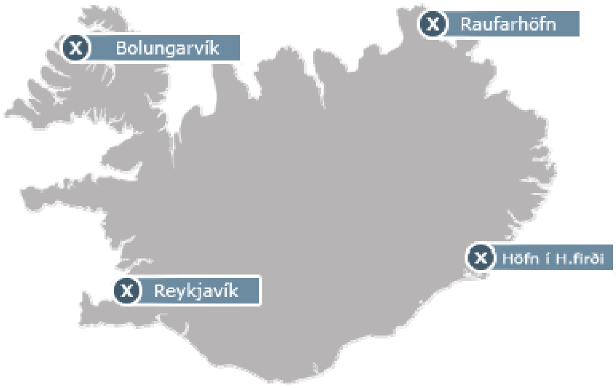

Fig. 1. Locations of IRSA's four gamma monitoring stations. Gamma background, rainfall, temperature, pressure, windspeed and wind direction is collected at these sites.

Material and methods

The gamma monitoring stations are equipped with a GammaTracer XL detectors from Saphymo[2]. Their measuring range is from 10 nSv/h - 10 Sv/h and their energy dependence is +/- 40% (over the range 45keV - 2MeV). The manufacturer provides the calibration (6% calibration error), and the tubes are recalibrated at each battery replacement (every 3-5 years). The measuring cycle is set to 10 minutes, with an estimated 1-sigma measuring error of approximately 5%. The gamma tracers are set to count for 10 minutes and write out to file.

The raw gamma dose rate data were at first cleaned to remove spurious data and obviously false readings. Further gaps (of order of a month and more) in the time stream are due to periods when the GammaTracer tubes undergo battery change and recalibration at the manufacturer's facilities. Due to growing pains (possibly a mistaken calibration) for the first year of operations, the first half year of data from the Reykjavik station has a clear upwards bias and was discarded off for the rest of the analysis. Cleaning these gaps out via automated algorithms proved tricky, so the time streams were trimmed 'by eye and hand'. Between 0.2 and 20% of the total data was thus discarded, most from the Reykjavik station.

For clarity and to reduce the effect of sampling irregularities, the data was collected into bins of 24 hours, and also into bins of 7 days.

We used the statistical computing environment R for this analysis.

Results

Summary statistics

The trimmed time streams are shown in figure 2 and a summary of the statistics from the cleaned data streams is given in Table 1. The bulk of the measurements values range from 40-60 nSv/hr, with a mean ranging from 44-49 nSv/hr and a standard deviation around 5 nSv/hr for the different stations.

[2] Manufacturer's information: `http://genitron.de/products/gamma_data_xl.html`

Table 1. Summary statistics of gamma dose rate data from the four stations. Note that data from each station come from strongly overlapping but not identical periods. All numbers in units of nSv/hr.

Station ID	Min	1Q	Median/Mean	3Q	Max
Bolungarvík	26	41	44 / 44.42	47	104
Höfn í Hornafirði	31	46	50 / 50.14	53	116
Raufarhöfn	30	47	50 / 49.97	53	161
Reykjavík	31	46	49 / 49.16	52	96

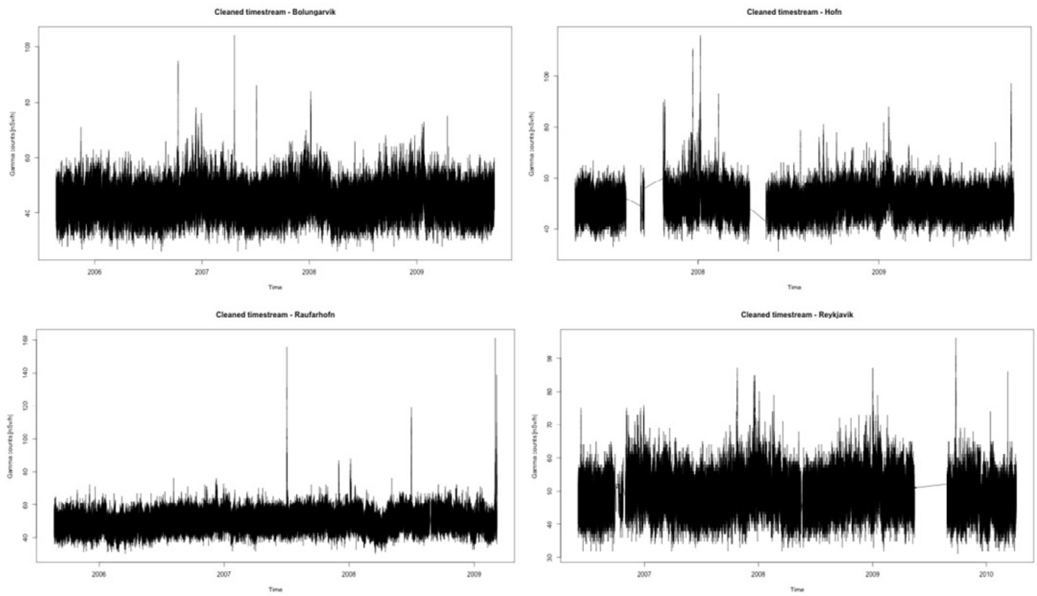

Fig. 2. Gamma dose rate time series data for the four stations, after pruning. The range can discerned, as can the larger gaps and some outlier values which were not obviously spurious.

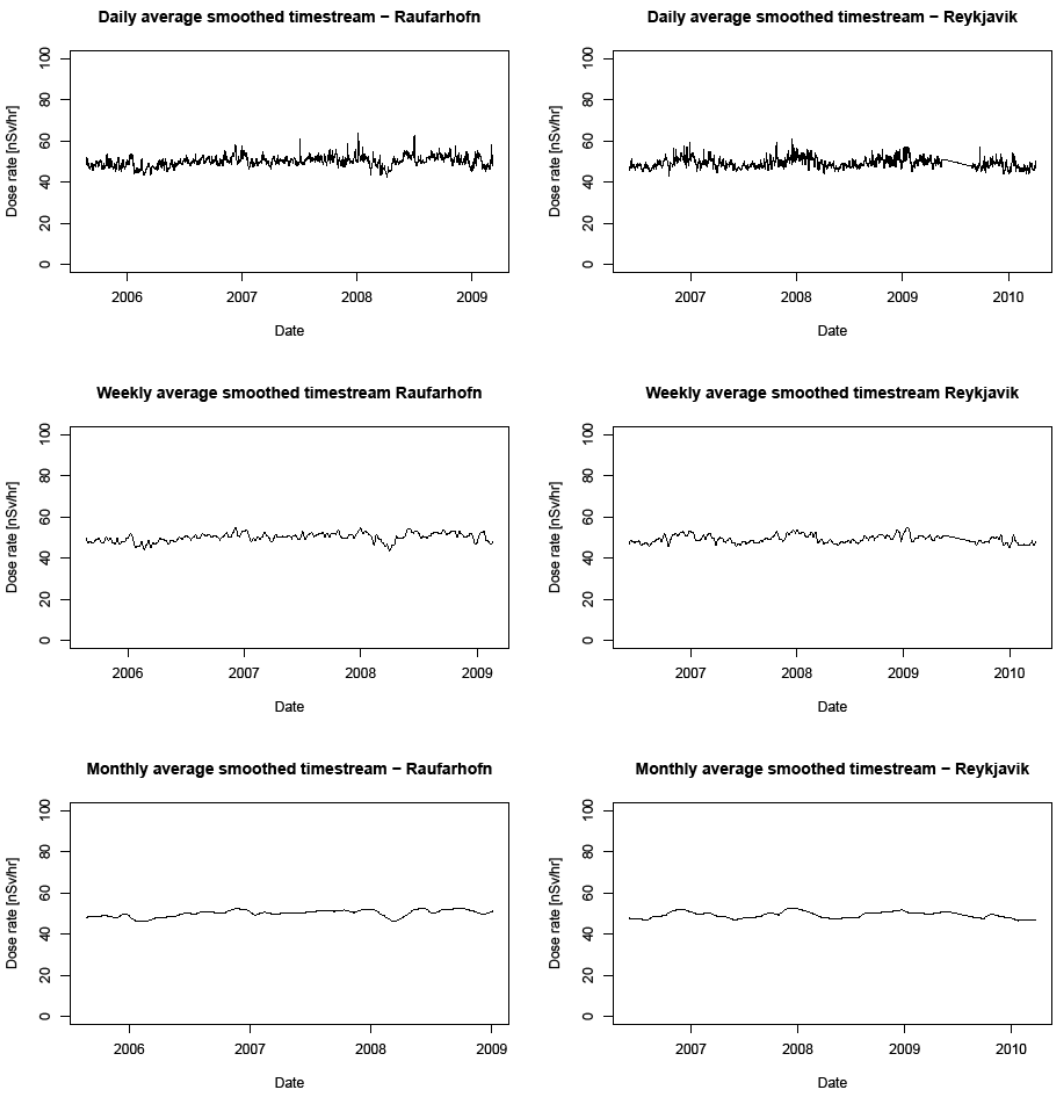

Fig. 3. 1-day, 7-day and 30-day moving average smoothing of cleaned gamma dose rate time series data for the Raufarhöfn (left) and Reykjavík (right) stations.

Dose rate distribution histograms

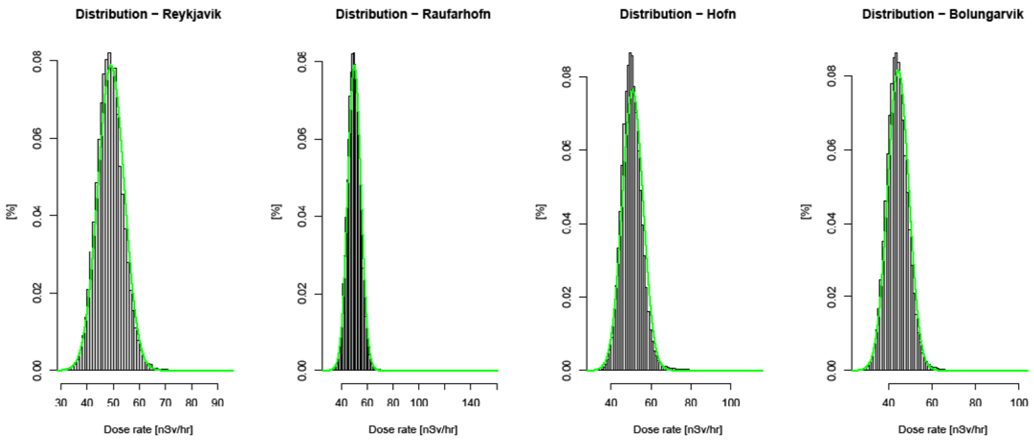

Fig. 4. Gamma dose rate distribution histograms for the four stations. The green line overlay shows a normal probability distribution with the data mean and standard deviation. The long tail towards higher values skews the distribution only slightly from a Gaussian.

Discussion & Conclusions

We see that environmental radioactivity levels are rather low and stable. The measurements are close to normally distributes, and the mild seasonality that can be spotted can be adequately explained with precipitation.

The data presented here will be made available to the network of gamma stations using the Eurdep data format. The data contain a many gaps and spurious values which need to be cleaned before a sensible analysis is made.

We thank Sigurður Emil Pálsson for his assistance and good advice.

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Localities in Montenegro with elevated terrestrial radiation

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Abstract

Research aimed to find localities in Montenegro with elevated terrestrial gamma background and recognize sources of their radiation was conducted during the period 2008-2009. For this purpose, 135 localities which have geological formations known to contain minerals with potentially high concentrations of U, Th and K, were selected throughout the country for dosimetric survey.

Knowing from earlier investigations that the average absorbed dose-rate in the air, 1 m above the ground, in Montenegro is 55 nGyh^{-1}, it was arbitrarily adopted that only localities with absorbed doses at least 50% above this average value will be considered as having relatively elevated radiation background. Field measurements have shown that 43 of the surveyed localities have such dose values. From these 43 localities, soil samples have been collected for further laboratory radiation characterization using a gamma spectrometry method. In that way, activity concentrations of ^{40}K, ^{232}Th, ^{235}U, ^{238}U, ^{226}Ra and ^{137}Cs were determined in the samples.

Among 10 sites with the highest radiation background in Montenegro (110-192 nGyh^{-1}), two sites are on andesite volcanic rock, one on fluvio-glacial sand and gravel, while the rest lie on bauxite deposits. Compared to the other known areas of high natural radiation background in the world, all these localities in Montenegro have a moderately elevated level of radiation background. Gamma spectrometry has revealed that a high content of K (up to 3341 Bqkg^{-1}) was the main source of elevated radiation at the sites on andesite rocks, and high U and Th content (up to 285 Bqkg^{-1} of ^{238}U, and up to 201 Bqkg^{-1} of ^{232}Th) at the sites on bauxite deposits.

Introduction

During the two-year period 2008-2009, within a project of the Montenegrin Academy of Sciences and Arts, a research was carried out with the aim to determine localities in Montenegro with relatively elevated terrestrial gamma background and to recognize sources of their radiation.

For planning field experiments, voluminous documentation on geology and pedology of Montenegro (Mirkovic et al. 1985; Djuretic et al. 1986) was used and analyzed. An earlier systematic investigation of terrestrial gamma radiation in Montenegro (Vukotic et al. 1997), performed 15 years ago, served as a base of this project. That investigation has shown that the average absorbed dose-rate in the air, 1 m above the ground, in Montenegro is 55 $nGyh^{-1}$ (according to the UNSCEAR 2000 Report, the average value for the other 7 countries of South-East Europe is 62 $nGyh^{-1}$). Therefore, we arbitrarily adopted that, within this project, a locality in Montenegro will be considered as having relatively elevated external radiation if the dose-rate in the air at that site is more than 50% higher than the country average, i.e. if the dose-rate is above 80 $nGyh^{-1}$. Soil samples were collected from such sites, and analyzed using a laboratory gamma spectrometry method.

In this article, only results for localities with dose-rates in the air which are at least twice higher than the country average ($D \geq 110$ $nGyh^{-1}$) are presented in detail.

Material and methods

Geological setting and selection of localities for survey

Before defining a sampling grid covering the territory of Montenegro and selection of localities where dosimetric measurements will be performed, a detailed analysis of structural-tectonic and structural-metallogenic regionalization of geological and ore formations was conducted. During this analysis, in planning field investigations, a special attention was directed towards the volcanic rocks and volcano-clastites of acid composition, Paleozoic sandstones and schists, flysh sediments, deposits of red and white bauxites, and Quaternary sediments (sands, clays, peats), for which there are indications they could contain minerals with elevated concentrations of U, Th and K. Less attention was given to carbonates (limestones, dolomites and marls) and to volcanic rocks of basic character.

Four geotectonic zones, with distinctive geological evolution during the last 200 million years, could be clearly recognized in Montenegro (Mirkovic et al. 1985), as presented in Figure 1. They are known as the Adriatic-Ionian Zone (JZ), the Budva-Cukali Zone (BZ), the High Karst Zone (VK), and the Durmitor Tectonic Unit (DTJ). The metallogenic zones in Montenegro correspond to these geotectonic units.

The coastal areas of the town of Ulcinj and partly of the towns of Tivat and Herceg Novi belong to the JZ zone, which is composed of Upper Cretaceous limestones, dolomites, dolomitic limestones; Middle Eocene foraminifer limestone and flysch; Middle and Upper Eocene flysch sediments and Middle Miocene sands, sandstones, clays and lithothamnium limestones.

The BZ zone encompasses the narrow coastal areas of the towns of Bar, Budva, Kotor and partly Tivat and Herceg Novi. Its geological structure consists of Mesozoic carbonates and eruptive rocks; Anisian and Paleogene flysch.

The central and southern parts of Montenegro belong to the VK geotectonic unit, whose geological structure is predominated by Mesozoic carbonate sediments, with occurrences of red and white bauxite formations, Triassic volcanic rocks, Paleogene flysch sediments and Quaternary sediments.

The DTJ zone encompasses the north-eastern part of Montenegro. Its terrains are composed of Paleozoic and Lower Triassic clastites; Triassic and Jurassic carbonates, with significant presence of Middle Triassic and Upper Jurassic volcanic rocks and volcano-clastites; sediments of Neogene age and Quaternary formations.

A sampling grid and a number of surveyed localities are planned for each of these four geotectonic unites individually. In total, 135 localities were selected in this manner for field investigations, 29 of them belonging to the JZ zone, 20 to the BZ, 38 to the VK zone and 48 to the DTJ, as it is presented in Figure 1. Geological, pedological and morphological characteristics of these localities were then described in detail.

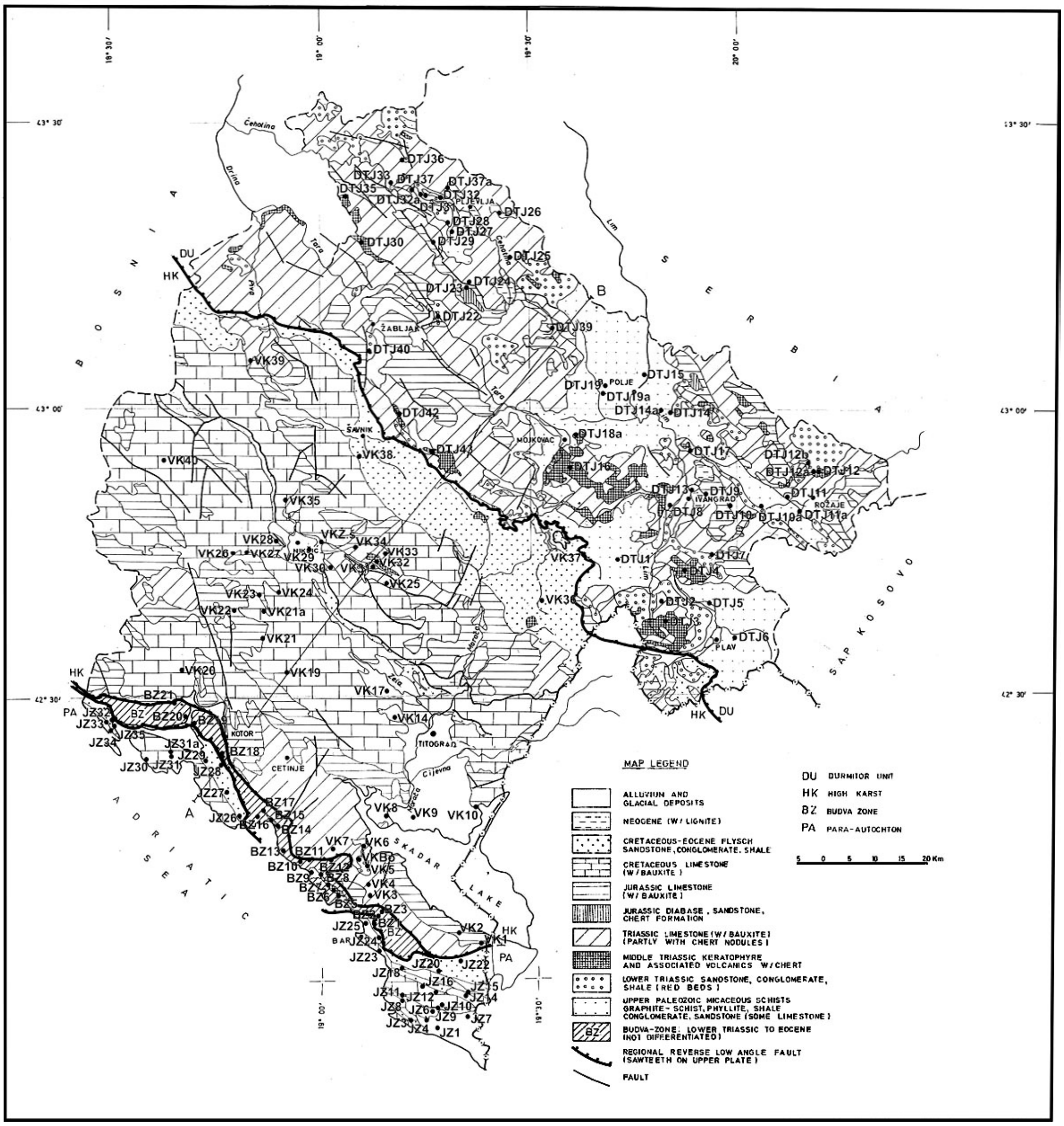

Fig. 1. Geological map of Montenegro, with indication of surveyed localities.

Dosimetry

Readouts of the gamma radiation count-rate meters CANBERRA Inspector (with NaI 1.5"x1.5" probe), VICTOREEN 190 SI (NaI 2"x2"), THERMO RadEye PRD (NaI probe) and THERMO FH 40 proportional counter were compared at several different sites under the same measuring conditions. This comparison has shown that the most consistent results were obtained with instruments VICTOREEN 190 SI and THERMO RadEye. Therefore, we decided to use the VICTOREEN 190 SI device for measurement of absorbed dose in the air, and to make control measurements, periodically, with the THERMO RadEye.

Dosimetry was then performed at all of the 135 selected localities, in the air 1m above the ground, during dry weather conditions. At each locality 10 measurements were carried out in an attempt to cover a broader area of the locality. The range of doses was then determined and the average dose-rate for the locality calculated on the basis of these 10 measurements. The GPS coordinates of the surveyed sites were also registered, as well as the geological and pedological observations.

Gamma spectrometry

Soil samples were taken, for a subsequent laboratory analysis, from all localities where an average dose-rate was found to be higher than 80 nGyh^{-1}. They were collected at the sites following a standard core sampling procedure (HASL-300). Sampled material was then ground, dried 8 hours at 100 °C, sieved through 2 mm mesh and packed into 1 liter Marinelli beakers. Analytical samples, prepared in this way, are counted 40 days later, after reaching radioactive equilibrium between ^{226}Ra and ^{222}Rn.

All these samples were analyzed using a low background gamma spectrometry system consisting of two HPGe detectors, which have 36% and 41% efficiency, 1.72 keV and 1.80 keV resolution (FWHM 1.33 MeV ^{60}Co), and 0.98 imps^{-1} and 1.22 imps^{-1} background count-rate at 40-2700 keV energy range, respectively. Activity concentrations of the following radioisotopes have been determined in the samples: ^{40}K (1460.75 keV), ^{232}Th (338.32 keV, 911.20 keV), ^{235}U (143.76 keV, 163.33 keV), ^{238}U (1001.03 keV), ^{226}Ra (295.22 keV, 351.93 keV, 609.31 keV, 1120.2 keV, 1764.4 keV), and ^{137}Cs (661.62 keV).

Results

An average dose-rate which is 50% higher than the country average was found at 43 localities: 5 in the JZ zone, 1 in the BZ, 18 in the VK, and 19 in the DTJ.

At only 10 of these localities the dose-rate is twice higher than the country average, i.e. D ≥ 110 nGyh^{-1}. These localities and characteristics of their radioactivity - dose-rates in the air and activity concentrations in the ground, are given in Table 1.

Discussion

It is evident from Table 1 that all localities with the most elevated radiation background belong to the VK zone, except one which is in the DTJ. There are none in the geotectonic zones JZ and BZ, which completely cover the coastal area of Montenegro and its hinterland.

As the highest dose-rate in the air 1 m above the ground was found to be 192 nGyh^{-1}, and while all the others are bellow 150 nGyh^{-1}, it can be concluded that these

10 localities with the highest external radiation in Montenegro, in comparison with other known areas in the world with high natural radioactivity (UNSCEAR 2000), form part of those areas with a moderately elevated radiation background.

Table 1. Dose-rates in the air and activity concentrations in the ground at the localities with the highest terrestrial background radiation in Montenegro.

Locality	D (nGy/h)	^{40}K (Bq/kg)	^{232}Th (Bq/kg)	^{235}U (Bq/kg)	^{238}U (Bq/kg)	^{226}Ra (Bq/kg)	^{137}Cs (Bq/kg)
VK/7a	192	2329 ± 75	51.2 ± 1.8	2.3 ± 0.9	64.8 ± 13.1	31.2 ± 1.1	5.6 ± 0.2
VK/20	125	27.4 ± 1.8	194 ± 6	4.1 ± 1.2	113 ± 14	73.0 ± 2.3	20.2 ± 0.6
VK/21a	131	159 ± 5	126 ± 4	7.8 ± 0.8	137 ± 13	92.3 ± 3	20.2 ± 0.7
VK/23	137	109 ± 4	129 ± 4	4.1 ± 0.7	73.5 ± 15.5	76.1 ± 2.4	< 0.43
VK/24	134	164 ± 6	169 ± 5	9.4 ± 1.3	151 ± 20	151 ± 5	5.2 ± 0.3
VK/25	149	173 ± 6	196 ± 6	2.7 ± 0.5	66.0 ± 9.5	48.8 ± 1.6	0.79 ± 0.11
VK/27	115	145 ± 5	166 ± 5	16.2 ± 1.1	285 ± 19	239 ± 7	1.40 ± 0.15
VK/33	110	109 ± 4	201 ± 6	4.9 ± 0.7	114 ± 13	66.8 ± 2.2	2.95 ± 0.17
VK/35	112	146 ± 5	198 ± 6	2.0 ± 0.5	38.0 ± 9.6	43.4 ± 1.4	4.3 ± 0.2
DTJ/42	148	3341 ± 106	71.9 ± 2.5	2.8 ± 0.7	65.0 ± 34.4	33.2 ± 1.1	0.41 ± 0.15

Two of 10 localities with the highest terrestrial radiation background in Montenegro, VK/7a (Virpazar settlement) and DTJ/42 (Savnik settlement), are characteristic with the presence of andesite volcanic rock of Middle Triassic, one (VK/20 near the town of Niksic) with fluvio-glacial gravel and sand, while the other 7 localities are above deposits of white and red bauxites (VK/25, VK/27, VK/33, VK/35 in the surroundings of Niksic, and VK/21a, VK/23, VK/24 in the surroundings of the town of Cetinje, towards Niksic).

Table 1 shows, and the same is with the other 33 samples analyzed using gamma spectrometry, that activity concentrations of ^{137}Cs are relatively low, mostly far below a level of 20 Bqkg^{-1}, which was the contamination level of the territory of Montenegro before the Chernobyl accident. This means that cesium of Chernobyl origin, after more than two decades of presence in this terrain, partly decayed and significantly migrated from surface into deeper layers of soil, and it is definitely not a cause of elevated radiation doses at the investigated 43 localities.

The average activity concentrations of ^{40}K, ^{232}Th, ^{238}U and ^{226}Ra for 7 countries of South-East Europe are 433 Bqkg^{-1}, 35 Bqkg^{-1}, 52 Bqkg^{-1} (for 4 countries), 35.5 Bqkg^{-1}, respectively (UNSCEAR 2000), while these values for Montenegro are somewhat lower: 246 Bqkg^{-1} for ^{40}K, 24 Bqkg^{-1} for ^{232}Th and 29 Bqkg^{-1} for ^{238}U (Vukotic et al. 1997). Accordingly, Table 1 reveals that the source of elevated radiation doses at the localities VK/7a and DTJ/42, which are characterized with the presence of andesite rock, is a high concentration of K (up to 3341 Bqkg^{-1}) and, in a certain measure, of U in the ground. At the 7 localities, where the bauxite is predominant, the elevated contents of Th (up to 201 Bqkg^{-1}) and U (up to 285 Bqkg^{-1} of ^{238}U) in the ground are the cause. This is in accordance with an earlier investigation of radioactivity of Montenegrin bauxites (Vukotic 1981) which has proved high concentrations of U and Th in them, and also with investigations of radon (Antovic et al. 2007; Vukotic et al. 2008) which have revealed that, in Montenegro, the highest indoor-radon level is in the town of Niksic (the main Montenegrin bauxite resources are on territory of this municipality), which is three times higher than the world average and two times higher than the average in the countries of South-East Europe.

Conclusions

The localities in Montenegro with the highest external radiation belong to those world areas with moderately elevated radiation background.

Cesium of Chernobyl origin, after more than two decades of presence in the terrains of Montenegro, partly decayed and migrated into deeper layers of soil, and is not a substantial cause of elevated radiation doses.

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Identifying the presence of orphan radioactive sources in dwellings from the vicinity of former uranium mines (Portugal): a methodological approach

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Abstract

Several radiological measurements were taken in two dwellings in the close vicinity of a former uranium mine (Urgeiriça, Central Portugal), in order to search for the presence of radioactive materials resulting from the mining activities. For the purpose, indoor radon concentration, gamma radiation flux, absorbed dose, radon gas flux from the floors and radon concentration measurements in crawl spaces were carried out.

The results showed important differences between the several parameters measured for both dwellings, reaching sometimes a few orders of magnitude. In one case, radioactive waste was identified in the crawl space, with the presence of these materials contributing to an excess dose of about 50 mSv/year, assuming as background the values measured in the second dwelling where no radioactive waste was found.

The methodology developed for this study proved to be a time/cost-effective procedure, capable of being applied in buildings of different construction types.

Introduction

For nearly one century, 61 uranium mines were exploited in central Portugal, generating 13 million tones of different kinds of wastes, accumulated normally in the vicinity of the exploitations. Over the years, some of these wastes were used as construction materials, usually in roads, but sometimes also in residential buildings; in this last case, the dose increase due to the exposure to ionizing radiation can be significant.

To assess the possible use of mining wastes in residential constructions, as well as the associated dose increase resulting from exposure to ionizing radiation, a methodology capable of being applied in buildings of different construction types and producing a time/cost-effective discriminating procedure was developed and tested. The methodology accounts for the challenge of discriminating between high radiation levels of purely natural origin, since the region is also radon prone, and those induced by the possible presence of radioactive materials resulting from mining activities.

Two dwellings located in the vicinity of the Urgeiriça mine (central Portugal) were used for test purposes. This was the most important uranium exploitation in Portugal, and also the place where industrial facilities were used to process ores extracted from all Portuguese mines. As a result, here can be found the largest amounts of mining wastes produced.

The uranium-bearing ore of Urgeiriça is of vein-type, striking N60ºE; it contains pitchblende and uraninite, as well as several secondary uranium minerals (Neiva, 1968). Hercynian porphyritic medium to coarse grained biotite granites are the dominant rocks observed in the area; non porphyritic two-mica granite rocks, as well as tertiary sedimentary, also occur in the region. Granites show uranium contents in the range 6 to 15 ppm, and soil gas radon measurements comprised between 29 and 593 kBq.m^{-3}, with a geometric mean of 110 kBq.m^{-3} (Neves et al. 2003). All granites are intersected by several sets of faults, with NNE to ENE dominant directions and frequently showing some degree of uranium enrichment (some dozens up to a few thousand ppm), resulting from the presence of secondary uranium phosphates (e.g. autunite, torbernite). In these faults and in their vicinity, soil gas radon measurements are frequently in the order of several hundred thousand up to a few million Bq.m^{-3}.

The referred geological setting results in a radon prone region, where indoor radon concentrations are in excess of recommended values for a large proportion of dwellings. In previous work, Neves et al. (2003) have studied 196 dwellings of the region with passive detectors, and obtained a geometric mean of 291 Bq.m^{-3} for indoor radon. Thus, distinguishing between natural or anthropogenic sources of indoor radon can be a challenge in this region.

Material and methods

In both dwellings used as a test case, a set of radiological data was collected, including the evaluation of indoor radon, gamma ray flux, absorbed dose, radon gas flux from the floors and also radon concentration in sub-slabs and crawl spaces.

Indoor radon concentrations were evaluated with CR39 detectors (n=7), exposed in several compartments for approximately 3 months; VOID labels were used to ensure that the detectors were not moved during the period of measurement (Fig. 1). After collection, the detectors were etched in a sodium-hydroxide solution during 270 minutes, at a temperature of 90ºC. The density of alpha impacts was estimated with an automated track-counting system (Radosys™), and the results converted to Bq.m^{-3} using appropriated factors, determined from the exposure of detectors in calibrated radon chambers.

Fig. 1. CR39 passive detector secured with a VOID label.

The gamma radiation flux from pavements, walls and ceiling materials of all rooms were measured using a NaI Saphymo^{TM} SPP2-NF gamma scintillation detector. The density of measurement points obtained in each room was controlled by the absorbed dose data; where this value was high, indicating the possible presence of radioactive materials, a more accurate screening was carried out, thus locating radioactive sources, its intensity and size. A total of 102 measurements were taken; results are given in cps.

The determination of the absorbed dose (external radiation) was carried out with an Exploranium™ GR130G, positioned at a height of approximately 1 meter from the floor. The values are given in µGy.h-1 and each measurement is representative of approximately 16 m2. A total of 16 measurements were performed. The purpose of this technique was to obtain an estimate of the environment gamma radiation that can immediately indicate the presence of radioactive materials, in the floor or walls of the analyzed room.

To determine radon gas flux from the floors, a radon-box was used as an accumulator, which remained sealed to the floor for 24h with the open surface downwards. The radon concentration was then measured with a Genitron-Saphymo™ AlphaGuard Pro (Fig. 2, left side) and radon gas flux from the floor/atmosphere interface were subsequently calculated. A total of 5 measurements were taken; results are given in Bq.m-2.h-1.

Radon concentration in crawl spaces was also measured with a Genitron-Saphymo™ AlphaGuard Pro, using appropriate probes to collect air samples (from 1.5m to 4.5m) through the outside vents (Fig. 2, right side). A total of 6 measurements were taken and the results are given in Bq.m^{-3}.

Fig. 2. Radon-box technique measurement with Genitron-Saphymo™ AlphaGuard Pro (left) and radon concentration measurement in a crawl space (right).

Results

Both dwellings were built in the 1950's with the same materials, by the same constructor and suffered minor refurbishment work over the years. Their plans are slightly different as a result of their size, but the construction techniques and project are basically the same.

Since both houses have wooden floor, they were built with crawl spaces beneath the floor, in some places more than 2 meters high.

Dwelling #1 is a two storey-house with 8 rooms in the ground floor and 2 rooms in the first floor. Indoor radon concentration was measured with CR39 detectors in the ground floor rooms D1 (living room), D2 (bedroom), D3 (bedroom) and D4 (dining room); results shown in table 1.

The minimum values for the absorbed dose, gamma radiation flux, radon flux from the floor and radon concentration in crawl space values were 0.20 μGy.h^{-}1 (D3), 250 cps (D7), 38.6 Bq.m^{-2}.h^{-1} (D1) and 5720 Bq.m^{-3} (D1), respectively. The highest absorbed dose, gamma radiation flux, radon flux from the floor and radon concentration in crawl space values were 0.69 μGy.h^{-}1 (D1), 1600 cps (D1), 314.7 Bq.m^{-2}.h^{-1} (D4) and 14160 Bq.m^{-3} (D6), respectively.

Table 1. Summary of the values measured for each room (mean values) in dwelling #1; n.d.-not determined; A-external radiation; B-gamma radiation flux; C-radon flux; D-radon concentration in crawl space; E-indoor radon concentration.

Room	A [μGy.h^{-1}]	B [cps]	C [Bq.m^{-2}.h^{-1}]	D [Bq.m^{-3}]	E [Bq.m^{-3}]
D1	0.60	859	38.6	7614	2948
D2	0.37	537	154.7	11341	3405
D3	0.20	310	60.1	n.d.	2371
D4	0.25	310	314.7	n.d.	2435
D6	0.43	838	n.d.	14160	n.d.
D7	0.40	650	n.d.	n.d.	n.d.
D8	0.31	453	24.2	n.d.	n.d.

Dwelling #2 is a one storey-house with 5 rooms built over a crawl space. Indoor radon concentration was measured with CR39 detectors in rooms D1 (bedroom), D2 (kitchen), and D4 (living and dining room); results are shown in table 1.

The minimum values for absorbed dose, gamma radiation flux and radon concentration in crawl space values were 0.14 μGy.h^{-}1 (D5), 160 cps (D1 and D5) and 1528 Bq.m^{-3} (D2), respectively. The highest values for absorbed dose, gamma radiation flux and radon concentration in crawl space values were 0.17 μGy.h^{-}1 (D3), 200 cps (D3) and 1629 Bq.m^{-3} (D3), respectively.

Table 2. Summary of the values measured for each room (mean values) in dwelling #2; n.d.-not determined; A-external radiation; B-gamma radiation flux; C-radon flux; D-radon concentration in crawl space; E-indoor radon concentration.

Room	A [μGy.h^{-1}]	B [cps]	C [Bq.m^{-2}.h^{-1}]	D [Bq.m^{-3}]	E [Bq.m^{-3}]
D1	0.15	175	n.d.	n.d.	351
D2	0.15	175	n.d.	1528	356
D3	0.17	188	n.d.	1629	n.d.
D4	0.15	177	n.d.	n.d.	312
D5	0.14	177	n.d.	n.d.	n.d.

Discussion

Given these results, it was possible to identify the presence of allochthonous radioactive materials in dwelling #1. These were only used in the crawl spaces beneath the floor, specially on it's dirt floor, foundation walls and footing; it's detection was possible due to the measurements of gamma radiation flux (maximum of 1600cps) and external radiation data (maximum of 0.69 μGy.h^{-}1).

Radon concentration in crawl spaces also showed very high values (maximum of *ca*. 14000 Bq.m^{-3}), especially if we take into account the mean height of these spaces, which is around 2m, and the fact that some degree of natural ventilation exists.

Given that no indications of the presence of radioactive wastes was found in dwelling #2, its radiological values were assumed as representative of background values, since they also correlate well with results from others studies carried out in similar geological settings where no mining activities were carried out (Pereira et al. 2003).

Conclusions

The results showed important differences between the several radiological parameters measured, reaching sometimes a few orders of magnitude. In dwelling #1, radioactive waste was identified in the sub-slab, with the presence of these materials contributing to an excess in dose of about 50 mSv/year, assuming as background the values measured in the second dwelling, where no radioactive waste was found. The very high indoor radon levels are the main contributors to this excess value (more than 95% of the total dose).

The methodology developed for this study proved to be a time/cost-effective procedure, and was subsequently applied to more than 75 dwellings of the region with good results.

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Bioremediation of land contaminated by radioactive material

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Abstract

Objectives: Radionuclide pollution arises as a result of many activities, largely industrial, such as mining and the provision of nuclear energy. These pollutants are discharged into the atmosphere and aquatic and terrestrial environments and may reach high concentrations, especially near the site of entry for point source emissions, and/or be transported between different environmental compartments. Metallic radionuclide has also entered the environment as a result of weapons-testing and accidents such as Chernobyl. The effects of radionuclide in ecosystems are not well understood, although a degree of understanding exists over their fate; it is primarily concerns over transfer along aquatic and terrestrial food chains that are of current economic and public-health significance. Microbial biotransformations of metallic radionuclides are of great importance in the biosphere and several have additional applications for bioremediation. Reactions mediated by microorganisms include solubilization from organic and inorganic complexes, compounds and minerals by the production of acids or chelating agents. The aim of our study consisted in elaboration of the new biotechnological method for environmental pollution risk reducing.

Methods: Screening the importance of soil micro-organisms on radionuclides mobility have been performed.

Results and conclusions: The influence of microorganisms on the environmental fate of radionuclides was elucidated, using some nonpathogenic strains of Penicillium sp. and Mucor sp., active producers of extra cellular pectolitic enzymes. Higher degree of radio nuclides insoluble compounds solubilizition, especially cobalt compounds, was observed under the influence of investigated strains in vitro (Invention nr. 3657 MD). Such mechanisms are important components of radionuclide biogeochemical cycles and should be considered in any monitoring analyses of environmental radionuclide contamination.

Establishment of research network for natural radiation exposure studies in Asia

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Abstract

A new project entitled gConstruction of natural radiation exposure study network h has been recently adopted in the Program of Promotion of International Joint Research under the Special Coordination Funds for Promoting Science and Technology operated by the Ministry of Education, Culture, Sports, Science and Technology of Japan. Eight institutions are involved in this project and the project will continue until March, 2012. The aims of the project are to assess the dose for natural radiation exposures using state-of-the-art measurement techniques in four Asian countries (China, India, Korea and Thailand) and their outcomes will be distributed worldwide. Conventional measurement techniques will be improved and be optimized. More scientific data and results will be obtained throughout this project. In particular, the following advanced technologies for inhalation exposures will be introduced:

1. Discriminative measurements of radon (^{222}Rn) and thoron (^{220}Rn) gases,
2. Evaluation of thoron decay products concentration,
3. Simple but effective particle size distribution measurements.

The measurement of the natural radiation background in a salt mine

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Abstract

The Unirea Salt Mine is situated at a depth of 208 meters beneath the surface of the earth that corresponds to a water-equivalent thickness of about 560 meters. It has a remarkable stability of microenvironment, characterized by a constant temperature over the years of 12.5 ±0.5 ºC and a relative humidity of 60-65 %.

In order to set up a calibration laboratory in this mine, for the measuring systems dedicated to the dosimetric measurements and environment, the natural radiation background had to be measured.

The measurements were made using two different dosimetric systems: An electronic type, consisting from two ratemeters: an Eberline FH 40 GL 10 and an AUTOMESS 6150 dose ratemeter with external scintillation probes; A TLD dosimetric system SD-TL type, which consists of the TL laboratory reader-analyze, 770 A type and thermoluminiscent detectors by LiF:Mg, Cu, P. The results obtained with the two systems are in good agreement.

Introduction

The measurement of the absorbed dose or of the integral ambient dose equivalent, and their rates, at the level of natural radiation background is a basic aspect of the dosimetry of the environment. In this case, an important problem to be solved in the calibration of the dosimetric measuring systems, is the low level of the measuring range, which has to include the values corresponding to the natural radiation background (50÷100) nSv·h-1.

In order to reach the adequate conditions to perform such calibrations, an ultralow level background laboratory is necessary. So, the "Horia Hulubei National Institute R&D for Physics and Nuclear Engineering" (IFIN-HH) decided to develop an ultralow level radiation background in the former Unirea salt mine, located in the town Slanic, at a distance of about 100 km from Bucharest.

The Unirea Salt Mine is located at a depth of 208 meters beneath the surface of the earth; taking into account the geological structure of the soil in the area of the mine, is was found that this depth corresponds to a water equivalent thickness of about 560 m, Margineanu et al. (Margineanu, 2008).

The salt mine is characterized also by a remarkable stability of the microclimate: a constant temperature, measured over several years, of 12.5± 0.50C and constant relative humidity, measured in the same conditions, of 60-65%. Of course, in order to set up a calibration laboratory in this mine, the first step was to perform accurate measurements of the absorbed dose and of the integral ambient dose equivalent and their rates, corresponding to the natural background in a hall where the new laboratory is located.

Material and methods

In order to perform accurate measurements of integral ambient dose equivalent $H^*_{(10)}$ expressed in mSv and of the rate $H^*_{(10)}$, expressed in nSv h-1, in the area were the laboratory was located, two different dosimetric system were chosen:
Two electronic dose ratemeters, with external scintillation probes;
One TLD dosimetric system, SD-TL type.

a) As for electronic dose ratemeter, we used many instruments and finally we have selected two of them:
   – an Eberline FH 40 GL 10 dose ratemeter, with the external probe FHZ 5023E. This probe has a large scintillation NaI detector of 3x3 inch. We calibrated it in terms of absorbed dose per impulse, with a value of the calibration factor of: F = (0.17±0.03) nGy·imp-1. The energy dependence of the response instrument in the ratemeter mode, is of maxim ±30% for the whole measured energy range. The conversion of units from absorbed dose, Gy, to ambient equivalent dose, Sv, was done by considering a conversion factor f = 1 Sv/Gy. It was used only for preliminary measurements.
   – a thermo Automess 6150, with the external probe 6150 AD-6/H having also a scintillation detector.

This instrument, metrologically certified by its producer, has the following characteristics, according the technical specification
   – measuring range (digital): 0.01µSv·h-1÷9.99 mSv·h-1, with a declared uncertainty of 10%;
   – energy range: 60 keV-1.3MeV; energy dependence of the response in this range: ±30%.

The reported results were obtained with this instrument.

The TLD dosimetric system, SD-TL type, consists of TL reader-analyzer, model RA-94 and the thermoluminiscent dosemeters with commercial thermoluminescent detectors type LiF:Mg,Cu,P, known as GR 200A. The dosemeter for environmental monitoring, designed and made in our laboratory, consists of a plastic cylindrical container with the external dimensions: Φ=50 mm and h = 9 mm. Each dosemeter is provided with three detectors. A detailed description of the system and of its dosimetrical characteristics is given in the paper by Stochioiu et al. (Stochioiu, 2009).

Measurement conditions

The electronic dose ratemeters measuring directly the dose equivalent rate $\dot{H}^*_{(10)}$, were used in several pre established significant locations inside the laboratory. The TLD system is calibrated in units of dose equivalent, $H^*_{(10)}$. The dose equivalent rate is calculated as the ratio of dose equivalent, mSv, and exposure time, h. In our case, two series of exposure, for periods of 39 d and 93 d were performed. In the case of the TL dosemeters, when processing the measured values, we took into account their irradiation during the transportation from Bucharest to the salt mine, and an appropriate correction was done. The reported values are the means obtained from the two exposition intervals.

For the purpose of the comparison of measurements and the establishment of consistence of results, the locations for the two types of measurements, electronic ratemeters and TLD system, were the same.

Results

Table 1 represents the measured values of the ambient equivalent dose rate [nSv·h-1], obtained with the two systems, respectively the electronic AUTOMESS 6150 and the TLD system SD-TL, in four representative locations and their comparison. These points were selected as the most representative from a large number of measurements areas of the mine.

Table 1. Mean ambient equivalent dose rate values , measured with the two systems and their comparison

Location	Mean ambient equivalent dose rate [nSv·h-1], AUTOMESS 6150 system	Mean ambient equivalent dose rate[nSv·h-1], TLD, SD-TL system	Difference, %
1	1.17±0.42	1.49±0.41	27
2	2.22±0.42	2.15±0.82	3
3	1.87±0.57	1.55±0.30	17
4	2.56±0.55	2.14±0.32	16

The reported uncertainties were calculated from the instruments' calibration factor uncertainties, respectively 10% and 7.5% and additional uncertainties: nonlinearity outside the certified interval, statistical uncertainty, etc.

Discussion

Such as it can be seen from the Table 1, two conclusions can be drawn:

1. The levels of background inside the underground laboratory are about 25 - 50 times less than the usual ground level, (50 -100) nSv h-1. The explanation is that the cosmic radiation is strongly attenuated by the rocks. The registered values are due to the residual content of radioactive elements existent in rocks, whose influence is also reduced by the existence of thick pure salt walls surrounding the laboratory. The influence of salt purity over the influence of 226Ra - 222Rn decay chain and of 40K was reported by Cristache and al. (Cristache, 2009), by using classical chemical analyses methods, Nenitescu (Nenitescu, 1982) and Neutron Activation Analysis.
2. The results obtained with two independent and different as operating principle dosimetric systems are in agreement within their measurement uncertainties.

Conclusions

– The low level of background in an underground laboratory was systematically studied, in order to characterize the environmental dose of the new laboratory.
– Two independent systems, an electronic and a TLD one were used. Their operating principles are different, but the results are in good agreement, what demonstrates the consistence of measurements.

Acknowledgements

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Personal monitoring of aircrew exposed to cosmic radiation

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Abstract

Everybody who flies frequently is exposed to elevated levels of cosmic radiation of galactic and solar origin and to secondary radiation produced in atmosphere. The annual average radiation dose to aircrew may become similar or even large than that of other occupationally exposed groups. The assessment of effective dose for aircrew members was performed by using programs EPCARD*. Altogether about 180 pilots were monitored during 2007 year and about 500 air crew members were monitored during 2008 years. In 2007 annual average effective dose (E) was 2.5 mSv and maximum personal dose was 4.0 mSv compared to next year annual average 2.3 mSv and maximum 3.7 mSv. Mobile Dosimeter Unit (MDU) Liulin Si spektrodosimeter and Exploranium GR 135 have been used for experimental measuring of levels of cosmic radiation on the airplane board.

Additionally to personal cosmic radiation monitoring, small human trial was conducted to monitor life style, nutrition and various clinical, biochemical, genotoxicity, immune and molecular markers from blood samples of exposed as well as control subjects and to investigate the possible effects of radiation exposure on human health.

Introduction

The recommendation of International Commission on Radiological Protection (ICRP) No. 60 from 1991 year (ICRP 60) included the exposure of the aircrew from the cosmic radiation among the occupational exposure. Council Directive 96/29/Euratom (CD96/29), the Basic Safety Standards (BSS) has been published on May 13, 1996 laying down basic safety standards for the health of workers and members of the general public against the dangers arising from ionizing radiation(BSS 1996). Article 42 of the directive on the protection of aircrews from EU member states have been transposed into the national legislation directives. The directive has been also incorporated into the Slovak regulation since 2000. Ministry of Health of Slovak

Republic funded the project “Radiation load of aircrew and biomonitoring of health risk of combined exposure to radiation and stress factor” which was conducted from 2006 to 2008. The main aims of the project were to introduce radiation monitoring (periodical monitoring system build on experimentally measurements as well as theoretically confirmed effective dose assessment) and to identify the effect of exposure on biological markers of genetic stability, DNA repair, oxidative stress and on immune markers. This contribution presents results of radiation monitoring.

Material and methods

The EPCARD* calculation program was used for evaluation of aviation personnel radiation load exposure. We created software for computing of data from large number of monitored members of aircrew. The interactive database was created for data on occupational exposure the aviation crew members for every month, as well as for annual exposure. This database enables data elaboration and analysis, the statistical evaluation, and the assessment of other radiation protection quantities.

Onboard aircraft exposure measurements were provided by Mobile Dosimeter Unit (MDU) – Liulin Si diode based spectrodosimeter and gamma spectrometer Exploranium GR 135. Measurements have been performed on an B737 aircraft Sky Europe Airlines at an altitude 12 000 m.

Results

Estimate of personnel radiation load

In 2007 we observed the radiation load of 181 pilots. The results show that the average annual effective dose is 2.5 mSv, the maximum annual effective dose is 4.0 mSv and the collective dose is 460.6 mSv.

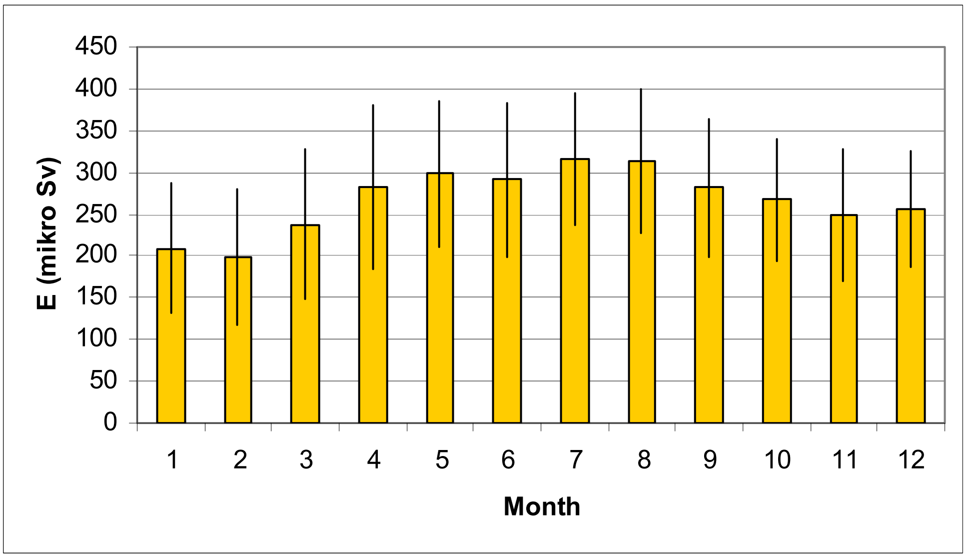

Fig. 1. Average monthly effective dose – pilots 2007.

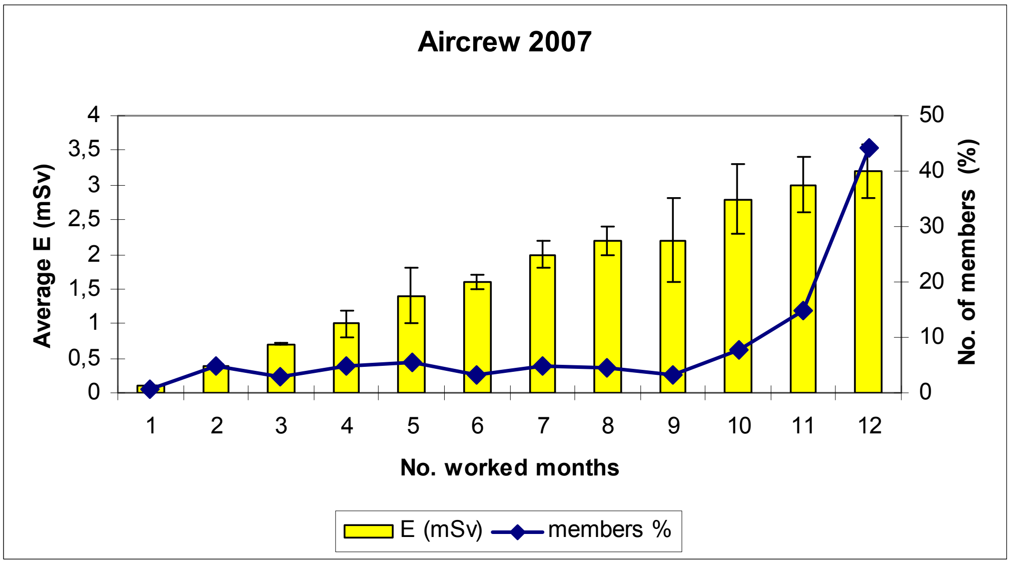

Fig. 2. The average effective dose by the number of months worked during the year 2007.

In 2008, we monitored the radiation load of 183 pilots. The results show that the average annual effective dose is 2.5 mSv, the maximum annual effective dose is 3.7 mSv and the collective dose is 463.6 mSv.

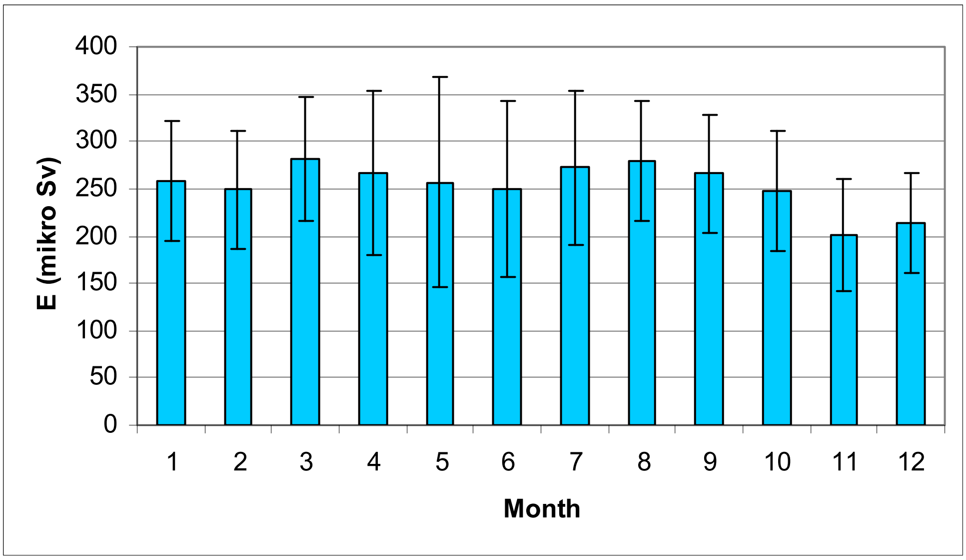

Fig. 3. Average monthly effective dose – pilots 2008.

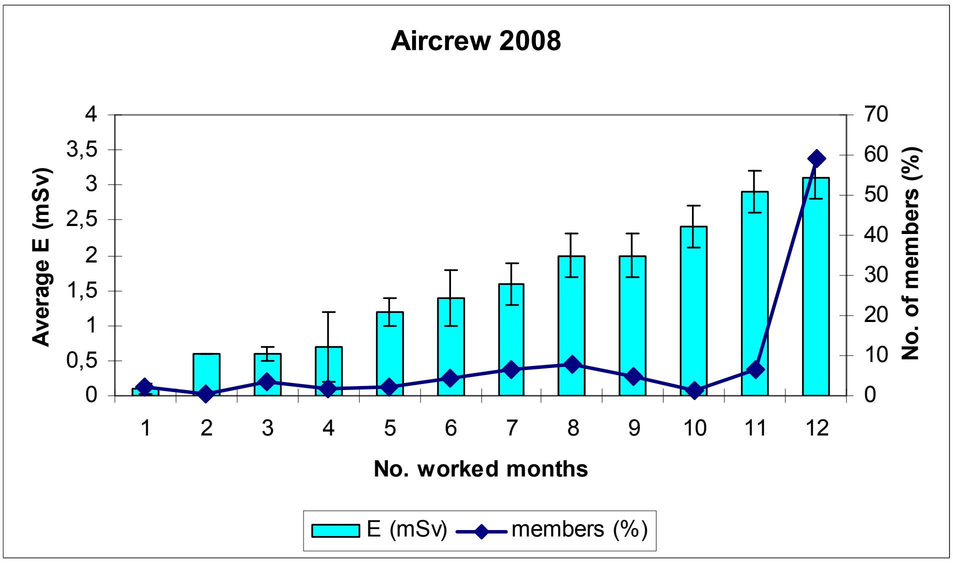

Fig. 4. The average effective dose by the number of months worked during the year 2007.

In 2008 we watched the radiation load 316 stewards too. The results show that the average annual effective dose is 2.2 mSv, the maximum annual effective dose 3.7 mSv and the collective dose is 693.1 mSv.

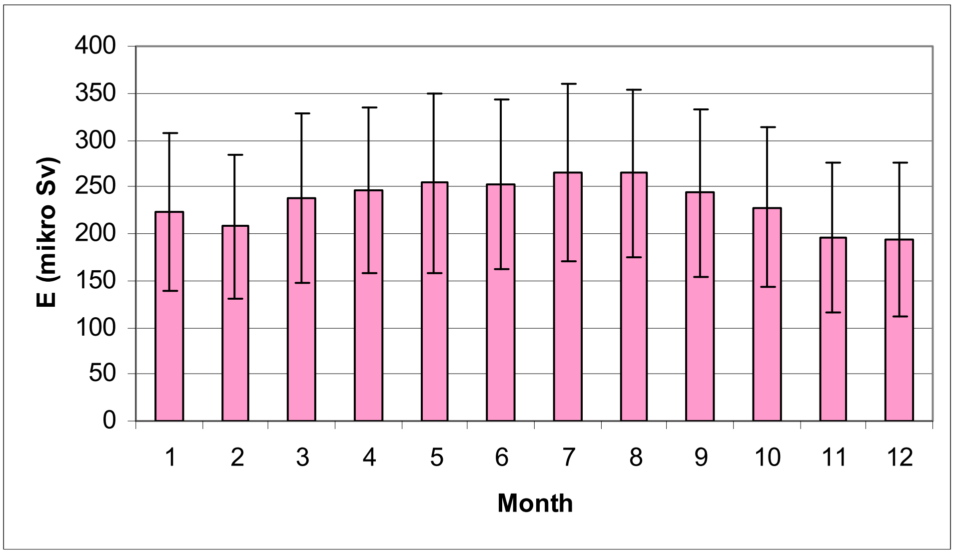

Fig. 5. Average monthly effective dose – stewards 2008.

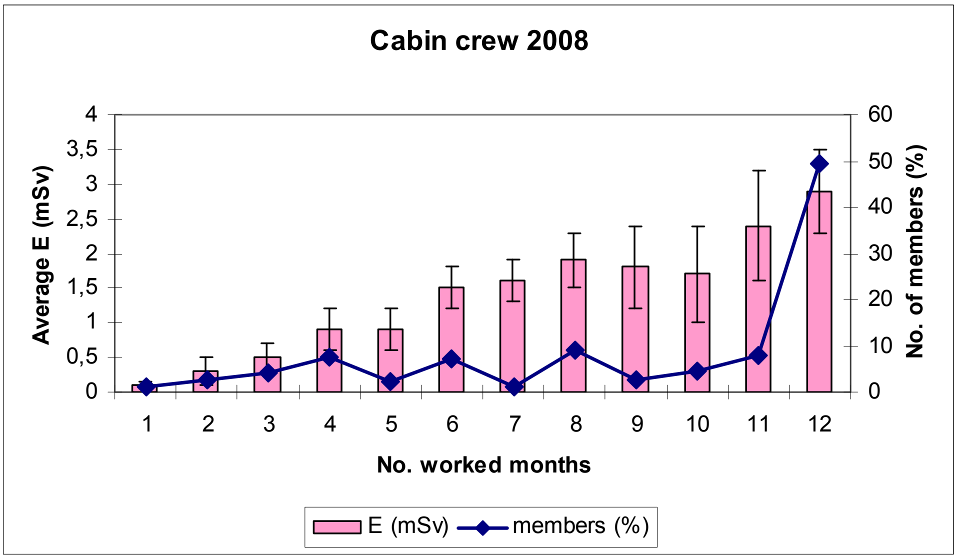

Fig. 6. The average effective dose by the number of months worked during the year 2008.

Table 1. Radiation load of aircrew members.

Year	Air crew Members	No members	Average E ± STD [mSv]	Maximum E [mSv]	Collective E [mSv]
2007	Pilots	181	2.5 ± 1.0	4.0	460.6
2008	Pilots	183	2.5 ± 0.9	3.7	463.6
2008	Stewards	316	2.2 ± 1.0	3.7	693.1

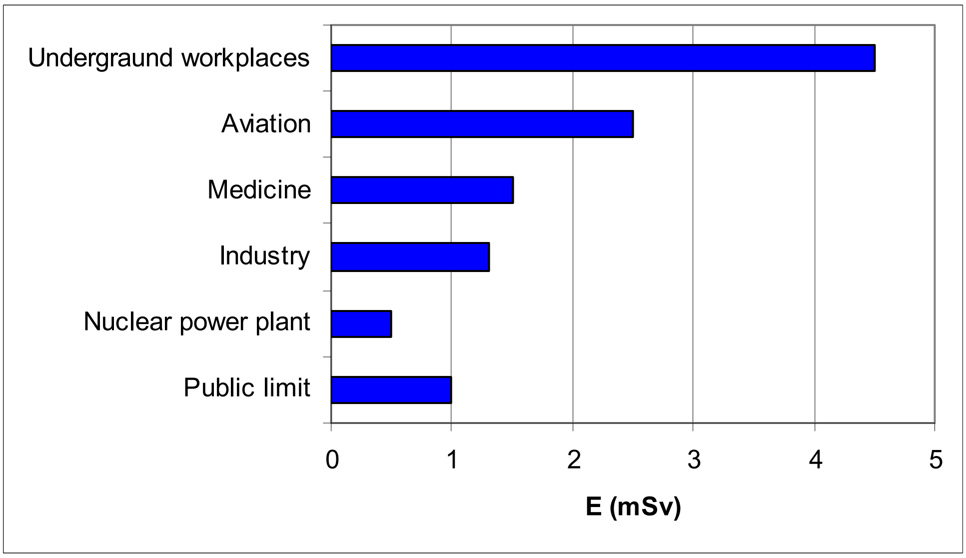

Fig. 7. Comparison of annual effective dose from different occupational exposure in Slovakia (Böhm 2007).

Experimental results

Table 2. Experimental data of ambient dose equivalent (May 2008).

Fly	Departure	Arrival	Cruise	H*1 [nSv]	H*1 [nSv]
BTS – FCO	12:52 13.05	14:12 13.05	1 h 20 min	200	1 747
BTS – LTN	17:40 13.05	19:45 13.05	2 h 05 min	410	4 208
BTS – MAN	12:37 14.05	14:50 14.05	2 h 13 min	500	4 382
BTS – KSC	09:03 15.05	09:33 15.05	0 h 30 min	25	283
KSC - DUB	10:20 15.05	13:20 15.05	3 h 00 min	780	7 025
BTS - BHX	12:00 16.05	14:10 16.05	2 h 10 min	470	3 824
BTS – HRG	20:10 17.05	23:45 17.05	3 h 35 min	640	5 231
HRG - BTS	01:10 18.05	05:15 18.05	4 h 05 min	730	5 869
BTS - ORY	06:38 26.05	08:50 26.05	2 h 18 min	460	3 460
BTS - ORK	09:45 27.05	12:22 27.05	2 h 37 min	710	6 123
ORK - BTS	13:30 27.05	16:00 27.05	2 h 30 min	690	5 771

BTS, FCO, etc - IATA code of airport (BTS – Bratislava, FCO – Rome, LTN – London, MAN - Manchester, KSC – Košice, DUB – Dublin, BHX – Birmingham, HRG – Hurgada, ORY – Paris, VIE – Vienna, ORK – Cork
Departure GMT +2
*H*1 – ambient dose equivalent gama rays (Exploranium 135)*
*H*2 - ambient dose equivalent neutrons and heavy charged particles (Liulin)*

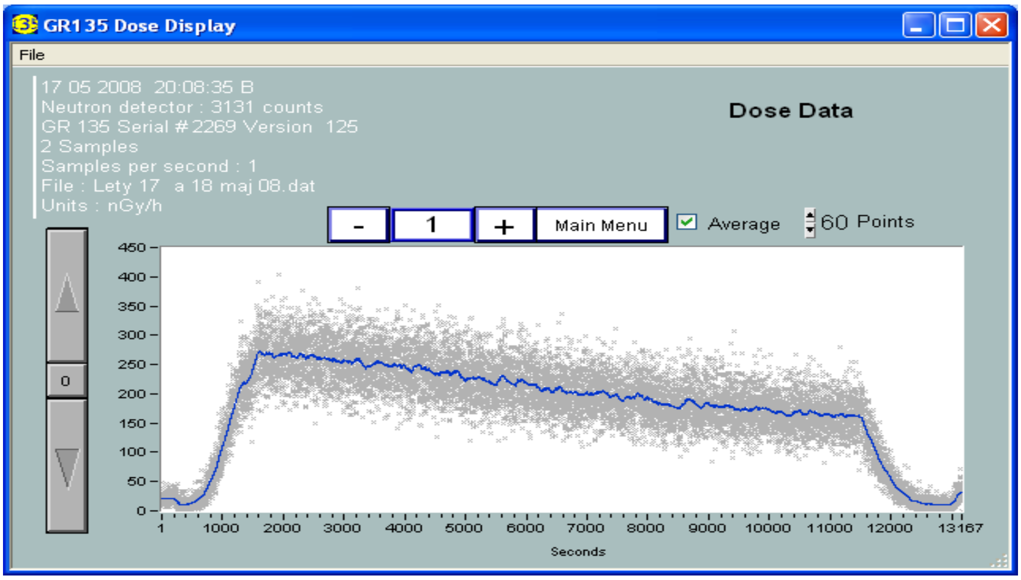

Fig. 8. Data from Exploranium 135, fly Bratislava – Hurgada.

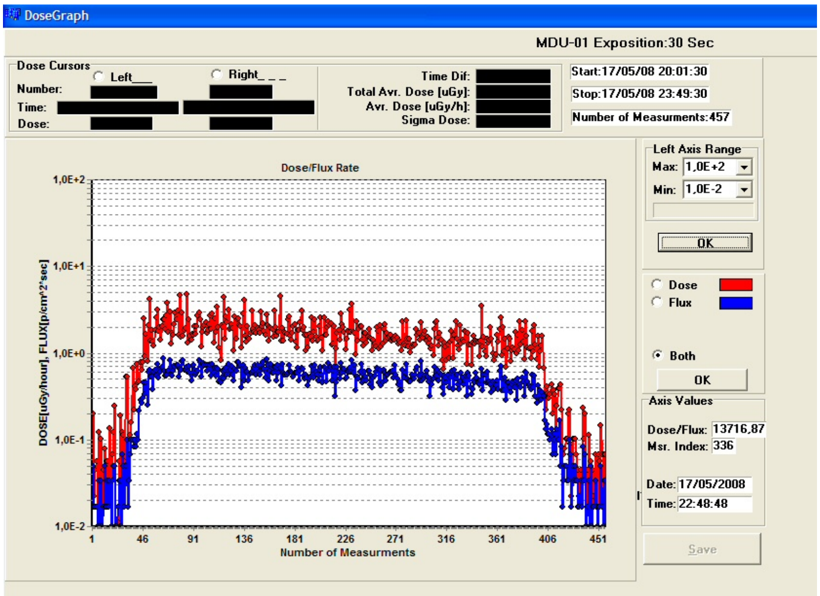

Fig. 9. Data fromMDU Liulin, fly Bratislava – Hurgada.

Discussion

Annual effective dose to aircrew members who had been working more than 3 months during the year always exceeds 1 mSv. In accordance with Article 42 Directive 96/29 Euratom - Each Member State shall make arrangements for undertakings operating aircraft to take account of exposure to cosmic radiation of air crew who are liable to be subject to exposure to more than 1 mSv per year. Experimental measurements and estimates of personal radiation load from exposure to cosmic radiation were carried out during the period of minimum solar activity. We assume that during the period of increased solar activity may be higher annual effective dose. The graphic outputs from used devices show that the measured ambient dose equivalent and the intensity of cosmic radiation are dependent not only on traffic levels but also on the geographical location. Cosmic ray intensity increases towards the magnetic poles of the Earth and decreases toward the equator in the same altitudes. This is due to the different intensity geomagnetic field at the Earth's equator and poles. Magnetic field of Earth protects us from the increased intensity of cosmic radiation. We use this fact to a large extent possible to optimize the radiation load of personnel planning for deployment of staff to the route. Reducing of radiation load of aircrew members cannot be achieved by shielding in aircraft construction or using of protective shielding devices. There is the only way optimization - to take into account the assessed exposure when organizing working schedules with a view to reducing the doses of highly exposed aircrew. During the flight planning it should be take into account that a dose rate of cosmic radiation grows with the increase of latitude and flight level.

Conclusions

The project allowed for the first time in Slovakia to carry out monitoring of personnel radiation load from cosmic radiation. The results obtained confirm the eligibility of the start up of radiation monitoring Slovak air carriers and justify the need for its continuation. The obtained results show that annual effective dose to members of air crews is highly significantly higher in comparison with radiation dose of workers exposed in other workplaces.

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Progress with the revision of the Euratom Basic Safety Standards and consolidation with other Community legislation

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Abstract

The revision of the Euratom Basic Safety Standards (Directive 96/29/Euratom) was undertaken to allow for the new ICRP Recommendations (Publication 103) as well as to consolidate all radiation protection legislation in a single BSS Directive. In line with ICRP the new Directive will allow for the three exposure situations: planned, existing and emergency. This required significant restructuring, together with the consolidation or recast. The new Directive will fully integrate all natural radiation sources, introducing more binding requirements for NORM industries, building materials, radon in dwellings and workplaces. The protection of the environment is now within the scope of the Directive. The Directive develops the concept of a graded approach to regulatory control, so that it is commensurate to the risk and to the effectiveness of such controls. In this context also the concepts of exemption and clearance have been worked out in more detail and new consideration is being given to the harmonisation of exemption and clearance levels. Further issues that emerged from the revision process are the definition and use of constraints and reference levels, as introduced by ICRP, and the principle of justification, in particular with regard to the deliberate exposure of people for reasons other than medical (e.g., for security screening). The revision and recast process was completed in February 2010. Harmonisation with the International Standards (IAEA and co-sponsors) is vigorously pursued in view of co-sponsorship by Euratom.

1. Consolidation of the Community legislation

The Community *acquis* in radiation protection, derived from Title II, Chapter 3 of the Euratom Treaty, constitutes a comprehensive and evolutionary legislative framework, with the Basic Safety Standards Directive (EU, 1996) as principal piece of legislation, and supplemented by eight binding instruments (Directives, Regulations, Decisions in addition to non-binding Recommendations and Communications).

The Commission undertakes the simplification of its "*acquis*" of Community legislation by the codification of related acts (without modification, e.g., amendments or complementary legislation) or recasting these if necessary (e.g., allowing for different definitions). Such recast will be undertaken for the five main Directives (all except shipment of radioactive waste (EU, 2006)). The very recent Council Directive

2009/71/Euratom on nuclear safety of nuclear installations was not considered for a recast at the present stage. In addition to the Basic Safety Standards, this recast thus concerns the following Directives:

- Medical applications of ionising radiation: Directive 97/43/Euratom (EU, 1997)
- Information in case of radiological emergency: Directive 89/618/Euratom (EU, 1989)
- Protection of "outside workers": Directive 90/641/Euratom (EU, 1990)
- Control of high-activity sealed radioactive sources and orphan sources ("HASS Directive"): Directive 2003/122/Euratom (EU, 2003).

The Basic Safety Standards Directive had introduced in 1996 some new features. A specific Title VII was introduced for the regulatory control of "work activities" involving natural radiation sources. With the exception of aircrew exposure, the Directive left the responsibility for the identification of work activities (NORM industries and workplaces with high radon concentrations) with national authorities. This flexibility was needed in order to achieve consensus on the inclusion of these new features at a time when there was little experience with such matters. The experience gathered since 1996, with transposition in national legislation (due by May 2000) and with operational implementation, demonstrated a need for enhanced harmonisation. Thus a revision of the Basic Safety Standards was undertaken amongst others to strengthen and broaden the requirements on natural radiation sources.

2. Revision of the Basic Safety Standards

The Community's Group of Experts established under Article 31 of the Euratom Treaty had established a four year work programme for the revision of the Basic Safety Standards. Initially it followed a topical approach. Working Parties took on board the redrafting of requirements on natural radiation sources, on exemption and clearance, and on a graded approach to regulatory control. Specific attention was given to the requirements on Education and training as well as to the definition of the responsibilities and qualifications of the Radiation Protection Expert, the Medical Physics Experts, and Radiation Protection Officer.

2.1. Exposure Situations

The revision of the Euratom Basic Safety Standards took account of the ICRP recommendations in Publication 103 (ICRP, 2008). While these do not necessarily require major changes in regulatory requirements, the Commission undertook to structure the requirements along the concepts of planned, existing and emergency exposure situations.

The distinction between the three exposure situations proved very helpful in structuring the Standards. However, very precise definitions are needed in a regulatory context, rather than the somewhat loose descriptive formulations in ICRP Publication 103.

For instance, ICRP defines a planned exposure situation merely as one involving the planned operation of sources. Existing exposure situations relate to "sources that already exists when a decision on control has to be taken". We strongly believe that when an activity significantly affects or alters an exposure situation caused by existing sources, such as NORM (Naturally Occurring Radioactive Material) or cosmic radiation, this is a planned exposure situation. Hence NORM industries and the exposure of aircrew are planned situations and the activities can be labelled as practices.

On the other hand, NORM materials with levels of activity concentration that are common in the earth's crust should be exempted from the requirements for practices. The boundary is established through identification of those types of NORM industries that should *a priori* be managed as a practice. The concept of a threshold in activity concentration, for the application of requirements for practices, matches the concept of exemption from such requirements. The new Directive has introduced the values of 1 Bq/g for U-238/Th-232 and 10 Bq/g for K-40 taken from IAEA RS-G-1.7 (IAEA, 2004), together with a graded approach to regulatory control using dose thresholds of 1 mSv and 6 mSv per year.

Truly existing exposure situations are those for which the exposure results from where you are, rather than what you do. Radon ingress in a dwelling, from soil, is not related to any activity, so it yields in general an existing exposure situation. Any exposure at work, not necessarily resulting from the work, is however the responsibility of the employer. High levels of radon in the workplace thus should be subject to a reference level, to a threshold for the management as a planned exposure situation (which may be the same as the reference level) and to the dose limit for workers.

There are other boundaries where pragmatic choices need to be made. The production or import of commodities clearly is an activity. However, if the radioactive substances arise from an existing exposure situation, then it is more convenient to manage such commodities in the same context. Hence, building materials and contaminated food are managed under the heading, “existing exposure situations”.

An emergency exposure situation eventually leads to the existing situation of living in a contaminated area. The delineation in this case requires a management decision.

In earlier drafts of the new Directive the provisions for emergency and existing exposure situations had been laid down in specific titles. Later if was found preferable to have, for example, all aspects of occupational exposure, including emergency workers and the follow-up to accidental exposure of workers, in a single title. Hence there is a title on “justification and regulatory control of planned exposure situations”, but the chapters on the protection of workers, patients and members of the public are no longer part of an overall title on “planned exposure situations” (as is the case with the current draft of the international standards.) It was proposed to have a clear 3 x 3 matrix structure with the categories of exposure (occupational, public and medical) on the one hand, and the three exposure situations on the other hand. (see Table 1).

Table 1. New matrix structure of the Euratom BSS.

OCCUPATIONAL EXPOSURE	PUBLIC EXPOSURE	MEDICAL EXPOSURE
Planned exposure situations	Planned exposure situations	Planned exposure situations
Emergency exposure situations	Emergency exposure situations	Emergency exposure situations
Existing exposure situations	Existing exposure situations	

2.2. Emergency exposure situations

The new management scheme for emergency exposure situations builds on recent guidance of ICRP (ICRP, 2009). The old approach of an emergency plan with different *intervention levels* is replaced by a more comprehensive system:

– threat analysis;
– overall emergency management system;
– emergency response plans for identified threats;
– pre-planned strategies for the management of each postulated event.

The key difference is that each strategy should aim at keeping doses below the *reference level*, optimising the available preventive and protective actions rather than justifying each action. New Annexes list the elements to be included in the management system and in the emergency response plan.

2.3. Natural radiation sources

The Working Party of the Article 31 Group of Experts on natural radiation sources undertook in the first place the harmonisation of the identification and regulatory control of NORM-industries. The Working Party agreed on a “positive list” of types of industries that will be subject to controls in all Member States. It will be the task of the national authorities to inform the concerned industries and to make sure that they understand the radiation protection issue and take, if necessary, appropriate measures to reduce exposures within the overall Health and Safety policy of the undertaking.

The industries (those listed and such other industries as identified at national level) will be requested to investigate activity concentration levels at any point of their process. On the basis of an assessment of occupational exposures in identified industries a graded approach to regulatory control will be applied. Where doses are all below 1 mSv per year, the practice is exempted. Where doses are in the range of 1 to 6 mSv per year the only requirement is to review whether optimisation calls for further reduction of exposures, and whether the exposures remain broadly the same over many years. Since there is in general no risk of accidental exposure, there is no need for individual dosimetry or medical surveillance. In the exceptional case that doses exceed 6 mSv per year the full set of requirements for classified workers will apply.

In the same way as for practices involving artificial radionuclides, the concepts of exemption and clearance should merge (even more so, since the output of one NORM industry often is the input to another). The values proposed for this purpose in earlier Community guidance (Radiation Protection 122 part II (EC, 2000)) were 0.5 Bq/g for (U238/Th 232 and 5 Bq/g for K-40, on the basis of an exemption criterion of 300 µSv per year for natural radiation sources. The values endorsed for the sake of international harmonisation (RS-G-1.7), twice as high as in (EC, 2000), are not always suitable for clearance of residues from NORM industries. The Euratom Directive therefore establishes unambiguously that the values in RS-G-1.7 for naturally occurring radionuclides apply neither to the recycling of residues into building materials nor to situations where there is a specific risk such as groundwater contamination.

While NORM industries and the exposure of aircrew are managed as planned exposure situations, a management system for existing exposure situations applies to:

– building materials;
– radon in dwellings and public buildings (workplaces are considered either as an existing or as a planned exposure situation).

An indicative list of types of building materials considered for control in view of the emitted gamma radiation has been included in an Annex. On the basis of earlier guidance (EC, 1999), requirements for the placing on the market and use of building materials have been incorporated in the Basic Safety Standards. The activity concentration index I :

$$I = C_{Ra226}/300\ \mathrm{Bq/kg} + C_{Th232}/200\ \mathrm{Bq/kg} + C_{K40}/3000\ \mathrm{Bq/kg}$$

may be translated into two classes of building materials depending on the use of the material and on whether a common reference level of 1 mSv per year (above background) would be exceeded. This is the only reference level that will be imposed through the Directive. Indeed, leaving a free choice to Member States would lead to very complex requirements to allow for free trade within the EU.

Currently, radon in dwellings is excluded from the scope of Directive 96/29/Euratom and is covered by a Commission Recommendation (EC, 1990). Recent epidemiological findings from residential studies demonstrate a lung cancer risk from indoor radon exposure at levels of the order of 100 Bq m^{-3}. ICRP is currently re-considering its earlier guidance on the dose conversion factors relating to concentrations of radon gas and its progeny in the decay chain. The ICRP Main Commission has issued a statement in November 2009 now proposing a maximum value for the reference level in dwellings of 300 Bq m^{-3}, in line also with the WHO handbook on indoor radon (WHO, 2009). The new Directive has incorporated this value for existing dwellings. The lower value for the reference level proposed by WHO, 100 Bq m^{-3}, is suitable as a long-term goal but a maximum value of 200 Bq m^{-3} for new dwellings has been maintained in the Directive.

A maximum value for the reference level for radon in workplaces has been set at 1000 Bq m^{-3}, in line with the ICRP Statement and with the current draft of the international standards. With the possible doubling of the dose conversion factor, a radon concentration of 1000 Bq m^{-3} would correspond to around 10 mSv per year, which is a high threshold for managing radon at work as planned occupational exposure and well above 6 mSv per year, used in the definition of Category A workers. It is expected that most Member States will set or maintain a reference level much lower than 1000 Bq m^{-3}.

Member States will be required to establish a national action plan which will cover radon in dwellings and in workplaces. The action plan will offer transparent information on the scope and objectives pursued at national or regional level, define the rationale for the conduct of surveys and for the delineation of radon-prone areas or other means of identification of affected buildings, and establish reference levels and building codes. An Annexe to the Directive gives an indicative list of items to be covered in the national action plans for radon.

2.4. Graded approach

The current system of regulatory control is a two-tier system: reporting of practices above exemption levels, and prior authorisation for certain categories of practices. IAEA (IAEA, 1996) had introduced a three-tier system: notification, registration and licensing. The Directive identifies which type of practices will be subject to each pillar, which general conditions need to be fulfilled and what is the content of requirements laid down upon registration or as part of a specific operating licence.

The current system for exemption of apparatus and of consumer goods relies very much on the concept of “type approval”. This concept was not worked out further and there is a lack of harmonisation of conditions for type approval and corresponding decisions in the EU. A system of mutual recognition (or at least allowance for) type approvals granted in other Member States has now been introduced.

Directive 96/29 had introduced exemption values in terms of activity (Bq) and activity concentration (Bq/g). In addition, the reuse or recycling of materials with negligible levels of contamination, especially arising from dismantling, could be authorised so that the materials would be released from regulatory requirements, subject to compliance with clearance levels. The clearance levels should be established in such a way that individual doses would be below about 10 μSv (and collective doses below 1 man Sv), taking Community guidance into account. Such guidance has been adopted by the Group of Experts for specific materials such as metals (scenarios for steel, copper and aluminium), buildings and building rubble, and default values for any type of material (EC, 2000).

Meanwhile the IAEA adopted similar guidance in RS-G-1.7 (IAEA, 2004), on the basis of scenarios to a large extent inspired by those underlying RP 122 part I. The IAEA levels were not specifically developed for the purpose of clearance, but it was suggested to use them for this purpose. For the sake of international harmonisation it was decided to introduce in the Directive the RS-G-1.7 values rather than those in RP-122. A study (EC, 2009) has investigated whether the differences between the two approaches and series of values has any significance in practical terms. In general the RS-G-1.7 values are equal to or higher than those in RP 122, but the differences can rather well be explained through the scenarios and assumptions. Nevertheless, several Experts consider that the Community guidance has a better scientific basis, and they are concerned with the increase of some of the values for artificial radionuclides.

The same concentration values now apply by default both to the concepts of exemption and clearance. It has been investigated whether lowering the exemption values will affect any consumer goods placed on the market. While the EC study (EC, 2009) concluded that there would be indeed no adverse practical consequences, it also highlighted some Member States’ reluctance to abandon the old numbers already incorporated in national law. In addition IAEA's transport standards committee (TRANSC) advocated keeping the existing values.

As part of the graded approach exemption from any requirements is built in at all levels of control. Specific exemption and specific exemption or clearance levels are a powerful tool in addition to the criteria for general exemption of practices from the scope of the requirements. Whenever the exempt activity concentration values laid down in Directive 96/29/Euratom, on the basis of Radiation Protection 65, would be preserved in national legislation, these values should be used only for moderate amounts of material, as defined in legislation or as specified by the competent authority.

2.5. System of protection

The overall "system of protection" mirrors the wording used in ICRP Publication 103. The main elements of the system of protection: justification of practices, optimisation of protection and limitation of individual doses are not essentially different from those in the current BSS. More weight is given to the principle of optimisation, subject to constraints and reference levels. The bands of constraints/reference levels proposed by ICRP (0-1 mSv, 1-20 mSv, 20-100 mSv) have been introduced explicitly, including the societal criteria that ICRP listed for each band (Table 5 of Publication 103).

The concepts of justification and authorisation in planned exposure situations were brought together in one title in view of the fact that they are the two main pillars of regulatory control. Also, the more elaborate requirements for the type approval of apparatus or consumer goods relate both to justification and to authorisation.

The concept of justification is described very much in the same terms as before. The so-called "medico-legal" exposures introduced in the Medical Directive (EU, 1997) have now been clearly identified as "non-medical imaging exposures" (deliberate exposure of individuals for other than medical purposes), and have been put under appropriate regulatory control. The need for justification of such practices, in three stages as for medical exposures, and for establishing associated conditions, has been worked out, including differentiation between procedures implemented by medical staff using medical equipment and procedures implemented by non-medical staff using non-medical equipments (e.g. security screening). While, in general, the annual dose limit and corresponding constraints for public exposure should apply, exceptions should be allowed for some specific non-medical exposure procedures carried out in a medical environment (e.g. drug search within the body).

The current dose limits for practices are kept, but the annual dose limit for occupational exposure will be simply 20 mSv per year. There should be no need for averaging over 5 years, except in special circumstances specified in national legislation.

On grounds of the precautionary principle, the Directive applies the optimisation principle also, where appropriate, to keep organ doses as low as reasonably achievable. The Scientific Seminar in 2008 (EC, 2008) on emerging evidence for radiation induced circulatory diseases indicated that epidemiological evidence is accumulating on an increased risk in circulatory diseases for cumulative doses higher than 0.5 Gy low-LET radiation.

In view of the conclusion of the Scientific Seminar in 2006 (EC, 2006) on the issue of radiation induced cataract and the further review of scientific literature performed by the Group of Experts, the dose limits to the lens of the eye should be reduced. ICRP will soon issue guidance on this matter and the Commission will take this guidance into account. Furthermore, the Directive requires the set-up of adequate systems for individual monitoring of (significant) doses to the lens of the eye.

3. Categories of exposure

3.1. Occupational exposure

A specific title covers all types of occupational exposure: of workers, apprentices and students, emergency workers, workers in identified NORM industries, aircrew and space-crew.

The graded approach to arrangements for occupational exposure is made more explicit, with a threshold of 1 mSv per year. The categories A and B workers are preserved. For workers in NORM industries, as part of the graded approach and if doses are in the range 1 to 6 mSv, it is sufficient to keep the exposures under review.

Emergency workers are subject to a dose limit of 50 mSv or, for specific cases identified in national emergency plans, an appropriate reference level. In the current Directive provision was made also for “specially authorised exposures”. This provision will now be applied to the exposure of space-crew.

3.2. Medical exposure

Medical exposure applies to the protection of patients, carers and comforters and volunteers in bio-medical research ("other individuals submitted to medical exposure"). There are very few changes to Directive 97/43, except the removal of “medico-legal” exposures and emphasis being given to the information to patients, to interventional procedures, diagnostic reference levels and dose indicating devices. A new feature is the introduction of accidental or “unintended” exposures.

3.3. Public exposure

There are little changes to the protection of members of the public. More precise requirements on the establishment of discharge authorisations have been introduced, as well as on the monitoring of discharges, with reference to a Commission Recommendation (EC, 2004).

3.4. Protection of the environment

The general objective of the Directive is the health protection of workers, members of the public and patients. The recast introduces complementary objectives on the control of sealed sources and on providing information to the public in the event of a radiological emergency. The health protection of the population and workers against the dangers of ionising radiation includes the protection of the human environment as a pathway from environmental sources to the exposure of man. In line with ICRP Publication 103 it is now felt that this should be complemented with specific consideration of the exposure of biota in the environment as a whole.

This extension of the scope of the Basic Safety Standards Directive will enable a better integration of the Euratom legislation with overall environmental legislation adopted under EC Treaty provisions, as well as the observance of international agreements, such as the OSPAR Convention on the protection of North-Atlantic waters, and meet the concerns of stakeholders.

While Chapter 3, “Health & Safety” of the Euratom Treaty only relates to the health protection of workers and members of the public, the policies for the protection of man and the environment should be coherent. For instance, environmental criteria as well as dose constraints should be considered for the authorisation of discharges of radioactive effluent.

Publication 108 of ICRP (ICRP, 2008) offers guidance on the definition of reference animals and plants, and the assessment of the impact of radiation on non-human species. The application of the principles of radiation protection on non-human species and ecosystems needs to be further developed however. The protection of the environment does not seem to warrant a high level of regulatory control, and the means for the demonstration of compliance should be proportionate to the expected relevance of the issue, in line with the graded approach. Also in view of the limited experience so far, the Commission envisages to leave enough time for transposition of these requirements in national law, pending the results of further research and international guidance of ICRP.

4. Prospects

The Group of Experts under Article 31 of the Euratom Treaty finalised the text of the new Directive in February 2010. The text of the Experts and their Opinion are the basis of a Commission proposal scheduled for the end of 2010. Adoption of the Commission's proposal by the Council may take another few years and, taking into account the time granted for transposition into national legislation, it may not be before 2014 that the requirements become truly effective.

Meanwhile the Commission is closely following the revision of the international Basic Safety Standards. As a result of the decision making rules in the European Union, the EC has so far never formally co-sponsored the international Standards. It is now envisaged to do so, in the same way as for the Safety Fundamentals. The aim is to harmonise as much as possible the definitions and requirements, both reflecting the ICRP Recommendations.

It should be emphasised, however, that the Euratom Standards and the international Standards will still look very different, on the one hand because the structures are not the same as well as the amount of detail in existing legislation; on the other hand because of the legally binding nature of the Euratom Standards, applicable to the 27 Member States of the European Union.

As a result of the rules of the recast procedure the current draft of the new Basic Safety Standards has in principle not introduced any requirements that had not been part, possibly in a different way, of the earlier Directives (except for the much broader requirements on natural radiation sources). In particular, no major changes have been made to the most recent Directive on High Activity Sealed Sources and orphan sources, except for the application of some of the requirements to any sealed sources, where this is considered to be good practice. The HASS Directive had introduced new safety, security and enforcement aspects, some of which have now been applied to all radiation sources. On the other hand there are still problems with orphan sources, and there have been important cases of contaminated metal being imported from third countries. Some requirements on orphan sources have therefore been strengthened, and a requirement on the notification of incidents with orphan sources or the contamination of metal has been introduced. Further international efforts are needed in this area to meet the conclusions of the Conference held in Tarragona in February 2009, striving for world-wide consensus on further legislative initiatives, in particular with regard to the restriction of trade in metal and scrap metal.

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Comparison of current clearance standards and their brief history

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Abstract

Over the last 30 years, governments and the nuclear and non-nuclear industry have funded efforts to assess the potential for tightening of materials cycles and minimisation of waste containing measurable levels of radionuclides. A number of studies have demonstrated the technical efficacy of converting various forms of radioactively contaminated materials into safe form and demonstrated that this can be achieved cost effectively and safely. However, until now clearance of materials considered safe has only occurred in small volumes, or within the nuclear industry for specific internal uses.

In the near future large amounts of radioactively contaminated and activated materials will be generated by the decommissioning of first generation nuclear power plants and the historic case by case approach will not be sufficient to handle the associated volumes. To minimise the waste arisings and encourage clearance of radioactively contaminated materials number of governments and international organisations have redrafted their regulations and guidance, allowing for a more fluent release of materials, from regulatory control.

The current approach taken by the international organisations and national authorities is risk based. It intends to provide protection to workers, public and environment alike. Although the International Atomic Energy Agency (IAEA), European Commission (EC) and United States Nuclear Regulatory Commission (USNRC) recommendations are in general agreement on the principles of how clearance should be applied there is still some way to go before clearance values are harmonised.

This study looks a the historic development of clearance standards over the last 30 years and compares the current risk based models and specific activity limits given by the IAEA, EC and the USNRC.

Introduction

Together the governments and nuclear and non-nuclear industry have tried to identify options for waste minimisation and tightening of materials cycles. Hence numbers of regulations and standards have been developed since the 1970s dealing with the de-minimis values. These approaches to exemption, exclusion and clearance are documented in reports published by International Atomic Energy Agency (IAEA), European Commission (EC), United States Nuclear Regulatory Commission (USNRC) and many other national regulatory bodies.

Exemption, exclusion and clearance are often discussed as coincident concepts. To a degree this is true as they are concepts to exempt radioactive materials from regulation, so that the effects of radiation on the human body are not subject to any further consideration.

Of these, the concept of exclusion is fundamentally applied to the radiation from anthropogenic origin so called “natural radiation”. There are various sources of “natural radiation” in the nature, uranium in soil, radon in air and radioactive potassium in bananas. Another source of “natural radiation” are cosmic rays, which bombard us relentlessly. The global average dose received from “natural radiation” is estimated to be 2.4 mSv/a. This radioactive dose is not subject to regulation and is hence excluded from regulation. The radioactive materials that are subject to exclusion are specified in IAEA International Basic Safety Standards (BSS) for Protection against Ionizing Radiation and for the Safety of Radiation Sources" (IAEA, 1996) and EC Basic Safety Standards (EC, 1996), as materials that are not suitable to be controlled or cannot be controlled.

There is an exception to the “natural radiation” rule: although not from anthropogenic origin radioactive fallouts on the earth surface due to nuclear tests are usually treated as exclusion, as they cannot be controlled.

The concepts of clearance and exemption, on the other hand, mean that the materials are exempt from regulation in terms of radiation protection. These concepts are possible to be applied for two kinds of materials; one not subject to regulation at the onset and the other subject to regulation at the onset but exempt afterward. Today, the former concept is termed exemption and the latter clearance.

IAEA had previously used the term of de-minimis to combine both of these concepts. In early 1980s, the term of exemption began to be used, but at this time it still included both concepts. The two concepts were first differentiated by IAEA around 1995, when both terms clearance and exemption were given in IAEA BSS, which was published in co-operation with international organizations of FAO (International Food and Agriculture Organization), ILO (International Labour Organization), WHO (World Health Organization), EC etc.

Materials requiring clearance

So what is material requiring clearance? In general it is material which is regulated, but due to low levels of radioactive materials could be exempted from regulation. From this arises another question what materials are regulated? In most developed regulatory systems regulated materials arise from utilisation of nuclear energy and radiation. The material composition and the type of contamination present will vary depending on its origin. Virtually all facilities which have been sites of nuclear activity will, at some point, become sources for radioactive materials which can be cleared. Nuclear power plants are by no means the only source of radioactive materials. For instance each step in the uranium ore processing results in uranium contaminated equipment and contaminated slag. The utilization of uranium for the production of power or weapons results in equipment contaminated not only with uranium, but also with fission and activation products. Fuel handling, storage, and reprocessing also result in more contaminated materials. Radioactive source manufacture facilities and laboratories undertaking nuclear research are another source, and yet more radioactive materials can be expected from accelerator facilities.

Albeit that there are a number of sources for radioactive materials the commercial nuclear power plants have by far the largest arisings. Previous research and regulatory databases can be utilised to assess the material composition, type of contamination present and volumes of contaminated materials. Generally the radionuclides found in nuclear power plants differ according to reactor type. According to research radionuclides resulting from activation of iron, ^{54}Mn, ^{59}Fe, cobalt, ^{60}Co, and zinc, ^{65}Zn (ICRP, 1983) were shown to be present at similar levels in pressurized water reactors (PWR) and boiling water reactors (BWR), unlike the activation nuclides for nickel and tin. The nuclides resulting from the activation of nickel, ^{58}Co and ^{63}Ni, were found to be present at a higher level in PWR's, whereas the tin activation nuclide, ^{125}Sb, was present at higher levels in BWR's. Contamination of materials by caesium isotopes, ^{134}Cs and ^{137}Cs is higher in PWRs, while strontium isotopes ^{89}Sr and ^{90}Sr are usually present at the same level (Dyer and Bechtold, 1994). A survey of reactor parts showed that the radionuclides which accounted for 1% or more of the total activity were ^{54}Mn, ^{55}Fe, ^{60}Co, ^{65}Zn, ^{125}Sb, ^{134}Cs, and ^{137}Cs (Schlienger et al., 1996). These are the key radionuclides for all materials arising from nuclear power plants.

History of clearance standards

Since the early 1980's IAEA have demonstrated an interest in the clearance of materials, when they established the basic principles, which underlie the current approach in their safety series document 89 (IAEA, 1988a). In 1996 IAEA further defined the issue of clearance when they published their first interim report on the recommended clearance limits (IAEA, 1996). In 2004 IAEA revised these clearance limits, which now include 1650 radionuclides (IAEA, 2004).

From 1996 EC has developed the idea of clearance further from the early 1980s attempts, when they were assessing the possibilities of recycling radioactive material and published a number of reports dealing with the issue. These reports address the issue for specific cases concerning a number of fields where radioactive material is generated. There is also a specific EC radiation protection report, RP-89, dealing with the radioactive scrap metal (EC, 1998).

The most recent EC report in 2002 EC RP-122 gives the specific clearance limits for 197 radionuclides, which could be present in any type of contaminated material (EC, 2002). These clearance limits added, modified and eliminated radionuclides compared with the previous EC clearance limits published in 1996 (EC, 1996).

The decision to apply the clearance criteria in the EC countries remains the responsibility of the competent authorities in the Member States, and they can have their own clearance values where appropriate. The guidance on clearance in RP-89, different from other EC guidance, has a caveat for a less stringent clearance limits allowing for slightly radioactive metal scrap, components and equipment from nuclear fuel cycle installations to be authorised for clearance to the public domain whenever recycling within the nuclear industry is not appropriate subject to meeting the clearance criteria (EC, 1998).

US Atomic Energy Commission (AEC), the predecessor of the USNRC, has since 1940's considered slightly contaminated materials, but the original 1957 rule for the protection against radiation did not include specific values for clearance of slightly contaminated materials. An amendment was later added which allowed the regulator to grant a licensee a permission to release materials on a case by case basis. During the 1970's USNRC began the process to set uniform standards for clearance of materials from regulatory control. These efforts have continued through the 1980's and 1990's (USNRC, 1999). The latest effort came in 2003 where the technical document declared that materials with low concentrations of radioactivity can be cleared and de-regulated (USNRC, 2003).

The final 2003 technical document NUREG 1640 includes specific activity limits for 115 radionuclides and a detailed discussion of the methodology used to estimate the resulting annual doses USNRC, 2003), which are given as a justification for the clearance limits. However, currently these clearance limits are only used for case by case clearance decisions.

According to these international and national policies clearance is seen as a logical and well-based scientific concept with the potential for avoiding unnecessary regulation. While number of studies have made it clear that clearance is justified and that its adoption is important for development of nuclear power and utilisation of radiation, the various clearance standards and conflicting research have alienated people from the decision making process making it more difficult to find acceptance for clearance.

The current developments in clearance standards seem to indicate, that harmonisation of clearance values is finally moving forward as two of the major safety standards collections have indicated that they are trying to harmonise their clearance values. The current indication is that IAEA will be implementing the clearance values given in R.SG-1.7 in their new BSS with EC following suite in their new BSS (Janssens 2009).

Comparison of current clearance standards

Current clearance standards do not agree well when it comes to the specific activity limits set for specific radionuclides. When comparing the IAEA, EC and USNRC documents there are significant differences for majority of the radionuclides listed; 1650, 197 and 115 respectively.

Comparison of the key radionuclides is presented in tabulated format in table 1, and in a graphical form in figure 1.

Table 1. Comparison of clearance values for key radionuclides.

Radionuclide	USNRC (Bq/g)	IAEA (Bq/g)	EU (Bq/g)
H-3	526	100	100
C-14	313	1	10
K-40	2,94	-	1
Mn-53	11400	100	1000
Mn-54	0,63	0,1	0,1
Fe-55	21700	1000	100
Co-60	0,19	0,1	0,1
Ni-63	21300	100	100
Sr-90	17,50	1	1
Mo-93	270	10	10
Tc-99	6,25	1	1
I-129	0,05	0,01	0,1
Cs-137	0,63	0,1	1
Ir-192	0,91	1	0,1

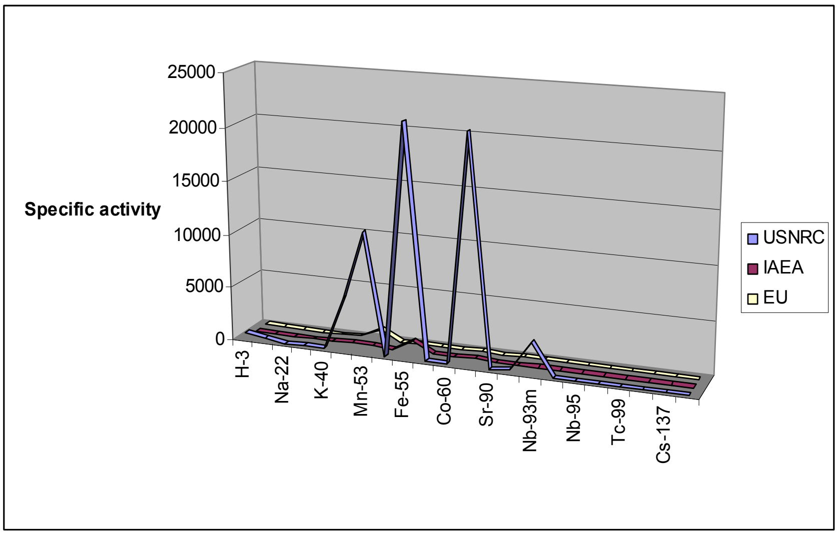

Fig. 1. Graphical comparison of clearance limits for key radionuclides.

Although the specific activity limits in the IAEA, EC and USNRC documents are not in agreement, they do agree on the principles of how clearance should be applied. The comparison of the content of these documents is given in table 2.

Table 2. Comparison of content of clearance standards.

	NUREG 1640	RS-G-1.7	EC RP 122
Waste material	All materials	All materials	Many kinds of materials
Nuclides	115	1650	197
Amount of material	Large	Large	Large
Disposition scenarios	Typical exposure scenarios for all materials	Typical exposure scenarios for all materials	Enveloping scenarios
Application constraints	No constraint on the origin and the type of materials	No constraint on the origin and the type of materials	No constraint on the origin and the type of materials
Standard dose	10 μSv/a	10 μSv/a	10 μSv/a

In all the documents the specific activity limits are based on the idea of negligible risk to the individual. All three standards have based their approach on the ICRP recommendation, that a dose of 10 μSv/a, which is 1% of the recommended public annual dose of 1 mSv, is negligible. By comparison, this means that the annual dose of 10 μSv is less than 0,5% of the natural radiation dose even in areas of very low background (~2 mSv/a) (UN, 2000), and less than 0,2% of the annual radiation dose in Finland (Muikku et al., 2007).

In the spirit of the concept of exclusion, clearance levels should be set at such levels that the resulting doses are small enough to be compared with doses from "natural radiation" so that the risk to the public health is so small it can be neglected.

Development of scenarios and dose evaluation

The clearance values in the current IAEA document RS-G-1.7 are derived numerically using 10 μSv/a for credible scenarios, 1 mSv/y for scenarios with low probabilities, and 1 manSv for the collective dose as the standard doses. The details of those derivations are shown in IAEA Safety Report Series No. 44 (IAEA 2004b).

In order to derive clearance levels, dose evaluations were performed by setting up required parameters for every pathway to be evaluated for the scenarios. Table 3 show evaluation pathways assumed for RS-G-1.7 clearance values.

Table 3. Evaluation pathways assumed by the IAEA.

Scenario	Group	Exposure path	Specific nuclides
Worker of repositories and other end-point facilities	Worker	External, inhalation and ingestion	^{239}PU and ^{241}Am (ingestion)
Worker of melting facilities	Worker	External, inhalation and ingestion	
Other workers (driver etc.)	Worker	External	
Peripheral residents of repositories or other facilities	Child	Inhalation and ingestion	^{3}H, ^{36}Cl, ^{41}Ca, ^{59}Ni, ^{63}Ni, ^{90}Sr, ^{99}Tc (ingestion)
Peripheral residents of repositories or other end-point facilities	Adult	Inhalation and ingestion	^{3}H, ^{36}Cl, ^{41}Ca, ^{59}Ni, ^{63}Ni, ^{90}Sr, ^{99}Tc (ingestion)
Peripheral residents of melting facilities	Child	Inhalation	
Residents of the house built with the contaminated materials	Adult	External	^{54}Mn, ^{60}Co, ^{65}Zn, ^{94}Nb, ^{134}Cs, ^{137}Cs, ^{152}Eu,^{154}Eu (external)
Peripheral residents of public facilities built with the contaminated materials	Child	External, inhalation and ingestion	
Residents consuming contaminated river fish and well water	Child	Ingestion	^{14}C, ^{129}I (ingestion)
Residents consuming contaminated river fish and well water	Adult	Ingestion	^{14}C, ^{129}I (ingestion)

The differences in the chemical and physical behaviour of radionuclides to be evaluated results in different doses for each nuclide at a given concentration, and hence each parameter is conservatively set up to include various postulated events. According to the dose-evaluation results for all evaluation pathways the clearance value is the minimum radionuclide concentration which is equal to the standard dose under any of the postulated events.

Table 3 also gives the major radionuclides which are critical for each of the pathways. Namely, it demonstrates that critical pathways are external exposure for gamma emitters (i.e.^{60}Co and ^{137}Cs), oral ingestion of beta emitters such as ^{90}Sr, and inhalation ingestion of alpha emitters such as ^{241}Am.

Concept of Limited Clearance

In accordance to the concept of clearance; materials, which are cleared are not subject to further regulation, but, as described in the development of EC clearance standards a concept of less stringent "limited" clearance exists, which provides more lax limits, and provides further conditions on the method of clearance or destination of cleared materials. This concept of "limited" clearance is also termed as "specific" clearance or "conditional" clearance.

As described earlier, the clearance level is usually set as the lowest concentration delivering the standard dose for the assumed exposure pathways. Therefore, if the assumed exposure pathways can be limited by imposing conditions on the implementation of clearance, the number of exposure pathways for the scenario evaluation will be reduced, and for the standard dose higher radionuclide concentration levels can be accepted. Similarly, if the amount of the materials for clearance is smaller, the clearance value will become higher.

As the USNRC did not gain acceptance for their Clearance approach outline in NUREG 1640 they are currently releasing materials under previous rules following the concept of “limited” clearance. There are also a number of other countries where this type of clearance approach has been incorporated into the legal regulations, such as Finland (STUK, 2008) and Germany (Sefzig, 2009).

Obstacles to clearance

Many countries face significant obstacles in adopting and applying clearance standards. The obstacles have been identified in a technical report published by the IAEA. The report deals with recycling of metals, but all of these factors could become obstacles for radioactive materials clearance aswell. Some of the factors are specific to a facility or a country and others are international (IAEA, 1988 b). In the context of clearance of metals these are:

- The availability of regulatory criteria giving activity levels for clearance and alternatives to applying the clearance levels
- The availability of technology and facilities to recycle the metals
- The availability of instrumentation to measure regulatory clearance levels and quality assurance programs to assure compliance with the criteria.
- The effect that recycling of materials will have on the extension of natural resources
- The economic implications including cost of decontamination, waste disposal, market value of recycled materials.
- The socio-political attitudes in the affected country or industry regarding the recycling/reuse of materials or components.

Of these factors the first is a legislative issue, which is for the governments to decide. The next two are technological issues, which are much dependant on the country’s infrastructure. The fourth is an international issue with stakeholder implications regarding sustainable development and conservation of natural resources. The second to last factor is a country specific issue with international economic dimensions. The last factor is a country specific issue which is strongly affected by stakeholder attitudes. In a modern western society it is unlikely that these first two factors would be significant obstacles to radioactive scrap metal recycling as readily available standards exist and there are ongoing operations around the world.

The third factor is a potential obstacle due to detector technology and associated cost implications, but these are discussed in detail in scientific publications and international guidance (Lubenau and Yusko, 1998, Clouvas et al., 2005).

Fourth issue is an international issue with strong stakeholder implications. However, in the current atmosphere of environmental conservation the considerate use of raw materials can be advantageous to radioactive scrap metal recycling if the stakeholders can be convinced of the safety of operations and the quality of the final product.

The fifth factor can become a hindrance if consumers do not accept the use of trace radioactive scrap metal in recycled metals, however the cost implications are unlikely to be an issue if burial cost avoidance is added to the financial calculations. The issue of cost benefit analyses is addressed by a number of scientific and articles and reports as well as national research (Menon et al., 1990, Hill et al., 1996, Yuracko et al., 1996, Rivera et al., 1996, Andreani and Bailo, 2000), the feasibility has also been demonstrated by the operational plants around the world.

The last factor can be, and it is the most serious obstacles to scrap metal recycling. To date a number of stakeholder involvement processes have been undertaken to convince the public of the benefits and safety of radioactive scrap metal recycling, but these stakeholder processes have not been successful in alleviating the concerns of the stakeholders and there is no free market for cleared radioactive scrap metal in the world.

Conclusions

Clearance regulation is provided to ensure the safety of the general public and the environment, and to ensure, that no one suffers health hazards from regulated radioactive materials. To a degree the numerous studies undertaken in developing the current clearance standards have demonstrated that it can be done safely and cost effectively, but the fact remains that significan differences are present in the clearance values eventhough they are working with the same standard dose of 10 µSv/a. This is mainly due to different approximations used in the exposure scenarios from which the limits have been derived.

This inconsistency makes it more difficult for regulators to convince stakeholders of the safety concerns associated with recycling of low activity scrap metal, and it would be more desirable to have consistent standards for materials rather than state of the art science.

With the indications coming from the IAEA and EC that they will be harmonising their clearance standards there seems to be some hope that in the future harmonisation for clearance values can be achieved, but this is still a long way away considering the effort that has been put into researching clearance standards and how jealously the assumptions behind clearance standards are discussed in the international environment. However if harmonised clearance standards can be agreed upon, it is more likely that the industries dealing with cleared materials would be more willing to include cleared materials in international trade.

According to IAEA clearance is: "a logical and well based scientific concept with the potential for avoiding costly and unnecessary regulation", and clearance of materials is demonstrated to be achievable through licensed operations around the world, but making it acceptable requires careful work from the international organisations and governments in justifying clearance and thorough stakeholder engagement and involvement at policy, strategy and facility levels.

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Current situation with application of new ICRP system of radiological protection in the Russian Federation: regulation and optimization issues

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Abstract
The current Russian basic regulative document, Radiation Safety Standards, needs updating, which must be carried out in accordance with common international practice. In the light of issuing of new ICRP principal recommendations in 2007, review of the IAEA Basic Safety Standards is in progress. In compliance with the national practice, the IAEA member states implement the Basic Safety Standards in a form of national systems of laws, norms and rules. In fact, the current national radiation safety standards can be updated in Russia only after new IAEA standards issuing. Nevertheless, even today, we must take the contemporary principles of development of the radiation safety system postulated in the ICRP recommendations into account. There are no stricter hygienic regulations in ICRP Publication 103 in comparison with those established in Russia, but there are some new conceptual provisions and concepts in it. The paper deals with the frame of national system of radiation safety regulation. Application of optimization and dose constraints in the course of Russian NPP operation is the specific case and its relevance is stressed in terms of future development of nuclear power engineering. Using the nuclear legacy regulation as an example, an importance of application of new exposure situations from ICRP Publication 103 has been shown.

Introduction
Radiation safety regulation consists of:

- Science based knowledge of ionizing radiation effects under development of the United Nations Scientific Committee on the Effects of Atomic Radiation;
- Conceptual model of the human safety and effective control of radiation sources and technologies being generated by the International Commission on Radiological Protection (ICRP);
- Legislative and regulative support of radiation protection and safety for different modes of nuclear energy using and another activity connected with radiation exposure to the human, being developed by the IAEA.

Development or change of each component has resulted in the necessity of the regulation system revision and its adequate submission at the international and national levels. The International Basic Safety Standards (BSS) have always followed the establishment of new ICRP recommendations. For example, the fundamental ICRP recommendations of 1990 issued as Publication 60 were the basis of the revised BSS, published in 1996. Then, in 1999, the Russian national radiation safety standards were developed.

The current Russian fundamental regulatory document – Radiation Safety Standards (NRB, 2009) – requires necessary revision, which must be carried out according to the established world practice. As it is well-known, issuing of the new ICRP fundamental recommendations in 2007 (ICRP, 2007), caused the revision of the IAEA BSS revision (IAEA, 2010). However, just before the official publication of new BSS, we must take into account the up-to-date principles of the radiation safety development as they are postulated in the ICRP recommendations, in our everyday practice.

Regulation of radiation safety and protection in Russia: regulatory bodies, legislative acts and regulative documents

In addition to operators directly responsible for radiological protection in Russia, as it is well known, there are some regulatory bodies, whose functions include regulation, licensing of the appropriate activity and its supervision, fig. 1.

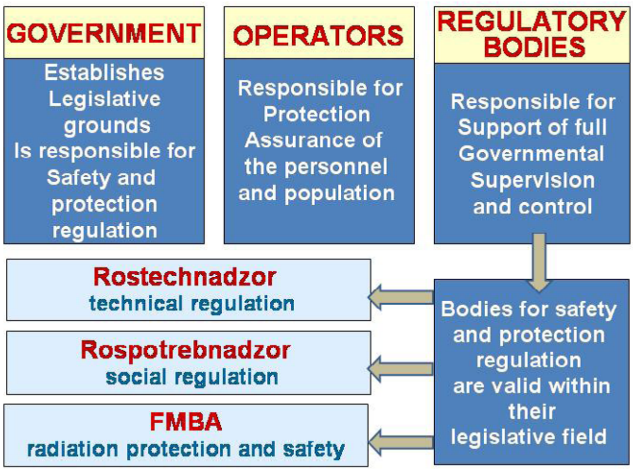

Fig. 1. State regulation of radiological protection in the Russian Federation.

According to the current Russian legislation, the Federal Medical-Biological Agency (FMBA of Russia) is responsible for medical and sanitary support as well as for the state sanitary epidemiological supervision. It covers organizations in some industrial branches with especially hazardous work conditions and the population of some Russian territories according to the list approved by the RF Government. This list includes all radiation hazardous facilities all over Russia (more than 400 facilities). One of the FMBA's principle functions is the state regulatory supervision of safety during nuclear energy using.

At that, some these functions are implemented through the system of the state sanitary and epidemiological regulation, to the extent that approval of the documents developed by the RF State Chief Medical Officer and their following registration in the

RF Ministry of Justice. Such regulations and sanitary rules are obligatory for observation by operators. Over recent 5 years, experts from the FMBA of Russia have developed more than 100 scientific and technical regulatory documents on radiation protection of nuclear workers and the public. 20 of such documents are ascribed to the category of sanitary rules and registered in the RF Ministry of Justice. They include sanitary rules for design and operation of nuclear power plants, including floating ones; regulations for surveillance of radiation safety during overall decommissioning/dismantlement of nuclear submarines; and the requirements for the Russian ports during enter and moorage of ships equipped with nuclear powered installations etc.

Principles of legislative regulation are postulated in laws and directives of the Government of the Russian Federation. National system of radiological protection and radiation safety assurance in the Russian Federation is based on two principal documents: Radiation Safety Standards (NRB, 2009) and Main Sanitary Rules for Radiation Safety and Protection (OSPORB, 1999). Figure 2 shows hierarchical frame of regulative and methodical documents as an example of radiological protection regulation of the personnel and population as applicable for the facilities of the State Atomic Energy Corporation “Rosatom”.

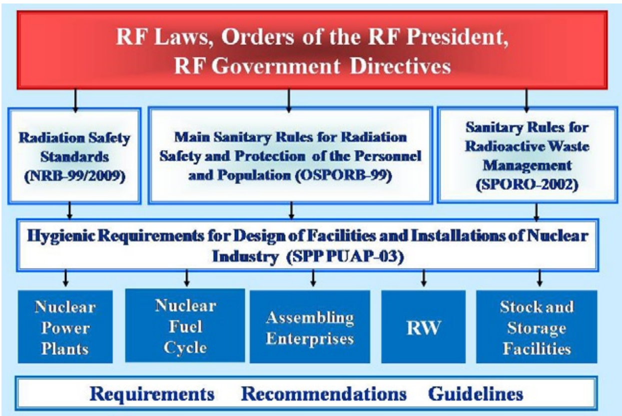

Fig. 2. Normative and methodic support of radiological protection of the personnel and population from the “Rosatom” facilities.

The legislative and regulative background established over the recent decade supports safe operation of nuclear facilities. However, its practical application revealed some problems as well. In the regulative documents, there are difficult questions to be solved connected with operation of the national nuclear industry, such as:

- remediation of sites contaminated due to past nuclear activity and “uranium legacy”;
- decommissioning of nuclear energy using facilities or justification of prolonged operational time of functioning facilities;
- radioactive waste management;
- assurance of effective emergency response;
- difficulties in adequate application of the optimization principle.

The mentioned problems can partially be solved on the base of issued ICRP principal recommendations - Publication 103.

Impact of the new system of the radiological protection on development of radiation safety and protection system in Russia

It is necessary to note that regardless the fact that the final BSS version is not finished so far, the principal Russian regulative document - Russian Radiation Safety Standards (NRB, 2009) has already been revised and is in force since 1 September 2009. Its amendment is due to not only Publication 103 issuing. Such kind of amendments is dictated by the termination of the ten-year operational time.

New version of the Russian Radiation Safety Standards keeps the previous structure. 300 different amendments have been made in total and the following in particular, up-to-date detriment adjusted nominal risk coefficients for stochastic effects after exposure to radiation at low dose rate have been given. In the section «Limitation of natural exposure» in addition to regulation of the construction raw materials, the finished production manufactured on its base has been added together with the mineral raw materials. Regulation of chemical fertilizers and agro-chemicals has been amended. In assessment of drinking water radiation safety the supplement says that regulation is made by adults, and intervention levels for the particular radionuclides are given as a special annex. In the section «Limitation of medical exposure», the criterion on discharge of the patient from the hospital after treatment with radiation sources is amended.

It should be noted that not all provisions of new ICRP system found their reflection in the revised document.

In Russia, the main actual task is to explain, how new ICRP conception of Publication 103 can be effectively and successfully applied in the regulation practice. The following new provisions of the radiation safety and protection system can be important for Russia:

- refusion of “practice and intervention” conception and its replacement by three types of exposure situation: planned, existing and emergency;
- introduction of dose constraint and reference levels for each exposure situation depending upon the category of individuals exposed;
- use of “representative individual” concept instead “critical group”;
- establishing of the eco-centric principle of radiation protection of non-human species and environment.

Some changes of the radiological protection system will cover the following issues as well review of weighting factors for relative radio-sensitivity of organs and tissues, selective use of collective dose value, etc.

Regardless of mentioned changes, together with the above new provisions of the radiological protection system, fundamental revision of current national NRB system is not assumed. Figure 3 illustrates evolution or dynamics of dose limit change in the ICRP documents and in the Russian documents regulated radiation safety, respectively. So, regarding to the professional exposure in planned situation, ICRP continues to advise to present dose limit as 20 mSv a^{-1} effective dose. The public exposure in planned exposure situation is either unchanged: ICRP continues to advise 1 mSv a^{-1} dose limit.

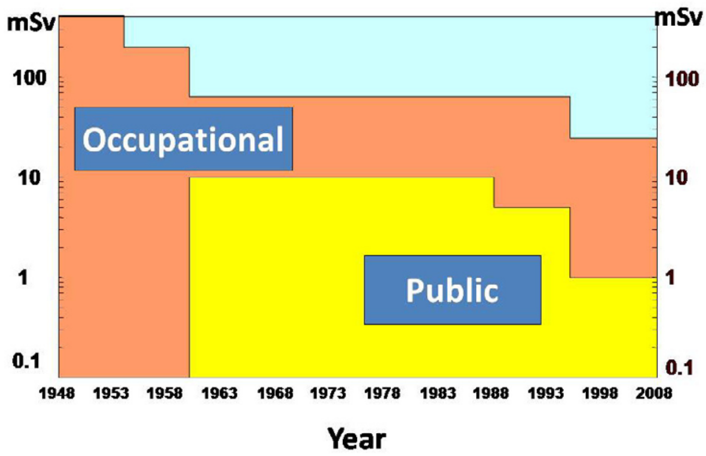

Fig. 3. Evolution of individual dose limits.

Thus, even before issuing of new IAEA international safety standards instead those in use today, it is already evident that national NRB will be the subject of evolutional amendments relating to «reconstruction» of the conceptual background of the radiological protection system, as well as change of digital values of dose coefficients.

In the nearest time, hard work should be performed on introduction of new recommendations into the national system for radiation protection and safety assurance. Therefore, after official issuing of ICRP Publication 103 in Russian, wide discussion is necessary both of the Russian terminology from the "Glossary" section and the whole text translated. Application of new conceptions and terms in the system of regulations must be weighted and one should avoid their formal application. When improving national regulative documents, one should analyze carefully national experience and take into account the up-to-date social and economic circumstances in Russia. The priority of changes from ICRP Publication 103 is to be specified to select those relating to the most important problem in nuclear industry.

Radiological protection optimization in the course of the Russian NPP operation

Information materials on the Russian NPPs operation are being demonstrated as a positive example of good practice of adequate application of the radiological protection optimization principle. In the regulative documents concerning this problem, the dose quote of 25% public dose limit, i.e., 0.25 mSv a^{-1} is written and observed over more than 10 years (SP AS, 2003). The dose quote in the national regulation assumes or is similar to dose constraint in the planned exposure situation. The current NPP 0.25 mSv a^{-1} dose quote value corresponds to the ICRP recommendations for dose constraint (of 0.3 mSv a^{-1}) established in the course of optimization of public radiological protection against existing or potential radiation source. This dose quote also corresponds to practice of other states, where nuclear power engineering is under development.

For the functioning Russian NPP, 250 μSv a^{-1} quote for the public dose is established, while for NPP under design and construction this value is 100 μSv a^{-1}. These quotes are being established for total public exposure due to radioactive gas-aerosol atmospheric discharges and liquid effluents in surface water reservoirs for NPP as a whole, regardless the amount of energy units on the industrial site. Table 1 shows the quote values for the public exposure originated from radiation factors (discharges and effluents) under route operation of the Russian NPPs.

Table 1. Quotes for the public exposure originated from discharges and effluents under route operation of the Russian NPPs, μSv per year.

Radiation factor	Functioning NPP	NPP under design or construction
Gas-aerosol effluents	200	50
Liquid discharges	50	50
Ttal	250	100

According to the regulatory document (SP AS, 2003), the approach is proposed for identification of permissible radionuclide releases into atmosphere based on the optimization principle and on the acceptable risk conception, when the lower border of the risk optimization area for the single radiation source (NPP, in this case) is assumed to be 10^{-6} a^{-1} (that corresponds to annual effective dose of about 10 μSv a^{-1}). For lower values, public doses are not to be reduced, i.e., we mean here application of additional radiation protection. The quote value is considered as upper border of potential public exposure due to radiation factors in the course of optimization of such public protection under condition of route (normal) NPP operation. Figure 4 illustrates the general ides of such approach.

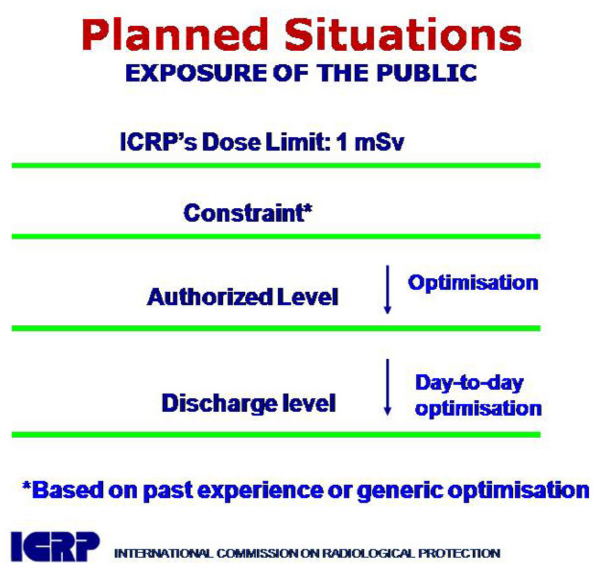

Fig. 4. Radiation protection optimization of the public living nearby the NPP.

Thus, permissible radionuclide releases due to the NPP operation result from optimization approach based on the “benefit-harm” analysis. Here, numerical values of permissible releases vary over the range between the upper and lower borders of optimization and they are not derived levels of dose quotes for the single radiation source (MG, 2004). When establishing the permissible annual releases of radioactive gases and aerosols into atmosphere, the fact was taken into account that noble radioactive gases made the main contribution (more than 98%) into the public dose under route of NPP operation. These are: argon, krypton, xenon and radionuclides ^{131}I, ^{60}Co, ^{134}Cs, ^{137}Cs (^{24}Na –for fast neutron type reactors, BN-600). Limitation and control of other radionuclide activities being detected in the NPP discharges seems to be unreasonable because of their trivial contribution into dose. Table 2 includes values of annual permissible discharges of radionuclides for the NPP equipped with different reactors taking into account ratios of the nuclide activities in such discharge and conditions of the discharge (heights of ventilation pipes).

Table 2. Annual permissible releases of radioactive gases and aerosols from the NPP into atmosphere.

Radionuclide, dimension	T3 with RMBK	T3 with WWR and BN
Noble gases, TBq*	3700	690
^{131}I (gaseous + aerosol forms), GBq**	93	18
^{60}Co, GBq	2.5	7.4
^{134}Cs, GBq	1.4	0.9
^{137}Cs, GBq	4.0	2.0
Note: * 1 TBq = 10^{12} Bq = 27 Ci; ** 1 GBq = 10^{9} Bq = 27 mCi		

The above mentioned permissible discharges are those of minimum significance and they observance guarantees that the public dose during rout operation mode will not be higher 10 μSv per year. Taking into account doses, which are the upper border of optimization, maximum permissible discharges are established for the functioning NPPs at the level of 20 permissible discharges, while for the NPP under design and construction – at the level of 5 permissible discharges. Values of maximum permissible discharges for all NPPs are 5 times higher than the permissible discharge.

If real release (discharge/effluent) from the NPP is lower than maximum permissible values, but higher than the permissible ones, then it is recognized that radiation impact of the NPP on the public and environment does not comply with the optimization principle. This is evidence of the manufacture culture violation and is to be considered with the purpose to eliminate the excess revealed. Excess of maximum permissible discharges or effluents is forbidden under route operation mode of the NPP, because this is violation of the sanitary norms and rules and can justify the suspension of the NPP operation.

Real releases from the Russian NPPs are not higher than 20% of permissible releases. Releases during power unit closures are much higher than those during normal operation at the stable power level (noble (inert) radioactive gases – up to 40%, ^{131}I – up to 70% of gross annual releases). At stable powers of the power units, measurement of the NPP release activities is of illustrative nature because of their low values.

Discharges and effluents due to the NPP operation at the level of about 100% of the permissible releases are undoubtedly acceptable; effective public doses confirm this fact for the population living at the NPP vicinity (Shandala N. et al., 2007). Table 3 includes doses to the public on the example of Volgodonsk NPP.

Table 3. Individual effective public doses at the supervision area of the Volgodon NPP, mSv a^{-1}.

Regions of the supervised area	Years of surveillance				
	2004	2005	2006	2007	2008
Cymlyansky	0.02	0.012	0.021	0.06	0.08
Volgodonsk city	0.015	0.01	0.09	0.017	0.019
Dubovsky	0.09	0.02	0.015	0.013	0.005
Zimovnikovsky	0.012	0.014	0.011	0.01	0.004

Thus, real discharges and effluents originated from the NPP operation are optimized and their reduction is not economically justified. The task is to keep the reached level of atmospheric effluents and discharges into water media, especially in the light of ambitious plans of nuclear power engineering development in Russia.

Optimization and regulation of the Russian nuclear legacy

Optimization of radiological protection in Russia is also important issue in remediation of nuclear legacy of sites and facilities of ex-USSR. Routine operations and discharges from radiation hazardous facilities do not contribute substantially to the public exposure and do not result in any significant consequences for health. In contrast, some legacy radioactive contaminations due to radiological accidents and emergency situations (Techa River, 1949; Kyshtim, 1957; Chernobyl, 1986) resulted in radiation doses to some population groups that significantly exceeded the permissible levels, Table 4 (Balonov M., 2008).

Table 4. Public exposure due to man-made radionuclide releases.

Source	Time period	Significant Nuclides	Mean dose (mGy, mSv)
Global fallout	1950 – 2020	^{137}Cs, ^{90}Sr, ^{131}I, ^{14}C, ^{3}H	1.1
Techa River, PA “Mayak”, Russia	1949 – 2020	^{90}Sr, ^{89}Sr, ^{137}Cs, etc.	50-2000
Chernobyl, USSR	1986 – 2056	^{131}I, ^{134}Cs, ^{137}Cs, ^{90}Sr	Up to 500; Thyroid – up to $10 \cdot 10^3$

Emergency exposure led to adverse health effects, such as, radiation sickness and a long-term increase in the incidence of cancer in the residents of the affected areas. The research institutions under the FMBA of Russia study consequences of contamination of Techa River, Southern Urals, due to both unauthorized radioactive discharges from the PA "Mayak" at the later 1940s and the accident at the "Mayak" in 1957. Following some radiological accidents, significant detriment of human health and environment has been observed. Consequences of defense activity are rather considerable for environmental safety at the sites of nuclear submarine allocation. Such kind of activity resulted in large amounts of the spent nuclear fuel and radioactive waste accumulated at the sites of temporary storage in the Russian Northwest and Far East.

What kinds of problems do arise in optimization context in case of existing exposure? Let us consider them on example the nuclear legacy regulation. The main Russian regulative document (NRB, 2009) presents the guidance (intervention criteria) with respect to the radioactively contaminated areas as a reference Appendix 5. Nevertheless, there is no comprehensive guidance for the existing exposure situation. The optimized protective and remedial measures are recommended at annual dose over the range 1 – 20 mSv; at dose >20 mSv, residence at the territory is forbidden. Prima facie, quite good general compliance with the ICRP Publication 103 is evident; however, application of the lower boundary (1 mSv/year) for the large-scale situation (Chernobyl, Kyshtym, Techa) seems to be inadequate. This can be explained by the Chernobyl Law adopted on the rise of democracy at the early 1990s and by wrong use of the dose limit in case of emergency and existing exposure situations (Shandala N. et al., 2009).

According to the ICRP Publication 103 (ICRP, 2007), environmental remediation can be considered as an existing exposure situation: exposure already exists, when decision is to be made on radiation protection. This kind of exposure situation includes prolonged exposure due to excess radiation background, after radiation accidents, following previous radiation substance handling (including nuclear weapon manufacturing and tests, peaceful nuclear explosions etc.). In many respects, consequences of the Chernobyl accident resulted in generation of such independent exposure situation. The previous ICRP Publication 82 (ICRP, 1999) and the IAEA documents (WS-R-3, WS-G-3.1) recommended the following principal provisions with respect to the remediation situation:

- Dose limits cannot be used, because they cannot be observed in each case.
- Criteria for the human protection – justification and optimization of intervention.
- Generalized criterion of non-intervention – non-exceeding annual effective dose to the public due to all environmental sources (including background) is 10 mSv. Intervention may be grounded above this level.

To develop the above mentioned provisions, the ICRP Publication 103 refuses the "intervention" concept and introduces the reference dose or risk level. The reference level is such level, above which radiation exposure is impermissible, and so optimized protective measures are to be taken, including those at dose below the reference level. The regulatory body establishes the reference level for the specific or typical situation. In case of the existing exposure situation, the reference level of annual effective dose within the range from 1 mSv to 20 mSv is proposed to be established. Higher levels are proposed to be applied in more large-scale situations.

Thus, in addition to review of actual Russian regulatory document (here we mean reasonability of introduction of three exposure situations instead practice and intervention available now), some special criteria of the environmental remediation are to be developed.

Conclusions

1. The ICRP Publication 103 has been translated into Russian.
2. There are no stricter hygienic regulations in ICRP Publication 103 in comparison with those established in Russia, but there are some new conceptual provisions and concepts in it.

3. The safety levels reached due to the optimization principle application in the field of radiation safety and protection at the Russian NPPs and at the areas of their potential impact on the public and environment can be recognized as quite good.
4. The actual Russian radiation protection system is not arranged enough in its documentation basis and differs from the international one, in particular, in inadequate application of dose limit for the planned exposure situations (1 mSv/year) under conditions of emergency and existing exposure situations. Introduction of three exposure situations in the Russian regulation can improve this situation.
5. Recommendations are as follows: 1) To introduce a chapter on radiation protection of the public in the existing (prolonged) exposure situation into the national radiation safety standards and put it in harmony with the international radiation protection system; 2) To promote development of the international connections on studying of good experience in the radiation protection is necessary.

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Stakeholder Involvement in Medical Practices – Report of the Heads of European Radiation Control Authorities HERCA

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Abstract
A working group of HERCA has reviewed a large variety of examples of stakeholder involvements in medical practices which have already been performed in the member states and in addition looked at possible stakeholder involvements for the future, where a leading role of the national radiation protection authority is needed. The authorities should take the lead to bring stakeholders together to solve today's challenge in a concerted manner. These challenges appear where different professional groups work together on new technologies and/or new processes. Examples can be found in radiation therapy (image guided radiotherapy, dealing with accidents), in nuclear medicine (PET/SPECT-CT), in radiology (going digital, patient dose optimization), screening, the complex of problems through self referral and many others.

Introduction
In diagnostic radiology radiation is used to form an image of a plane or volume, or in nuclear medicine to visualize the function of an organ, and the radiation dose to the patient is just an unwanted side effect. In radiation therapy radiation as such is used to control cancer growth for cure or palliation. We have seen a tremendous technological development both in diagnostic and therapeutic applications of radiation. This growth and development is mainly of great benefit to the patients as individuals and to the society as a whole, but it also causes a strong increase of medical radiation exposure of the population which is of concert for radiation protection reasons. Only a close collaboration between all the stakeholders will allow this dose increase to be understood and kept under control.

Material and methods

The Heads of the European Radiation Control Authorities HERCA have first met in 2007 to discuss the common problems in radiation protection in Europe. 22 member states of the European Union and Norway and Switzerland identified the key issues which needed to be worked on. 5 main topics have been chosen to be looked at more closely in therefore created working groups. The task of Working Group 5 was to analyse the situation of stakeholder involvement in medical practices and give recommendations to national radiation protection authorities of how to identify the relevant stakeholders and how to proceed in involving them in radiation protection issues.

Results

The working group has reviewed a large variety of examples of stakeholder involvements which have already been performed in the member states (see Annex A) and in addition had look on possible stakeholder involvements for the future, where a leading role of the national radiation protection authority is needed.

The participating experts from HERCA member states have discussed the role of the authority in the involvement of stakeholders, have tried to identify relevant parties and to make some recommendations and to give examples of stakeholder involvement.

Definition and identification of stakeholders in medical practices

A stakeholder is someone who is (or should be) entitled to have an interest in radiation protection in medicine. To give a better overview, stakeholders are split into three groups *Justification, Optimization* and *General*.

Justification	• Medical doctors, medical societies and associations • Patients, patient organizations • Legislator
Optimization	• Medical doctors, medical physicists, technicians, other medical staff • Manufacturers and suppliers, staff undertaking installation and maintenance • Hospital directors
General	• Patients and their relatives • Patient ombudsman • members of the public • Insurance and social security • Legislator and health authorities

In the group *Justification* the stakeholders are involved in the process before a prescription has been issued by the medical doctor. Subsequently in the group *Optimization*, the stakeholders are carrying out the procedure.

The overall goals of stakeholders involvement

The national authorities competent in the field of in radiation protection should take the lead in bringing the different stakeholders together (stakeholder platform), analyzing the different interests and needs (stakeholder analysis) and motivate them to participate actively in optimizing medical radiation exposure (stakeholder participation).

The goals, which should be achieved by this process, are summarized in the following table:

Justification	• Specialized Medical doctors in charge of radiotherapy, radiology, nuclear medicine, etc are aware of the principle of justification (ICRP Publication 103 ….) • Specialists and generalists use the guidelines "good practices" (Guide du bon usage, Orientierungshilfe, EU referral criteria RP 118…) • Medical doctors and dentists (generalist) are aware of the justification of radiological examination • Patients are informed of the risks and the benefits of the procedure
Optimization	For the application of the principle of optimization, medical doctors in charge of radiotherapy, radiology and nuclear medicine, physicists and technicians are in charge of : • State-of-the-art technical equipment, quality assurance and maintenance • Optimized procedure for the realization of the examination (i.e. diagnostic reference levels, handbooks of good practice…) • Regularly trained staff • Defined responsibilities in the process (rights and duties) • Resources for radiation protection in all radiological procedures • Feedback culture (safety information system, continuous improvement, self auditing….)
General	• Manufacturers care about radiation protection • Awareness and knowledge in radiation protection (public, patients …) • Well-accepted authorities, who inform, coach and train all users…. • organizations preparing standards are taking care of radiation protection

Discussion

There is no doubt that important benefits can result from interactions between stakeholders and regulatory authorities.

These interactions can vary significantly from one situation to another. Their importance, intensity, purpose and expected outcome depend on well-known factors (benefits, detriments, risks – real as well as perceived). Associated circumstances (e.g. sense of urgency, potential crisis situation, current political engagements of governments) also affect the interactions, as well as socio-economical factors (e.g. need for improved business conditions). In addition, the expertise and resources available to the authorities, as well as the potential need for improved regulatory framework, play a role on the interactions with the stakeholders.

The purposes and expected outcomes of the involvement can be any of these: information exchange, development of tools (books, guidelines, training), advice and expertise, recommendations, legal texts and instruments, direct or indirect involvement in decision making as well as in activities related to authorization and control, prevention of incidents and accidents, preparedness, and response.

The WG agrees with the recommendations concerning stakeholder involvement formulated during the 10th European ALARA Network Workshop:

"Regulatory bodies have an important role to play in facilitating stakeholder involvement, and are encouraged to establish mechanisms for communicating with relevant parties and encouraging their participation. This may for example include seminars, consultation exercises, public meetings, internet forums, etc."

- Stakeholders should be consulted as widely as possible, whenever acceptable (there may be security related restrictions for some types of interactions) and manageable. Authorities should try to be clear on the purpose of the involvement and on the output expected from the interaction (transparency).
- Authorities should try whenever possible to build structural mechanisms for consultation with stakeholders.

One could find several kinds of roles for radiation protection authority when working with stakeholders and getting their involvement to reach an appropriate level of safety. Roles could be for example:

1. Forerunner

The national authorities competent in the field of in radiation protection should initiate, co-ordinate and monitor safety related research and development. The results of the research will support its regulatory functions and also stakeholders in using justified and optimized methods and equipment and to apply and adopt optimized work procedures.

The Radiation protection authority could promote research projects where stakeholders are partners in trying to achieve common goal. The authority could suggest ideas for topics to be investigated and initiate and motivate research projects which will be performed by research groups, universities and research institutions as well. Also small scale studies together with educational institutes and students are valuable.

Especially in the medical sector the opinion leaders are in universities, major hospitals, and educational institutes and in specialized research institutes. Co-operation with these stakeholders will provide a solid base for further work in radiation protection and will grant credibility for radiation protection authority in further actions.

Specialists of the Radiation protection authority should participate in international congresses to get understanding of the newest applications of ionizing radiation.

One has to investigate the needs of education and training together with the users of radiation in discussions and meetings with different kind of specialists and organizations. Surveys together with educational organizations of the needs are valuable as well. Based on this information Radiation protection authority could propose further education for radiation users at universities and training institutions.

2. Rule maker

According to IAEA GR-S-1 "In order to fulfil its statutory obligations, the Regulatory authority shall define policies, safety principles and associated criteria as a basis for regulatory actions". To get commitment from stakeholders for the regulation it is useful to prepare basic regulations using consultation with stakeholders. Sometimes pre-stage in preparing regulation is to perform preliminary study or research project to get better understanding of the status quo. Here you could involve some key stakeholders in the work. The results of the projects could serve as basic information for the foundation of the regulation.

Many times it is practical to invite experts from professional societies to join in to the working groups to prepare specific regulatory documents. Furthermore it is important to call for comments from all involved stakeholders on the drafts of the documents and take these opinions into account in the final version, if appropriate from the point of view of radiation protection.

3. Advisor

Radiation protection authority shall provide guidance to the operator on developing and presenting safety assessments or any other required safety related information. This is according to the IAEA GS-R-1. It is important to make surveys which kind of guidance is needed by license holders and specialists in the field. This is also to get a better mutual understanding of the common goals in radiation protection and to prepare proper guidance. When preparing such guidance and information it is useful to prepare guidance together with the specialists working in actual practices.

4. Communicator

Radiation protection authority shall introduce the regulation and guides for all radiation users (organizations, experts) and involved parties. All radiation users should have information on new regulations and guides soonest possible. Good practice is to mail this information based on the registers of the stakeholders if any. Good service for stake-holders is to provide all kind of general information on radiation protection for stakeholders on the www-pages of the authority.

An effective method is to organize regular discussions and meetings on current topics with different specialists and organizations.

How to communicate risk is a particular challenge, it should be prepared with other stake holders like medical doctor organizations, hospital directors, patient organizations, press bodies. Communicate the risk in advance (booklet), communicate the risk after an incident/accident (info unit important). In some cases (e.g. contamination, permanent source implants), communication should not be limited to explaining what the risk is, but should allow people to contribute to radiation protection (how to behave in order to limit exposure of oneself and/or others and/or environment).

5. Supervisor

Radiation protection authority should have high reputation in its work. To reach this goal authority should have competent experts in all applications of radiation practices with good understanding of radiation protection. The authority should inform all stakeholders of its activities transparently to get a well known status among users of radiation. When recognized status is reached the supervisory work will be more effective and stakeholders will be cooperative to reach optimal radiation protection.

6. Connection to other Health authorities

Radiation protection authority should have a strong connection to health authorities; they are a very important stakeholder. Health authorities obviously have responsibilities for all aspects of health care provision; while the radiation protection authorities have selected duties. There are many different national approaches. In some countries the health authorities are giving expert advice on radiobiology, where in others the competence is within the radiation protection authority. In some countries the responsibility for staff protection and patient protection may be in different authorities.

Where is the justification question set in the national regulatory framework; In the RP legislation, or in a more generic legislation? For example, are other health authorities involved in the justification question? For example breast screening program in Finland, the justification question is judged in a medical – ethical trial.

Examples of areas where other health authorities are involved in addition to the radiation protection authority:

- Justification
- Patient safety
- How patient data are stored, frequency of examinations and treatments, dose data, and other information needed for surveys
- Financing
- Following up of epidemiological data
- International standards

7. Network creator

Information transfer between stakeholders is an important part of a learning process in radiation protection. Users of radiation can learn good practices from each other and also learn from the mistakes of other stakeholders. Radiation protection authority could activate this information transfer by creating discussion forums for example in regular meetings with target groups. Discussion forums on web pages might be useful as well. Data banks of incidents and accidents could be created and made available through the network.

8. Trainer

Radiation protection authority could act as a trainer of trainers on special topics. In this meaning one could educate and train some opinion leaders of stakeholders who can distribute the information on radiation effects, radiation protection and regulation and guides among other specialists.

9. Role to verify and improve radiation protection

Radiation protection authority should verify from time to time the situation at the particular point of time for example by making surveys with stakeholder groups on special topics related to radiation protection. The results of such verification offer useful material to create fruitful discussion with stakeholders in the meetings to improve the process of radiation protection.

Conclusions

The progressively but steadily increasing exposure to ionizing radiation associated with medical applications should be specifically addressed. The lack of awareness of the general public and some medical professions in relation to medical exposures needs to be addressed. Consultations with all stakeholders (medical devices manufacturers, medical staff, hospital managers, structures involved in the organization of national health systems, patients) should be organized in order to address this complex problem and try to find appropriate solutions.

Concerning medical devices, consultations with manufacturers and suppliers should be established to address the radiation protection issue directly at the source. Additionally, a discussion (on a national or international level), allowing better understanding and more transparency concerning the EC specifications for medical devices, should be initiated with the responsible EC structure.

Consultations concerning the notification of medical events and conditions for reporting those events to the medical community and the public should be further developed (medical staff, hospital managers, structures involved in the organization of national health systems, patients).

Health detriment and radiation protection management

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Abstract

Introduced in 1977 by ICRP, the concept of health detriment is a complex construction based on scientific information and on expert judgement. It gives an estimate of the total harm to health to individuals and their descendants as a result of an exposure to radiation, assuming a linear-non-threshold dose-effect relationship. It allows to compare risks induced by different types of radiation exposure situations and to put into perspective with other health risks.

Understanding its meaning is not straightforward, even for experts. It is in fact necessary to analyse its components and associated value judgements, and their evolution to catch this meaning. In addition, the application of this concept for different exposure situations is still a matter of debates in the radiation protection community. For instance: Does it represent a potential number of health effects for an exposed population? Does it give an estimate of the probability of occurrence of a radiation-induced effect for an individual?

The aim of this presentation is to recall the basis of this concept and its evolution and to discuss the key stakes regarding its usefulness in the radiation protection system. For instance, for occupational exposures, although the radiation detriment is no more the central indicator for establishing exposure limits, it is still quite useful to compare the risk associated with these limits with the levels of risk associated with exposure to different carcinogens and with occupational injuries. In addition, it is a key element for addressing the probability of causation for occupational diseases associated with ionizing radiation. Its use is also a matter of debate regarding radioactive waste management or environmental exposures, for which the main concern refers to exposure of future generations and large populations. Nevertheless, this concept is useful in this domain for assessing the level of protection on the basis of the current radiological protection criteria.

Norwegian strategy to fulfil the OSPAR Radioactive Substances Strategy objectives

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Abstract

At the 1998 Ministerial meeting of the OSPAR Commission, the Contracting Parties agreed to a strategy with regard to radioactive substances, the OSPAR Radioactive Substances Strategy (RSS). It states that the ultimate aim is: substantial and progressive reductions in discharges and to achieve concentrations in the environment near background values for naturally occurring radioactive substances and close to zero for artificial radioactive substances by 2020. Several of the Contracting Parties are far from meeting the 2020 objective. The oil and gas industry represent a substantial part of the TENORM-discharges to the North-East-Atlantic. A substantial part of the discharges come from the Norwegian sector. The Norwegian Radiation Protection Authority (NRPA) regulates the oil and gas industry on the Norwegian sector. NRPA uses a variety of means to contribute to fulfil the RSS objectives by 2020. NRPA collaborate with the competent authorities from the other oil and gas producing countries in the North Sea area to work towards a more harmonized regulation. The Norwegian zero discharge goal for the oil and gas industries has been extended to contain radioactivity, it is also recommended to perform a new assessment of injection of produced water on the two offshore platforms with the highest discharges of radioactive substances. We work towards a strengthened regulation of radioactive waste and discharges by a closer collaboration with the Norwegian Climate and Pollution Agency (former Norwegian Pollution Control Authority). As part of this work the legislation is currently under revision, and a new regulation based on the Pollution Control act is proposed. It is important to continue and develop the national monitoring programme of the industry, as well as investigate possible cleaning technologies and initiate cooperation between industry, regulators and researchers. Norway has at present special focus on TENORM. One of the first repositories for final storage of TENORM waste from the oil and gas industry was commissioned in 2008.

Introduction

Radioactive substances are used or found in several research institutions, hospitals and industries in Norway and on the Norwegian Continental Shelf. Some of these are well known such as the oil and gas industry on the Continental Shelf and in the research reactors at Kjeller and in Halden, the latest is part of an OECD project called OECD

Halden Reactor Proctect, as well as the use in medical diagnostic and therapy in hospitals. In addition NRPA has started a mapping of other industries where there can be discharges of radioactive substances with waste water or generating of radioactive waste of any category as part of the production. The discharges from the oil and gas installations on the Continental Shelf are far the largest radioactive discharges in Norway.

OSPAR

The OSPAR Convention for the Protection of the Marine Environment of the North East Atlantic was agreed in 1992. Countries that either have a North East Atlantic coast or discharge into the OSPAR maritime area via their rivers are Contracting Parties to the Convention. The North Sea including Skagerak, the Norwegian Sea and the Barents Sea are all within the area covered by the Convention. Totally 15 countries and the EU commission have ratified the Convention.

At the 1998 Ministerial meeting of the OSPAR Commission, the Contracting Parties agreed a strategy with regard to radioactive substances (see Box 1).

Box 1: OSPAR Radioactive Substances Strategy (RSS)*

Overall objective:
To prevent pollution of the maritime area, as defined under the Convention, from ionising radiation, through progressive and substantial reductions of discharges, emissions and losses of radioactive substances. The ultimate aim is to achieve concentrations in the environment near background values for naturally accruing radioactive substances and close to zero for artificial radioactive substances. In achieving this objective, the following issues should, inter alia, be taken into account:

- legitimate uses of the sea
- technical feasibility
- radiological impacts to man and biota

Intermediate objective (2020)
By the year 2020, the OSPAR Commission will ensure that the discharges, emissions and losses of radioactive substances are reduced to levels where the additional concentrations in the marine environment above historical levels, resulting from such discharges, emissions and losses, are close to zero.

* Radioactive Substance Strategy of the OSPAR Commission for the Protection of the Marine Environment of the North East Atlantic, 1998.

Norwegian Environmental Policy and Goals

In the Report No 21 (2004 – 2005) to the Storting "The Government's Environmental Policy and the State of the Environment in Norway" one of the national goals says that discharges of radioactive materials from Norwegian sources shall be limited to levels that do not have an adverse effect on the natural environment.

The Report No 8 (2005 – 2006) to the Storting "Integrated Management of the Marine Environment of the Barents Sea and the Sea Areas off Lofoten Islands" states that the targets for limiting inputs and concentrations of radioactive substances in the Barents-Lofoten area are.

> *The environmental concentrations of hazardous and radioactive substances will not exceed the background levels for naturally occurring substances and will be close to zero for man made synthetic substances. Releases and inputs of hazardous or radioactive substances from activity in the area will not cause these levels to exceed.*

In the "Integrated Management of the Marine Environment of the Norwegian Sea", Report No 37 (2008 – 2009) to the Storting is an analogous target is expressed.

With regard to radioactive substances are the goal expressed in these two Reports to the Storting more or less identical with the objective of the OSPAR Strategy. In the report to the Storting 26 (2006 – 2007) "The Government's Environmental Policy and the State of the Environment in Norway" it is referred to the Norwegian obligation to OSPAR, expressed as.

> *As a member of OSPAR Norway have the obligation to prevent pollution of the maritime area from ionising radiation through progressive and substantial reductions of discharges. The goal is that the level of naturally occurring radioactive substances shall be close to the background levels. Norway shall as part in OSPAR before 2020 secure that the discharges of radioactive substances be reduced to levels that contribute to concentrations in the environment beyond historical levels as a result of such discharges are near to zero.*

In the same Report to the Storting the Norwegian government also states that there should be a mapping of discharges, supplies and levels of radioactive materials in the Norwegian Coastal Current and the nearest maritime areas from both oil and gas activities and other sources, and that a need for strengthening of the knowledge of the consequences of the discharges were needed.

Norwegians efforts to reducing the discharges

Acts and regulations

The Norwegian radiation Protection Authority (NRPA) is the competent national authority in the field of radiation protection and nuclear safety in Norway. The NRPA administers two acts along with associated regulations:

- Act of Nuclear Energy Activities (No. 28 of 12 May 1972)
- Act of Radiation Protection and Use of Radiation (No. 36 of 12 May 2000)

- Regulations No.568 of 9 May 2003 on Application of the Act on Radiation Protection and Use of Radiation on Svalbard and Jan Mayen
- Regulation No 1362 of 21 November on Radiation Protection and Use of Radiation (Radiation Protection Regulations)

In addition the NRPA's preparedness mandate is laid down in a Royal Decree of 17 February 2006

When the Norwegian radiation protection legislation was revised at the end of the 1990s, resulting in the Radiation Protection Act of 2000, it was built to a large extent on the prevailing international recommendations. The main focus of radiation protection has traditionally been the protection of man from harmful radiation doses, which also was the main focus in the Radiation Protection Law. However, the Radiation Protection Act also embodied increased environmental awareness of radiation by including the environment in the purpose of the law alongside the goal of preventing the harmful effects of radiation on human health. The Radiation Protection Regulations, adopted pursuant to the Radiation Protection Act, went into effect in 2004. It has special provisions concerning radioactive emissions and waste. Even though the regulations are limited it includes requirements for approval of emissions of radioactive substances, waste facilities, and the export and import of radioactive waste. The Ministry of the Environment was authorised to function as the administrative appeal body under these provisions in 2006.

In recent years, an increased awareness of radiation as an environmental problem has gradually grown both in the international co-operation in the radiation area and in Norway. With respect to the relation between humans and environmental protection, the presumption has long been that as long as people are protected, then the environment is protected. An expression of this prevailing perception is given in the recommendations from International organisations like the Commission on Radiological Protection (ICRP), the International Atomic Energy Agency (IAEA) and the Nuclear Energy Agency (NEA). Corresponding perceptions are also found in EU documents.

As a part of the work of strengthening the radiation protection administration for the external environment, the Norwegian Radiation Protection Authority, Climate and Pollution Agency, the Ministry of Health and Care Services and the Ministry of the Environment have performed a review of the challenges of regulation of radioactive substances in the environment and the roles of the current bodies of regulations. There is a need for a more comprehensive set of regulations than the present regulations on protection of the environment. To meet these challenges a new radiation protection regulation is put forward. The regulation will be under and pursuance of Pollution Control Act. This regulation will result in a regulation of radioactive pollution will be based upon fundamental environmental principles such as prevention and the precautionary principle, the polluter pays and the best possible technologies. This new regulation is proposed to be put into force 1 January 2011.

The zero discharge project

The objective of zero discharges to sea of potentially environmentally hazardous substances from the petroleum activities was established in the Report No. 58 (1997 – 98) to the Storting "Environmental policy for a sustainable development". Based on the petroleum activities' substantial and increasing discharges to sea, the authorities identified a need to formulate a strategic, general objective that could contribute to reducing the discharges beyond that which followed from national and international objectives for reduction of oil and chemical discharges

In the Report No.26 (2006-2007) to the Storting the Norwegian Government announced that in 2009 evaluation of the progress of this project and whether further measures were needed to ensure that the zero discharge targets were achieved for oil and naturally-occurring substances discharged with produced water from the offshore petroleum industry. The Government also wanted to evaluate the need of efforts to reduce the discharges of naturally occurring radioactive materials (TENORM) from the petroleum industry.

To establish a basis to answer this questions the Climate and Pollution Agency and the Norwegian Petroleum Directorate were asked by the Norwegian Ministry of Environment and the Norwegian Ministry of Oil and Energy in 2008 to an evaluation of the above mentioned tasks. This was carried out in cooperation with the Norwegian Radiation Protection Authority. Data prognosis of discharges from the oil and gas installations on the Norwegian Continental shelf were collected from the companies together with estimates of costs for injection of produced water as an alternative to discharges. In addition an evaluation of technical ways to accomplish the injection and The influence of the marine environment of the discharges with special focus on the most vulnerably areas in the North Sea and the Norwegian Sea was also considered together with an evaluation of technical challenges of injection of produced water.

So far no harmful influence on living species in the marine environment can directly be related to discharges of radioactive substances with produced water on the Norwegian Continental Shelf. There is a need for more knowledge of long time effects. It is, however, known that radioactive substances have similar properties and potential environment-related harmful effects such as heavy metals and other environmental toxins.

The concentrations of radioactive substances in produced water can vary considerably between the field and these together with the great variation in the volume of produced water discharged will result in a great variation of the amount of radioactive materials discharged from the fields. The two installations Troll B and Troll C in the northern part of the North Sea are among the installations with the highest discharges of produced water. There is also relative high concentration of radioactive substances in the water discharged from these installations resulting in large discharge of radioactive materials from this field, see figure 1. The discharged radioactive substances from the Troll field count for about 40 % of the total Norwegian discharges.

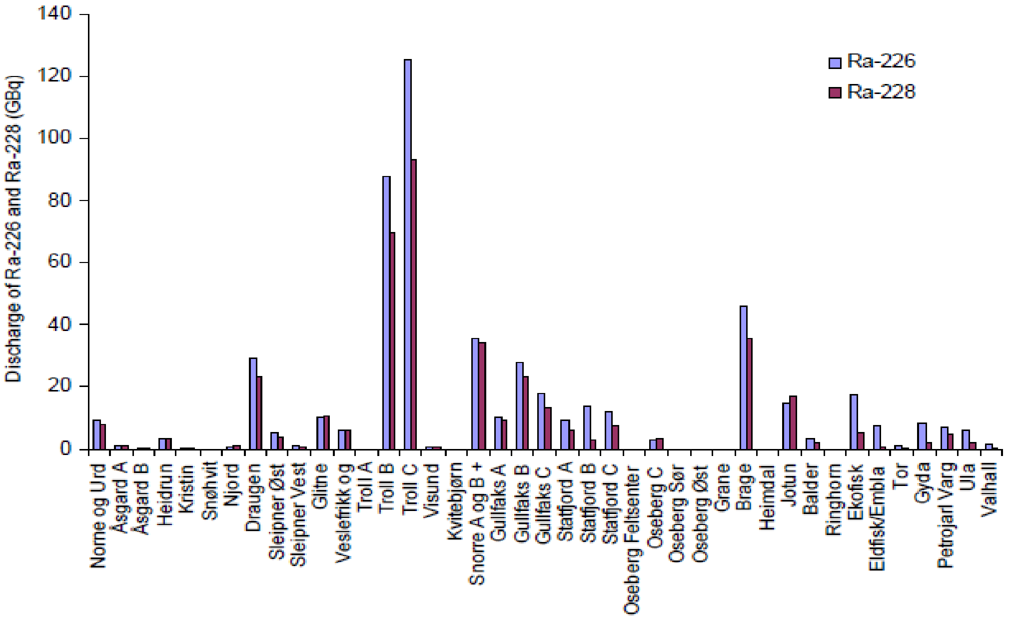

Fig. 1. Estimated discharged activity of ^{226}Ra and ^{228}Ra from Norwegian oil and gas fields in 2007 (NRPA 2009).

Based on the above mentioned evaluations two advices regarding radioactivity in the zero discharge project were given to the Ministry of Environment:

- to include radioactivity (TENORM) in the zero discharges target
- carry out new evaluations of the possibility of injection of produced water from the Troll B and C platforms due to the large amount discharged

Reporting

All activities that hold a permit to discharge radioactive substances to the environment have an obligation to report on the discharges from the activities every year. The reports should include information of the complete discharges of the various nuclides that have been discharges and also the concentration of the nuclides. For land based activities the recipient to which the discharges take place be described too. A description of how the use of best available technology is implemented as well as treatment of radioactive materials before discharges, to reduce the concentration of radioactive substances, should also be included.

Monitoring

National Programme

The Norwegian Radiation Protection Authority has since 1999 had a program for monitoring of radioactivity in the marine environment, both in coastal areas and the open sea (Radioactivity in the Marine Environment, RAME). Samples of seawater, sediments and biota are collected every third year in the North Sea including Skagerak, The Norwegian Sea and the Barents Sea. That is samples are collected from one of the areas each year. In addition samples are collected at permanent coastal stations, at Bear Island, Hopen and Svalbard, and in selected Norwegian fjords. The samples are analysed for both naturally occurring radioactive substances like radium and for some dedicated substances from earlier tests of nuclear weapons in the 1950ies and –60ies, from the accident of Chernobyl and discharges from the reprocessing facility at Sellafield in UK and Cap de La Hauge in France.

In addition the Climate and Pollution Agency is coordinating a task to establish a long-term monitoring programme for the Norwegian maritime areas. The main purpose of this programme is:

- Calculation of the entry of environmental harmful substances including radioactive materials from various sources, divided in up to 12 regions
- Calculation of transport and concentration of these substances in the marine area
- Monitoring of environmental harmful substances in biota and sediments

Monitoring around oil and gas installations on the Norwegian Continental Shelf

The oil and gas companies operating on the Norwegian Continental Shelf have an obligation to carry out monitoring of the water column and the sediments in the neighbourhood of their installations. This monitoring shall give an overview of the environmental conditions and map the development and trends in the environment around the oil and gas installations. The Norwegian Shelf is divided in several regions, see figure 2. Surveys of taking sediment samples is carried out in each region every third year. For the water column the monitoring consist of two parts; condition monitoring in which samples of fish are analysed every third year, and the other part which is being developed is an effect study and the frequency of monitoring near by the different installations is not yet decided. In addition to demonstrate trends the results are the basis for expectations for future developments. In the later years the monitoring programme has included analyses of sediment samples for radioactivity for a few stations.

A process to develop the monitoring of radioactivity around the oil and gas installations has to been started and for 2010 the number of stations where sediment samples will be taken is expanded compared with the previous years.

The results from the monitoring will, among other things, be used for:

- evaluation of the risk for environmental damage and ecological effects
- verification of models for calculating the environmental risk as a function of existing and expected discharge from the offshore industry.
- verification of laboratory based research

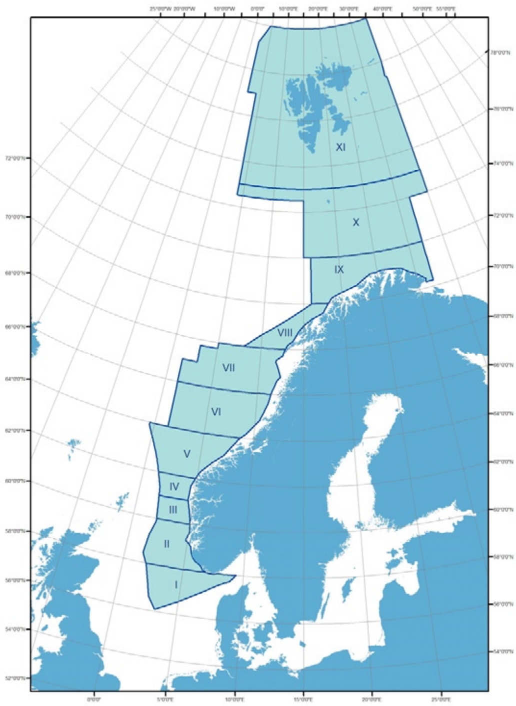

Fig 2. Overview of regions for monitoring of sediments on the Norwegian Continental Shelf.

Regulation of discharges

Generally

All approvals/permissions issued to companies who discharge radioactive substances instruct the companies to examine all their procedures and routines carefully to identify any measure to reduce their discharges of radioactive substances. The results of such an examination should be reported to NRPA within a certain deadline.

NRPA should carry out a total review of all approvals/permissions to secure that the discharges limits are in according to the existing discharges and that they does not open up for significant increase in the future.

If relevant, request for environmental monitoring is a part of the approvals/ permissions, and the results of this monitoring shall be reported to NRPA within a certain deadline.

All approvals/permissions to discharges should contain requests for use of best available technology, to avoid discharges if possible or keep the discharges as low as possible.

Research nuclear reactors

The company responsible for the two Norwegian research reactors (IFE) should be requested to carry out risk analysis of the discharges for both the facility in Halden and at Kjeller. As a part of new discharge permits IFE has to evaluate potential measures to reduce the discharges and by that reduce the radiation doses to the public as well as the environment. In a new discharge permit discharge limits based on the activity of the various nuclides has to be established. As the only Norwegian company IFE have discharge limits given in dose limits, other companies have discharge limits given in nuclide activity.

Oil and gas industry

New fields shall, as a general rule, be developed with the objective of zero discharges of produced water containing radioactive substances under normal operations. For new fields that are developed with a tie in to existing installations discharges of produced water should only be allowed under normal operation if it is needed of serious safety, technical economic or environmental reasons.

On fields in production where injection of produced water is a existing solution it is important that the amount of produced water is kept at the prevailing level or is increased if there is no serious technical, safety, environmental or economical reasons for reducing or stop the injection.

Development of technology for reduce the content of radioactivity in the produced water has to be carried out in a cooperation of operating companies on the Norwegian Continental Shelf, the suppliers of purify technology and research institutions.

Hospitals, universities and research institutions

A joint evaluation of all discharges of radioactive substances from these kinds of institutions to map doses to people and the environment should be carried out. Based on the results of such an evaluation the requirement of further measures to reduce the discharges should be developed.

Waste management

All waste management companies should have a permit with requirements that contribute to as low discharges or other kind of radioactive pollution from the activities as possible. The new regulation mentioned earlier describes procedures for how to handle waste containing radioactive materials with various content of activity. For radioactive waste with origin offshore that have concentrations above certain limits should be sent to a special repository for final disposal. Also for waste with lower activity there are requirements for handling and final disposal.

Mapping of discharge from other sources

NRPA has started several projects to map any discharges or other kinds of radioactive pollution from mines other kinds of industry which may lead to radioactive pollution These projects form a basis for further development of regulations of radioactive waste management and discharge.

Auditing

Inspections and auditing are important to verify that the companies or other institutions that have approvals/permits for discharges or handling radioactive waste from NRPA are operating in conformity with the regulations and permits. These activities should be carried out in accordance to a yearly established plan.

References

NRPA, 2009. Radioactivity in the Marine Environment 2007. Results from the Norwegian National Monitoring Programme (RAME). StrålevernRapport 2009:15. Østerås: Norwegian Radiation Protection Authority.

The Integrated Management System – to ensure an overall safety

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Abstract

GNS (**G**esellschaft für **N**uklear-**S**ervice mbH) as a subsidiary company of the German utilities E.ON, RWE, EnBW and Vattenfall is charged with ensuring the radioactive waste management for their nuclear facilities. Moreover GNS also provides its products and services to foreign customers.

"Safety first" – this is the key note of our vision to ensure sustainable radioactive waste management, meaning first of all the protection of people and the environment.

For this reason GNS has implemented an integrated management system strictly based on the requirements of the IAEA Safety Requirements GS-R-3 "The Management System for Facilities and Activities" /1/ in early 2009.

The management system integrates safety, health, environment protection, security, quality and also economic elements.

Our company policy including the respective guidelines reflects the top-ranking of a strong safety culture within our management system.

An adequate organisational structure and the process organisation needed to fulfil the purpose of our company have been implemented always keeping in mind the meeting of the safety requirements. Continuous improvement of the organisation and processes is guaranteed, especially regarding the further development of our safety culture.

The paper presented in this abstract is to demonstrate in which way an integrated management system, strictly following the requirements of the above mentioned IAEA Safety Requirements, has been developed and implemented. The structure and the daily working of such an integrated system will be illustrated, benefits and advantages will be presented as well as the experiences gained up to the moment. But also challenges that appeared when developing such an integrated system with the emphasis on "safety first" will be addressed.

Introduction

GNS (**G**esellschaft für **N**uklear-**S**ervice mbH) as a subsidiary company of the German utilities E.ON, RWE, EnBW and Vattenfall is charged with ensuring the radioactive waste management for their nuclear facilities. The activities range from the planning, design, manufacturing and services in the field of radioactive waste and nuclear casks to the handling, transport and interim storage of flasks/casks for irradiated fuel as well as the decommissioning/dismantling of nuclear plants or the preparation of final storage. Moreover GNS also provides its products and services to foreign customers.

All these activities are governed by GNS' vision
"as a centre of expertise of the German energy supply companies, to sustainably ensure nuclear waste management while maintaining the highest safety standards at competitive terms".

This vision has been specified by the company's policy taking into account nuclear safety and radiation protection, health and safety, quality and environment factors as well as by the related principles, the very first being "we are aware of our responsibility for safe and reliable waste management". It is supported by the "Code of Conduct" accepted by the management and the employees.

Irrespective of this voluntary commitment GNS also has to satisfy legal requirements, technical rules and standards, requirements defined in licences and approvals all intending to keep safety objectives as e. g. shielding, leak tightness, under criticality, prevention of release of radioactive substances and so to protect men and environment from adverse effects of radioactivity.

But – how to meet these requirements in day-to-day business? Could an adapted management system help? And if yes, what type of management system should it be?

Material and methods

Why an Integrated Management System?

An adapted management system has to provide an instrument to achieve a company's vision and objectives. It defines the organisational structure as well as the process organisation needed to fulfil the company's tasks and the respective requirements. It has to conform to the specific and accepted management rules and standards. It has to integrate already existing procedures and systems or modify them, where necessary, as the already existing quality management system and environmental system, both being accepted for a long time by third parties. And it has to be suitable for day-to-day business.

So GNS decided for the implementation of a – process related - **I**ntegrated **M**anagement **S**ystem (IMS) strictly following the requirements of GS-R-3 /1/ taking into account the requirements of radiation and environmental protection, occupational health and safety, quality control and cost-effectiveness in the planning, design and management of projects and processes. The IMS "covers all specifications, regulations and organisational aids that are provided within the company to manage the relevant tasks for the success of the company under controlled conditions and to control and improve the achievement of objectives" (KTA 1402 /2/). The IMS focuses on the processes that are important to the company.

But – which are the processes important to the company?

Process identification

In co-operation with the management board GNS processes have been identified, the ones classified as being the most important to the company's success as defined in the vision and the policy have then be divided up into management processes, main processes, supporting processes and "other processes".

Management processes are those necessary for the overall regulation of the company and are primarily dedicated directly to the Management Board.

The main processes cover all activities to fulfil the essential tasks in relation to the customers and to a decisive extent determine the economic performance of GNS and the importance of the company for the shareholders in accordance to GNS' vision and policy.

Supporting processes are necessary to achieve the goals of the main processes. The supporting processes provide the prerequisites for the working of the main processes or define procedures which apply to all processes in the same way.

"Other processes" focus on procedures and workflows for other overall elements of the IMS.

Organisation structure to ensure an Integrated Management System

The organisation structure of GNS always follows the essential tasks in relation to the customers and so the main processes. Where necessary the organisation structure has been adapted to other relevant processes within the context of establishing the IMS and therefore focussing on the different interested parties. Persons with special responsibility for elements or processes within the IMS have been authorised.

Documentation of the Integrated Management System

The structure and the processing of the IMS in principle have been described in the IMS Manual /3/. Details are fixed in supplementary documentation.

Results

The processes identified as the most important are shown in figure 1.

The requirements originating from the elements of the IMS (as there are radiation protection and nuclear safety, environmental protection, occupational health and safety, quality and cost-effectiveness) as well as the requirements and the satisfaction of the interested parties (customers as well as shareholders, authorities and experts, employees, suppliers and sub-contractors, members of society) basically determine GNS' processes.

The structure of the processes shown in figure 1 are described in detail in further process visualisations and completed by descriptions or instructions, where necessary. Stipulations are documented to guarantee the keeping of the requirements originating from the different aspects of the IMS. All documents are made available to all employees by an adequate database. The structure of the IMS documentation is shown in figure 2. This figure also demonstrates the correlation between the company's policy or external influences (legal requirements, licences, approvals etc.) and the management system of GNS.

The IMS as means to keep the company's policy and principles has been discussed and communicated to the management and the employees.

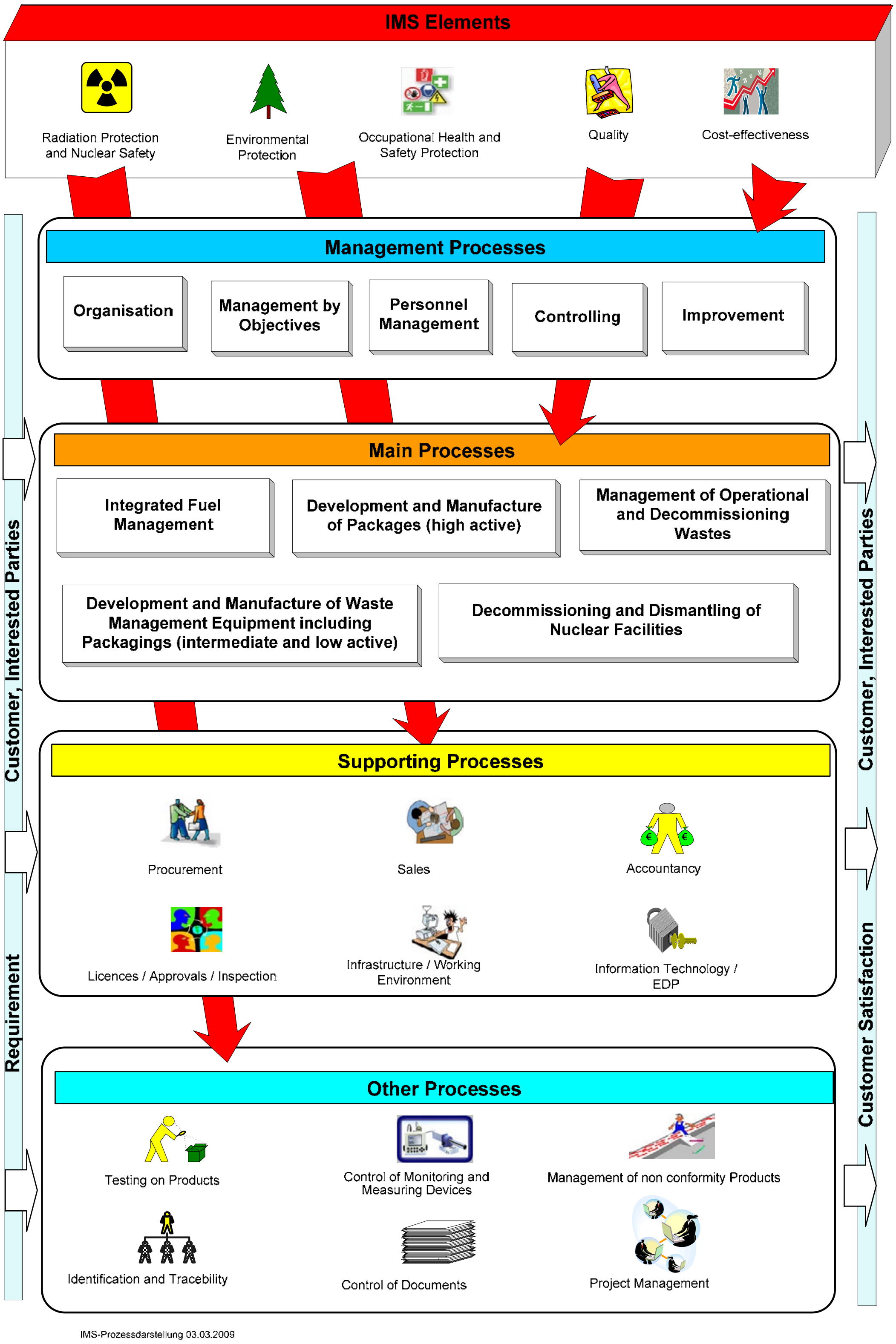

Fig. 1. Process Model of the Integrated Management System.

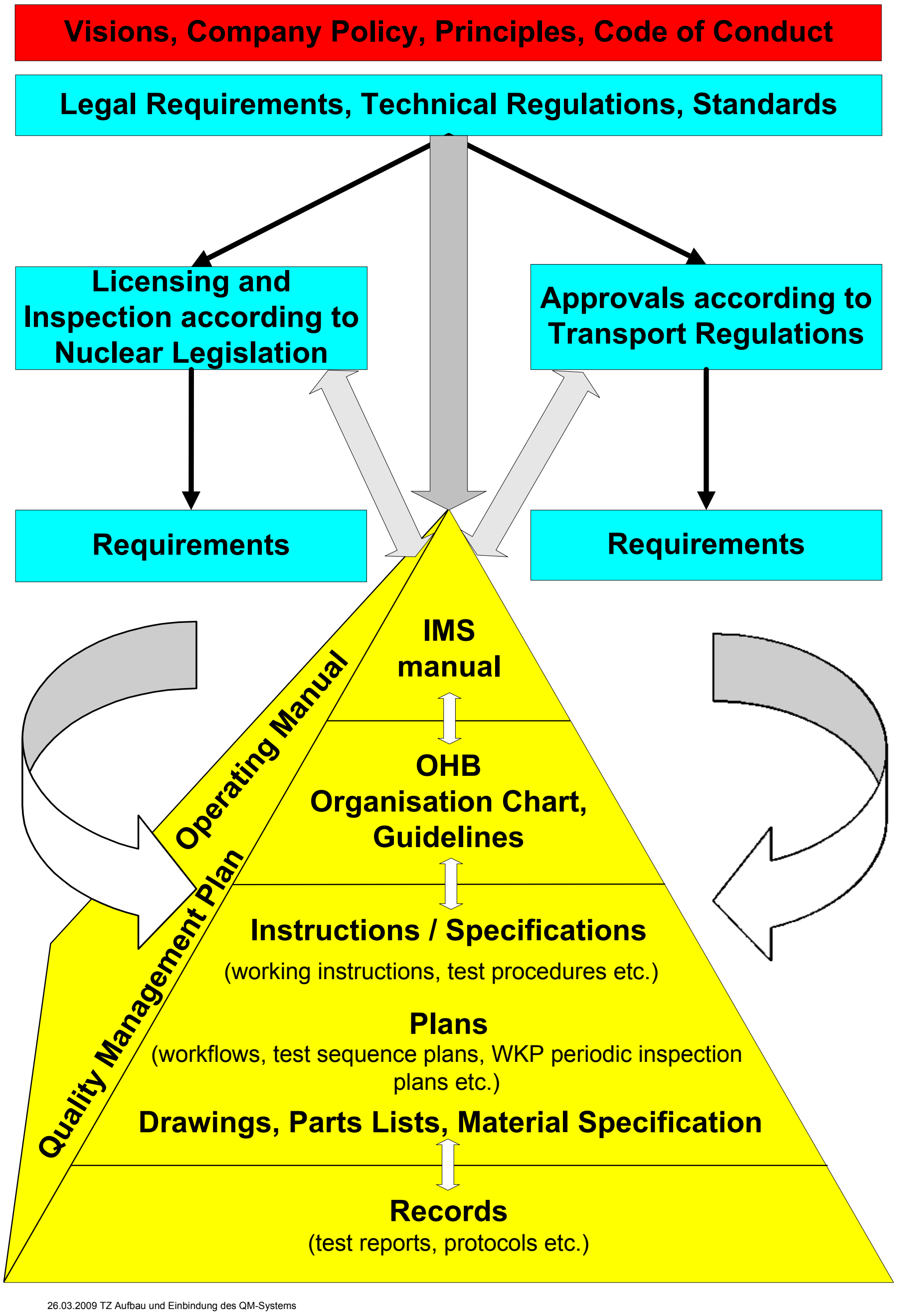

Fig. 2. Structure and Integration of the Management System.

Discussion

What did change regarding the day-to-day business when introducing the integrated management system as presented before?

The IMS is the result of an appropriate combination and continuous development of already existing systems (the quality or environment management system). The principles of the functioning of these systems, proven for a long time, have been adopted and, where necessary, modified according to experiences made. Only the factors systematically to be taken into account when planning, designing and managing processes and projects have been extended according to the company's policy and principles. This fact has supported a high acceptance of the IMS by the management and the employees.

Benefits and advantages of an IMS are highly detectable. The integration of all relevant aspects into the internal procedures has made clear the correlations between them. It has led to a higher sensitivity of the employees for the different requirements and their responsibility for fulfilling them within the frame of their process and project activities. This has also led to a continuous assessment and improvement of the implemented processes and so the management system. Moreover all organisational structures within GNS that could be concerned by a project or a process are involved in the respective planning, designing and managing.

Systematically taking into account all factors that may influence a process, a product or a project also lead to a higher legal certainty with regard to the legal requirements, standards and rules to be kept.

But – how to decide in case of contradictory requirements that originate from the different factors, e. g. aspects of cost-effectiveness versus those of environment protection requirements? The interests have to be balanced, but safety shall never be compromised. The optimum solution has to be found while fulfilling the legal requirements or such coming from licences or approvals. Due to the long lasting experience and practise with operating nuclear facilities or performing activities with radioactive material this spirit was already there in the whole company and its deliberately expression in the IMS Manual was not new for the staff.

Conclusions

The implementation of an integrated management system focussing on the processes important for the company with regard to its policy and principles has been successful. It has been accepted by the management and the employees. Nevertheless continuous sensitisation for as well as discussion and assessment of the needs of such a system as well as of the single factors is required and realised. So it will be and is integral part of continuous improvement.

Regarding the parts of quality management and environment management the IMS has already been accepted as appropriate system by the certification body. The IMS has also been accepted as an adequate system to manage safety (as synonym for quality within the nuclear industry) by third parties as authorities.

References

/1/ GS-R-003; IAEA, Safety Requirements, Management System for Facilities and Activities, Vienna, 2006

/2/ KTA 1402 (Draft Safety Standard in Preparation); KTA, Management System for the Operation for Nuclear Facilities, November 2007

/3/ IMS-Manual – GNS, Manual on GNS' Integrated Management System, Rev. 0, April 2009

Possible implications of new Basic Safety Standards – a Swedish viewpoint

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Abstract

The 2007 Recommendations of the International Commission on Radiological Protection (ICRP) [1] formally replaced their earlier 1990 Recommendations. The International Atomic Energy Agency and the European Commission presently update and review their *Basic Safety Standards* (BSS) [2, 3] within their frameworks, jurisdiction and international roles. The Swedish regulatory system will be influenced and steered by these emerging documents, especially by the Euratom BSS since these are binding for the Member States. With the proposals existing as of spring 2010 as starting point, we offer our view on the possible implications the new Standards will have on the radiation safety regulations of Sweden.

Introduction

Both the *International BSS* and the *Euratom BSS Directive* relate to the health protection of patients, workers and the public from the dangers of ionising radiation. The standards apply to any planned, existing or emergency exposure situation which involves a risk from exposure to ionising radiation.

The new *Standards* reflect changes introduced in ICRP 2007 main recommendations:

- Updating the radiation detriment based on latest scientific information of the biology and radiation exposure,
- Moving from a process-based protection approach (practices, interventions) to a situation-based approach, basing the system on type of exposure situations (planned, emergency, existing),
- Emphasis on the use of optimisation and dose constraints/reference values in all exposure situations, and
- More focus on the protection of the environment, developing a framework to demonstrate the radiological protection of the environment.

The fundamental principles of radiological protection: *Justification, Optimisation, Application of dose limits*; and the Commission's *dose limits* for effective dose and equivalent dose from all regulated sources in planned exposure situations are kept.

Furthermore, the new *Standards* reflect the societal needs, experience feed-back, and value judgements which have led to inclusion or updates of:

- Requirements on national/regional education, training, and competence,
- Security issues and concerns influence threat analysis and the national emergency planning,
- The justification and treatment of non-medical imaging,
- Regulation of planned or existing exposure situations involving naturally occurring radioactive materials (NORM) and radon in dwellings and workplaces.

Influence on Swedish legislation and regulations

In the following a non-exhaustive treatment of the possible influence of the *Standards* in Sweden is given. Focus is on the Euratom BSS Directive since it is binding for the EU member states.

Exposure Situations

The use of the three exposure situations to identify and determine the appropriate protection needs will influence the present legislation. The control of *planned exposure situations* (includes *practices*) is since earlier well developed and justification and optimisation were required in the Swedish regulation. The new concepts with existing and emergency exposure situations with different approaches to justification and the use of reference levels in the optimisation of protection must be reflected in new legislation. The choice between planned exposure situations and existing exposure situations is not always obvious for all activities which must be taken into account.

Optimisation of the protection and safety in new exposure situations.

For *emergency exposure situations* and *existing exposure situations*, optimisation shall be used to enhance protection. In practice, existing regulations and procedures for emergency preparedness, management of radon exposure and others, classified as existing exposures, will be reviewed. For emergency preparedness this will involve the planning phase (*optimised protection options* related to different events) or decision-making in emergency situations (release of contaminated land, radiation doses to public and emergency workers, etc.). In *existing exposure situations*, protective measures shall be optimised and not merely result in exposure levels below an action value. *Reference values* will have to be developed if they do not already exist. For *existing exposure situations*, any changes in regulatory approach will in Sweden often involve several authorities.

Graded Approach

Especially the *Euratom BSS Directive* now put emphasis on *a graded approach*. The protective measures should be more commensurate with the actual risk estimates. Though for many practices the Swedish authority uses a graded approach in licensing and supervision, it is more of praxis than based on legal documents. This means that a larger diversity of possible legislative and regulatory means must be introduced. In Sweden, the radiation protection legislation will have to introduce the concepts of notification and registration, the existing options of licensing or not licensing are not sufficient. Furthermore, the actual requirements must be proportional; the regulatory body must develop regulations and/or license conditions with a larger degree of versatility. The need for prioritising of practices and sources suitable for registration is obvious and also to more frequently exempt activities or products from control. The proposed general clearance levels are expected to be complemented by authorised release from practices based on the criteria for clearance given in the BSS.

Education and Qualified Experts

Education and good knowledge of sciences and the existing radiation protection philosophy is needed in radiation protection work. Sweden has, as an example, good education in radiation physics. Most of these students become medical physicists and a high number of these are employed in Swedish hospitals but some are employed as qualified experts in the nuclear industry or research institutions. The SSM has issued *general advices* in its Code of Status on suitable educations and competence for operational staff at nuclear facilities and for qualified radiation protection experts. Regulations exist connected to education for the medical and the nuclear sector as well as in many more specific activities (e.g. radiography). However, a general overview of existing advice and regulations must be done and in some cases more specific requirements could be needed. The Euratom BSS directive is now more precise in defining the concepts and tasks of *radiation protection experts* and *radiation protection officers* and this must be accounted for and amendments in regulations introduced as necessary. Another factor that makes education and training important is the on-going generational change at the licensees as well as at the authority.

Regulation of NORM

What earlier was specified as *work practices* in the existing *Euratom BSS Directive*, will now, after the member states have performed national investigations of the needs for regulation and control, require regulation. In many areas, such as radon in residences and work places, Sweden is since many years prepared with regulations. In other areas, changes in legislation and regulations are foreseen. The SSM is for the moment preparing regulations on how NORM material emerging in constructional projects (roads, buildings, tunnels etc…) should be handled if the quantities are smaller than 100 tonnes per year. Further regulatory activities are to be expected; one issue is how the requirement on how optimisation under *reference levels,* is ensured for such *existing exposure situations*.

Justification

An issue of major importance is the justification of activities involving radiation protection. The justification process in the medical field (overall justification of methods and justification of the individual examination, qualifications and responsibilities of referring medical practitioner, radiological medical practitioner) is an unceasing source of importance. The justification of consumer products (e.g. use of smoke detectors) is also a relevant topic and an even more "hot topic" is the justification of non-medical imaging in relation with legal or insurance purposes, theft detection, security or anti-smuggling purposes. The two *Basic Safety Standards* have different approaches to these issues but the common denominator is that such activities, should they occur, must be justified and regulated according to national praxis. The *idea* is to allow for justified use of non-medical imaging however applying strict regulation. How to deal with these complex situations should be further analysed and dealt with, purpose for purpose, situation for situation.

Conclusions

Sweden is well prepared for implementing the new requirements as formulated in the international radiation protection standards, especially the Euratom BSS Directive. Nevertheless quite some work will be needed to review and amend the existing legislation and regulatory framework. Emphasis will have to be put on the new requirements of optimisation in all exposure situations, applying a graded approach and the requirements of education and radiation protection experts. Regarding NORM, some regulations exist but more is foreseen. Finally, we would like to underline that the issue of justification needs emphasising, both in medical as well as in non-medical imaging.

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IEC standards for measurement of environmental radiation

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Abstract
This paper presents the IEC/SC 45B "Radiation Protection Instrumentation" and its 15 standards for measurement of environmental radiation that have been published or that are under development or revision. Two types of standards are considered: general standards for environmental measurement instrumentation and standards for airborne radioactivity measurements. The first type covers gamma radiation ratemeters for environmental monitoring (IEC 61017-1&2), equipment for monitoring of radionuclides in liquid effluents and surface waters (IEC 60861), mobile instrumentation for the measurement of photon and neutron radiation in the environment (IEC 62438) and in-situ photon spectrometry systems using a germanium detector (IEC 61275). The second types concerns equipment for continuous monitoring of radioactivity in gaseous effluents (IEC 60761 series), monitoring equipment of atmospheric radioactive iodines (IEC 61171) and radioactive aerosols (IEC 6117) in the environment, radon compensation for radioactive aerosol monitors (IEC 61578) and equipment for monitoring radioactive noble gases (IEC 62302) and airborne tritium (IEC 62303).

Introduction
Different international organizations (ICRP, ICRU, IAEA, ISO, IEC…) develop standards, recommendations, reports or other international documents concerning radiation protection and particularly for measurement of environmental radiation. This paper will present IEC, its Sub Committee 45B "Radiation protection instrumentation" and the scopes of the 15 international standards that IEC/SC 45B has developed for measurement of environmental radiation.

IEC/TC 45/SC 45B
The International Electrotechnical Commission (IEC) was founded in 1906 and is the leading global organization that prepares and publishes international standards for all electrical, electronic and related technologies. These standards serve as basis for national standardization, as references when drafting international tenders and contracts, and for conformity evaluation of instrumentation.

A total of 76 countries are now participating in the IEC family (56 members and 20 affiliates from developing countries). Each country participates through its national committee.

There are 179 Technical Committees (TC) and Subcommittees (SC), and more than 700 working groups (WG) that carry out the standardization work for IEC. The working groups are composed of representatives from all over the world who are experts in their own field and who are members of research and testing laboratories, regulatory agencies, academia, manufacturers, and user organizations.

Technical Committee 45 "Nuclear instrumentation" addresses standard development for instrumentation specific to nuclear applications. Its Sub Committee 45B "Radiation protection instrumentation" covers all the fields of radiation protection instrumentation for measurements under normal and accident conditions of external and internal individual exposure, to workers, the public and in the workplace and environment.

SC 45B has more than 200 experts from 22 participating and 12 observer countries. There are currently 47 publications in force and 14 projects are in development. The SC has liaisons with IAEA, ISO, ICRP and other organizations. Its standards are considered by CENELEC for adoption as European ones.

SC 45B currently has the following active working groups:

- B5: Measurement of environmental radiation;
- B8: Pocket active electronic dose equivalent and dose equivalent rate monitors;
- B9: Installed equipment for radiation and activity monitoring in nuclear facilities;
- B10: Radon and radon decay products measuring instruments;
- B14: Passive integrating dosimetry systems for monitoring of external radiation;
- B15: Illicit trafficking control instrumentation;

SC 45B, through its WGs B5 and B13, was charged with the development of international standards for environmental measurement instrumentation. WG B13 "Measurements of Airborne Radioactivity" was recently disbanded and its activity was transferred to WG B5.

General standards for environmental measurement instrumentation

The general standards for measurement of environmental radiation are developed within WG B5.

IEC 61017 "Portable, transportable or installed X or gamma radiation ratemeters for environmental monitoring"

The first standard to be considered is IEC 61017 "Portable, transportable or installed X or gamma radiation ratemeters for environmental monitoring" published in 1991. This standard has two parts: part 1 "Ratemeters" and Part 2 "Integrating assemblies". Being rather old, a revision of this standard is expected to start in 2010. Both parts will be combined into one standard covering transportable and installed assemblies. Portable (rate) meters are already covered by IEC 60846-1 (2009) developed within WG B8.

IEC 61017 is applicable to portable, transportable or installed assemblies intended to measure environmental air kerma rates from 30 nGy h^{-1} to 10 μGy h^{-1} due to X or gamma radiation of energy between at least 50 keV and 1.5 MeV. If the assembly is to be used to measure air kerma rates in the area surrounding a nuclear reactor producing 6 MeV photons it will be necessary to determine the response at that energy.

For the purpose of radiation protection these assemblies comprise at least a detection sub-assembly (e.g., ionization chamber, GM counter tube, scintillation detector, etc.), a measuring sub-assembly including a display device, which may be connected together either rigidly or by means of a flexible cable or incorporated into a single assembly. The installed assembly may also comprise a continuous recorder (e.g., chart or magnetic cassette recorder or telemetry equipment). The requirements of this standard are also applicable to assemblies that use integration of the ionization current, count-rate, etc. to enable a mean air kerma rate to be indicated or determined.

This standard does not apply to thermoluminescence dosemetry systems or other passive integrating devices and it does not provide for the measurement of beta radiation.

IEC 60861 "Equipment for monitoring of radionuclides in liquid effluents and surface waters"

IEC 60861 Ed. 2 (2006) defines technical requirements for equipment used for monitoring of alpha, beta or gamma emitting radionuclides in liquid effluents and surface waters, provides some general guidance as to the possible detection capability of such equipment and indicates when and where its uses may be practicable.

This standard is applicable to equipment for continuous monitoring of the activity in liquid effluents which could be released in the environment during normal operations and in environmental waters.

It is applicable to water monitors intended to measure the volumetric activity or count rate due to radionuclides in the liquid and its variation with time and to actuate an alarm when a limit value of volumetric activity or count rate in water is exceeded.

IEC 60861 does not apply to equipment specifically for use in accident conditions that may require additional capabilities.

This standard is restricted to equipment for continuous monitoring of gross alpha or gross beta of maximum energy higher than 150 keV or gamma activity in liquid effluent streams and environmental waters. It does not deal with sample extraction and laboratory analysis.

The second edition of this standard from 2006 cancels and replaces the first edition published in 1983 and the first edition of IEC 61311 published in 1995. This edition, with respect to the previous edition, takes into account the main technological evolutions, notably the feasibility of continuous monitoring of alpha radioactivity in liquids and tests for electromagnetic compatibility.

IEC 62438 "Mobile instrumentation for the measurement of photon and neutron radiation in the environment"

IEC 62438 (2010) is applicable to mobile radiation detection systems used for the detection, quantification and identification of photon and/or neutron emitters in the environment. This includes point and distributed radiation sources.

In general, mobile instrumentation systems for nuclear radiation measurements in the environment are comprised of detectors, detector signal processors, position sensing devices, on-board data recording, operational monitoring, and real time display/alarm capabilities. In addition, advanced systems may provide data streams that can be transmitted by telemetry to operation centres.

This standard cancels and replaces IEC 61134, issued in 1992. The scope of IEC 61134 was restricted to exploration for geological deposits of potassium, uranium and thorium. IEC 62438 incorporates the range of currently available detector technologies and incorporates neutron monitoring. This standard also relates to a wide range of mobile platform applications including environmental, emergency response, security, in addition to geological.

IEC 61275 "Measurement of discrete radionuclides in the environment - In-situ photon spectrometry system using a germanium detector"

IEC 61275 (1997) is applicable to a portable or transportable photon spectrometry assembly using a high purity germanium (HPGe) detector to survey in situ, generally at 1 m above ground level, areas in the environment for discrete radionuclides. Such equipment is used to make rapid assessments of activity levels and corresponding free air exposure rates from photon emitting radionuclides. Measurement results may be used to develop guidance for subsequent follow on action including radiological assessments, sampling and monitoring programmes.

This standard does not apply to mobile measurement systems that are covered by IEC 62438.

IEC 61275 is currently in revision on CD level with expected publication in 2012.

Standards concerning the measurements of airborne radioactivity

These standards have been developed for many years within WG B13 of SC 45B before this working group was merged with WG B5.

IEC 60761 series "Equipment for continuous monitoring of radioactivity in gaseous effluents"

IEC 60761 Ed. 2 published in 2002 consists of the following parts, under the general title "Equipment for continuous monitoring of radioactivity in gaseous effluents":

- Part 1: General requirements
- Part 2: Specific requirements for radioactive aerosol monitors including transuranic aerosols
- Part 3: Specific requirements for radioactive noble gas monitors
- Part 4: Specific requirements for radioactive iodine monitors
- Part 5: Specific requirements for tritium monitors

IEC 60761-1 defines acceptable forms of such monitoring, provides general guidance as to the possible range of measurement and capability of such equipment as may be envisaged, and indicates when and where its uses may be practicable. This standard is applicable to equipment for continuous monitoring of radioactivity in gaseous effluents during normal operations and during anticipated operational occurrences. IEC 60761-1 does not apply to equipment specifically for use in accident conditions. Such equipment may require additional capabilities. This standard is restricted to equipment for continuously monitoring radioactivity in gaseous effluent. It does not deal with sample extraction and laboratory analysis.

IEC 60761-2 is applicable to equipment intended for simultaneous, delayed or discrete sequential measurement of aerosols in gaseous effluents discharged into the environment. It is applicable to equipment designed to fulfil the following functions:

- the measurement of the volumetric activity of the aerosols in gaseous effluents and/or the released total activity of aerosols ;
- the actuation of an alarm signal when either a predetermined volumetric activity or a predetermined total released activity of aerosols is exceeded.

This equipment is intended for measurement over a wide range of activity, including very small quantities in the presence of a much larger natural background. The daughters of ^{222}Rn (radon) and ^{220}Rn (thoron) are naturally occurring aerosols contributing to the natural background. The discrimination against natural activity can be an important problem when monitoring low level activity. In order to provide more and better information, complementary or retrospective laboratory analysis of the filters after collection may be performed.

IEC 60761-3 is applicable to equipment intended for simultaneous, delayed or discrete sequential measurement of radioactive noble gas in gaseous effluents discharged into the environment. It is applicable to radioactive noble gas effluent monitors intended to fulfil the following functions:

- the measurement of the volumetric activity of radioactive gases in the gaseous effluents at the discharge point and its variation with time;
- the actuation of an alarm when a predetermined volumetric activity or a predetermined total released radioactivity is exceeded;
- the determination of the gas activity discharged over a given period and/or information on the composition of a mixture of different gases in the discharge.

Radon is a natural radioactive noble gas. Its measurement is not included in this standard. The presence of radon, or its decay products, may interfere with the measurement of other (artificial) radioactive gases.

IEC 60761-4 is applicable to equipment intended for the simultaneous, delayed or discrete sequential measurement of radioactive iodine in all forms. When the effluent is sampled for measurement, iodine bound on aerosols is generally collected on a pre-filter which should be analyzed separately in a laboratory to provide a complete measurement.

It is applicable to equipment designed to fulfil the following functions:

- the measurement of the volumetric activity of iodine and compounds of radioactive iodine in gaseous effluents or of the released iodine total activity;
- the actuation of an alarm signal when either a predetermined concentration or a predetermined total released activity due to iodine or its compounds is exceeded.

This equipment is intended for measurement over a wide range of activity in the presence of other radionuclides in the gaseous effluent, including naturally occurring radioactive nuclides. Discrimination against other radionuclides can be important in measuring low levels of radioactive iodine.

This standard considers both the use of iodine collection media such as activated charcoal and the direct measurement of iodine in stacks or ventilation ducts.

IEC 60761-5 is applicable to equipment intended for the simultaneous, delayed or discrete sequential measurement of tritium in any gaseous form in gaseous effluents discharged into the environment. It is applicable to equipment designed to measure the concentration of tritium in the gaseous effluents at the discharge point and its variation with time and actuate an alarm when a predetermined volumetric activity or a predetermined total released radioactivity is exceeded. The equipment may also be used for the determination of the tritium activity discharge over a given period.

IEC 61171 "Monitoring equipment - Atmospheric radioactive iodines in the environment"

IEC 61171 (1992) is applicable to equipment intended for transportable or installed use for monitoring, as a function of time, airborne radioactive iodines (e.g., ^{131}I, ^{125}I) in the environment (outside of buildings or facilities, at heights typically from one to a few meters above the surface) of a nuclear facility during normal operations, during anticipated operational occurrences or during accident conditions. For the purpose of this standard, monitoring includes the continuous sample collection with, if it is required, the capability to automatically initiate sampling.

This standard does not include equipment intended for monitoring radioactivity associated with gaseous effluents (at the stack). This type of equipment is covered in IEC 60761-1 through IEC 60761-5. This standard does not include monitoring for ^{129}I.

Specific chemical species of radioactive iodines may be selectively collected by specialized sampling equipment according to the requirements specified by agreement between manufacturer and user. Such samples should be in the form appropriate for laboratory analysis. The radioactive iodines may be gaseous, vapour or in aerosol form. IEC 61171 is restricted to equipment for monitoring radioactive iodines in the atmosphere and does not address sample extraction and subsequent laboratory analysis. This standard does not specify tests with atmospheric radioactive nuclides.

IEC 61172 "Monitoring equipment - Radioactive aerosols in the environment"

IEC 61172 (1992) is applicable to transportable or installed equipment for continuous monitoring of radioactive aerosol in the environment for both normal and accident conditions. For the purpose of this standard, monitoring includes the continuous sample collection with, if desired, the capability to automatically initiate sampling. In particular the standard is applicable to equipments designed to fulfil the following functions:

- determination of the activity per unit volume of radionuclides in the form of aerosols, either per unit time together with the variations as a function of time, or integrated over a longer time period, e.g., 24 h and measurement of the volume sampled.
- actuation of an alarm signal when either a predetermined high concentration or a predetermined time integrated concentration of an aerosol activity is exceeded.

The aerosol monitoring can be made both by continuous measurements during sampling and by measurement after collection of the sample. In the measurement after collection, the sample is removed from the air sampler and analysed in a laboratory. This procedure of measurement has to be followed in special cases such as the activity assessment of specific radionuclides and is not within the scope of this standard. The continuous method of measurement is the more widely used and consists in collecting the radioactive samples by drawing air through a filter and measuring the aerosol activity accumulated on it with a detector which is close to the filter. The filters are suitably chosen and periodically changed according to the conditions of use. In some cases automatic filter sequences are used to avoid the build-up of radioactivity or excess of dust on the filter. Also the type of detectors depends on the various conditions such as the type of radiation, its energy and the level of activity.

The discrimination against natural radioactivity (such as airborne radon and thoron daughters) can be an important problem in monitoring low-level radioactivity in air especially with transuranic elements.

IEC 61578 "Effectiveness of radon compensation for radioactive aerosol monitors including transuranic aerosols "

IEC 61578 "Calibration and verification of the effectiveness of radon compensation for alpha and/or beta aerosol measuring instruments - Test methods" was published in 1997 but is currently in revision on CD stage with a new title "Effectiveness of radon compensation for radioactive aerosol monitors including transuranic aerosols".

This standard is applicable to type test methods which allow calibration and measurement of the effectiveness of radon decay product compensation for radioactive aerosol monitors. IEC 61578 in his 2nd edition defines aerosol characteristics used in these tests and applies the following test methods permitting the measurement of:

- the sensitivity of the monitor relative to alpha and/or beta defined man-made aerosols,
- the reference response of the monitor relative to alpha and/or beta defined man-made aerosols,
- the response of the monitor relative to radon decay product aerosols,
- the effectiveness of radon compensation,
- the response of the monitor relative to a mixture of aerosols constituted by natural and man-made radioactive emitters.

IEC 62302 "Equipment for sampling and monitoring radioactive noble gases"

IEC 62302 (2007) directly compliments IEC 60761-1 and IEC 60761-3. This international standard is applicable to equipment used for sampling and continuous measurement of radioactive noble gases in the workplace, in gaseous effluents discharged into the environment as well as in the environment itself. Monitoring by definition is the process of continuous and real-time measurement. The processes of sampling or taking samples for retrospective laboratory analysis are included in this standard.

IEC 60761-3 that will be complimented by IEC 62302, is applicable to installing portable and transportable equipment for sampling and monitoring radioactive noble gases, only in gaseous effluents, while IEC 62302 expands coverage to include monitoring all possible locations where radioactive noble gases could present a radiological hazard. The equipment is designed to be operational during normal operation conditions as well as under emergency conditions, both during and following an accident. Depending on the nature of the emergency conditions it may be necessary to install specially designed equipment for normal operational conditions and other equipment for emergency conditions.

IEC 62303 "Equipment for monitoring airborne tritium"

IEC 62303 (2008) is applicable to equipment used for sampling and continuous measurement of tritium in the workplace, in gaseous effluents discharged into the environment as well as in the environment itself and it is applicable to installed, portable and transportable equipment.

The object of this international standard is to establish mandatory general requirements and to present examples of acceptable methods and equipment for continuously monitoring and/or sampling airborne tritium. IEC 60761-5 which is complemented by IEC 62303, is applicable to equipment for sampling and monitoring tritium only in gaseous effluents, while IEC 62303 expands coverage to include monitoring all possible locations where tritium could present a radiological hazard. The equipment is designed to be in operation during normal operation conditions as well as under emergency conditions, both during and following an accident. Depending of the emergency conditions, it might be necessary to install specially designed equipment for normal operation conditions and other equipment for emergency conditions.

IEC 62303 is applicable to tritium samplers and tritium monitors intended to provide the following functions:

– measurement of the volumetric activity of tritium and its variation with time in the workplace, in gaseous effluents at the discharge point and in the environment;
– actuation of an alarm when a predetermined volumetric tritium activity or tritium concentration or a predetermined total activity of released tritium is exceeded;
– determination of the total tritium activity discharged over a given time;
– sampling and analysis of air or gas containing tritium.

Conclusions

The object of the presented standards is to establish performance requirements, to give examples of acceptable test methods and to specify the general characteristics, general testing procedures, radiological, mechanical, electrical and environmental characteristics of the equipment.

IEC/SC 45B with its 15 published international standards greatly contributes for the high quality of existing instrumentation for measurement of environmental radiation. Compliance with such standard requirements provides the manufacturers with internationally acceptable specifications and the device users with assurance of the rigorous quality and accuracy of the measurements.

Experts from all countries are welcome to contribute to the IEC work (the national committees should be contacted for registration). The next SC 45B meeting will be in October 2010 in Seattle (US).

Acknowledgements

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Investigation radiation hygienic monitoring in the Russian NPP vicinity

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Abstract
Investigation monitoring in the NPP vicinity includes radiation hygienic monitoring and population health monitoring. Radiation hygienic monitoring is the system of the comprehensive and dynamic surveillance including the long-term continuous control of radiation hygienic situation parameters and doses of residents in the near-by areas of NPP. In the reference points of the NPP surveillance area and in the comparison area, the specialized rules are elaborated. These rules include the types of environmental media, scope and periodicity of sampling, methodological and technical requirements, etc. Two approaches are used for population health assessment: epidemiological and cohort. Medical demography characteristics are based upon indices of the birth rate, general malignant neoplasm mortality, and infant and childhood mortality. Morbidity is used as quantitative and qualitative index of the population health.

The radiation hygienic situation in NPP surveillance areas is generally satisfactory and stable. The individual population risk for all radiation sources in surveillance areas of three examined NPPs is $(2.5\text{-}3.1)\cdot10^{-4}$ cases per year. The individual risk related to the NPP operation is $0.4\cdot10^{-12}$ (Volgodonsk NPP) to $3.2\cdot10^{-8}$ (Novovoronezh NPP) cases per year. Thus, this risk is hundreds and even million times lower if compared to unconditional acceptable risk – $1\cdot10^{-6}$. The assessment of the health baseline as well as in the comparison areas was elaborated applying data of state medical statistics within more than ten years. The peculiar attention was attracted to the malignant neoplasm morbidity, their incidence rate and dynamics. The developed comprehensive investigation monitoring in the NPP vicinity should be the necessary part of the nationwide system of the public radiation protection regulation under nuclear renaissance, because it provides the opportunity to get the modern objective assessment of the NPP impact in the environment and population health.

Introduction
The basic social requirement for the atomic power is the provision of convincible evidences of the safety warrantees of both the plant operation personnel and residents of the NPP affected area. Since the USSR atomic industry start-up, the Burnasyan Federal Medical Biophysical Centre (before 2008 – Institute of Biophysics) of the Federal Medical Biological Agency of Russian Federation was the leading institution

responsible for the medical issues of the industry. One of basic activities of the Burnasyan Federal Medical Biophysical Centre is the scientific support of the radiation protection of the public.

The analysis of medical effects of the Chernobyl accident has obviously demonstrated the high importance of the establishing so-called “zero” baseline of the population health. Such information is necessary to evaluate possible medical effect of many year operation of atomic facilities and to clarify the scale of the health impact. Such assessments require vast comprehensive hygienic studies within the long period of time. Thus, the necessity to organize the rigorous social hygienic monitoring (SHM) in the NPP vicinity is occurred, which is the system of state surveillance, analysis, assessment, and prognosis of the population health and habituated environment as well as the correlation of the population health versus the habituated environment factor exposure. According to the Russian Federation Law on Sanitary and epidemiological well-being of the population, SHM is the major mechanism to regulate this well-being. The SHM implementation is of specific significance, taking into account special document regulates this work and attributes it not only to sanitary and epidemiological authorities and institutions, but also to other ministries and agencies [1]. Present SHM system presumes the development of the unified system of collection, management, and assessment of the information regarding the environmental contamination and population health indices.

The purpose of the present paper is to describe methodology and current status of SHM results obtained in the NPP vicinity. SHM includes radiation hygienic monitoring and population health monitoring.

Ideology & methodology of the research

The main stages of SHM in the NPP vicinity are as follows:

- Conducting of background “zero” examination of public health and radiation hygienic situation before the NPP start-up;
- Dynamic monitoring of important parameters of public health and hygienic parameters (including radiation factor) during the whole NPP operation period;
- Comparison of observed changes of public health and environment against “background” values;
- Observation of changes of public health within 5-10 years after decommissioning NPP or reactor units.

Following eight SHM directions are included for the NPP vicinity:

1. Radiation hygienic situation specifications and population doses
2. Medical demography features of the populations
3. Population morbidity analysis
4. Reproductive health status
5. Endemic diseases
6. Children health status
7. Societal psycho-physiological status of the population and mass media relations
8. Monitoring data bank development.

The scientific ideology of the this research is investigation radiation hygienic monitoring (RHM), which is the system of the comprehensive and dynamic surveillance including the long-term continuous control of radiation hygienic situation parameters and doses of residents in the near-by areas of NPP and other radiation facilities of the first category [2].

The obvious question on the differences between research RHM and radiation control elaborated by different agencies and organizations is occurred. It is clear that radiation situation monitoring in the NPP surveillance area is elaborated by the NPP operator and designer, hydrometeorology service, and sanitary and epidemiological surveillance service. Results of such control attributed to the agency responsibility as well as elaborated under the radiation hygienic certification of territories have some disadvantages. These disadvantages include the rare application of radiochemical techniques, low periodicity of examinations, insignificant number of comparison points, and, in some cases, the absence of the comparison area and significant change of the list of objects subjected to the control. Finally, the radiation factor is not considered together with other non-radiation environmental factors affecting the human and the ranking of these factors is not provided. Unfortunately, low levels of radiation hygienic parameters are not recorded in practice. The indicated disadvantages of the practical radiation monitoring can be corrected by the implementation the detailed investigation RHM.

To elaborate RHM in the reference points of the NPP surveillance area and in the comparison area, the specialized rules are elaborated. These rules include the types of environmental objects, scope and periodicity of sampling, methodological and technical requirements, and inter-relations of interested parties. Table 1 provides a number of aspects of elaborated RHM rules on the sampling and examined parameters of the radiation hygienic situation in the NPP vicinity. RHM rules (environment section) presume the usage of NPP surveillance network data obtained by State Hydrometeorology Committee and Automated System of Radiation Situation Control as well as the original own data [3].

Table 1. Periodicity of sampling and examined parameters in the control points of the NPP surveillance area (routine operation) and in the comparison area.

Supervision object	Periodicity of information receipt	Parameters	Value dimensionality to be measured
Environment			
Gamma background in the area	Random inspection (weekly)	Exposure dose rate of external γ radiation in the area	μR/h
		Effective dose rate of external γ radiation	mSv/y
Atmospheric air (aerosol content in surface air)	Daily, monthly	ß-radioactivity, ^{90}Sr, ^{137}Cs, ^{134}Cs. Content of inert radioactive gases, ^{131}I, ^{60}Co	Bq/m^3
Density of radioactive fallout	Monthly	Total ß-radioactivity, ^{90}Sr, ^{137}Cs, ^{131}I	Bq/m^2/month
Soil	Annually	Total ß-radioactivity, ^{90}Sr, ^{137}Cs, ^{40}K	Bq/kg, kBq/m^2
Gramma grasses or natural growth	Annually (during period of vegetation)	Total ß-radioactivity, ^{90}Sr, ^{137}Cs, ^{40}K, ^{131}I	Bq/kg

Table 1. Continued.

Supervision object	Periodicity of information receipt	Parameters	Value dimensionality to be measured
Environment			
Water of open ponds including water reservoir-cooler	Two times a year (random inspection)	Total α- and ß-radioactivity, ^{90}Sr, ^{137}Cs. In water reservoir-cooler: ^{131}I, ^{60}Co, ^{90}Sr	Bq/l
Underground sources of water (using observation wells)	Monthly (random inspection)	Total α- and ß-radioactivity, ^{90}Sr, ^{137}Cs, ^{3}H	Bq/l
Drinking water (using draw well and water supply systems)	Quarterly	Total α- and ß-radioactivity, ^{90}Sr, ^{137}Cs	Bq/kg
Water-plants and sea-floor sediments of open ponds	Annually (random inspection)	^{90}Sr, ^{137}Cs, ^{40}K in water-plants – total ß-radioactivity	Bq/kg
Foodstuffs of local manufacture			
Bread or bread flour	Annually	^{90}Sr, ^{137}Cs	Bq/kg
Grain (wheat)	Annually	^{90}Sr, ^{137}Cs	Bq/kg
Milk	Two times a year (in summer and in winter)	^{90}Sr, ^{137}Cs, ^{131}I	Bq/l
Meat (pork, beef, mutton)	Annually	^{90}Sr, ^{137}Cs	Bq/kg
Poultry meat	Two times a year	^{90}Sr, ^{137}Cs	Bq/kg
Freshwater fish (river and lake ones)	Annually	^{90}Sr, ^{137}Cs	Bq/kg
Potatoes	Annually (during harvesting)	^{90}Sr, ^{137}Cs	Bq/kg
Edible roots (beet, carrot)	Annually (during harvesting)	^{90}Sr, ^{137}Cs	Bq/kg
Vegetables (cucumbers, tomatoes, onion, cabbage)	Annually (during harvesting)	^{90}Sr, ^{137}Cs	Bq/kg
Gourds (melon, water-melons)	Annually (during harvesting)	^{90}Sr, ^{137}Cs	Bq/kg
Food pot-herbs and leaf vegetables	Two times a year (during period of vegetation)	^{90}Sr, ^{137}Cs, ^{131}I	Bq/kg
Garden fruits and berries (at places of harvest)	Annually (during harvesting)	^{90}Sr, ^{137}Cs	Bq/kg

When choosing the comparison area, following circumstances should be taken into account:

a) hygienic characteristics similarity:
   – soil and sub-soil type;
   – plant species;
   – chemical content of water in superficial reservoirs and underground waters;
   – conditions of foodstuff production;
   – burdens of natural and (or) global fallout radionuclides.

b) the positioning outside the radiation facility affecting area;

c) medical assistance peculiarities (number of specialized medical doctors and availability of medical equipment etc.).

We would like to address population health monitoring, which requires the scientific justification of index selection and population health assessment criteria. These criteria should correspond to the following conditions:

– Applicability for the population assessment;
– Correspondence to the long-term observation tasks, namely, the simplicity, the applicability for the large number of individual examinations, reliable quantification, and objective qualitative characteristics;
– Future possibility for the assessment of possible NPP radiation impact in population health.

Two approaches are used for population health assessment: epidemiological approach based upon the health evaluation via medical statistics data and cohort clinical approach to include health assessments via detailed examination of critical population groups and critical body systems.

Medical demography characteristics are based upon indices of the birth rate, general malignant neoplasm mortality, infant and childhood mortality.

Morbidity is used as quantitative and qualitative index of the population health to assess the dissemination and structure of major diseases in the population of the NPP vicinity. To evaluate health status, medical statistics data are applied and dynamic indices within many years are analyzed.

The index reflecting the population health as a whole is the reproductive health. Reproductive health changes are most specific to the unfavorable factor impact in human health. Basic parameters of the reproductive health are as follows:

a) obstetric and gynecological status (rate and character of the pregnancy termination, gynecological and oncological morbidity)

b) reproduction function state (spontaneous abortions, still birth, early neonatal mortality) and newborn condition (morbidity, inherited developmental defects, etc.).

The child organism at growth and development phase is specific to the peculiar sensitivity, so the children under the radiation risk compose the population critical group. In the framework of the elaborated children health monitoring, statistical data on pediatric assistance are used as well as the detailed clinical examination results in some groups of children.

The “baseline” data on leukemia and thyroid morbidity are essential for the comprehensive assessment of NPP vicinity resident health to use these data as the reference for the following radiation exposure effects. Thyroid examinations presume endocrinologist evaluation and ultrasound imaging to obtain sizes and echo structure as well as the iodine deficiency assessment via urinalysis.

Material and methods

The investigation monitoring been implemented in a number of Russian NPPs including Kalinin, Volgodonsk and Novovoronezh NPPs. When implementing monitoring the following tasks were elaborated:

1. The dynamic acquisition of necessary, sufficient, and confident information on controllable radiation parameters of the environment and on radionuclide burdens in foodstuff and water.
2. Investigation of the foodstuff consumption.
3. Assessment of the external and internal exposure doses in population.
4. Revealing current changes of radiation hygienic situation and prognosis of possible consequences in the population.
5. Providing the information for the managerial decision making to keep radiation doses as low as reasonably achievable.
6. Providing the information for local authorities and local centers of State Sanitary and Epidemiological Surveillance on radiation hygienic situation in the surveyed territory.
7. Forming the databases on the examined parameters.
8. Providing the information for local residents of the NPP vicinity.

More than 670 environmental and foodstuff samples in 48 settlements positioned both in the surveillance areas and in the comparison areas, in cooling ponds and other water reservoirs of the NPP vicinity was investigated (Table 2).

Table 2. The amount of investigations, which were carried out nearby NPPs.

NPP	Quantity of points	Number of proof samples		
		Foodstuffs	Drinking water	Other objects of environment
Volgodonsk	22	270	26	24
Kalinin	15	151	29	20
Novovoronezh	11	98	22	12
	48	519	77	56

Results of the Investigation Monitoring

The comparison to the current standards is also provided. The radiation hygienic situation in NPP surveillance areas is generally satisfactory and stable:

– Outdoors gamma dose rate is in the range of background fluctuations for such territories;
– ^{90}Sr and ^{137}Cs specific activity in outdoor water reservoirs is in the range of radionuclide content in water reservoirs of the Central Russia;
– ^{90}Sr and ^{137}Cs burdens in drinking water is below intervention levels for 135 and almost 300 times, respectively; total alpha and beta activity is below permissible levels;

– ^{90}Sr and ^{137}Cs burdens in foodstuff products and drinking water are 100-1000 times below permissible levels;
– ^{90}Sr and ^{137}Cs burdens in foodstuff products and drinking water (NPP surveillance areas) are similar to these in other regions of the country.

Table 3 provides total doses in NPP vicinity residents due to all sources of ionizing radiation. As it follows from Table 3, the natural background irradiation results to 69-71 % of total effective dose from all sources. The dose impact of global fallout and NPP exposure is 0.2-0.3% in the vicinity of Volgodonsk, Kalinin and Novovoronezh NPPs [4].

Table 3. Total effective doses in the population in the NPP vicinity, mSv/year.

Component of dose, mSv/y	Volgodonsk NPP	Kalinin NPP	Novovoro-nezh NPP	Assessment criteria
^{137}Cs and ^{90}Sr man-caused background	0,007	0,0079	0,010	Russia – 0.022 world – 0.007
NPP	0,0000006	0,00044	0,0017	0.01
Natural sources	2,4	2,9	2,3	2.4
Medical sources	1,0	1,3	1,2	Russia – 1.0 world – 0.4
Sum of all sources	3,4	4,2	3,5	Russia – 3.5 world – 2.8

The assessment of the health baseline in the vicinity of NPPs as well as in the comparison areas was elaborated within more than ten years. The peculiar attention was attracted to the malignant neoplasm morbidity, its incidence rate and dynamics. Specifying children health and the morbidity structure is estimated. Data obtained did not show any significant terms of worse health state of adults and children.

At present time, the radiation hygienic monitoring in the NPP vicinity can provide the radiation risk assessment (Table 4). The individual population risk for all radiation sources in surveillance areas of three examined NPPs is $(2.5\text{-}3.1)\cdot10^{-4}$ cases per year. The individual risk related to the NPP operation is $0.4\cdot10^{-12}$ (Volgodonsk NPP) to $3.2\cdot10^{-8}$ (Novovoronezh NPP) cases per year. Thus, this risk is hundreds and even million times lower if compared to uncondtional acceptable risk – $1\cdot10^{-6}$.

Table 4. Individual life span risk of stochastic effects in residents of the NPP surveillance area, year^{-1}.

Risk due to all the sources	Risk due to NPP operation (R_{NPP})	UAR/ R_{NPP}
Volgodonsk NPP		
$2{,}5\cdot10^{-4}$	$4{,}0\cdot10^{-13}$	2,5 million times
Kalinin NPP		
$3{,}1\cdot10^{-4}$	$6{,}6\cdot10^{-9}$	150 times
Novovoronezh NPP		
$2{,}6\cdot10^{-4}$	$3{,}2\cdot10^{-8}$	30 times

Conclusion

The developed comprehensive investigation monitoring in the NPP vicinity (a) should be the necessary part of the nationwide system of social hygienic monitoring, (b) provides the opportunity to get the modern objective assessment of the NPP impact in the environment and population health, and (c) can be widespread to other radiation facilities in this country.

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Radio-ecological criteria and norms during remediation of the nuclear legacy facilities in the Russian Northwest

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Abstract

Remediation of the sites for temporary storage (STS) of the Spent Nuclear Fuel (SNF) and Radioactive Waste (RW) is one of the most important challenges for the Russian Northwest region. The prime task is to develop radiation environmental regulations on justification of radiation safety assurance during remedial operations at the STS. According to legislative and normative acts of the Russian Federation regulating management of radioactively contaminated territories after identification of the site contamination level at the radiation facility, one of three decisions can be made: conservation of the site; renovation of the site and buildings (brown lawn);unlimited use of the site (greenfield).The criteria and regulations for the SNF and RW STS facilities and site have been developed, which are suitable for each remediation option. At the same time, the environmental models have been taken into account; reference levels have been developed expressed in radiation parameter units, which could be measured during radiation control and monitoring: surface beta and alpha contamination of the STS buildings, gamma dose rate, radionuclide specific activity in soil, annul activity concentration of ground water, radionuclide contents in hydrobionts.

Introduction

Two technical bases of the Northern Fleet were created in the Russian Northwest in the 1960s at Andreeva Bay in the Kola Peninsula and Gremikha village on the coast of the Barents Sea. They maintained nuclear submarines, performing receipt and storage of radioactive waste (RW) and spent nuclear fuel (SNF). No further waste was received after 1985 and the technical bases have since been re-categorized as sites of temporary storage (STS).

Remediation of sites and facilities of the STS of SNF and RW in Andreeva bay and Gremikha village on the Kola Peninsula is one of regulatory functions of the Federal medical-biological agency (FMBA of Russia). The work has involved the Russian Federation Burnasyan Federal Medical Biophysical Centre, which is technical support organization of the FMBA of Russia. In thus work took part the Norwegian

Radiation Protection Authority (NRPA) in frame of the Norwegian government's Plan of Action to improve radiation and nuclear safety in northwest Russia.

Main tasks within the FMBA – NRPA cooperation consist of:

- Independent detailed analysis of the radiation situation at and near the STSs.
- Radiological threat assessment to determine priority issues for regulatory attention.
- Radiological control and monitoring of the environmental conditions.
- Development of a regulatory documentation system ensuring radiation protection observance of workers and the public, including radiation-hygienic criteria and standards of rehabilitation of contaminated territories.

In order to obtain comprehensive information with respect to current radiation circumstances at STS (independent from regulatory point of view), radiation-hygienic monitoring of STS facilities has been carried out.

Material and methods

Over 2005-2008, more than 180 samples of environmental media, local foods and drinking water were collected in Andreeva Bay and Gremikha village expeditions; moreover, personal dose monitoring was implemented. Gamma-spectrometry and radiochemical methods were applied in sample measurements.

Characterization of SevRAO facilities in Andreeva Bay and Gremikha

The STS Andreeva Bay is located on Kola Peninsula in the Barents Sea coastal strip (Motovsky gulf, west bank of Zapadnaya Litsa bay). The nearby settlements are: Bolshaya Lopatka (2.4 km); Nerpitchie village (1.8 km); Zaozersk city (8 km). The population is 15 700, the majority of which are military estates. The facility holds about $1.3 \cdot 10^{17}$ Bq of SNF and $6{,}0 \cdot 10^{14}$ Bq of RW.

The STS Gremikha is located on Kola Peninsula in Chervyanaya Bay of the Barents Sea. The nearby settlements are: Gremikha village (0.7 km from the site) and Ostrovnoy city (1.2 km). The population is 3500 (mainly, former soldieries and their families). The facility holds about $1.3 \cdot 10^{16}$ Bq of SNF and about $3.3 \cdot 10^{13}$ Bq of RW.

Up to now, a large amount of SNF contained in 88 unloaded cores, as well as 17558 tons of solid radioactive wastes and 3042 tons of liquid radioactive wastes have been accumulated in Andreeva Bay and Gremikha.

Specification of areas within the STS territory

With the purpose of radiation protection of workers and the public, the following areas are specified on-site and around the STS site:

- Controlled access area (CAA) – SNF and RW store facilities are situated here and radiation-hazardous operations are performed here too;
- Uncontrolled (free access) area (UA) – Facilities intended for work supplying in CAA;
- Health protection zone (HPZ) – This is an area of administrative and technical provision of the STS;
- Supervised area (SA) – This is an area surrounding the STS, where radiological monitoring is carried out to guarantee radiation safety and protection for the public.

The member of the public must not stay within the first three areas.

Radiation situation on-site the STS in Andreeva Bay

The accomplished examinations showed that gamma dose rates within the STS territory varied over a wide range: in CAA - from 0.2 to 140 µSv/h-1; in UA - from 0.2 to 12 kBq·kg-1; in HPZ - from 0.1 to 0.2 kBq·kg-1. Within SA, gamma dose rates varies from 0.063 to 0.14 µSv/h-1 with an average value of 0.12 µSv/h-1, which does not differ from the levels typical for the territories of Northwest Russia and in the Murmansk region, in particular. The results of selective personal dose monitoring show that external exposure gamma rates of the public and workers of group B (individuals who are not working directly with the sources of ionizing radiation, but who, due to their working place location, can be exposed to radiation) due to natural and man-made sources of ionizing radiation are, respectively, equal to 0.8 and 0.9 mSv/y. Internal public radiation doses associated with intake of radionuclides with food are 14 µSv·y-1. The total effective radiation doses to the public living in the STS's SA of Andreeva Bay(due to natural and man-made radionuclides) are estimated to be approximately 0.8 – 0.9 mSv·y-1, that is not more than the actual norms.

The highest level radioactive contamination of soil on-site induced by man-made radionuclides is observed in the area of the old technological pier and around some SNF store facilities, where ^{137}Cs specific activity reaches 5.7 107 Bq·kg-1, and that of ^{90}Sr is 5.7 106 Bq·kg-1. ^{137}Cs and ^{90}Sr concentrations in soil within HPZ and SA is at the background level typical for "clean" Russian Northern areas and does not exceed 36 Bq/kg and 4 Bq/kg, respectively.

Radiation situation on-site the STS in Gremikha

Gamma dose rate within CAA varies from 0.2 to 500 µSv·h-1 (maximum values are 4 times more than those in Andreeva bay); in UA – from 0.2 to 12 µSv·h-1 and levels within approximately 80% of the territory do not exceed 5 µSv·h-1. In HPZ and SA (in Ostrovnoy and Gremikha) it varies from 0.09 to 0.2 µSv·h-1, i.e., within fluctuation limits of natural background of this region. The results of selective personal monitoring of the people living and working (workers group B) due to natural and man-made sources of ionizing radiation in the STS area show that the external exposure gamma dose rates are 0.7 mSv·year-1(for public) and 0.9 mSv·y-1 (for worker group B). Internal public radiation doses due to intake of ^{137}Cs and ^{90}Sr with food are approximately 14 µSv·y-1, which is significantly lower than acceptable levels.

Within the industrial site, man-made contamination is observed in top-soil due to ^{137}Cs, ^{90}Sr and, in small concentrations, ^{60}Co, ^{152}Eu, and ^{154}Eu. In SA (including Gremikha and Ostrovnoy), ^{137}Cs and ^{90}Sr contents in soil are mainly within background level (1 – 50 Bq·kg-1). In some cases, at local parts outside the settlements, observed levels exceed background values by up to 100 Bq·kg-1 by ^{137}Cs.

Results of radiation situation assessment at STS in Andreeva bay and Gremikha

Radiation monitoring of the environmental media showed considerable exceeding of typical background values of ^{137}Cs and ^{90}Sr radionuclide concentrations (in the SSZ coastal strip) in seaweeds, bottom sediments and vegetation. An exceeding is also observed in some cases in the STS SA environmental media in comparison with background values. Preliminary results of sorption experiments of radionuclides on local soil and ground waters suggest that radionuclide migration from highly contaminated areas on site, via groundwater flow pathways, is possible. This leads to permanent entry of radioactive substances into the off-shore marine environment.

According to radiation monitoring of catches in the STS off-shore marine environment, the concentration of ^{90}Sr and ^{137}Cs in fish is in the range 0.7 - 13 Bq·kg-1 for ^{90}Sr and 0.4 – 35 Bq·kg-1 for ^{137}Cs, respectively, being significantly lower than actual Russian accepted radiation contamination levels. With the purpose of radiation exposure restriction during large-scale STS remedial work, FMBA established a public radiation dose quota; this quota is 100 µSv·y-1 due to effluents and 30 µSv·y-1 due to radioactive substance discharges (table 1).

Table 1. Public radiation dose quota due to effluents and to radioactive substance discharges under conditions of STS facility normal operation.

Sources of exposure	Quota, µSv·y-1
Gas-aerosol discharges	100
Intake with seafood	300
Reserve for unregistered sources	200

Conclusions

Environmental radiation monitoring demonstrated significant excess (in comparison with typical background values) of ^{137}Cs and ^{90}Sr contents at local parts of the coastal strip of the STS health protection zone in seawater, seaweeds, bottom sediments, vegetation and soil.

Results of radionuclide sorption examination in soil and ground water permit to assume the presence of effective migration from contaminated areas via groundwater, causing radioactive inflow into offshore marine waters. Having in mind a possibility of further contamination of the STS area, dynamic surveillance is needed of the radiation situation both at routine activity, and at SNF and RW removal [2].

The described work carried out under joint FMBA and NRPA Project, devoted to regulation of the public radiation and nuclear safety during STSs operations, current output has included the following documents:

– InitialThreat Assessment for the situation at STS sites.
– Guidance “Criteria and norms on remediation of STS sites and facilities contaminated with man-made radionuclides”.
– Guidance “Hygienic requirements for personnel and public radiation safety guaranteeing at the stage of designing the work with SNF and RW at STSs”.

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Quality management system of in-vivo measurement (IVM) lab at Karlsruhe Institute of Technology (KIT) – accreditation and experience

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Abstract
The in-vivo monitoring lab (IVM) at Karlsruhe Institute of Technology operates one whole body counter and three partial body counters. IVM is an approved lab for individual monitoring for incorporation according to German regulation. In 2007 a web based quality management system has been set up and all the method and procedures used at IVM have been described in a set of documents. Several document classes are (e.g. standard operation procedures SOP) are used for this. For each class of document a template satisfying the formal requirements is used. In 2007 several internal audits were used for fine-tuning the system before an external audit, which finally granted accreditation to the IVM, was held. The system is kept flexible and can thus be easily adapted to new situations (e.g. organisational changes). Only a sparse amount of the work time needs to be spent to maintain the running system. It also leaves enough freedom in the daily routine work at IVM and found acceptance by the employees, quickly. After two years of working with the quality management we can say it was worth the trouble and time required in setting it up.

Introduction
The in-vivo monitoring laboratory (IVM) at Karlsruhe Institute of Technology (KIT) operates one whole body counter with NaI(Tl) detectors in a stretcher geometry and three partial body counting systems, two with Phoswich and one with HPGe-detectors. Routine monitoring and special measurements of workers at the KIT campus north (either staff of KIT, or of external companies located at the campus) and for external customers are done at IVM. The laboratory is used in scientific studies, e.g. studies on caesium content of the population, and in education and training of students and radiation protection workers. IVM is an approved lab for individual monitoring by direct assessment of incorporated radionuclides according to German regulation. One of the requirements for the approval is an poof of competence via accreditation on ISO/IEC Standard 17025. A quality management system (QMS), which was successfully audited and granted accreditation, has been set up at IVM in 2007.

Material and methods

The quality management system (QMS) is based on ISO9001 certified system of the central safety department. All issues dealing with management requirements (i.e. chapter 4 of ISO 17025) are met by this system. The IVM team added documentation of the routine work and the quality assurance procedures in the lab meeting the technical requirements (i.e. chapter 5 of ISO 17025).

A web based quality management software (CAS Teamworks 2010) is used for handling and controlling of all relevant documents and the workflow for all the steps of document issuing and approval. Employees can log in to the system from their workplaces simply via a web browser. Using an online system ensures that everyone is working with the same revision of a given document. Thus "errors" caused by invalid or expired documentation are minimised. In principle one should work only with the electronic documentation and avoid printing the documents. The latter is not forbidden, but hardcopies are valid only if the revision number on the hardcopies and the revision number in the electronic system are identical. For each controlled document the roles of editor, examiner, approver and the persons that need to acknowledge the final document need to be assigned. In the document properties displayed in the system action links for each step of the document control workflow are provided. The persons involved are informed by the system's messaging interface or via an email with a link to the document in the QMS, when their duty is due. The final approval for the release of a document is given by the quality management representative and a process owner of the procedure involved. This assures a final check of the document and ensures that the system is kept consistent.

Table 1. Main Document Classes in the Quality management system.

Class		Content
QMH	Quality Management Manual (Qualitätsmanagementhandbuch)	Description of formal organization, quality objectives and procedures
MB	Description of Method used (Methodenbeschreibung)	Description of non-standardized methods used in the laboratory
VVP	Protocol of verification of validation of methods (Verifizierungs- und Validierungsprotokoll)	Description of verification and validation of methods used
VA	Generic Procedures (Verfahrensanweisung)	Description of general processes and workflow
SAA	Standard Operating Procedure (Standardarbeitsanleitung)	Operations are described step by step
HMS	Specification of tools and materials (Hilfsmittelspezifikation)	Only tools and materials used in the methods are specified
QA	Reporting Forms (Qualitätsaufzeichnung)	Standardized Reporting Forms for perseverative reporting

Several classes of documents are used in the quality management system, an overview of the most important ones is given in table 1. Central document of the quality management system is a quality management manual (QMH) which provides information of the duties and the formal organization of the institution described (e.g.

ISF or IVM), as well as the quality objectives and policy. The generic processes used are described in process instructions (VAA), more detailed instructions are provided in standard operating procedures (SAA). The tools and materials used in the processes are described in specification sheets (HMS) only if they need to be calibrated or are relevant for the quality of the process or of its product. Templates satisfying all formal requirements of ISO 17025 are provided for each class of documents. Each document is derived from these templates and given a unique ID of the type “Class Lab Number” e.g. “SAA IVM 100” and a meaningful title. Using this ID the documents can easily refer to others and thereby link procedures during a workflow.

The generic principles of our measurements were described in three documents as non-standardized methods:

- Determination of incorporated radionuclides by gamma spectroscopy in the whole body counter
- Determination of incorporated radionuclides by gamma spectroscopy in the partial body counter with Phoswich detector.
- Determination of incorporated radionuclides by gamma spectroscopy in the partial body counter with germanium detectors

The general information and the scientific background of the techniques used in the laboratory is provided in separate documents (MB). These descriptions were required because no reference document (e.g. an ISO standard) on in-vivo monitoring techniques are available. Scientific publications (IAEA 1996, ICRU 2003) were the basis for these documents. Based on past experience and successful participation in intercomparison exercises these methods were verified and validated. This has been documented in a VVP-document.

The full workflow from customer enquiry to final billing including measurement and reporting has been described in a generic procedure (VAA IVM 003). A translated flowchart as provided in this document is shown in figure 1. Two other generic procedures describe the principal workflow for measurements and dose assessments. Detailed step by step descriptions for the single parts of these workflows are given in standard operating procedures, which are linked by their document IDs in the generic procedures. The single documents are focussed on (small) steps of the work (e.g. daily quality assurance measurement), thus only small adoptions of the documentation are required when changes in parts of the procedures are made. The minimum requirements which need to be fulfilled before an enquiry is accepted are described in a standard operation procedure. A counselling interview with a customer is conducted to check these requirements prior to the first measurement ordered. Two documents describe the application of the reference procedure for internal dose assessments as reported in German regulation. All other laboratory (e.g. quality assurance of the balances in the lab) and administrative work (e.g. invoicing, archiving, …) is described in separate SAA documents. An overview of all quality assurance procedures and their frequencies (e.g. daily, weekly, ….) is provided in a document of the quality manual. There all the according standard operating procedures and reporting forms are linked by their document ID. This document provides an overview of the QA procedures at the lab. If changes in on of these procedures are made, only the according SOP (SAA) documents for this special procedure needs to be adapted.

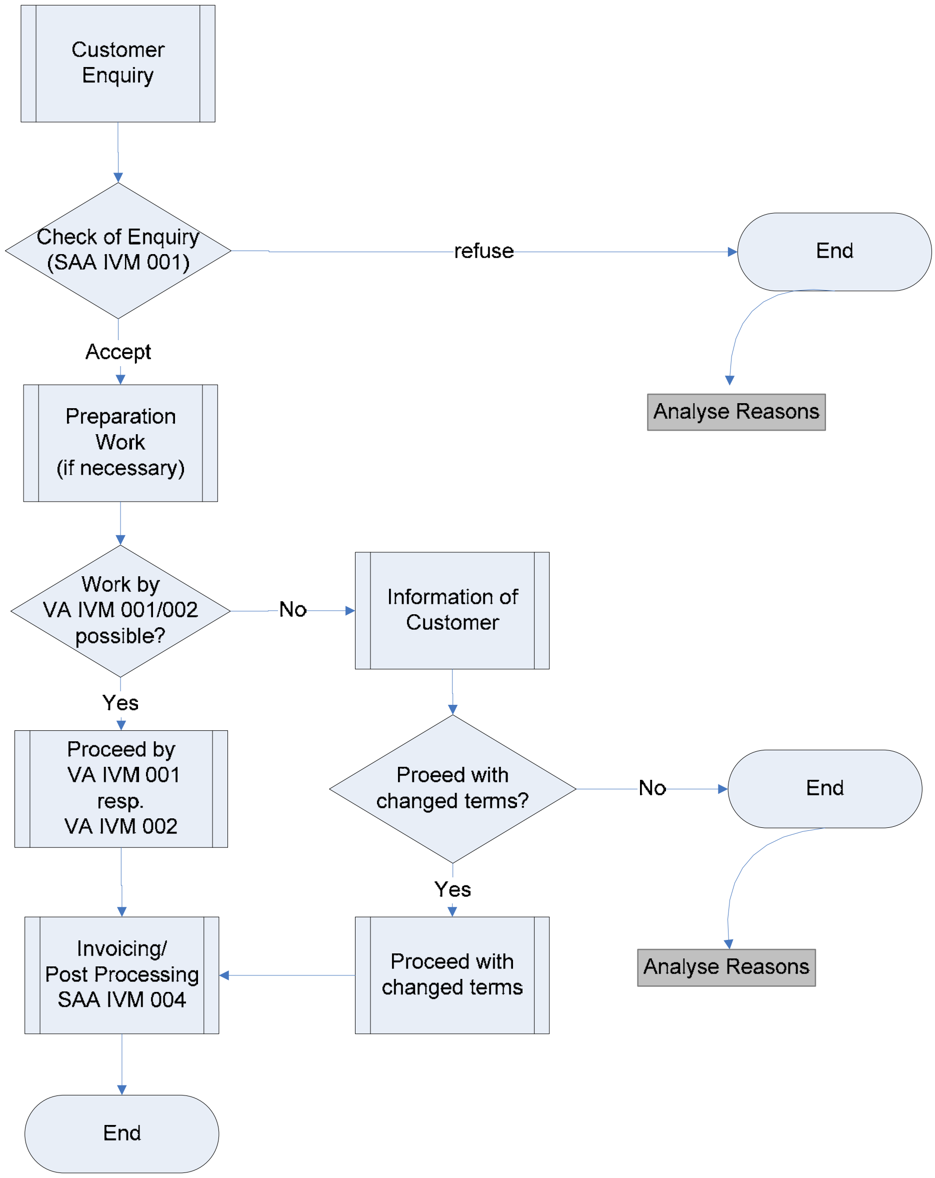

Figure 1. A translation of the flowchart for the generic procedure from customer enquiry to final billing. A short description an the ID of the document providing detailed procedures is given in the VA document.

The whole system including the generic ISO 9001 parts and special documentation of IVM has been set up flexible and could easily be adopted to new situations, like recent organizational changes (e.g. the founding of a new Institute for Radiation Research at KIT to which the IVM was then assigned). Here only minor changes in the existing documentation were needed to adapt the whole system to the new situation.

Conclusions

All the mechanisms required in ISO 17025 chapter 4 are met by the ISO 9001 certified quality management system our accreditation is based on. Thus we "only" needed to add our laboratory specific parts to the system. The experiences from the accreditation of the spectrometry lab in 2006 saved lots of time and discussions to be spent on very basic issues, like e.g. the handling of electronic data. We carefully reviewed and improved our processes during planning and writing the documentation. All processes in the lab are now well documented, the former hidden experiences of the staff are now "publicly" available and are assured against getting lost over time. The quality management documentation now serves as a manual of the work in the laboratory and is under constant review. The discussions and explanations to our quality management team, which was non-expert in the field of in-vivo monitoring, were helpful in putting everything in clear words. Thus our final documentation became more comprehensible for people without in-depth knowledge of the laboratory. The materials developed for the QMS is also used for the education and training of new personnel e.g. the students of Karlsruhe university of cooperative education, who do parts of their practical studies in radiation protection at the in-vivo monitoring laboratory (IVM).

After more than two years of working with the quality management we can say it was worth the trouble and time required in setting it up and gave an overall benefit for the lab. Only a sparse amount of the work time needs to be spent to maintain the running system. It also leaves enough freedom in the daily routine work at IVM and found acceptance by the employees, quickly.

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Setting up of a Molecular Imaging Unit in biomedical research centres

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Abstract

Molecular imaging techniques have become important tools for the clinical diagnosis of several diseases. These techniques are today in a highly mature state in the clinical field and are now being rapidly developed for use in biomedical (preclinical) research. Among currently available techniques are those allowing acquisition of high resolution anatomical images (CT, MRI), while others offer high sensitivity physiological/ molecular imaging (PET, SPECT, optical). However, used separately each technique offers limited information, and therefore the emphasis of imaging applications for research is on multimodal imaging, wherein images from different techniques are combined to yield an image of high resolution and sensitivity.

Although the use of radioactive isotopes in biomedical research is declining overall, their use in molecular imaging techniques is increasing. Currently, the most developed molecular imaging technique in research, and the most significant from the perspective of radiation protection (RP), is the microPET, combined with anatomical imaging, mainly by CT. In order to set up this technique (or others, such as SPECT) in a biomedical research centre, the RP requirements associated with the handling of high energy gamma sources (PET) and X rays (CT) must be met (equipment, shielding, dosimetry, waste management, training, etc.). These measures also need to be evaluated and adjusted to meet the specific requirements of research centres in terms of biosafety, animal health and welfare, etc. This situation thus complicates RP in this kind of facility.

The aim of this study is to briefly describe the most important imaging techniques and their application in biomedical research, and to present an example of the setting up of a unit or laboratory specialized in these techniques in centres dedicated to pure biomedical research (not associated with a healthcare centre).

This study has been conducted by specialists from the RP and molecular imaging fields.

Introduction

Biomedical research addresses increasingly complex problems related to the biochemical processes that occur in living organisms. As in clinical research, medical imaging techniques are excellent tools to study these processes. A molecular image can be defined as a visual representation, characterization, and quantification of biological processes at the cellular and subcellular levels within living organisms without disturbing the system under study. It has been demonstrated that the visualization and quantification of the function of certain organs of laboratory animals by means of molecular images is a very important tool in the study of human diseases, and also for the discovery and development of new drugs and biochemical probes. As a consequence, animal models are being developed on a translational scale for the generation and the development of tools suitable for the diagnosis and/or treatment of this type of diseases.

Molecular Imaging Techniques with implications for radiation protection

Different Molecular Imaging techniques are combined in this single discipline with a common aim: to change the way in which biological research is carried out. Techniques such as, among others, PET, single photon emission computed tomography (SPECT), digital autoradiography, magnetic resonance (MR), MR with spectroscopy, bioluminescence, fluorescence, and echography, are under constant development. Lately, considerable effort has been directed towards the development of these non-invasive high-resolution imaging techniques for their use in small animals. These miniature systems are not an unnecessary fashionable trend, since they have a better resolution and are generally cheaper than their counterparts for clinical use; several systems can be placed in one laboratory and can be shared among different disciplines. Nevertheless, there are still challenges that need to be solved, such as: trying to obtain an image of a mouse that weighs 30 g and compare it with that of a human who weighs 70 kg, differences in size and volume, the spatial resolution necessary to collect useful anatomical and/or functional data, and the time needed to obtain the images (1). In small animal research the main goal is to obtain the maximum signal possible using the minimum amount of molecular probe, and to locate this signal as precisely as possible in terms of temporal and spatial resolution, using a single system that is able to produce a three-dimensional image that simultaneously presents anatomical and functional information (2). Below, we will evaluate each of the imaging techniques that have major repercussions for radiation protection.

Positron Emission Tomography (MicroPET)

By means of PET the metabolic route can be visualized that a given molecule follows after its incorporation into the organism, generally by means of intravenous administration of a radiolabelled drug. Various biological molecules are labelled with positron-emitting isotopes and follow their normal metabolic route, moving to the sites where they are metabolized (2). Throughout their course, and also from the sites were they are stored and eliminated, they emit a radioactive signal that can be detected from the outside. Detection is done by means of a positron camera or a PET camera. At present, the basic radiolabelled drug used in this technique is fluorodeoxyglucose

(FDG), a glucose analogue labelled with Fluorine-18 (Figure 1). This tracer is relatively unspecific and, although it allows to locate and to study in detail the uptake of the molecule in tissues with high glucose consumption (heart, brain, tumours), deposits may also be produced in other sites that are not of interest, such as points of inflammation. Nevertheless, FDG can be obtained commercially very easily. The use of PET is limited by a poor access to other types of radiolabelled drugs specific for other applications (synthesis of proteins and nucleic acids, hypoxia, immunoPET, etc.); even though such drugs have been developed or are being developed, access to them is difficult if one does not possess a cyclotron or does not have one nearby. Drugs radiolabelled with other isotopes (Ga-68, Cu-64, I-124) are also being developed. At the moment the main applications of PET, both in research and in the clinic, are oncology for localizing and monitoring tumours, and cardiology and neurology for the detection of specific pathologies.

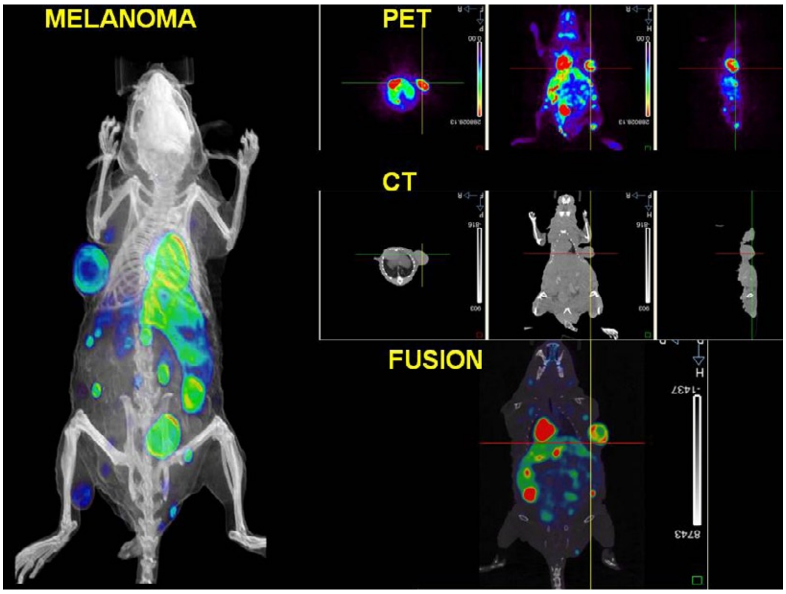

Fig. 1. Image of a study of a transgenic mouse with metastasized melanoma using PET, CT, and a fusion of both. Multiple F18-FDG deposits corresponding to metastases are observed.

Single photon emission computed tomography (MicroSPECT)

SPECT is a technique that is analogous to PET. The difference resides in the type of isotope used, which in the case of SPECT are single photon gamma emitters. Labelling, administration, and acquisition are similar to those for PET, but in this case the technique offers a lower resolution and sensitivity. However, there is a greater variety of tracers and isotopes, since most radiolabelled drugs used for SPECT are obtained using a Tc99m generator, with a longer half life and allowing to work *in situ*. SPECT is applied to the same cases as PET, and in addition for the control of renal and osteoarticular function, but since there is a greater number of approved radiolabelled drugs available, the field of application extends to a larger number of benign pathologies and functional studies.

Computerised Tomography (MicroCT)

The images of computerised tomography are based on the different absorption by tissues when X-rays pass through them. CT is not a technique that provides information at the cellular or molecular level *per se*, and therefore it is not considered a molecular imaging technique. Nevertheless, this technique is of great importance for the anatomical location of functional images (PET and SPECT) and for their reconstruction. This is even more so if one takes into account that the molecular probes are increasingly specific. The radiation deposited in the experimental animals is not insignificant (0.6 Gy/scan, corresponding to 5% of the LD50 in mice), which limits obtaining images in series (3).

Model for applying Molecular Imaging techniques in biomedical research centres

In this part of the paper we will develop a model for the design and management of working with imaging techniques in a research centre not associated with clinical facilities. What is indicated under each point is an option that the authors consider appropriate to be in operation or in phase of development in real centres.

Radioactive Facility

The development of the imaging techniques mentioned above (PET, CT) implies the handling of non-sealed radioactive isotopes and the use of equipment emitting ionizing radiation (RX), which is why, according to the Spanish legislation, it is necessary to have a radioactive facility (RF) authorized by the competent authorities.

In Spain many biological research centres follow a similar RF model. The model includes a central laboratory and authorized areas in the research laboratories. It is a model designed for the use of non-sealed sources both *in vitro* and *in vivo*, although depending on the type of centre and activity it may include equipment emitting ionizing radiation (irradiators, diffractors, imaging equipment, etc.). According to this model, there is a central laboratory designed for the manipulation of high activities (limits of activity per essay from 10 to 30 mCi, depending on the type of isotope) or high-risk isotopes (volatile iodides, high energy gamma emitters). This laboratory is also used for the management and control of the RF and contains the central radioactive waste depots. The RF further includes authorized areas in the research laboratories. These are small areas in each laboratory that requires it, and are specifically placed and equipped for the exclusive handling of limited activities of radioactive isotopes (between 0.5 and 2 mCi per assay, depending on the isotope). In addition to these we may also find areas where emitting equipment is stored; these areas are set up according to the requirements of the equipment (access control, shields, etc.). The Molecular Imaging Unit (MIU) is considered as another authorized area, to which the appropriate measures of control and protection are applied, based on the types of isotopes handled and the techniques developed. The research centres normally have personnel specifically assigned to the management and control of the facility, with an official license (for operation or supervision) granted by the competent authority.

Central Laboratory of the Radioactive Facility

As mentioned before, this laboratory is especially equipped for the development of techniques that imply the handling of high activities or high-risk isotopes. The laboratory should be classified as a controlled area with radiation and contamination hazards. It must have access control and control of atmospheric pressure (negative pressure), as well as activated carbon filtration of all air extracted from the laboratory.

The laboratory has both the basic laboratory equipment and all the necessary means of protection for the development of the established techniques (fixed and movable shields, containers, containment equipment, detectors, emergency devices, etc.).

Since the radioactive waste deposits are included in it, in this laboratory the whole spectrum of registration, preparation, and elimination of radioactive waste or waste material with radioactive contents is realised.

With respect to imaging techniques, in this laboratory the control and registration of the incoming commercial radioactive material is carried out by personnel in charge of Radiation Protection management, and the labelling of tracers is done in situ; these activities are developed by research personnel with specific training in Radiation Protection.

The laboratory has the necessary specific protection equipment to carry out the indicated activities. It must have manipulation cells, movable shields, shielded containers, general laboratory equipment, materials that enable the proper course of action in emergency situations, etc. For manipulation cells, one can opt for mixed and shielded biosafety cabinets. These are conventional biosafety cabinets that have been modified by adding activated carbon filters and the necessary shielding in the outer and frontal panels. They allow working under sterile conditions, and at the same time they protect the operator from radioactive gases or vapours, as well as of aerosols contaminated with pathogenic biological agents that may be used in the research.

This model is being successfully used in various centres.

Molecular Imaging Unit

As mentioned before, a research centre that wants to develop this type of techniques must have a Unit specialized in molecular imaging. The MIU must provide the research groups with the latest imaging techniques. This unit must be in charge of the preparation of the doses, administration to the animals, obtaining the images, and their analysis. It must therefore have specifically trained personnel. Because of the nature of the work done, this unit must be intimately linked to the Animal Facility, ideally being located close to this facility. The unit must also have specific facilities both for handling the animals, which must form part of the Animal Facility, and for the analysis of the images, which should be outside the facility but close to it.

Design of the facility for handling radioactive material and animals

Normally, one starts out with a defined space that must be adapted to the requirements of this type of techniques. Especially with PET isotopes, it is more efficient in economic terms to take advantage of the distances instead of using thicker shields as a means of protection against radiation, or to place the different areas next to low occupation spaces, such as corridors, technical areas, etc. Of course, one should also consider the isotopes to be used and the activities to be manipulated; the latter depends on the study model, since large animals usually require higher activities.

In this space we must establish a room for the imaging equipment, which must fulfil the installation and environmental requirements for operation of the equipment (temperature, humidity, maximum incline, electric power, etc.), and also the requirements for handling animals: space for an anaesthesia table, gas provision, etc. Preferably adjacent to it, there should be a control room from which both the specimen and the equipment can be seen. Close to this area of the facility, to avoid transfers over long distances, a room must be located where the animals are injected and where they will stay during the incorporation of the tracer, and to which they are returned and where they will remain until it can be considered safe to manipulate and house them with the rest of the animals. Normally, this area requires the most shielding of the entire facility and must provide sufficient space for the amount of studies to be done. This is the area with the highest contamination hazard because of the injection procedure and the animal excrements. To avoid transfers of radioactive material over long distances, it is also advisable to locate close to the injection room the area for dose preparation, where the radioactive material from the providing laboratories is received and divided into single doses. The floor of the laboratory must be able to sustain the weight of the of radiation protection material that will be installed in it, which usually is around 1000 kg/m2. Optionally, depending on the characteristics of the isotopes to be used, it may be necessary to have an area for the temporary storage of waste, possibly including animal carcasses until they have decayed. If only PET isotopes are used, storing for 24-48 hrs in wastebaskets may be sufficient. When (e.g. Ge-Ga) generators are used, it will be necessary to have a greater storage capacity.

As a general norm, air of these areas should not be recirculated to the rest of the facility, and the needs of the animals must determine the quality of the air supplied to these areas. All surfaces of the areas described above must be finished in such a way that they allow easy cleaning and decontamination.

Equipment

The area of animal manipulation will have to contain the necessary equipment for the development of these techniques. Below we will list the most important equipment to consider.

- Imaging equipment: select according to the technique to be used. The ideal is to use an integrated multimodal PET/CT or SPECT/CT equipment. At the moment integrated or associated PET/NMR equipment is being developed. The type of studies will determine to a large extent the type of technique and therefore the type of equipment.
- Anaesthesia Table: sized according to the type of study animal.
- Leaded screen for handling the animals, especially rodents. It has to be made in such a way that it can be easily decontaminated.
- Manipulation cell / lead castle: easily decontaminatable, with access for hands and material, shields adapted to the activities to manipulate (30-50 mm Pb is usually sufficient), with an adequate system of air circulation and filtration if the type of isotope to be manipulated requires it, and sufficient illumination. It is very

recommendable to include a properly shielded housing for the ionization chamber of the activimeter.

- Activimeter: appropriate for the isotopes to be used and in the range of the activities handled in the laboratory.
- Radiation Monitor: with a probe sensitive to the levels of design of the facility, typically from 0.1uSv/h to 20 mSv/h
- Contamination Monitor: with a probe suitable for the emission of the isotopes of the facility and with a surface not less than 100cm2.
- Wastebaskets or furnishings for temporary waste storage: easily decontaminatable and with sufficient capacity for the amount of material to be used.

Work norms

For working with PET/SPECT isotopes and the radiation emitting equipment, the general norms for working with radioactive isotopes apply. These include norms regarding the operator: personal protection (gloves, lab coat, etc.), abiding by the norms of hygiene, use of appropriate shields, correct use of the dosimeter, etc.; norms regarding the work area: signposting, order and cleanliness, containment, monitoring, access control, etc.; and norms regarding the surroundings: contamination and radiation monitoring and correct waste management.

Specific norms for PET/SPECT isotopes must also be applied, such as the use of specific movable shields (screens, dose dispenser, syringe protectors, etc.) or the use of means for increasing the distance between source and hands (forceps). Of special importance for labelling animals with PET/SPECT is prior cold training, to carry out the tasks in as little time as possible. Labelled animals must be housed in suitably shielded areas during a period of not less than 24 hours (this can be modified depending on the characteristics of the isotopes used) to allow the isotopes to decay before incorporating the animals again into the colony of the Animal Facility.

To guarantee the correct fulfilment of the norms and hence the protection of the operators, operational monitoring of the facility must be done by means of periodic inspections of control of the fulfilment of the operational norms, both in the central laboratory and in the authorized areas, including the MIU. These inspections include monitoring of contamination, waste management, appropriate conditions of order and cleanliness, filling in of the registers, correct maintenance of the specific monitors, etc. These inspections are carried out by the personnel in charge of the management of Radiation Protection.

Personnel. Functions and classification

The personnel implied in the manipulation of PET isotopes and radiation emitting equipment is as follows (references to degrees and licenses refer to what is established in the Spanish legislation):

- <u>Personnel in charge of Radiation Protection management</u>: Made up of the Responsible of the RF (Supervisor in charge or Head of Radiation Protection), and a team of assigned technicians, all holding an operator license in the application field of the RF at issue. These personnel will be in charge of the reception and registration of the commercial radioactive material and the operations of monitoring and control of the RF.

- Molecular Imaging Unit: Made up of a Person in charge holding a Supervisor license and a team of technicians with operator licenses, all in the field of application of the RF at issue. These personnel will be in charge of the development of the different imaging techniques that involve treating animals with radioactive material or handling radiation emitting equipment, including labelling the animals, the acquisition of images, and their subsequent analysis. This will be the only personnel authorized to carry out this type of techniques.
- Research Personnel: They will carry out the techniques of labelling experimental tracers with PET or SPECT isotopes. Depending on the activities and isotopes handled, they may or may not need to hold an operator license in the field of application of the RF at issue.

As a general norm, all exposed personnel will be considered as exposed workers of category B (according to the Spanish legislation), since it is improbable that they will receive doses above 6 mSv or 3/10 of specific limits in a calendar year. Depending on the animals handled and the doses administered, workers who carry out functions of dispensing and administering radiolabelled drugs may be classified as occupationally exposed personnel of Category A (who may receive doses above 6 mSv or 3/10 of specific limits in a calendar year).

Training

Besides knowledge acquired in training courses for operators or supervisors of Radioactive Facilities, all personnel involved in handling PET or SPECT isotopes must receive specific in-house training. This training will consist of giving different seminars in which information will be included on the general safety norms for working in a laboratory, the handling of radioactive PET/SPECT isotopes (radiological hazard, use of shields, detectors, etc.), operation, and the Emergency Plan of the RF. The personnel of the Imaging Unit dedicated to labelling the animals will receive additional training in handling research animals. These personnel must also first do practical and specific training, developing the techniques of animal handling under the same conditions as when using radioactive material, for example using portable shields (screens, syringe protectors).

Dosimetry

Dosimetry of occupationally exposed personnel of Category A and B must be done by means of individual dosimeters. In our case, we use thermoluminescence dosimeters. The whole body dose must be controlled during the entire working day by means of lapel dosimeters. For the personnel who directly handle the radioactive isotopes, the hand dose must also be controlled by means of ring dosimeters. After an accident an additional control by means of internal dosimetry must be done.

The aim is to keep the doses received by the personnel who handle the animals and the vials or syringes with radiolabelled drugs as low as possible. In the case of handling small animals, whole body dose levels can be obtained that are similar to those of the other exposed workers in research, normally below the public dose (1 mSv/year). Hand doses can be kept at very low levels (100 times below the established annual limit, 500 mSv/year). When working with large animals it is tried to obtain dose levels that are less than those received by the personnel who apply these techniques in the clinical setting, since the doses used for large animals are of the same order as those applied to humans.

Working with research animals. Small and large animals.

Animals used as experimental models for research are, generally, rat, mouse, rabbit, and pig. Mice are the first species of choice in the scientific community for imaging studies. Nevertheless, the study of certain pathologies in mice limits the translation to humans. However, the ease with which genome can be manipulated, the possibility to work with large sample sizes, and the reduced costs make them an ideal species for study. On the other hand, the ease with which they can be handled, the low need for space, and the relatively low doses of radiolabelled drugs they receive in studies simplify the management in radiation protection regarding shielding, housing of the labelled animals, and waste management. In contrast, the use of large animals entails the inoculation of high activities of radiolabelled drugs, to which the difficulty of handling the animal is added, which will have to be anaesthetised and gavaged for the possible collection of contaminated urine (15% of the radiolabelled drug) after injection. Administration of radiolabelled drugs, both to small and to large animals, is best done with the animal under anaesthesia and using appropriate means of protection like shields, or using systems to restrain the animal to reduce the risk of self-inoculation. After the imaging study, there must be a shielded space reserved in the radioactive facility to house the animals until their decay. This module, with peripheral shielding and preferably with individual cells in which movable shields (screen type) must be installed, must have an adequate system for the containment of urine/excrements/ potentially contaminated bedding. Special mention must be made of the handling of urine, which is the major vehicle of elimination of radioactive material. Therefore, a containment system must be implemented adjusted to the volumes and activities used, such as containment tanks or deposits.

Discussion

Molecular imaging techniques are a new and very useful approach in biomedical research. Nevertheless, a technique that is useful for all fields of biomedical research does not exist. The technique must be selected that adjusts best to the needs of each type of disease studied. Multimodality in molecular imaging is the present and the future of these techniques in research and in the clinic. The use of hybrid equipment simplifies and accelerates image analysis and obtaining results.

The use of PET isotopes implies an increased risk with respect to the isotopes used most commonly in research, since it involves handling isotopes with higher energies and higher activities than the ones generally used in research. This becomes even more significant if research is done on large animals, where the doses used are similar to those used in humans. Keeping the doses low is achieved by an appropriate design and application of movable shields, as well as by the specific work norms. Hand doses can also be maintained at very low levels (approximately 100 times less than the annual limits for extremities) by using specific movable shields, by increasing the distance from the source, by avoiding direct handling of the source, and by using accessories (e.g. forceps for the handling of vials) whenever possible. In our case, the facilities for working with large animals (pigs) are in the phase of design and construction, so that we do not have results of the dosimetry of this type of work. The objective is to obtain dose values below those of people working in clinical facilities, using the means indicated above adapted to the type of animal used.

To obtain these results, specific prior "cold" training is vital. For the selection of technicians who are going to do the animal labelling, priority must be given to previous experience in the manipulation of experimental animals, which could be complemented with the appropriate training in radiation protection. Handling laboratory animals and administering drugs to them demand special skills to obtain optimal results (injection in the minimum time possible with the smallest number of attempts and the greatest safety for the operator), that can only be achieved through training and experience. To this the difficulty is added to conduct these operations using specific shielding (syringe protectors). Therefore, the optimal situation would be to have a person with experience in animal manipulation who has sufficient time for prior training in the use of such shieldings. In Spain, training in radiation protection has been regulated for a long time, and it is therefore much more developed than training in animal handling. For this reason it is deemed that it is faster and easier to acquire an extensive and detailed training in radiation protection that complements the experience in animal handling.

Conclusions

At present, PET/CT is the molecular imaging technique that is developed furthest in research, with a greater projection and offering results of higher quality, although other techniques are being developed for specific applications, such as SPECT/CT or PET/MRI.

According to our experience, in the work with small research animals (mice), as far as the dosimetry of the personnel who handle these isotopes is concerned, it is possible to obtain whole body dose levels (lapel dosimetry) similar to the other researchers, which are very much below the annual dose limits of exposed workers and normally close to the public dose levels. Hand dose levels very much below the established annual limits for extremities can also be obtained.

Prior cold training and the selection of personnel are vital to attain the goal of the safe use of these isotopes and to maintain the doses low. It has been observed that it is preferable to prioritize previous experience and training in animal handling, which is then complemented with training in Radiation Protection. Working with large animals complicates the design considerably, and therefore all systems affected must be studied carefully to establish the most appropriate measures to take (shields, housing, control of effluents, etc.) in each case.

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Testing of sealed radioactive sources at BAM

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Abstract
Requirements and test programs for sealed radioactive sources are specified in international standards for safety in transport and in use.

Sealed sources which are approved as special form radioactive material according to the Transport Regulations, IAEA Safety Standards TS-R-1, must be able to withstand mechanical (9 m drop, percussion and bending) and thermal (800°C heat) tests without loss of radioactive content.

The International Standard ISO 2919 provides a set of tests which classifies the sources for their safety in use. Performance tests specified in this standard are temperature (high and low), external pressure, impact, vibration and puncture tests. Each test can be applied at different levels of intensity depending on typical usage.

As a criterion of pass or fail, leakage testing has to be done after each test.

The poster gives an overview of BAM's comprehensive test equipment and experience in testing sealed radioactive sources.

Introduction
Requirements and test programs for safety evaluation of sealed radioactive sources are specified in international safety standards for transport and for use. BAM is the competent authority for approvals for special form radioactive material and for type testing of devices with inserted radioactive sources and possesses a comprehensive test equipment and experience in testing for many years./1/

Regulations and standards for testing

Transport regulations
Special form radioactive material is defined in IAEA –Regulations TS-R-1 /2/ as an in-dispersible solid material or a sealed capsule containing radioactive material. The design must be able to withstand severe mechanical and thermal tests without undue loss or dispersal of radioactive content. As a consequence the predicted hazards after a severe accident are minimised and the transport of a greater activity in a Type A - package is permitted. Tests specified in Para 705-709 of TS-R-1 are

– impact test
– percussion test
– bending test
– heat test

A different specimen may be used for each test. As a criterion of pass or fail, leakage testing according to ISO 9978 has to be done.

Regulations for use

The International Standard ISO 2919 /3/ provides a set of tests which classifies the sealed source for their safety in use. Performance tests specified in this standard are

– temperature test (high and low),
– external pressure test
– impact test
– vibration test
– puncture test
– bending test.

Each test can be applied at different levels of intensity depending on typical usage (Table 1). The criterion for passing or failing is also the leak-tightness.

Table 1. List of test conditions (ISO 2919:1999).

Table 2 — Classification of sealed source performance (5 digits)

Test	Class						
	1	2	3	4	5	6	X
Temperature	No test	– 40 °C (20 min) + 80 °C (1 h)	– 40 °C (20 min) + 180 °C (1 h)	– 40 °C (20 min) + 400 °C (1 h) and thermal shock to 20 °C	– 40 °C (20 min) + 600 °C (1 h) and thermal shock to 20 °C	– 40 °C (20 min) + 800 °C (1 h) and thermal shock to 20 °C	Special test
External pressure	No test	25 kPa absolute to atmospheric	25 kPa absolute to 2 MPa absolute	25 kPa absolute to 7 MPa absolute	25 kPa absolute to 70 MPa absolute	25 kPa absolute to 170 MPa absolute	Special test
Impact	No test	50 g from 1 m or equivalent imparted energy	200 g from 1 m or equivalent imparted energy	2 kg from 1 m or equivalent imparted energy	5 kg from 1 m or equivalent imparted energy	20 kg from 1 m or equivalent imparted energy	Special test
Vibration	No test	3 times 10 min 25 to 500 Hz at 49 m/s² (5 g_n) 1)	3 times 10 min 25 to 50 Hz at 49 m/s² (5 g_n) 1) and 50 to 90 Hz at 0,635 mm amplitude peak to peak and 90 to 500 Hz at 98 m/s² (10 g_n) 1)	3 times 30 min 25 to 80 Hz at 1,5 mm amplitude peak to peak and 80 to 2 000 Hz at 196 m/s² (20 g_n) 1)	Not used	Not used	Special test
Puncture	No test	1 g from 1 m or equivalent imparted energy	10 g from 1 m or equivalent imparted energy	50 g from 1 m or equivalent imparted energy	300 g from 1 m or equivalent imparted energy	1 kg from 1 m or equivalent imparted energy	Special test

1) Acceleration maximum amplitude

Specimens that comprise special form radioactive material may be subjected alternatively to ISO 2919 tests if these are the more severe ones. The ISO 2919 temperature class 6- test with 1 h at 800°C is in any case more severe than the IAEA TS-R-1 heat test with 10 min at 800°C. Special form radioactive material with a small mass may be except from 9 m drop impact. Capsule designs with a mass less than 200 g may be alternatively subjected to the ISO 2919 impact class 4- test, and for designs with a mass less than 500 g the impact class 5- test may be equally acceptable.

Test methods, equipment and experience

BAM cannot test radioactive samples. All tests have to be done with dummy sealed sources, i.e. the source capsule has the same design, same manufacturing methods and is made from exactly the same material as those of the sealed source that it represents, but containing, in place of radioactive material, a substance resembling it as closely as practical achievable in physical and chemical properties.

Impact test (9 m drop)

For impact test of special form radioactive material the specimen should be dropped from a height of 9 m onto a flat unyielding target so as to suffer maximum damage. Because a definite angle of impact could be important for the damage result BAM use a magnetic release device for a momentum free drop. A high-speed camera (image frequency 5.400/s, 1024x1024 pixel) is used to keep track of the impact position.

Impact test 1 m, percussion test, puncture test

For impact and puncture test according to ISO 2919 and for percussion test according to the IAEA regulations for special form radioactive material the same equipment is used by BAM. (Fig.1)

This test device has also a magnetic release system to assure a momentum free drop, and is suitable for hammer masses up to 5 kg. Impact tests with higher hammer masses are performed at the 9 m drop facility.

Most of the different steel hammers for the tests in different classes (Table 1) are available (Fig.2).

For percussion test with special form designs the specimen shall be placed on a sheet of lead and struck by a steel bar of 1.4 kg from a height of 1 m. A fresh flat surface of lead shall be used for each impact. The bar shall strike the specimen so as to cause maximum damage.

As an example, in case of small welded Ir-192 sources used for the brachy therapy the bar should strike the weld-seam with its rounded edge to cause maximal bending stress by pressing the source only partly into the soft lead (Fig. 1 b).

For ISO 2919 impact test class 4 the specimen is placed on a steel anvil with a mass of at least 20 kg, and is struck by a hammer of 2 kg from a height of 1 m. In most cases strains are much higher than in case of the IAEA percussion test, were an only 1.4 kg hammer drives the specimen in the soft lead sheet, excepting cases with additional bending stress as described above. Beside the striking position, also the form of simulated content can have an influence on damage results. For example Ir-192 sources could be filled with one long iridium wire or several shorter iridium-cylinders. Under impact the edges of the cylinders can partly push trough the thin capsule wall.

Steel hammers for puncture test bear a fixed pin with a diameter of 3 mm and a height of 6 mm and can also be attached at the magnetic release device .The mass depends on the test-class and varies between 1 g and 1 kg. If source design has more than one vulnerable area, tests have to be carried out on each of them.

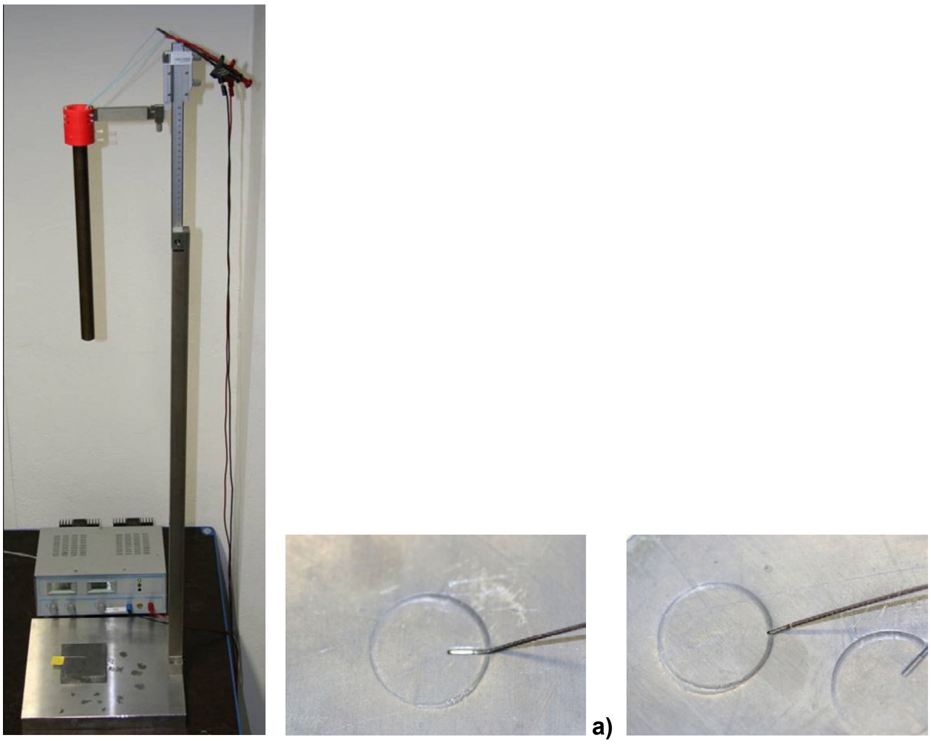

Fig. 1. Percussion test on a Ir-192 source for brachy therapy (1.4 kg from 1 m on a sheet of lead) with different impact positions a) less damage , maximum bending stress in the flexible wire; b) maximum bending stress in weld seam of the source.

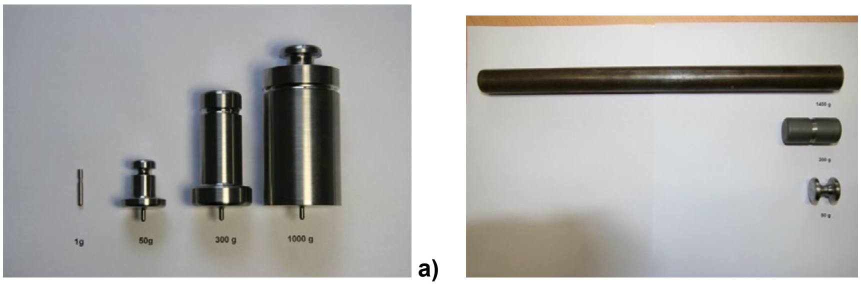

Fig. 2. Examples for steel hammers for a) puncture and b) impact or percussion test.

Temperature tests

An approval for special form radioactive material requires a heat test in air at a temperature of 800°C for 10 min.

Temperature tests according to ISO 2919 include a high–temperature test at temperatures of 80 to 800°C for 1h and a low-temperature test to -40°C for 20 min. The class 6 test (-40°C for 20 min, 800°C for 1 h) as an adequate alternative to the IAEA heat test (800°C for 10 min) is often employed by BAM.

BAM uses for upper temperature test a furnace with a capacity of 10 kW for a maximum temperature of 1200°C to be reached in 2.5 h. Attention should be paid to positioning the specimen in the middle of the furnace and to fitting the thermocouple directly on the specimen surface. In case of designs with a low-melting content, for example distance pieces made of aluminium, an eutectic reaction can happen and can cause a damage of the capsule material and in worst a leakage.

A climate chamber (-80°C until 190°C) is used for low temperature test.

Bending test

Bending tests shall apply only for long, slender sources. IAEA regulations require this for sources with both a minimum length of 100 mm and a length to minimum width ratio of not less than 10. The free end of a rigidly damped specimen has to be struck by a 1.4 kg steel bar. BAM uses a common clamping tool for fixing the specimen and a tripod to adjust the drop height.

Bending tests according to ISO 2919 shall apply to sealed sources having a length to minimum width ratio of 15 or greater. Test procedure is different from this described above. BAM has up to now never carried out this ISO bending test.

Vibration test

BAM features an electro-dynamic shaker with up to 26 kN peak force and frequencies of 5-2000 Hz with acceleration and force control for performing the vibration test according to ISO 2919. Up to now none of the tested sealed sources have shown any signs of damage after these tests done by BAM. Often applicants abstain from this test.

External pressure test

A hydraulic power unit with a maximum operating pressure of 345 MPa , operating medium water, is used for the external pressure test, therewith BAM is able to carry out also tests up to 170 MPa .

Leakage test as criterion for pass or fail

Compliance with all mechanical and thermal tests has to be determined by the ability of the sealed source to maintain its leak-tightness after each test performed.

BAM uses non –radioactive procedures based on a relationship between volumetric leakage rates and loss of radioactive material with test procedures according to the International Standard ISO 9978 /4/.

A volumetric leakage test is only applicable if there is a minimum void within the capsule (Table 2). In particular in the field of medicine sources become smaller and the need for application of volumetric leak test methods with very small void increases.

BAM possesses comprehensive equipment for performing various kinds of helium tests, vacuum bubble and nitrogen bubble tests.

Table 2. Minimum void in capsule for different volumetric leak test methods.

Leak test method	Minimum void in capsule [mm³]
Vacuum bubble	
- glycol or isopropyl alcohol	10
- water	40
- Pressurized bubble with isopropyl alcohol	10
Liquid nitrogen bubble	2
Helium pressurization	10

Conclusions

The International Standards ISO 2919 and the IAEA Safety Standard TS-R-1 provide a set of mechanical and temperature tests for assessing the safety of sealed radioactive sources in transport and use. BAM possesses comprehensive knowledge, experience and equipment for performing nearly all of these tests.

Since BAM is the competent authority for approval of special form radioactive material tests according to the IAEA regulations have always the priority. But depending for their work load test labs are always ready to carry out also the classification tests according to ISO 2919. About four weeks after incoming of specimens, a test program for one source design can be finished in general.

BAM actively works in further development of test methods and regulations and its' representatives take part in national and international working groups and boards dealing with this topic.

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/4/ International Standard ISO 9978:1992, Radiation protection-Sealed radioactive sources-Leakage test methods

Radiological criteria's for patients discharge following a radionuclide therapy or brachytherapy with implanted sealed radionuclide sources

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Abstract

Dose criteria for limitation of exposure incurred by persons helping the patients or living with patients discharged from hospitals following radionuclide therapy or brachytherapy with implanted sealed radionuclide sources have been proposed for national Russian regulation. By means of a conservative dosimetry model, the values of operational radiological criteria for patient discharge from hospital are substantiated basing on the standards of permissible effective dose for population – 1 mSv and for persons helping the patient or living with him – 5 mSv. Two sets of whole body activity for radionuclides I-125, I-131, Sm-153 and Re-188 used in Russia for therapy, as well as dose rate near patient body were received. The smallest one was included in the new Russian Standards for Radiation Safety (SRS-99/2009). Observance of suggested criteria will ensure radiation safety of people in near environment (family, close friends et al.) of the discharged patient.

The new ICRP recommendations and the role of stakeholders

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First lessons from the Montbéliard Radiation Protection Pilot Project: towards the development of a local expertise in radiation protection

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Abstract

Since 2004, the Montbéliard Urban Community (CAPM), in cooperation with CEPN, is engaged in a Radiation Protection Pilot Project whose main objectives are to improve the radiation protection of the inhabitants of the Montbéliard Country and to promote the creation of a pole of competence in the field of radiation protection in the territory.

This Project is based on a global approach addressing all aspects of radiation protection. It is divided into five sections: three ones are related to exposure situations (medical exposure, exposure to radon, exposure in case of nuclear or radiological events), two others concern the development of the radiation protection culture among the population.

The Project is designed to promote active participation from various stakeholders at the local level, working in cooperation with national and international experts. In particular, local actors work in close relationships with major radiation protection experts in France: namely the French Institute of Radiation Protection and Nuclear Safety (IRSN) and the French Nuclear Safety Authority (ASN). Besides, CAPM has entrusted CEPN with a number of tasks to help manage and facilitate the project.

In 2009, the main partners of the Radiation Protection Pilot Project decided to analyse the feedback experience from this initiative. This paper focuses on the main results and perspectives of the Project and presents the first lessons that have been drawn. In particular, lessons about the possible involvement of territories in radiation protection management are developed. Conditions needed to favour a local expertise in radiation protection are notably discussed.

Introduction

Everyday, the public is exposed to radioactivity from various sources – radon, medical practices, releases from nuclear facilities… Such exposure, no matter how significant, raises questions and sometimes gives rise to worry or even anxiety. The populations concerned and the professionals and local authorities responsible for risk prevention and management often find it hard to assess the impact that such situations can have. This is largely because they are not involved in the management of their "radiological situation" and have insufficient knowledge of basic principles of radiation protection

That being the case, how can members of the population and professionals working at the local level gauge risk levels and determine what steps need to be taken to control these forms of exposure? How can these people be associated with the assessment and management of radiological risks, with the support of experts and authorities? What can be done to promote a shared radiation protection culture that will allow people to contribute to their own protection?

Based on these observations and questions, local experts in radiation protection from the Montbéliard territory decided in April 2003 to look into the possibility of developing a radiation protection skills centre within the Montbéliard area. A Radiation Protection Pilot Project was then set up in March 2004 with the objectives to improve the radiation protection of the inhabitants of the Montbéliard Urban Community and to promote the creation of a pole of competence in the field of radiation protection in the territory. Considering the absence of nuclear installations in the region, this decision reflected a strong political engagement from the local authorities to address risks related to radiation exposure.

In 2009, five years after the beginning of the Project, it was decided to analyse the feedback experience from this initiative. This paper focuses on the main results and perspectives of the Project and presents the first lessons that have been drawn. In particular, lessons about the possible involvement of territories in radiation protection topics are developed (Bataille et al., 2010).

Structure of the Radiation Protection Pilot Project

The Radiation Protection Pilot Project is based on a global approach designed to tackle all aspects of radiological risks and focuses on any form of radiation exposure, from natural, industrial or medical sources. It is implemented according to thematic sections related to different fields of radiation protection:

- Analysis and management of the risk associated with radon,
- Radiation protection in the medical field,
- Management of radiological risks in the event of a nuclear or a radiological emergency situation,
- Training in radiation protection in the medical field,
- Development of a scientific and technical culture in radiation protection.

The Project is designed to promote active participation from various stakeholders at the local level, working in cooperation with regional, national and international experts. In particular, local actors work in close relationships with major radiation protection experts in France: namely the French Institute of Radiation Protection and Nuclear Safety (IRSN) and the French Nuclear Safety Authority (ASN). The Montbéliard Urban Community (CAPM) coordinates the Project and has entrusted CEPN with a number of tasks to help manage and facilitate the project.

Main results and perspectives of the Radiation Protection Pilot Project

During the last five years, numerous actions have been performed in the framework of each of the sections of the Radiation Protection Pilot Project. Main outputs and perspectives are briefly summarized below.

Analysis and management of the risk associated with radon

The Montbéliard Urban Community is part of the Doubs department, which has been defined by French authorities as a priority area for the control of radon (JO, 1999). In this context, CAPM has wished to launch a specific "radon" section to learn more about radon-related risks on its territory and to initiate any protective action considered as necessary. This section began in 2005 as part of a broader objective aiming at improving the air quality in dwellings.

Work in this section began with awareness campaigns targeting local players likely to be involved in radon measurements as well as the inhabitants of the Montbéliard Urban Community (articles in the local press, distribution of brochures on radon). CAPM then decided to benefit from a "radon map" of its territory to obtain a clear idea of the scale of radon risk. Several measurement campaigns of radon concentration were also performed from the end of 2006, first in dwellings and then in public buildings.

CAPM made the choice to develop its own measurement capacities: both campaigns were performed by inspectors of the CAPM Health Office who were trained by IRSN. Local elected representatives were particularly involved in the organisation of the measurement campaigns: they were solicited to contact inhabitants and owners of public establishments and participated in communicating results of measurements. Through these campaigns, all involved actors gained knowledge and competences in radon risk management.

This approach allowed CAPM benefiting from a better knowledge of the radon risk in its territory and identifying buildings where radon concentration was too high. CAPM has then engaged several actions to help owners to implement remediation actions with the support of experts from IRSN and the Swiss Office of Public Health.

Today, the section "Analysis and management of the risk associated with radon" is continuing through new measurement campaigns that will allow completing the mapping of the territory. Otherwise, it is envisaged that CAPM takes part in a working group whose objectives would be to analyse feedback experiences from local French communities that have decided to deal with radon risk. The aim could be to produce guidance for others communities that would like to initiate a similar approach. Finally, the Montbéliard territory will host the next "radon workshop" organised by the French Society of Radiation Protection (SFRP) in Spring 2011.

Radiation protection in the medical field

This section was launched in 2004 by the Montbéliard hospital management team. It was organised to address problems raised by new regulatory requirements applying to medical physicists and radiation protection experts in two areas: the organisational aspects of radiation protection and medical physics and the optimisation of radiation protection of patients.

In particular, the new requirements imposed the need to have a medical physicist on hand in the nuclear medicine and diagnostic radiology departments (JO, 2004a) and to consult a radiation protection expert (JO, 2003). This triggered the idea of exploring the possibility of setting up a unit dedicated to medical physics and radiation protection. As part of this work, the Montbéliard hospital solicited CEPN to gather and analyse feedback experiences from hospitals where such units were in place. Conclusions of this study are still discussed (Badajoz et al., 2008).

As far as radiation protection of patients is concerned, the Montbéliard hospital and CEPN have participated in a working group created by the French-speaking Swiss Society of Radiation Protection since 2009. The objective is to elaborate new documents to improve information of patients undertaking medical procedures using ionising radiations.

Even if the works described above gave interesting results, the staff from the Montbéliard hospital had difficulties to devote time to the Radiation Protection Pilot Project. The partners of the Project also decided to extend this section beyond the hospital. In this context, the Montbéliard Urban Community now intends to develop, with the help of CEPN, the "Medical Radiation Protection School of the Montbéliard Territory". This project will mainly aim at:

– Supporting initial and continuing radiation protection training and research,
– Informing health professionals and patients,
– Developing a radiation protection culture among health professionals and the public.

The "Medical Radiation Protection School of the Montbéliard Territory" will be opened to any interested structure, in particular:

– Local actors involved in the "health" domain in the Montbéliard territory,
– Regional, national and international institutes/organisms involved in radiation protection in the medical field.

Managing radiological risks in the event of a nuclear or a radiological emergency situation

The objectives of this section are to understand the local stakes associated with the management of a radiological accident and to develop practical actions to improve accident and post-accident preparedness in the Montbéliard territory. This section is implemented from a multiple risk perspective, since the Montbéliard Urban Community has to draft Local Community Emergency Plans (as required by the French law; JO, 2004b) taking into account any risk that could affect its territory.

In order to identify the potential consequences of a nuclear accident or a radiological event on its territory, the CAPM, with the technical support of CEPN and KIT (Germany), is using the Decision Aiding Tools developed over the last decades within the European Framework Programme for emergency and rehabilitation preparedness and management. Simulation results are then coupled with data provided by the CAPM Geographic Information System (GIS), which allows determining practical actions to be better prepared to a radiological event.

To date, first studied scenarios showed that the Montbéliard territory could be quite affected. In case of a medium accident at the Fessenheim Nuclear Power Plant (located at 70 km from Montbéliard), the territory would have to implement foodstuffs restrictions on some products (milk, leafy vegetables) during limited periods of time. In case of a severe accident, it would have to set up emergency countermeasures. Finally, an accident of a truck transporting a caesium-137 source would have very localised consequences, mainly in the nearest agricultural areas and rivers.

As a consequence, the Montbéliard territory engaged some practical actions to be better prepared to a radiological event. Buildings that could accommodate populations that would have to be evacuated have been identified. Their location and characteristics have been entered in the GIS of the Montbéliard territory. An assessment of stocks of iodine tablets, available on the territory, has also been performed.

Today, CAPM is pursuing this project. It will notably take part in the development of the French doctrine on post-accident management (CODIRPA) managed by ASN and become a member of the European Platform on Preparedness for Nuclear and Radiological Emergency Response and Recovery (NERIS), recently launched.

Training on radiation protection in the medical field

This section has been initiated and managed by the University of Franche-Comté with the objective to promote links between training, public research and local competences. Because of the lack of personnel in radiation protection in hospitals in France, the proposed trainings sessions mainly concern radiation protection in the medical field. Two training courses have thus been created:

– A professional degree in medical dosimetry and radiation protection (DORA) was created in 2005. 15 to 20 dosimetrists are trained each year.
– A training module on radiation protection of patients was implemented early 2009, addressing radiographers.

Through the creation of these training courses, the “Training” section allowed a densification of links between the University and IRSN. Today, the University would like to implement a more formal collaboration.

Otherwise, the University envisages to enlarge its training sessions and to develop a “medical radiation protection” option for existing radiation protection masters. All these trainings will be integrated in the “Medical Radiation Protection School of the Montbéliard Territory” described above.

Developing a scientific and technical culture in radiation protection

Drawing on its long experience in promoting scientific culture, the Pavillon des Sciences de Franche-Comté was the driving force behind the implementation of a “Scientific and Technical Culture” section. The objective is to raise the profile of a scientific and technical culture related to radioactivity and radiation protection among the general public and, more specifically, among pupils in the Montbéliard area.

A good example of this section’s activities is the elaboration of the exhibition entitled “Did you say radiation protection?”. This exhibition is the result of a collaboration between artists and radiation protection experts who have produced a number of works of art as well as films that are screened on information terminals. The exhibition was opened in Montbéliard in October 2007. Since this date, it travels all around the world and has already welcomed more than 20,000 visitors.

Since 2007, the work of this section is enhanced through the organisation of “radiation protection workshops” in secondary schools. The goal is to involve pupils in activities approaching radiation protection topics in a concrete manner and favouring an understanding of radiation protection stakes in daily life. The workshops are animated by teachers of the secondary schools in partnership with local and national experts in radiation protection. They favoured a multidisciplinary approach: radiation protection is dealt with through physical and biological sciences, but also through history, philosophy or plastic arts. Every year, all the participating schools gather in a final seminar at the end of the scholar year. The last edition took part in March 2010 in Paris and gathered more than 200 pupils, including children from 7 French schools and 7 foreign schools.

First lessons learned from the Radiation Protection Pilot Project

In addition to the results obtained in the different sections, the activities carried out as part of the Pilot Project have led to a real improvement in the skills of local players involved in various aspects of radiation protection. For example, CAPM employees have acquired new skills and widened their range of expertise: they were trained to radon measurement and were taught to handle simulation tools to be better prepared to radiological events.

Thus, today, several sections of the Radiation Protection Pilot Project have reached a "state of maturity":

– Results acquired in the sections "Analysis and management of the risk associated with radon" and "Managing radiological risks in the event of a nuclear or a radiological emergency situation" allowed a national and international recognition of the competences developed in radiation protection within the territory. Activities will now continue in relationships with national players, particularly IRSN and ASN.
– Activities initiated within the section "Developing a scientific and technical culture" are now developed beyond the frame of the Radiation Protection Pilot Project in an autonomous manner. This section constitutes a major success of the Project and showed that this initiative was answering to a real lack of information and dialogue opportunities on radiation protection topics and stakes.

Besides, other sections, notably the one related to "Radiation protection in the medical field", got restricted results between 2004 and 2009. Today, they benefit from a renewed context and should be further developed from 2010 through the creation of the "Medical Radiation Protection School of the Montbéliard Territory".

Although the experience of the Radiation Protection Pilot Project is not ended, first lessons about the development of a local expertise in radiation protection can be proposed. Indeed, the Project has allowed identifying some elements needed for the development of local competences in radiation protection, among which:

– The importance to benefit from local radiation protection experts, capable of initiating and guiding the various actions. Each section was indeed piloted by a local expert who was involved in the concerned field and benefited from a deep knowledge of local characteristics.
– The necessity to link radiation protection actions with local stakes or projects. For instance, we can remind that: the "Radon" section was initiated as a part of actions concerning the air quality in dwellings; the section about "Managing radiological risks in the event of a nuclear or a radiological emergency situation"

was accomplished in a multi-risk perspective as part of drafting Local Community Emergency Plans.

– The necessity of a definite commitment and support of the local elected representatives. Every year, since 2004, the state of advancement of the Project is presented to local elected representatives. To date, they have always expressed their unanimous support to the Project.
– The importance of a strong articulation with national and international radiation protection experts. Indeed, it is important to underline that the Project could not have been developed in this way if it did not benefit from interactions and help of national and international radiation protection players.

If the experience of the Montbéliard Urban Community has allowed to show that the development of a local expertise in radiation protection is possible, it also allowed to demonstrate that it can lead to convincing results: the different actions performed at the local level have been very well received by the inhabitants of the Montbéliard territory and were all the more efficient. For instance, as far as the management of the radon risk is concerned, owners of houses are listening to advice of the staff of CAPM, because they know them and trust them. Similarly, in the course of the presentation of the exhibition “Did you say radiation protection?” in Montbéliard, visitors were sensitive to the information which, mostly, made echoes to local realities.

However, as mentioned above, it remains to underline the limits of a local expertise related to radiation protection issues: indeed, actions developed by the stakeholders of the Montbéliard territory were sometimes slowed down by a lack of means (equipment to measure radon) or of knowledge (about regulation relating to distribution of iodine tablets). They were also sometimes confronted to public policies that are not well adapted to local actors expectations.

All these elements emphasize the necessity to work on the modalities of articulation between local and national expertises in radiation protection (adaptation of language, definition of roles of each actor…) and on the needs of mediation/facilitation between these two levels, which appear complementary. Drawing on this observation, CAPM and IRSN, with the support of CEPN, have recently decided to officialise their cooperation through the signature of a specific protocol. They engaged to focus their efforts to articulate at best their respective actions and to contribute to a general improvement of radiation protection in the fields already covered by the Radiation Protection Pilot Project.

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Involvement of local stakeholders in the long-term surveillance of radioactive waste disposals

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Abstract
The sustainability of radioactive waste disposals' surveillance systems is a key factor when it comes to long-term surveillance of these facilities. Local stakeholders have shown concerns on this matter, especially when dealing with intergenerational transmission of knowledge and memory of installations.

As a European cooperative research program, COWAM in Practice (CIP) addressed these issues from a practical point of view. From 2007 to 2009, CIP involved a wide range of stakeholders from five countries, including members of NGOs and industry, elected officials, national experts and authorities.

Among CIP outputs were the local stakeholders' expectations regarding long-term surveillance, and considering two main issues: on the one hand the long-term environmental and health surveillance, on the other hand the practical implementation of "reversibility" for geological disposal.

The paper addresses the governance issues associated with the practical implementation of long-term surveillance systems. A sustainable surveillance system, involving local actors in the decision making process, should allow them to monitor the waste inventory, the disposal evolution, the implementation of 'reversibility', the potential environmental and health impacts as well as the financial aspects. Beforehand, an effective involvement requires the identification of concerned local stakeholders, their technical and political empowerment, as well as establishing cooperation between the local and national institutions involved in the surveillance.

The radiation protection community has an important role to play to support local stakeholders' involvement in long-term surveillance, mainly in ways of training, skills and capacity building, but also by taking into account public concerns in the disposal design and for the organisation of its follow-up. Such a support would contribute to improve the sustainability of the surveillance system as well as that of the vigilance.

Introduction

Radioactive waste introduce a new time dimension in the field of risk management. The radioactive period of some radionuclides leads to consider time scales ranging from several thousand to several million years. These time scales imply adapting risk management processes to long or very long-term. Reflections have been led over the past decade on safety as well as societal issues associated with this long-term risk management. In fact, the process of long-term waste management generates new responsibilities that the current generation has to hand over to future generations for pursuing this management, maintaining and organizing a sustainable monitoring system and preserving the waste memory. Although the operator of the waste disposal has the responsibility of developing and implementing a robust monitoring system, the development of a citizen vigilance is a key element for improving the sustainability of the surveillance over the long-term period. This citizen vigilance is notably characterised by the handover of responsibilities of the current generation to the following ones.

Within European projects COWAM[1] and more particularly COWAM in Practice (CIP), dedicated to the improvement of radioactive waste management's governance, working groups involving experts, authorities, waste managers and local stakeholders were set up to discuss the stakes associated with long-term inclusive governance in this field. CIP cooperative investigations helped notably to elaborate local stakeholders' expectations regarding arrangements for long-term vigilance, in the light of two major concerns: first, protection, now and over time, of health and the environment around waste installations; second, practical implementation of 'reversibility' for geological disposal.

This paper presents the main results of these investigations with regard to the involvement of local stakeholders in long-term surveillance systems and discusses some challenges for the radiation protection community to improve and facilitate this involvement, notably through their participation to the building of stakeholder competences in terms of radiation protection.

Long-term surveillance of radioactive waste disposal

Whatever the type of radioactive waste management facility (geological disposal, short-term or long-term storage), the generic term of 'surveillance' can include several aspects of the protection system, which may also vary with time, such as:

- The technical monitoring of the facility environment (control of radionuclides discharges in the environment);
- The technical maintenance of the facility, the reassessment of the safety level over time and the management of any actions on site, including possible retrieval of waste;
- The surveillance of human activity in and around the site;
- The preservation and transmission of know-how concerning waste management;
- The organisation of a multi-level vigilance;

[1] COWAM 1 (2000-2003), COWAM2 (2004-2006), CIP (2007-2009). More information on : www.cowam.com

- The training of the next/new generation(s) who will take over the radioactive waste management facility site.

As it has been shown in the COWAM 2 project, there is no unitary definition of 'long-term'. What 'long-term' means largely depends on the chosen perspectives – who defines it, what the context is, what it is defined for. Therefore, it is essential to first delineate what is at stake in terms of time dimension when dealing with long-term surveillance. Regarding radioactive waste management, two long-term perspectives may be considered: a technical perspective and a societal perspective.

From a technical point of view, long-term is a concern for the operators of radioactive waste disposals, the safety authority and radiation protection community as there is a need to assess the performance of protection systems of the waste disposal over periods of time of about several thousand of years and beyond, that is assessing notably:

- The geological evolution;
- The robustness of waste package and disposal;
- The evolution of radioactivity and thermal characteristics of the waste packages with time, its potential transfer to the environment on long-term periods and the associated safety criteria.

However, from the societal perspective, considering timescales of the order of several thousands of years is meaningless. It is not possible to envisage how the society will be organised in the far future. A reasonable approach to cope with the long-term duration of waste radioactivity is, for the current generation, to create management and governance processes favouring a continuous transmission to the next generation(s) of a "safety heritage" (composed of know-how, protection options, procedures, resources…) in order to ensure the continuity of waste management.

These management processes may evolve with time, but the current generation needs to consider how they can be set up in order to achieve a number of 'missions' to be transmitted to the next generation(s). It will be the responsibility of the next generation(s) to continue and/or reconsider these processes and to adapt them with the aim of ensuring the realisation of these different missions.

Citizen vigilance and local stakeholder involvement in the long-term surveillance system

The COWAM 2 project has shown that the development of citizen vigilance is a key factor for improving the sustainability of surveillance systems over the long-term. This development is achieved through the involvement of local stakeholders in the surveillance system in order to:

- At the planning stage, address their questions and concerns and find a mutual understanding of the issues at stake;
- Identify potential improvements of the monitoring system and management of the nuclear waste installations;
- Favour in this way citizen confidence in the "institutional" monitoring system;
- Work out the specific contributions that citizen vigilance can make, identify the practical aspects of implementation, and, if it is necessary, build up local structures to play this role.

The cooperative research program COWAM in Practice (CIP) allowed revealing the stakeholders' expectations regarding their practical involvement in the surveillance system. The main aspects of interest concern the following aspects:

– Follow-up of the waste inventory;
– Surveillance of the integrity over time of the disposal system (waste packages, installation…);
– Health and environmental surveillance;
– The decision-making process associated with the practical implementation of reversibility for a deep geological waste disposal.

It has to be noticed that the later topic about the role of reversibility on the long-term surveillance was particularly discussed within the CIP project, as the 2006 French law on radioactive waste management asks for a period of reversibility for deep geological disposals.

Follow-up of the waste inventory and the integrity of the disposal system

The issue of keeping an inventory of all radioactive waste contained in the disposal facility is raised at various levels, including:

– From an ethical point of view, ensuring that the memory of the inventory is kept alive over the course of time is seen as a duty of today's generation toward future generations, who have a right to know of their legacy;
– With regard to risk management, it is essential to know with sufficient accuracy what the content of the disposal facility is in order to be able to assess the level of protection provided by the systems according to the health and the environmental aspects.

Special tracking systems are set up by waste management operators to follow-up the content of the disposal facility, throughout the entire operational phase and subsequently to maintain this knowledge in the long-term to ensure that it is passed on to future generations. For example in France, national follow-up of radioactive materials and waste has been mandatory since the 2006 law (JORF, 2006). It gives rise to the elaboration by a pluralistic group of a 'National Plan for Management of Radioactive Materials and Waste' (ASN, 2006).

Beyond the national follow-up, to reinforce the sustainability of this system, the involvement of local stakeholders in the follow-up of the waste inventory is of importance as it creates a 'local knowledge' of the facility, easing the transmission to the future generations. For this purpose, a specific pluralistic body, involving producers of radioactive waste, the disposal facility operator, representatives of institutional and non-institutional actors (safety authorities, expert bodies, associations, etc.) as well as local actors (local associations, elected people…) could be created, allowing the sharing of the information on a regular basis.

For a better efficiency of this local follow-up it appears necessary to create dialogue opportunities between the various actors in charge of elaborating the waste inventory system and the local stakeholders in order to 'co-construct' meaningful indicators. This may be done within a National Plan for Management of Radioactive Materials and Waste, as implemented in France. This means, on the one hand, to facilitate the understanding by the stakeholders of the criteria used by the operator (or authorities) for checking the inventory of waste packages and, on the other hand, to allow these stakeholders to address their expectations concerning indicators making sense for them which could be added to the classical surveillance plan. The followed indicators might cover not only the radiological content of packages but also data on the potential presence of chemicals, package design, how they are positioned in the facility, etc.

The relevance of the followed indicators with regard to long-term surveillance can also be increased by putting them into perspective with the inventory of existing (or future) radioactive waste facilities in their own country and abroad[2].

Another ‘technical aspect’ of the surveillance system concerns the follow-up of the integrity over time of the disposal facility (including waste packages, installation, geological surrounding evolution, etc). Here also, the involvement of local stakeholders will contribute to improve their understanding and knowledge of the waste facility, favouring their participation to the long-term vigilance. This aspect of the surveillance implies to set up a reference point of the disposal system, characterised by a set of indicators that would enable to compare the state of the installation to the evolution planned at the design stage of the disposal. Such follow-up might also be an opportunity to reduce the uncertainties inherent to the predictions of long-term modelling tools.

This follow-up leads questioning related to the diffusion of a radiation protection culture, that is, how to share the assessments made by safety authorities with the local stakeholders. One should note that the meaning of the criteria to be used is currently discussed between ICRP and the radioactive waste management community. Such a discussion will have to be also open to local stakeholders to share a common view on the evaluation of the robustness of the waste management system.

Health and environmental surveillance

Within the CIP project a strong demand on the part of local actors emerged concerning potential health or environmental impacts around nuclear waste installations. In fact, the development of a surveillance system on health and environment can favour the local and national vigilance on the long-term evolution of the site and will provide an assessment of the level of protection achieved for the current period. The follow-up will allow to assess the evolution of this level of protection overtime. It is also an opportunity to define a reference point which could be used in case of incidental situation.

According to the stage of development and operation of the nuclear waste installation, various levels of involvement for the local stakeholders in this surveillance system can imagined: they could participate in the definition of the reference point before the creation of the installation, then key milestones need to be identified in the long-term surveillance programme for which they may be involved and develop more intensive actions.

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2 For example, during the CIP project, stakeholders have shown great interest in collecting feed-back on the closed disposal of low and intermediate level radioactive waste in France (Centre Manche) as well as the currently operating one (Centre de l’Aube). It can be noticed that good practices and difficulties – and even failures – are equally interesting to learn from past experiences to identify key issues and prepare better technical and governance approaches..

It has to be recognized that health and environmental impacts are quite difficult to evaluate. Among the emerging questions are the following:

– How to estimate the long-term evolution of environment and health?
– How to set up a global approach on health impacts which includes the other nuclear installations and the other exposures to ionising radiation (natural sources and medical exposures)?
– How to identify other causes of health or environmental impacts (potential presence of other industries, presence of chemical releases, pollutions,…)?
– How the results of the waste inventory will have an impact on the surveillance of the integrity of the facility and the waste package?
– How to put the estimated potential impacts into perspective with the regional, national and international situation concerning evolution of health?

To cope with these questions, the local stakeholders expressed a clear interest to establish cooperation / partnership with national institutes in charge of health and environmental surveillance. The creation of these partnerships would contribute to address their concerns and to find a mutual understanding of the issues at stake, to identify indicators making sense for the local stakeholders and to identify potential improvements in the surveillance system and management of the nuclear waste installations.

It was also recognized that it is essential to have access to pluralistic expertise (from national and international levels) when characterizing the reference points (for health and environment) and elaborating the surveillance system. Due to all the difficulties inherent to the impact assessment methodologies (uncertainties, monitoring results interpretation, statistical limits, …), it is thus essential to confront experiences and to share with all the concerned actors (waste operator, authorities, national experts, local and national stakeholders, ….) elements aiming at a better comprehension of these methodologies, their advantages and limits, in view of an evolution of the methodologies if it appears to be necessary.

The decision-making process associated with the practical implementation of reversibility for a deep geological waste disposal

As indicated above, in the French context, any geological disposal of radioactive waste must be designed in order to respect a period of reversibility of at least 100 years. This period of reversibility may have an impact on the long-term surveillance as the storage is maintained ‘opened’ and the waste management process may evolve during this period of time.

Discussions between experts and local stakeholders within the CIP project lead to the proposal that, from a practical point of view, the aim of maintaining a period of reversibility is to be able to choose, throughout this entire period, between three major options: continue to maintain reversibility, retrieve packages or initiate the closure of all or part of the disposal facility (see fig.1). This proposal allowed to shape issues associated with the practical implementation of reversibility, which are now discussed with the waste operator and the authorities.

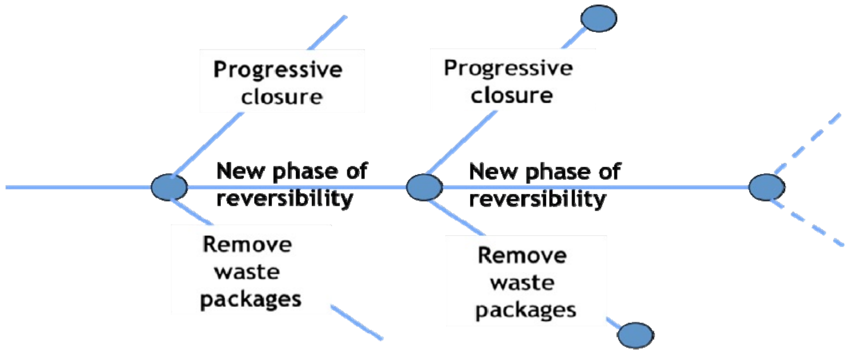

Figure 1. Illustration of the decision-making process throughout the period of reversibility for a geological radioactive waste disposal facility.

The issue of the identification of decision-making criteria for choosing which option to implement is not discussed here. While a first set of criteria will be proposed by the current generation, it will be down to the future generations affected by these decisions to consider or modify these criteria and potentially to draw up other decision-making criteria in view of the context in the future. It is however essential to determine before the start of the operating period what could be the structure of the decision-making process associated with these options: Who will be involved in assessing the situation? Who will make the decision? How often will the situation be reviewed? etc.

There is strong demand on the part of local stakeholders to be involved in the decision-making process associated with reversibility options. They are not asking to take the decision themselves but to be involved in assessing the disposal facility and ensuring that their expectations, especially insofar as they concern decision-making criteria, are effectively taken into account. The involvement of local stakeholders in actually drawing up the assessment and decision-making processes will also serve to make it more effective and, thereby, make the decisions taken more acceptable and sustainable.

The period of reversibility can also be seen as a period during which the disposal is maintained 'opened', giving the opportunity for a close monitoring of the packages, structures and the environment. This period is of particular interest for local stakeholders insofar as it can be used to collect elements aiming at improving the monitoring and surveillance system, reinforce the confidence in the safety scenarios or to clear up any doubts and uncertainties.

During the reversibility period, the inventory of waste packages and the monitoring of the integrity of the waste facility will be all the more important as the results of this surveillance might be an input to decide on the three mentioned options.

Organising regular meetings, within the framework of the decision-making process related to reversibility, to decide between the options for the future of the disposal facility also serves to highlight the question of how new developments in standards should be taken into account (safety standards, radiological protection standards, environmental protection, etc.) and how to ensure the capacity of technical and organisational systems to adapt to such developments. As long as checks are scheduled (and performed) it may also be possible to bring some equipment up to standard (and even replace obsolete equipment) and to some extent make adjustments in the facility in line with new safety requirements or new techniques.

Challenges for the radiation protection community

The radiation protection community has an important role to play to support local stakeholders' involvement in long-term surveillance systems, mainly in ways of training, skills and capacity building.

Thus, with a view to playing an active role when participating to the various aspects of the long-term surveillance system and fulfilling their task of ensuring vigilance, it is essential for local stakeholders to develop the necessary competences to be able to express their expectations and concerns not only about technical issues but also on governance ones. It is also essential to favour the development of governance processes allowing them to participate in the surveillance.

As far as radiation protection is concerned, the main challenge is then to foster the dissemination of a 'radiation protection culture' among the local stakeholders, and engage a cooperation with stakeholders when developing aspects related to long-term surveillance. For this purpose, various types of actions could be foreseen such as:

- Proposing dedicated education and training programmes to the stakeholders aiming at increasing their competences in the various aspects of the surveillance system (basic knowledge of radiation protection, explanation of the criteria used to characterize the waste inventory, what are the key aspects of environmental surveillance etc.);
- Developing, in co-operation with the stakeholders, criteria and indicators to be followed-up for assessing the health and environmental impacts of the waste facility;
- Favouring co-operation and partnership between local stakeholders and radiation protection experts from national institutions or other organisations;
- Setting up a system to ensure intergenerational transfer of knowledge within the local stakeholders (for example, by building specific radiation protection education programmes for local schools, …).

In order to facilitate the involvement of local stakeholders in the waste inventory follow-up, radiation protection professionals could provide them regular reports, invite them to attend periodic inspections of the inventory, propose specific expertises to answer their concerns or even propose to call pluralistic expertises if some specific agreements can be found with operator and authorities to access to the waste inventory files or to directly make complementary measurements.

To assist in the comprehension of long-term surveillance indicators, it could be relevant to propose to local stakeholders (members of the Local Liaison Committee or other concerned representatives of local stakeholders to discuss and analyse with radiation protection professionals the safety reports linked with the radioactive waste facility. The interest of such discussions is both to share the meaning of indicators and the way of assessing the level of protection.

With regards to health and environmental surveillance, radiation protection professionals could be involved in the discussions, with local stakeholders, on the reference levels set up for the impact assessment and take into account local concerns and habits in the elaboration of the exposure scenarios.

Another challenge for the radiation protection community will be to adapt if necessary the monitoring system to respond to the additional surveillance criteria/indicators identified by the local stakeholders.

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Individual response to communication about the August 2008 ^{131}I release in Fleurus: results from a large scale survey with the Belgian population

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Abstract
Nuclear and radiological events, such as the one referred to in this article, have taught us that communication is one of the most important challenges in emergency management. In the crisis reported here, communication to local authorities, local actors and population proved to be challenging during this crisis.

On Friday, August 22nd, 2008, a release of 131I started from the Institut des Radioelements (IRE) in Fleurus, Belgium. The release was not noticed until after the weekend, and people living in neighbouring areas were informed no sooner than 6 days after the onset of the incident. It was only then that the Belgian authorities activated a nuclear emergency plan, taking protective actions for the population.

In the summer of 2009, SCK•CEN initiated a large-scale survey on risk perception, using computer assisted personal interviews (CAPI). The survey focused on nuclear risk communication, with one chapter on reception and acceptance of messages during the Fleurus crisis. To the representative sample of the Belgian population, we added a sample of 100 persons living in the zone impacted by the Fleurus incident.

It discusses how people living in the vicinity of Fleurus perceived and evaluated communication the local crisis, and examines the degree to which these perceptions vary according to the information sources attended to, and the media used. Finally, we compare the Fleurus data emergency risk communication with the results for the general population.

Introduction: reconstruction of the event
On Friday, August 22nd, 2008, ^{131}I started to be released from IRE, in Fleurus, Belgium. IRE produces isotopes for the medical sector, by extraction of fission products from highly enriched uranium targets irradiated in reactors such as the BR2 in SCK•CEN in Mol, Belgium. The production is a batch process and, after processing, waste liquids are collected in waste tanks. On that Friday, contents of smaller waste tanks were transferred to a larger tank, starting a chemical reaction and leading to the release of ^{131}I to the stack, through filter batteries. The bulk of the release took place over the weekend, after which smaller quantities kept being discharged for several days,

amounting to a total of 50 GBq ^{131}I. The release was not noticed till after the weekend, when a safety engineer started his work on Monday morning. The problem was first examined in cooperation with the technical support organisation, BEL-V, and the Belgian agency for nuclear control (AFCN-FANC) was not notified about the release until 5:15 p.m. on Monday evening.

On Tuesday, August 26th, the agency made its first mention of the event on its website, announcing that inspectors were being sent to the installation to examine the situation. More details were given in a press release the same afternoon that classified the event as a “serious incident” (INES-3), and that announced that production in the IRE as well as the neighbouring MDS Nordion facility were temporarily stopped. The statement specified the occurrence of a "very small release" to the environment that neither called for activation of the Belgian nuclear emergency plan nor for measures to protect the environment. The press release announced follow-up measurements by the Belgian automatic radiological measurement network TELERAD.

It may be worth mentioning that, in fact, four additional mobile TELERAD stations were placed on and near the Fleurus site that, due to the characteristics of the release, were not able to register the release. The fact that only pure 131 Iodine was discharged at a low concentration during a relatively long period, made it impossible for the TELERAD network to capture dose rates increases.

On Wednesday August 27, AFCN-FANC (the Belgian agency) informed IAEA about the incident, stating *“The waste division of the IRE has performed a transfer of liquid radioactive waste from one tank to another one. Immediately afterwards – for reasons still unknown – radioactivity was released through the stack. The quantity of radioactivity released into the environment is estimated at 45 GBq I-131, which corresponds to a dose of 160 microsievert for a hypothetical person remaining permanently at the site’s enclosure. This incident did not cause a contamination of the personnel, and their dose limits were neither exceeded.”* [BVS 2008]

On Thursday August 28, the first results for three environmental samples became available. These samples, taken by the agency in the close vicinity of the IRE, and analyzed in the lab of the scientific institute of Public Health ISP/WIV, shed new light on the situation. Radioactivity was established at up to 5000 Bq/kg for one grass sample, suggesting that the intervention limits for the food production might have been exceeded and, therefore, activation of the Belgian emergency plan was needed. On Thursday evening governmental web sites issued the first recommendations to the population: People living in a zone of 5 km north east of the IRE were neither to eat fruits and vegetables from their own gardens, nor to use rain water. The press publicized these recommendations. Moreover, local and provincial emergency managers received a FAQ list specifically compiled for the occasion [Crisiscenter 2008]. The local police and the city of Fleurus communicated directly with the concerned population on Friday August 29th and an information phone line was opened. Further characterisation of potential water, air, grass, vegetables and milk contaminations was carried out.

On Saturday August 30th, new measurements somewhat alleviated the concerns. These allowed a reduction of the area for which protective measures were recommended to 3 km-area north east of the IRE. In addition, they confirmed that no significant contamination of milk had occurred. (Maximum measurement was 17

Bq/liter where the intervention level is 500 Bq/liter). Nevertheless, a large-scale campaign to do thyroid measurements was announced to start on Monday, September 1st. These measurements assess a potentially enhanced risk for thyroid cancers and are especially important for children [Public Health 2008].

In the next week, the crisis management team focused on complete certainty that the release had stopped. It requested that the IRE installed extra filters between the waste tank and the stack, and it further monitored the situation by environmental sampling, and active charcoal air sampling. By Saturday September 6th, there were no more uncertainties: No measurements of over 100 samples of vegetables and fruits had been near the intervention levels [FAVV2008], the maximum was 86 Bq/kg, while the intervention level is 2000 Bq/kg, however the WHO has a recommendation to take action when baby food exceeds 100 Bq/kg, and at no time there had been a need to trigger direct protective action for the population (sheltering, evacuation or intake of iodine tablets). Therefore, recommendations to avoid eating fruits and vegetables from the gardens were lifted, and the emergency plan scaled down to a U1 level. Nevertheless, thyroid testing for possible contamination continued to be offered to the local population, with the idea that this could be re-assuring. The medical examinations, performed on Monday September 1st and Tuesday September 2nd, prioritized the most sensitive persons, children and pregnant women. A total of 1320 persons were examined, and no signs of contamination found.

Media coverage of the incident

The event was covered by all mass media in Belgium, both in the French speaking southern part, and in the northern Dutch speaking part of the country. The event was no headline news, but it remained a daily news items for several weeks. The news items were mostly informative, based on the information provided by the crisis management or informed by interviews with the important actors: crisis managers, experts, managers from the installation and local and national politicians. The incident was not sensationalised in the mass media; there was neither amplification of negative messages nor stigmatisation of the technology. Discussions focused on the responsibility for the incident, with a focus on the unacceptable late response of the authorities and the lack of information to the population in the first days. Another issue of discussion was the failure of the automatic measuring network TELERAD to register the release; this failure was interpreted as another instance of poor performance of this network that had received some negative press before. Blame of the IRE management for their lack of communication was another theme, framed as an illustration of the existing problematic relation between IRE, the local authorities and the local population [Fleurus 2008]. Part of the news framing was also on the expected shortage of medical isotopes, a consequence of the incident.

Public opinion survey: material and methods

The Belgian Nuclear Research Centre SCK·CEN conducts periodically large-scale public opinion surveys among a representative sample of the Belgian population [Carlé and Hardeman, 2003; Van Aeken et al., 2007; Perko et al., 2010]. Alongside with recurrent issues such as risk perception, confidence in authorities and other actors, and the use of nuclear energy, the surveys include detailed research sections on topics such as expert functioning, emergency planning, food safety or risk communication.

The data for this study originate from the 2009 SCK•CEN survey.

1. Background Variables
2. Risk perception and confidence in authorities for 17 risk items
3. Attitude towards:
3.1. Science and technology
3.2. Stakeholders engagement in decision process related to industrial installations with risks
3.3 Nuclear energy
4. Acceptance of legal norms for food products
5. Use of different information media
6. Evaluation of different actors in the nuclear domain
7. Risk perception of an accident in a nuclear installation
8. Safety behaviour and anomy
9. Knowledge about the nuclear domain
Newsflash: TV-report about topics related to preparedness for nuclear emergencies
10. Reception & acceptance of messages communicated in the TV-report
11. Reception & acceptance of the iodine predistribution / information campaign
12. Reception & acceptance of messages communicated by authorities during the Fleurus event in 2008
13. Consumer's attitude towards food with radioactive contamination

Fig. 1. Main topics of the SCK•CEN risk barometer survey in 2009.

The content of the entire survey is summarised in Fig.1, more details can be found in Perko et al. (2010). In this paper we present results relevant for the respondents' perception of the communication during the IRE ^{131}I release incident in Fleurus.

Sampling procedure

Field work in the SCK•CEN 2009 survey was performed by a market research bureau using CAPI - computer assisted personal interviews. These interviews were carried out at the home of the respondent; the average duration was 43 minutes.

The bulk data of the 2009 survey consisted of 1031 interviews in French or Dutch. This sample is representative for the Belgian adult population with respect to province, region, level of urbanization, gender, age and professionally active status.

In the general Belgian population, 163 out of the 1031 persons remembered the IRE incident, the questions about what they remembered, to what degree they agreed with the communication during the crisis, and how what media and information sources they used were asked to this subset.

In addition, we asked the same set of questions to a supplementary set of 104 persons from the Fleurus area, the municipality in the French speaking part of Belgium, where the IRE incident happened.

In the next paragraphs, the results for the specific Fleurus group are compared with those from the 16% of the Belgian population who remembered the incident. We provide the results from these directly involved communities as a second line in the graphs, to allow comparison.

Results

Reception of the crisis management communication during the incident

The first set of five questions measures the degree to which people still remember what happened. During the incident, which took place about 11 months prior to this survey, all issues covered in the questionnaire, had received extensive media coverage.

Table 1. What do people remember about the incident?

Question	Sample	Correct answer	Incor. answer	Incor. answer	Incor. answer	No answer
What was the main pollutant?		Radioiodine	Tritium	Caesium	Uranium	
	BE 163	19%	4%	4%	15%	58%
	FL 104	63%	1%	1%	1%	35%
For what purpose was it produced?		Medical	Indus- trial	Energy		
	BE 163	58%	15%	7%		20%
	FL 104	33%	11%	21%		36%
What is the risk of a large intake of Radio-iodine?		Thyroid cancer	Lung cancer	Skin cancer		
	BE 163	68%	7%	2%		23%
	FL 104	77%	3%	0%		20%
What countermeasures were adviced ?		No fresh food from garden	No dairy products	To evacuate		
	BE 163	67%	9%	14%		10%
	FL 104	74%	1%	14%		11%
What was measured in the children?		Radioiodine in thyroid	Blood ana- lysed in lab	Children's urine test		
	BE 163	19%	21%	6%		53%
	FL 104	3%	11%	18%		68%

Less than 20% of the people in Belgium who remember the incident with a radio-active release know it was radio-iodine which was accidently released, but in Fleurus, over 60% remembers the radio-iodine. Remarkably, more people in the general population than in Fleurus connect the incident to the production of radio-iodine for medical purposes. This might be because the subject of a shortage of radio-iodine in the hospitals was in the press during a long time, and it was connected to the incident in Fleurus and the long shutdown of the facility. The majority of the respondents, both in the population in the Fleurus area and also in the part of the population that remembers the incident, are aware of the link between radio-iodine and thyroid cancer. About 80% of both groups remember correctly that people were advised no to consume fruits and vegetables from their gardens if they lived in an area situated up to 5 km north east of the facility. Neither the local population nor the part of the general population, who remembered the incident, appear to have been aware of the thyroid measurement campaign.

Acceptance of the crisis management communication

We asked respondents to indicate their agreement with six statements regarding the information and guidance given during the crisis Six statements were proposed, to which the respondents had to state their agreement degree on a 5-point Likert scale (from 1="strongly disagree" to 5="strongly agree"). Again, the results presented in graph. 1 come from the 16% of the Belgian population who remembered the incident and are compared with the Fleurus sample.

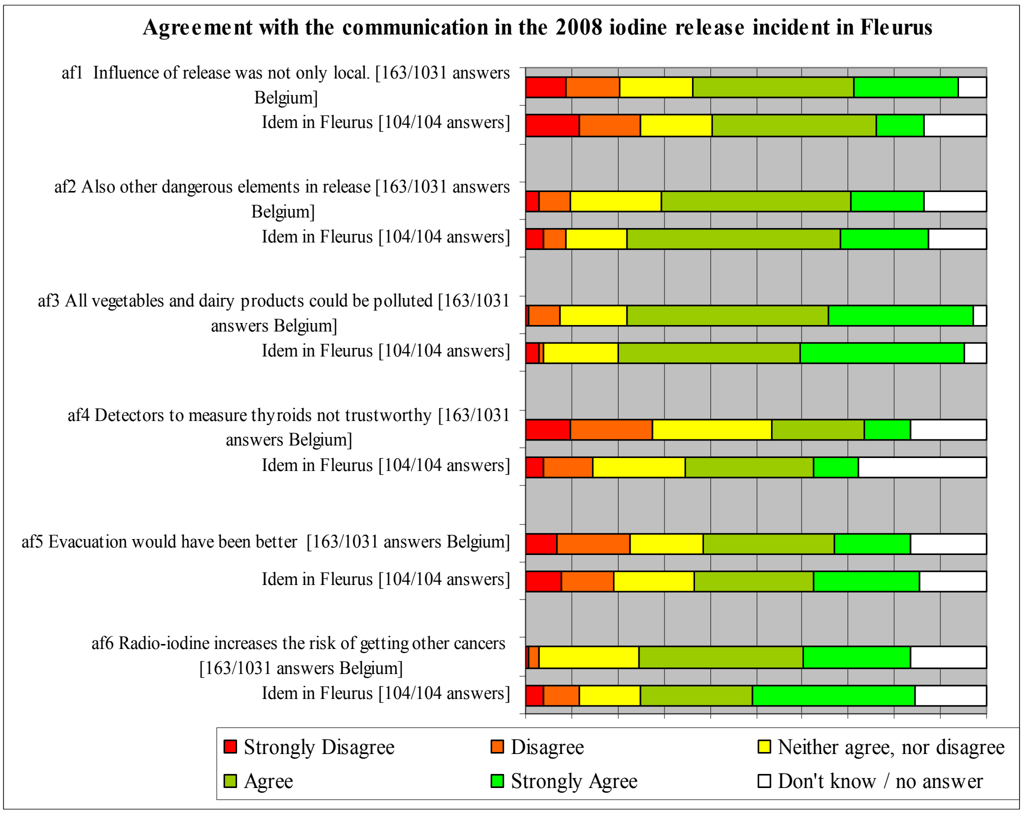

Fig. 1. Agreement with the crisis management communication in the 2008 iodine release incident in Fleurus, Belgium.

Most people think that the incident had consequences for a larger area than was communicated by the authorities (first statement). There is little difference between the people from Fleurus and the general population of Belgium. A large majority of people think that the situation was worse than communicated by the authorities (statements 2 and 3), agreeing with the view that other dangerous elements were released (60% to 80% of the samples), or that other products than garden fruit and vegetables from a larger area could have been affected (50% to 80% of the respondents). Agreement with the last statement also indicates that people think the situation is worse: they believe that other health effects will result from the release.

The population is more balanced on whether evacuation would have been a better countermeasure: about 50% agree, against 20 % disagree and 30 % people with no clear opinion. And finally, statement 4 shows that people don't really trust the physical measurements from the detectors, although the agreement with the statement ".results from detectors used for the measurements of presence of radio-iodine in the thyroid are not completely trustworthy" is lower than for the other statements, between 30 and 40% agree. Remarkably there are no big differences between the general population and the people in Fleurus who were in the affected zone.

Information channels used during the incident.

In this paragraph we look at the media use during the incident in Fleurus. First we ask the respondents what media and information sources they used, and then we ask their appreciation for each information source, specified by how credible, how timely and how extensively they evaluated each source. The first part lists the mass media, the second part personal contacts and other information sources. The results are presented separately for the two parts.

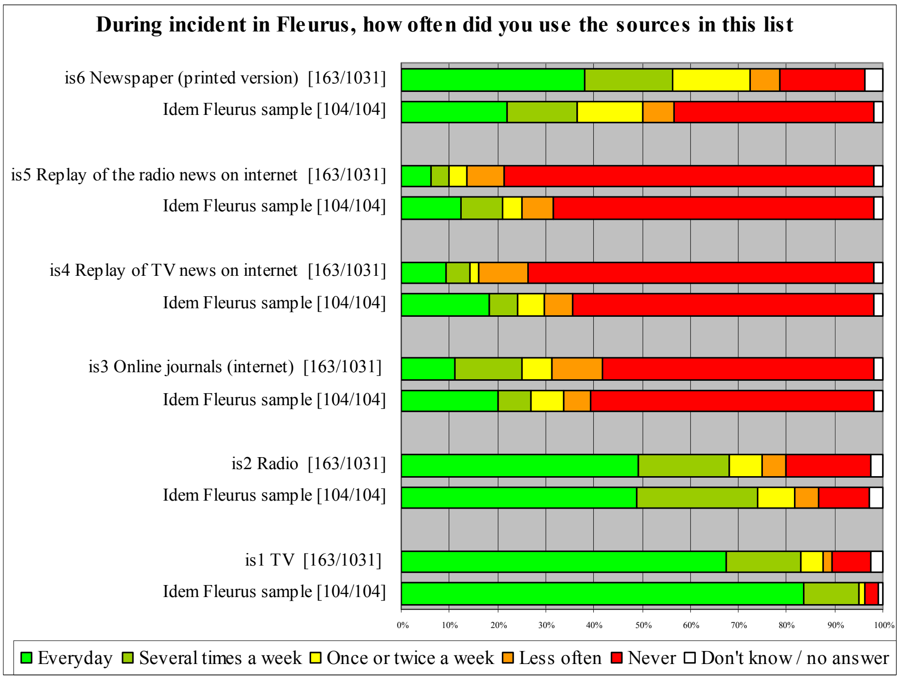

Fig. 2. Information channels used during the 2008 iodine release incident in Fleurus, part A.

In Figure 2 we see that television was the number one mass communication channel, over 80% of the respondents in Fleurus say the watched TV daily, while nearly 70% of the population in the rest of the country indicate they followed the Fleurus incident daily on TV. Radio is clearly the second most important information channel, around 50% both in the local population and in the rest of the country followed the news daily on the radio, and another 20 % did so several times a week.

Newspapers are the third mass communication channel, consulted daily or several times a week by 55% of the general population (or at least by the part of the general population who remembered the incident). For the local population, newspapers were less important, 50 % of the Fleurus population says they read the newspaper never or less than once a week during the incident. We differentiated between 3 types of internet use: internet online journals, replays of TV news on internet and replay of radio news on internet. In general internet is used less than the other media; over 60% of the respondents indicate they never used internet in any of the three ways specified. About 25% of the Fleurus respondents say they used internet journals daily or at least several times a week, and 22% used the replays of TV and radio news as often. The responses for the general population indicate a marginally lower use of the internet sources.

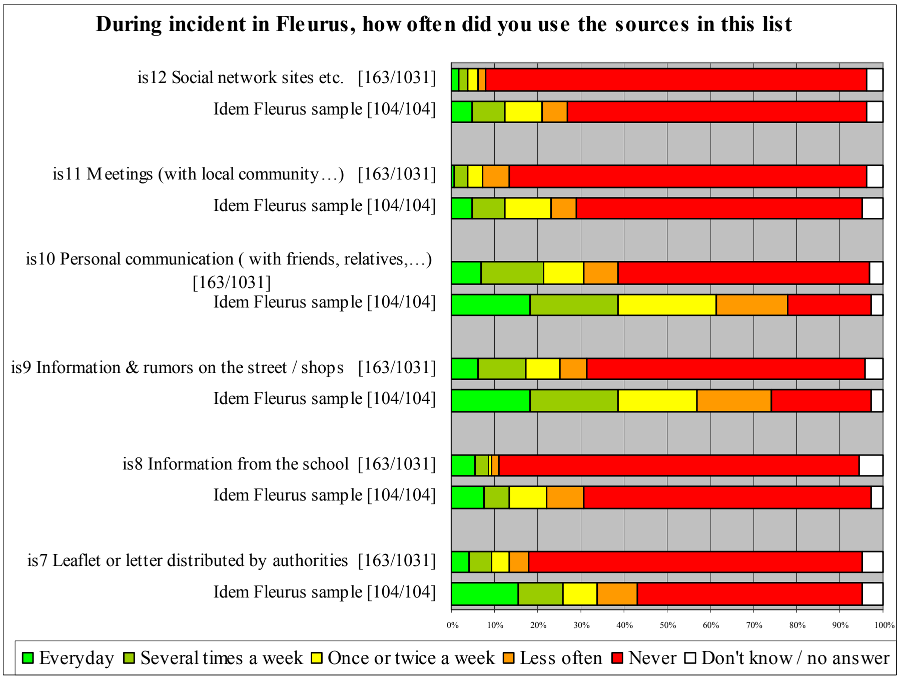

Fig. 3. Information channels used during the 2008 iodine release incident in Fleurus, part B.

While even in Fleurus, TV and radio news were important sources of information, here personal communications such as discussions with friends and relatives, and information and rumours in the street and in local shops etc. were relatively important also: about 40% of the Fleurus sample self-reported to have turned to others daily or several times a week. Other sources, such as letters or leaflets from authorities, local information communications by the schools were used as information by 30 to 40% of the local population at least once a week.

In the next series of questions, we asked the respondents to evaluate the information sources on credibility and being timely. The questions were only asked for the communication sources for which the respondents previously indicated some use, therefore, the total number of answers is different for each source. Results are presented in Tables 2 and 3.

Table 2. Evaluation of credibility and timeliness of media during 2008 iodine release incident in Fleurus Part A.

Sources	#	Credi bility Little	Credibility Average & No Answer	Credi bility Much	Timeli-ness Little	Timeliness Average & No Answer	Timeliness Much
TV (Belgium)	146	14%	39%	47%	13%	37%	50%
TV (Fleurus)	100	46%	30%	24%	49%	26%	25%
Radio (Belgium)	130	10%	42%	48%	9%	42%	48%
Radio (Fleurus)	90	42%	31%	27%	40%	36%	24%
On line Journals (internet) Belgium	68	13%	40%	47%	12%	37%	51%
On line Journals (internet) Fleurus	41	41%	46%	12%	41%	46%	12%
Replay of TV news on internet (Belgium)	43	16%	42%	42%	14%	37%	49%
Replay of TV news on internet (Fleurus)	37	43%	46%	11%	41%	43%	16%
Replay of Radio news on internet (Belgium)	35	14%	46%	40%	17%	43%	40%
Replay of Radio news on internet (Fleurus)	33	36%	48%	15%	30%	55%	15%
Newspaper (printed version) Belgium	128	8%	43%	49%	11%	40%	49%
Newspaper (printed version) Fleurus	59	27%	47%	25%	29%	44%	27%

A first observation is that credibility of the media is considered lower in the Fleurus sample than in the general population. Furthermore, the general population expresses lower confidence in TV and radio news when answering to questions about the Fleurus incident than to more general questions on nuclear accidents, as they occurred earlier in the questionnaire. People from Fleurus have a negative evaluation of the timeliness of news in all media, which may reflect the delay of communication by both authorities and media: It took 5 days after the start of the release before it came first in the news. The general population has a more positive view on how fast the media gave information about the crisis.

Table 3. Evaluation of credibility and timeliness of media during 2008 iodine release incident in Fleurus Part B.

Sources	#	Credi bility Little	Credibility Average & No Answer	Credi bility Much	Timeli-ness Little	Timeliness Average & No Answer	Timeli-ness Much
Leaflets & letters by authorities (Belgium)	29	14%	21%	34%	21%	41%	38%
Leaflets & letters by authorities (Fleurus)	45	46%	24%	58%	24%	58%	18%
Information from the school (Belgium)	18	17%	39%	44%	17%	28%	56%
Information from the school (Fleurus)	32	19%	28%	53%	19%	31%	50%
Information and rumors on the street etc (Belgium)	51	18%	31%	51%	18%	31%	51%
Information and rumors on the street etc (Fleurus)	77	23%	23%	53%	21%	26%	53%
Personal communication (discussion with friends, relatives...) Belgium	63	29%	27%	44%	30%	29%	41%
Personal communication (discussion with friends, relatives...) Fleurus	81	22%	30%	48%	23%	26%	51%
Meetings (with local community…) Belgium	22	18%	36%	45%	18%	59%	23%
Meetings (with local community…) Fleurus	30	37%	50%	13%	40%	47%	13%
Social network sites (facebook,…) Belgium	13	23%	62%	15%	15%	38%	46%
Social network sites (facebook,…) Fleurus	28	21%	50%	29%	21%	57%	21%

Information given by the school is evaluated as the most credible information (over 50% high or very high credibility) by the small subset of the Fleurus sample that indicated schools as a source of information (one third). Informal discussions with friends and rumours on the streets or local shops etc. are considered more credible than the information gained from information spread by the authorities (written information, public meetings). Compared to the general population, those from Fleurus whom the crisis concerned most, evaluate communications by both governmental officials and the press as less credible; they put more trust in information given by the school, and gained from personal conversations.

Direct communication and certainly personal communication is evaluated more as timely than media coverage by radio and TV; an exception forms the “town hall” meeting that was *not* perceived to be timely. The results for the Fleurus sample are for these communications also more consistent with the responses from the people from the general population.

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Science and values in radiation protection: Summary of two NEA Workshops

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Abstract

For some time the Nuclear Energy Agency (NEA) has been examining the evolution of decision making in radiological protection, studying both the stakeholder involvement aspects, and the judgemental and scientific uncertainty aspects. As part of this work the NEA held a workshop in January 2008 to examine how science, social values and judgement are combined to arrive at acceptable and sustainable radiological protection decisions in a transparent fashion. The second workshop in this series was held in November 2009.

The first workshop addressed three topics of emerging science: individual sensitivity; non-targeted effects; and circulatory diseases. These were discussed in a "*What if?*" format, that is, if research in these areas were to result in significant changes in our understanding of risk, what would we do and how would uncertainty and judgement be taken into account. The results of the dialogue among the attending regulators, scientists and NGOs were seen as having very positively improved mutual understanding of issues, viewpoints and possible implications, and as beginning to identify elements of a process and framework for the better integration of the social and scientific dimensions of radiation protection.

The second science and values workshop will take a similar approach to the first, but addressed three topics that are currently presenting challenges to radiological protection policy, regulation and application in a "*What now?*" format: management of medical exposures; management of radon exposures; and implications of new scientific evidence on cardiovascular disease risks from radiation exposure. These topics represent areas where current approaches to radiation protection have either not fully yielded the desired results (i.e. radon and medical exposures), and/or have not yet been perceived as being sufficiently scientifically supported to warrant change in the current approach (i.e. cardiovascular diseases). Thus, while the objective of this workshop was not to develop detailed recommendations as to new approaches, it did develop significant results:

- Relevant stakeholders in each area presented and exchanged experience related to their viewpoints and relevant values, increasing their levels of mutual understanding to facilitate development of common approaches forward;
- Social and scientific rationale and justification for adopting new approaches to radiation protection in each of these areas was discussed (tipping point);

- Practical approaches to improving radiation protection in each area was discussed based on national experience, and in relation with public health issues; and
- Process and framework elements that could enhance radiation protection in these three areas by better integration of social and scientific aspects were identified.

This paper will summarise the results of this workshop, and indicate what further work could be pursued in this area.

Introduction

Radiological protection was founded on and continues to be a combination of science and value judgments. As described by Lauriston S. Taylor, radiation physicist and pioneer in the field of radiation safety, and founding member of both the International Commission on Radiation Units and Measurements (ICRO) and the International Commission on Radiological Protection (ICRP), radiological protection is *"... not only a matter for science. It is a problem of philosophy, and morality, and the utmost wisdom."* (Taylor 1957).

In the development of radiological protection policy and its practical application there is always a need for radiological protection policy makers, practitioners and other stakeholders to better understand the evolving interactions between science and values. At the same time, there is also a need for radiological protection scientists to better understand the broad processes of radiological protection decision making and to better interact with these processes particularly by furnishing input coming from their research. Rolf M. Sievert, the Swedish physicist who laid the foundations of modern radiation physics, expressed clearly the dynamic link between policy, research and application: *"The establishment of maximum permissible radiation levels is a non scientific task, which must be based primarily on scientific knowledge and judgment."* (United Nations 1958).

Achieving such a mutual understanding among radiological protection policy makers, practitioners, researchers, industry, non-governmental organisations, etc. is expected to facilitate prioritisation of research and framing of decision making in the future and to improve the quality and robustness of radiological protection.

The Committee on Radiation Protection and Public Health (CRPPH), a standing technical committee of the OECD Nuclear Energy Agency, has long been aware of a need to develop a shared understanding of emerging challenges for radiological protection among scientific and regulatory communities and other concerned stakeholders. In 2007 the CRPPH Expert Group on the Implications of Radiological Protection Science (EGIS) published a report on "Scientific Issues and Emerging Challenges for Radiation Protection" (OECD-NEA 2007). This report represents a broad summary of key scientific challenges that could arise from ongoing research on radiological protection. Additionally, the CRPPH Expert Group on the Collective Opinion (EGCO) published its report on "Radiation Protection in Today's World - Towards Sustainability" (OECD-NEA 2007). That report identifies key emerging challenges to the radiological protection system in order to assist decision makers at all levels to better address these within their relevant contexts.

In follow-up to these activities and in response to the existing need to foster mutual understanding among all concerned stakeholders, the CRPPH initiated a longer-term process of reflection on scientific and societal issues that might challenge radiological protection in the coming decade. The first step in this process was a workshop organised in collaboration with the Radiation and Nuclear Safety Authority of Finland (STUK) to address some of these issues. This first workshop entitled "Science and Values in Radiological Protection" took place on January 15-17, 2008 in Helsinki, Finland1. In this workshop more than 60 scientists, researchers, representatives of regulatory authorities, political decision-makers and other experts from 22 countries (including countries not members of the OECD) gathered together to discuss new trends in radiological protection.

A second workshop in the series was then organised in collaboration with France's IRSN (Institut de Radioprotection et de Sûreté Nucléaire) and sponsored by the French Ministry of Ecology, Energy, Sustainable Development and Land Use Planning (MEEDDAT). This second "Science and Values" workshop took place on 30 November - 2 December 2009 in Vaux-de-Cernay, France2. Attending were more than 70 participants from 19 countries. The following types of institution were represented: public health authorities; regulators; international bodies; industry; hospitals; scientific expert organisations; universities.

The objective of these meetings was to develop a more shared understanding between the various stakeholders with regard to the mixing of science and its uncertainty with social values and their diversity in achieving decisions that are sustainable and acceptable when radiological risks are involved. At both Helsinki and Vaux-de-Cernay, selected key challenges to radiation protection were explored by plenary presentations3. In-depth discussion then took place among participants in topical break out sessions. The present report introduces the objectives, aims and topics of each workshop, and then summarises findings and outcomes of the respective break out sessions.

Material and methods

Each workshop was structured to foster participants having a complete overview of the "state-of-the-art" with respect to the subjects addressed, and to allow a significant amount of time dedicated to discussion of the value issues involved in decisions regarding these topics. To achieve this, each workshop was structured around a number of invited plenary presentations addressing selected emerging challenges to radiological protection. These presentations were followed by facilitated break out sessions that examined those challenges in greater detail, in particular their potential implications from the perspectives of science, values and interpretation. Participation in each break out session was predetermined so as to achieve balance in terms of the spectrum of stakeholders participating in the workshop.

Each workshop was structured around a few scientific and regulatory issues. The 1st Science and Values Workshop addressed the following three topics, which had been identified in the EGIS report (OECD-NEA 2007):

- Non-targeted effects.
- Individual sensitivity.
- Circulatory diseases.

[1] http://www.nea.fr/html/rp/helsinki08/welcome.html
[2] http://www.nea.fr/html/rp/vaulx_de_cernay09/welcome.html
[3] A subset of the plenary presentations from each workshop can be viewed online at the hyperlinks provided in footnotes 1 and 2 of this document. The remaining presentations were based on preliminary data, or on data awaiting journal publication elsewhere, and were thus unsuited for online publication.

The moderated breakout session discussions followed a "*what if*" approach to determine the nature and significance of the potential implications of the various emerging issues or challenges. Where appropriate, participants identified the need for further research and/or analysis to aid in better understanding the challenge and how/if it may be accommodated. Discussion was facilitated by a set of proposed questions adapted to each "challenge" topic, reproduced below for the topic of "non-targeted effects":

- Why are non-targeted effects a relevant topic?
- What do we know now about non-targeted effects?
- What do we not know now about non-targeted effects that we would like to know?
- What are the scientific issues?
- What are the regulatory issues?
- What approach(es) should be followed to address scientific issues raised above?
- What would we do differently if we knew now what we would like to know?
- What could or should we do now while we wait for the answers to these questions?

During the 1st workshop a "*What if?*" approach was used to discuss the scientific aspects in some key areas of emerging radiological protection science. The results of the dialogue among the attending regulators, scientists and NGOs were seen as having very positively improved mutual understanding of issues, viewpoints and possible implications. The discussion was also seen as having begun to identify elements of a process and framework for the better integration of the social and scientific dimensions of radiological protection.

Re-emphasising that radiological protection is a combination of science and value judgments, the CRPPH agreed at its sixty-sixth annual meeting to continue the useful discussions begun in Helsinki by organising the second workshop. Here, however, the committee agreed to focus on radiological protection issues that are currently facing us, and that continue to pose challenges to our world today. As such, the second workshop in this series was designed to address a series of current radiological protection issues from the standpoint not of "*What if?*", but rather, "*What now?*". With this, the 2nd Science and Values workshop continued along these lines, and addressed the following three topics:

- Radon as a public health issue.
- Medical exposures in diagnostic and screening procedures.
- Radiation-induced vascular effects.

During the discussion of these issues participants presented and discussed their relevant national objectives and experience in the designated area, and brainstormed on approaches to better achieve their objectives when faced with emerging data. The break out sessions engaged participants in addressing the following central questions:

1. Which issues need further elaboration before deciding whether it is necessary or appropriate to change the current approach?
    - Identification and discussion of science issues
        - What level of effect is being discussed?
        - What are the uncertainties involved and how well characterised are they?
    - Identification and discussion of practical issues
        - What would a change in regulation impact?
        - What would be the magnitude of such changes?
    - Identification and discussion of value issues
        - Balance of risks and benefits?
        - Precautionary judgement?
2. What aspects weigh on decisions regarding possible change?
    - Implications for regulation, industry and health care sectors
    - Practical implications for application
        - Resources needed.
        - Adapting to a significant change in approach.
        - Education and training needs.

Results

The results of these two workshops are being summarised in an NEA report that will be published in the fall of 2010, however the most significant conclusions are presented briefly here.

Non-targeted Effects (1st Science and Values Workshop)

Non-targeted effects referred to those effects seen in cells not directly hit by radiation, such as the so-called bystander effects and genetic instability. These effects were characterised as being seen mostly at a cellular level, and as such having a somewhat uncertain effect on the overall detriment to the organism. It was felt that better understanding of non-targeted effects would very likely not affect the overall level of risk, but rather would better explain the point of origin of the risk.

Individual Sensitivity (1st Science and Values Workshop)

At this workshop the individual sensitivity that was discussed referred broadly to the sensitivity that may be caused by genetic imperfections, rather than the sensitivity differences seen between men and women, or between children and adults. As a result of discussions, there seemed to be no need to radically modify the current approach RP. However, in emergency situations and in medical diagnostic and therapy situations, it was suggested that some consideration be given to refocusing protective actions taking individual sensitivity into account.

Cardiovascular Disease
(1st Science and Values Workshop)

The discussion of cardiovascular disease at the 1st Science and Values workshop suggested that new epidemiological data, in particular from the Mayak workers, would be published not long after the workshop, and would broadly show agreement with data from the Hiroshima / Nagasaki Lifespan study, suggesting that cardiovascular diseases could be provoked by both acute and chronic exposures. Taking this detriment into account, at that point simply based on LSS data, there could be a need to lower current dose limits by 30-50%, with strong emphasis on optimization. However, the workshop concluded that the science is still evolving and there was no need at that point to recommend change to the system of radiological protection.

Management of Domestic Radon Exposure
(2nd Science and Values Workshop)

In discussing this topic and exchanging experience regarding national approaches, it was noted that most countries focus actions on high-concentration homes. This suggests that the "values" aspect of the management decision has focused on protecting those most exposed. Epidemiology suggests, however, that most lung cancers occur in homes with lower concentrations, such that revisiting national protection strategies could be of value. Optimisation actions could include the promotion of protective actions for new buildings (e.g. building codes, pre-sale criteria), tying radon strategy to other programmes (e.g. smoking, energy efficiency, indoor-air quality), integrating radon exposure management into the public health system, rather than the other way around. It was noted that remediation of radon in dwellings is a long-term process

Management of Diagnostic Medical Exposures
(2nd Science and Values Workshop)

With the significant increase of patient exposures in many countries, the justification of diagnostic procedures, and the optimisation of protection were seen as key issues to discuss and improve. It was suggested that physicians tend to focus very strongly on the diagnosis and treatment of an individual's health issues, suggesting that the driving value is to give priority to health risks today. There is evidence, however, that a significant fraction of diagnostic procedures are not justified at the individual level, and that protection is not optimised. Justification seems to be the critical step. Greater awareness and use of exiting tools (e.g. diagnostic guidelines), and focus on sensitive populations (e.g. children) seem to be the areas of focus

Cardiovascular Disease
(2nd Science and Values Workshop)

Because the scientific study of cardiovascular diseases seemed to be evolving rapidly, this topic was chosen to be revisited at the 2nd Science and Values workshop. Radiation-induced cardiovascular disease is well known in high-dose radiation-therapy patients, however risks are broadly viewed as deterministic, without sufficient scientific knowledge to categorise otherwise. As such, the value driving decisions seems again to be to give priority to health risks today, in particular since these effects seem to be deterministic rather than stochastic. However, new epidemiological studies suggest that chronic, lifetime exposures may, at somewhat lower doses, cause cardiovascular diseases. Mechanisms of such damage, and identification of “target organ” are still topics of research. Again, it was judged that the science is still evolving and there was no need at that point to recommend change to the system of radiological protection. It was, however noted, that with the same level of knowledge regarding stochastic effects of radiation in the 1950s, a precautionary approach was none-the-less adopted at that point, without the need for science to “prove” that a risk existed. As such, the workshop suggested that there is a need to increase awareness of this issue, and to reinforce scientific studies.

Conclusions

The CRPPH has recognised that mutual understanding on the scientific evidence and on the radiological protection values and practice is important both for obtaining optimal protection, and for identifying the gaps in knowledge that are most relevant for radiological protection. Most of the participants at both Science and Values workshops agreed that while there is no immediate need to change the current principles, extended dialogue among all concerned stakeholders is necessary in order to facilitate integration of challenging scientific phenomena into existing regulatory frameworks. This type of exchange forum between regulators and scientists was welcomed and could serve as a model way of moving forward. As such, the CRPPH is preparing the 3rd Science and Values workshop, to be held in the second half of 2011 in Japan, hosted by the Japanese government.

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Lessons learned in radiological protection during the dismantling of nuclear facilities

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Abstract
This paper presents the lessons learned and recommendations deriving therefrom in relation to radiological protection and the application of ALARA programmes during the performance of dismantling projects. It also identifies specific problems arising and the result of the solutions designed.

Introduction
The Spanish national radioactive waste management agency Empresa Nacional de Residuos Radiactivos (ENRESA) has carried out the dismantling of the Vandellós 1 Nuclear Power Plant and the experimental reactor and other research facilities at the Centre for Energy-Related, Environmental and Technological Research (CIEMAT).

During dismantling work various radiological protection challenges arise; application of the ALARA principle is fundamental.

The principles governing the application of the ENRESA ALARA policy in the projects and activities under its responsibility are established in the AL (G) document, "Manual for application of the ALARA principle in ENRESA projects and activities", and in the ALARA programmes specific to the different projects. These documents establish the methods for the development, implementation and evaluation of the ALARA programmes applicable to the engineering and work performance phases of the facility Dismantling and Decommissioning Plans.

The application of this programme is carried out by introducing recommendations during the phase of drawing up of the documentation of interest, as well as through the performance of ALARA audits during the different activities and phases of the project.

The radiological criteria (collective dose, individual dose, dose rates and levels of surface and airborne contamination) for the detailed application of the ALARA methodology to a specific task are specified in the ALARA work control and management and dose reduction and minimisation procedures.

Specific aspects of dismantling
The objectives of Radiological Protection and ALARA, the basic criteria for application and the regulatory responsibilities are the same in normal operation of the facilities as in dismantling; however, the aspects relating to the licensing documentation, planning, adaptation of the organisation, production objectives, regulatory requirements, the

necessary human and material resources and the mechanisms for coordination with the rest of the organisation have clearly differentiated characteristics that must be taken into account.

Furthermore, there are initial conditioning factors that fundamentally shape the approach and methodology for the application of Radiological Protection, these being the type of facility, the time elapsing since shutdown, the final level of dismantling foreseen, the remaining surveillance systems and the possibilities of their being reused during the project.

All this requires adaptation of the organisation to the project objectives, this implying the organisation operating in a matrix configuration, an increase in the number and diversity of the participating organisations and the involvement of agents and workers not used to working in radiological scenarios.

As regards the Radiological Protection (RP) personnel, these conditioning factors materialise through a multiplication of interfaces and dependences, this including involvement in the "production process", which is not habitual at operating facilities.

For RP these initial conditions imply assessment of the usefulness of the available equipment and of the possible need for them to be adapted or complemented, specifically as regards documentation, facilities and instrumentation.

These conditioning factors pose various specific challenges, such as the following:

- Dismantling implies modifying the physical configuration of the systems existing at the installation (ventilation, effluents, etc.); variation of the routes for equipment, wastes and materials; the loss of auxiliary media (cranes, hoisting devices, etc.); the appearance of new radiological zones (stores, decontamination workshops, radioactive waste collection areas, etc.).
- Dismantling implies continuous variation of the radiological status of the installation due to the rupturing of system confinements, continuous work in areas of radiological risk and the appearance of new radiological zones (due to the movement and storage of the radioactive material generated).
- Dismantling implies modifying the working techniques due to the use of industrial cutting and demolition equipment, equipment for the moving of large loads and large-scale decontamination techniques not used during normal operation.
- Dismantling implies a variation of the radiological risks: the risk of exposure to transuranic elements increases, as does the variety of the chemical forms and sizes of contaminant particles, although the risk of exposure to other radionuclides, such as radioiodines, noble gases and, in general, all short-lived radionuclides, is reduced.
- Dismantling implies adapting the application criteria to the ALARA criterion. Converting classical criteria into a tool for efficiency is an essential objective, and for this the most practical approach is to require that contractors and the organisation overall fulfil their part of overall responsibility, offering the help required to simplify the methods for the analysis and assessment of alternatives, prioritising the optimisation criteria regarding strict compliance with the initial dose objectives.

- Dismantling implies contemplating the risk of internal contamination by alpha emitters, for which reason this should be taken into account: special individual perception of this risk; the existence of strict regulatory requirements; the significant dosimetry repercussion in the event of incorporation; the repercussion on external doses of using protective equipment slowing down job performance; the need for specific protective and measuring equipment; the need to implement complex internal contamination controls through the bioanalysis of urine and excreta.
- Dismantling implies adapting the radiological protection training of the workers, taking into account that many of them will never have worked before with ionising radiations and that disassembly, cutting, demolition and decontamination tasks require non-habitual behaviour patterns, precautions and protections. Likewise, the workers should understand the risk of internal contamination and not underestimate its importance.
- Dismantling implies a combination of radiological and conventional risks due to the existence of radiological environmental contamination and the presence of toxic materials and fumes. Likewise, the use of radiological protection equipment may increase the risk of accidents due to diminished visual or hearing capacity.
- Dismantling implies the extended use of radiological protection equipment and resources, both human and instrumental. The equipment is required to be robust and transportable and to provide extended detection capabilities allowing for the real-time measurement of airborne contamination and for remote control and data acquisition.

Lessons learned and recommendations

Indicated below are certain of the problems specific to dismantling detected and the solutions successfully developed and applied. The experience acquired during the dismantling of Vandellós I and of the facilities included in the PIMIC dismantling project constitutes the main source of information.

Administrative and organisational aspects

The main problem detected is that the contracting specifications do not establish sufficiently the ALARA responsibilities of the contractor and that ALARA aspects are not adequately considered in assessment of the bids. These shortcomings must be overcome when performing the work.

Furthermore, as ALARA solutions are not taken into account in the specifications, the collective doses assessed are overestimated and at times it is difficult to justify the variations between the estimated and actual doses before the Regulatory Authority.

The solution consists of establishing the ALARA commitments and requirements in the specifications and bids and of taking into account ALARA proposals in the licensing documentation dose estimates.

Other organisational aspects have also been observed relating to the documentation required of the exposed workers. These lead to delays in the process of registering the worker at the facility for him to perform his work.

This might be solved by including the requirements and the obligation of the contractor to meet them in the specifications and contracts, including penalties for cases of non-compliance.

Another aspect that needs to be improved is training; on the one hand to provide better understanding of the project documentation by the works organisation and, on the other, to provide better basic training on radiological protection for the workers performing the work.

In the dismantling operations managed by ENRESA, enormous efforts have been made regarding overall accountability, analysis, multidisciplinary planning, the involvement of the contractors and specific initial and on-going training, with highly positive results. Likewise, the contractors have been required to make available technical personnel with ALARA training and experience, this having facilitated coordination in this respect with the organisation.

Technical aspects

The following are particularly significant as regards the technical aspects to be taken into account:

- Initial radiological characterisation of the site: this is not normally oriented towards radiological protection needs for performance of the work, but towards estimation of residual activity, isotopic streams and scaling factors. Although an initial radiological classification is established, this is not valid since it corresponds to a static situation that might vary a great deal when disassembly work begins, with unforeseen areas of radiological risk emerging. It is necessary to extend and intensify the initial campaigns when the work begins, especially in those areas in which there is little historic know-how.
- Control of airborne contamination. The special characteristics of dismantling activities (cutting, breakdown, scarification, demolition, opening of active systems, transfer, transit storage, etc.) very often imply the release and dispersal of contaminants in the form of aerosols in the working environment. Occasionally this contamination poses non-habitual problems such as the presence of alpha particles, this requiring special controls implying sampling, detection and the implementation of exhaustive precautions, confinement and the use of special individual protective equipment.
- Confinement and ventilation: The technique of constructing and deploying rigid confined enclosures (SAS) (figure 1.) is efficient for the confinement of contamination when work is performed with important levels of negative pressure and air renewal. The elimination of releases of airborne contamination outside these SAS during work underlines this. These enclosures must guarantee an important level of negative pressure with respect to the adjacent areas, also ensuring a flow of air from clean compartments to more contaminated areas. There must also be conditioned areas for the loading, closure and removal of the containers and drums used for dismantling and secondary wastes. The use of transparent materials for their construction facilitates supervision, control and communications between the interior and the exterior of the enclosure. Personnel accesses and exits should be provided, with sufficient space for the implementation of sequential dressing and undressing stages, with the

corresponding stepwise arrangement, allowing for action with sufficient convenience and safety (these are the phases implying the highest radiological risk). In this same respect it is essential that there be a rigorous individual discipline as regards the sequence of undressing, for which reason increasing the awareness of all the participating workers becomes especially relevant.

Fig. 1. Rigid confined enclosure.

- Working techniques: The introduction of improvements in working techniques in the performance of tasks potentially generating or increasing the generation of surface and airborne contamination is another aspect for which important efforts have been made.
  - The use of special equipment for the removal and pre-conditioning of active liquids is to be recommended, as is the use of breathing equipment and systems for the filtering of contaminated dust, guaranteeing confinement of the particles and preventing subsequent risks of exposure to internal contamination.
  - The removal of liquids remaining in the bottom of tanks using venturi suction systems transferring the liquid and sludges directly to a confined drum, in which the air extracted passes through a HEPA filter before being vented, and the use of a similar system for the removal of highly contaminated dust, replacing the drum with a system equipped either with a sleeve filter or a cartridge filter in a plastic case (disposable without the need for handling) and a final HEPA stage, both easily disassembled and replaced, has proven to be much safer than the normal practice of using vacuum cleaners, the operation of which may pose considerable risks.
  - Also identified as a moment for special attention is the replacement of spent filters from portable ventilation units, and ventilation systems in general, especially when they have been used in areas with alpha contamination. Certain cutting techniques generate large quantities of smoke and aerosols that saturate the filters in a short time, making it necessary to replace them very frequently, a process that if carried out without protection or with inadequate handling may lead to the external or internal contamination of the workers.

- It would be interesting to study the possibility of improving the service lifetime of PVU filters by incorporating a coarse pre-filtering system of some kind, capable of retaining coarse particles and part of the contamination. This would the need for filter replacement operations and the associated risks. Reducing the number of spent filters with high levels of contamination (specifically alpha) would also reduce the risks associated with the necessary subsequent processes of disassembly and conditioning as waste.
- The objective of necessarily limiting the use of rotary or thermal cutting equipment or methods is to reduce the risk of dispersal and resuspension of contaminated aerosols. However, following assessment of the operational profitability of applying thermal cutting and its authorisation on the basis of tests performed for the purpose, it has been demonstrated that this may be used if adequate protection and efficient ventilation systems are available. The effectiveness of these aggressive but extraordinarily fast techniques makes them an interesting and controllable option from the radiological point of view, as long as the personnel are duly capacitated in their use. In the specific case of breaking down tanks, the solution to make thermal cutting compatible with the confinement of the contaminated fumes generated consisted of incorporating suction systems inside the tanks themselves, such that practically all the fumes and aerosols generated are immediately absorbed and filtered. To optimise the process, all the incomplete cuts were initially addressed, maintaining the geometric integrity of the tanks as long as possible in order to make the vacuuming more effective. This system kept the areas outside the SAS free from airborne contamination throughout all the work.
- The use of peelable paints (figure 2.) has been effective to fix the existing surface contamination and to protect already decontaminated areas. Their application and handling has not posed any problems.

Fig. 2. Peelable paint applicated.

  - As an alternative to thermal or rotary methods (plasma, blowtorch, cutting disk) for highly contaminated items, hydraulically operated manual shearing devices of different capacity have been used successfully for the cutting of pipes and profiles of small and intermediate dimensions. This system reduces the risk of resuspension , provides an initial sealing of the cut edges of the pipes and allows for a far better rate of work than manual disassembly and de-flanging and cutting with saws. It also reduces the risk of fire and does not produce fumes.

- Special protective equipment (figure 3): on the basis of analysis of the radiological characteristics and risks associated with work with high levels of surface and/or airborne contamination (especially if artificial alpha contaminants are included), the following is deemed to be necessary:
  - Availability and use of personal means to protect against contamination (surface and airborne), guaranteeing a sufficient degree of isolation under any conditions. For this reason, the use of special protective equipment of known characteristics and methods of use has been widespread for certain activities, this having proven to be useful in installations and tasks of comparable characteristics. In our case, this has consisted of the use of integral semi-autonomous systems with a supply of exterior breathable air and incorporating a mask and P3 filter connected to the supply line. This equipment allows a continuous level of protection to be afforded during the process of removing protections, guaranteeing that at no time during the undressing phase is the worker exposed to contamination by inhaling.

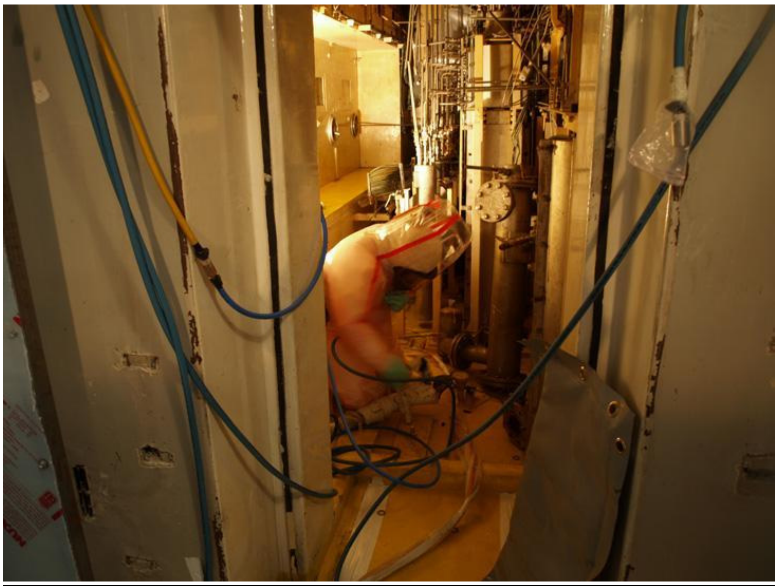

Fig. 3. Special protective equipment.

  - The specific conditions of use of this equipment, and the use of a breathable air system, make it advisable to deliver specific training to the personnel intervening in the different phases of the work.

- Special surveillance techniques: continuous airborne alpha and beta contamination measuring equipment has been used. This equipment was used for the dismantling of equipment that had been directly affected by operational activities implying the handling, reprocessing or storage of irradiated nuclear fuel. The advantage of using equipment of this type has been the possibility of remotely and centrally controlling the environmental conditions of the work places, with real-time display. The equipment has been of great help in tasks involving a specific risk of alpha contamination, since it immediately warns of any unexpected increase in the levels of airborne contamination, facilitating interruption of the work and the evacuation of the affected personnel. As a complement to the use of this equipment, environmental volumetric sampling has been performed, analysing the filters in the laboratory to confirm the alarms. It is recommended that nasal swabs be performed periodically and whenever there are incidents involving possible personal contamination affecting the face.
- Precautions in the handling of radioactive waste and materials: Management, including the handling and conditioning of radioactive materials and waste, constitutes one of the most important scopes of work during dismantling. It is essential to take into account that many of the radiological risks identified in the performance of tasks with materials are reproduced, with specific characteristics, in many of the activities relating to their management once they have become wastes. The experience acquired makes it possible to identify certain processes in which these risks are clearly manifested, and for which preventive and control measures should be established. Among the most significant are the inspection of containers, the immobilisation of non-compactable wastes and the compacting of wastes.
- Hot particles (discrete radioactive particles): The presence of discrete high specific activity radioactive particles has not been a risk detected in the dismantling of Vandellós and PIMIC, due to the type of facility and the time elapsing between the shutdown of the installations and the initiation of dismantling activities. However, in future light water NPP dismantling projects, the appearance of this radiological problem is highly likely due to the essential task of cutting highly activated metallic materials (reactor internals). This should be taken into account from the very design phase and consideration should be given to the need to develop programmes and procedures for the prevention, location, characterisation, control and elimination of the problem (specific portable detectors, integral checking gate monitors on two levels), as well as to dose calculation tools. In accordance with the international recommendations, the control programmes set up should be aimed at preventing equivalent organ dose limits from being exceeded (fundamentally the skin) and at preventing the appearance of deterministic effects such as inadvertent exposure to hot particles.

Conclusions

– The dismantling of a nuclear power plant requires the RP programme set up during normal operation to be entirely adapted. This adaptation covers from documentary aspects to increased human and technical resources and the reorientation of operating practices.

- The changing situation during work performance means that RP control must be continuous, without forgetting the importance of ensuring the commitment of the entire organisation to minimising doses.
- The ALARA principle should be taken into account from the moment of planning the work, and all the players involved should participate in its application.

- Regardless of the level of risk of external radiation that might exist at the facility, maximum attention should be paid to the risk of internal contamination, which increases greatly with respect to normal operation because of the disassembly tasks themselves, the removal from service of the ventilation and confinement systems and interventions on equipment and components never before touched.
- Adequate planning of the tasks requires an in-depth understanding of the operating and radiological history of the facility, although in all cases the radiological status of the installation needs to be updated when addressing its dismantling.

With due consideration given to the above, it may be stated that the RP applied will be effective and that doses will be as low as reasonably achievable.

Remediation of TENORM residues: Professional risk assessment and public risk perception

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Abstract

In the last three decades, members of different disciplines (such as sociology, radiation protection) have published a vast number of studies on risk perception and acceptance with respect to nuclear power plants and nuclear waste repositories.

However, bridging the gap between what radiation protection experts consider to be an objective risk and the subjective risk perception by non-scientists is a challenging task. This is especially true for perception of radiation risks which might derive from so-called TENORM (Technologically Enhanced Naturally Occuring Radioactive Materials) during the remediation process of polluted industrial sites.

In many cases, remediation of TENORM residues is affected by the people concerned facing inconsistent information (on the actual risks) and/or overlapping legal regulations. The remedial actions aimed at the protection of the residents often raise further concerns. As a consequence even the radiation monitoring required might cause distrust.

This research project was established to elucidate the underlying reasons for communication problems between experts and affected members of the public.

In case studies from two remediation sites in Germany (a residential area on a polluted former industrial site in Hanover and remediation of uranium tailings by WISMUT in Saxony) the risk assessment of experts and the risk perception of laypersons, resp., are analysed on the basis of interviews conducted with experts in the field of radiation protection and remediation, as well as with residents living on former industrial sites polluted with radioactive substances.

Furthermore, primary sources including governmental documents and radiation protection regulations, and press coverage are investigated.

The first results from the analysis of the data collected in this qualitative approach are presented here.

Introduction

Everyday perception of radioactivity is often shaped by the discussion on how safe or risky nuclear power plants or nuclear waste repositories are (Renn 1998; Sjöberg and Drottz-Sjöberg 2009; Slovic 1996). Only when ionising radiation occurs in people's living environment, their houses or gardens, they realise that also other forms of radioactive substances exists, the so-called technically enhanced naturally occuring radioactive materials (TENORM).

The management of TENORM residues is affected by a number of conflicts between the several dimensions of a remediation (Fig. 1).

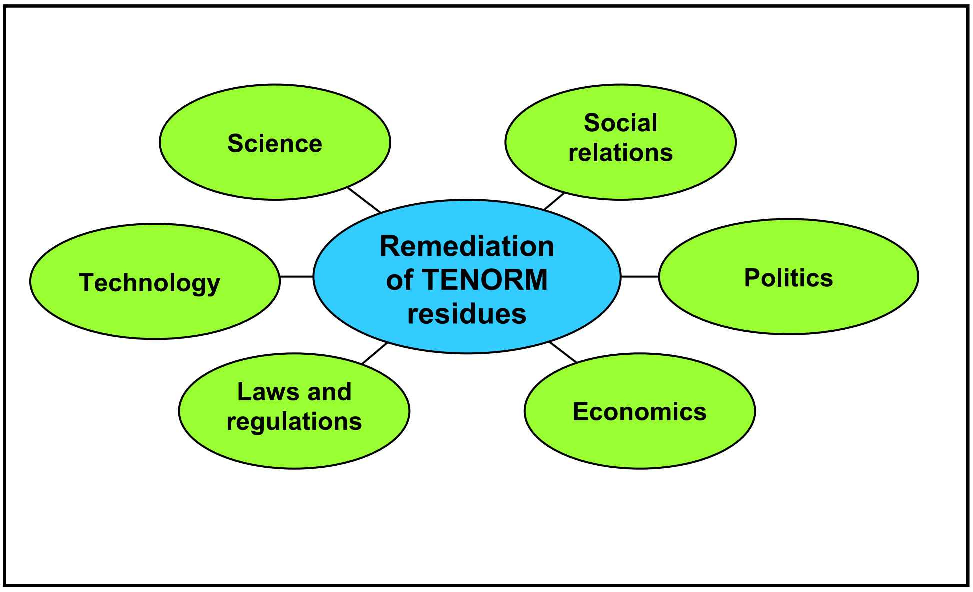

Fig. 1. Dimensions of remediation of TENORM residues.

For example: Contradictory statements by experts and media reports lead to reservations on the part of the residents affected. Errant communication between experts and politicians, and overlapping legal framework raise further confusion. As a consequence, the residents experience remediation actions as new dangers. Whenever authorities or experts try to protect the residents of polluted areas, the residents often consider the actions as unsettling rather than of avail (Wynne 1987).

In the field of radiology as well as sociology, many researchers account the fact that residents and experts often percceive the risk of radiation in very different ways (Slovic 2002). But until now, research on TENORM remediation lacks a clear concept for the management of TENORM residues which combines the professional risk assessment of the experts with the the risk perception of the residents.

In our project, funded by the German Federal Ministry of Education and Research (BMBF), we work out criteria for an evaluation of TENORM remediation. These criteria do not only include radiological but also societal aspects. Using as examples the remediation of an urban district of the provincial capital Hanover and the remediation process of the former area of the uranium mining industry in Saxony, Germany, we study the risk assessement and risk perception of TENORM-remediation. Our aim is to investigate the following questions:

- What are the different understandings of risk and safety for experts and lay persons involved?
- Which basic assumptions underly the predictions of experts?
- How does the familiarisation with ionising radiation, like in an uranium mining area, influence the perception of radiation risk?
- Which social factors determine an appropriate and accepted remediation of TENORM?
- And which interactions between the stakeholders have a recognisable impact on TENORM remediation?

Material and methods

We have chosen a transdisciplinary approach which combines the examination of the scientific as well as the social factors of remediation processes. The collaboration of the Insitute for Radioecology and Radiation Protection and the Institute of Sociology allows to combine scientific and sociological expertise for the examination of the perception and assessment of risk.

Based on analyses of the stakeholders` network of residents, experts, administrators, journalists, politicians etc. we focus on two contrasting cases for remediation of TENORM. The first case concerns the contamination of parts of an urban area in Hanover. The polluted area lies in the very center of the city and strikes one of the most favourable residential areas. Parts of the district were built on the property of an abandoned chemical plant, in which, among other things, uranium and thorium had been processed. In 2008, more than one hundred years after the demolition of the plant, the industrial inspectorate found residues of uranium, thorium and their daughter products in the soil, as well as heavy metals in remarkable concentrations.

The second case refers to the remediation of one of the world`s-biggest uranium minig sites in the last century. From 1946 to the end of 1990 extensive uranium mining operations took place in Saxony and Thuringia (eastern Germany). 220.000 tonnes of uranium were produced and supplied to the former Soviet Union. At the mining and milling sites, 48 waste rock piles were accumulated, containing a total of 300 million cubic meters of mine spoils. Furthermore, 160 cubic meters of suspended ore-processing residues were disposed of in 14 slurry ponds. Due to the fact that uranium was mined in a densely populated and intensively used agricultural area, living conditions for many people in the affected area were impaired significantly. After almost fifteen years of remediation, this process is a striking example for the chances, but also for the pitfalls, deriving from TENORM remediation in Germany (Hagen and Jakubick 2006).

We chose these two cases to compare the impact of an unforeseen confrontation with TENORM like in Hanover with the effects of familiarisation to TENORM residues like in Saxony and Thuringia.

Our research is based on three interconnected methodological approaches. First, we analyse different standard texts in use by the involved experts, e.g. manuals and other standard works on radiation protection, environmental law and occupational health. The analysis of these documents is crucial for a deeper understanding of the experts` risk assessment, their basic assumptions and terminology. Secondly, we will conduct interviews with selected experts who are directly involved. Thirdly, qualitative interviews with residents in the affected areas will complete these analyses.

First results

A first analysis of interviews, media reports and other material already depicts several obstacles constricting the interaction between the different stakeholders.

First, stakeholders use scientific terms such as ´risk`, ´hot spot`, ´contamination` and `remediation` in very different ways. 'Hot spot' e.g.; is a good example for how a term changes its meaning in different contexts. We can see in our case that investigating radiologists use 'hot spot' to name locations with radiation levels higher than the surrounding areas. But when politicians or journalists use 'hot spot' in public discussion, they connect the term with 'danger`. Thus, politicians and residents demand removal of every 'hot spot', even if there is no need from the scientific point of view.

Another area of conflict are the lacks in the legal framework in NORM-regulations in Germany. For instance, there are no specific constraints or even reference levels for indoor radon concentrations. The scientific discussion about the level from which on radon causes cancer is still going on. Therefore, the decision makers involved in remediation processes have to rely on diverging recommendations given by different organisations such as WHO, the German Federal Office for Radiation Protection (BfS) and the ICRP, respectively.

Regarding residents` risk perception we observe that two main factors having a substantial impact – factors beyond concerns about physical damage. First, residents fear economic damage, namely the reduction in value that their properties suffer. As a consequence they are also worried about the impending social decline of their neighborhood, and therefore a recognisable damage in their social status.

Conclusion and future work

Having started our project in January 2010 the first results already indicate that communication problems exist on different levels of the stakeholder interactions in an on-going TENORM remediation process. Barriers become prominent between radiation experts and politicians, between radiation experts and decision makers in the administration, and between scientific risk experts and residents, respectively. In our future work we will try to reveal further relevant communication obstacles, and track to which extent the different obstacles are interconnected.

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Thyroid measurement campaign after an accidental release of 45 GBq ^{131}I in Fleurus, Belgium

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Abstract

On August 22, 2008, a cloud of ^{131}I was released from the Institut des Radioelements (IRE), a producer of medical isotopes located in Fleurus, Belgium. The peak of the release took place in the weekend of 23-24 August and was followed by a smaller, continuous release which lasted for several weeks.

After reports of the incident in the press, concern grew among the local population about possible health effects of this release of radioactivity.

As a consequence of this concern, the Belgian public health authorities organised a thyroid measurement campaign for the local public to reassure the public that no detectable dose was acquired in the thyroid, especially for small children and pregnant women.

The Belgian Nuclear Research Centre SCK•CEN made a very substantial contribution to the thyroid measurement campaign. It provided 4 of the 5 measurement teams, each one equipped with a high-purity germanium detector and the necessary hardware and software. The University of Liège provided another team with a NaI detector.

All detectors had been calibrated in advance and were checked regularly during the measurement campaign.

The thyroid measurement campaign took place on Monday September 1st and Tuesday, September 2nd 2008 in the village of Lambusart close to Fleurus. More than 1000 people were measured, including all children from the local schools. No thyroid contamination was detected.

This paper elaborates on the preparation of this measurement campaign and the practical execution. It draws important lessons for the preparedness for similar situations in the future, both with respect to public information and technical and organisational aspects.

Introduction

During a radiological or nuclear emergency, it is of utmost importance to maintain (or establish) public thrust in the authorities for a good implementation of countermeasures and to prevent ill-based panic.

On August 22, 2008, a cloud of ^{131}I was released from the Institut des Radioelements (IRE), a producer of medical isotopes located in Fleurus, Belgium. The peak of the release took place in the weekend of 23-24 August and was followed by a smaller, continuous release which lasted for several weeks.

After reports of the incident in the press, concern grew among the local population about possible health effects caused by this release of radioactivity.

As a consequence the Belgian public health authorities decided to organise a thyroid measurement campaign for the local public to reassure the public that no detectable dose was acquired in the thyroid. Especially small children and pregnant women, the most sensitive group, were envisaged.

The Belgian Nuclear Research Centre SCK•CEN and the University of Liège were asked to execute the technical part of the thyroid measurement campaign. SCK•CEN provided 4 out of 5 measurement teams, each one equipped with a high-purity germanium detector and the necessary hardware and software. The University of Liège provided another team with a NaI detector and associated hardware and software.

Description of the measurement campaign

Goal (technical and societal)

The technical goal of the measurements was to measure at least a ^{131}I contamination in the thyroid that would result in an effective dose of 2 mSv for a child.

Based on the assumption that the intake of ^{131}I took place via inhalation, Absorption Type F, the predicted values for activity in the thyroid per inhaled activity [ICRP 78] and the ICRP dose conversion factor for ^{131}I in the thyroid (inhalation, AMAD 1 µm, Absorption Type F [ICRP 71]) a measurement time of 1 minute was derived in order to be able to measure a 2 mSv thyroid equivalent dose for a child, equivalent to a 100 Bq activity in the thyroid 8 days after the major release.

An analysis of the measured gamma-spectra after the measurement campaign revealed that the obtained detection limits were in the range of 20-90 Bq. The observed variation is due to differences in the detectors, mainly the efficiency.

The maximum detection limit of 2 mSv is well below the Belgian intervention level of 10 mSv thyroid equivalent dose for children and pregnant women.

Based on the known release of ^{131}I, it was considered as extremely unlikely that a measurable contamination could have occurred. However, in view of the concern among the local population, it was decided by the authorities to organise this measurement campaign in order to reassure the public. The campaign focussed on children, but adults that desired to be checked were given the opportunity to be measured after the children were measured.

Used equipment

Four detectors were used:

- 1 LEGe detector with electronics and Genie2000 analysis software
- 1 MAC Ge detector with electronics and Genie2000 analysis software
- 2 Ge detectors with ISOCS efficiency calibration.

The acquisition, analysis and calculation of the detection limits was done with Genie2000 (Canberra) software. Detection limits were calculated with the Currie method. The detection limit is stated as the smallest net signal that can reliably be quantified. It can be calculated for the radionuclides not found in the spectrum, as well as for those that have been found. Under ideal laboratory conditions (low background, long measurement time), a detection limit as low as 1 Bq can be achieved, but this was not the case for this measurement campaign as explained below.

Calibration was performed with a ^{133}Ba source in a thyroid neck phantom as shown in figure 1. The measurement setup of the University of Liège was also checked with this calibration phantom for consistency of the measurement results.

Figure 1. IRSN thyroid phantom for intercomparison (left), SCK•CEN thyroid phantom for calibration (right).

Practical execution of the measurements

The 4 detectors were placed in two changing rooms of the sports hall *'Salle Omnisport'* of the village of Lambusart, close to Fleurus. This village is located in the main wind direction of the release. The setup for the measurements consisted of placing the person with his/her thyroid gland (located just under the Adam's apple in the neck) in front of the detector at 2-5 cm distance (figure 2). A measurement was then started for 1 minute during which the person had to stay as still as possible. The measurement is based on the detection of the 364.5 keV gamma peak of ^{131}I.

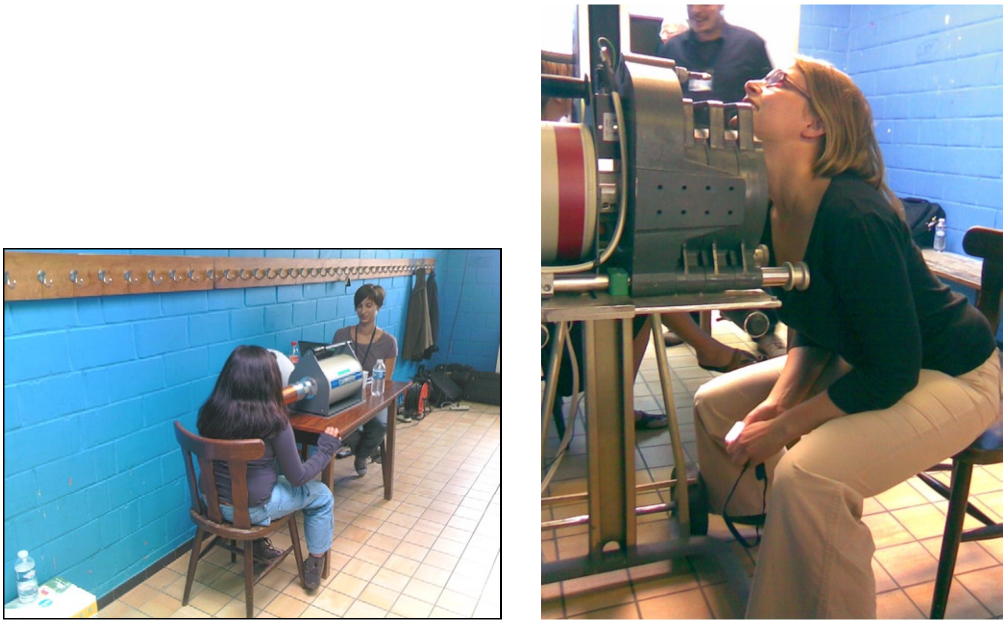

Figure 2. Thyroid measurement setup

After the gammaspectrum is acquired, the analysis is done immediately and indicates if any activity of ^{131}I was detected in the thyroid. No activity detected means that with a confidence of 95%, the activity in the thyroid is lower than the calculated detection limit.

Measuring in an unshielded room and reducing the measurement time to 1 minute increased the detection limits considerably compared to the above-mentioned ideal laboratory conditions. The use of different detectors and the varying background activity in the environment resulted in detection limits ranging from 20 to 90 Bq for the detection of ^{131}I..

The result of the measurement was immediately communicated to the person involved and it was indicated on the result sheet with an identification number. 'SCK•CEN' was written on the results sheets of persons measured by any setup of SCK•CEN. When no activity was detected, a result of 'risque nul' (zero risk) was indicated on the result sheet of the measured person. This was done on demand of the responsible from Public Health although scientifically we could not state the activity was zero.

Target group

The authorities decided to focus first on the most vulnerable group, children and pregnant women. All children from the schools in the vicinity of Lambusart were transported to the sports hall to be measured, while also all pregnant women were requested to come to the measurement location.

After these groups had been measured, the other members of the local public could come and let themselves be measured.

SCK•CEN performed 876 recorded measurements, all with a negative result. Additionally about 200 persons were monitored with equally negative results, but without recording the results in writing. This was done at their explicit request.

Chronology

On Friday August 29th the public health authorities asked the SCK•CEN representative at the federal crisis centre whether SCK•CEN would be capable to organise a thyroid measurement campaign for the local population of Lambusart, with an estimated 3000 inhabitants. The same date SCK•CEN wrote a note that stated that such a campaign would be possible.

On Saturday August 30th SCK•CEN received an order of the Ministry of Public Health to provide measurement teams for the thyroid measurement campaign. The same day the relevant experts were contacted to prepare this campaign.

On Sunday August 31st two available measurement chains were prepared. Soon it was realised that these two setups would not be sufficient to measure 3000 people and it was decided to include the two additional ISOCS systems to have four systems available. The complete team was assembled, consisting of 4 measurement teams with one operator and one guide, an expert in electronics, a team leader and a radiological expert for informing people.

Monday September 1st at 5:45 a.m. arrival of the team at SCK•CEN for the last preparations. Departure to Lambusart at 6:30 a.m.for arrival at Lambusart at 8:30 a.m.. Set up of equipment took 1 hour and final briefing of the teams at 9:30 a.m. Start of the measurements at 10:00 a.m. From 10:00 a.m till 01:00 p.m. measurement of limited numbers of children and pregnant women. After 13:00 all inhabitants could come and have themselves measured. Between 16:00 and 19:30 there was a peak, resulting in relatively long queues. After 20:00 the campaign was finished. Several team members had to stay for an information session for the local population from 21:00 till 23:00. After this session the public health authorities decided to extend the measurement campaign at the next day. Departure at 24:00.

Tuesday September 2nd, two measurement teams were sent instead of four. Measurement campaign was executed from 16:00 till 20:00. The team consisted of two measurement teams of two persons, a team leader, a radiological expert and a communication expert. No electronics expert was present, but a third Ge detector was present in case replacement is necessary.

Interactions with/reactions of the public

Most members of the public did not ask more information. However, when an initiative was taken to provide them with more information, the majority reacted positively and asked for more information. A considerable minority considered the campaign as a cover-up of the authorities, but nevertheless accepted a measurement.

The minority that asked more information on its own initiative was concerned whether the measurements itself could harm them and wanted to know more about the health effects of ^{131}I. Less questions were asked about how the measurements were performed and how reliable the measurements were.

Performing measurements on small children is a trick on its own. The size of the Ge detectors was rather impressive for the youngest (0-5 years) and required distractors like laboratory gloves blown as balloons. For that group, the NaI detector of the University of Liège was less frightening since its size was much smaller, but its efficiency was also lower.

An additional disadvantage was that September 1st is the first schoolday, so most teachers were not yet acquainted with the children and had not yet established a bond of trust.

The vast majority considered SCK•CEN and the University of Liège as impartial and therefore trusted the quality and outcome of the measurements. This would not have been the case if IRE, which caused the release, would have carried out the measurements.

The vast majority of the team on the first day consisted of native Dutch-speaking people, with one exception. Although many of them spoke reasonably well French, communication with the French-speaking population led sometimes to misunderstandings, certainly when it concerned children. It was decided for the second day to compose teams with a better French-speaking ability.

Possible improvements

It has to be admitted that the existing measurement equipment just had been calibrated for thyroid measurements, rather as a coincidence. Actions are ongoing now to extend calibrations for a range of isotopes (^{131}I, ^{137}Cs, ^{60}Co, ^{235}U and ^{238}U) and contamination types (thyroid, whole body, lungs).

There was a clear demand of the general public to be informed, but the initiative should come from the experts. The public is reluctant to ask experts about their expertise, but is eager to listen and learn when something is explained. Therefore communication to the public should be part of the overall strategy, explaining first of all of the health effects of radiation and secondly of the principles of the measurement method.

To establish and maintain experience in these kind of campaigns, regular exercises should be undertaken as part of the overall emergency planning exercises. The different types of measurements (thyroid, Whole Body Counting, lungs) should be trained to understand the specific problems that occur in practice.

Conclusions

During a measurement campaign, the public needs to be informed actively, since the majority will not ask information, but is nevertheless interested when more information is provided. The information given should focus first on the health effects of radiation and secondly on the radiation measurement principles.

Very young children (age 0-5 years) should be measured in a reassuring environment. Toys, ballons and lively colours will distract their attention from the impressive measurement equipment.

Measurements should be executed by institutes that are not related with the institute responsible for the radiological problem and that are considered by the general public as independent and impartial.

Calibrations of existing measurement equipment for specific cases are time-consuming and cannot be performed in times of crisis. These calibrations should be performed for all different types well beforehand and should be checked by intercomparison exercises.

To maintain the operationality of the measurement teams, exercises should regularly be organised to test also the aspect of measuring large number of people.

In a multilingual country measurement teams should be partly composed of native speakers of the local language to provide a clear communication with them.

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Designing of stakeholder meetings for consensus development through the stakeholder involvement in the field of nuclear energy utilization

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Abstract

In order to develop trustful relations and to harmonize with society for the use of nuclear energy and the operation of nuclear facilities, it is essential for the state governments and electric power utilities to design the appropriate measures for communication and dialogue with stakeholders. The one-way information distribution should be avoided and the two-way information exchange system should be build up between the administration and the public.

In this paper, we focused on the designing of the stakeholder meetings and comparatively studied the cases in Japan, France and UK. As the result, it is found that there are differences in the selection of stakeholders and the designing of the meetings. These differences are probably relating with the social backgrounds and cultural differences in the countries. The experiences in the French system would be informative for the Japanese stakeholder meetings.

In Japan, one of the biggest issues has been the finding of the candidate site for the high-level waste disposal facility. Toyo town in Kochi prefecture applied as the candidate, for the first time in January 2007. However, due to the strong opposition movement developed in Toyo town and Kochi prefecture, the application was cancelled. Based on the results of this case, we considered the problems of the stakeholder involvements and the differences in the designs of the stakeholder meeting between the existing nuclear facilities and the planning facilities.

Introduction

In the past one and half decades, a number of troubles, failures, accidents and manipulations have occurred in the nuclear industries in Japan. In spite of the world wide nuclear Renaissance, good relationship between the public and the nuclear industries has not yet been created in Japan. To eliminate the distrustful concern of the public against the operations of nuclear facilities, and to develop good social relations, it is important to facilitate the communications of two-way talks among stakeholders. In the present work, we comparatively analyze the stakeholder meetings established with the aim of safety assurance of the nuclear facilities in the hosting region in France, Japan and UK, paying attention to the designing of the stakeholder meeting.

Subject of Analysis

The approaches and the efforts have been made in various countries to establish and to run the stakeholder meetings in order to ensure the safety and the transparency on the operation of the nuclear facilities in the local area. The present study surveys and analyzes the designing of the stakeholder meetings developed. We selected the following stakeholder meetings for the analysis in the present work, paying attention to the participating members.

Stakeholder meetings in Japan

– Community Meeting for Securing Transparency of Kashiwazaki-Kariwa Nuclear Power Station (hereinafter referred to as "The Community Meeting")
– Monitoring and Evaluation Conference on Environmental Radiation

Stakeholder meetings in France

– CLI AREVA NC La Hague
– CLIS de Fessenheim

Stakeholder meetings in UK

– West Cumbria Sites Stakeholder Group (WCSSG)
– Sizewell A&B Stakeholder Group

Method for Investigation

We conducted the literature survey on the design and the laws and regulations relating to the stakeholder meetings in Japan, France and UK.

We also conducted the interviews to survey the activities, the membership and the regulations for the following stakeholder meetings: The Community Meeting (Japan); CLI AREVA NC La Hague (France); CLIS de Fessenheim (France); West Cumbria Sites Stakeholder Group (WCSSG) (UK).

Analysis

Japan

The establishment of the stakeholder meetings is not required by laws and regulations for the operation of nuclear facilities in Japan. The municipalities hosting nuclear facilities establish the Regional Committee for Monitoring and Evaluation of Environmental Radiation, in order to protect health and safety of local residents. Its purpose is to evaluate the results of the monitoring data around the nuclear facilities. In addition, Niigata prefecture and other local authorities establish a stakeholder meeting in the area located Kashiwazaki-Kariwa Nuclear Power Station.

1. Community Meeting for Securing Transparency of Kashiwazaki-Kariwa Nuclear Power Station (The Community Meeting)

In July 2000, a former employee of an affiliate company of General Electric reported the Ministry of Trade and Industry the dishonesty in the inspection records of Unit No.1 of Tokyo Electric Power Company's Fukushima Daiichi Nuclear Power Plant. In August 2002, the Nuclear and Industrial Safety Agency announced to the public that there were a total of 29 improper inspection records, in which the four cases occurred in Kashiwazaki-Kariwa Nuclear Power Plant (Community Meeting 2002, 2003a).

In order to prevent the reproduction of the occurrence and to restore the trustful relations, Tokyo Electric Power Company proposed to set up a third party organization in both Fukushima and Niigata Prefectures, where nuclear power plants are located. However, Niigata Prefecture, as well as the municipalities of Kashiwazaki-city, Nishiyama-town, and Kariwa-village, had misgivings about the proposal, sensing that the electric power utilities might be using it as nothing more than a public relations activity. They came to the realization that a public-participation type conference in the host region was necessary to prevent future dishonest actions by the electric power utilities and to enhance transparency in the operation of nuclear power plants. Thus, in 2003, the "Community Meeting for Securing Transparency of Kashiwazaki-Kariwa Nuclear Power Station" (hereinafter referred to as "The Community Meeting") was established on a voluntary basis by local authorities (Community Meeting 2002).

The purpose of The Community Meeting is to conduct monitoring from the viewpoint of local residents, placing emphasis on securing transparency of the activities at the power station, in order to prevent the re-occurrence of dishonest actions in Kashiwazaki-Kariwa Nuclear Power Plants.

Membership of The Community Meeting is to be composed of a maximum of 25 members residing in Kashiwazaki City or Kariwa Village, recommended by the 21 organizations approved by the regional community (Community Meeting 2003a, 2003b, 2008).

These organizations include in Kashiwazaki-city, the Energy Forum in Kashiwazaki, the Kariwa village Society of Commerce and Industry, Association to protect Kariwa village from nuclear power plant, and so on. The membership of The Community Meeting consists of nuclear power promoters, neutral parties, and opponents. This is characteristic of The Community Meeting. The site operator (Tokyo Electric Power Company) and the regulatory bodies participate as elucidators in the regular conference of The Community Meeting.

2. Monitoring and Evaluation Conference on Environmental Radiation

As the prefectures that host nuclear facilities have a duty to protect the health and safety of their residents, they are involved in surveying environmental radiation in the vicinity of these facilities. The "Guidelines for Monitoring Environmental Radiation" stipulate that monitoring shall be uniform between local public entities and electric power utilities, and that evaluation of monitoring results should, depending on local situations, be made by a monitoring and evaluation organization consisting of stakeholders, including local public entities and residents. In accordance with these regulations, monitoring and evaluation conferences on environmental radiation have been organized in the prefectures hosting the nuclear facilities. Japanese laws do not stipulate the membership and number of members of Monitoring and Evaluation Conference on Environmental Radiation.

The membership of these conferences consist of key players, mayors of cities, towns, and villages in the region, and representatives of local industries (Monitoring and Evaluation Conference on Environmental Radiation 2007). The memberships of

Monitoring and Evaluation Conference on Environmental Radiation in Aomori-prefecture and Niigata-prefecture are shown in Table 1.

France

The local information committees (CLI: Commission Locale d'Information) are established in the area where the nuclear facilities are located in France. The purpose is to engage in dialogue and communicate with stakeholders, in order to nurture a committed relationship among stakeholders. CLIs was established on the basis of the notification dated 4 September 1981 from the minister of industry and the notification dated 15 December 1981 from the Prime minister. Then, the establishment of CLI is required under article 22 of Nuclear Security and Transparency Act in 2006. There are about 30 CLIs around the civil nuclear facilities in France. This study chooses 2 CLIs as the subjects of analysis: CLI AREVA NC La Hague and CLIS de Fessenheim. That is because they have operated for about 30 years and they are still active.

The constitution of CLI is stipulated in the legislative decree dated 12 March 2008 (No. 2008-251). The membership and their component ratio are defined under article 5.

1. CLI AREVA NC La Hague

CLI AREVA NC La Hague was established on the basis of the notification dated 4 September 1981 from the minister of industry (Hervé 1981). Originally its name was CSPI (Commission Spéciale et Permanente d'Information). CSPI changed its name to CLI AREVA NC de La Hague after enacting Nuclear Security and Transparency Act in 2006.

As of 2010, CLI AREVA NC La Hague consists of 23 local assembly members, 6 representatives of environmental protection groups, 6 representatives of labour unions, 3 representatives of chamber of commerce, 1 representative of constabulary, 6 certified person in nuclear engineering and communication. Total number is 45. The site operator (AREVA) and safety authority are not members but observers. The membership and the component ratio are defined under article 5 of the legislative decree dated 12 March 2008 (No. 2008-251).

2. CLIS de Fessenheim

CLIS de Fessenheim was established in 1977. This was before the notification dated 15 December 1981 from the Prime minister (Mauroy 1981). Fessenheim nuclear power station is the first site where a PWR was constructed in France. At that time, local residents concerned about the PWR which was different from the existing gas-cooled reactor. Therefore CLIS de Fessenheim has the purpose to respond the concerns among local residents and to conduct surveillance of Fessenheim nuclear power station. This is because CLIS de Fessenheim has a character "S" which means "surveillance."

There are 40 members in CLIS de Fessenheim as of 2010. The membership and the component ratio are defined under article 5 of the legislative decree dated 12 March 2008 (No. 2008-251).

The feature of membership of CLIS de Fessenheim is participation from other country. Fessenheim nuclear power station is located near the German border, so that 2 members from Germany participate (Conseil Général Hault-Rhin 2008).

UK

In UK, Local Liaison Committee (LLC) and/or Site Stakeholder Group (SSG) are established around the area where nuclear facilities are located. Various stakeholders (municipalities, labour unions, environmental protection groups, and so on) are included in the members of LLC and SSG. The purposes are to share the information, to have a dialogue and to communicate among stakeholders. This study chooses 2 SSGs as the subjects of analysis: West Cumbria Sites Stakeholder Group (WCSSG) and Sizewell A&B Stakeholder Group.

1. *West Cumbria Sites Stakeholder Group (WCSSG)*

The Sellafiled Local Liaison Committee (SLLC) was established to provide information and a forum of local community around the area of the Sellafiled reprocessing plant. It was replaced to the West Cumbria Sites Stakeholder Group (WCSSG) in April 2005. WCSSG provides a forum for representation of local community interests and is the interface between the community, the site operators and the Nuclear Decommissioning Authority (NDA). The establishment of WCSSG is on the basis of NDA Guidance (NDA 2009). It is not a law or a regulation.

WCSSG has six sub-committees, detailed aspects of the Sellafield and Low Level Waste Repository sites, including operational issues, environment health, emergency planning and socio-economic impacts.

WCSSG membership reflects the representational structure of the local community and its interests. The membership is as follows:

- Elected representatives of the local community.
- Appointed representatives of relevant organisations such as regulators, local authorities, unions, emergency and health services.
- A representative of the NDA.
- Representatives of the site contractor/operator.
- Independent advice to support members as appropriate.
- Representation from members of the public and local environmental groups.

The core membership of WCSSG is shown in Table 1. It consists of the representatives of the site operators, regulators, local councils, fire services, police, farmers union and so on (WCSSG 2006).

2. *Sizewell A&B Stakeholder Group*

The Sizewell A&B Stakeholder Group was established and run around the area of Sizewell A&B Nuclear Power Station. The purpose of the Sizewell A&B Stakeholder Group is to:

- inform the public of activities of Sizewell site,
- act as a conduct for two-way information provision and flow,
- act as a “clearing-house” for community concerns by providing independent interpretation of information.

The membership list of the Sizewell A&B Stakeholder Group is shown in Table 1. The Sizewell A&B Stakeholder Group is characterized by inclusion of the representatives of local council and the site operator, as well as WCSSG.

Results

Table 1 shows the design of the stakeholder meetings already-described.

Table 1. Design of the stakeholder meetings around the local areas of the nuclear facilities in Japan, France and UK.

Stakeholder meeting	Country	Numbers	Membership
The Community Meeting	Japan	within 25	Local residents living in Kashiwazaki City or Kariwa Village, recommended by the 21 organizations approved by the regional community
Monitoring and Evaluation Conference on Environmental Radiation in Aomori-prefecture	Japan	within 80	25 experts 8 local key figures 2 prefectural assembly members 8 mayors in municipalities 21 representatives of concerned associations 6 prefectural government's employments
Monitoring and Evaluation Conference on Environmental Radiation in Niigata-prefecture	Japan	21	10 experts 5 representatives of concerned associations* 1 governor of Niigata Prefecture 1 mayor of Kashiwazaki city 3 prefectural government's employments
CLI AREVA NC La Hague	France	45	23 local assembly members 6 representatives of environmental protection groups 6 representatives of labour unions 3 representatives of chamber of commerce 1 representavie of constabulary 6 certified person in nuclear engineering and communication
CLIS de Fessenheim	France	40	20 local assembly members 6 representatives of labour unions 7 representatives in the business community and experts 7 representatives of environmental protection groups
West Cumbria Sites Stakeholder Group (WCSSG)	UK	51	Members and officers of the local councils Representatives of the local parish councils Representative of the local supply chain Representative of the economic regeneration programme Representative of local NGO groups Representatives of Health Authorities Representative of Cumbria Constabulary Representative of Cumbria Fire Service Representative of Isle of Man Department of Local Government and the Environment Representative of Health and Safety Executive (NII) Representative of Environment Agency Representative of DEFRA Representative of FSA Representative of Government Office North West Representative of United Utilities Representative of National Farmers Union Representative of Whitehaven and District Trades Council Representatives of Site Operators (ie: Sellafield Ltd, LLWR) Representatives of labour unions

* The concerned associations are local medical association, the local chamber of commerce, the fishery association, the farmers cooperative association and so on.

Table 1. Continued.

Stakeholder meeting	Country	Numbers	Membership
Sizewell A&B Stakeholder Group	UK	22	2 representatives of Leiston-cum-Sizewell Town Council 1 representatives of Aldeburgh Town Council 1 representatives of Saxmundham Town Council 1 representatives of Aldringham-cum-Thorpe and Knodishall Parish Councils jointly 1 representatives of Middleton and Theberton Parish Councils jointly 1 representatives of Westleton Parish Council and Dunwich Parish Meeting jointly 1 representatives of Suffolk Coastal District Council 1 representatives of Suffolk County Council The Member of Parliament for Suffolk Coastal Parliamentary Constituency or their representative 1 representatives of the Suffolk Association of Local Councils 1 representatives of the Sizewell Residents 1 representatives of the Leiston Business Association 1 representatives ofg the National Farmer's Union and Country Landowners' & Business Association jointly 2 representatives of the staff of Sizewell A 2 representatives of the staff of Sizewell B

The feature of The Community Meeting is inclusion of anti-nuclear groups in membership. On the other hand, Monitoring and Evaluation Conference on Environmental Radiation do not have any members of anti-nuclear groups. But it is charactered by consideration to local industries, because many representatives of the local industry associations participate in the Conference.

In France, the membership and the component ratio are defined under article 5 of the legislative decree dated 12 March 2008 (No. 2008-251). The result of analysis found that the ratio adhere fundamentally to the composition of the notification dated 15 December 1981 from the Prime minister. CLI consists of local assembly members, representatives of environmental protection groups, labour unions, certified person in nuclear engineering and communication, and so on. Local assembly members are as the position of representatives from local residents.

The memberships between a nuclear power station and a spent fuel reprocessing plant are the same in France.

WCSSG has the members of local fire services and police. WCSSG has a sub-committee on the emergency plan. At the nuclear emergency in UK, Police has to play a central role, so that WCSSG has the members of police. On the other hand, fire services and police are not the core members of Sizewell A&B SSG and CLI, because they do not work with them in the emergency situation.

The participants of these stakeholder meetings in Table 1 are not selected by independent organizations.

Discussion

As the result of analysis, the following characteristics are found.

- Each stakeholder meetings has the members as the representatives from local residents, who are affected by the consequence of the facility accidents and concerned with the activities of nuclear facilities.
- The Community Meeting intentionally excludes the representatives of labour unions and the site operator from membership. This is because that the role of The Community Meeting is to represent the opinions of the residents.
- Anti-nuclear groups are not participating in Monitoring and Evaluation Conference on Environmental Radiation.
- The experts participate in CLI and Monitoring and Evaluation Conference on Environmental Radiation. OECD/NEA points out the necessity of participation of neutral persons who are reliable and having high expertise in relating technical fields (OECD/NEA 2004).
- In CLIS de Fessenheim, 2 members are from Germany, reflecting the regional situation.

We think that following items are important at the designing of the stakeholder meetings: well-balanced selection of various stakeholders and the consideration of local characteristics and individual situations.

In Japan, one of the biggest issues has been the finding of the candidate site for the high-level waste disposal facility. Nuclear Waste Management Organization of Japan (NUMO) has been in charge of disposal business. Toyo town in Kochi prefecture applied to NUMO as the candidate of the study area, for the first time in January 2007. The Japanese government approved the initiation of preliminary literature investigations to survey the previous records of the earthquakes and so on. However, due to the strong opposition movement developed in Toyo town and Kochi prefecture, the application was cancelled. A number of opponents gathered in Toyo town from distant areas. The results indicates that multi stakeholders meeting is needed in the site area, to develop appropriate consensus among stakeholders. In the designing, the stakeholder meeting should have the core stakeholder which represent the core area of the region hosting the facilities. The distance from the nuclear facilities would be one of the factors to define core stakeholders.

The membership constitution of CLI is the same as one for other nuclear facilities. In France, the design of the stakeholder meeting for the construction of a future new reactor is the same as that of existing nuclear facilities. In the case of a future new nuclear facility, just like the high-level waste disposal facility, based on the result of Toyo town case, it may be necessary to design the core stakeholders and distinguish it from other stakeholders in the future energy policy making.

Conclusion

The result of analysis from this study reveals that the designing and selection of participating stakeholders should be different facility by facility in accordance with the purpose and expecting roles of stakeholder meetings.

We think a diversity of stakeholders and its selection are important to realize the broad-ranging discussion and dialogue. If the stakeholder meeting like CLI and SSG will be introduced into Japan, it is deemed desirable to flexibly design the meeting and the membership reflecting local characteristics, like CLIS de Fessenheim and Monitoring and Evaluation Conference on Environmental Radiation in Aomori and Niigata prefectures.

The study on the stakeholder meetings in other counties such as North European countries would be interesting.

Note

For this survey, we contacted the secretariat of the Community Meeting on 28 September and 5 October 2007, by e-mails. We interviewed the president of the Community Meeting on 29 October 2007 in Yokohama, and on 8 August 2008 in Tokyo. We interviewed the secretariat of CLI AREVA NC La Hague in Cherbourg on 21 February, 2008. We participate sat in on a regular conference on 17 December 2009 in order to grasp the situation. We interviewed the secretariat of CLIS de Fessenheim in Colmar on 15 January, 2009. We interviewed 2 members of WCSSG in Cleator near Sellafield reprocessing plant on 2 March 2010. We are thankful for the help of all concerned.

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Information technologies in radiation protection legal regulative in Serbia

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Abstract
Books, documents, lows and other regulatory papers were accessible only for the limited number of users. Development of information technologies enable easy and quick access to radiation protection regulatory as well as to the other documents not only to the experts but also to the public. In this paper we gave example of information technology application in radiation protection regulative of Serbia

Introduction
Books, reports, texts of laws and accompanying documents were available only for restricted number of experts and governmental clerks. Development of information technologies (IT) enabled digitalization of libraries and archives and made them also available to the lawyers, journalists and to general public. From mid 1990 legislature started with internet communication in aim to enable free or semi-free access to their activities and documents not only for employees but also to the public. Internet is only a part in evolution of so called E-government and two-way communication between governmental structures and citizens.

Investigations in our country have shown that citizens were generally satisfied with electronic services which also have included possibility of getting information about document drafts, radiological situation in the country as well as about some actions with direct impact of public security. Investigations also have shown that there was some doubt about security information. In building of Information system we started from EU good practice in European Parliament, US Congress and House of Commons in UK. We gave a copies of web sites relevant for the radiation protection.(EC,2005)

Material and methods

Legislative bodies and Web
At the very beginning legislative webs have been created to support the three main goals: Activities and distribution of the documents; Communication with citizens and Transparency. It was the big step forward as at the time of Chernobyl accident there was a strict state embargo on all information. From the internal point of view webs allow to the staff and members of legislative bodies to improve organization and

responsibility hierarchy and also to be inform in time about all meetings, agendas, documentations and monitoring data. In the other side citizens can be better informed about all relevant legislation, final documents, drafts and changes, monitoring data which could be freely shared but also they can find brief description about the legislative procedures, voting and possibility to ask a questions. Some political parties have their own sites dedicated to radiation protection but they are not the part of official information system and their data are not valid for taking an actions and preventive measures. Pressures of press, various lobbyists as well as general public have a great influence on amount of data and information available to the public which enable continuous transparency in legislative bodies work.

Citizens and E-government

Generally, about 62 % of citizens which have internet assess used web site of National assembly to find published or drafted laws and rules of procedures. Less than 20 % know that there is a possibility to find monitoring data at the site of the Ministry of the Environment. Unfortunately, only about 25 % citizens of Serbia have internet access. Although the internet became the central part of civilize society it is obvious that the further development is necessary. This development must offer interactive approach and enable to the public to have more active role in legislative process. In radiation protection it is not so easy as the citizens, journalist and other interested groups must have at least basic knowledge in radiation protection quantities and units and dose limits. Such educative materials are not available at our official sites linked to the governmental institutions. Citizens expect not only to have information about radiation protection legislative but also to expose their own opinion and to debate. Truly the possibility of interactive approach is very poor at governmental official sites. From that point of view it is necessary to look critically on interactivity and some proposals and remarks given during the public discussion of vital radiation protection law: Law of radiation protection and nuclear security.

New standards for information systems in radiation protection

Information and documents available at official sites have to fulfilled some basic standards: accuracy, timeliness, completeness, clarity and content and links. Design of information system must fulfilled five basic criteria but also have to estimate the basic requirements for links between E-government and citizens and to identify the best possible practice.

Interactive two-way approach has a challenge in the nature of inputs and the fate of all remarks and numerous questions of citizens. Some questions are easy to be answered but some of them are complex and take a lot of time and engagement of experts. Additional complication is terminology as laws and accompanying documents use special juristic and professional terminology hardly understandable to average citizen. Users of information system are very differ in knowledge, juristic education and in capability to understand legal processes., therefore Information system have to be built in multiple layers "in depth" suitable for all categories of users: members of legislative bodies, students, professors, various lobbyists, pres, Citizens Associations, foreign governments and their accredited representatives as well as average citizens. This system is not built jet in our country.

Sites are multilingual, in Serbian (Latin and Cyrillic) and in English.

Objectives of designed information system in radiation protection legislation based upon (Griffith, 2006) can be summarized in eight points:

1. Development of coordinated system which fulfill all requirements of legislator and accompanying agencies and bodies.
2. Assurance of free or semi-free access to information and texts obtained from authorized sources,
3. Assurance of integrated system for location of the most important data in legislative
4. Assurance of coordinated interactive system which will avoid duplicating
5. Assurance of tracking the new technologies and continuous site innovation
6. Assurance of reliable and safety technical environment for interactive system
7. Establishment of systems and procedures for long-life stability and reliability of information system in legislation
8. Assurance of appropriate support of new and temporary users as well as experts

Results

Tracking system recommended by legislator is tracking from the initiative (in Ministry) to adoption in National Assembly. Tracking system gives also the short or complete text of legal document. The tracking is realized through: internal portal for lawyers, constitutional court site, competent ministries, National Assembly of Republic of Serbia official site and VINCA Institute of Nuclear Sciences site.

Internal portal for lawyers is dedicated portal for professionals which offers information about: primary and secondary juristic sources (legislative, court decisions, journals, data base, libraries and browsers etc), relevant institutions (faculties of law, NGOs, institutes, lawyers bars and chambers, courts, international institutions and individuals. The most important portal is *find a law* www.pronadjipravo.com.

Fig. 1. Home page of Portal for lawyers (www.pronadjipravo.com).

Constitutional court side offers information related to competence of this court and gives short educative texts about processing cases at this court. www.ustavni.sud.rs

Fig. 2. Home page of Constitutional court (www.ustavni.sud.rs).

Agency for Radiation Protection and Nuclear Security is in the process of establishing. All activities related to radiation protection and nuclear security regulation are still in competence of the Ministry of Science and Technological Development. This Ministry was a proponent of all laws and legal documents dedicated to radiation protection.

Fig. 3. Home page of Ministry of Science and Technological development (www.mntr.gov.rs).

National Assembly of Republic of Serbia is the prime legislative body which adopts all laws and legal documents. Assembly site gives a numerous information and data about assembly activities including complete text of laws in procedures and adopted laws, The disadvantage of this site is extremely poor and non-updated interactivity. www.parlament.rs

Fig. 4. Home page of National assembly (www.parlament.rs).

Carrier of supervisory functions in radiation protection and nuclear security is Ministry of Environment and Spatial Planning. The most important part of this site is link to the System of early warning of radiation accident which gives data about ambient dose equivalent rate at every 30 minutes.

Fig. 5. Home page of Ministry of Environment and Spatial Planning (www.ekoserb.gov.rs).

Significant link bank in chain of radiation protection legislation is the only nuclear institute on our country: VINCA Institute of nuclear sciences. On Vinca site it is possible to find a lot of useful information related to Institute activities. In this very moment when the decommission of one reactor is in progress it is possible to find monitoring data from the Institute territory which are complementary with previous data.

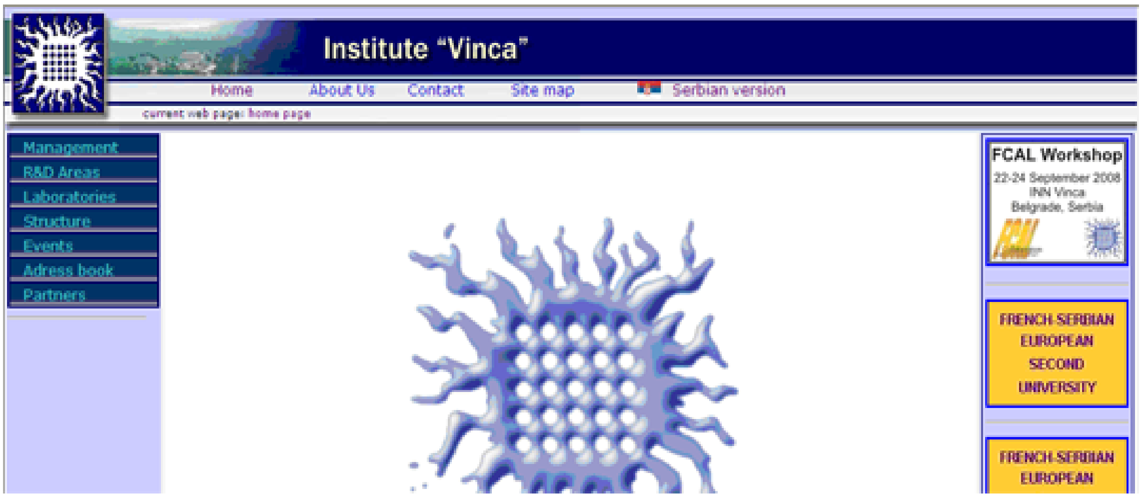

Fig. 6. Home page of VINCA Institute of Nuclear Sciences (www.vinca.rs).

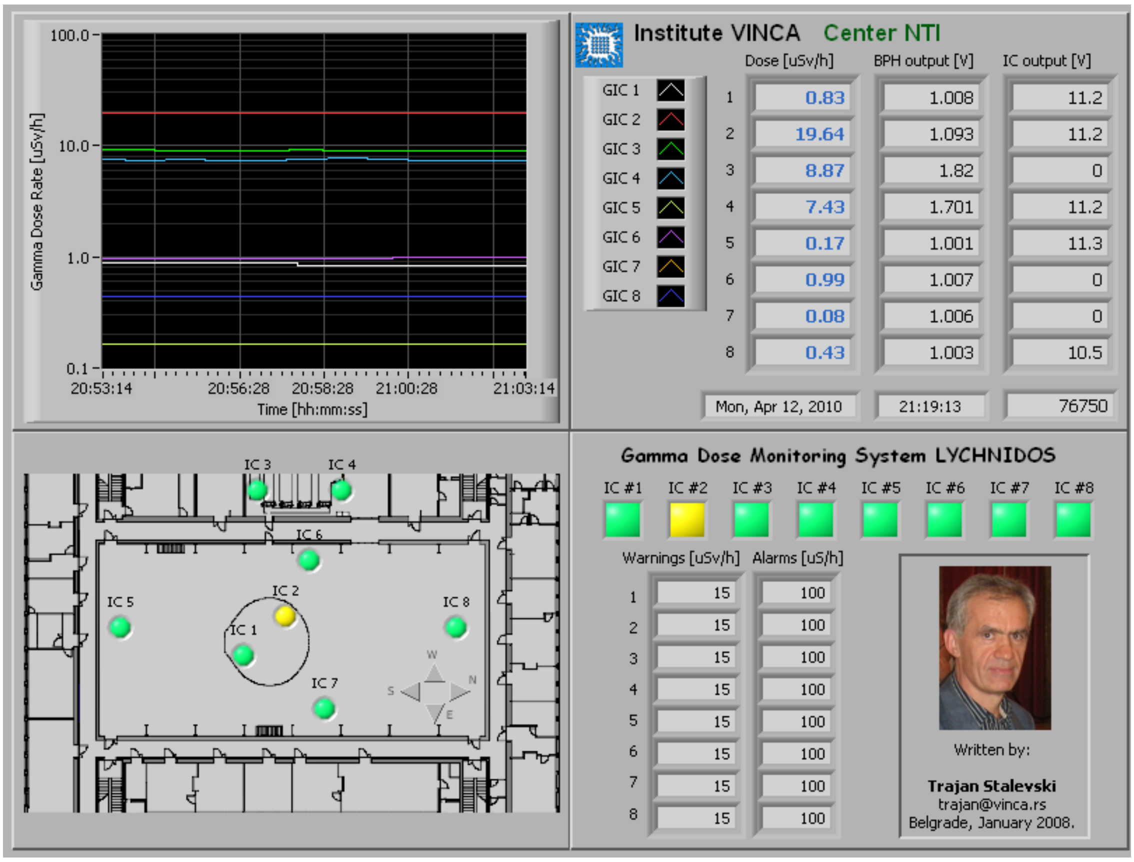

Fig. 7. Home page of VINCA Institute of Nuclear Sciences Reactor monitoring system (www.vinca.rs).

Discussion and conclusion

Source of information in Serbian radiation protection legislation can be summarized in five points: Resumes given at Ministries and National Assembly sites related to short information, explanations and short texts; Proposals of Commissions and Boards; Reports of the Boards; Home Page of National Assembly with new information and education resources. The necessity of information system in legislative actions is obvious and only question is how this system has to designed and implemented. Implementation of good practice means realization of four elements: Resumes of law proposals; Integration of all available sources; Implementation of mechanisms of data management and Continuous testing and improvement of all services. (Garson, 2004)

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30 years of the Croatian Radiation Protection Association

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Abstract
Croatian Radiation Protection Association (CRPA, www.hdzz.hr/) was founded in 1979 and in 1992 became a regular member of the International Radiation Protection Association (IRPA). The 30th anniversary of CRPA is an opportunity to review the work of the society and to point out the main goals for the future. Seven national symposia of CRPA with international participation were organized after 1991. All presentations were published as full papers in printed Proceedings. In 2001 CRPA organized the IRPA Regional Congress in Dubrovnik, Croatia. Members of CRPA take part in organization of international IRPA congresses and in the work of IRPA General Assembly. CRPA has been actively involved in the meetings of the European radiation protection societies since 2004 and organized the 6th meeting in Zagreb in 2009. On the celebration of the 30th birthday of CRPA the first award for best work in the field of radiation protection was given to the young scientist.

Introduction
The Croatian Radiation Protection Association (CRPA) was founded in 1979 in the frame of the Yugoslav Radiation Protection Association. Since 1991 CRPA has been an independent organization and in 1992 became a regular member of the International Radiation Protection Association (IRPA). CRPA adopted the IRPA Code of Ethics. CRPA has today a membership of more than 150 scientists and specialists from Croatia from various areas of radiation science.

CRPA is a public organization with the purpose of promoting and developing scientific, educational and cultural activities in the field of radiation protection and related fields of science. The main activities are organisation of scientific and popular lectures, organisation of national symposia, co-organisation of regional meetings, and participation at international meetings and conferences.

The main goals of all these activities are stimulation and support of basic and applied scientific research; stimulation and support of the application of scientific results in commercial and non-commercial activities; initiatives and execution of improvement of education in schools, colleges and universities in Croatia; popularization of science, its results and their applications; and affirmation of the scientific method and manner of thought.

National symposia

Seven national symposia of the Croatian Radiation Protection Association with international participation were organized after 1991: in 1992, 1994, 1996, 1998, all in Zagreb, in 2003 and 2005 in a small resort nearby, Stubičke Toplice, and in 2008 in the North-Adriatic sea-side resort Opatija. The proceedings of all symposia were printed in Croatian with extended abstracts in English [CRPA, 1992; 1994; 1996; 1998, 2003, 2005, 2008]. The next, 8^{th} Symposium of CRPA with international participation will be held in the Koralj Hotel, Krk, Island Krk, 13 – 15 April 2011.

The topics of interest for the the radiation protection community can be observed through the relative abundance of papers grouped in various research subjects (Fig. 1). All papers presented at the first four symposia (1992 – 1998) are grouped together and then compared with the papers of the 5th, 6th and 7th Symposia.

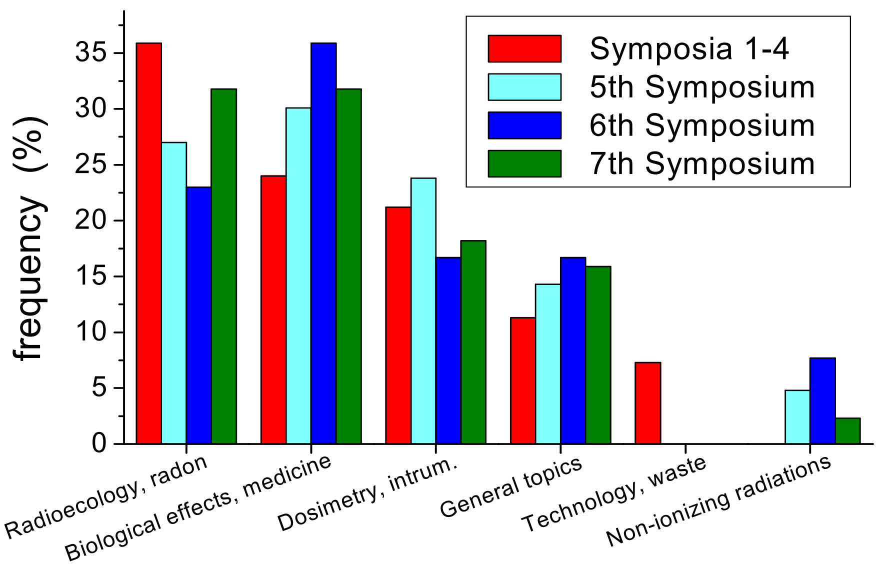

Fig. 1. Relative abundance of topics in CRPA proceedings.

Radioecology (including radon and related topics) and biological effects of radiation and medicine are the most abundant subject areas, each representing about 30% (with slight variations, ±5%) of all contributions. Dosimetry (including radiation physics and chemistry, methods and instrumentation) is represented by about 20% of the contributed papers, and general topics (including legislation, education, public awareness) by 15%.

At the first four symposia problems concerning radioactive waste technology and disposal were represented by 5% of the papers, but no papers on these topics are presented at the last three symposia. Instead, papers dealing with non-ionizing radiations represent about 5% of contributions. Non-ionizing radiations, their effects and medical/health implications became more abundant mostly due to huge expansion of cellular telephony and other applications of non-ionizing radiations (ultra sound, microwave, photo-therapy, etc), as well as due to our ever increasing awareness of their possible damage.

IRPA congresses

Members of CRPA regularly participate at regional, European and world IRPA congresses. Seven regional IRPA congresses for Central and Eastern Europe were held in the period 1992 – 2008 and the average number of Croatian participants (excluding the one in Dubrovnik 2001) is 10, as well as the number of contributed abstracts. The average number of Croatian participants at the four world IRPA congresses since 1996 (IRPA-9 through IRPA-12) and two European IRPA congresses is 15 with 16 presentations on the average.

CRPA organized the IRPA Regional Congress on Radiation Protection in Central Europe with the theme "Radiation Protection and Health" in Dubrovnik, Croatia, in 2001, with 180 presentations and 229 participants from 29 countries [CRPA, 2001]. Full papers were published as proceedings on CD-ROM [CRPA, 2002].

International cooperation

CRPA has been actively involved in the meetings of the European radiation protection societies since 2004 and organized the 6th meeting of this kind in Zagreb, October 26, 2009 with participation of 23 representatives of 13 European radiation protection societies. The topics included reports on current IRPA activities and IRPA-12 Congress (Buenos Aires, Argentina, 2008), status of the Radiation Protection Culture initiative, announcements of several training courses and workshops, current status of organization of 3rd European IRPA Congress and IRPA-13 Congress (Glasgow, 2012), as well as the status of national awards for young scientist and professionals in the field of radiation protection and radiation science.

30th Anniversary of CRPA

The main celebration of the 30th birthday of CRPA was held on October 27, 2010. The program included presentations of activities of the CRPA and a very interesting lecture on the early days of radiation protection in Croatia. All former president of the CRPA were awarded and Prof. dr. Božo Metzger has been elected as the first Honorary member of CRPA. The first award for the best paper of a young scientist published in 2008 – 2009 was officially given to Martina Rožmarić-Mačefat who is going to compete for the first European award for young scientists at the 3rd European IRPA Congress. Exhibition of scientific equipment and publications related to radiation protection was also organized.

Concluding remarks

In future CRPA will continue to organize regular national symposia and encourage its members to participate at IRPA congresses. It will increase the efforts in organization of lectures for both scientific community and general population. It will also keep its presence in various international activities of radiation protection societies with emphasis on improvement of cooperation with the societies in the neighbouring countries.

Acknowledgments

We are grateful to all members of CRPA who actively participate in the activities of the society.

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Public acceptance of radiocontamination in food products: what can we learn for a better decision-making?

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Abstract

This contribution investigates consumer's risk perception and attitude towards food products containing residual radioactivity. The data originate from a large-scale public opinion survey carried out in Belgium in the summer of 2009 on the basis of computer assisted personal interviews. It is assumed that the products under discussion satisfy the legal norms concerning the maximal allowable levels of radioactive substances in food.

Risk perception of residues of radioactivity in food products, consumer's attitude and a number of factors that could influence it are explored in our study: acceptance of food legal norms, attitude towards nuclear energy and psychometric factors such as disaster potential and tampering with nature. The results are interpreted in the context of practical implications for the decision making process in nuclear emergency management.

Introduction

After a nuclear event with subsequent contamination of the environment, one of the key issues to be dealt with is management of contaminated food production systems. The main challenges are mitigating the health effects to the population consuming such products and bringing social reassurance, while limiting economic loss and restoring normal life.

The European norms (also adopted in the Belgian legislation) for marketing food products after a future radiological accident are the Council Food Intervention Levels (CEC, 1989). These are subject to approval or modification by the experts of Article 31 of the EURATOM Treaty in the first 3 months after the accident. In practice, it is likely that the type and scale of the release, as well as various socio-economic factors, will bear a heavy influence on the decisions taken with regards to food safety. While public acceptance of the policy is important for the obvious social and political reasons, consumer's attitude towards potentially affected food products may have important economic consequences. A better insight in consumer's attitude can thus contribute to a better preparedness and a more efficient decision-making.

Residual radioactivity in food is not a matter of daily concern for the consumer. Some lessons can yet be drawn from past experiences with food contaminations of other nature, which show that consumption levels for products affected by a food chain crisis decreased in both claimed and reported consumer behaviour (Verbeke, 2001).

The first aim of this study is to investigate general risk perception of residual radioactivity in food. The second is to explore related consumer's attitude in a post-accidental framework. We thus tackle the following research questions:

- are risk perception or confidence in authorities for managing risks from residues of radioactivity in food different than for harmful substances in food in general?
- what is the consumer's attitude towards radioactive contamination of meat?
- what is the relationship between consumer's attitude and potentially explanatory variables, such as acceptance of food legal norms, attitude towards nuclear energy or psychometric risk characteristics.

Throughout the paper, it is assumed that the food products under discussion satisfy legal norms concerning maximal allowable levels of radioactive substances in food and therefore can be freely marketed.

The interpretation of results aims at highlighting practical implications for nuclear emergency management and for the associated decision-making process.

Material and methods

Starting from 2002, the Belgian Nuclear Research Centre SCK·CEN conducts periodically large-scale public opinion surveys among a representative sample of the Belgian population (Carlé and Hardeman, 2003; Van Aeken et al., 2007; Perko et al., 2010). Alongside with recurrent issues such as perception of various risks, confidence in risk regulators or the use of nuclear energy, the surveys include detailed research sections on topics such as emergency planning, food safety or communication.

The data in this study originate from the SCK•CEN survey carried out in 2009.

1. Background Variables
2. Risk perception and confidence in authorities for 17 risk items
3. Attitude towards:
    - 3.1. Science and technology
    - 3.2. Stakeholders engagement in decision process related to industrial installations with risks
    - 3.3 Nuclear energy
4. Acceptance of legal norms for food products
5. Use of different information media
6. Evaluation of different actors in the nuclear domain
    - 6.1. Knowledge, competence and confidence in various actors
    - 6.2. Confidence in authorities for management of nuclear installations
    - 6.3. Confidence in authorities for radioactive waste management
7. Risk perception of an accident in a nuclear installation
8. Safety behaviour and anomy
9. Knowledge about the nuclear domain
    - 9.1 Protective measures in nuclear emergencies
    - 9.2 General knowledge about nuclear technology and nuclear energy
    - 9.3 Experiences with the 'nuclear'

Newsflash: TV-report about topics related to preparedness for nuclear emergencies

10. Reception & acceptance of messages communicated in the TV-report
11. Reception & acceptance of the iodine predistribution / information campaign
12. Reception & acceptance of messages communicated by authorities during the Fleurus event in 2008
13. Consumer's attitude towards food with radioactive contamination

Fig. 1. Main topics of the SCK•CEN risk barometer survey in 2009.

The content of the entire survey is summarised in Fig.1, but more details can be found in Perko et al. (2010). In this paper we resume to presenting results relevant for consumers' risk perception and attitude towards food with radioactive contamination.

Sampling procedure

The field work in the SCK•CEN 2009 survey was performed by a market research bureau using CAPI - computer assisted personal interviews. These interviews were carried out at the home of the respondent, the answers being directly recoded and stored on a portable computer's hard disk. The bulk data of the 2009 survey consisted of 1031 interviews in a chosen language (French or Dutch). This sample is representative for the Belgian adult population with respect to province, region, level of urbanization, gender, age and professionally active status. Stratified sampling was applied by i) cross-tabulating the 11 Belgian provinces with four levels of urbanisation; ii) drawing a selected sample of communities from each resulting cell; and iii) random selection of dwellings in the 108 communities. Interviews were organised such that predefined quota were respected for gender, age (divided in three categories), professionally active status and social class.

In addition, the 2009 edition included 104 interviews organized in the Fleurus area (commune in Wallonia, the French speaking part of Belgium), where the authorities advised against eating fresh vegetables and fruit from the gardens after the iodine release event in August 2008. Results for this specific population group will be compared -where appropriate- with those obtained for the French-speaking part of the Belgian population and the region of Wallonia, respectively.

Scaling

Wherever latent concepts were measured by more than one item, in order to reduce measurement errors and/or to capture multiple aspects of that concept, exploratory factor analysis using principal axis factoring was performed to examine each of the scales. In addition, the reliability of the scales was assessed with Cronbach's alpha reliability coefficient. The high reliability estimates (> 0.7) and factor loadings (> 0.5) obtained for each of these scales suggest high construct validity. The content of the constructed scales will be discussed in later subsections.

Risk perception and confidence in authorities

Perception of risk from "*residues of radioactivity in food"* and confidence in authorities for managing this risk were measured by direct questions, alongside with other risk topics: "*How do you evaluate the risks for an ordinary citizen of Belgium for …"* and "*Please state how much confidence you have in the authorities for the actions they undertake to protect the population for …*". In the results section these shall be presented and discussed in comparison with risks from harmful substances in food in general.

Acceptance of food legal norms

The term "food legal norms" was used with the meaning of maximal amounts of toxic substances allowed in that food products. This was briefly explained by the interviewer through a short introductory text: "*The government plays an active role by limiting the* *quantity of toxic substances a particular foodstuff may contain. These upper limits are laid down by the government as legal norms. Such a norm may tell us for instance how much dioxin may chicken meat contain or how much preservatives may be present in cookies*".

Based on previous research (Van Aeken et al., 2006), a set of six items related to legal norms was used, reflecting the content, the enforcement and the legitimacy of the norms. These items were formulated as statements with respect to which the respondents expressed their level of agreement on a 5-point Likert scale, ranging from 1="strongly disagree" to 5="strongly agree". Kaiser-Meyer-Olkin measure of sample adequacy (0.81) and Bartlett's test of sphericity (χ^2=2585, df = 15, Sig.<0.001) suggest that conducting a factor analysis on these items is appropriate (see results in Table 1). Two norm items were inverted to enable the calculation of the reliability coefficient. Factor analysis revealed one factor, all items having loadings larger than the cut-off value 0.5.

Table 1. Factor loadings and scale reliability for the acceptance of food legal norms.

Acceptance of food legal norms (N=935)	Factor loading Principal axis factoring Rotation: Oblimin	Reliability Cronbach's alpha
Legal norms for food products offer sufficient protection to all citizens, including children and the elderly.	.818	
A food product that complies with the legal norms can be safely consumed.	.772	
Legal norms for food products are the result of sound reasoning by the government.	.745	0.85
There is sufficient control on food products.	.728	
(inv) The legal norms are not strict enough.	.575	
(inv) The government is inadequately organized to secure food safety	.530	

Attitude towards nuclear energy

Attitude towards nuclear energy was first assessed by three general statements to which the respondents had to state their agreement degree on a 5-point Likert scale (from 1="strongly disagree" to 5="strongly agree").

Table 2. Factor loadings and scale reliability for the attitude towards nuclear energy.

Attitude towards nuclear energy (N=892)	Factor loading Principal axis factoring Rotation: Oblimin	Reliability Cronbach's alpha
(inv) Opinion about nuclear energy (INVERTED scale)	.818	
Benefits of nuclear energy outweigh disadvantages	.772	0.84
Keeping NPP's open necessary for a secure energy supply	.745	
(inv) Reduction of NPP's is good cause	.728	

These items assessed the extent to which the respondents agreed that benefits/advantages of nuclear energy outweigh the disadvantages, that keeping NPP's open secures energy supply and that reduction of the number of NPP's in the E.U. is a good cause (first three items in Table 2). Subsequently, opinion about nuclear energy was measured through a direct question whether the respondent was in favour of nuclear energy or not (from "1=very much in favour" to 5="very much opposed").

Before conducting a factor analysis (see results in Table 2), two items were inverted such that high scores indicate always a strong support for nuclear energy. A scale was formed with these items using the resulting factor.

Consumer's attitude towards meat products with radioactive contamination below legal norms, after a nuclear accident

In 2006 an extensive study (Van Aeken et al, 2007; Turcanu et al., 2007; Turcanu et al., 2008) was dedicated to the management of contaminated milk. In the 2009 edition of the SCK•CEN barometer, the focus was on meat products. These generally represent an important part of the daily diet in Belgium: almost half (47%) of the respondents consider meat as important or very important daily food product, whereas 29% think it is moderately important.

In order to assess consumer's attitude towards meat products with residual radioactivity, a short introductory text was used to set the framework: "*Suppose that after a nuclear accident there is a contamination of meat with radioactive substances, and that the authorities ensure that radioactivity level is below the legal norms*". Consumer's attitude was then measured by the level of agreement with the statement: "*I would buy meat and consume it as usual in the first few months after the accident*", on a 5-point Likert scale ranging from 1="strongly disagree" to 5="strongly agree".

Characteristics of risk perception of meat products with radioactive contamination below legal norms, after a nuclear accident

Ten items selected and adapted from the extended psychometric model (Sjöberg, 2000) were used to asses the latent constructs behind risk perception of radioactive contamination in meat products after an accident in a nuclear installation. Respondents were asked to express their level of agreement with these statements, introduced with the following text "*Even for a contamination below legal norms, the presence of radioactivity in meat after an accident in a nuclear power plant…*"

Measurements were done on a 5-point scale ranging from 1="strongly disagree" to 5="strongly agree", high scores indicating strong adherence to a psychometric risk characteristic. Three characteristics of risk were measured: "Disaster potential" (four items), "Tampering with nature" (three items) and "New and unknown risk" (three items).

Kaiser-Meyer-Olkin measure of sample adequacy (0.92) and Bartlett's test of sphericity (χ^2=7298, df=45, Sig.<0.001) indicate the suitability of conducting a factor analysis. Three factors were extracted (see Table 3), based on the theory (Sjöberg, 2000), scree plots analysis, explained variance in the data and eigenvalues larger than or close to 1.

Table 3. Factor loadings and scale reliability for psychometric characteristics of risk perception of meat products with radioactive contamination below legal norms, after a nuclear accident.

Psychometric risk characteristic	Factor loading Principal axis factoring Rotation: Varimax	Reliability Cronbach's alpha
Disaster potential (N=917)		
Has large consequences for consumers	.782	
Will cause cancer for consumers	.802	0.95
Will harm children and future generations	.804	
Will have effects that cannot be reversed for consumers	.775	
Tampering with nature (N=942)		
Is the result of an activity which is contrary to nature	.738	
Is the result of humans trying to influence the basic processes and structures of nature	.874	0.89
Is a warning that much worse things might happen	.572	
New and unknown (N=944)		
Has effects that are hard to understand for consumers	.667	0.81
Has effects that are not sufficiently known by science	.652	
Would be something new and unknown to consumers	.690	

Results

Risk perception and confidence in authorities for residues of radioactivity in food

Outside of the context of a nuclear accident, "*residues of radioactivity in food*" are generally perceived as a low to moderate risk for an ordinary citizen of Belgium: 47% of the respondents perceive it as a low or very low risk and 27% as a moderate risk.

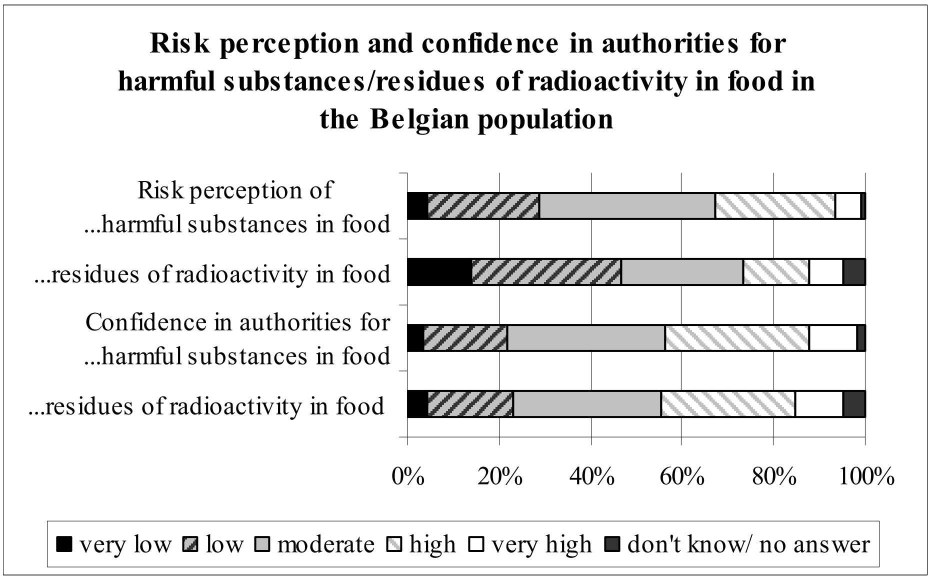

Fig. 2. Risk perception and confidence in authorities for harmful substances/ residues of radioactivity in food in the Belgian population.

Compared to this, perception of harmful substances in food (in general) is slightly higher (see Fig. 2), which suggests that there is no a priori fear of radioactivity in food.

Concerning confidence in authorities for managing risks from residues of radioactivity in food, 40% of the respondents expressed high or very high confidence, while 32% a moderate level of confidence. The difference compared to “harmful substances in food” is considerably smaller than for risk perception (see also Fig. 2).

The same analysis for the population sampled in the area of Fleurus (Fig. 3) reveals that risk perception is more clearly outspoken for “radioactivity in food” than for “harmful substances in food” in general: the percentage of people with a moderate risk perception is lower for risks from residual radioactivity in food. However no clear tendency towards a high or a low risk perception is observed.

Confidence in authorities is again similar for the two types of food risks.

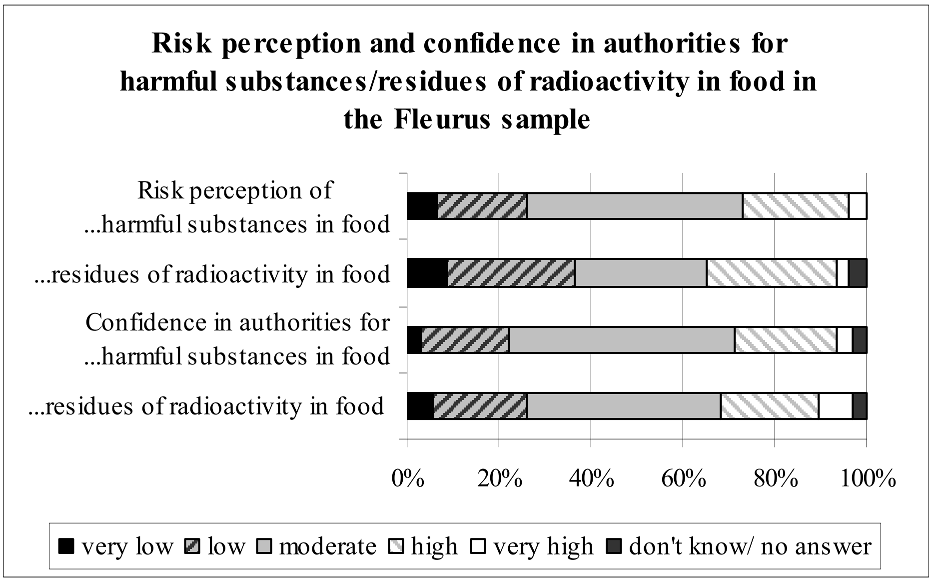

Fig. 3. Risk perception and confidence in authorities for harmful substances/ residues of radioactivity in food in the Fleurus area.

Looking at residues of radioactivity in food, we notice that only 36% of the respondents from the Fleurus sample have low or very low risk perception compared to 43.8% in the French speaking part of the population and 51% in Wallonia; 30% of the respondents from the Fleurus sample have high or very high risk perception against only 9.1% in the French speaking population and 18% in Wallonia. With respect to the confidence in authorities, this accounts for 26% of respondents with low or very low confidence and 29% with high or very high confidence in authorities in the Fleurus area, which is quite in line with results from the French speaking population and Wallonia region: 29% with low or very low confidence among the French speaking and 31% in Wallonia, against and 30% with high or very high confidence in authorities among the French speaking and 33% in Wallonia.

Consumer's attitude towards meat with radioactive contamination

Almost half of the respondents (48%) stated their disagreement with buying and consuming meat as usual, even if authorities ensure that contamination is below legal norms (see Fig. 4). About 17% of the respondents do not have a definite opinion, while 30% agree or strongly agree with buying and consuming such meat products as usual.

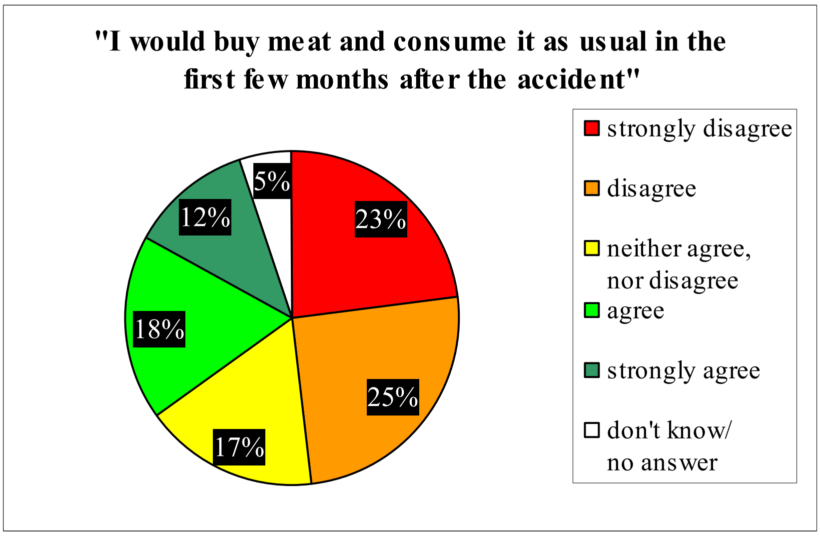

Fig. 4. Consumer's attitude towards meat with radioactive contamination below legal norms.

The study of potential associations between consumer's attitude and socio-demographic characteristics revealed significant associations with language (χ^2=50.10, p<0.001), education level (χ^2=51.24, p<0.001), region (χ^2=56.45, p<0.001), habitat (χ^2=47.96, p=0.001) and importance of meat in the daily diet (χ^2=137.6, p=0.001).We note that no significant associations with gender, social class or age were detected.

Respondents with primary education or with university degree strongly disagree with buying and consuming meat products with radioactive contamination below legal norms more often than people with other education levels: 40.5% with university education and 31.5% with primary education strongly disagree with buying and consuming meat with radioactive contamination, while for other education levels this is only between 20 and 25%. However, strong agreement decreases with the education level (from 20.2% for primary school to 4.1% among the holders of a university diploma) and increases when moving from urban to rural environment. In large cities only 18.1% respondents agree or strongly agree with consuming such meat products, while in other habitats this accounts to more than 35%. Brussels stands out as expressing the strongest disagreement (33.7% strongly disagree) with buying and consuming meat products as usual, followed by Wallonia (27.5%) and Flanders (21%).

Strong agreement with buying and consuming meat products as usual when contamination is below legal norms also grows with the importance of meat in the daily diet, e.g. 30.8% of those for whom meat is a very important food product strongly agree with buying and consuming meat as usual, while this is valid for only 5.2% of the respondents who state that meat is moderately important in the daily diet.

Potential predictors for consumer's attitude towards meat with radioactive contamination

Ordinal logistic regression was employed to study potential predictors for the consumer's attitude using the R package (library Design). The regression coefficients and the results of the Wald test for significance of the coefficients are presented in Table 4.

Table 4. Results of ordinal logistic regression for consumer's attitude (N=722)

Cut-off point/ Independent variable	Coefficient	Std. error	Wald Z stat.	P
y>=disagree	0.14	0.24		
y>=neither agree, nor disagree	-1.28	0.24		
y>=agree	-2.18	0.25		
y>=strongly agree	-3.62	0.27		
Importance of meat in daily diet	0.376	0.068	5.51	<0.001
Acceptance of food legal norms	0.289	0.083	3.48	<0.001
Attitude towards nuclear energy	0.202	0.084	2.41	0.016
Disaster potential	-0.756	0.082	-9.18	<0.001
Tampering with nature	-0.549	0.076	-7.19	<0.001
New	-0.329	0.085	-3.88	<0.001
Likelihood ratio χ^2=283; df=6; Sig.<0.001(statistically significant model); c-statistic C=0.74; Pseudo-R^2 = 0.35 (moderate effect size).				

Psychometric risk characteristics are negatively associated with the tendency to accept food products with contamination below legal norms, whereas e.g., importance of meat in the daily diet is positively associated with the latter. For every unit increase in the importance of meat in daily diet, we expect a 0.4 increase in the expected log odds to move to a higher category of consumer's attitude (more acceptance), i.e. an increase in odds by a factor exp(0.376)= 1.45[1].

Discussion

The study of general risk perception of residual radioactivity in food shows that this risk is not a matter of special concern in the Belgian population. In the Fleurus sample, perception of this risk is slightly higher than overall in the French speaking population or in Wallonia. However, this cannot be pinpointed as a direct effect of the radioactive iodine release incident in 2008, since no measurements of risk perception were performed prior to the incident. In what concerns confidence in authorities, similar results were obtained for risks from residues of radioactivity in food and harmful substances in food, both for the Belgian population and for the Fleurus sample.

In a post-accidental situation, even if food products complying with legal norms may be deemed as fit for consumption, this does not necessarily lead to consumer's willingness to buy food products with residual radioactive contamination, which confirms the findings of the 2006 survey (Turcanu et al, 2007). When asked if a food product complying with legal norms can be safely consumed, an overwhelming 68% of the respondents agree or strongly agree with this statement. However, only 30% agree

[1] The log odds scale is given by logit(p)=log(p/(1-p)), where p/(1-p)= odds of an event with probability p

or strongly agree with buying and consuming meat products with radioactive contamination below legal norms after a nuclear accident.

Significant effects on consumer's attitude towards meat products with contamination below legal norms in a post-accidental context were detected for psychometric risk characteristics, importance of meat in daily diet, acceptance of food legal norms and attitude towards nuclear energy. For instance, respondents with higher acceptance of food legal norms or less adherence to the psychometric risk characteristics studied are more likely to agree with consumption of such food products.

Conclusions

A good preparedness is the cornerstone of an efficient response to emergencies, one of the important issues to be dealt with being the management of contaminated food. Our study aimed at gaining a better insight in consumer's attitude towards food products with radioactive contamination. Results show that, although radioactivity in food is generally not perceived as a high risk, a large scale post-accidental contamination would bring about complex policy issues. Effective communication and information of the public in such a circumstance are certainly needed and could help alleviate social concerns.

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Lifetime health risk of paediatric exposures to ionizing radiation

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Radiation protection of embryo-foetus in diagnostic imaging

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Justification and optimization of paediatric CT

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Unjustified CT examination in young patients: a survey at Oulu University Hospital

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Introduction

Shortly after publication of the European Commission's directive 97/43/EURATOM, justification was considered to be the challenge of the decade (1). Ten years later, it has been speculated that the process of justification is sometimes weak or even nonexistent (2).

The radiation doses from computed tomography (CT) examinations are among the highest in diagnostic radiology, yet CT is being increasingly utilized. According to the referral criteria for imaging recommended by the European Commission, imaging methods without ionizing radiation, such as ultrasound (US) and magnetic resonance imaging (MRI), or methods with low-dose radiation should be considered whenever justified (3). Particular attention should be paid to young patients, since the radiation-induced lifetime risk of cancer mortality is higher at younger age until approximately 35 years of age (4).

The basic aim of our study was to determine whether previous CT examinations done at our university hospital on patients under the age of 35 years had been justified, and if not, whether other, more justifiable imaging modalities had been available, or if any modality was needed. Based on the study, a more comprehensive development project related to justification assessment was implemented.

Materials and Methods

Altogether 148,988 examinations were performed in the Department of Diagnostic Radiology of Oulu University Hospital, Oulu, Finland, in 2005. 16,975 of the examinations (11%) were done using CT, and 2,367 (14%) of the CT examinations were done on patients under the age of 35 years. The main groups of examinations were CT of the head, thorax or lungs, lumbar (and sacral) spine, abdomen or upper abdomen, trauma, cervical spine, nasal sinuses and body (thorax and abdomen) (Table 1).

The examinations analysed in this study were CTs of the head (50 patients), lumbar (and sacral) spine (30), abdomen or upper abdomen (30), trauma (30), cervical spine (30) and nasal sinuses (30) (Table 1). The final study thus included 200 examinations. Images falling in these categories were extracted from the electronic patient files of our hospital consecutively from the beginning of the year 2005. CTs of the thorax or lungs and body were excluded from the study because there is no good alternative for these examinations.

Patient files, clinicians' referrals and indications of the examinations were analysed by a specialist in radiology with good experience. Using this information and the referral criteria for imaging recommended by the European Commission (3), it was decided whether the examinations had been justified, and if not, whether some other, more justifiable imaging modalities would have been available. After that, other specialists in radiology went through the information collected and expressed their opinion; if necessary, consensus was used.

Table 1. CT examinations performed on patients under the age of 35 years in 2005 and the number of cases analysed in the study.

CT examinations	Number of CT examinations n	CT examinations analysed n
Head	1,063	50
Thorax or lungs	241	
Lumbar and sacral spine	130	30
Abdomen or upper abdomen	123	30
Trauma	117	30
Cervical spine	110	30
Nasal sinuses	100	30
Body	80	
Other	403	
Total	**2,367**	**200**

Results

About 30 per cent of all the 200 examinations evaluated were non-justified (Table 2). Twenty-three of the 30 CT examinations of the lumbar spine (77%) were considered not justified. Twenty cases could have been replaced by MRI, and three patients would not have needed any radiological examination (Table 2). Symptoms of disk syndrome, suspicion of spinal stenosis and control of spinal lymphoma in young patients may indicate MRI. Trauma and control of fixation indicate CT.

Eighteen of the 50 CT examinations of the head (36%) were not justified. All of them could have been replaced by MRI. MRI should have been performed in elective cases. CT is indicated in trauma or some other acute cases, such as suspicion of intracranial bleeding or acute stroke.

CT was not justified in 11 of the 30 CT examinations of the abdomen or upper abdomen (37%). Five of the cases could have been replaced by MRI, four by US and one by fluoroscopy. One patient would not have needed any radiological examination. Two patients had unspecific hepatic lesions at US, which should have indicated MRI instead of CT. Other patients in this group were so variable that no classification could be done; the analysis had to be done on a case-by-case basis.

Six of the 30 CT examinations of the nasal sinuses (20%) were not justified. Five of them could have been replaced by MRI, while one would not have needed any other examination but CT of the head. CT was considered to be justified especially if operation of the sinuses was being planned, since there is a need for accurate delineation of the bony structures for functional endoscopic sinus surgery (FESS). However, five of the unjustified cases also had rhinitis or sinusitis, but there was no information about operation plans in the referral.

Only one of the 30 CT examinations of the cervical spine was not justified. The patient would not have needed any CT examination of the cervical spine in addition to one of the lumbar spine. Other cases were traumas and a control of fixation, which indicated CT. All the 30 CT examinations of trauma were justified because the traumas were high-energy ones.

In our study 21 out of the 200 patients were children (15 or under 15 years old) (11%). There were three unjustified cases in this group (3/21, 14%).

Table 2. The number of unjustified CT examinations out of total number of cases analysed and the possibility of other modalities to replace CT.

CT examination	Unjustified / All n (%)	Possibility of other modalities to replace CT			
		MRI n	US n	Fluoroscopy n	No examination needed n
Lumbar spine	23 / 30 (77%)	20			3
Abdomen	11 / 30 (37%)	5	4	1	1
Head	18 / 50 (36%)	18			
Nasal sinuses	6 / 30 (20%)	5			1
Cervical spine	1 / 30				1
Trauma	0 / 30				
Total	**59 / 200 (30%)**	**48**	**4**	**1**	**6**

New interventions and follow-up

As a consequence of our study, we wanted to change our practice and introduced various interventions and organized follow-up of the project. The interventions were mostly introduced in 2006-2007.

We provided education for the staff of the Department of Radiology, other personnel working with ionizing radiation in our area and the referring practitioners in our hospital. The education consisted of the risks and doses of radiation, indications of different examinations, the process of justification and legislation on radiation protection. The radiology staff in other hospitals in Northern Finland could also be reached through videoconferences. We also provided info cards containing information on radiation and justification for the referring practitioners in the Oulu area and for the people working with radiation in our hospital, and the cards will be handed out to medical students every year.

The referral criteria for imaging recommended by the European Commission were distributed into the different areas of the Department of Radiology. We also made some new recommendations for the use of CT for the referring practitioners and the radiologists at our hospital. Their content is roughly the following: 1. MRI is the primary examination of the head. CT examination is only indicated in acute cases. 2. MRI is usually the primary examination of the lumbar spine in young patients. 3. Clinicians are recommended to consult a radiologist before sending a request form for abdominal CT in the case of a young patient.

Our study revealed that most of the unjustified cases could have been replaced by MRI. Shortage of MRI capacity may partly have contributed to the poor results of justification. We addressed this by purchasing a new MR system. We have also reported about the project at conferences and in medical journals both in Finland and abroad (5).

We have followed up justification of the CT of the lumbar spine in our department. We wanted to monitor that area in particular, as it showed the poorest justification in our study. The total number of CTs of the lumbar spine has decreased and the justification has improved in patients under the age of 35 years during the years 2007-2009 when compared to the levels in 2005 (Table 3). We have also followed up the ratio of examinations with and without ionizing radiation in our department. Since 2006, the ratio also improved towards examinations without radiation until the year 2008. There was a slight opposite change again in 2009 (Table 4).

Table 3. CT examinations of the lumbar spine performed on patients under the age of 35 years in 2005 and in 2007-2009 and the number of unjustified lumbar CTs out of total number of cases analysed during the same years.

	2005	2007	2008	2009
CT of the lumbar spine (n)	130	37	38	24
Unjustified lumbar CT (n) / All (n) (%)	23/30 (77%)	2/4*/** 11/19 (58%)*/**	2/4*/** 5/20 (25%)*	1/1*/** 2/20 (10%)*

* in different units of the department; time period for the control audit varied between 9-11 months in different years
** less than 20 patients found for the control

Table 4. The ratio of examinations with and without ionizing radiation in the follow-up.

	2005 %	2006 %	2007 %	2008 %	2009
Ratio	77:23	76:24	74:26	74:26	75:25

Discussion

It is estimated that about 50% of the global collective dose of radiation is caused by CT with its relatively high doses (3). There were about 3.9 million medical x-ray examinations performed in Finland in 2005. About 7% were CT scans, and there were 30% more CT examinations in Finland in 2005 compared to 2000 (6). At Oulu

University Hospital, there were 19% more CT examinations in 2005 compared to 2000. It has been assumed that although the risk of radiological examination to a single individual is small, the exposed global population is large and increasing, which may result in significant long-term public health problems (7). It is therefore important to have good indications for CT or to utilize US or MRI or examinations with lower doses whenever possible.

The utilization of radiology is accepted as part of medicine, especially after careful justification. Despite the rules and recommendations defined in the legislation on medical radiation, suspicions of inappropriate use of radiological examinations and less selective use of diagnostic CT have been reported. Some paediatric radiologists have estimated that about one-third of CT examinations are unnecessary (7,8). With the help of the retrospective analysis we wanted to find out whether the number of CT examinations done on young patients could have been reduced with better justification. For the analysis, we chose CT examinations that could be replaced by other investigations, even ones not involving any radiation.

Most of the unjustified examinations, 77%, appeared to fall into the group of lumbar CT. The dose of radiation from lumbar CT is about 300 times the level of thorax PA x-ray. Most of these unjustified cases could have been replaced by MRI. Thirty-seven per cent of the cases in the group of abdominal CT were unjustified. The dose of radiation from abdominal CT examination is about 500 times that of a single thorax PA x-ray (3). Five of the unjustified cases could have been replaced by MRI, four by US and one by fluoroscopy. Thirty-six per cent of the cranial CT studies were deemed unjustified. All these 18 examinations should have been replaced by MRI. The dose of radiation from CT of the head is also about 115 times that of a thorax PA x-ray (3). There were fewer unjustified cases in the group of CT examinations of the nasal sinuses or the cervical spine, and all cases in the trauma group were justified.

To our knowledge, there are only few other studies about the justification of examinations causing radiation. In 2001, Clarke et al. reported about the possibilities of MRI to replace CT examinations. This team had more patients and subgroups than we did, and more than 70% of the CT examinations could have been replaced by MRI (9). In another report concerning CT examinations of the abdomen, pelvis and lumbar spine, the last of these was often recommended to be replaced by MRI (10). One study reports about 60% justification of CT examinations according to the request forms; in particular, US could have been useful as a preceding or alternative investigation (11). According to the Swedish national survey on justification of CT examinations approximately 20% of all examinations were not justified. The degree of justification varied strongly with the organ examined, moderately with prescriber affiliation and weakly with geographical region (12).

In our study, 21 out of the 200 patients were children. There were three unjustified cases in this group (14%). Ionizing radiation always increases the statistical risk of cancer mortality. The risk is higher at younger age because the expected lifetime is longer than at older age. Division of the cells is also fast and the organs are particularly sensitive to radiation at a younger age (4,13). There are only few published audits of justification of radiological examinations in children. The Swedish national survey on justification of CT examinations reported that the degree of justification is lower for younger patients. However, the total number of paediatric examinations was small in that study (12). There is also an audit of CT for the evaluation of mild to moderate paediatric trauma. Paediatric patients had significantly more CT scans than adults (14).

As a consequence of our study, we wanted to change our practice by introducing new interventions. It is known that awareness of radiation is often deficient and that radiation risks are frequently underestimated (15). We provided education and an info card containing information on radiation and justification for the staff of the Department of Radiology, other personnel working with ionizing radiation in our area and the referring practitioners.

The referral criteria for imaging recommended by the European Commission were distributed to different areas of the Department of Radiology. We also made some new recommendations for the use of CT for the referring practitioners and the radiologists at our hospital. Regular use of referral guidelines can also lead to a reduction in the number of request forms and ultimately to a reduction in patient exposure to ionizing radiation (3). Our study revealed that most of the unjustified cases could have been replaced by MRI. Because shortage of MRI capacity may partly have contributed to the poor results, we addressed this by purchasing a new MR system. In order to give other centres an opportunity to make use of our conclusions, we have reported about the project at different conferences and in medical journals.

The follow-up has so far mainly concentrated on the justification of CT of the lumbar spine and the results are striking. In the near future, we plan to follow up indications for CT examinations. We expect that better awareness of justification also in other areas of imaging will be reached by both the personnel working in the area of radiology and the referring practitioners. We also hope that the project will have an impact on other hospitals and health care centres in Northern Finland, other units in Finland, and other countries, too.

In conclusion, justification of CT examinations in young patients seemed to be inadequate. However, justification could be improved by interventions - education, use of referral guidelines and increased MRI capacity. The main goal of the whole project has been radiation protection of the patients, and the results have been promising.

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Screening or selective imaging in paediatric dentistry: from panoramic to CBCT

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Abstract
Dental radiography is one of the most frequent forms of medical X-ray imaging. A large proportion of examinations are performed in the paediatric age group, principally for the detection of dental decay (caries) and developmental anomalies. While most dental X-ray examinations have a very low radiation dose, the introduction of Cone Beam CT (CBCT) brings a higher dose range to dental imaging. Clinical screening of children is good practice, but radiographic screening cannot be justified. Referral (selection) criteria for imaging have been developed by various authorities and organisations, but these have a variable basis upon scientific evidence and uncertain acceptance amongst dentists. Provisional evidence-based referral criteria for CBCT are available through the European SEDENTEXCT project. Considerable research is needed to refine the existing referral criteria for dental radiology and to monitor their adoption in primary dental care.

Introduction
In developed countries, it has been estimated that 20% of X-ray examinations are performed by dentists (UNSCEAR, 2000). This figure conceals considerable national variation, ranging from the highest (839 per 1000 population) in Japan to below 100 per 1000 for many countries. In the United Kingdom it was estimated in 2002 that over 9 million intraoral and over 3 million panoramic radiographs were taken annually (Hart and Wall, 2002). Although some dental practice is conducted in hospitals and community (public health service) clinics, most takes place in dental offices. It is notable, therefore, that the usual practice of dental radiography is one of self-referral, in which the dentist decides radiography is necessary, selects a particular type of X-ray examination, performs the justification process and, often, also performs the examination. In most instances, an external influence on prescription of radiographic examinations is financial, through national or private health insurance schemes.

Another important aspect of dental practice is that a large proportion of radiographic examinations are performed on younger age groups (UNSCEAR, 2000), including children, reflecting two clinical needs. First, dental caries (decay) is a particular risk soon after the eruption of teeth, both for the deciduous and permanent dentitions. Secondly, dental development in childhood is frequently complicated by eruption problems (impaction of teeth), abnormal number of teeth (hypodontia or hyperdontia), discrepancy in size between the teeth and the available space in the mouth (crowding or spacing) and aesthetic problems. These developmental disturbances may necessitate orthodontic treatment. The prevalence of these clinical problems is such that "screening" or routine X-ray examinations have been adopted by some dentists.

Traditional dental radiographic techniques are limited in scope, being intraoral, panoramic and cephalometric techniques. Relative to most medical X-ray examinations, each of these is associated with a low radiation dose (European Commission, 2004; Ludlow et al 2008). Recently, however, Cone Beam CT (CBCT) has become available to dentists, at an equipment cost that makes it affordable. CBCT is associated with a higher radiation dose than is seen with traditional radiographic techniques, raising important challenges in justification and optimisation (SEDENTEXCT, 2009).

Radiographic screening in paediatric dentistry

Screening can be defined as "the testing of a symptomless population in order to detect cases of a disease at an early stage" (Bandolier, 2004). The basic principles of screening are:

- The condition is common and disabling, the natural history is known and that there is a recognisable latent or pre-symptomatic phase.
- The screening test is reliable, valid and repeatable, is acceptable and easy to perform, is sensitive and specific and low cost.
- Treatment should be effective and available, and that there is an agreed policy on who to treat.

How do these criteria fit with paediatric dentistry? As detailed above, there are two broad categories of "common or disabling" conditions for which a dentist (or public health dental service) might choose to screen: dental caries and its sequelae and anomalies of growth and development of the dentition requiring orthodontic management

Dental caries is a common disease in some populations, being associated with low socio-economic status, inadequate fluoride exposure, poor oral hygiene, poor diet and numerous other less frequent risk factors (Faculty of General Dental Practitioners UK, 1998; Pendlebury et al, 2004). In low caries risk populations, however, caries has become a rare condition in children. Nonetheless, the natural history of dental caries is well understood and the disease has a long pre-symptomatic phase and therefore might be seen to comply with the first criterion of a screening programme. Similarly, treatments and care pathways for dental caries are readily available and have well established efficacies, whether these relate to remineralisation of early, non-cavitated, lesions or restoration of carious cavities in teeth.

A significant proportion of children in developed countries may seek orthodontic treatment (Pendlebury et al, 2004; Isaacson et al, 2008). Because of this, clinical screening of children at around 8-10 years has become accepted practice. The addition of radiographic examinations to the clinical screening procedure has, however, become common. The aim of the dentist or orthodontist is to ensure the development of a functional and aesthetic dentition. It is important to remember that an impacted or missing tooth does not necessarily prevent this. The subjectivity of dental aesthetics means that some patients may find entirely acceptable anomalies for which others may demand treatment. On the other hand, the natural history of anomalous dental development and the pre-symptomatic phase are compatible with the first screening criterion. Treatments are effective and available and criteria for treatment have been developed based on clinical findings, notably the DAI (Dental Aesthetic Index; Cons et al, 1986) and IOTN (Index of Orthodontic Treatment Need; Brook & Shaw, 1989). Such indices typically identify that up to about half of children are in need of orthodontic treatment (Chestnutt et al, 2006; Manzanera et al, 2010).

From the above, therefore, it can be seen that a reasonable person might argue that radiographic screening of children is justifiable because the conditions and diseases of greatest prevalence in this age group are of significance and amenable to effective treatment. The question that remains relates to the screening methods that are used. In dental practice, the primary method of diagnosis is always clinical, after which special tests can be selected. The most widely used special test in dentistry remains imaging using X-rays.

Diagnostic tests for dental caries

While clinical examination of clean dry teeth, with good lighting, can reveal dental caries lesions, several high quality reviews indicate that posterior bitewing radiographs are an essential adjunct (Hanlon PM, 1985; Kidd & Pitts, 1990; Henderson & Crawford 1995; Weerheijm 1997; Pitts 1996). Bitewing radiographs (Fig. 1) reveal significantly greater proportions of caries lesions than clinical examination alone in both high and moderate caries risk populations (Faculty of General Dental Practitioners UK, 1998; Pendlebury et al, 2004). The diagnostic accuracy of bitewing radiographs for caries diagnosis has been subject to a systematic review by Bader et al (2001). They reported mean sensitivities ranging from 30-66% (depending on lesion size and site) and specificities of 76-95%. For low caries risk populations the evidence for screening using bitewing radiographs is less convincing (Hintze & Wenzel, 1994).

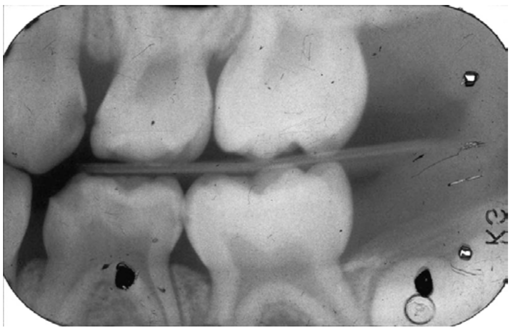

Fig. 1. A bitewing radiograph from a 6 year-old patient in the deciduous dentition.

While there is a well understood body of evidence surrounding the use of bitewing radiographs for dental caries diagnosis, there is less evidence for the role of panoramic radiographs in this respect. The reason for the limited volume of research in this field is, in part, because of the inferior imaging characteristics of panoramic radiography for detection of the small demineralisations of tooth enamel that represent caries lesions. As such, there is not a great pressure to perform research comparing an inferior (panoramic) with a superior (bitewing) technique. Panoramic radiographs have lower resolution and often show overlap between teeth that obscure lesions. Thus, caries diagnosis accuracy is lower than for bitewing radiographs (Scarfe et al, 1994; Rushton & Horner, 1996; Akkaya et al, 2006). Overlying this, image quality is often compromised by poor technique in primary dental care facilities (Rushton et al, 1999). Nonetheless, while emphasising the need for individualised prescription of radiographs, US guidelines still recommend "posterior bitewings with panoramic exam" as appropriate at all paediatric ages subsequent to the eruption of the first permanent molar at around 6 years of age (American Academy on Pediatric Dentistry, 2008).

The advent of CBCT to dentistry raises the potential of its use for the full range of diagnostic applications, including caries diagnosis. The current evidence suggests that limited (small field-of-view) CBCT has a similar diagnostic accuracy to conventional radiography for the detection of caries in posterior teeth *in vitro*, but that the representation of caries depth may be superior (Akdeniz et al, 2006; Haiter-Neto et al 2008; Tsuchida et al 2007; Qu et al, 2010). It should be emphasised, however, that the CBCT systems available to dentists vary enormously in their imaging characteristics, including resolution and field-of-view. In clinical practice, an important additional problem of using CBCT is the presence of artefacts from high atomic number dental restorations that will substantially reduce specificity. Because of these findings, the only currently available evidence-based guidelines on dental CBCT state that "CBCT should not be used as a routine method of caries detection and diagnosis" (SEDENTEXCT, 2009).

Diagnostic tests for anomalies of dental development

The usual radiographic examination performed for the assessment of the developing dentition is the panoramic radiograph (Fig.2). It is an effective method of identifying most dental developmental anomalies, as demonstrated by several surveys of unselected populations (Bergstrom, 1977; Locht, 1980; Ignelzi at al, 1989). The panoramic radiograph may be supplemented by selected intraoral radiographs and, where orthodontic treatment is planned, by a lateral cephalogram. Beyond initial diagnosis, patients who go on to orthodontic treatment may receive repeated X-ray exposures (Hujoel et al, 2006).

The evidence that panoramic radiography can identify dental anomalies should not be seen as a justification for imaging. Many anomalies are of purely documentary interest and the important aspect is whether the radiological findings influence management. Research suggests that in a large proportion of cases the radiographic evidence makes no contribution to diagnostic thinking (Atchinson et al 1991; Han et al 1991). While understandable emphasis has been placed here on the use of screening panoramic radiographs, the role of cephalometric radiographs in management of orthodontic patients has also recently been questioned (Nijkamp et al, 2008). Research performed 20 years ago provided compelling evidence that panoramic radiographic screening for the purpose of assessing malocclusion and timing of treatment was ineffective. In their key paper, Hintze et al (1990) demonstrated that a 2-3 minute clinical examination of an unselected 11-12 year old population, followed by selected panoramic radiographic examination, could identify all children except one in need of immediate orthodontic treatment (a nosological sensitivity of 97%) while excluding 94% of "healthy" children. In a further study in a larger population, Hintze & Wenzel

(1990) reported that screening panoramic radiography resulted in 67% of children being exposed for no benefit.

Fig. 2. A panoramic radiograph from a 9 year-old patient in the mixed (deciduous and permanent) dentition.

It is clear that, for some years, the research evidence has been against the concept of screening panoramic radiography of children for development of the dentition and orthodontic assessment, but in favour of *clinical* screening supplemented by radiographs when indicated. Nonetheless, the practice of radiographic screening and "routine" radiography persists amongst some dentists, particularly in the United States, possibly driven by fears of litigation if "something is missed".

Orthodontics involves the bodily movement of teeth through bone, using appliances that exert a steady force upon the former. This is therefore "three-dimensional" treatment. As such, the arrival of CBCT into dental practice has been greeted enthusiastically by orthodontists as offering an opportunity to have accurate three-dimensional knowledge of tooth position before, during and after orthodontic treatment. For assessment of facial bone shape, position and inter-relationships, there must be a high accuracy of measurements made with CBCT. Several studies have addressed this, reviewed by the SEDENTEXCT Guideline Development Panel (SEDENTEXCT, 2009), usually using direct calliper measurement of dried skulls as a reference standard. Differences between CBCT-derived measurements and the reference standard appear to be small and are unlikely to be clinically significant. Studies are not, however, available for all CBCT machines on the market. In considering the use of CBCT in orthodontics, a clear distinction must be made between localised CBCT (using a restricted field-of-view for a specific purpose) and large volume CBCT in which the facial bones are viewed in their entirety.

For localisation of unerupted teeth, CBCT can be reasonably assumed to have high accuracy. Teeth are relatively large objects, having good contrast with the surrounding bone. It is obvious that a three-dimensional imaging technique with acceptable measurement accuracy and little distortion will identify position of teeth with high diagnostic accuracy. As such, it is not surprising that no formal study has been performed that compares diagnosis of unerupted tooth position using CBCT and conventional radiographs. As conventional ("medical") CT is used in some centres for localisation of unerupted maxillary canine teeth (Alqerban et al, 2009), the dose advantage seen with CBCT means that CBCT may be preferred (SEDENTEXCT, 2009). Fig. 3 shows an image from a localised CBCT examination. A similar argument can be made in favour of CBCT in the assessment of cleft palate, for which conventional CT is the usual examination.

Fig. 3. Images from a localised CBCT examination for the localisation of an unerupted permanent canine tooth in the maxilla (upper jaw).

In contrast to this positive assessment of CBCT, the use of large volume (craniofacial) CBCT as a routine means of assessing growth and development of the dentition and the face in orthodontics causes concern. Compared with a "conventional" examination, using a panoramic and a lateral cephalometric examination, large volume CBCT gives a higher radiation dose, sometimes considerably so. The SEDENTEXCT Guideline document, based on systematic review, described the literature on this subject as "strong on hyperbole and short on evidence of significant clinical impact" (SEDENTEXCT, 2009), with case reports and non-systematic reviews claiming improvements in diagnosis and treatment in orthodontics for CBCT without any objective justification (e.g. Kau et al, 2005; Cattaneo & Melsen, 2008) . The concerns over the spreading use of CBCT as a treatment standard in orthodontics is compounded by anecdotal evidence of large volume CBCT datasets being used solely for the purposes of reformatting into panoramic and cephalometric images, a practice condemned in a recent Health Protection Agency Guideline document in the United Kingdom (Holroyd & Gulson, 2009).

Referral criteria instead of screening

Probably the first set of guidelines dealing with justification of radiology in dentistry, including paediatric uses, were those produced in the United States in 1987 (Dental Radiographic Patient Selection Criteria Panel & Joseph, 1987). These "expert panel" guidelines emphasised that individualized radiographic examinations should be prescribed based upon the patient's signs, symptoms, and history using selection [referral] criteria that increase the likelihood that the patient will benefit from the radiographic examination.

In Europe, most recommendations relating to Justification seem to be general guidance that imaging should be selected on an individual patient basis after a history and examination have been performed, relying on individual dentists' judgements rather than a set of specific referral criteria (European Commission, 2004; Martínez-Beneyto et al, 2007; Peltola et al, 2009). More detailed guidelines have appeared sporadically from European and national dental organisations, notably in the UK (Faculty of General Dental Practitioners UK, 1998) and France (Haute Autorité de Santé, 2006). Apart from national initiatives, guidelines on imaging have also appeared from national (American Academy on Pediatric Dentistry, 2008; Isaacson et al, 2008) and international (Espelid et al, 2003) specialist organisations. On close examination, however, many of these sets of referral criteria are expert panel based, rather than developed using a structured method of literature review. This no doubt explains why different sets of referral criteria may disagree.

In the UK, the Faculty of General Dental Practice (UK) produced evidence-based referral criteria for dental radiography in 1998 (Faculty of General Dental Practitioners UK, 1998.), revised them in 2004 (Pendlebury et al, 2004) and is now working on its 3rd edition. Unlike the US and previous European guidelines, these were very clearly "evidence-based", following a specific methodology which, at best, included systematic review. These guidelines acted as the model for the development of the European Guidelines on Radiation Protection in Dental Radiology (European Commission, 2004). Unfortunately, such is the rapidity with which CBCT has developed and proliferated in dentistry that those guidelines did not address it in any way. This provided the reason for the initiation of the SEDENTEXCT project under the European Atomic Energy Community's Seventh Framework Programme, which has recently developed evidence-based guidelines on CBCT use in dentistry, including referral criteria (SEDENTEXCT, 2009).

Conclusions

The nature of clinical dentistry, based largely in independent practice, increases the chances of inappropriate use of X-ray-based imaging. Conventional radiography has a low dose when appropriately optimised, although the increasingly available CBCT has significantly higher radiation doses. While clinical screening of children for the most common dental disease (caries) and developmental conditions requiring orthodontic intervention is appropriate, radiographic screening of children is unjustifiable on the basis of the research evidence. Instead, referral (selection) criteria have been developed by various organisations to encourage rational prescription of imaging that promotes improved treatment outcomes. Many of these guidelines have a low evidence base, however, and considerable research is needed to refine the existing guidance and to monitor their adoption in primary dental care.

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The IAEA OSART programme and radiation safety related findings during recent OSART missions

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Abstract

The IAEA operational safety review team OSART for nuclear power plants in Member States have been reformed responding to the changes of the environment of both the industry and Member States needs. The IAEA safety standards for Operations are being applied when conducting the Agency's safety services. Our goal is to ensure that the issues and trends resulting from industry operating experience and the Agency safety services can be effectively communicated to Member States and be used to further reform our safety standards and services. Radiation Protection is an important area of OSART review. The relevant Safety Guide serving as a basis for this area is "Radiation Protection and Radioactive Waste Management in the Operation of Nuclear Power Plants", IAEA Safety Standards Series No. NS-G-2.7. The objective of this presentation is, by referring to the above Safety Guide, to discuss the various findings, recommendations, suggestions and good practices from recent OSART missions in different countries that should to be considered in order to improve Radiation Protection in the Operation of NPPs. The following RP areas are assessed during OSART mission: Organization and function, Radiation work control, Control of occupational exposure, Radiation protection instrumentation, protective clothing and facilities, Radioactive waste management and discharges and Radiation protection support during emergencies.

Radiation safety in new build

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Abstract

STUK reviewed the utility Teollisuuden Voima Oyj's (TVO) application for the Construction Licence of the Olkiluoto 3 nuclear power plant unit in 2004-2005. Based on this review STUK prepared its statement on safety together with a safety assessment report of the new plant to the Government. STUK has continued reviewing the detailed design during the construction of the new plant unit. By virtue of the Nuclear Energy Act (990/87) and the Government Decree on the Safety of Nuclear Power Plants (733/2008), Radiation and Nuclear Safety Authority (STUK) issues detailed regulations, YVL Guides, concerning the safety of nuclear power plants. Several YVL Guides deal with radiation safety (site, abatement of releases, worker radiation protection, emergency arrangements, etc). The paper will discuss some radiation safety related requirements in the design of a new Finnish NPP and their implementation in the licensing documentation.

Introduction

The licensing process of a new nuclear power plant in Finland is shown in Figure 1. The project of the fifth Finnish nuclear power reactor was formally started in May 1998 with Environmental Impact Assessment (EIA) process. Then the utility TVO submitted the application for a Decision in Principle in November 2000. The Finnish Government made the decision in January 2002, which Parliament ratified in May 2002.

Late 2003, TVO proposed the plant site to be Olkiluoto and made a contract with a consortium of AREVA (former Framatome ANP) and Siemens AG to build an EPR (European Pressurised Water Reactor). TVO submitted the application for Construction Licence in the beginning of 2004 and the licence was granted by the Government in February 2005. The commercial operation of the new plant unit is expected to take place in 2012 instead of spring 2009 as originally planned. Longer construction schedule is due to the delays in detailed civil work and system design and problems met in the construction and manufacturing of main components. The Operating Licence evaluation process takes approximately one year.

There are also several new nuclear power plant projects ongoing in Finland. Finnish Government and the Parliament are making decisions during 2010 on three applications for a Decision in Principle.

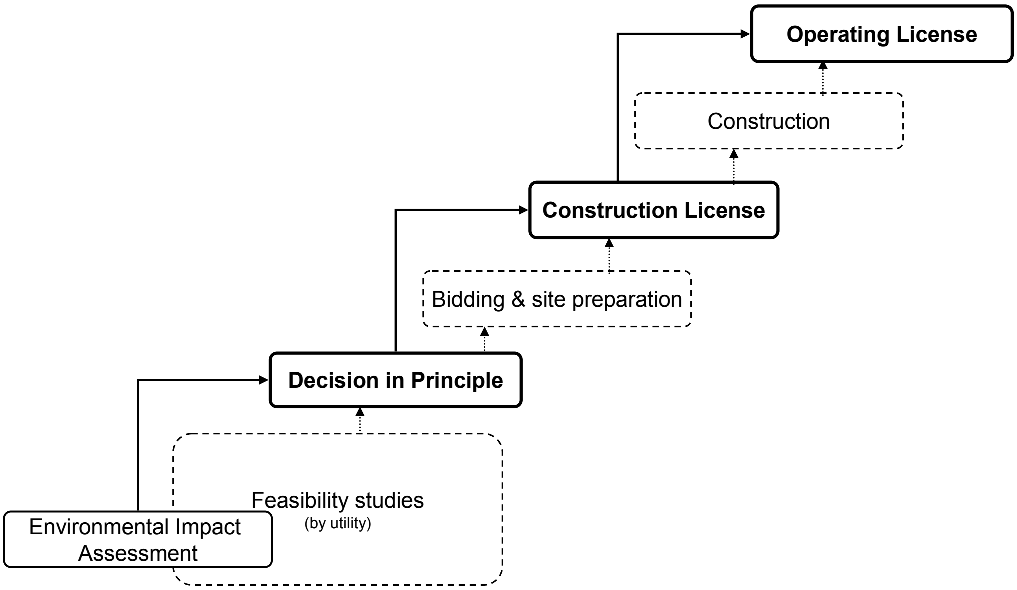

Figure 1. The licensing process of a new nuclear power plant in Finland.

Construction Licence application review at STUK

At the same time as the application for Construction Licence was sent to the Ministry Employment and the Economy (former of Ministry of Trade and Industry), TVO submitted the required licensing documentation to the Radiation and Nuclear Safety Authority (STUK). According to the Finnish Nuclear Energy Decree Section 35, these documents include:

- Preliminary safety analysis report (PSAR)
- Probabilistic risk assessment of the design stage
- Proposal for a Classification Document
- Description of quality management during construction
- Preliminary plans for physical protection, emergency preparedness, and safeguards
- Arrangements for the regulatory control
- Other reports that STUK considers necessary.

Based on the review of these documents, STUK prepared its statement on safety and a safety assessment report, which were submitted to the Ministry in January 2005. STUK's positive statement on safety was a prerequisite for the Government to grant the Construction Licence. In the statement, STUK indicated specific observations on and some further demands for the plant safety.

Radiation safety in YVL Guides

The Guide YVL 7.18, "Radiation safety aspects in the design of NPPs", was updated in 2003 and is now again under revision. This Guide includes radiation safety requirements to be taken into account in the nuclear power plant layout and system design. The Guide covers plant's normal operation, accident situations including severe accidents and aspects of decommissioning. Other relevant radiation guides during the construction licence review were:

- YVL 1.10, "Safety criteria for siting a NPP"
- YVL 7.1, "Limitation of public exposure in the environment of and limitation of radioactive releases from NPPs"
- YVL 7.2, "Assessment of radiation doses to the population in the environment of a NPP"
- YVL 7.3, "Calculation of the dispersion of radioactive releases from a NPP"
- YVL 7.5, "Meteorological measurements at NPPs"
- YVL 7.6, "Monitoring of discharges of radioactive substances from NPPs"
- YVL 7.11, "Radiation monitoring systems and equipment in NPPs".

Further relevant guides for the operating licence review are:

- YVL 7.4, "NPP emergency preparedness"
- YVL 7.7, "Radiation monitoring in the environment of NPPs"
- YVL 7.8, "Environmental radiation safety reporting of NPPs"
- YVL 7.9, "Radiation protection of NPP workers"
- YVL 7.10, "Monitoring of occupational exposure at NPPs".

All of these guides are currently under revision as the whole structure of the STUK's YVL guides will be changed. However, most of these guides were just recently updated, so not major modifications are expected in the revision. The target date for the updated guides is the end of year 2011.

Radiation safety of NPP workers (ALARA)

In the regulatory guide YVL 7.18, a design upper limit for an annual personnel collective dose of 0.5 manSv per 1 GW of net electric power averaged over the plant life is set forth for new reactors. In the European Utility Requirements (EUR) document, the target for annual collective effective dose averaged over the plant life is set as 0.5 manSv per reactor unit.

There are two operating nuclear power plants in Finland; two boiling water reactor (BWR) units at Olkiluoto site and two pressurised water reactor (PWR) units at Loviisa site. They were commissioned between 1977 and 1981. Average personnel collective radiation doses per reactor for existing Finnish NPPs and average values for BWRs and PWRs for the years 1998-2008 are shown in Figure 2. The collective dose at the Olkiluoto NPP has been clearly under the international reference values of the BWR reactors. On the other hand, the comparison of the collective dose at the Loviisa NPP to the PWR reactors does not give such an excellent result. Average collective doses per reactor of the German Konvoi generation NPPs (Emsland 1, Isar 2 and Neckarwestheim 2) and French N4 generation NPPs (Chooz B1 and B2, statistics only from the year 2001) are also shown in Figure 2. The statistics would indicate that the collective dose in the EPR could be low.

The PSAR review of STUK experts on plant radiation sources, shielding, lay-out and radiation protection arrangements, as well as STUK's topical inspection made to the vendor (designer) pointed out some aspects where the applicant and the vendor shall enhance efficient communication between different expert and designer groups. STUK's review on the plant radiation monitoring systems description indicated that the system design covered well the main requirements of YVL 7.11. The detailed design of the systems is now reviewed during the construction phase.

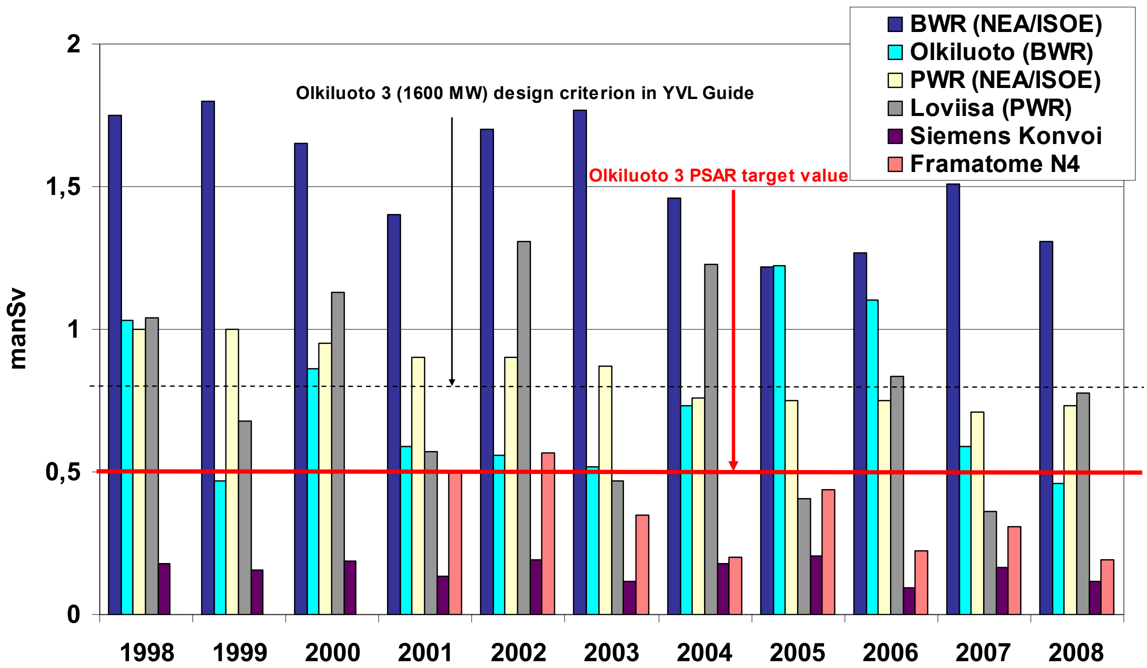

Figure 2. Average personnel collective radiation doses per reactor.

Minimising discharges of radioactive materials during normal operation

The reactor, systems and components containing radioactive substances shall be designed in such a way that releases of radioactive substances and the radiation exposure of the population living in the vicinity of the plant can be kept low. Systems which are capable of cleaning fluids and gases containing radioactive substances shall effectively limit radioactive releases and shall be designed according the principle of Best Available Techniques (BAT). Radioactive releases from a nuclear power plant during normal operation are, to a great extent, determined by leakage from the nuclear reactor fuel rods, reactor coolant and its fission and corrosion products, maintenance operations and waste management (including purification and retention of exhaust gases and liquids).

Based on STUK's review on PSAR and detailed design of the systems, due consideration has been given to the minimisation of discharges of Olkiluoto unit 3. Over the past years fuel leaks at the German and French reference plants have been minor. The same is expected to apply to Olkiluoto 3's reactor fuel. The materials for the reactor cooling circuit have been selected and the water chemistry designed with a view to minimising the creation of radioactive corrosion products. The reactor coolant purification system, the processing system for gaseous wastes, the storage and processing system for liquid wastes and the processing system for radioactive concentrates are based on technology used at the German reference plants, with improvements based on operating experience. The purpose of these systems is to limit the releases of radioactive materials into the environment. These systems are designed with due regard to the Best Available Techniques. Adequate exhaust air filters have been designed for installation in the ventilation systems.

Accident analyses, severe accident management and on-site radiation safety

Analyses of radiological consequences of postulated accidents, design basis accidents, design extension conditions and severe accidents were presented in PSAR and reviewed by STUK. The results from the analyses were well below the dose limits defined in the Government Decree 733/2008 the Safety of Nuclear Power Plants.

At Olkiluoto unit 3, severe accident management is based on reliable depressurisation of the primary circuit and proper containment functions with a unique core catcher. These measures will prevent any major release of radioactive materials (Cs-137 release in excess of 100 TBq shall have a probability less than $5 \cdot 10^{-7}$/a). With regard to protection against external threats, Olkiluoto unit 3 is constructed according to the design features which provide safety and prevent significant releases even in the case of a crash of a big passenger aircraft.

In a nuclear power plant, on-site habitability during accident situations has also to be taken into account in the design. The regulatory guide YVL 7.18 requires analyses of the magnitude and location of the possible radiation sources and evaluation of doses in different accident management and emergency preparedness measures. It shall be shown during the design process, that these doses do not exceed the normal dose limits of a radiation worker, i.e. 50 mSv. STUK's assessment on Olkiluoto 3's PSAR showed that adequate shielding and lay-out arrangement existed in the design. A more detailed review was done during the construction phase based on the detailed design of the systems and structures. It showed that the analysed doses of workers were well below the radiation worker's dose limit.

Regulatory oversight during construction

Regulatory oversight during Olkiluoto 3 construction ascertains that the plant is built according to the design and quality criteria approved in the construction licence phase, and that the prerequisites for high quality end result exist and the licensee is getting prepared for the commissioning and operation of the plant. STUK is currently finalising the review of the detailed design of the systems, structures and components and that includes also verifying that radiation safety requirements are fulfilled.

STUK has also established a construction inspection program to inspect TVO's project progress and implementation. One of the inspections focuses on the consideration of radiation safety issues, i.e., fulfilment of radiation safety criteria in the design and plans for radiation protection programme during plant commissioning and operation. In addition, STUK has performed inspections on the plant vendor and two of them concerned also radiation safety aspects.

Conclusions

Several STUK's YVL Guides deal with radiation safety. STUK has reviewed the application for the Construction Licence of the Olkiluoto 3 nuclear power plant unit in 2004-2005 and is currently finalising the review of the detailed design of the systems, structures and components. This review work includes also verifying that radiation safety requirements are fulfilled. The experience gained in the Olkiluoto 3 regulatory oversight project is taken into account in the revision of the STUK's YVL Guides and in the planning of the next possible oversight projects.

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The new IAEA Basic Safety Standards (BSS) and their implementation for the operation of nuclear installations

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Abstract

After the issue of ICRP 103 /1/ the process of revising the IAEA Basic Safety Standards (BSS) /2/ started. It follows the policy to adopt all the new ICRP considerations as far as possible. This should not be a problem as the ICRP itself stated "stability and continuity" as the headline for its new recommendations.

Indeed, there are few changes in ICRP: the risk factors are nearly the same and became even smaller, the dose quantities remain and, what is essential, the three basic principles justification, optimization and limitation endure. But there are changes. The process based approach is no longer used, there is new terminology and dose constraints were seen as central part of radiation protection not corresponding to the long lasting practice.

The revision of the BSS triggered a lively discussion in which the operators of nuclear installations participated. The large number of comments on the drafts shows that some remains to be done. Clarifications were already possible during the joint meeting of RASSC and WASSC in November 2008. It was e.g. made clear that not an optimized state has to be assured but the process of optimization. Unfortunately this decision was later cancelled, a situation which is not acceptable. It was also revealed that sometimes it was not yet clear who is responsible and what is a requirement and what a guideline. This could be approved in the following drafts, but the reformatted drafts failed to concentrate on "real" requirements.

For operators of nuclear facilities it is essential that changes in the BSS lead to higher safety and not to higher bureaucracy only. The demand for dose constraints for all sources that could be drawn from the text formally would be an example for that.

The drafting of the new BSS is in the hands of the IAEA and its co-sponsors. The operators need to focus on the discussions in the Safety Standards Committees and on comments. This is an important part of the process and we noticed that experience from practice is welcomed.

Material and methods

The basis for the analysis of the possible changes in radiation protection and their implications to the daily practise has been and still is the drafts of the revised BSS. The last is now Draft 3.0, posted for comments by the IAEA Member States until the end of May 2010. The plan is to finish the work at the end of the year, so the discussion here and during the Meetings of the 4 Safety Standards Committees of the IAEA are the last opportunity to correct things which might be misleading, not adequate or imprecise.

Results

The European nuclear industry has taken the opportunity to participate in the discussions about the revised BSS by a rather new organisation, the European Nuclear Installations Safety Standards Initiative (ENISS). ENISS was given an observer status in 3 Safety Standards Committees (NUSSC, RASSC, WASSC). From the very beginning ENISS commented the drafts of the revised BSS. The ENISS comments focused on 2 major issues: the provisions in the BSS about optimization and the prominent role of the Dose Constraints (DC). All other remarks have been of less importance, sometimes editorial or were accepted by the drafters.

Discussion

In addition to justification and the setting of dose limit, optimisation has been one of the basic principles in radiation protection for a long time now. This has not changed, even with the publication of ICRP 103. The main message of ICRP 103 is "continuity and stability". Persons active in radiation protection will be happy about this message, since it confirms the practical experience in radiation protection so far and calls for its continuation. However, it is in the nature of optimisation not to lead to a standstill. Optimisation means asking yourself constantly if you did everything to keep the risk of radiation exposure as low as reasonably achievable "taking into account all prevailing circumstances".

The implementation of these radiation protection principles has led to a constant decrease of actual exposures for workers. This is clearly indicated by statistics published by the Federal Office for Radiation Protection, BfS, /3/. They are on such a low level that occasionally the issue is raised if any further decrease could still be possible while maintaining the optimisation principle or if minimisation starts or has already started to dominate as a new principle.

> *"From 2003 to 2007, the mean annual dose of all persons exposed decreased by an average of 3% per year. Simultaneously, the number of persons exposed increased by an average of 4% per year. The collective dose remained almost constant. Measured in the increase of persons exposed, this presents a relative decrease in the dose. However, this is considerably less than from 1999 to 2003, when as a result of the new Radiation Protection Ordinance entering into force an absolute decrease in the dose could be observed in spite of a considerable increase in the number of the persons exposed.* ***This may indicate that for the majority of the employees the average annual exposure is becoming as low as reasonably achievable by optimisation measures."***

Fig. 1. Quotation from the BfS Report /3/, bold letters by the authors.

Nevertheless, we state: Radiation protection is on an excellent level implementing the principle of a continuous improvement in the form of optimisation, and a necessity to change the radiation protection concept is therefore not recognisable.

Dealing more intensely with the new basic recommendation of the ICRP of 2007 (ICRP 103), you will find that the ICRP has changed the system in spite of the expectations mentioned above. With the new and specific emphasis on "dose constraints" already introduced in the ICRP 60, these are now seen as one of the most important means of radiation protection optimisation, if not the most important one. This, however, may create a difference between the radiation protection optimisation and the ALARA principle. The reasons for this are difficult to understand.

If you ask around in practical operation, and try to find out where dose constraints are being used, you will learn that they can hardly be found. However, often operational dose restrictions are found that serve to control compliance with dose limits, such as maximum daily doses, maximum monthly doses, permitted dose fractions for external and internal exposure. Furthermore, there are action levels for individual doses that trigger measures when reached, for instance, monitoring. Finally, also collective dose restrictions are defined as standard, in part even as goals for the operation related radiation protection or as decision gates for the planning of radiation protection measures. So one can find a host of dose-oriented criteria of which none complies fully with the "dose constraints" introduced by the ICRP.

It is of interest that Article 7 of the Council Directive 96/29/EURATOM /4/ points moderately to dose constraints: "Dose constraints should be used, where appropriate, within the context of optimisation of radiological protection”. The implementation of this Council Directive 96/29/EURATOM into the German Radiation Protection Ordinance does not even include dose constraints. Now, the question is, what does the ICRP aim at by emphasising the dose constraints and will this lead to an improvement of the radiation protection in practical operation?

According to the ICRP, dose constraints are to be determined relative to sources. By definition, they are lower than dose limits. Contrary to the non-compliance with limits a non-compliance with dose constraints should not be a legal offence but needs to be excluded by adequate planning. If the dose constraint is a solely prospective planning measure or is to be used as well for retrospective evaluation of practical operation is controversial and not clearly indicated by ICRP statements.

The definition of "dose constraints relative to a source" creates a problem: Which source is meant? According to the ICRP, a radiation source is everything leading to radiation exposure. This may be an enclosed radiation source, with several kBq or TBq, a nuclear power plant in its entirety or even individual activities within the plant, such as the replacement of a valve, or an X-ray machine in a dental practice. Also the invasive X-ray diagnostics and even the radon exposure in a residential building qualify a "source" according to the ICRP. Already a first glance at the variety of possible radiation exposures shows you there are just as many options to define radiation sources. Does this mean several thousand dose constraints should be determined? For this, the ICRP does not provide selection criteria. So far, for the Council Directive 96/29/EURATOM the magic word "where appropriate" - even connected with the word "should" - was helpful. Such a restriction seems to be indispensable in the future as well; otherwise, a questionable bureaucracy is to be feared with regard to dose constraints. Indeed the BSS contains a lot of phrases such as “as appropriate”, “where appropriate”, “if appropriate”, otherwise DC’s could not be accepted at all.

For the application, it is imperative to take the dose levels into account. The significance of constraints for very small doses, for example in the range µSv, must be considered low from the outset.

Ionisation smoke detectors are a typical example, for which a dose constraint would not make sense. Here, other radiation protection measures apply based on the optimisation principle, e.g. only use as much activity as necessary, make the design robust and subject of design approval. As the example shows, it is worthwhile to develop criteria, when and where dose constraints are even "appropriate". In the discussion of new BSS, these considerations have not been included yet. Probably, the definition of dose constraints is only reasonable when there are actual design options regarding radiation protection, for example with complex projects or within the design phase of a source.

Which reasons are given by the ICRP for this particularly emphasised position of the dose constraints?

Dose constraints should limit inequity thus avoiding unacceptable high individual exposures as a result of the radiation protection optimisation.

So far, this objective was attributed to the dose limits, i.e. any exposure below these limits was basically acceptable. It is practise that exposures close to the limit values will be avoided. This is also a preventive action to avoid l exceeding the limits. It is customary as well to assign a higher priority to the reduction of exposures close to dose limits, and also a higher alpha value for quantitative considerations. These two procedures have proven to be efficient in reducing exposures at the upper level. To differentiate the acceptance even below the dose limits, maybe even quantitatively supported by DC’s, is very questionable.

Occupationally exposed persons know about possible hazards caused by exposure. They also know how to influence the level of exposure by one's own behaviour. Apart from that, the development of the exposures has been showing a decreasing tendency for many years, as already specified above, and is now on a level clearly below the limits /5/. Higher exposures have also decreased continuously. In radiation applications in the medical field approx. 1%, in the industrial field approx. 2% of the exposed persons are exposed "to a higher degree", i.e. they receive doses of more than 1 mSv. For the few exposure groups with higher exposures, in the first place this affects flight attendants and pilots (approx. 9% of the persons monitored are in the dose range between 1 and 6 mSv), it may be reasonable to consider specific measures to reduce these exposures. However, this reasoning should not justify a change of the entire protection philosophy, as would be provided by a global introduction of dose constraints.

Dose constraints should prevent exceeding dose limits in case of exposures to multiple sources.

This goal has always existed. Its achievement is simple. The personal dosimetry applies to professionally exposed persons. For severely changing exposure situations, the official personal dosimetry performed at longer intervals is supplemented by an operation related dosimetry, which can be evaluated immediately. Thus, intervention to prevent exceeding limits even under consideration of several sources is always possible.

Public exposures are caused by discharges of radioactive material and direct radiation. In Germany, these exposures are controlled by considering the already existing radiological burden within the licensing procedure specified by the Radiation Protection Ordinance. Other countries may have similar regulations. In extreme cases, this regulates new discharges or new direct radiation fractures stricter than the ones already approved for a site. This must be accepted by someone arriving later. There is absolutely no problem for public exposure in the vicinity of such plants. So, even control of multiple exposures does not require the introduction of dose constraints

Apart from that, and this is even stipulated by the ICRP in publication 103, there is usually only an exposure to one source (meaning here one plant, one application technology) both for occupational and public exposure.

Dose constraints are to exclude solutions, which are not considered optimal.

In connection with the principle of optimisation, this presents a problem: How should one know, which is the optimised solution prior to the implementation of the optimisation itself? A way out could be considered the so-called "best practices" for certain applications of ionising radiation, from which preferred solutions could be derived. Strictly speaking, however, general specifications contradict the ALARA principle according to which the circumstances of the individual case must be taken into account. Missing criteria and methodical references for a derivation of dose constraints are another relevant problem. Unfortunately, ICRP 103 is of no help here.

Dose constraints could be understood as being results derived from the optimisation process. Here it is also easy to envision that for comparable technologies or activities dose constraints defined by reference numbers could be useful. This, however, contradicts the ICRP statement that optimisation is required below the dose constraints. Thus, certain arbitrariness exists in the definition of dose constraints, and this is definitely misfortunate.

Another problem is that the ICRP does not provide clear information as to who is to be responsible for finding and setting the dose constraints. ICRP considers the plant operator responsible for optimisation, but recognises a certain necessity to co-ordinate dose constraints with the authorities. Here, there is a lot of room for discussion with regard to the future integration into European and national regulations, and in particular with regard to the subsequent practical operation, which is not of much help to radiation protection.

Ensure that protection and safety are optimized.

The second item of importance for the discussion of the revised BSS is the inadequate formulation of the optimization principle in the BSS, stating that “operators/licensees have to ensure that protection and safety are optimized”. We still believe that this is a crucial item for the implementation of radiation protection in practice. The principle of optimization based on ICRP 103 is correctly described in the introductory chapter 1 of the BSS as a process. In the main text of the BSS optimization has several times been reduced to the misleading phrase "to ensure that protection is optimized". This could create difficulties in practice as there are no clear criteria for the "optimized solution". As we believe, there is no principal difference among the radiation protection experts about the optimization principle, we suggested to change the corresponding formulations. In the RASSC/WASSC Meeting in November 2008 this change was accepted unanimously. However, during the next meetings of the drafting group a backchange was decided. Unfortunately in the following meetings of the Safety Standards Committees a deep going discussion of this issue did not take place.

To explain the concerns with the too short wording of the BSS regarding optimization some questions may enlighten the situation:

- What is optimization in legal terms?
- What are the criteria of being optimized?
- When and how often do operators have to demonstrate that protection and safety is optimized?

Optimization in legal terms

As described in chapter 1 of the BSS according to ICRP optimization is one of the principles of radiation protection and it is definitely a process.

"The optimization of protection and safety, when applied to the exposure of workers, members of the public and comforters and carers of patients undergoing radiological procedures, is a process for ensuring that the magnitudes and likelihood of exposures and the numbers of individuals exposed are as low as reasonably achievable, taking social and economic factors into account. This means that the level of protection should be the best under the prevailing circumstances, maximizing the margin of benefit over harm, and will thus not necessarily be the option with the lowest risk or dose. Optimization is a forward-looking iterative process requiring both qualitative and quantitative judgements, and may be used, if appropriate, in conjunction with individual source-related values of dose or risk that serve as boundaries in defining the range of options in optimization."(BSS Draft 3.0 para 1.14)

ICRP has issued so-called foundation documents in connection with ICRP 103 and the one dealing with optimization is the ICRP Publication 101 /6/. It is clearly stated there that optimization is a process and not dedicated to a specific result. The outcome of the optimization process will always be specific under the prevailing circumstances. From the legal point of view a legal requirement can therefore only be directed towards the process or the principle. The nuclear industry has made time and again a proposal for changing the text into "have to ensure that protection and safety are subject to an optimization process". As said this was already accepted in November 2008.

A counterargument was since then that having a process only is too soft and does not guarantee that results of the process will be implemented. Although this argument is not very convincing one could in addition demand in the BSS that results of the optimization process have to be implemented. From the standpoint of radiation protection practice this is logical anyway.

The current situation in many radiation protection legislations is that only the principle of optimization is fixed legally. A prominent example is the Council Directive 96/29/EURATOM, where Article 6 says, that ”each Member State shall ensure that in the context of optimization all exposures shall be kept as low as reasonably achievable, economic and social factors being taken into account.” The German Radiation Protection Ordinance e.g. indeed contains only this principle. Note that the provision is directed to the Member States only. The BSS addresses not only the regulator and competent authorities but also operators.

Criteria for being optimized

Optimization in the textversion of the ALARA Principle may be interpreted differently: one may stress the word ”low”, asking for exposures in the µSv-range or even Zero-dose, one may stress the ”reasonably” as operators are often said to do so, or one may stress the ”achievable”, as engineers might willing to do. Social factors may be seen differently from a workers point of view and a stakeholders point of view. These examples easily show the great variety of interpretation of being optimized. There are no clear criteria. Getting a licence may be an indicator for being optimized, otherwise an authority would probably not have issued the licence. Getting a type approval is also an indicator. The question than remains what to do during operation. The BSS do not answer this questions and give no indication for the fulfillment of its strict requirement to ensure being optimized. But such a situation is dangerous for any development of protection. Operators might be forced not to change their operational mode as the fear not to get a renewed licence.

When and how demonstrate ”being optimized”?

The optimized solution today may be obsolete tomorrow. The circumstances change permanently. For any radiation protection expert in the field it is clear that one cannot change the rules everyday. Operation needs stability. The BSS do not contain details of how to ensure the optimized solution. In the planning of the framework of IAEA standards a guide about optimization is not to see. Without guidance on optimization it will be difficult to implement the requirement. The experience so far goes along the line of having a process of optimization. With the new formulation of the optimization requirement in the BSS the proven system will be questioned and the outcome is unclear. A situation which is not suited to foster protection and safety.

Conclusions

In summary, it becomes clear that dose constraints according to the ICRP version are needed neither in Germany nor in the EU or elsewhere to achieve excellent radiation protection. Care should rather be taken to avoid new bureaucracy from developing that would do more harm than good.

Dose constraints, defined and applied flexibly and with a sense of proportion make sense when used by the operators of nuclear plants as management tools for a proven practical operation and as a result of optimisation considerations. It may be reasonable to determine a few operation related values. Optimisation, however, is by far more than the determination of such dose constraints.

The shortening of the optimization principle to the requirement "to ensure that protection and safety are optimized" is inadequate, not in line with the ICRP philosophy and the guidance given in ICRP 102. The drafters of the BSS need to come back to the decision from 2008 and reinstall the formulation " ensure that protection and safety are subject to an optimization process".

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/2/ DS 379, International Basic Safety Standards for Protection against Ionizing Radiation and for the Safety of Radiation Sources, Draft 3.0, 2010

/3/ BfS-SG-12/09; The Occupational Radiation Exposure in Germany in 2007, Report of the Radiation Protection Register, page 23:

/4/ Council Directive 96/29/EURATOM laying down basic safety standards for the protection of the health of workers and the general public against the dangers arising from ionizing radiation, 1996

/5/ "Information by the Federal Government: Environmental Radioactivity and Dose Exposure in 2007", page 27

/6/ ICRP Publication 101, The Optimisation of Radiological Protection, 2005

Radiation protection culture in the nuclear industry

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Occupational radiation exposure – an overview on the exposure of the workers in facilities of the nuclear fuel cycle

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Abstract

Workers are subject to radiation exposure in several industries. This contribution will provide an overview of the exposure of workers in facilities of the nuclear fuel cycle. Following a first overview, details on the exposure in nuclear power plants in operation as well as under decommissioning are addressed. Based on recent data, trends in the exposure of workers will be discussed and some examples will be given on how experiences contribute to improvements concerning the exposure of workers.

The data available show, that with time in general the exposure of the individual worker decreases. This is a result of a manifold effort by all parties involved in the radiation protection in nuclear facilities, which is mainly based on a consequent experience feedback from past operation to improve design and operation and regulations relevant for the nuclear sector.

Introduction

In its 2008 report to the United Nations General Assembly, the United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR) estimates, that today more than 22.8 million workers are exposed to ionizing radiation (UNSCEAR 2008). About 13 million workers are subject to ionizing radiation from natural sources. The remaining 9.8 million workers belong to all sectors of artificial sources, 75 % of them related to the medical sector. Among the 25 % of the 9.8 million workers are those working in facilities of the nuclear fuel cycle (nuclear sector).

This contribution to the Third European IRPA Congress addresses worldwide trends in the occupational exposure in facilities of the nuclear sector. For that, a first overview on the situation worldwide is given, followed by a focus on the situation in nuclear power plants as – in general – one of the main sources for exposure. The factors which influence the occupational exposure and its decrease at nuclear power plants are manifold. As an example for more details and influencing factors, some German trends in occupational exposure in nuclear power plants are explained.

Trends in the occupational exposure in the nuclear sector worldwide

In its 2000 report UNSCEAR has published data on the occupational exposure related to work in different types of facilities related to the nuclear sector (UNSCEAR 2000). The data published cover the years 1975 until 1994 (Note: Currently, an update of this report is under preparation which will be published early 2010 but was not available at the time of preparation of this contribution). The nuclear sector comprises facilities for uranium mining and milling, for uranium enrichment and conversion, for fuel fabrication, for operation of nuclear power plants (NPP), for reprocessing and for research within the nuclear fuel cycle. Although waste management is another branch in the nuclear sector within the UNSCEAR survey only little data become available separately as they are mainly covered in the other branches mentioned.

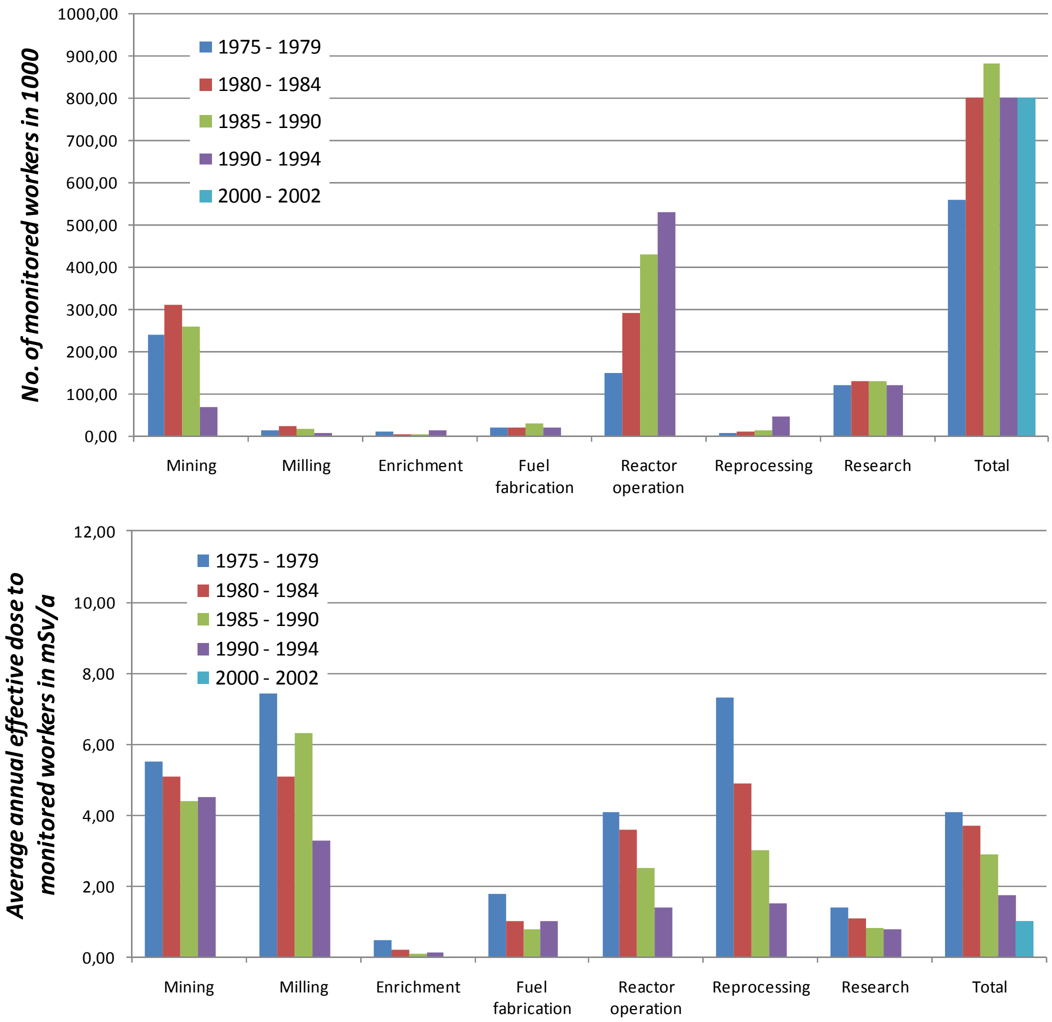
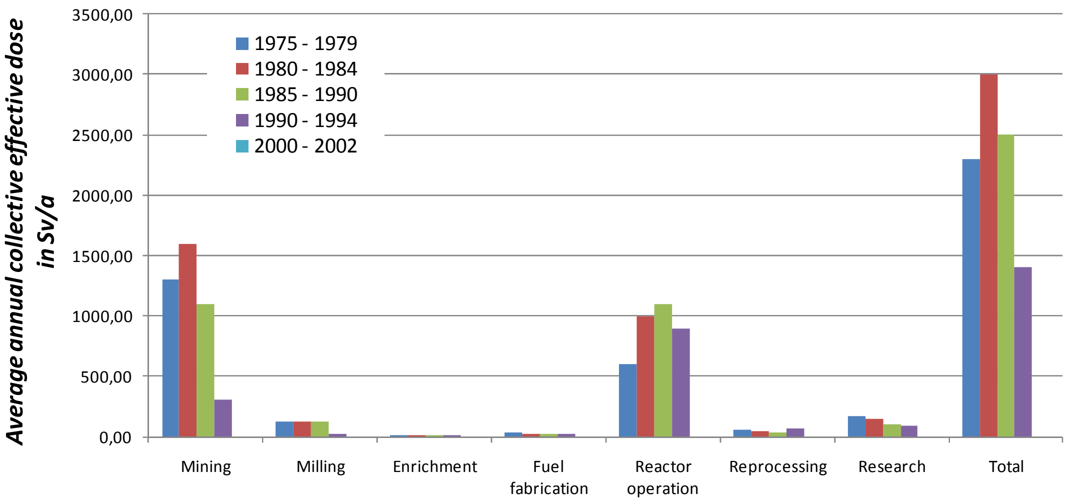


Fig. 1. Number of monitored workers in thousands and average annual collective doses for the different branches in the nuclear sector.

Fig. 1 and Fig. 2 present the trends of estimated numbers of monitored workers, estimated average annual collective doses, the resulting average annual effective doses of monitored workers and the fraction of monitored workers with an annual effective dose of more than 15 mSv for the different branches within the nuclear sector. The data

1975 – 1994 are taken from the 2000 report of UNSCEAR (UNSCEAR 2000), the data 2000 – 2002 are taken from the 2008 report (UNSCEAR 2008), but not for all items 2000 – 2002 numbers were available.

The UNSCEAR data were obtained by surveys of occupational radiation exposure, requesting information from various national authorities and institutions and by collecting supplementary data, e.g. from the Information System on Occupational Exposure (ISOE) of the OECD Nuclear Energy Agency (NEA). As a matter of fact the data reported differ due to national differences e.g. in the statutory dosimetry systems, on reporting of exposed or monitored workers, on the national reporting levels. In addition, not for each period and for all nuclear branches data for all facilities have been supplied. Especially in case of the mining branch, some estimates became necessary to achieve a more complete view on the worldwide situation for the data of the period 1990 – 1994, resulting in some larger uncertainties in the number of monitored workers and the related average annual collective effective dose. Nevertheless, the data prepared allow important insights in the occupational exposure of workers in the nuclear sector.



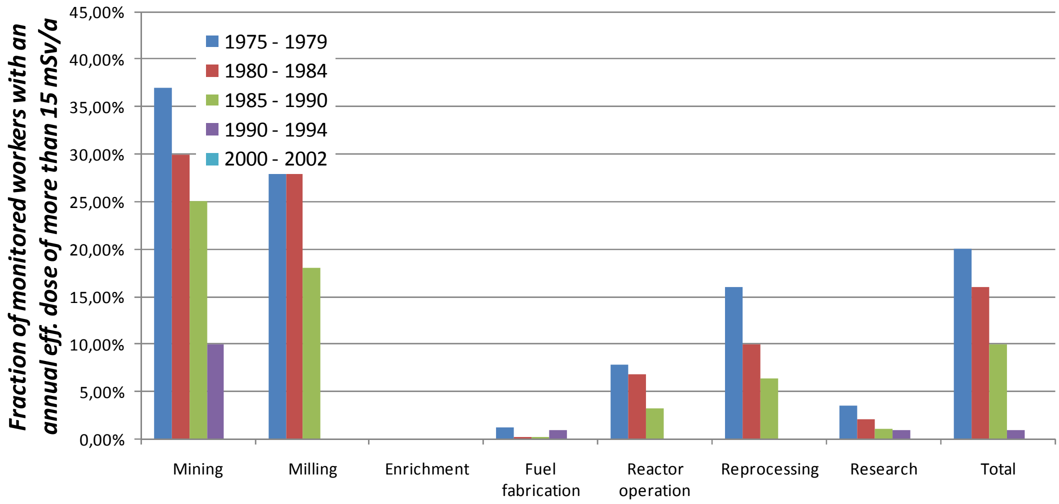


Fig. 2. Average annual effective doses of monitored workers and fraction of monitored workers with an annual effective dose of more than 15 mSv/a for the different branches in the nuclear sector.
(Note: for several branches the fraction for doses of more than 15 mSv is close to 0 in 1990-1994)

Independent from the uncertainties mentioned, the figures and numbers show, that

(a) NPP operation (reactors) is the branch with the highest number of monitored workers in the last period but not with the highest average annual effective dose to monitored workers,
(b) for all branches a decrease in the annual collective effective dose to monitored workers can be observed in the last periods,
(c) the fraction of monitored workers with an annual effective dose of more than 15 mSv decreases over time, but for the mining and milling branches the fraction is still the highest.

As such the data show a general worldwide trend of decreasing occupational exposure by ionizing radiation in the nuclear sector. The average annual effective doses of monitored workers in the nuclear sector decreased from 1975 to 1994 from 4.1 mSv to 1.75 mSv; the recent data of the 2008 report indicate a further decrease to about 1 mSv.

Details on the occupational exposure in nuclear power plants worldwide

While UNSCEAR collects and analyses data on all aspects on the effects of ionizing radiation, the Information System on Occupational Exposure (ISOE) of the OECD Nuclear Energy Agency (NEA) focuses on the occupational exposure of workers in NPPs. Established in 1992 by the OECD-NEA, the overall goal of the ISOE is to facilitate the optimisation of worker radiological protection in NPPs through collection and assessment of relevant data and the exchange of experiences on radiation protection in NPPs. As of January 2010, 63 operators of NPPs in 27 countries are official participants of the ISOE, representing 311 NPPs in operation and 40 NPPs under decommissioning. The central database of ISOE contains data for many more NPPs in operation (401 NPPs) and under decommissioning (about 80). As such the ISOE database represents the largest database on data on the occupational exposure in NPPs under operation.

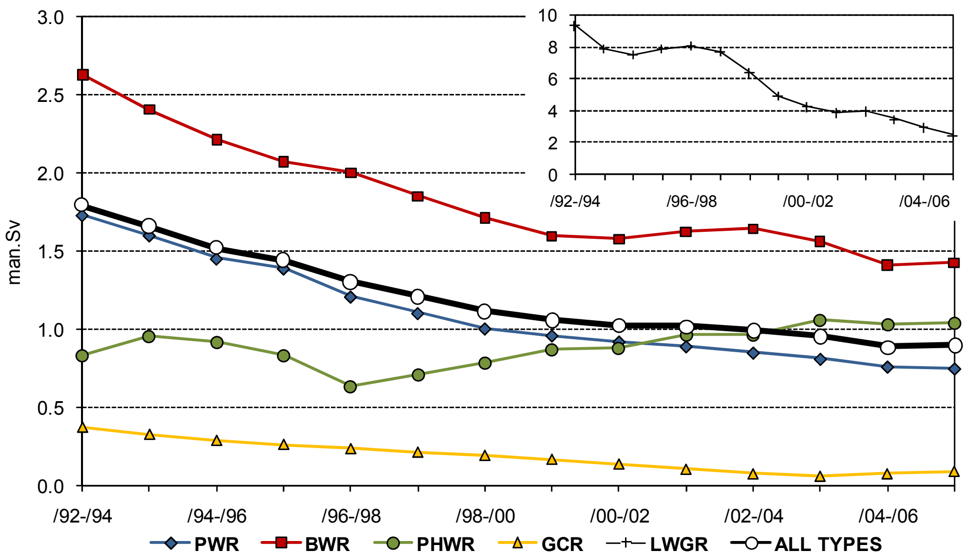


Fig. 3. Three years rolling average annual collective effective dose per nuclear power plant for different reactor types (Ahier 2008, ISOE 2009).

Fig. 3 shows the three years rolling average annual collective effective dose for all NPPs participating in ISOE and for different reactor types. The data show a decreasing trend in the exposure, which supports the conclusions drawn by the UNSCEAR. Nevertheless, it should be taken into account, that the data of the different reactor groups are based on the data of different numbers of NPPs. E.g. for the year 2007 for PWR group (including VVER) about 230 nuclear power plants provided data, while for BWR about 70 nuclear power plants, for PHWR about 30 and for LWGR only 1 NPP provided data (ISOE 2008). Accordingly extraordinary work during outage or refurbishment in an individual NPP has different impact on the averaged dose depending on the reactor group. Such an influence can be recognized in the data of the PHWR, which is representing the CANDU reactor type. Since 1999 (ISOE 2001) an additional NPP provides its data which are significant higher than the average at that time. In combination with major work in other NPPs this resulted in an increase of the average dose overlaying any decreasing trend due to improvements made.

The database data contains also data on the occupational exposure at NPPs under decommissioning. Fig. 4 shows the average annual collective effective dose and the number of the related reporting NPPs under decommissioning (ISOE 2009). The evolution of the average dose with time may be interpreted as a decreasing trend. In fact, such an interpretation is not correct, as – different to the operation of a NPP – the exposure strongly depends on kind and amount of decommissioning work of a year which change during progress of a decommissioning project dramatically. The increase of the reporting NPPs is related to the fact, that ISOE promotes the collection of decommissioning related data within the last years and that new decommissioning projects were started.

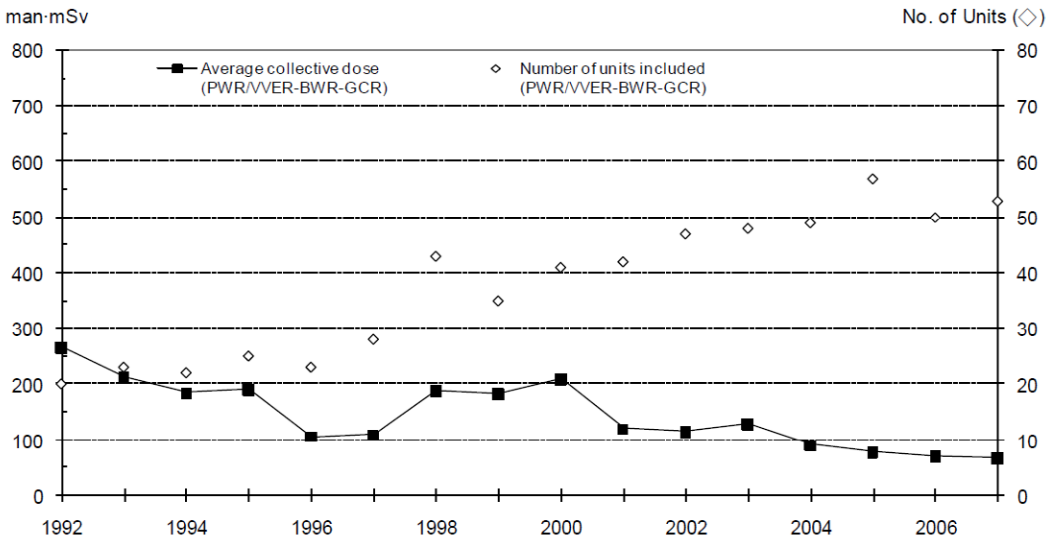


Fig. 3. Three years rolling average annual collective effective dose per nuclear power plant for different reactor types (ISOE 2009).

However, the data available for NPPs under decommissioning show that the average annual collective effective dose is much lower than in case of NPPs in operation. The ratio varies strongly from year to year at least due to the changing decommissioning work.

A spotlight to a national situation – examples from Germany

As already discussed in the context of the UNSCEAR data, main branches for occupational exposure in the nuclear sector are mining and milling and nuclear power plants. Today, in Germany no mining and milling facilities of the nuclear fuel cycle are in operation anymore. 17 nuclear power plants at 12 sites are in operation and 17 nuclear power plants (at 13 sites) are under decommissioning while 2 nuclear power plants were completely dismantled meanwhile.

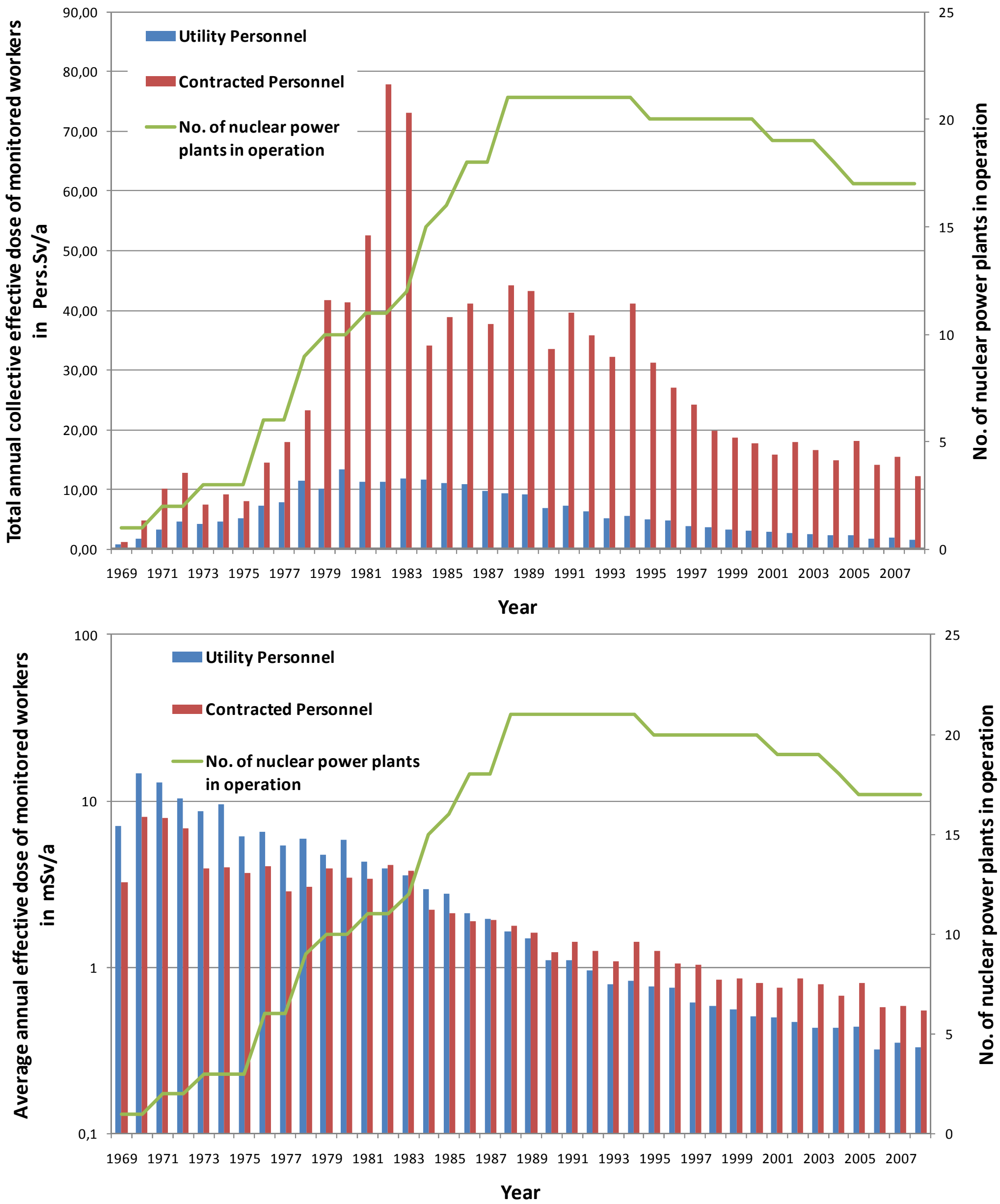


Fig. 5. Total annual collective dose and average annual effective dose of monitored workers for German nuclear power plants in operation.

Fig. 5 and Fig. 6 show the total annual collective dose and the average annual effective dose to monitored workers for all German NPPs in operation and under decommissioning. Both figures are based on data from the occupational dosimetry systems operated by the operators of the NPPs. Thus contracted workers are attributed to each single NPP even, if they entered two or more NPPs during the year. Accordingly, the number of contracted workers in the figures is higher than the number of real persons or the related number of workers registered in the German statutory dosimetry. As a consequence, the average annual effective dose for monitored workers is lower than in case of using the official data of the statutory dosimetry.

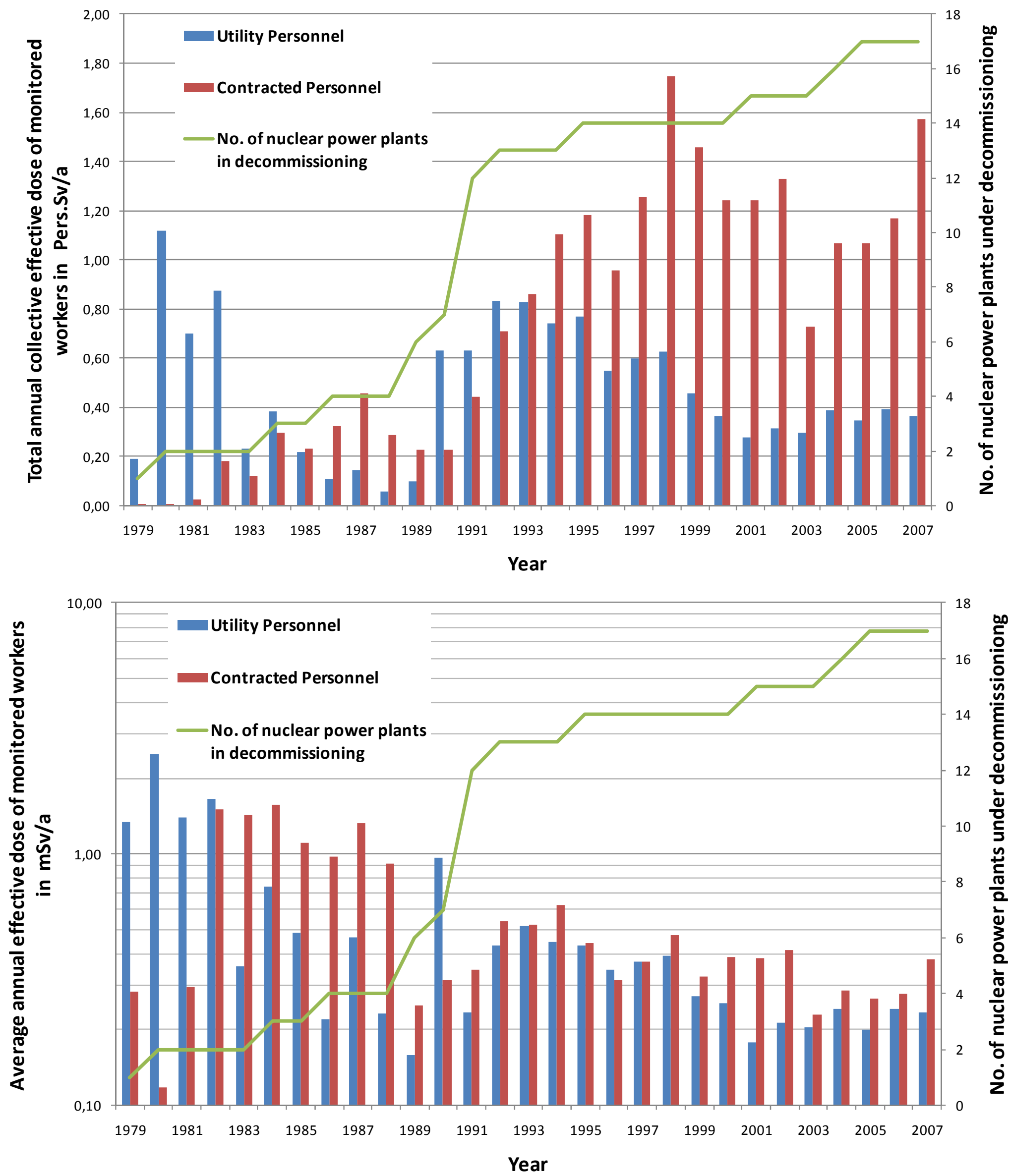


Fig. 6. Total annual collective dose and average annual effective dose of monitored workers for German nuclear power plants under decommissioning.

Fig. 5 shows a clear long term trend of lower total annual collective doses and average annual effective dose of monitored workers. In general terms, this trend is due to a consequent experience feed back and radiation protection work planning process (so called IWRS radiation protection planning) during operation and due to improvements in the design of newer NPPs. For some NPPs, back fitting activities were performed during operation improving also the radiological work conditions. Especially the long term trend for the average annual effective dose of monitored workers indicates that in general the working conditions for workers changed and improved. A further analysis of the distribution of the individual annual effective dose of the monitored workers in German NPPs confirms that the fraction of workers with a higher dose decreases (e.g. Fig. 7 on the distribution of the fraction of monitored utility workers). Since 2001 exposures above the German dose limit above 20 mSv/a, did not occur.

Fig. 6 shows the numbers for nuclear power plants under decommissioning. A trend similar to that of Fig. 5 can not be recognized as was already mentioned during discussion of data from the ISOE. This is due to the fact that the annual effective dose for each nuclear power plant strongly depends on the decommissioning work and the related radiological conditions and which change form year to year, following the overall work planning and decommissioning strategy for the NPP. Obviously, the type, inventory and operational history of the NPP influence the radiological conditions. As such no trends can be expected. But, improvements e.g. due to experience feedback take place and can be identified on the level of an individual NPP. Comparing the data of German NPPs under decommissioning it turns out as a rule of thumb that the decommissioning related average annual effective dose of monitored workers is about 10% to 20% of that for operating, but this ratio depends on the NPP and – as mentioned – the work to be performed.

The influence of the design of the German NPP on the occupational exposure during operation can be easily recognized in Fig. 8 in which the average annual collective effective dose of the works in German NPPs, belonging to a specific design generation, is presented. While the figure considers all commercial NPPs in operation in former West Germany the six NPPs of the former East Germany are not considered, as they belong to another design generations of Russian design. The NPPs of PWR type are divided into 4 generations; during design of the PWRs of the fourth generation (so called Konvoi reactor) all previously made experiences, esp. from operation of PWRs of the first generation, were considered (e.g. on the use of material with low neutron activation, design of compartments to separate components with high dose rates from those with low dose rates to reduce exposure during later maintenance work, design requirements for low dose rates at frequently accessed locations). As such, the Konvoi reactors offer well improved radiological conditions due to design.

Fig. 8 illustrates also the different contributions from the German NPPs to the total annual collective effective dose of the German NPPs in operation and the related annual effective dose of monitored workers. As an example on the different situation in the individual NPPs, for 2008 the average annual effective dose for monitored utility workers varied between 0.04 mSv/a and 0.78 mSv/a. As already mentioned, the individual contributions may change significantly, typically more for the earlier generations, as they are due to the height of their dose contribution more sensible to changes in workload than the Konvoi reactors.

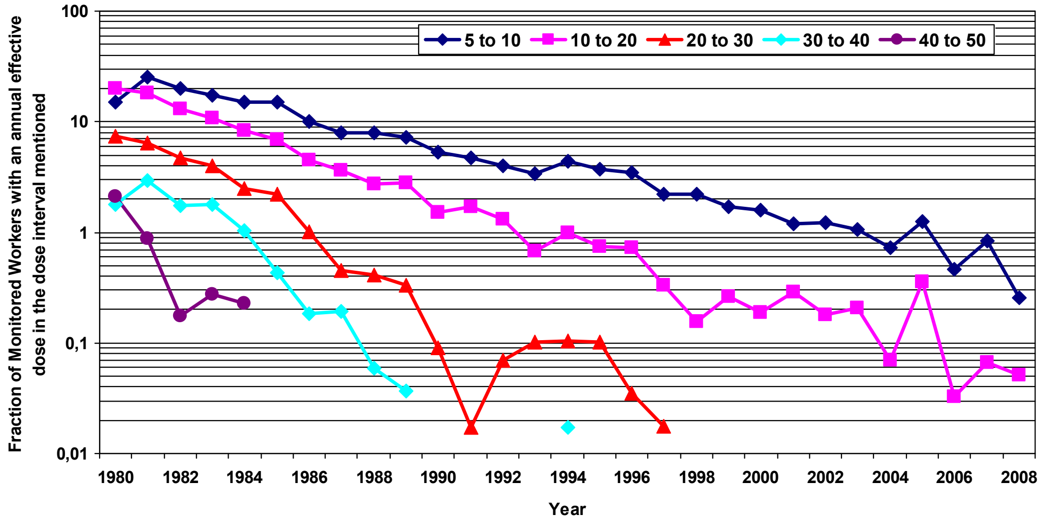


Fig. 7. Distribution of the fraction of monitored utility workers with a specified annual effective dose for German nuclear power plants in operation (in %).

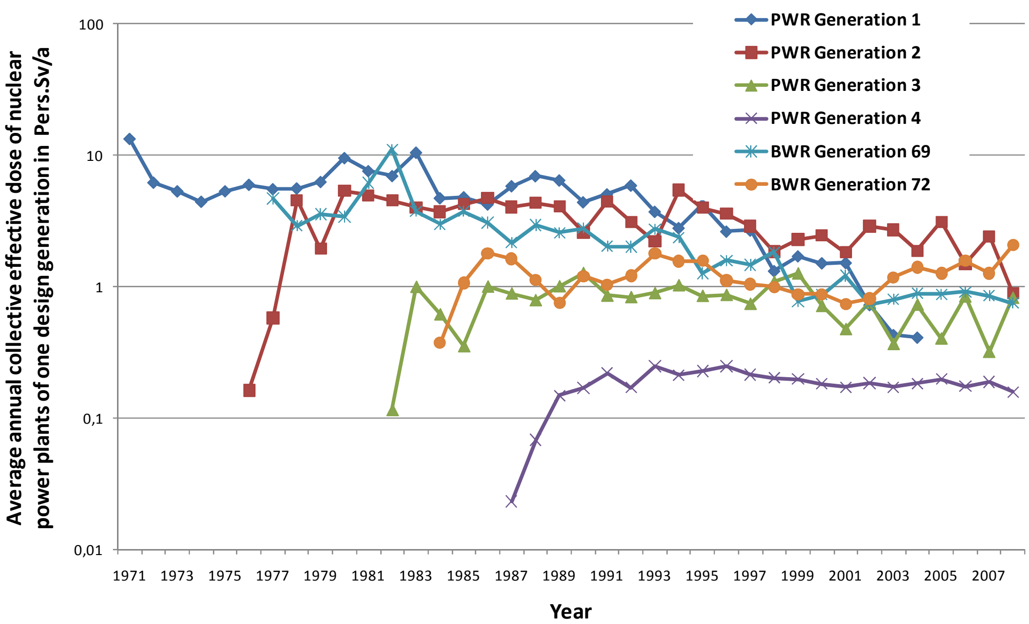


Fig. 8. Average annual collective dose of workers, averaged for German nuclear power plants of similar design generation.

Conclusions

Recent international data of UNSCEAR show, that the occupational exposure for workers in the nuclear sector has decreased during the last three decades. Although these data, especially those related to the mining and milling branch within the nuclear sector, are connected with some uncertainties, but the trend shown can be regarded as realistic.

When focussing on the situation within NPPs, the international data of the ISOE publicly available allow a more specific analysis of the situation including a separation between NPPs in operation and under decommissioning. For the common reactor types

PWR (including VVER) and BWR, a decreasing trend on the average annual collective effective dose for nuclear power plants in operation can be observed; the data for the group of NPPs of reactor type PHWR show an increasing trend, which is due to the increasing number of reporting NPPs in ISOE and the contributions of single NPP with high doses on the average value.

Finally, best insight can be obtained by analysis of national data allowing a high degree of differentiation. As such an example, the data on the occupational exposure in German NPPs show again a decreasing trend for the total annual collective effective dose of German NPPs in operation and for the average annual effective dose of the monitored workers. Depending on the type and generation of the NPP, the improvements during operation are different due to the resulting radiological conditions. Improvements can be observed for all NPPs but the extent of improvement, e.g. expressed in terms of dose savings, depends on the radiological conditions – typically savings are higher in NPPs of the first generations. The national example shows also that an adequate design will ensure best improvements in radiation protection.

Concerning the occupational exposure in NPPs under decommissioning, the available data and their trend are dominated by the specific situation during decommissioning; different to operation, the annual work for a NPP under decommissioning changes dramatically with completely different radiological conditions. As such, the annual collective effective dose of a NPP under decommissioning changes significantly due to the nature of work from year to year. As a consequence, no trends in the annual data can be expected which allow an easy conclusion on radiation protection improvements. But, experiences show, that such improvements take place, resulting e.g. in a much lower real exposure of monitored workers than was expected during planning of the decommissioning.

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Radiation protection issues in fuel-manufacturing

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Floating nuclear power reactors – Fiction or future?

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Abstract

Russia's current floating nuclear power plant (FNPP) plans were initiated nearly 20 years ago, with a competition to develop small-scale power plants for the Russian Arctic. While the locations where the first plants will be sited have been altered several times, the basic design of the prototype plant remains the one that won the competition in 1994: a variant of the Russian icebreaker reactor. This paper describes the development in the Russian plans for building floating nuclear power plants (FNPP), and the status for the ongoing construction project. There have been two major changes since 1994: Russia has promised to fuel the reactors with low enriched uranium (LEU) fuel – whereas icebreakers use 36–90% enriched uranium fuel – and exporting FNPPs has become a key goal for its developers. The project now under realization has also been redesigned to increase safety. Attention to proliferation-resistance has also been highlighted in many Russian presentations on the FNPPs, making a switch back to HEU less likely. Redesigns to increase safety have also been widely touted. While there are many detailed reports about the fuel design, though, there is no data on performance – indeed, it is not clear that performance has indeed been sufficiently demonstrated. There are reports (without data) that testing of the fuel has occurred in other Russian reactor types, however, and Russian designers appear satisfied with the current fuel design. Russian has also approached the IAEA in order to have the Agency involved in assessing the safety of the plant design. Despite these developments, though, there are still several major questions about Russia's FNPP plans that raise concern. Much of the FNPP project remains shrouded in secrecy. Economic calculations, which will surely effect expenditures on safety, security, etc., are unknown – the total cost estimates given vary widely, and do not appear to include security, transport, or back end costs

Assessment of potential consequences of possible radiological accidents in the seas of the Northwest Region of the Russian Federation

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Abstract

There are the sites for the storage and decommissioning of the retired nuclear submarines, their nuclear reactors, spent nuclear fuel and radioactive waste, located in Northwest Russia. Many of those sites are not isolated and protected. There are also nuclear shipbuilding facilities, naval objects and nuclear powered vessels (submarines, icebreakers) in the seas. All these sites present potential threat of radiological releases. This was the reason for implementation of the International Project "Enhancement of Radiation Monitoring and Emergency Response System in the Murmansk Region" (Project), managed by the European Bank for Reconstruction and Development. The Nuclear Safety Institute (NSI) of the Russian Academy of Sciences was the main executor of the Project. Possible scenarios and consequences of accidental radioactive contamination of coastal waters of Northwest Russia are discussed in this work. The scenarios assume the location of the potential accidents in the White Sea and in one of the Bays of the Barents Sea. The modelling of possible consequences of radiological accidents was implemented for water objects that strongly differ in size (from 5 to 300 kilometres), tides, currents and characteristic times of water exchange with the Arctic Ocean. The modelling of migration of the radioactive substances was carried out with the use of computer model developed by the NSI specialists. This model is based on the well-known three-dimensional Princeton Ocean Model.

Introduction

There are a lot of seas, gulfs and bays in the world that have been contaminated by radioactivity or are in danger of such contamination. The nuclear Navy facilities and especially nuclear submarines are the main sources of the danger.

The Soviet Union built about 250 nuclear submarines, more than 30 nuclear maintenance service vessels and coastal maintenance bases. The lifetime of most of the submarines and ships is over. The infrastructure proved to be unprepared for the necessary rate of decommissioning. It has resulted in the fast accumulation of the storage facilities for the retired nuclear submarines with spent nuclear fuel on board. The condition of the hulls of the submarines is worsening, presenting a threat of radioactive contamination of the environment. In Russia the sites for the storage and decommissioning of retired nuclear submarines, their nuclear reactors, spent nuclear fuel and radioactive waste are located at the coasts of the Barents Sea, White Sea, Kamchatka peninsula and at the Russian Far East (Takano et al., 2001; Nikitin et al., 1996; Compton et al., 2003; Vysotsky, 2008).

It is necessary to mention that the hazards are not hypothetical. For example, in 1985 the reloading of nuclear fuel at the nuclear submarine in the Chazhma Bay in the Russian Far East resulted in the accident with discharge of the large amount of radioactivity (Sivintsev, 2000).

Four Russian and two American nuclear submarines sank during combat duty. They lay on the ocean bed throughout the world. One Russian decommissioned submarine sank in the Barents Sea during transportation. Moreover several reactor units of the nuclear submarines were dumped with nuclear fuel on board (Reistad, 2006; IASA, 2003).

There are also regions of ocean that are radioactively pure nowadays, but with intense traffic of nuclear submarines. Underwater collisions of submarines represent another potential threat (IASA, 2003).

Thus, the geography of potential radioactive contamination of the World Ocean is broad enough. Integration of a radioactivity transport model into the Princeton Ocean Model (POM) could give the necessary tool to nuclear safety specialists.

At present in Northwest Russia, especially at the coast of the Barents Sea in the Murmansk Region, there are a lot of sites for the storage and decommissioning of the retired nuclear submarines, their nuclear reactors, spent nuclear fuel and radioactive waste. The highest radiation potential (~45% of total in the region) have nuclear submarines, located at the naval bases in the Murmansk region. The radiation potential of the storage sites in this part of Russia is higher than activity of long-lived radionuclides, released during the Chernobyl accident (Sarkisov, 2009). Some of those sites are not isolated and protected. This was the reason of implementing of the international Project "Enhancement of Radiation Monitoring and Emergency Response System in the Murmansk Region" (Project), managed by the European Bank for Reconstruction and Development (Amozova, 2007). Under this Project two crisis centres were established and equipped by the software necessary for nuclear safety specialists. Within the frame of the Project specialists of Nuclear Safety Institute of the Russian Academy of Sciences implemented initial improvements of the POM code.

Description of the regions of modelling

The Sayda Bay is situated in the Murmansk Region of the Russian Federation. The size of the Bay is about 5.6 km from east to west and 5.1 km from south to north (see Figure 1). The Sayda Bay is a restricted territory and is not used economically by local population. The bay is connected by narrow neck with Kola Bay of Barents Sea, There is no major rivers flow into the Sayda Bay. The tides in this bay are considerable. There is hydro-meteorological station at the Yekaterininskaya Harbor in the Kola Bay near the mouth of the Sayda Bay. The recorded tides are up to the 4 meters.

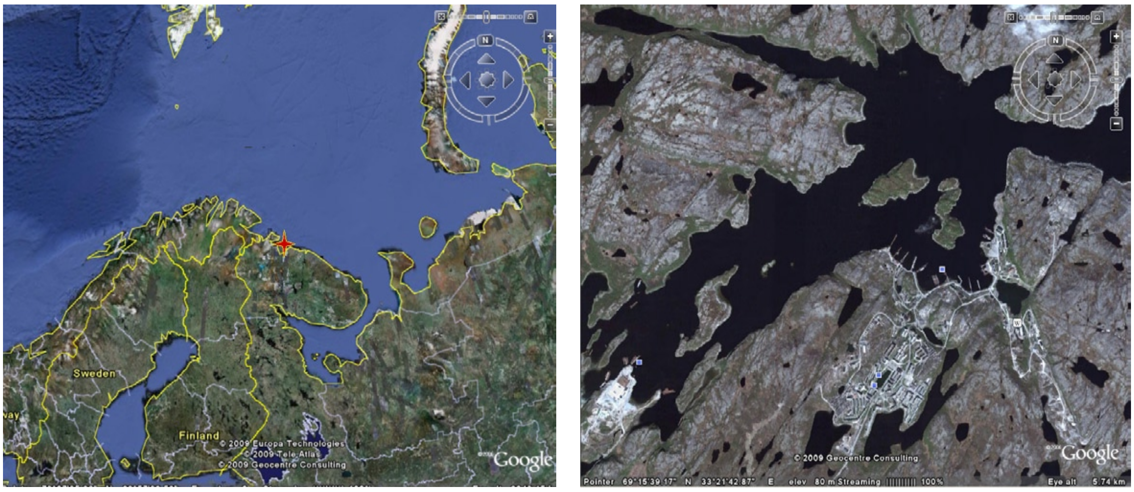

Fig 1. Location (left) and satellite photograph (right) of the Sayda Bay.

White Sea belongs to the internal seas of the Arctic Ocean. In the north it connects to the Barents Sea through the Gorlo and Voronka straits. The area of the sea is $8.7*10^{10}$ m^2, the volume of water - about $6*10^{12}$ m^3, the average water depth - 67m, and the greatest - 350m. (Figure 2, left). Annual runoff averages $2.2*10^{11}$ m^3.

The horizontal circulation of waters of the White Sea is formed under the combined effect of wind, river runoff, tides, compensation flows, so it is diverse and complex in detail. The resulting motion forms counterclockwise movement of waters under the influence of the Coriolis forces, peculiar to the seas of the Northern Hemisphere (Figure 2, right) - (Dobrovolskiy 1982).

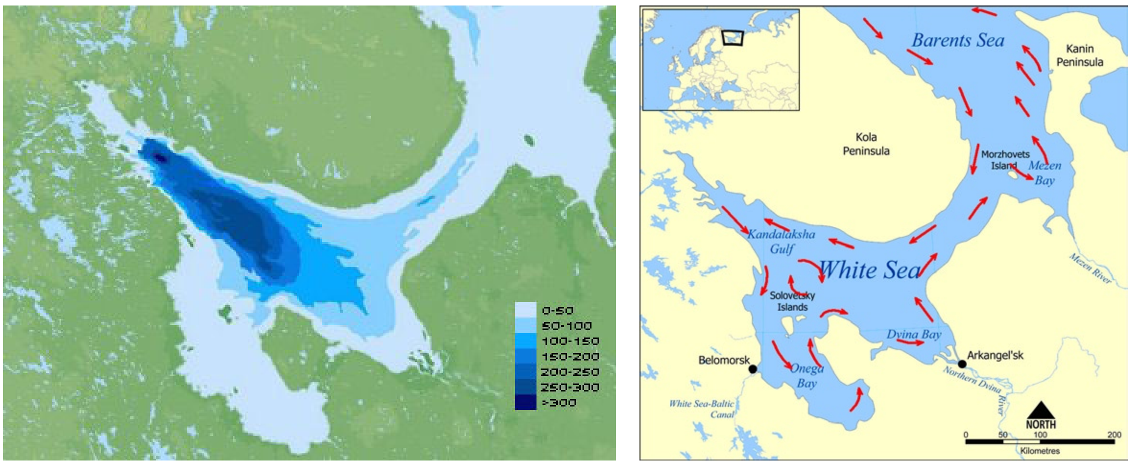

Fig 2. Depths (left) and a constant surface currents (right) in the White Sea.

The velocities of the constant currents are small and usually equal to 0.1 – 0.15 m/s in narrow waters and reach 0.3 – 0.4 m/s around capes. In some areas tidal currents have much higher speeds. In Gorlo strait and Mezenski Bay they reach 2.5 m/s, in Kandalaksha bay - 0.30 – 0.35 m/s, in Onega Bay - 0.8 – 1.0 m/s. In the sea basin velocity of the tidal currents is approximately equal to velocity of the constant currents (Dobrovolskiy 1982).

Improvements of the POM necessary to enable modelling of the radioactivity transport

Radioactive substances in the marine environment are transported in solute and sorbed on particles, first of all, on the finest fractions of the suspended matter. Modelling of radioactivity migration in the solute can be done by analogy with salinity. It is necessary to mention that calculation of salinity as a function of time and space is included into the POM.

In order to model radioactivity sources it is necessary to apply appropriate boundary conditions. The most probable types of radioactive contamination sources of the marine environment are as follows:

- A permanent point source on the seabed. It can include leakage of radioactivity from a sunken submarine, a piece of nuclear fuel or another radioactive fraction of a ship as it was in the Chazhma Bay accident.
- A permanent point source on the coast. It can include leakage from a coastal radioactively dangerous object or the mouth of a contaminated river.
- A short-term or instantaneous point source. It can include accidental radioactivity discharge from a nuclear vessel, submarine or from a coastal facility. It might even be deliberate discharge of liquid radioactive waste into the sea.
- A spatial surface source. Such sources can occur in case of an accident with a radioactivity discharge to the atmosphere and subsequent fallout on the sea surface. The area of fallout can be vast, its duration can be prolonged and the intensity can be extremely variable.

There are no significant difficulties in modelling of permanent sources. Boundary conditions should add the necessary amount of radioactivity each time step to the appropriate cells of the grid. Modelling of instantaneous point sources is also easy. But for this type of sources it is very important to spin up the model before adding the radioactivity. The spatial sources of radioactivity are most difficult for modelling as source characteristics require close interaction of models of contaminant dispersion in the marine environment and atmosphere.

It should be mentioned that POM computer code has no special module for the calculation of radioactive contaminant dispersion. Therefore authors of the paper have decided to use for this purpose seawater salinity forecast approaches used in POM code (Krylov 2009). It should be taken into account that unlike the salinity, the radioactive contaminants can be not always considered conservative. In most cases it is not necessary to take into account radioactive decay for long-lived radioactive substances, like ^{137}Cs or ^{90}Sr, as their half-life time is 30 years. And if the time of simulation is much less the decay can be neglected. But for accidents with a discharge into marine environment radionuclides with the half-life time of several days, like ^{131}I, it is very important to take into account their radioactive decay. For this purpose it is necessary to add the exponential decrease of the amount of radioactivity throughout the calculation grid.

Taking into account daughter radionuclides is much more difficult task. Daughter radionuclide can be much more dangerous than the maternal one. For example ^{241}Pu decays to ^{241}Am which is much more dangerous to the humans and the environment. In most cases daughter radionuclides are taken into account implicitly - dose coefficients of maternal radionuclide include effects of daughter ones. But if migratory characteristics of mother and daughter radionuclides differ significantly it may be necessary to model daughter radionuclides explicitly.

There is the mode splitting mechanism in the POM to use appropriate time steps for fast external gravity waves and slow internal gravity waves (internal and external modes). As the variability of radionuclides concentration fields can differ significantly from variability of fields of currents, salinity, temperature it is reasonable to develop further the mode splitting and to develop the «radioactivity mode» (the analogue of the «internal mode») enabling calculation of transport of radioactivity with a different time step.

To take into account the transport of sorbed radioactive substances it is necessary to take into consideration the processes of particle sedimentation, resuspension and transport of suspended particles and of course processes of sorption and desorption of radioactivity on suspended matter. The coupling of sediment transport model with the POM model was done by Dr. Wang (Wang, 2002; Wang et al., 2005).

Let us assume that transport of radioactivity sorbed on different fractions of suspended matter can be described with the use of one effective fraction of suspended matter with the effective sorption and physical characteristics. The models taking into account several fractions of suspended matter are similar. They are more detailed but more vulnerable to inaccuracies due to uncertainty in the values of characteristics of all of the fractions.

Explicit dynamic modelling of sorption and desorption is the difficult problem because even in case of first-order approximation of the processes there are three parameters (the rates of reversible sorption, irreversible sorption and the rate of desorption) that are unknown functions of the chemical composition of the water and suspended particles. Accurate assessment of the sorption and desorption rates is a complex problem.

Fortunately, in many cases explicit modelling of the sorption and desorption processes can be avoided. Equilibrium between radionuclides sorbed on suspended particles and in solution is reached relatively quickly (from hours to tens of hours) (Ferronski 1977; Venicianov, 1983). The assumption of the equilibrium enables one to model migration of sorbed radioactive substances with the use of distribution factors. Values of the factors can be measured or if it is impossible they can be estimated on the base of recommendations of the International Atomic Energy Agency (IAEA, 2001).

In this case following expressions can be used:

$$C_{sorbed} = K_d C_{dissolved} \quad (1)$$

$$C = S_{eff} C_{sorbed} + C_{dissolved} \quad (2)$$

$$Q = K_d S_{eff} Q_{suspended\_matter} + Q_{dissolved} \quad (3)$$

Where: C – total specific activity of a radionuclide in the water, Bq/m^3; $C_{dissolved}$ – specific activity of the dissolved phase, Bq/m^3; C_{sorbed} – specific activity of the sorbed phase, Bq/kg; Q – total flux of a radionuclide, Bq/s; $Q_{dissolved}$ – flux of a radionuclide in the dissolved state, Bq/s; $Q_{suspended\_matter}$ – effective flux of suspended matter, kg/s; K_d – distribution coefficient, m^3/kg; S_{eff} – effective concentration of the suspended matter, kg/m^3.

Interaction between the radioactivity in the boundary layer and in the seabed can be taken into consideration by the boundary conditions. The principal processes on this border are particle sedimentation and resuspension. Diffusive mass-transfer between seabed and water is usually less significant.

And at last but not least, it is necessary to take into account that many-hour simulation can be unacceptable in case of real accident when decisions are to be made very promptly. Prompt modelling of radioactivity transport in some cases can be achieved by separating calculation of radioactivity transport from calculation of currents and transport of particles. I.e. it is reasonable to develop capability of calculation of radioactivity transport with the use of calculated beforehand varying in time current fields.

The improved computational code was used to model consequences of the hypothetical radioactivity discharge into the Sayda Bay and White Sea. To model tides we have developed the subroutine taking into account 21 fundamental harmonics (Rotter, 2000). It can be used for other sites if parameters for the harmonics are known. The tide model was successfully validated against the data of water levels at the Yekateriniskaya Harbor (69.2000° N, 33.4667° E) and Port Kem (64.9833° N, 34.7833° E), using data provided at http://www.mobilegeographics.com/.

Calculations for the Sayda Bay

Wind and tidal forcing was taken into account. As it was mentioned above, the main objective of modelling the Sayda Bay was to find out possibility of quick transport by sea of significant quantity of radioactivity out from the Sayda Bay in case of an accident at the temporary storage facility of reactor units. Thus the modelling period of several days is most important because this time is necessary to establish full-scale radiation control in the Sayda Bay and taking appropriate population and environmental protecting measures.

The uniform computational grid with 89 steps (63m each) from East to West and 77 steps (66 m each) from South to North was used. In vertical direction non-uniform sigma coordinate computational grid with 7 steps from water surface to bottom was used. Outline of the modelling area is shown at the Figure 3. Southwest part of the area was chosen for more detailed analysis.

The velocity fields and spatial distribution of radioactivity were calculated for a 120 hours period after the accident. The source of radioactivity was considered instantaneous, located near the sea surface and comparable in order of magnitude with the radioactive inventory of a reactor unit of a nuclear submarine.

The sorption and radioactive decay were not taken into account. Neglecting sorption approach will be accurate for radionuclides tending to migrate in solute (strontium, technetium, uranium, iodine, tritium, etc). For radionuclides tending to be sorbed on particles (plutonium, cesium, cobalt, etc) it can lead to somewhat over-dispersion of the radioactivity.

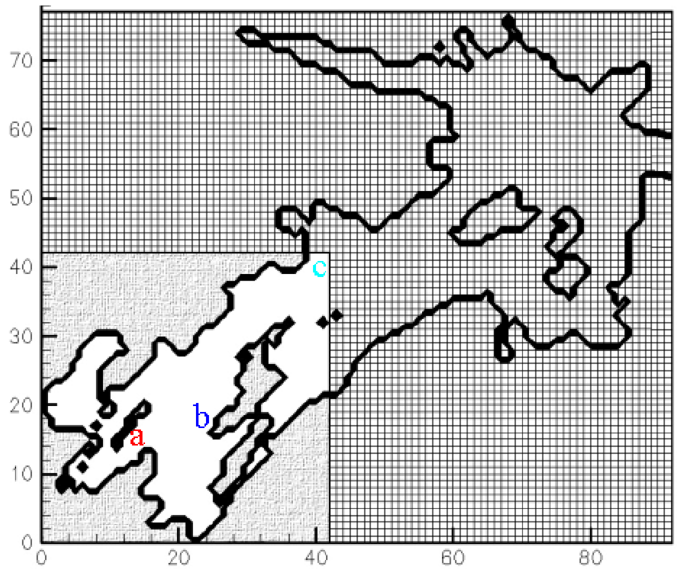

Fig 3. Outline of the modelling area. The red ("a") point indicates the place of the hypothetical accident. The blue ("b") and the pale blue ("c") indicate reference points at the distance of 570 and 2300 meters from the point of accident accordingly.

As it was mentioned above, for instantaneous sources it is important to spin up the model before adding the radioactivity. Concentration of the radioactive substance was calculated for the Southwest part of the Sayda Bay in one and in five days after the hypothetical accident. The calculation results have shown that two and five days of the preliminary spin up gave almost identical results. Therefore two days of spin up are enough for the scale of the Sayda Bay and all subsequent modelling was conducted with two days of preliminary spin-up.

The modelling was conducted for different weather conditions and tide phases in the period of the hypothetical accident. Figure 4 shows that the accident, taking place in the time of low-tide results in much wider dispersion of radioactivity than the one, taking place in the time of high-tide. The influence of wind direction and speed on the processes of radioactivity dispersion in the bay has been estimated. Calculations showed that these factors have significantly less influence than tidal effects.

In addition, we calculated the area of radioactive contamination of the bay with the values of the concentration of radioactive substances exceeding certain prescribed standard. As one of the options of such calculations it was assumed that the total activity of ^{137}Cs in water revenues in the accident was 37 TBq (1 kCi). Further, based on the allowable concentration of this nuclide in foods intended for international trade (1 kBq / kg) (IAEA, 2003) and the factor of accumulation of ^{137}Cs in marine fish in comparison with its content in sea water (100 times) (IAEA, 2001), it was found that within 5 days after the accident, the area of the bay with a water contaminated above allowable level will not exceed 2-5% of its total land area (depending on the time of the accident with regard to tidal currents and surface wind velocity). In addition, this area of contamination would be far from the bay mouth and radioactivity will not be discharged into the open part of the Barents Sea.

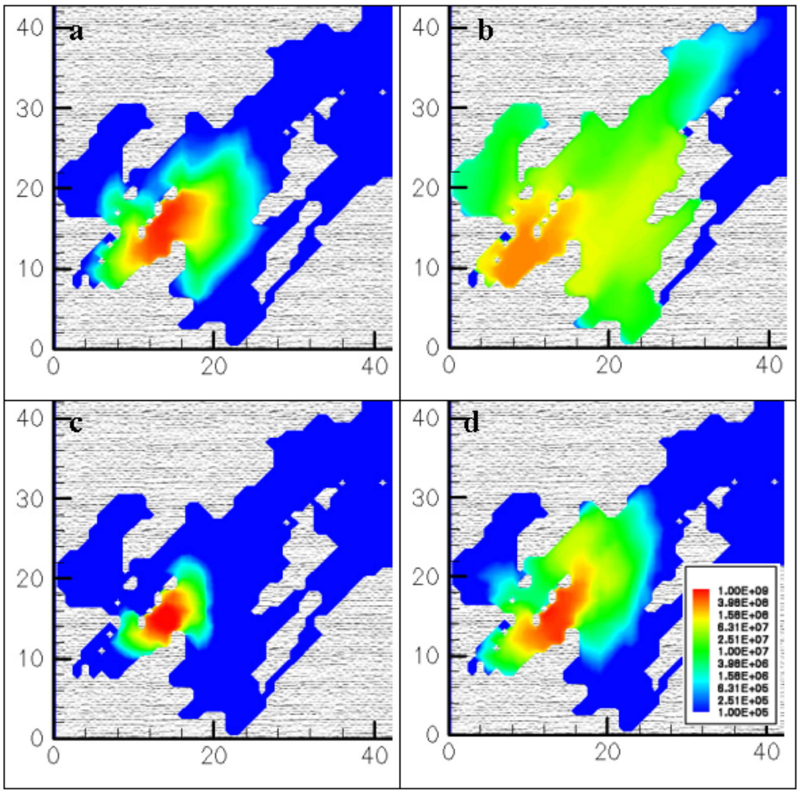

Fig 4. Predicted radioactive pollution of the surface layer depending on the phase of tide at the moment of accident, rel. units. No wind. "a" and "b" – low-tide (0,67 m). "c" and "d" - high-tide (3,56 m). "a" and "c" – 24 hours after the accident. "b" and "d" – 120 hours after the accident.

Calculations for the White Sea

Testing of the calculation module was implemented also for the White Sea. Calculations were performed on two versions of a computational grid with the number of cells 91*93 and 182*186, with the size of cells varied from 3.5 – 5.0 to 1.7 – 2.5 km. With the increase in cell size and the large area of the calculated zone, it was decided to increase also the duration of the pre-calculations, which amounted to 10 days. One of the variants of the calculations is shown in Figure 5. The calculations assumed that the intake of radioactive material occurs in August during the peak of the tidal wave and wind speed above the sea surface amounting to10 m. The wind direction - from the Gorlo strait into the sea. As it is clear from the Figure for this variant of calculation the contamination spot moves along the south-eastern part of Kola Peninsula due to the continuous surface current in this sea area and wind-induced recession. Calculations on the size of contamination zone, with the values of ^{137}Cs concentration in sea water above allowable levels, similar to those described above for the Sayda Bay, showed that within a few days the size of such zone is practically reduced to zero, which is connected with a significantly large scale of actual mixing zone of sea water.

Analysis of simulation of contaminants dispersion in the marine environment by the example of the small sized Sayda Bay showed that tides appear to be the main factor affecting the speed of dispersion processes, especially in the first few days after the accident.

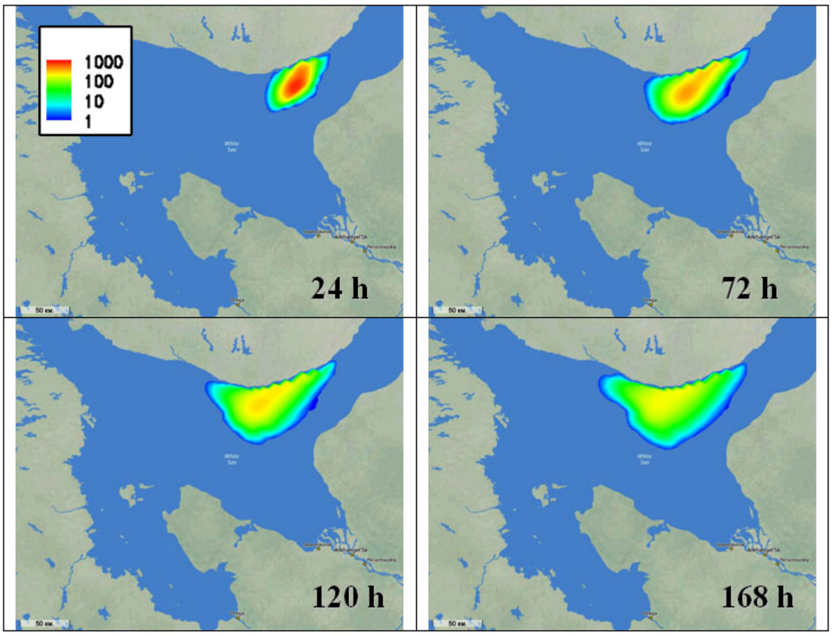

Fig 5. Calculated values of radionuclide concentrations in the surface layer of the White Sea, depending on the time after the emergency discharge, rel. units.

As far as modelling of the hydrological processes in the White Sea are concerned, besides the tide currents the significant factors are the gradients of the temperature and salinity as well as stable currents.

Conclusions

Modelling of the contaminant dispersion processes in the Sayda Bay and water area of the White Sea has shown that the improved computational code gives reasonable results on radioactivity transport in the marine environment. In the present state the model is applicable for periods up to several days for radioactive substances that tend to be sorbed by particles and for longer periods for substances that tend to migrate in the solute.

From the author's point of view the most important tasks for the future are:

- to study the influence of air and water temperatures at various sea depths on the change of rates of contaminant dispersion;
- to implement validation of the hydrodynamic model based on measurements of the sea surface elevation, fields of the currents velocities and salinity in different parts of the White Sea;
- to make possible modelling of radioactive decay and separate migration of maternal and daughter radionuclides if their sorption characteristics differ significantly;
- to enable modelling of radioactivity transport on suspended particles and to enable the use of fields of currents calculated beforehand.

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ICRP *Publication 103* and beyond

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Abstract
This paper focuses on ICRP *Publication 103*, the 2007 Recommendations of the International Commission on Radiological Protection, which lays out the system of radiological protection for all exposure situations and exposure types. In addition, subsequent ICRP publications which delve more deeply into specific aspects of this system are reviewed to some extent, including: ICRP *Publication 104* Scope of Radiological Protection Control Measures; ICRP *Publication 105*, Radiological Protection in Medicine; ICRP *Publication 108*, Environmental Protection — the Concept and Use of Reference Animals and Plants; ICRP *Publication 109*, Application of the Commission's Recommendations for the Protection of People in Emergency Exposure Situations; and an ICRP publication in press titled Application of the Commission's Recommendations to the Protection of People Living in Long Term Contaminated Areas After a Nuclear Accident or a Radiation Emergency.

Introduction
The International Commission on Radiological Protection (ICRP) is an independent, international organization that advances for the public benefit the science of radiological protection, in particular by providing recommendations and guidance on all aspects of protection against ionizing radiation.

ICRP was established in 1928 by the International Society of Radiology (ISR) to respond to growing concerns about the effects of ionizing radiation being observed in the medical community.

In preparing its recommendations, ICRP considers the fundamental principles and quantitative bases upon which appropriate radiation protection measures can be established, while leaving to the various national protection bodies the responsibility of formulating the specific advice, codes of practice, or regulations that are best suited to the needs of their individual countries.

ICRP offers its recommendations to regulatory and advisory agencies and provides advice the intended to be of help to management and professional staff with responsibilities for radiological protection. Although ICRP itself has no formal power to impose its recommendations, in fact legislation in most countries adheres closely to ICRP recommendations. In addition, the International Atomic Energy Agency (IAEA) International Basic Safety Standards for Protection against Ionizing Radiation and for the Safety of Radiation Sources (commonly referred to as "the BSS") is based heavily on ICRP recommendations, and the International Labour Organisation (ILO) Convention 115, Radiation Protection Convention, General Observation 1992, refers specifically to the recommendations of ICRP. Effectively, ICRP recommendations form the basis of radiological protection practice, programmes, regulations, and international standards and guidance worldwide.

ICRP has published well over one hundred publications on all aspects of radiological protection. Most address a particular area within radiological protection, but a handful of publications, the so-called fundamental recommendations, each describe the overall system of radiological protection. The system of radiological protection has been developed by ICRP based on (i) the current understanding of the science of radiation exposures and effects and (ii) value judgements. These value judgements take into account societal expectations, ethics, and experience gained in application of the system.

The 1990 Recommendations of ICRP (ICRP *Publication 60*) form the basis of the current IAEA BSS, and are also the foundation of most radiological protection practices, programmes and regulations worldwide. The 2007 Recommendations of ICRP (ICRP *Publication 103*) recently replaced the 1990 Recommendations. They are the result of nearly a decade of development and several major worldwide public consultations.

The 2007 Recommendations replace the 1990 Recommendations and update, consolidate, and develop additional guidance on the control of exposure from radiation sources issued since 1990. They reflect a more up-to-date understanding of the science behind radiological protection and evolving societal expectations.

The System of Radiological Protection

The system of radiological protection described in the 2007 Recommendations is an evolutionary change from that described in the 1990 Recommendations. This evolution is necessary in order for the system to remain current with our evolving understanding of the relevant scientific findings, and also to continue to reflect current societal norms. For example, the revised radiation and tissue weighting factors reflect updated scientific knowledge, while a greater emphasis on environmental protection reflects a heightened social awareness of the importance of this area. As well, practical application of the system can point out areas for improvement.

What follows is a brief review of the system of radiological protection as described in the 2007 Recommendations and subsequent publications. Much of this is, necessarily, taken almost directly from ICRP *Publication 103*.

Scope

The system of radiological protection applies to all exposures to ionising radiation from any source, regardless of its size and origin. However, the system can apply in its entirety only to situations in which either the source of exposure or the pathways leading to the doses received by individuals can be controlled by some reasonable means. Some exposure situations are excluded from radiological protection legislation, usually on the basis that they are unamenable to control with regulatory instruments, and some exposure situations are exempted from some or all regulatory requirements where such controls are regarded as unwarranted. ICRP *Publication 104* elaborates on the scope of radiological protection control measures.

Health effects of ionising radiation

An understanding of the health effects of ionising radiation is central to the system of radiological protection. Following a review of the biological and epidemiological information on the health risks attributable to ionising radiation ICRP concluded that the distribution of risks to different organs/tissues has changed somewhat since 1990. However, assuming a linear response at low doses, the combined detriment due to excess cancer and heritable effects remains essentially unchanged at around 5% per Sv (see Table 1). Embodied in this current estimate is the use of a dose and dose-rate effectiveness factor for solid cancers which is unchanged at a value of 2. In addition, following prenatal exposure, the cancer risk will be similar to that following irradiation in early childhood, and a threshold dose exists for the induction of malformations and for the expression of severe mental retardation.

The assumption of a linear dose–response relationship for the induction of cancer and heritable effects, according to which an increment in dose induces a proportional increment in risk even at low doses, continues to provide the basis for the summation of doses.

Table 1. Detriment-adjusted nominal risk coefficients (10^{-2} Sv^{-1}) for stochastic effects after exposure to radiation at low dose rate.

Exposed population	Cancer	Heritable effects	Total
Whole	5.5	0.2	5.7
Adult	4.1	0.1	4.2

From ICRP *Publication 103*, Table 1.

Dose limits

The dose limits (see Table 2) remain unchanged from those recommended in 1990. However, it is recognized that further information is needed and revised judgements may be required particularly in respect of the lens of the eye. In addition, emerging data on possible excess risk in non-cancer diseases (e.g., cardiovascular disorders) are being watched closely.

Table 2. Dose limits in planned exposure situations.

Type of Dose Limit	Occupational	Public
Effective dose	20 mSv per year, averaged over defined periods of 5 years, and 50 mSv in any single year[a]	1 mSv in a year
Annual eququivalent dose in: Lens of the eye Skin Hands and feet	 150 mSv 500 mSv 500 mSv	 15 mSv 50 mSv –

[a] Additional restrictions apply to the occupational exposure of pregnant women.
From ICRP *Publication 103*, Table 6.

The use and calculation of equivalent and effective dose

Effective dose is intended for use as a protection quantity. The main uses of effective dose are the prospective dose assessment for planning and optimisation in radiological protection, and demonstration of compliance with dose limits for regulatory purposes.

Although the use of equivalent and effective dose remains unchanged, a number of revisions have been made to the methods used in their calculation. Reviews of the range of available data on the relative biological effectiveness of different radiations, together with biophysical considerations, have led to changes to some of the values of radiation weighting factors (see Table 3 and Figure 1). In addition, the distribution of risks to different organs/tissues has changed somewhat since 1990, particularly in respect of the risks of breast cancer and heritable disease, resulting in revised tissue weighting factors (see Table 4) intended to apply as rounded values to a population of both sexes and all ages.

Another change is that doses from external and internal sources will be calculated using reference computational phantoms of the human body based on CT images (the first appearing in ICRP *Publication 110*), replacing the use of various mathematical models. For adults, equivalent doses will be calculated by sex-averaging of values obtained using male and female phantoms.

Note that calculation of equivalent and effective dose use models and parameters selected from a range of experimental investigations and human studies through judgements, and meant to apply to a population rather than individuals. For individual retrospective dose and risk assessments, individual parameters and uncertainties have to be taken into account.

Collective effective dose is used for optimisation, predominantly in the context of occupational exposure. It is not intended as a tool for epidemiological risk assessment, and it is inappropriate to use it in risk projections. The aggregation of very low individual doses over extended time periods is inappropriate, and in particular, the calculation of the number of cancer deaths based on collective effective doses from trivial individual doses should be avoided.

Table 3. Radiation weighting factors.

Radiation type	Radiation Weighting Factor, w_R
Photons	1
Electrons and muons	1
Protons and charged pions	2
Alpha particles, fission fragments, heavy ions	20
Neutrons	A continuous function of neutron energy (See Figure 1)

From ICRP *Publication 103*, Table 2.

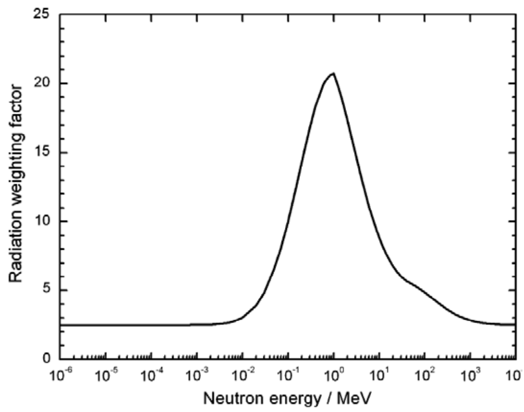

Fig. 1. Radiation weighting factor, w_R, for neutrons versus neutron energy (from ICRP *Publication 103*, Fig.1).

Table 4. Tissue weighting factors.

Tissue	Tissue weigting factor, w_T	$\sum w_T$
Bone-marrow (red), Colon, Lung, Stomach, Breast, Remainder tissues*	0.12	0.72
Gonads	0.08	0.08
Bladder, Oesophagus, Liver, Thyroid	0.04	0.16
Bone surface, Brain, Salivary glands, Skin	0.01	0.04
TOTAL		1.00

* Remainder tissues: Adrenals, Extrathoracic (ET) region, Gall bladder, Heart, Kidneys, Lymphatic nodes, Muscle, Oral mucosa, Pancreas, Prostate (male), Small intestine, Spleen, Thymus, Uterus/cervix (female). From ICRP *Publication 103*, Table 3.

Planned, existing and emergency exposure situations

The system of radiological protection described in the 2007 Recommendations recognises planned, existing and emergency exposure situations. These are intended to cover the entire range of exposure situations, and replace the previous categorisation into practices and interventions.

Planned exposure situations (which include situations previously categorised as practices) encompass sources and situations that have been appropriately managed within the system of radiological protection. Protection during medical uses of radiation is also included in this type of exposure situation. Recommendations for planned exposure situations are substantially unchanged from these provided previously.

Existing exposure situations include naturally occurring exposures as well as exposures from past events and accidents, and practices conducted outside the system of radiological protection. In this type of situation, protection strategies will often be implemented in an interactive, progressive manner over a number of years.

Indoor radon in dwellings and workplaces is an important existing exposure situation, and is the subject of current high-priority work for ICRP. In November 2009 ICRP issued a Statement on Radon. The intention is to publish this Statement in the Annals of the ICRP with an accompanying report on assessment of lung cancer risk from radon, after undertaking a public consultation on the two documents together.

The statement reaffirms that, for planned exposure situations, any workers' exposure to radon incurred as a result of their work, however small, shall be considered as occupational exposure. However, because of the ubiquity of radiation, the direct application of this definition to radiation would mean that all workers should be subject to a regime of radiological protection. Therefore the use of 'occupational exposures' is limited to radiation exposures incurred at work as a result of situations that can reasonably be regarded as being the responsibility of the operating management.

Emergency exposure situations include consideration of emergency preparedness and emergency response. Emergency preparedness should include planning for the implementation of optimised protection strategies which have the purpose of reducing exposures, should the emergency occur, to below the selected value of the reference level. During emergency response, the reference level would act as a benchmark for evaluating the effectiveness of protective actions and as one input into the need for establishing further actions.

Occupational, public and medical exposures

The system of radiological protection continues to distinguish among three categories of exposure: occupational, public, and medical. Table 5 shows these three categories of exposure, the three exposure situations, and the dose constraints and reference levels applicable in each circumstance.

Table 5. Dose constraints and reference levels used in the system of radiological protection.

Type of situation	Occupational exposure	Public Exposure	Medical Exposure
Planned exposure	Dose limit Dose constraint	Dose limit Dose constraint	Diagnostic reference level (Dose constraint[b])
Emergency exposure	Refrerence level	Reference level	n/a
Existing exposure	n/a[a]	Reference level	n/a

[a] Exposures resulting from long-term remediation operations or from protracted employment in affected areas should be treated as part of planned occupational exposure, even though the source of radiation is 'existing'.
[b] Comforters, carers, and volunteers in research only.
From ICRP *Publication 103*, Table 4.

Fundamental principles of radiological protection

The three key principles of radiological protection have remained unchanged in the 2007 Recommendations. The principles of justification and optimisation apply in all exposure situations. The principle of application of dose limits applies only for doses expected to be incurred with certainty as a result of planned exposure situations. These principles are defined as follows:

- The Principle of Justification: Any decision that alters the radiation exposure situation should do more good than harm.
- The Principle of Optimisation of Protection: The likelihood of incurring exposure the number of people exposed, and the magnitude of their individual doses should all be kept as low as reasonably achievable, taking into account economic and societal factors.
- The Principle of Application of Dose Limits: The total dose to any individual from regulated sources in planned exposure situations other than medical exposure of patients should not exceed the appropriate limits specified by the Commission.

Emphasis on optimisation

The 2007 Recommendations emphasise the key role of the principle of optimisation. This principle should be applied in the same manner in all exposure situations. Restrictions are applied to doses to a nominal individual (the Reference Person), namely dose constraints for planned exposure situations and reference levels for emergency and existing exposure situations. Options resulting in doses greater in magnitude than such restrictions should be rejected at the planning stage. Importantly, these restrictions on doses are applied prospectively, as with optimisation as a whole. If, following the implementation of an optimised protection strategy, it is subsequently shown that the value of the constraint or reference level is exceeded, the reasons should be investigated but this fact alone should not necessarily prompt regulatory action. This emphasis on a common approach to radiological protection in all exposure situations should aid application of the system of radiological protection in the various circumstances of radiation exposure.

Table 6. Framework for source-related dose constraints and reference levels with examples of constraints for workers and the public from single dominant sources for all exposure situations that can be controlled.

Bands of constraints and reference levels[a] (mSv)	Characteristics of the exposure situation	Radiological protection requirements	Examples
Greater than 20 to 100	Individuals exposed by sources that are not controllable, or where actions to reduce doses would be disproportionately disruptive. Exposures are usually controlled by action on the exposure pathways.	Consideration should be given to reducing doses. Increasing efforts should be made to reduce doses as they approach 100 mSv. Individuals should receive information on radiation risk and on the actions to reduce doses. Assessment of individual doses should be undertaken.	Reference level set for the highest planned residual dose from a radiological emergency.
Greater than 1 to 20	Individuals will usually receive benefit from the exposure situation but not necessarily from the exposure itself. Exposures may be controlled at source or, alternatively, by action in the exposure pathways.	Where possible, general information should be made available to enable individuals to reduce their doses. For planned situations, individual assessment of exposure and training should take place.	Constraints set for occupational exposure in planned situations. Constraints set for comforters and carers of patients treated with radiopharmaceuticals. Reference level for the Highest planned residual dose from radon in dwellings.
1 or less	Individuals are exposed to a source that gives them little or no individual benefit but benefits to society in general. Exposures are usually controlled by action taken directly on the source for which radiological protection requirements can be planned in advance.	General information on the level of exposure should be made available. Periodic checks should be made on the exposure pathways as to the level of exposure.	Constraints set for public exposure in planned situations.

[a] Acute or annual dose.
From ICRP *Publication 103*, Table 5.

The relevant national authorities will often play a major role in selecting values for dose constraints and reference levels. Guidance on the selection process is provided by recommending bands of constraints and reference levels (see Table 6).

Emphasis on optimisation using reference levels in emergency and existing exposure situations focuses attention on the residual level of dose remaining after implementation of protection strategies. This residual dose should be below the reference level, which represents the total residual dose as a result of an emergency, or in an existing situation, that the regulator would plan not to exceed. These exposure situations often involve multiple exposure pathways so protection strategies involving a number of different protective actions will have to be considered. However, the process of optimisation will continue to use the dose averted by specific countermeasures as an important input into the development of optimised strategies.

Radiological protection of the environment

In the past, ICRP concerned itself with the environment primarily with regard to the transfer of radionuclides through it because this directly affects the radiological protection of human beings. In the 1990 Recommendations ICRP stated that "the standard of environmental control needed to protect man ... will ensure that other species are not put at risk. Occasionally, individual members of non-human species might be harmed, but not to the extent of endangering whole species or creating imbalance between species."

However, in its 2007 Recommendations, ICRP acknowledges that that it is now necessary to demonstrate, directly and explicitly, that the environment is being protected. Therefore, it is necessary to develop a clearer framework to assess the relationships between exposure and dose, and between dose and effect, and the consequences of such effects, for non-human species, on a common scientific basis. This is further developed in ICRP *Publication 108*.

Recent ICRP Publications which Further Elaborate the System of Radiological Protection

ICRP *Publication 104*: Scope of Radiological Protection Control Measures

In principle the system of radiological protection applies to all exposures to ionizing radiation. However, in practice the measures undertaken to control these exposures must be limited based on practical considerations. ICRP *Publication 104* discusses the scope of radiological protection control measures, and describes certain tools (e.g. exemption, exclusion and clearance) that can be used to manage the scope.

ICRP *Publication 108*: Environmental Protection – the Concept and Use of Reference Animals and Plants

The 2007 Recommendations acknowledge that that it is now necessary to demonstrate, directly and explicitly, that the environment is being protected, rather than simply assume that adequate protection of humans is sufficient to protect the environment. Therefore, it is necessary to develop a clearer framework to assess the relationships between exposure and dose, and between dose and effect, and the consequences of such effects, for non-human species, on a common scientific basis.

To this end, ICRP *Publication 108* sets out some high-level ambitions with regard to environmental protection. To aid in demonstrating whether these ambitions are being achieved, and help optimise the level of effort that might be expended on environmental protection, and ICRP has developed a set of Reference Animals and Plants (RAPs) and derived consideration reference levels (DCRLs). ICRP does not propose the application of dose limits to Reference Animals and Plants.

Recent ICRP Publications Elaborating on Specific Exposure Categories and Situations

One of the first publications following the 2007 Recommendations was ICRP *Publication 105* on Radiological Protection in Medicine. This publication describes the application of the system of radiological protection for medical exposures, an exposure category quite different from occupational and public exposures by virtue of the fact that most of the time, most of the benefit and the detriment apply to a single individual: the patient. In the other two exposure categories, typically one group receives most of the detriment (e.g. the workers in a facility), while another group receives most of the benefit (e.g. the public at large receiving electrical power).

Two publications each examine one of the three exposure situations: ICRP *Publication 109* examines emergency exposure situations, and the ICRP publication in press titled Application of the Commission's Recommendations to the Protection of People Living in Long Term Contaminated Areas after a Nuclear Accident or a Radiation Emergency examines the system of radiological protection as applied to existing exposure situations.

Other Current Initiatives and Future Work

For more information on current and future work of ICRP the reader is referred to another paper prepared for this congress by the same author titled “International Commission on Radiological Protection – recent publications, current initiatives and future work”.

Acknowledgements

Preparation of this paper drew extensively on the hard work of ICRP members through several ICRP publications shown in the references section. The author would like to acknowledge the dedication of those ICRP members who contributed to these publications, and thus recognize their important contribution to this paper.

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Radiation protection metrology and measurements

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Abstract
Quantitative assessments of external and internal exposure need reasonable state-of-the-art physical / biological / ecological models and reliable quantities and measurement methods and instruments. Fundamental basis for dose and activity measurements are solid instruments and adequate measurement methods.

In this paper, the appropriate quantities and units as far as necessary measurement chains to assure traceability from international primary standards to end-user measurement facilities are given. The basic concepts and practical implementation of uncertainty assessments in activity and dose measurements methods are presented. Finally main aspects of quality assurance in the field of radiation protection measurements are discussed.

The target group for this information are newcomers as far as experienced radiation protection experts, radionuclide and dosimetry metrologists and end-user of measurement instruments in radiation protection.

Introduction
Accurate and high-quality measurements of ionizing radiation and radionuclide activities are required in a wide range of industrial, scientific and medical applications where they are critical relating to human health and safety. Dose and activity measurements for radiation protection are stringent in its accuracy and reliability requirements. Generally this means that the uncertainty of measurements for radiation protection around the internationally agreed dose limits should not exceed a few percent.

To provide an international metrological basis for measurements in radiation protection and other sectors (e.g. medical diagnostics and therapy), the 11th *General Conference of the Meter Convention* decided to establish the Ionizing Radiation section at the *Bureau International des Poids et Mesures*, BIPM, in 1960.

The main activities of the BIPM in the field of ionizing radiation are to maintain the international reference standards for dose and activity measurements [1]. These standards are used in the BIPM key comparisons and their development and improvement is a major part of the international metrological research and development programme. The ionising radiation section of the BIPM also undertakes calibrations for national laboratories, and participates in international comparisons.

At the European level *European Association of National Metrology Institutes* EURAMET is acting as the regional metrolo¬gical organisation. It is coordinating the metrological activities of the European National Metrology Institutes (NMI's) and Designated Institutes (DI's) of the European Union including the Joint Research Centres of the European Commission, EFTA and EU Accession States. The objective of EURAMET is to promote the coordination and development of metrological activities and services and support the European Metrology Research Programme EMRP. EURAMET is working in Technical Committees; on of them is the TC Ionising Radiation. This TC is divided the Sub-Fields photon dosimetry, radioactivity, and neutron measurements. The basic activities are jointly done by co-operative scientific and technical projects e.g. joint research projects and comparisons.

In Fig. 1 the international, European and national (for example the Austrian) organisations supporting the quality of measurements for radiation protection are schematically shown. In addition to the metrological infrastructure there is a support in legal metrology (International Organization of Legal Metrology OIML; European Cooperation in Legal Metrology WELMEC; national/Austria: Federal Office for Metrology and Surveying BEV) and for accreditation (International Laboratory Accreditation Cooperation ILAC; European co-operation for Accreditation EA; National/Austria: Federal Ministry of Economy, Family and Youth, BMWFJ) established.

	International	Europe	National e.g. Austria
Metrology	BIPM	EURAMET European Association of National Metrology Institutes	Willkommen im BEV
Legal Metrology	OIML	European Cooperation in Legal Metrology WELMEC	Willkommen im BEV
Accreditation	ilac	EA European co-operation for Accreditation	bmwfj Federal Ministry of Economy, Family and Youth

Fig. 1. International, European and national metrological infrastructure.

Fig. 2 shows the international organisations and network between society's and economics' needs together with science and technological support for the provision of quality in measurements. Additional to ILAC the *International Accreditation Forum, Inc. (IAF)* acts as the world association of Conformity Assessment Accreditation Bodies and other bodies interested in conformity. In the field of technological standardisation the *International Electrotechnical Commission* IEC (Technical Committee 45 *Nuclear* *instrumentation*, Sub Committee 45B *Radiation protection instrumentation*), the *International Organisation for Standardisation* ISO (Technical Committee 85 *Nuclear Energy*, Sub Committee 2 *Radiation Protection*) and the *Telecommunication Standardization Sector of the International Telecommunication Union* ITU-T (for non-ionising radiation) support the development and implementation of measurement radiation protection instruments and methods.

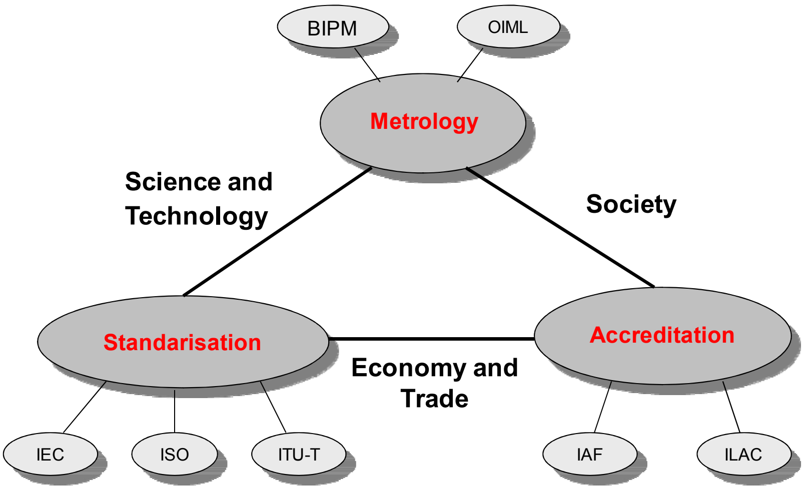

Fig. 2. International organisations and network to ensure quality in measurements.

Basic metrological items and its definitions

In this section the most significant basic definitions of items for metrology and measurement are given. The items / definitions are largely compatible to them given in the International Vocabulary in Metrology (ISO/IEC, 2007; VIM, 2008).

Quantity

… Property of a phenomenon, body, or substance, where the property has a magnitude that can be expressed as a number and a reference.

A reference can be a measurement unit, a measurement procedure, a reference material, or a combination of such. Symbols for quantities are given in ISO 80000 and IEC 80000 series Quantities and Units.

Measurement unit; unit

… Real scalar quantity, defined and adopted by convention, with which any other quantity of the same kind can be compared to express the ratio of the two quantities as a number.

Measurement units are designated by conventionally assigned names and symbols. In some cases special measurement unit names are restricted to be used with quantities of a specific kind only. For example, the measurement unit ‘second to the power minus one’ (1/s) is called hertz (Hz) when used for frequencies and becquerel (Bq) when used for activities of radionuclides.

Metrological traceability; traceability

… Property of a measurement result whereby the result can be related to a reference through a documented unbroken chain of calibrations, each contributing to the measurement uncertainty.

For this definition, a 'reference' can be a definition of a measurement unit through its practical realization, or a measurement procedure including the measurement unit for a non-ordinal quantity, or a measurement standard. The abbreviated term "traceability" is sometimes used to mean 'metrological traceability'. The full term of "metrological traceability" is preferred if there is any risk of confusion.

Metrological traceability chain; traceability chain

… Sequence of measurement standards and calibrations that is used to relate a measurement result to a reference.

A metrological traceability chain is defined through a calibration hierarchy.

Metrological traceability to a measurement unit; metrological traceability to a unit

… Metrological traceability where the reference is the definition of a measurement unit through its practical realization.

The expression "traceability to the SI" means 'metrological traceability to a measurement unit of the International System of Units'.

Calibration

… Operation that, under specified conditions, in a first step, establishes a relation between the quantity values with measurement uncertainties provided by measurement standards and corresponding indications with associated measurement uncertainties and, in a second step, uses this information to establish a relation for obtaining a measurement result from an indication.

A calibration may be expressed by a statement, calibration function, calibration diagram, calibration curve, or calibration table. In some cases, it may consist of an additive or multiplicative correction of the indication with associated measurement uncertainty. Calibration should not be confused with adjustment of a measuring system, often mistakenly called "self-calibration", nor with verification of calibration. Often, the first step alone in the above definition is perceived as being calibration.

Calibration hierarchy

… Sequence of calibrations from a reference to the final measuring system, where the outcome of each calibration depends on the outcome of the previous calibration.

Measurement uncertainty necessarily increases along the sequence of calibrations. The elements of a calibration hierarchy are one or more measurement standards and measuring systems operated according to measurement procedures. For this definition, the 'reference' can be a definition of a measurement unit through its practical realization, or a measurement procedure, or a measurement standard.

Measurement standard; etalon

… Realization of the definition of a given quantity, with stated quantity value and associated measurement uncertainty, used as a reference

A "realization of the definition of a given quantity" can be provided by a measuring system, a material measure, or a reference material. A measurement standard is frequently used as a reference in establishing measured quantity values and associated measurement uncertainties for other quantities of the same kind, thereby establishing metrological traceability through calibration of other measurement standards, measuring instruments, or measuring systems.

A primary measurement standard is a measurement standard established using a primary reference measurement procedure, or created as an artifact, chosen by convention (eg. Kilogramm artefact) than again a secondary measurement standard is established through calibration with respect to a primary measurement standard for a quantity of the same kind.

Measurement uncertainty; uncertainty

… Non-negative parameter characterizing the dispersion of the quantity values being attributed to a measurand, based on the information used.

Notes:

Measurement uncertainty includes components arising from systematic effects, such as components associated with corrections and the assigned quantity values of measurement standards, as well as the definitional uncertainty. Sometimes estimated systematic effects are not corrected for but, instead, associated measurement uncertainty components are incorporated. The parameter may be, for example, a standard deviation called standard measurement uncertainty (or a specified multiple of it), or the half-width of an interval, having a stated coverage probability.

Measurement uncertainty comprises, in general, many components. Some of these may be evaluated by Type A evaluation of measurement uncertainty from the statistical distribution of the quantity values from series of measurements and can be characterized by standard deviations. The other components, which may be evaluated by Type B evaluation of measurement uncertainty, can also be characterized by standard deviations, evaluated from probability density functions based on experience or other information (ISO/IEC Guide 98-3:2008).

Does quantities and their units

In this section the most significant quantities and units used for radiation protection purposes are assembled. The given items and their definitions are compiled from current international standards and documents (ICRU, 1985; 1998; IEC 60050-393, 2003; IAEA, 2007; ICRP 103, 2007; ISO 12749-1, 2010).

Absorbed dose, *D*

… Quotient of $\mathrm{d}\bar{\varepsilon}$ by d*m*, where $\mathrm{d}\bar{\varepsilon}$ is the mean energy imparted to matter of mass d*m* thus

$$D = \frac{\mathrm{d}\bar{\varepsilon}}{\mathrm{d}m} \tag{1}$$

The unit of absorbed dose is J·kg^{-1}. The special name for the unit of absorbed dose is gray (Gy). The mean energy imparted $d\bar{\varepsilon}$ is the expectation value of the energy imparted by ionizing radiation to the matter in a given volume. The unit of the mean energy imparted is J.

dose equivalent, *H*

… Product of *D* and *Q* at a point in tissue, where *D* is the absorbed dose and *Q* is the quality factor for the specific radiation at this point, thus

$$H = D \cdot Q \tag{2}$$

The unit of dose equivalent is joule per kilogram (J kg^{-1}), and its special name is sievert (Sv).

Ambient dose equivalent, *H**(*d*)

… Dose equivalent that would be produced by the corresponding aligned and expanded field in the ICRU sphere at a depth *d* on the radius opposing the direction of the aligned field.

This quantity is defined at a point in a radiation field and is used as directly measurable proxy (i.e. substitute) for effective dose for use in monitoring of external exposure. The recommended value of d for strongly penetrating radiation is 10 mm.

The ICRU sphere is a sphere with 30 cm diameter consists of ICRU tissue. This is a material with a density of 1 g cm^{-3} and a mass composition of 76,2 % oxygen, 10,1 % hydrogen, 11,1 % carbon, and 2,6 % nitrogen (ICRU 39, 1985)

Directional dose equivalent, $H'(d, \vec{\Omega})$

… Dose equivalent that would be produced by the corresponding expanded field in the ICRU sphere at a depth *d* on a radius in a specified direction $\vec{\Omega}$.

This quantity is defined at a point in a radiation field and is used as directly measurable proxy (i.e. substitute) for equivalent dose in the skin for use in monitoring of external exposure. The recommended value of d for weakly penetrating radiation is 0,07 mm. In practice $\vec{\Omega}$ is chosen in direction of maximum *H*', indicated as *H*'(*d*).

Personal dose equivalent, H_p (*d*)

… Dose equivalent in the 'ICRU sphere' at an appropriate depth, d, below a specified point on the human body.

The unit of personal dose equivalent is joule per kilogram (J·kg^{-1}) and its special name is sievert (Sv). The specified point is usually given by the position where the individual's dosimeter is worn. In most cases of penetrating radiation d = 10 mm is chosen: $H_p(10)$. For weakly penetrating radiation d = 0,07 mm is chosen: $H_p(0{,}07)$. In a unidirectional field, the direction can be specified in terms of the angle, α, between the direction opposing the incident field and a specified normal on the phantom surface.

Effective dose

… Sum of the weighted equivalent doses in all tissues and organs of the body.

The effective dose is expressed in units of joules per kilogram (J·kg^{-1}) with the special name sievert (Sv). The effective dose is not directly measurable. It can be approximated by operational dose equivalent quantities for external exposure and by dose conversion models (dose conversion factors) in the case of intake (inhalation, ingestion) of radionuclides. The effective dose is used in dose limitation / restriction concepts in radiation protection. Effective dose is a measure of dose designed to reflect the amount of radiation detriment likely to result from the dose. Values of effective dose from any type(s) of radiation and mode(s) of exposure can be compared directly.

Activity quantities and their units

In this section the most significant quantities and units used for radiation protection purposes are assembled. The given items and their definitions are compiled from current international standards and documents.

Activity

… Quotient, for an amount of radionuclide in a particular energy state at a given time, of $\langle \mathrm{d}N \rangle$ by d*t*, where $\langle \mathrm{d}N \rangle$ is the expectation value of the number of spontaneous nuclear transitions from this energy state in the time interval of duration d*t*:

$$A = -\frac{\langle \mathrm{d}N \rangle}{\mathrm{d}t} \tag{3}$$

The unit of activity is s^{-1} with the special name *becquerel* (Bq).

Activity concentration;

… Quotient of activity by either the total mass or the total volume of the sample.

This quantity is expressed either in becquerels per kilogram (Bq/kg) or in becquerels per cubic-meter (Bq/m³). For a gas, in some cases it is necessary to indicate the temperature and pressure conditions for which the activity concentration, expressed in becquerel per cubic metre, is measured, for example standard temperature and pressure (STP). For solid material, sometimes it is necessary to indicate the sample moisture or dryness conditions for which the activity concentration, expressed in becquerel per kilogram, is measured, for example Bq/kg material dried at 105 °C.

Specific activity; massic activity

… Quotient of activity by the total mass of the sample.

This quantity is expressed either in becquerels per kilogram (Bq/kg).

The distinction in usage between specific activity and activity concentration is controversial. A common distinction is that specific activity is used with reference to a pure radionuclide (e.g. 1 g ^{226}Ra has the specific activity of 37·10^{9} Bq). Activity concentration (which may be activity per unit mass or per unit volume) is used for any other situation (e.g. when the activity is in the form of contamination in or on a material).

Volumic activity; volumetric activity

… Quotient of the activity by the total volume of the sample.

This quantity is expressed in becquerels per cubic metre (Bq/m^3).

Surface activity

… Quotient of the activity by the total area of surface of the sample.

This quantity is expressed in becquerels per square metre (Bq/m^2).

Measurement uncertainties in radiation protection

In most cases, a measurand Y is not measured directly, but is determined from N other quantities $X_1, X_2, ..., X_N$ through a functional relationship f:

$$Y = f\left(X_1, X_2, ..., X_N\right) \tag{4}$$

An estimate of the measurand Y, denoted by $\bar{y}$, is obtained from Y using input estimates $\bar{x}_1, \bar{x}_2, ..., \bar{x}_N$ for the values of the N quantities $X_1, X_2, ..., X_N$. Thus the output estimate $\bar{y}$, which is the result of the measurement, is given by

$$\bar{y} = f\left(\bar{x}_1, \bar{x}_2, ..., \bar{x}_N\right) \tag{5}$$

When the input quantities are not correlated, the combined standard uncertainty $u_c(\bar{y})$ is the positive square root of the combined variance $u_c^2(\bar{y})$, which is given by

$$u_C(\bar{y}) = \sqrt{\sum_i \left(\frac{\partial f\left(\bar{x}_1, \bar{x}_2, ...\bar{x}_N\right)}{\partial X_i}\right)^2 \cdot u^2\left(\bar{x}_i\right)} \tag{6}$$

When the input quantities are correlated, the appropriate expression for the combined variance $u_c^2(\bar{y})$ associated with the result of a measurement is

$$u_c^2(\bar{y}) = \sum_i \sum_j \frac{\partial f}{\partial X_i} \frac{\partial f}{\partial X_j} \cdot u\left(\bar{x}_i, \bar{x}_j\right) = \sum_i \left(\frac{\partial f}{\partial X_i}\right)^2 u\left(\bar{x}_i\right) + 2\sum_{i=1}^{N-1} \sum_{j=i+1}^{N} \frac{\partial f}{\partial X_i} \frac{\partial f}{\partial X_j} \cdot u\left(\bar{x}_i, \bar{x}_j\right) \tag{7}$$

where $u\left(\bar{x}_i, \bar{x}_j\right) = u\left(\bar{x}_j, \bar{x}_i\right)$ is the estimated covariance associated with $\bar{x}_i$ and $\bar{x}_j$. The degree of correlation between $\bar{x}_i$ and $\bar{x}_j$ is characterized by the estimated correlation coefficient

$$r\left(\bar{x}_i, \bar{x}_j\right) = \frac{u\left(\bar{x}_i, \bar{x}_j\right)}{u\left(\bar{x}_i\right) u\left(\bar{x}_j\right)}$$

where $r\left(\bar{x}_i, \bar{x}_j\right) = r\left(\bar{x}_j, \bar{x}_i\right)$, and $-1 \leq r\left(\bar{x}_i, \bar{x}_j\right) \leq +1$. If the estimates $\bar{x}_i$ and $\bar{x}_j$ are independent, $r\left(\bar{x}_i, \bar{x}_j\right) = 0$, and a change in one does not imply an expected change in the other.

Equation (6) and its counterpart for correlated input quantities, Equation (7), both of which are based on a first-order Taylor series approximation of the measurand Y, express what is termed the law of propagation of uncertainty.

In most cases, the best available estimate of the expectation value $\langle X_i \rangle$ of an input quantity X_i that varies randomly, and for which independent observations have been obtained under the same conditions of measurement, is the arithmetic mean or average $\langle x_i \rangle$ of the observations. Thus, for an input quantity X_i determined from independent repeated observations, the standard uncertainty $u(\bar{x}_i)$ of its best estimate $\langle x_i \rangle$ is $u(\bar{x}_i) = s(\bar{x}_i)$, with $s^2\langle \bar{x}_i \rangle$ is the experimental variance or the estimate of the variance $\sigma^2\langle X_i \rangle$. For convenience, $u^2(\bar{x}_i) = s^2\langle \bar{x}_i \rangle$ and $u(\bar{x}_i) = s\langle \bar{x}_i \rangle$ are sometimes called a Type A variance and a Type A standard uncertainty, respectively.

In Tab. 1 a worksheet for the practical implementation of the measurand's uncertainty estimation from uncorrelated (independent) input quantities based on formula (4) to (6) is given.

Table 1. Worksheet for the estimation of the expanded uncertainty *U* of the output quantity (measurand) *Y* from estimates of independent uncorrelated input quantities X_i.

input quantity	estimate of input quantity	evaluation type	sensitivity coefficient c_i	standard uncertainty	square products
X_i	$\bar{x}_i$	A or B	$\dfrac{\partial f(\bar{x}_1, \bar{x}_2, \bar{x}_3, \ldots)}{\partial X_i}$	$u(\bar{x}_i)$	$\left(\dfrac{\partial f(\bar{x}_1, \bar{x}_2, \bar{x}_3, \ldots)}{\partial X_i}\right)^2 \cdot u^2(\bar{x}_i)$
output quantity / measurand	output estimate of measurand		sum of square products: $u_\mathrm{C}^2(\bar{y})$		
			square root of the sum: $u_\mathrm{C}(\bar{y})$		
Y	$\bar{y} = f(\bar{x}_1, \bar{x}_2, \bar{x}_3, \ldots)$		coverage factor k:		
			expanded uncertainty: $U(\bar{y}) = k \cdot u_\mathrm{C}(\bar{y})$		

Quality management of radiation protection measurements

The state-of-the-art main source for laboratory quality management is the European standard EN ISO/IEC 17025:2005. In this standard the main conformity requirements of an appropriate QM system for testing and calibration laboratories are given:

- **Management requirements:** organisation, management system, document control, review of requests, tenders and contracts, subcontracting of tests and calibrations, purchasing services and suppliers, services to the customer, complaints, control of nonconforming testing and/or calibration work, improvements, corrective action, preventive action, control of records, internal audits, and management reviews.

- **Technical requirements:** general, personnel, accommodation and environmental conditions, test and calibration method validation, equipment, measurement traceability, sampling, handling of tests and calibration items, assuring the quality of tests and calibration results, reporting the results.

Especially for measurements in radiation protection and radiation safety the required measurement traceability is explained in IAEA (2008):

"7.28. To be sure that the measurement results will comply with international standards, each measurement device that has an influence on the results should be calibrated before being put into service and at defined intervals afterwards. The standards used for these calibrations should be traceable to the International System of Units (SI). In some cases — e.g. in connection with ^{222}Rn — the only means of providing confidence in measurements is through participation in suitable international intercomparison exercises.

7.29. Calibration services have to trace their standards and measuring instruments to the SI System by means of an unbroken chain of calibrations or comparisons linking them to relevant primary standards for the SI units of measurement. For measurement services, this traceability can be achieved by using a calibration service."

This metrological requirement must be strictly implemented in the QM system and assured in the day-to-day laboratory practice and documentation.

Conclusions

Measurements in radiation protection and radiation safety for the assessment of external and internal exposure need reliable measurement instruments and methods. To ensure high-quality measurements, an appropriate quality management system has to be implemented and followed in the day-to-day practice. Careful attention has to be paid to the use of appropriate quantities, the reasonable estimation of measurement uncertainties and the metrological traceability of the measurement instruments.

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External dosimetry and individual monitoring

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Abstract
Individual monitoring is required by international regulations to demonstrate the compliance of dose limits. This paper gives an introduction which types of personal dosemeters are in use and how these dosemeters are calibrated in the operational dose quantities $H_p(10)$ and $H_p(0.07)$. In addition type test requirements for different dosemeters stated in recent international standards (IEC) are summarised and compared.

General principles and requirements
The main principles for the radiation protection system are the following three issues:

- **principle of justification**
- **principle of optimization of protection**
- **principle of application of dose limits**

Based on the last principle monitoring of the individual exposure of persons constitutes an integral part of any radiation protection programme. Especially the principle of dose limitation requires reliable assessment of individual doses.

Legal requirements and international recommendations
The fundamental principles for the operational protection of occupationally exposed persons, apprentices and students are laid down in EU Council Directive 96/29/Euratom Basic Safety Standards (BSS) [(2)]. The requirements for dose assessments and the corresponding dose limits are mandatory specified in these BSS based on international recommendations by the International Commission on Radiological Protection (ICRP) and reports of the International Commission on Radiation Units and Measurements (ICRU).

Dose limits and dose quantities
In these regulations dose limits, dose constraints, and reference levels for different situations are stated. In the BSS, the requirements for dose assessment and for dose limits are given in terms of the protection dose quantities, effective dose, equivalent dose to the lens of the eye, equivalent dose to the extremities and equivalent dose to local skin. These protection quantities are difficult to assess and impossible to measure directly. The BSS state that operational dose quantities for external radiation, personal

dose equivalent for personal monitoring, and ambient dose equivalent and directional dose equivalent for area monitoring, are to be used for monitoring for operational protection purposes. All these measurements are required to be traceable to international recognise metrological institutions maintaining primary standards for the relevant basic dose quantities. This traceability is normally ensured by proper calibration of the dosemeter by an accredited calibration laboratory.

The relationship between the different quantities is implicitly given by the exact definition of those (Table 3). For practical reasons conversion factors are published (in most cases determined by MC calculation) for special well defined conditions or procedures (e.g. for the purpose of calibration in a well-defined set up).

One important difference between all these quantities is the fact that the definition considers different concepts: the basic physical quantities are defined in any point of the radiation field. The operational quantities are defined in a point in a special phantom (ICRU sphere or the human body of a person wearing a personal dosemeter) in an aligned and expanded radiation field (Figure 1). The protection quantities are doses averaged over an organ/tissue or weighted over the whole body of a person.

Table 1. Annual dose limits given in the BSS [(2)].

Limiting quantity (protection dose quantity)	**Exposed workers** (aged over 18)	**Apprentices and students** (aged between16 and 18)	**Public**
Effective dose	100 mSv / 5a and 50 mSv on a single year;	6 mSv	1 mSv
Equivalent dose for the lens of the eye	150 mSv	50 mSv	15 mSv
Equivalent dose for the skin and extremities	500 mSv	150 mSv	50 mSv

Table 2. Summary of the system of different types of dose quantities.

Dose quantities		
basic physical quantities	**operational quantities**	**protection quantities**
Fluence (m^{-2}) Kerma (Gy) Absorbed dose (Gy)	Ambient dose equivalent (Sv) Directional dose equivalent (Sv) Personal dose equivalent (Sv)	Equivalent dose for organs and tissues (Sv) Effective dose for the whole body (Sv)
Realised by primary standards	Measured with a calibrated routine dosemeter	Quantity for which dose limits are stated
Defined by ICRU		Defined by ICRP

The detailed definitions and explanations of the system of basic quantities, protection quantities and operational quantities are summarised in Table 3.

Table 3. Definition and SI units of dose quantities (definitions of ICRU and ICRP).

Definition	Equation	SI unit
The **fluence, Φ** is the quotient of dN by da, where dN is the number of particles incident on a sphere of cross-sectional area da	$\Phi = \frac{\mathrm{d}N}{\mathrm{d}a}$	m^{-1}
The **Kerma, *K*** is the quotient dE_{tr} by dm, where dE_{tr} is the sum of the initial kinetic energies of all charged ionizing particles liberated by uncharged ionizing particles in a volume element of mass dm	$K = \frac{\mathrm{d}\,E_{\mathrm{tr}}}{\mathrm{d}\,m}$	$J.kg^{-1}$ special name: gray (Gy)
The **absorbed dose, D** is the quotient of $\mathrm{d}\bar{\varepsilon}$ by dm, where $\mathrm{d}\bar{\varepsilon}$ is the mean energy imparted by ionizing radiation to matter of mass dm	$D = \frac{\mathrm{d}\bar{\varepsilon}}{\mathrm{d}m}$	$J.kg^{-1}$ special name: gray (Gy)
The **equivalent dose, $H_{T,R}$** , is the absorbed dose in an organ or tissue multiplied by the relevant radiation weighting factor, where $D_{T,R}$ is the absorbed dose averaged over the tissue or organ T, due to radiation R and w_R is the radiation weighting factor for radiation type R	$H_{\mathrm{T,R}} = w_{\mathrm{R}} \cdot D_{\mathrm{T,R}}$	$J.kg^{-1}$ special name: sievert (Sy)
The **effective dose, E,** is a summation of the equivalent doses in tissue, each multiplied by the appropriate tissue weighting factor, where H_T is the equivalent dose in tissue T and w_T is the tissue weighting factor for tissue T.	$E = \sum_{\mathrm{T}} w_{\mathrm{T}} \cdot H_{\mathrm{T}}$	$J.kg^{-1}$ special name: sievert (Sy)
The **dose equivalent, *H*** is the product of Q and D at a point in tissue, where D is the absorbed dose and Q is the quality factor at that point,	$H = Q \cdot D$	$J.kg^{-1}$ special name: sievert (Sy)
The **ambient dose equivalent, *H**(*d*)** at a point in a radiation field is the dose equivalent that would be produced by the corresponding expanded and aligned field in the ICRU sphere at a depth, *d*, on a radius opposing the direction of the aligned field	$H^*(d)$	$J.kg^{-1}$ special name: sievert (Sy)
The **directional dose equivalent, *H′*(*d*, *Ω*)** at a point in a radiation field is the dose equivalent that would be produced by the corresponding expanded field in the ICRU sphere at a depth, d on a radius in a specific direction, Ω.	$H'(d,\Omega)$	$J.kg^{-1}$ special name: sievert (Sy)
The **personal dose equivalent, $H_p(d)$** at a point in a radiation field is the dose equivalent in soft tissue at an appropriate depth d, below a specified point on the body.	$H_p(d)$	$J.kg^{-1}$ special name: sievert (Sy)

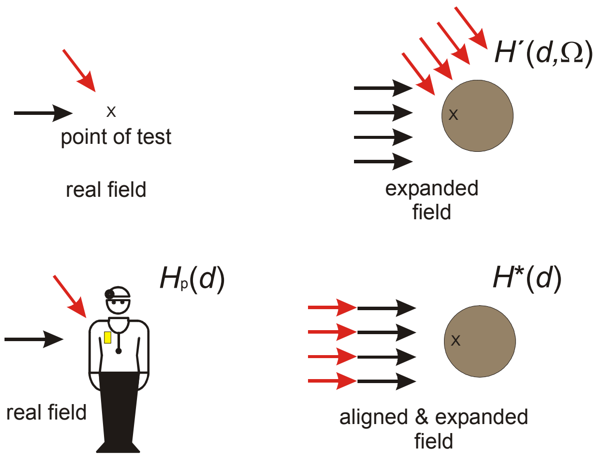

Figure 1. Schematic example of expanded and aligned radiation fields used in the definition of the operational quantities H*(d) and H‘(d,Ω), defined in the ICRU-sphere. The quantity H_p(d) is defined in the real field in a point in the human body wearing a dosemeter.

Quality factor Q and Radiation weighting factors w_R

The radiation weighting factor is defined as a factor w_R by which the tissue or organ absorbed dose $D_{T,R}$ is multiplied to reflect the higher biological effects of neutron, proton and alpha radiation compared to low LET-radiation. Considerable changes of the previous values were recently introduced by ICRP 103 [(11)]. Especially the weighting factor for neutrons was decreased by a factor of two for neutron energies below 1 MeV.

In the early 1960s, radiation weighting in the definition of radiological protection quantities was related to the radiation quality factor Q, as a function of LET. In ICRP 60 [(8)] the method of radiation weighting was changed in the definition of the protection quantities equivalent dose and effective dose. ICRP selected a general set of radiation weighting factors (w_R) that were considered to be appropriate for application in radiological protection. Today the quality factor Q however is still used for defining the operational quantities H*, H’ and H_p.

Tissue weighting factors

The tissue weighting factor w_T is defined as a factor by which the equivalent dose to a tissue or organ H_T is multiplied in order to account for the relative stochastic detriment resulting from the exposure of different tissues or organs. Recent ICRP publication 103 [(10)] recommended in 2007 new tissue weighting factors for 14 organs and tissues and for a remainder (Figure 1).

New factors in ICRP 103

ICRP 110 [(9)] compares conversion coefficients for effective dose to air kerma for ICRP 60 and ICRP 103 values. For photons no significant differences especially in the lower energy region were stated. For neutrons mainly the change of the quality factor influences the conversion coefficient for energies below one MeV by a factor of 2.

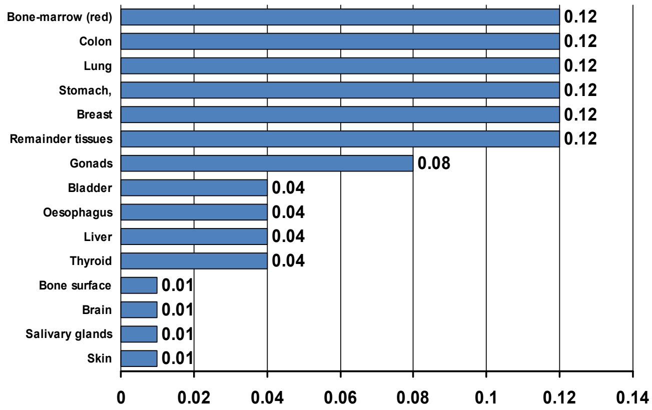

Figure 2. Tissue weighting factors (ICRP-103).

Assessment of the limiting quantities by the corresponding operational quantities

The *protection dose quantities* - defined by the ICRP - form the basis for dose limitation. These quantities are not directly measurable. The exposure can only be assessed by calculations when both the irradiation geometry and the radiation field parameters (type of radiation, intensity, spectral energy distribution etc.) are known. For routine monitoring, measurements with personal dosemeters calibrated in the operational quantities are performed. The suitability of this procedure to give a reasonable estimation of the appropriate limiting quantity was investigated by a joint task group of the ICRP and the ICRU. The results are published both as ICRP publication 74 [(10)] and ICRU report 57 [(7)]: in these publications the dose measured by the operational quantities were compared with the dose of the corresponding protection quantities (effective dose, equivalent dose, all weighting factors from ICRP 60). Some results are given in Figure 3. The influence of the changed neutron radiation weighting factor from ICRP 103 is added in addition. These diagrams show that the operational quantities for photons for irradiations from the front side (AP) always give a conservative estimate of the effective dose. This is valid for neutrons below 10 MeV too, when the new neutron radiation weighting factors from ICRP 103 are applied.

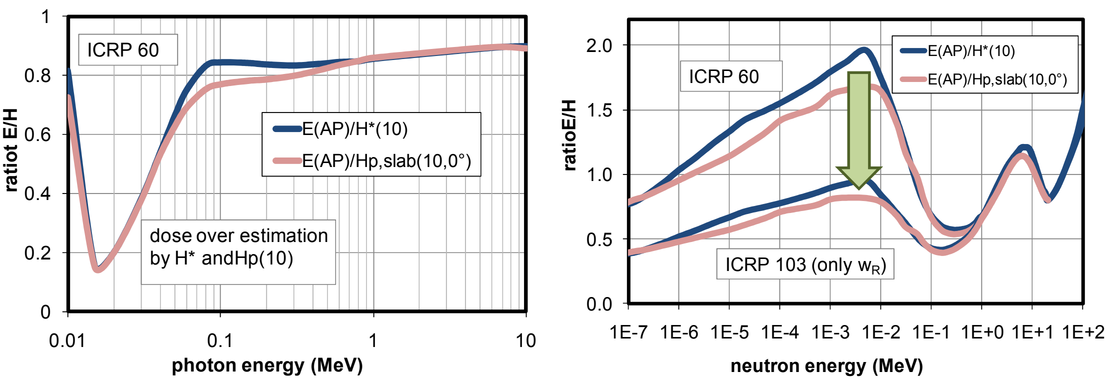

Figure 3. Ratio of the effective dose E and ambient dose equivalent H*(10) and personal dose equivalent $H_{p,slab}(10,0°)$ for photons and neutrons in AP irradiation geometry (ICRP60 weighting factors, except right figure where the new ICPR radiation factors for neutrons are used for the two lower lines) from [(10)] and [(7)].

Detection techniques

For routine individual monitoring active and passive dosimetry systems are used. Active systems give the worker an instant indication of both accumulated dose and sometiomes dose rate. If an alarm function is available these devices can be used simultaneously as an integrating dosemeter and as an alarm dosemeter. Active dosemeters are suitable also for short monitoring periods and low dose values. Recent investigations show that some active dosemeters are not capable to measure reliably doses in pulsed radiation fields. Significant under responses were reported [(4)]. These systems are therefore not recommended/allowed for specific tasks. Passive integrating systems are used normally for longer monitoring periods (weeks and months). These dosemeters are commonly issued by individual monitoring services (IMS). Implementation of a quality management system and sometimes formal accreditation are often national requirements for these services. Pulsed radiation fields generally do not cause problems for the applied measuring techniques. Table 4 shows an overview of the different dosimetry systems.

Table 4. Summary of different types of personal dosemeters

System type	External photon/beta radiation	Neutron radiation
Passive integrating systems	Photographic film Thermoluminescence (TLD) Optically Stimulated Luminescence (OSL) Radio-Photo-Luminescence (RPL) Direct Ion Storage (DIS)	TLD albedo dosimeters Track etch techniques Direct Ion Storage (DIS) bubble detectors
Active systems	APD (different detectors) (3) (Direct Ion Storage - DIS)	APD (different detectors)
Wearing position	trunk (whole body), extremity	trunk (whole body),
Dose quantity	$H_p(10)$, $H_p(0,07)$, (in future $H_p(3)$?)	$H_p(10)$,

Several international standards exists for area and personal dosemeters. A summary of most recent IEC standards and a comparison of the main requirements for these dosemeters are given in Table 5 and in [(1)]. Fulfilling these requirements limits the measuring uncertainties for the measured dose. Uncertainty estimations result in typical expanded uncertainty (k=2) of ±25% for photon doses at higher dose levels (some mSv) [(10)].

Table 5. recent international IEC standards for area and personal dosemeters.

IEC Standards	Radiation protection dosemeters		
	Personal dosemeter		Area dosemeter
	active	passive	active
Standard ID	IEC-61526 (2005)	IEC-62387-1 (2007)	IEC-60846 (2002)
Dose quantity	$H_p(10)$, $H_p(0,07)$		$H^*(10)$, $H'(0,07)$, $[K_a]$
Minimum. dose range	100 µSv to 1 Sv		4 orders of magnitude
Minimum. dose rate range	0,5 µSv h^{-1} to 1 Sv/h	-	4 orders of magnitude
Energy- and angular response	-29%, +67% (photons) -35%, +300% (neutrons) (±60°)	-29%, +67% (±60°)	±40% (±45°)
Linearity	±15%	-9%, +11%	±20%
Coefficient of variation	15% to 5% (11 µSv) (photons) 25% to 5% (5,1 mSv) (neutrons)	15% to 5% (1,1 mSv)	15% to 5% (11 µSv)
Temperature dependence	±15%	-17%, +25%	±10% (inside) ±20% bis 30%
Intrinsic error	No requirements		±20%

Calibration in terms of the operational quantities

Calibration in general is the procedure to establish a relationship between the indicated value of a measuring instrument and the conventional true value of the quantity to be measured under well-defined conditions.

Radiation protection instruments for external radiation are calibrated in well-defined reference radiation fields. Recommended fields for calibration as well as the calibration methods are defined in different documents of the International Organization for Standardization (ISO) for photon [16, 17, 18], neutron [13, 14, 15] and electron radiations [19, 20, 21]. These reference radiation fields are generally defined in terms of the basic physical quantities since only these are directly traceable to primary standards.

Conversion coefficients

For the calibration in terms of the operational quantities, conversion coefficients are necessary relating the basic physical quantities to the operational quantities. Based on the results of radiation transport codes and appropriate mathematical models both ICRU [7] and ICRP [10] recommend a set of conversion coefficients for photons, neutrons and electrons. These published values are however restricted to monoenergetic radiation fields. When fields with spectral distributions are used a spectrum weighted conversion coefficients must be applied.

According to the definition of the operational quantities the conversion coefficients for $H^*(10)$, $H'(10, \Omega)$ and $H'(0.07, \Omega)$ are derived from calculated dose distributions in the ICRU sphere.

Since the personal dose equivalent is defined in the individual body of the person wearing the dosemeter, individual conversion coefficients would be required. Since this is not practical, conversion coefficients for $H_p(10)$ and $H_p(0.07)$ are calculated only for three standard phantoms representing typical wearing positions (trunk, wrist and ankle, finger) of common used dosemeter types. All these phantoms are composed of ICRU-tissue with a density: 1 g/cm³ and a mass composition of: 76.2% oxygen, 11.1% carbon, 10.1% hydrogen and 2.6% nitrogen. For the calibration of personal dosemeters similar shaped calibration phantoms are used – due to practical reasons the material of these phantoms however is different (see Table 6)

Calibration method

The calibration method and the application of the conversion factor are carried out in principal in the following way:

- Selection of the dosemeter to be calibrated and the calibration conditions (calibration quantity, radiation quality, direction of incidence etc.)
- Selection of a suitable reference radiation field and a point of test (parameters like radiation quality, intensity, field size and homogeneity etc. needs to be considered)
- Determination (measurement) of the value of the appropriate basic physical quantity in the point of test (without the dosemeter and any calibration phantom)
- Calculation of the value of the required operational quantity by application (multiplication) of the corresponding conversion factor. This result is considered to be the conventional true value.
- Positioning of the dosemeter (and a phantom if required) with its reference point at the point of test, irradiating the dosemeter and reading the indicated value.
- Calculation of the calibration factor of the dosemeter defined as the ratio of the conventional true value (step 4) to the indicated value (step 5).

Table 6. Phantoms for the calibration of personal dosemeters

Calibration phantoms				
Name	**Shape and dimension**	**Material**	**Calibration quantity**	**Wearing position of dosemeter**
Water slab phantom	Slab: 30 cm x 30 cm x 15 cm	PMMA walls (100 mm, on front side 2,5 mm thick) filled with water	$H_p(10)$; $H_p(0,07)$	trunk
Pillar phantom	Cylinder: Diameter 7,3 cm Length 30 cm	PMMA walls (2,5 mm on circum-ference, 10 mm on faces sides thick) filled with water	$H_p(0,07)$	wrist, ankle
Rod phantom	Cylinder: Diameter 1,9 cm Length 30 cm	PMMA	$H_p(0,07)$	finger

Calibration phantoms

Depending on which dosemeter type is calibrated, special calibration phantoms must be used. The calibration of area monitoring instruments in terms of $H^*(d)$ or $H'(d, \Omega)$ is carried out free in air. No calibration phantoms are needed.

The calibration of personal dosemeters in terms of $H_p(d)$ is carried out always on an appropriate calibration phantom. The function of these phantoms is the production of backscattered radiation. Three different calibration phantoms are recommended depending on the wearing position of the personal dosemeter. Details for the water slab, pillar and rod phantom are given in Table 6. When using calibration phantoms the corresponding conversion factor must be applied.

Summary

Individual monitoring is required by international regulations to demonstrate the compliance of dose limits. Whole-body and extremity dosemeters are calibrated in the operational dose quantities $H_p(10)$ and $H_p(0{,}07)$. The question if $H_p(3)$ should be used in addition is still under discussion. In most cases it is assumed that these operational quantities are a conservative estimate for the limiting quantities effective dose and equivalent dose. Recent changes of the radiation weighting factor for neutrons by a factor of two improved the applicability of $H_p(10)$ for the estimation of effective dose for neutrons. International standards for personal dosemeters state minimum requirements for these dosemeters. Fulfilling these requirements limits the measuring uncertainties for the measured dose and guarantees reliable dose measurements.

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Radiobiology – Evaluation of health risks after ionising radiation

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Abstract
For radiological protection two classes of effects are grouped: deterministic and stochastic effects. Deterministic effects with threshold doses are mainly tissue effects, dose response is well-known from clinical experience and animal experiments. For stochastic effects (cancer and hereditary effects) no threshold dose is assumed. The knowledge of mechanisms is necessary for solving these open questions. DNA damage and its repair, apoptosis, adaptive response, bystander effects, genomic instability are important in this connection and will be discussed. Radiations with different LET as well as biological systems with different sensitivities like prenatal development and genetic diseases will be of high interest.

Clinical auditing and quality assurance

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Abstract
In the jungle of the concepts of quality management, with a diversity of approaches and procedures in order to improve and maintain high quality, the meaning of concepts can easily be confused with each other and this has been particularly true for clinical audit. Clinical audit is a systematic review of the procedures in order to improve the quality and the outcome of patient care, whereby the procedures are examined against agreed standards of good practices.

While it certainly is an important part of the overall quality assurance activities, it should not be confused with either overall quality assurance or quality control programmes. It should neither be confused with external quality assessments such as accreditations or certification of quality systems, or with regulatory inspections.

In Europe, the Council Directive 97/43/EURATOM introduced the concept of clinical audit to medical RADIOLOGICAL (diagnostic radiology, nuclear medicine and radiotherapy)[1] procedures. Recently, the European Commission has published further guidelines on clinical audits (Report Radiation Protection 159). In this refresher course, the purpose, scope and methods of clinical auditing are presented in accordance with the EC guidelines. The role of clinical audit as a part of overall quality assurance is explained and its relation to the concepts of quality management clarified.

Introduction
It has been estimated [UNSCEAR, 2000] that worldwide there are about 2000 million x-ray studies, 32 million nuclear-medicine studies and over 6 million radiation therapy patients treated annually, and the numbers are constantly increasing. The use of radiation for medical diagnostic examinations contributes over 95 % of the man-made radiation exposure and is only exceeded by natural background as a source of exposure [UNSCEAR, 2000]. Furthermore, the dose to patients for the same type of examination differs widely between centres, suggesting that there is considerable scope for management of patient dose. About 40 to 60 % of all cancer patients are treated at least once during their disease with radiotherapy and more than half of these with curative intent. The difference between the dose that is required to achieve local control and the dose that can cause serious side effects is often quite small.

[1] "RADIOLOGICAL", written in capital letters, is used throughout this paper to denote all three fields of application: diagnostic radiology, nuclear medicine and radiotherapy. When only diagnostic radiology is concerned, the term is written in small letters ("radiological"). .

The above facts have stressed the importance of proper justification, optimization and quality assurance procedures in order to ensure the high quality and safety of RADIOLOGICAL procedures. A lot of attention has been paid to the quality management in the medical use of radiation, and worldwide there has been a tendency to establish quality systems and introduce appropriate quality audits. Among the regulatory efforts, the Council Directive 97/43/EURATOM (the MED directive; Article 2 and Article 6(4)) [European Commission, 1997] introduced the concept of *clinical audit* for medical RADIOLOGICAL procedures. According to the MED directive, clinical audits shall be implemented in accordance with national procedures.

In the jungle of the concepts of quality management, with a diversity of approaches and procedures in order to improve and maintain high quality, the meaning of concepts can easily be confused with each other and this has been particularly true for *clinical audit*. While it certainly is an important part of the overall quality assurance activities, it should not be confused with either quality assurance or quality control programmes. It should neither be confused with external quality assessments such as accreditations or certification of quality systems, or with regulatory inspections.

In this refresher course, the purpose, scope and methods of clinical auditing are presented in accordance with the EC guidelines. The role of clinical audit as a part of overall quality assurance is explained and its relation to the concepts of quality management clarified.

EC Guidelines on clinical audit

The concept of clinical audit is not a new one but has long been applied in many health care practices. In the European Commission (EC) directive 97/43/EURATOM (MED) [European Commission, 1997], this concept has been introduced for the assessment of medical RADIOLOGICAL practices. The MED-directive defined clinical audit as

> "a systematic examination or review of medical radiological procedures which seeks to improve the quality and the outcome of patient care, through structured review whereby radiological practices, procedures, and results are examined against agreed standards for good medical radiological procedures, with modifications of the practices where indicated and the application of new standards if necessary".

EU Member States are required to implement clinical audits "in accordance with national procedures" (Article 6.4 of the MED). Due to the high variation between the approaches of the Member States in its implementation, in 2007-2008, the EC conducted a special project to provide guidance on clinical auditing for an improved implementation of Article 6.4 of the MED.

The EC guidelines [European Commission 2009] provide a general framework for the Member States in order to establish sustainable national systems of clinical auditing of RADIOLOGICAL practices (diagnostic radiology, nuclear medicine and radiotherapy). The guidelines are sufficiently flexible and will enable the Member States to adopt the model of clinical audit with respect to their national legislation and administrative provisions.

The EC guidelines document introduces the basic principles of clinical audit (objectives, coverage, standards of good practice etc) aiming at clarifying its profound meaning and recommended application. It defines the topics which should be covered while the criteria of good practice are discussed only on generic levels. It discusses the interrelation of clinical audit with other audit systems such as certification of quality systems, accreditation, peer review and quality award, and also its interrelation with regulatory control. Finally, it gives general advice for the practical implementation of audits, including organization of audits, recommendations for auditors, models of financing, national coordination and the role of scientific and/or professional societies and regulatory authorities.

It is important to recognize that the EC guidelines are not a legal requirement. The guidelines will only give recommendations and highlight some possible "national procedures" as expected by the MED.

Simultaneously with the European development, the IAEA has developed comprehensive audit programmes under the term of clinical audit [IAEA 2007; 2010]. Further, several dosimetry or quality audit programmes traditionally applied in the field of radiotherapy (e.g. by the IAEA [Izewska et al. 2004] or the ESTRO [Ferreira et al. 2000]) have been recognized to form an important part of clinical audit. There are also several national efforts of clinical auditing providing examples of potential approaches, e.g. the AuditLive system in the UK [The Royal College of Radiologists 2009] and the Finnish nationwide organization [Soimakallio 2003]. The general framework published in the EC guideline is well supported by these other international and national developments.

Essentials of clinical audit

The general purpose of any clinical audit is to

- improve the quality of patients' care
- improve the effective use of resources
- enhance the provision and organization of clinical services
- further professional education and training

With these objectives, clinical audit is an integral part of the overall quality improvement process and should be considered as an integral part of quality management and clinical governance. In the conceptual frame of quality concepts, clinical audit is a type of quality audit, which is one of the methods to implement overall quality assurance, which itself is one of the tools of quality management as shown in Fig. 1.

Clinical audit should cover the whole clinical pathway, and address the three main elements of the RADIOLOGICAL practices: structure, process, and outcome For RADIOLOGICAL procedures, the priorities should be as shown in Table 1. Clinical audits should address both the critical issues of the radiation protection for the patient and key components of the overall quality system.

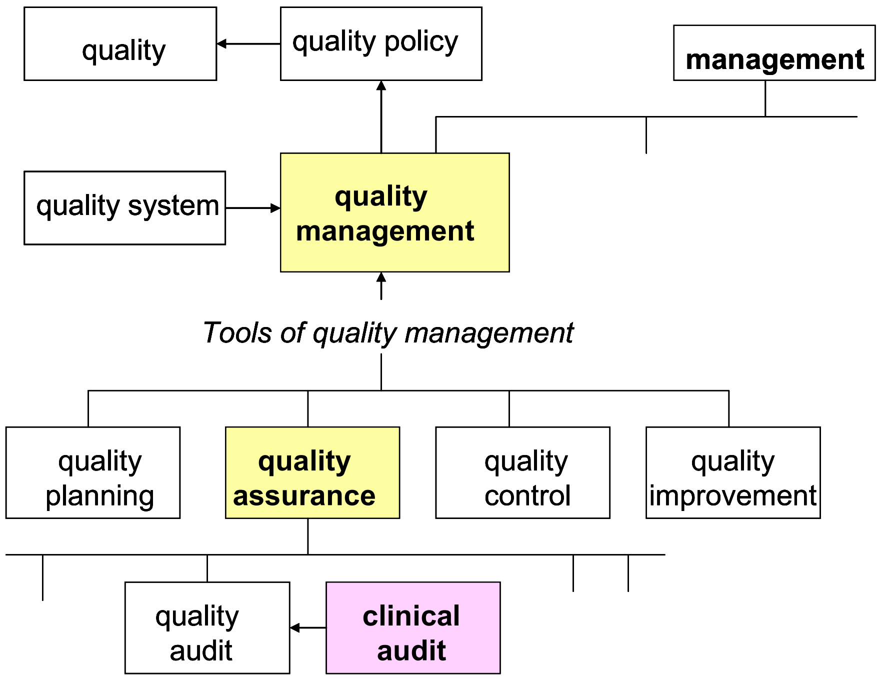

Fig. 1. Clinical audit in the "jungle" of quality management concepts; quality management scheme based on EN ISO 8402 [ISO 1995].

Table 1. The priorities of clinical audit of RADIOLOGICAL practices.

Structure	The mission of the unit for RADIOLOGICAL practices Lines of authorities and radiation safety responsibilities Staffing levels, competence and continuous professional development of staff, in particular for radiation protection Adequacy and quality of premises and equipment
Process	Justification and referral practices, including referral criteria Availability and quality of examination and treatment guidelines (protocols, procedures) Optimization procedures Patient dose and image quality in diagnostic radiology and nuclear medicine procedures, and comparison of patient dose with nationally accepted reference levels Procedures for dose delivery to the patient in radiotherapy (beam calibrations, accuracy of dosimetry and treatment planning) Quality assurance and quality control programmes Emergency procedures for incidents in use of radiation Reliability of information transfer systems
Outcome	Methods for the follow-up of outcome of examinations and treatment (short term and long term)

Clinical audit aims at continuous improvement of the medical practices. Therefore, it should be carried out regularly and it should be ensured that the audit cycle is completed (Fig.2). The general audit cycle consists of selecting a standard of good practice, assessing and comparing local practice with accepted standards, implementing change when necessary, and re-auditing after a certain time. Regular re-audits will improve the quality or give reassurance that a good quality is maintained.

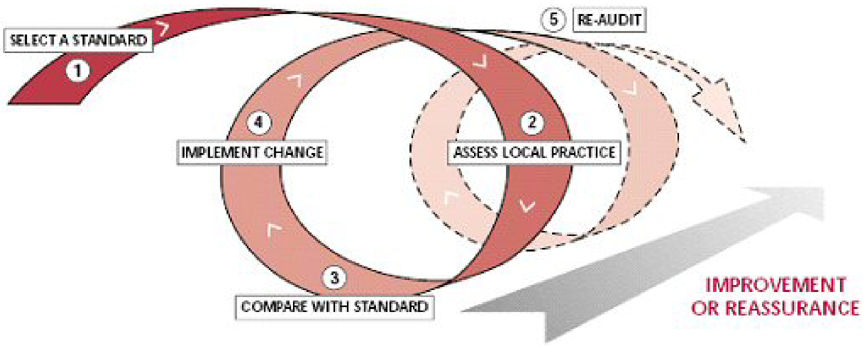

Fig.2 The audit cycle. Reprinted from Goodwin R, de Lacey G, Manhire A (eds). Clinical Audit in Radiology: 100+ Recipes, 1996 by permission of The Royal College of Radiologists.

It is evident from the definition that clinical audit is a truly multi-disciplinary, multi-professional activity. It must be carried out by auditors with extensive knowledge and experience of the radiological practices to be audited, i.e., they must generally be professionals involved in clinical work within these practices. Further, the general understanding of the concept "audit" implies that the review or assessment is carried out by auditors independent of the organizational unit or practice to be audited.

Both internal audits and external audits should be implemented. These should be of equal importance and supplement each other. *Internal* audits are undertaken within a given health care setting by staff from the same institution, while the audit findings can be also externally reviewed. In small health care units, internal audits would rather be self-assessments. *External* audits involve the use of auditors who are independent of the radiology department/institution. External audits may help identify other, unrecognised areas for improvement. They are needed to remove possible “blindness” of internal experts to recognize weaknesses of own unit and to give more universal and broader perspectives. External auditors should also possess better benchmarking skills in relation to the assessment.

The *standards of good practice* should be derived from evidence based data, long term experience and knowledge gained. In practice, these can be adopted from legal requirements, results of research, consensus statements, recommendations by learned societies or local agreements (if there is no other more universal reference). International medical/scientific/professional societies could play an important role in developing such standards. In a case where a consensus of the good practice is not easily achieved, the criteria of good practice adopted should be regarded as giving only preliminary orientation, and the results of audit should then be used like a benchmarking tool to achieve improved evidence and possible adjustment of the chosen criteria.

Clinical audits can be of various types and levels, either reviewing specific critical parts of the RADIOLOGICAL process (partial audit) or assessing the whole process (comprehensive audit). Dosimetry audit is an important example of partial audits and should be within the scope of a comprehensive clinical audit, as assured dosimetry is a vital component of accurate clinical practice. The comprehensive clinical audit must include the full patient pathway from referral to follow up.

Audits can address various “depths” of the procedure, from generic features to details of a given examination or treatment. Auditing the detailed practice for a given examination or treatment can usually mean only a few selected processes per audit run. Full details of the procedures should be assessed at least for the items of the procedure where a reasonable consensus on a good practice can be achieved.

Before starting the clinical audit the critical areas should be identified and the objectives agreed. For internal audits, the objectives are set by the management of the health care unit to be audited. For external audits, the detailed objectives should be agreed between the auditing organization and the unit to be audited, and should be based on any legal requirements on audit programmes, as well as on any recommendations by national coordinating organizations or by health professional and/or scientific societies when available.

The practical organizing of *external* clinical audits can be through site visits of an audit team or, for a limited part of practices with relevant documented or measurable data, by the collection of data via mail or internet, with central assessment of the data. Examples of the latter technique are the assessment of the quality of CT referrals by mailed questionnaire [Almen et al., 2009] and a national audit of provision of MRI services [Barter et al., 2008]. For comprehensive audits, a site visit is needed and should comprise a series of interviews, observations, document and data reviews, measurements, collection of data samples and analysis. Due to the multidisciplinary nature of the audit, a team of auditors is usually needed, comprising different professionals - radiologist, medical physicist, radiographer etc - depending on the scope of the audit. Besides the basic clinical competence, the auditors should receive specific training on the general audit procedure and techniques, as well as the agreed audit programme and the criteria of good practices to be applied.

Once the clinical audit has been completed and the auditor’s report with recommendations is available to all staff, the unit should respond to the recommendations with an agreed timeline for improvement. This is important not only to achieve maximum benefit from the audit but also to retain the respect and motivation of the staff for subsequent re-audits.

In summary, clinical audit is an important tool of quality improvement and can have a major impact on developing the medical RADIOLOGICAL practices in compliance with the most recent data on good practices. The audits will yield multiple benefits to the health care system such as:

- improvement of practice
- recognition for quality and awareness of good practices
- recognition of outdated practice
- motivation of staff to increase quality
- improvement of local standards and adherence to national standards
- prevention against litigation

- improvement of communication within the institution,
- revealing weak points and
- promoting development of quality systems.

Relationship with other audits

While it is obvious that clinical audit has some similarities with other activities for the development and control of RADIOLOGICAL practices, it is imperative not to confuse it with such activities as

- research
- quality control program for equipment
- regulatory inspection nor any other regulatory activity.

Further, clinical audit should not be confused with other external quality assessments systems, broadly categorized as [European Commission 2009]:

- certification by International Standards Organization (ISO)
- accreditation
- professional peer review –based schemes
- award seeking such as European Quality Award and their national variants (i.e. European Foundation for Quality Management (EFQM) Excellence Model).

Alhough the methodology and terminology of these external review systems differ, a constant movement towards collaboration and convergence of those models has been observed [European Commission 2009]. The purpose of these other activities should be properly understood, and conversely to duplicating any efforts, clinical audits should be developed to supplement the other activities.

Research is a systematic investigation to increase the sum of our knowledge. For clinical audit, the aim of research is to determine what a good practice is, while audit itself should ask the question: "Are we actually following good practice?"

The *quality control* of RADIOLOGICAL equipment aims at ensuring adequate performance characteristics and safe operation throughout the lifetime of the unit, and the responsibility for the checking lies solely on the health care organization, the user of the equipment or the practitioner. Any audit or external control does not remove or release this responsibility while, however, can give desirable confidence on the results of the quality control.

A legislative and statutory framework is needed in each country to regulate the safety of facilities and activities, including medical use of radiation. The *regulatory requirements* will generally depend on the level of risk or complexity associated with the medical use, as determined by the regulatory body. In radiotherapy, for example, the high risk profile justifies maximum regulatory efforts, including regulatory inspections. The purpose of such an inspection is to verify that various detailed requirements for radiation protection are being met. The methods of verification can include both documentary assessments and verification measurements. While the verification procedures can be partly similar to some clinical audit procedures, the basis of the review and the use of the results are quite different (Table 2) and clinical audits should not be confused with *regulatory inspections*.

Table 2. Main differences between clinical audit and regulatory inspection.

	Clinical audit	Regulatory inspection
Focus of review	"Agreed standards for good medical RADIOLOGICAL procedures". These are often not requirements but recommendations to the users. There may be more than just one agreed standard.	Legislative and statutory framework (laws, statutes and other regulations). These are unambiguous and usually binding requirements to the users.
Use of the results	Auditor's report, with the findings and recommendations, is given to the user. *The auditor cannot enforce any actions*, but the actions are solely decided by the user.	The non-compliance with specified conditions and requirements leads to *enforcement actions* by the regulatory body. The regulatory inspector may impose on the spot corrective requirements to the user.

A quality system in conformance with international quality standards such as the ISO 9001 [ISO, 2000] is generally considered a good basis for the overall quality management in a RADIOLOGICAL unit. To ensure that the local quality system conforms to the specifications of the quality standard, *certification* by a certification body can be acquired by special quality (system) audits carried out regularly by quality system experts. These experts are usually not clinical experts and the assessment does not address the quality of the clinical practice but solely its conformance with the general quality rules. Conversely, in clinical audits, clinical experts concentrate on the evaluation of the conformance of the clinical practice to the agreed good practice. The difference between quality system audits and clinical audits is demonstrated further in Fig. 3.

While clinical audit is not a systematic assessment of the quality system, the quality system documentation is a prerequisite for meaningful evaluation of the practices against standards of good practices. Therefore, an important supplementary benefit of clinical audits has been noted to be that their implementation has speeded up the development of appropriate quality systems in health care units.

The system of *accreditation* deals with the competence of the unit to perform certain practices, in accordance with given standards. This can become very close to the aims of clinical auditing, while usually the scope is narrower and limited to the definite standards. In RADIOLOGICAL practices, such standards are either very general or not very common and limit the applicability of accreditation in the sense of replacing the clinical audits.

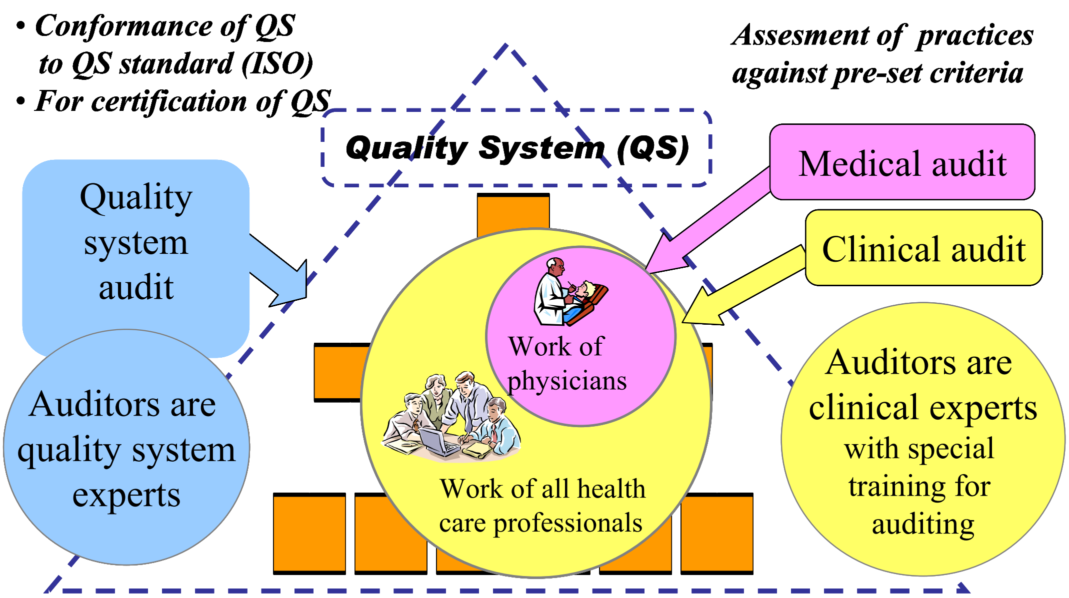

Fig. 3. Quality audits for a health care organization: Clinical audit versus quality system audit.

Conclusions

Clinical audit is a systematic review of the procedures in order to improve the quality and the outcome of patient care, whereby the procedures are examined against agreed standards of good practices. It is an important tool of quality improvement and can have a major impact on developing the medical RADIOLOGICAL practices in compliance with the most recent data on good practices. While it is an important part of the overall quality assurance activities, it should not be confused with either quality assurance or quality control programmes. It should neither be confused with external quality assessments such as accreditations or certification of quality systems, or with regulatory inspections.

Acknowledgements

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Natural radiation environment and NORM

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Abstract
We are exposed to natural radiation all the time and everywhere. However, the types and levels of exposure vary considerably from "insignificant background" to situations where workers or members of the public receive doses which would not be accepted in any planned uses of radiation sources. In some cases, human activities have modified or even caused new pathways of exposure to natural radiation sources.

The refresher course will introduce the most common sources of natural radiation, as well as, types and levels of exposures delivered to workers and the members of the public. It will also highlight key aspects and challenges in reducing exposures and in introducing regulatory requirements for e.g. radon in workplaces, natural radioactivity in building materials and industries involving NORM (Naturally Occurring Radioactive Material).

The aim of the refresher course is to provide a general overview on natural radiation environment and NORM and to highlight practical aspects in managing related exposures.

Introduction
This refresher course introduces the most common sources of natural radiation, as well as, types and levels of exposures delivered to workers and the members of the public. The aim is also to highlight practical aspects in managing related exposures by means of regulation and regulatory control.

The United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR) has provided a comprehensive overview on sources and levels of exposures to natural radiation. The following chapters are mostly summaries condensed from the UNSCEAR 2000 report (UNSCEAR, 2000). Because condensing always imply personal preferences it is recommended to refer to the full report for complete information.

The current EU Basic Safety Standards Directive (BSS, 1996) covers work activities which involve the presence of natural radiation sources. The EU Member States are required to identify by means of surveys or by any other appropriate means work activities which lead to significant exposure of workers or members of the public. Corrective measures to reduce exposures and a system of radiological protection shall be applied to occupational and public exposures, where appropriate.

In this refresher course, the also Finnish experiences in implementing the above mentioned EU provision are discussed (separate Power-Point slides) as examples of imposing regulatory requirements and controls on exposures to natural radiation.

Cosmic radiation

Cosmic rays

The earth is continually bombarded by high-energy particles that originate in outer space. These cosmic rays interact with the nuclei of atmospheric constituents, producing a cascade of interactions and secondary reaction products that contribute to cosmic ray exposures. The cosmic ray interactions also produce a number of radioactive nuclei known as cosmogenic radionuclides.

Galactic cosmic rays consist of a nucleonic component, which in aggregate accounts for 98% of the total, and electrons, which account for the remaining 2%. The nucleonic component is primarily protons (88%) and alpha particles (11%), with the remainder heavier nuclei. These primary cosmic particles have an energy spectrum that extends from 10^8 eV to over 10^{20} eV.

Another component of cosmic rays is generated near the surface of the sun by magnetic disturbances. These solar particle events are comprised mostly of protons of energies generally below 100 MeV and only rarely above 10 GeV. These particles can produce significant dose rates at high altitudes, but only the most energetic affect dose rates at ground level.

The most significant long-term solar effect is the 11-year cycle in solar activity, which generates a corresponding cycle in total cosmic radiation intensity. At times of maximum solar activity the field is at its highest and the galactic cosmic radiation intensity is at its lowest.

The high-energy particles incident on the atmosphere interact with atoms and molecules in the air and generate a complex set of secondary charged and uncharged particles, including protons, neutrons, pions and lower-Z nuclei. The secondary nucleons in turn generate more nucleons, producing a nucleonic cascade in the atmosphere.

The pions generated in nuclear interactions are the main source of the other components of the cosmic radiation field in the atmosphere. The neutral pions decay into high-energy photons, which produce high-energy electrons, which in turn produce photons etc., thus producing the electromagnetic, or photon/electron, cascade.

Exposure at ground level

The world average effective dose from cosmic radiation is 380 μSv/a. Dose is influenced by geomagnetic latitude, altitude, solar activity and shielding effect of buildings.

At high altitude areas doses may be significantly higher than at sea level. For example, in La Paz, Bolivia, which is located at the altitude of 3,9 km, the effective dose is about 2 mSv/a.

The effective dose rate at the equator is about 34 nSv/h and at polar regions about 43 nSv/h. Variation during the 11-year solar activity period is about 10%. At high altitudes the variation is more significant than at sea level. The shielding factor for buildings vary between 0.4 - 1, the average being about 0.8. Values like 0.4 may apply to some lower storeys of substantial concrete buildings. For houses with minimum vertical shielding (e.g. small wooden houses) the shielding factor is close to 1.

Exposures at aircraft altitudes

Aircraft passengers and crew are subject to cosmic radiation exposure rates much higher than the rates at ground level. Total exposure on a given flight depends on the particular path taken through the atmosphere and flight time; that is, it depends on the duration of exposure at various altitudes and latitudes. Complicating the situation is the fact that the exposure associated with any flight path may vary with time.

There are two possible approaches to dose assessment under these circumstances: area and/or individual monitoring for each flight, and determining the radiation fields as a function of time and space and calculating the effective dose for any flight path.

Commercial subsonic aircraft generally have cruising altitudes of 7 to 12 km. The annual number of hours flown by crew members varies. The range is 300 - 900 hours per year, with an average of about 500.

For altitudes of 9 - 12 km at temperate latitudes, the effective dose rates are in the range 5 - 8 µSv/h, such that for a transatlantic flight from Europe to North America, the route dose would be 30 - 45 µSv. At equatorial latitudes, the dose rates are lower and in the range of 2 - 4 µSv/h.

Cosmogenic radionuclidies

The interactions of cosmic-ray particles in the atmosphere produce a number of radionuclides, including ^{3}H, ^{7}Be, ^{14}C, and ^{22}Na. Production is greatest in the upper stratosphere, but some energetic cosmic-ray neutrons and protons survive in the lower atmosphere, producing cosmogenic radionuclides there as well.

Except for ^{3}H, ^{14}C, and ^{22}Na, which are elements with metabolic roles in the human body, the cosmogenic radionuclides contribute little to radiation doses and are mainly of relevance as tracers in the atmosphere and in hydrological systems after deposition. The annual effective doses from cosmogenic radionuclides are about 12 µSv from ^{14}C, 0.15 µSv from ^{22}Na, 0.01 µSv from ^{3}H, and 0.03 µSv from ^{7}Be.

Terrestrial radiation

Naturally occurring radionuclides of terrestrial origin (also called primordial radionuclides) are present in various degrees in all media in the environment, including the human body itself. Only those radionuclides with half-lives comparable to the age of the earth, and their decay products, exist in significant quantities in these materials.

Irradiation of the human body from external sources is mainly by gamma radiation from radionuclides in the ^{238}U and ^{232}Th series and from ^{40}K. Some other terrestrial radionuclides, including those of the ^{235}U series, ^{87}Rb, ^{138}La, ^{147}Sm, and ^{176}Lu, exist in nature but at such low levels that their contributions to the dose in humans are small.

External exposures outdoors

The levels of external exposures from terrestrial radionuclides are related to the types of rock from which soil originate. Higher radiation levels are associated with igneous rocks, such as granite, and lower levels with sedimentary rocks. There are exceptions, however, as some shales and phosphate rocks have relatively high content of radionuclides.

Three components of the external radiation field, namely from the gamma-emitting radionuclides in the ^{238}U and ^{232}Th series and ^{40}K, make approximately equal contributions to the externally incident gamma radiation dose to individuals in typical situations both outdoors and indoors.

The activity concentration of ^{40}K in soil is an order of magnitude higher than that of ^{238}U or ^{232}Th. The worldwide median values are 400, 35, and 30 Bq/kg, and the population weighted values are 420, 33, and 45 Bq/kg for ^{40}K, ^{238}U, and ^{232}Th, respectively. The population-weighted average absorbed dose rate in air outdoors from terrestrial gamma radiation is 59 nGy/h and typical variation at different locations is from 10 to 200 nGy/h. Some areas with much higher levels exist, for example Guarapari in Brazil and Kerala in India (thorium baring monazite sand deposits).

In addition to variations from place to place, the ambient background gamma dose rate in air at any specific location is not constant in time. It is subject to fluctuation, in particular from the removal of radon progeny in air by rainfall, soil moisture and snow cover. Washout and rainout of radon progeny from air may result in the short-term enhancement, by 50% - 100%, of the gamma ray dose rate in air. The extent of the elevation depends on rain interval, as well as, the rainfall amount. Snow cover depresses the background level by about 1% for each centimetre of snow.

External exposures indoors

Indoor exposure to gamma rays, mainly determined by construction materials, is inherently greater than outdoor exposure if earth materials have been used; the source geometry changes from half-space to a more surrounding configuration indoors. When the duration of occupancy is taken into account, indoor exposure becomes even more significant. Buildings constructed of wood add little to indoor exposures, which may then be comparable to outdoor exposures.

The world-wide population-weighted average is 84 nGy/h with national averages ranging from 20 to 200 nGy/h. The indoor to outdoor ratios range from 0.6 to 2.3, with a population-weighted value of 1.4. Thus indoor exposures (absorbed dose rate in air from terrestrial gamma radiation) are, in general, 40% greater than outdoor exposures.

Effective dose

To estimate annual effective doses, account must be taken of (a) the conversion coefficient from absorbed dose in air to effective dose and (b) the indoor occupancy factor. The average numerical values of those parameters vary with the age of the population and the climate at the location considered. Assuming 0.7 Sv/Gy for the conversion coefficient from absorbed dose in air to effective dose received by adults and 0.8 for the indoor occupancy factor, i.e. the fraction of time spent indoors and outdoors is 0.8 and 0.2, respectively, the components of the annual effective dose are determined as follows:

Indoors: 84 nGy/h × 8,760 h × 0.8 × 0.7 Sv/Gy = 0.41 mSv
Outdoors: 59 nGy/h × 8,760 h × 0.2 × 0.7 Sv/Gy = 0.07 mSv

The resulting worldwide average of the annual effective dose is 0.48 mSv, with the results for individual countries being generally within the 0.3 - 0.6 mSv range. For children and infants, the values are about 10% and 30% higher, in direct proportion to an increase in the value of the conversion coefficient from absorbed dose in air to effective dose.

Internal exposures other than radon

Internal exposures arise from the intake of terrestrial radionuclides by inhalation and ingestion. Doses by inhalation result from the presence in air of dust particles containing radionuclides of the ^{238}U and ^{232}Th decay chains. The dominant component of inhalation exposure is the short-lived decay products of radon, which because of their significance are considered separately under section “Radon and decay products”.

Doses by ingestion are mainly due to ^{40}K and to the ^{238}U and ^{232}Th series radionuclides present in foods and drinking water. It should be noticed that potassium concentration in the body is under homeostatic control and, therefore, the dose due to ^{40}K remains essentially constant, irrespectively of the amount of intake.

Inhalation intake of radionuclides of the ^{238}U and ^{232}Th decay chains (other than radon and its decay products) makes only a minor contribution to internal exposure. These radionuclides are present in air because of resuspended soil particles. Typical concentrations in air are 1 - 2 µBq/m^3 corresponding to a dust loading of 50 µg/m^3 and ^{238}U and ^{232}Th concentrations in the dust in the range 25 - 50 Bq/kg. The age-weighted annual effective dose is about 6 µSv from inhalation of uranium- and thorium-series radionuclides in air.

Ingestion intake of natural radionuclides depends on the consumption rates of food and water and on the radionuclide concentrations. Concentrations of naturally occurring radionuclides in foods vary widely because of the differing background levels, climate, and agricultural conditions that prevail. There are also differences in the types of local foods that may be included in the categories such as vegetables, fruits, or fish.

The total effective dose from inhalation and ingestion of terrestrial radionuclides is 310 µSv, of which 170 µSv is from ^{40}K and 140 µSv is from the long-lived radionuclides in the uranium and thorium series. It should be noted, however, that variations may be very significant. Some ground waters and waters from bed rock (drilled wells) may contain very significant levels of natural radionuclides including radon. Levels leading to doses up to several tens of mSv have been observed.

Radon and decay products

Radon (^{222}Rn) and thoron (^{220}Rn) are the gaseous radioactive products of the decay of the radium isotopes ^{226}Ra and ^{224}Ra, which are present in all terrestrial materials. Radon and its short-lived decay products in the atmosphere are the most important contributors to human exposure from natural sources.

The half-life of ^{222}Rn is 3.824 d. It has four short-lived decay products: ^{218}Po (3.05min), ^{214}Pb (26.8min), ^{214}Bi (19.9 min), and ^{214}Po (164 µs). Both polonium isotopes are alpha-emitters. The relatively short half-life of thoron (55.6 s) means that it does not have much time to travel from its production site to the immediate environment of human beings.

The evaluation of exposure to radon and thoron and their decay products must take account of the actual activity concentrations of the various alpha-emitting radionuclides in the decay series. An equilibrium factor F is defined that permits the exposure to be estimated in terms of the potential alpha energy concentration (PAEC) from the measurements of radon gas concentration. This equilibrium factor is defined as the ratio of the actual PAEC to the PAEC that would prevail if all the decay products in each series were in equilibrium with the parent radon. However, it is simpler to evaluate this factor in terms of an equilibrium equivalent radon concentration, C_{eq}, in the following manner:

$$F = C_{eq}/C_{rn}$$

$$C_{eq} = 0.105\ C_1 + 0.515\ C_2 + 0.380\ C_3\ (^{222}\text{Rn series})$$
$$C_{eq} = 0.913\ C_1 + 0.087\ C_2\ (^{220}\text{Rn series})$$

where the symbols C_1, C_2, and C_3 are the activity concentrations of the decay progeny, namely ^{218}Po, ^{214}Pb, and ^{214}Bi, respectively, for the ^{222}Rn series and ^{212}Pb and ^{212}Bi (C_1 and C_2) for the thoron series. The constants are the fractional contributions of each decay product to the total potential alpha energy from the decay of unit activity of the gas.

In this way, a measured radon concentration can be converted to an equilibrium equivalent concentration (EEC) directly proportional to PAEC. This provides a measure of exposure in terms of the product of concentration and time. The EEC can be converted to the PAEC, when desired, by the relationships 1 Bq/m^3 = 5.56 ·10^{-6} mJ/m^3 = 0.27 mWL (222Rn) and 1 Bq/m^3 = 7.6·10^{-5} mJ/m^3 = 3.64 mWL (thoron).

The value of the equilibrium factor may vary considerably depending on the prevailing circumstances, even from as low as 0.1 up to 0.9. However, typical value of the equilibrium factor in indoors is 0.4 and outdoors 0.6. If equilibrium factor is not determined with measurements, these values can usually be assumed.

Radon as source of indoor radon and exposure is discussed in a separate refresher course "Indoor radon sources, remediation and prevention in new construction". Therefore, characteristics of radon, its behaviour and exposures caused are not further discussed here.

Industrial activities

There are a number of circumstances in which materials containing natural radionuclides are recovered, processed, used, or brought into position such that radiation exposures result. This human intervention causes extra or enhanced exposures.

Industry uses many different raw materials that contain naturally occurring radioactive materials, sometime abbreviated NORM. These raw materials are mined, transported, and processed for further use. The consequent emissions of radionuclides to air and water lead to the eventual exposure of humans. Industries and materials for which NORM-issues have been of interest are, for example:

- Phosphate processing
- Metal ore processing
- Zircon sands

- Titanium pigment production
- Fossil fuels
- Oil and gas extraction
- Building materials
- Thorium compounds
- Scrap metal industry

The natural radionuclides present in the raw materials or wastes of these industries are those of the ^{238}U and ^{232}Th series. Releases are mainly to air or water, although landfills after dredging or wastes disposed on land may also provide pathways of exposure.

Both external and internal exposures may result from naturally occurring radionuclides released by industrial activities. In general, installations are located away from residential areas, and because external radiation levels decrease with distance from the plant, local residents are not significantly exposed. The workers, however, may receive low doses in connection with ore stock piles or waste deposits. Estimated and measured doses are in the range 0.1 - 300 µSv/a from direct exposures.

Inhalation contributes to exposure only in the vicinity of the plant, particularly with mineral sands-processing plants. Doses depend on distance and could be up to 50 µSv/a for office workers in a building just outside the plant site. In most other industries the contributions to total dose appear to be negligible.

Because most food products consumed by individuals are produced in large agricultural regions, possible dose from ingestion of radionuclides are small. For a typical situation, a small population in the vicinity of an elementary phosphorus plant, the calculated dose would be of the order 100 µSv/a. More generally, the estimated doses would be 1 - 10 µSv/a. Ingestion doses that could result from discharges of wastes to water are negligible compared to those by the other pathways.

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Internal dosimetry and individual monitoring

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Abstract
A description is given of the main principles and methods of internal dosimetry. Factors that are unique to internal dosimetry (e.g. protraction of doses over time) are discussed, and the implications for the dose quantities used in internal dosimetry are explained. The principles underlying the use of biokinetic models are discussed, and ICRP's models of the respiratory tract and human alimentary tract described. As an example of the current approach to systemic modelling of the elements, the biokinetic model for plutonium is described. Lastly, the use of individual monitoring for the retrospective assessment of internal doses is discussed.

Introduction
Internal dosimetry addresses the determination of radiation doses resulting from the intake of radioactive materials into the body by inhalation, ingestion, absorption through the skin or *via* wounds. Internal doses may be determined for particular organs or tissues in the body, or for the whole body. It is not possible to measure internal doses directly, although the activity of a radionuclide in the body can be measured. *Individual monitoring* for internal contamination is the direct or indirect measurement of the amount of a radionuclide in the body, and the retrospective assessment of internal doses using the results of such measurements.

In this paper, the main steps in the calculation of internal dose are first summarised. Subsequent sections describe the dose quantities used in internal dosimetry, the method used to calculate internal doses, and the biokinetic models used to describe the movement of radionuclides through the respiratory and gastro-intestinal tracts. As an example of the current approach to modelling the biokinetics of radionuclides after uptake from blood, the biokinetic model used to predict the systemic behaviour of plutonium is described. Finally, individual monitoring for internal contamination is discussed.

This paper aims to provide a summary of the basics of internal dosimetry and individual monitoring for internal contamination, and to give an account of recent developments in the field. It is aimed at radiation protection professionals who may have limited experience of internal dosimetry, or who wish to learn about recent advances.

Assessment of Internal Doses

The risk or severity of any adverse effects on health resulting from an intake of a radionuclide is in almost all cases determined by making an assessment of the internal radiation dose. Risks to health are then evaluated making use of information on the relationship between risk and dose such as that provided by epidemiology studies, notably the follow up of Japanese A-bomb survivors.

Since internal doses cannot be measured directly, they have to be calculated using mathematical models describing the behaviour of radionuclides in the body. The main steps in a calculation of internal dose resulting from a specified intake (i.e. a prospective assessment of dose) are as follows:

- Modelling of the **intake** process to determine the location(s) at which the radionuclide is initially deposited or through which it transits (e.g. regions of the respiratory tract; organs in the gastro-intestinal tract), and the amounts deposited in those locations
- Modelling of the **clearance** of the radionuclide from the initial site of deposition, to determine the amount cleared and rates of clearance
- Modelling of the **uptake** of the radionuclide from the initial deposition site to the blood and then to the rest of the body
- Determination of the amounts then **deposited** in the various organs and tissues of the body, and the rates at which the radionuclide is removed from these organs and tissues
- Calculation of the amounts of **energy deposited** in each target organ as a result of the decay of the radionuclide in each source organ
- Calculation of the **absorbed dose** received by each organ over time, from the amount of energy deposited per unit mass of the organ
- Calculation of the **equivalent dose** received by each organ over time, determined by applying the appropriate radiation weighting factors to the absorbed dose for each radiation type, so allowing for the different effectiveness of each radiation type in causing stochastic effects
- Calculation of the **committed equivalent dose** for each organ, by summing the equivalent dose received from the intake over a fixed time period (e.g. 50 years for occupational exposure)
- Calculation of the committed **effective dose**, determined by applying the appropriate tissue weighting factors to the committed equivalent dose for each organ and then summing, so allowing for the different radio-sensitivities of the organs for stochastic effects

Dose Quantities

The calculation of doses from radionuclides within the body (internal emitters) is less straightforward than with external radiation. A number of factors that are specific to internal dosimetry influence the dose quantities that are used, as described below.

Internal doses are protracted over time

Individuals cease to receive doses from external radiation when they leave the radiation field. In contrast, radionuclides within the body continue to deliver dose until they are removed from the body by normal biological processes or until the amount remaining is reduced to a negligible level because of radioactive decay. Thus, ICRP recommends that the quantity to be controlled should be the *committed dose*, defined as the dose incurred in the fifty years following intake for an adult, and from intake to age 70 years for children (equation 1).

$$D(70-t_0)=\int_{t0}^{70}\dot{D}(t)dt \qquad (1)$$

where "*D_dot*" is the absorbed dose rate at time t, and t_0 is the age at intake

Organs are the sources of irradiation

A particular organ may receive a dose either as a result of radionuclide decay within the same organ, or in other organs. For the purposes of a dose calculation, an organ in which radionuclide decay is taking place is called a *source organ* (denoted S), while an organ that receives a dose is called a *target organ* (denoted T). All organs in which radionuclides are present are in principle source organs, and all source organs are also target organs. Other organs may or may not be target organs, depending on how penetrating the radiation type (e.g. α, β, γ) is.

Radiosensitivity of organs is variable

Some radiation types are more effective than others in causing stochastic effects such as cancer. For radiation protection purposes, the absorbed dose associated with each radiation type is multiplied by a *radiation weighting factor*, w_R, which reflects these differences. In general, the decay of any radionuclide gives rise to the emission of more than one type of radiation, and so the absorbed dose from each radiation type must be weighted separately and then summed. The result is the *equivalent dose*, defined in equation (2) for a particular target organ T, and denoted H_T.

$$H_T=\sum_R w_R\overline{D}_{T,R} \qquad (2)$$

where D is the absorbed dose, which is averaged over the organ T, and the summation extends over all radiations R. The unit of equivalent dose is the sievert (Sv).

The radiation weighting factors, w_R, represent judgments regarding the potential relative biological damage of radiation R, taking into account all of the possible target organs or tissues (e.g. the lung) and all the possible health outcomes (e.g. lung cancer). In part, they reflect the density of ionization within the target organ, which is associated with the linear energy transfer (LET) of the radiation. Clearly, the use of radiation weighting factors represents a very simplified view of the health effects of different types of radiation. Their use is justified provided it is clear that the resulting assessed doses are only to be used for radiation protection purposes (for example, to confirm that doses are well below dose limits). The values of w_R on which current legislation is based are given in ICRP Publication 60 (ICRP, 1991), while ICRP's current recommended values are given in ICRP Publication 103 (ICRP, 2008).

Dose distributions in the body may be very inhomogeneous

It is not unusual for doses to target organs to differ by several orders of magnitude. Nevertheless, for radiation protection purposes, it is useful to derive a single dose quantity that broadly reflects the overall risk of stochastic effects on health. If all the organs and tissues of the body had the same sensitivity to ionising radiation, then a "mean dose" could be determined simply by taking a weighted average of the organ/tissue doses, with the weight factors equal to the ratio of the organ/tissue mass to the total body mass. However, there are significant differences between the radio-sensitivities of the organs/tissues, and these are reflected in the set of *tissue weighting factors* specified by ICRP. The sum of the weighted organ doses is the *effective dose*, E, which is defined as:

$$E = \sum_T w_T H_T \qquad (3)$$

where the summation extends over the organs/tissues assigned tissue weighting factors w_T. The values of w_T on which current legislation is based are given in ICRP Publication 60, while ICRP's current recommended values are given in ICRP Publication 103.

Calculation of internal dose

The committed equivalent dose rate at time t in target tissue T can be expressed as:

$$\dot{H}_T(t) = c \sum_S \sum_j q_{S,j}(t) \cdot SEE(T \leftarrow S;t)_j$$

where $q_{S,j}(t)$ is the activity of decay chain member j present in the source region S at age t; $SEE(T\leftarrow S;t)_j$ is the equivalent dose in the target tissue T per nuclear transformation in region S at age t for radionuclide j ; and c is any numerical constant required by the units of q and *SEE* (the Specific Effective Energy).

The committed equivalent dose in the target tissue T accumulated by age 70 y due to a single intake of a radionuclide j at age t_0, $H_T(70-t_0)$ is:

$$H_T(70-t_0) = \int_{t_0}^{70} \dot{H}_T(t)dt = c \cdot \int_{t_o}^{70} \sum_S \sum_j q_{S,j}(t) \cdot SEE(T \leftarrow S;t)_j \, dt$$

For each radionuclide, the *SEE* at age t takes into account the contributions of each radiation emitted by the radionuclide weighted by the appropriate radiation weighting factor. The *SEE* is :

$$SEE(T \leftarrow S\,;\,t) = \frac{1}{M_T(t)} \left[\sum_i E_i \, Y_i \, w_{R,i} \, AF(T \leftarrow S\,; E_i\,, t) + w_{R,\beta} \int_0^\infty Y(E) \, E \, AF(T \leftarrow S\,; E, t) \, dE \right]$$

where E_i is the energy of the ith discrete radiation emitted by the radionuclide with intensity Y_i per nuclear transformation, $M_T(t)$ is the mass of the target tissue T at age t, $w_{R,i}$ is the radiation weighting factor applicable to the ith radiation, $AF(T\leftarrow S;\, E_{i,\,t})$ is the absorbed fraction, which is the fraction of the energy E_i emitted in S that is absorbed in T for an individual of age t, and $Y(E)\,dE$ denotes the number of electrons in the beta or positron spectrum, with energy between E and $E+dE$.

In most cases for non-penetrating radiations (alphas, betas, positrons, electrons), energy is assumed to be deposited only in the region where the radionuclide resides, and so *AF* is equal to 1 when S and T are identical and zero otherwise. However, in some dosimetric models where the location of target cells is given special consideration (e.g. bone, the alimentary tract and the respiratory tract), the calculation of *AF* is more complicated, generally being undertaken using Monte Carlo methods in radiation transport models.

For penetrating radiations (gammas and X-rays), *AF* generally has to be calculated for all possible combinations of source and target organs. Calculations are performed using a computer model of the body often termed a "phantom" which describes the geometric relationship between the different tissues and organs of the body. Two types of phantom are currently in use: mathematical phantoms made up of arrangements of geometric shapes (e.g. the MIRD phantom); and voxel phantoms, derived from computed tomography (CT) or magnetic resonance images (MRI) obtained from high resolution continuous scans of a single individual. The MIRD phantom was used for dosimetry calculations for many years, but voxel phantoms are now coming into general use.

Biokinetic Models

The mathematical models describing the movement of radionuclides in the body are called biokinetic models. All of the models currently in use are compartment models which use first order kinetics to describe the transfer of material between compartments. Models contain a number of compartments, each representing a particular organ, part of an organ or tissue. A compartment is assigned a deposition or uptake fraction that determines its initial content (if any), and each path out of a compartment is assigned a rate constant which determines the removal rate *via* that pathway. The sum of the rate constants of the pathways leaving a compartment determines the retention half-time (also known as the biological half-time) for that compartment. A simple model may contain only two or three compartments linked by a single pathway in a linear chain. In more complex models, there may be networks of interconnected compartments in which recirculation between compartments may occur.

Biokinetic models describe the distribution of activity within the body as a function of time. Radioactive decay can be included, and the model can then be used to calculate the number of decays that take place in each compartment as a function of time. For single intakes of radionuclides with short physical half-lives or biological half-times, such as ^{131}I, almost all the dose is delivered over a period of days or weeks, whereas for radionuclides such as ^{239}Pu, with long radioactive decay half-lives and biological half-times, dose is delivered over the lifetime of the individual.

Ideally, model parameter values are determined from human studies, but for many elements, animal data must be used, and for some elements where no data is available, parameter values must be determined from data on chemically similar elements.

Biokinetic models can be used to describe the processed of intake, distribution, retention and excretion. Radionuclides entering the body by inhalation or ingestion cause the respiratory tract and gastro-intestinal tract to be irradiated. (Other organs will also be irradiated if the radiation is penetrating). Entry through contaminated wounds will also result in local irradiation of tissues. The subsequent behaviour of the radionuclide depends on the element concerned and its chemical form, which determine the solubility of the radionuclide and the extent to which it dissolves and is absorbed into blood. The chemical nature of the element determines the distribution between the organs of the body, and the retention times in each organ. If a radionuclide is an isotope of an element that is normally present (e.g. Na, K, Cl) then it behaves like the stable element. If it has similar chemical properties to an element normally present, then it tends to follow the metabolic pathways of that element, although its rate of movement between the various compartments in the body may be different (e.g. ^{90}Sr and ^{226}Ra behave similarly to Ca, ^{137}Cs and ^{86}Rb similarly to K). The behaviour of other radionuclides depends upon their affinity for biological ligands and other transport systems in the body. Radionuclides entering the blood may distribute throughout the body (e.g. ^{3}H, ^{40}K, ^{137}Cs); they may selectively deposit in a particular tissue (e.g. ^{131}I in the thyroid; ^{90}Sr in bone); or they may deposit in significant quantities in a number of tissues (e.g. ^{144}Ce, ^{210}Po, ^{239}Pu, ^{241}Am).

In addition to the direct entry of radionuclides by absorption to blood, insoluble materials may be moved slowly through the lymphatic system in the form of small particles. Movement along lymphatic vessels can lead to accumulation of radionuclides in regional lymph nodes, e.g. the tracheo-bronchial lymph nodes following movement from the lungs or the axillary lymph nodes following movement from a wound in the hand. Particles may slowly reach the bloodstream and are then rapidly removed from the circulation by phagocytic cells of the reticulo-endothelial system in the liver, spleen and bone marrow.

This paper describes some of the biokinetic and dosimetric models developed by the International Commission on Radiological Protection (ICRP). Dose coefficients for intakes of radionuclides (that is, dose per unit intake values) derived using these models and published by ICRP (ICRP, 1989, 1993, 1994b, 1995a, 1995b), are incorporated into legislation in the European Union and others parts of the world.

The ICRP Human Respiratory Tract Model

The ICRP model of the Human Respiratory Tract, HRTM (ICRP, 1994a), applies explicitly to all members of the population, utilising reference parameter values for 3 month-old infants, 1, 5, 10 and 15 year old children and adults. The model can be used both for dosimetry and for the interpretation of individual monitoring data (such as measurements of activity in the lungs).

Physiology

The respiratory tract (Figure 1) is represented by five regions. The extrathoracic (ET) airways are divided into the ET_1 region, the anterior nasal passages, and the ET_2 region, which comprises the posterior nasal and oral passages, the pharynx and larynx. The other regions are within the lungs. They are the bronchial region (BB: trachea, generation 0 and bronchi, airway generations 1-8), the bronchiolar region (bb: airway generations 9-15), and the alveolar-interstitial region (AI: the gaseous exchange region). Lymphatic tissue is associated with both the extrathoracic and thoracic airways (LN_{ET} and LN_{TH} respectively).

Intakes and deposition

The HRTM can be used to calculate doses per unit exposure to radionuclides in air as well as doses per unit intake. Intake depends on ventilation, i.e. the product of the breathing frequency and tidal volume of the lungs. This is the main aspect of the model that depends on age and level of exercise. The model can determine doses per unit exposure for four levels of exercise: sleep, sitting, light exercise, and heavy exercise.

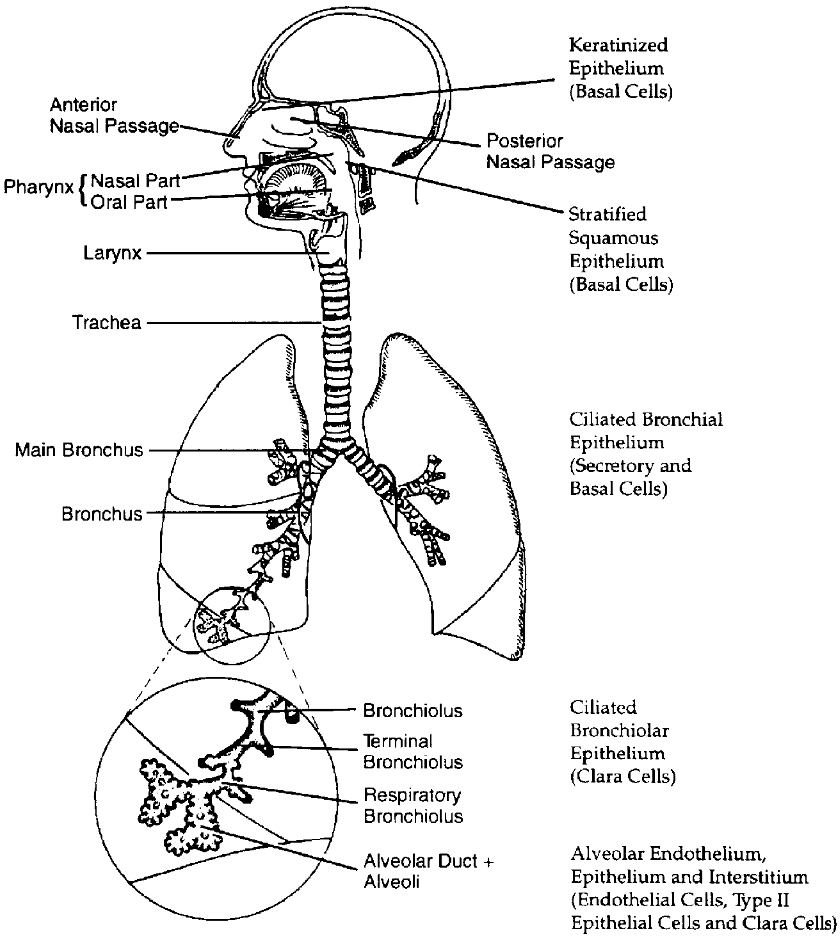

Fig. 1. Respiratory tract regions.

Deposition is calculated for each region of the respiratory tract taking account of both inhalation and exhalation. This is done as a function of particle size, breathing parameter values and/or work load, according to the age and sex of the subject. Deposition parameter values are specified for a particle size range of 0.006 µm AMTD[1] to 100 µm AMAD[2]. Default deposition parameter values for individuals are based on average daily patterns of activity. In general, deposition in the extrathoracic airways is greater for a 5 µm AMAD aerosol (the default value for occupational exposure), and conversely penetration to the alveoli is greater for a 1 µm AMAD aerosol (the default value for environmental exposure).

[1] Activity Median Thermodynamic Diameter (AMTD) is used for particle sizes < 0.5 µm, where thermodynamic processes dominate. AMAD (defined below) is used for larger particles sizes where aerodynamic processes (gravitational sedimentation, inertial impaction) dominate.
[2] Activity Median Aerodynamic Diameter (AMAD), defined such that 50% of the activity of an aerosol is associated with particles with aerodynamic diameters greater than the AMAD.

The model can be used for intakes of gases and vapours as well as particulate material. Three classes are defined: SR-0 for insoluble and non-reactive materials for which uptake/retention is negligible; SR-2 for very soluble and reactive materials for which absorption to blood is assumed to be complete; and SR-1, an intermediate category.

Clearance

The HRTM describes three clearance pathways. Material deposited in ET_1 (the front of the nose) is removed by extrinsic means such as nose-blowing. In other regions, clearance is competitive between (a) particle transport to the gastrointestinal (GI) tract and the lymphatic system, and (b) absorption into blood. Particle transport to the GI tract occurs by mucociliary clearance. It is assumed that particle transport rates are the same for all materials. Absorption into blood is material-specific and is assumed to occur at the same rate in all regions except ET_1, where none occurs.
Figure 2 shows the compartment model for particle transport. It is assumed that the AI deposit is divided between AI_1, AI_2 and AI_3 in the ratio 0.3:0.6:0.1 (half-times of about 30d, 700d and 7000d). The fraction of the deposit in bb and BB which moves slowly (bb_2 and BB_2) is particle size dependent (up to 50% for <2.5 μm) with 0.7% retained in the walls (bb_{seq} and BB_{seq}). A small fraction is retained in the wall in the ET_2 region (ET_{seq}) and the rest is cleared rapidly to the GI tract. Simultaneous absorption to blood occurs as a competing process in all compartments except ET_1.

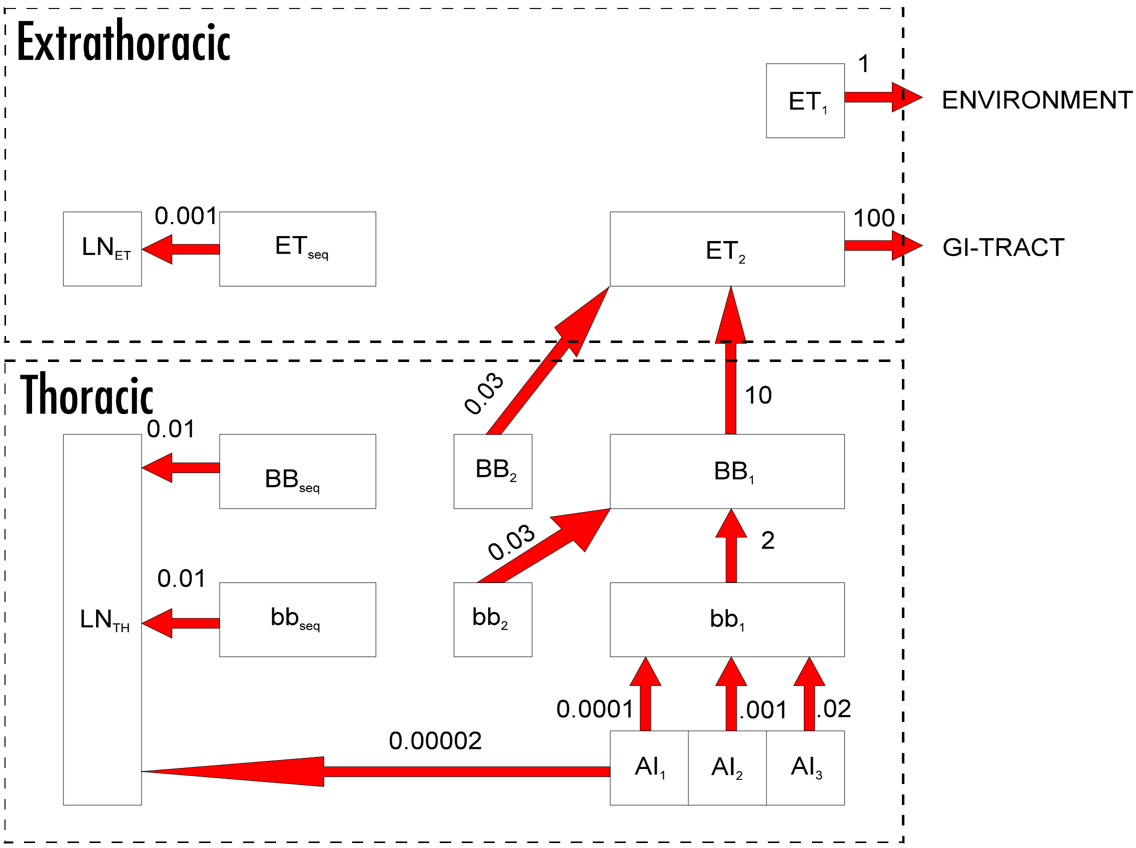

Fig. 2. Particle transport in the HRTM (ICRP Publication 66). Transfer rates are in units of d^{-1}.

Absorption

Absorption to blood depends on the chemical nature of the element and the chemical form deposited but is assumed to be independent of deposition site (with the exception of ET_1). The model allows for changes in dissolution and absorption to blood with time. The use of material-specific dissolution rates is encouraged but defaults are given; i.e. Types F (fast), M (medium) and S (slow). Type F has a single clearance component, whereas Types M and S have two components. Expressed as approximate half-times, these absorption rates are: Type F - 10 min (100%); Type M - 10 min (10%), 140d (90%); Type S - 10 min (0.1%), 7000d (99.9%).

For the example of a 1 µm AMAD aerosol, the total amounts reaching blood from the respiratory tract as predicted by the HRTM are 24% (F), 9% (M) and 1% (S) of the total inhaled. In each case, a further fraction of the inhaled material reaches blood by absorption from the gut after mechanical clearance and swallowing; this fraction will be greatest for Type F materials (most soluble) and least for Type S materials.

Dosimetry of the respiratory tract

In the respiratory tract model (as well as in models for the bone and gastro-intestinal tract), special attention is given to the locations of both source tissues (where radionuclides reside) and targets (tissues whose cells are thought to be at risk of cancer). Radionuclides are known to be deposited in the fluids lining the airways or alveoli, from where they can be absorbed to blood, perhaps residing in the airway wall for some time, or cleared to other regions of the lung by the action of mucus and cilia. There is good evidence that the target cells for cancer induction are the basal and secretory cells in the epithelia of the conducting airways, and certain cells of the alveolar region such as the Clara cells. To calculate doses to the sensitive tissues, cylindrical models of the airways are used, with the various cells modelled as concentric layers. Calculations are performed using Monte Carlo simulations of radiation transport. To calculate the dose to the lungs as a whole, a weighted average of the doses to the tissues is determined, independently of their relative masses. In the current model, the tissues are given equal weighting. For calculations of effective dose, the tissue weighting factor for the lungs (w_T = 0.12) is apportioned equally to the AI, BB and bb regions (i.e. 0.04 each).

Table 1. Committed doses from inhalation of radionuclides (1 µm AMAD aerosol) by adults (Sv Bq^{-1} intake).

Nuclide	Principal emissions	Half-life	Absorption Type	Equiv. Dose to lungs	Effective dose	% lung dose [1]
^{131}I	β,γ	8d	F	6.0×10^{-11}	7.4×10^{-9}	<1
^{137}Cs	β	30y	F	4.3×10^{-9}	4.6×10^{-9}	11
^{210}Po	α	138d	M	2.6×10^{-5}	3.3×10^{-6}	95
^{238}U	α	4.5×10^{9}y	M	2.2×10^{-5}	2.9×10^{-6}	91
^{239}Pu	α	2.4×10^{4}y	M	3.3×10^{-5}	5.0×10^{-5}	8
^{239}Pu	α	2.4×10^{4}y	S	8.7×10^{-5}	1.6×10^{-5}	65

1. % of effective dose contributed by equivalent dose to lungs

Table 1 gives examples of committed equivalent doses to the lungs and committed effective dose. The lung dose varies from 9 x 10^{-5} Sv Bq^{-1} for insoluble compounds of ^{239}Pu (i.e. Type S) to 6 x 10^{-11} Sv Bq^{-1} for ^{131}I, i.e. by over five orders of magnitude, reflecting the different retention times in the lung and the energy and type of radiation emitted.

Similarly, the contribution of lung dose to the effective dose varies from high values of about 90% (^{210}Po, ^{238}U) to a value of <1% for ^{131}I. The rapid absorption of Type F materials from the lungs to blood makes lung dose less important compared to doses to other tissues - the thyroid in the case of ^{131}I. For insoluble ^{239}Pu the lung dose dominates the effective dose (65%) due to its long retention in lung, whereas this is not true for more soluble Type M forms (8%) since in this case doses to systemic tissues are higher.

ICRP Human Alimentary Tract Model

Until 2007, the model of the gastrointestinal tract used by ICRP was that described in Publication 30 (ICRP, 1979). This is a simple model, containing four compartments in a linear chain representing the stomach, small intestine and upper & lower large intestines. In 2007, the Publication 30 model was superseded by ICRP's Human Alimentary Tract Model, HATM (ICRP, 2007), and future ICRP publications will use this model. However, current EU legislation contains dose coefficients generated using the Publication 30 model. This section describes the main features of the HATM, shown in Figure 3.

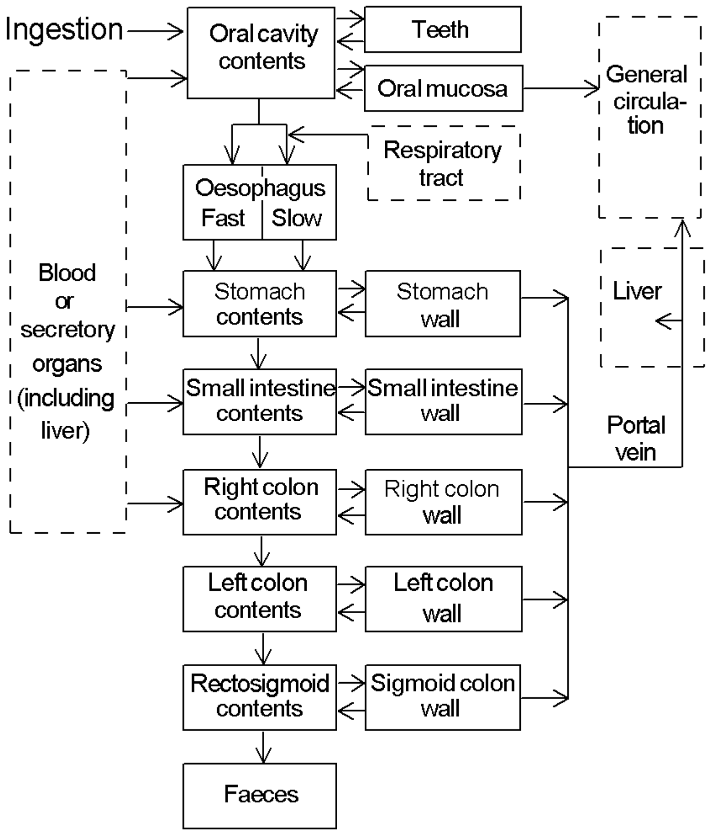

Fig. 3. Structure of the HATM. The dashed boxes are included to show connections with the respiratory tract model and systemic models.

The increased complexity reflects the wider scope and greater degree of realism of the HATM as compared to the Publication 30 model. The main features of the HATM are:

- Age-dependent parameter values are specified for the dimensions of alimentary tract regions and associated transit times of contents through the regions, as are gender-dependent parameter values of dimensions and transit times for adults.
- Retention of radionuclides in the mucosal tissues of the walls of alimentary tract regions is included. For some radionuclides, such as isotopes of iron and related elements, there may be significant retention in the intestinal wall.
- Explicit calculations are made of dose to target regions for cancer induction within each alimentary tract region. The sensitive cells for cancer induction are thought to be the stem cells in the base of the crypts which are likely to receive negligible doses from α-emitting nuclides in the gut lumen and lower doses for most beta emitters.
- All alimentary tract regions are included. Doses are calculated for the oral cavity, oesophagus, stomach, small intestine, right colon, left colon and recto-sigmoid (including the rectum). Colon doses are combined as a mass-weighted mean.
- Transit times are specified for food and liquids, as well as for total diet, for the mouth, oesophagus and stomach.

Table 2. Comparison [1] of ingestion dose coefficients derived using the HATM and ICRP Publication 30 models for adult males.

Nuclide	Equivalent doses [1]			Effective dose [1]
	Stomach	Right colon [2]	Left colon [3]	
^{90}Sr	1.0	0.2	0.1	-6%
^{106}Ru	1.2	0.2	0.1	-60%
^{239}Pu	0.8	0.4	0.2	0%

1. Equivalent doses compared as a ratio (HATM : Pub 30), effective doses as a fractional change (%).
2. Compared with the upper large intestine of the Publication 30 model.
3. Compared with the lower large intestine of the Publication 30 model.

Table 2 compares doses calculated using the HATM and Publication 30 models for three examples. For the beta-emitting nuclides ^{90}Sr and ^{106}Ru, the substantial reduction in colon doses in the HATM reflects reduced Specific Absorbed Fractions (SAF). For ^{239}Pu, there is no direct dose to alimentary tract regions using the HATM; all of the dose is due to absorbed ^{239}Pu deposited in soft tissues. The extent to which differences in equivalent doses to the tissues of the alimentary tract affect the effective dose depends on the relative contribution of these tissues to the effective dose. For ^{90}Sr, effective dose is dominated by skeletal doses (bone surfaces and red bone marrow) and there is little difference in the effective dose calculated using the HATM and Publication 30 models. For ^{106}Ru, doses to alimentary tract regions contribute significantly to effective dose with the result that the value obtained using the HATM is about half of that using the Publication 30 model. For ^{239}Pu, as for ^{90}Sr, doses to alimentary tract regions make only a small contribution to effective dose, the dominant contributors being liver and skeletal tissues.

Biokinetics and dosimetry for plutonium

The absorption of Pu and other actinide elements from the GI tract is very low; default fractional absorption (f_1 values) recommended by ICRP for adults are 10^{-5} for Pu oxide, 10^{-4} for Pu nitrate and similar compounds and 5 x 10^{-4} for other chemical forms and environmental forms of Pu. For inhalation, the default absorption assumption is Type M for Pu (Type S for Pu oxide).

After absorption to blood, the principal sites of deposition of Pu and related elements are the skeleton and liver. In Publications 56 and 67 (ICRP 1989, 1993) a physiologically-based age-dependent model was developed and applied to Pu, Am and Np and subsequently to Th and Cm (ICRP 1995a,b). The model is shown in Figure 4.

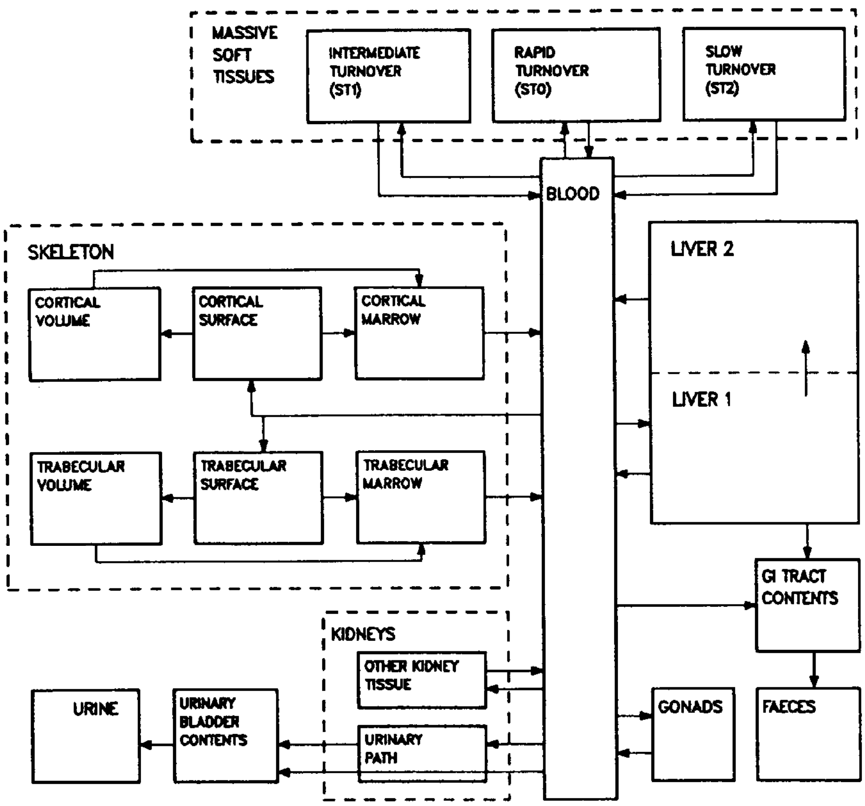

Fig. 4. Biokinetic model for the actinide elements: Pu, Am, Cm, Th, Np.

The radiation-sensitive parts of the skeleton are the red bone marrow, for leukaemia induction, and a layer of cells (the endosteal cells) on the inner bone surface, for induction of bone cancer. In adults, red bone marrow is contained in the trabecular or “spongy” bone which constitutes, for example, the vertebrae and the ends of the long bones. Cortical or compact bone, which constitutes the shafts of the long bones, contains only inactive fatty marrow in the adult. The model treats cortical and trabecular regions of the skeleton separately, with initial deposition on bone surfaces. A specific aspect of the biokinetics of Pu and related elements is that incorporation into bone volume results from burial surface deposits of new bone.

Pu removed from the bone surfaces may be retained in bone marrow. Accordingly, the model includes transfer to bone volume and to bone marrow. It also includes transfer from bone volume to bone marrow due to resorption. Activity is eventually released from bone marrow to blood and becomes available for recycling to all tissues including the skeleton. For simplicity, rates of bone growth and resorption are all taken to be the same and show the same age-dependence.

The liver has two compartments for all elements except Am and Cm, for which one compartment is used. Activity taken up by Liver 1 is transferred to Liver 2 except for a small proportion excreted in bile to the GI tract. Activity is lost to blood from Liver 2. Figures 5 and 6 illustrate the retention of Pu in the skeleton and liver predicted by the model for different ages at intake.

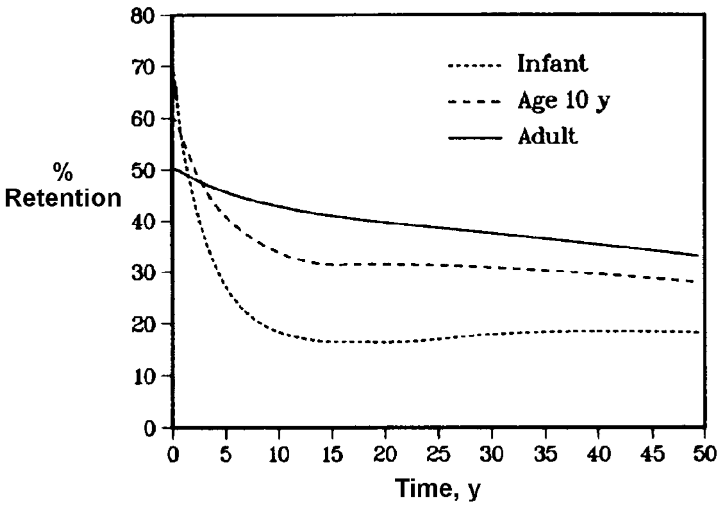

Fig. 5. Model predictions for the retention of Pu in the skeleton as a function of time after entry to blood (% total entering blood).

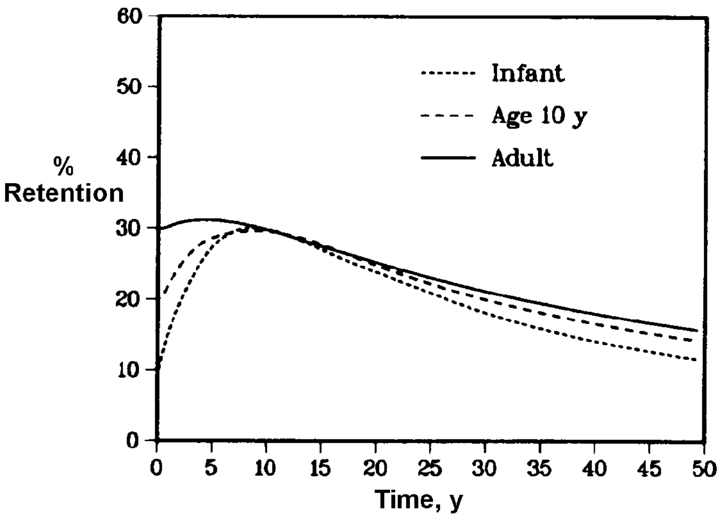

Fig. 6. Model predictions for the retention of Pu in the liver as a function of time after entry to blood (% total entering blood).

Table 3 shows dose coefficients for the ingestion or inhalation of ^{239}Pu. Committed effective doses after ingestion are due very largely to contributions from doses to skeletal tissues and liver with smaller contributions from doses to the gonads (testes or ovaries).

Table 3. Committed effective doses from the ingestion or inhalation of ^{239}Pu, 10^{-6} Sv Bq^{-1}.

Age	Ingestion *	Inhalation (1 µm AMAD)†	
		Type M	Type S
3 months	4	80	43
1 year	0.4	77	39
5 years	0.3	60	27
10 years	0.3	48	19
15 years	0.3	47	17
Adult	0.3	50	16

*Environmental forms; $f_1 = 5 \times 10^{-4}$, 5×10^{-3} for infants

‡Type M is default; Type S = e.g. Pu oxide

After inhalation, doses to the lungs are also important, particularly for Type S materials. Doses are lower after ingestion than inhalation because of the small amount absorbed to blood (most activity passes unabsorbed through the GI tract; >99%). Doses per unit intake are greatest at younger ages because of the smaller tissue masses, greater relative uptake in the skeleton (and greater f_1 value in infants).

Monitoring for ^{239}Pu is usually based on urine monitoring followed by mass- or alpha-spectrometry. Modern laboratories can detect levels less than a few mBq per litre. Faecal monitoring can also be used with some advantage, but there are often acceptability problems with workforces. Plutonium-239 emits 17 keV X-rays during alpha decay and these emissions can be detected by whole-body counting.

In general ^{239}Pu, like other alpha-emitters, presents problems for monitoring since intakes of a relatively small mass can lead to dose limits being exceeded. Excretion levels resulting from a small mass are low, and the non-penetrating nature of alpha emissions make detection difficult. In some cases the presence of other Pu isotopes, ^{241}Am, and fission products can make detection easier because of higher energy emissions but, of course, the composition of the mixture needs to be established.

Individual monitoring for internal contamination

The previous sections have described methods for calculating doses from a known intake received by a reference person using standard phantoms and biokinetic models. This is a prospective assessment of dose, because it answers the question “What would the dose be if a person receives an intake of x Bq?”. Usually a unit intake (1 Bq) is assumed and the results are known as dose coefficients (Sv Bq^{-1}) and are published in standard compendia (e.g. ICRP, 1994b). To obtain estimates of doses to individuals the dose coefficients must be multiplied by the known intake (Bq). For environmental intakes this can be derived from measurements of concentrations in foodstuffs (e.g. Bq kg^{-1}) and habit data giving the typical consumption of the foodstuff in a set time period.

In occupational settings however, workers are monitored on a regular basis and the results of measurements on their bodies or excreta can be used to estimate intake. This is a retrospective assessment of dose, because it answers the question "What is our best estimate of the dose, give that x Bq (or y Bd d^{-1}) has been measured in the lungs (or in urine, etc)?". This section gives general information on the design and use of these methods; it is taken largely from ICRP Publication 78 (ICRP, 1997).

Methods of Individual Monitoring

The purpose of this section is to describe briefly the main measurement techniques, their advantages and their limitations. In most cases, individual monitoring for intakes of radionuclides may be achieved by body activity measurements, excreta monitoring, air sampling with personal air samplers, or a combination of these techniques. The choice of measurement technique will be determined by several factors: the radiation emitted by the radionuclide; the biokinetic behaviour of the contaminant; its retention in the body taking account of both biological clearance and radioactive decay; the required frequency of measurements; and the sensitivity, availability, and convenience of the appropriate measurement facilities.

Routine monitoring programmes usually involve only one type of measurement if adequate sensitivity can be achieved. For some radionuclides, only one measurement technique is feasible, e.g. urine monitoring for intakes of tritium. For radionuclides such as plutonium isotopes that present difficulties for both measurement and interpretation, a combination of techniques should be employed. If different methods of adequate sensitivity are available, the general order of preference in terms of accuracy of interpretation is: body activity measurements; excreta analysis; personal air sampling. Results of monitoring of the working environment (area monitoring) may provide information that assists in interpreting the results of individual monitoring, e.g. information on particle size, chemical form and solubility, time of intake.

The results of workplace monitoring for air contamination may sometimes be used to estimate individual intakes. However the interpretation of the results of measurements from air sampling in terms of intake is not simple and may be misleading. The most common form of representative sampling is by using fixed samplers at a number of selected locations intended to be reasonably representative of the breathing zone of the worker.

Monitoring in relation to a particular task or event may often involve a combination of techniques so as to make the best possible evaluation of an unusual situation, for example, a programme of both body activity and excreta measurements.

In some cases of suspected incidents, screening techniques (such as measuring nose blow samples or nasal smears) may be employed to give a preliminary estimate of the seriousness of the incident.

In vivo measurements

The IAEA (IAEA, 1996) has given guidance on the direct measurement of body content of radionuclides. Direct measurement of body or organ content provides a quick and convenient estimate of activity in the body. It is feasible only for those radionuclides emitting radiation that can escape from the body. In principle, the technique can be used for radionuclides that emit: X or γ radiation; positrons, since they can be detected by measurement of annihilation radiation; energetic β particles that can be detected by measurement of bremsstrahlung; and some α-emitters that can be detected by measurement of the characteristic X rays.

Many facilities for the measurement of radionuclides in the whole body or in regions of the body consist of one or a number of high efficiency detectors housed in well-shielded, low-background environments (IAEA, 1996). The geometrical configuration of the detectors is arranged to suit the purpose of the measurement, e.g. the determination of whole-body activity or of activity in a region of the body such as the thorax or the thyroid. The skull or knees may be used as a suitable site for measurement of radionuclides in skeleton.

Care must be taken to remove surface contamination before body activity is measured. For routine measurements, determination of whole-body content is often adequate for radiological protection purposes.

Commonly encountered fission and activation products, such as ^{131}I, ^{137}Cs and ^{60}Co, can be detected with comparatively simple equipment at levels that are adequate for radiation protection purposes. Such simple equipment may consist of a single detector, viewing the whole body or a portion of the body, or, for iodine isotopes, a small detector placed close to the thyroid. The advantage of simple equipment is that it may be operated at the place of work, thereby avoiding the time required to visit a remote whole-body monitoring facility. Measurements may then be made more frequently so that any unusually large intake would be recognised soon after it had occurred.

In contrast, high sensitivity techniques are needed for monitoring a few radionuclides at the levels that are required for protection purposes. Examples are the α-emitting radionuclides such as plutonium isotopes.

The activity present in a wound can be easily detected with conventional β-γ detectors if the contaminant emits energetic γ-rays. In the case of contamination with α-emitting radionuclides, detection is more difficult since the low energy X rays that follow the α-decay will be severely attenuated in tissue; this effect is more important the deeper the wound.

Analysis of Excreta and other Biological Materials

In some cases, excreta monitoring may be the only measurement technique for those radionuclides which have no γ-ray emission or which have only low energy photon emissions. Excreta monitoring programmes usually involve analysis of urine, although faecal analysis may be required in some circumstances, for example where an element is preferentially excreted via faeces or to assess clearance of insoluble material from the respiratory tract. Other samples may be analysed for specific investigations. Examples are nose blow or nasal smears as routine screening techniques or blood, in the case of suspected high level contamination.

The collection of urine samples involves three considerations:

- Firstly, care must be taken to avoid adventitious contamination of the sample.
- Secondly, it is usually necessary to assess the total activity excreted in urine per unit time from the sample provided. For most routine analyses, a 24 hour collection is preferred but, if this is not feasible, it must be recognised that smaller samples may not be representative. (Tritium is a particular case for which it is

usual to take only a small sample and to relate the measured activity concentration to the concentration in body water.)

- Thirdly, the volume required for analysis depends upon the sensitivity of the analytical technique. For some radionuclides, adequate sensitivity can be achieved only by analysis of several days' excreta.

Radionuclides that emit γ-rays may be determined in biological samples by direct measurement with scintillation or semiconductor detectors. Analysis of α- and β-emitting radionuclides requires chemical separation followed by appropriate measurement techniques. Measurement of so-called total α or β activity may occasionally be useful as a simple screening technique, but there is no method that will determine accurately all the α and β activity in the sample. The technique may be used in routine monitoring situations where intakes are expected to be very low compared with annual limits. The results would not be interpreted quantitatively, but would be used to provide confirmation of satisfactory conditions, an unusual result indicating the need for further investigation which would include radiochemical analysis. Total activity measurements may also be useful following a known contamination event or to identify those samples that merit early attention. Measurements of total α or β activity cannot be used in quantitative evaluations of intake or committed effective dose, unless the radionuclide composition is known.

Measurement of activity in exhaled breath is a useful monitoring technique for some radionuclides. In practice this method may be useful for ^{226}Ra and ^{228}Th since the decay chains of both these radionuclides include gases which may be exhaled. It can also be used to monitor $^{14}CO_2$ formed *in vivo* from the metabolism of ^{14}C-labelled compounds.

Air Sampling

A Personal Air Sampler (PAS) is a portable device specifically designed for the estimation of intake by an individual worker from a measurement of time-integrated concentration of activity in air in the breathing zone of the worker. A sampling head containing a filter is worn on the upper torso close to the breathing zone. Air is drawn through the filter by a calibrated air pump carried by the worker. Ideally, sampling rates would be representative of typical breathing rates for a worker (~1.2 m3 h^{-1}). However, sampling rates of current devices are only about 1/10 of this value. The activity on the filter may be measured at the end of the sampling period to give an indication of any abnormally high exposures. The filters can then be retained, bulked over a longer period, and the activity determined by radiochemical separation and high sensitivity measurement techniques. An estimate of intake during the sampling period can be made by multiplying the measured integrated air concentration by the volume breathed by the worker during the period of intake.

A PAS does not provide information on particle size. Nevertheless, it is important either to determine the particle size distribution of the inspirable material or to make realistic assumptions about it, since it can have a marked effect on deposition fractions in the respiratory tract, and hence on dose estimates. All samplers are size selective to a greater or lesser extent, under- or over-sampling at particular particle sizes, and this can result in errors in intake estimation. The aspiration efficiency of a PAS should therefore be determined to indicate whether corrections are necessary.

Static air samplers (SAS) are commonly used to monitor workplace conditions, but can underestimate concentrations in air in the breathing zone of a worker, typically by a factor of up to about 10. Nevertheless, if SAS devices are sited appropriately, a comparison of PAS and SAS measurements can be used to define a PAS:SAS air concentration ratio which can be used in the interpretation of SAS measurement for dose assessment purposes. It should however be recognised that the use of SAS is a relatively indirect method for assessing doses, and use of the results to estimate individual dose requires a careful assessment of exposure conditions and working practices. Apart from their potential use for dose estimation, SAS devices can also provide useful information on radionuclide composition, and on particle size if used with a size analyser such as a cascade impactor.

Estimation of Intake and Dose

In an investigation of an incident for which the time of intake is known, the intake can be estimated from the measured results using the values of measured quantities predicted by the biokinetic models for a unit intake. If only a single measurement is made, the intake can be determined from the measured quantity, M, by:

$$Intake = \frac{M}{m(t)}$$

where m(t) is the predicted value at the time of intake t. The intake can be multiplied by the dose coefficient to give the committed effective dose; this can then be compared with the dose limit or some investigation level based on dose.

If the measurement indicates that an investigation level has been exceeded, further investigation is required. The nature of the investigation will depend upon the circumstances and the extent to which the investigation level is exceeded. The following should be considered:

- repeated measurements to confirm or refine the initial evaluation, and
- the use of additional monitoring techniques.

The predicted values can then be scaled to obtain the best fit to the measured data points. The best fit is usually taken to be that fit which minimizes the sum of the squares of the residuals, a residual being defined as the number of standard deviations separating a measurement from the fitted curve. The intake is then equal to the value by which the predicted values are scaled. Many laboratories have developed software for fitting the intake in this, and other more mathematically advanced, ways, for example HPA's IMBA program.

When a positive measurement is detected by routine monitoring, say at monthly intervals, certain assumptions have to be made about the time of intake when calculating doses. It is often assumed that intake took place in the middle of the monitoring interval of T days. If the dose is large further measurements would be necessary. It should be noted that an intake in a preceding monitoring interval may influence the actual measurement result obtained.

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Optimisation of radiation protection for pediatric and adult patients in radiography and computed tomography

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Introduction

The need for optimization of radiation protection of patients in diagnostic medical imaging became clear already some 30 years ago. Scientists that pioneered in radiation protection observed that large variations in radiation exposure of patients existed for one and the same diagnostic examination. This triggered many national and international field studies. The field studies that have been carried out in the United Kingdom are particularly well known, and the observed steady reduction of third quartile adult patient dose values from three reviews of UK national patient dose data for the mid-1980s; the year 1995; and the year 2000 illustrates that improvements in acquisition protocols and imaging equipment resulted in substantial optimization of radiation protection [Wall]. The need for optimization of radiation protection of young patients became evident when the number of clinical indications for CT examination of children grew rapidly mainly due to the enhanced speed of the CT scanners around the year 2000. It was observed in the United States that many CT scans of children were performed with acquisition protocols that were developed for adults, and that used too high exposures for small children [Brenner et al.]. This led to special efforts worldwide in optimization of pediatric CT examinations; the Alliance for Radiation Safety in Pediatric Imaging of the United States of America (USA) encourages on its website [www.pedrad.org] increased awareness of opportunities to lower radiation dose in pediatric CT procedures. A recently published field study in the USA revealed CT that as the major contributor to the population dose [Mettler et al.]. In 2006, Americans were exposed to more than seven times as much ionizing radiation from medical procedures as was the case in the early 1980s. Radiation exposure due to medical examinations in the United States is now assessed at about 3 mSv per year, the USA is the first country were the population exposure is dominated by exposures from diagnostic imaging, whereas in other countries the population exposure is dominated by exposure to natural sources of ionizing radiation (Figure 1). The high levels of radiation exposures from CT as were observed in the USA are not (yet) observed in Europe. Nowadays the manufacturers of CT scanners offer improved opportunities for optimization of radiation protection in CT examinations; the biggest challenge for the future in optimization of CT seems now to be achieving proper (evidence based) justification of the referral for CT examinations. It has to be established which CT examination really contribute to the proper clinical care of patients, and which referrals are mainly based on too defensive diagnostic strategies.

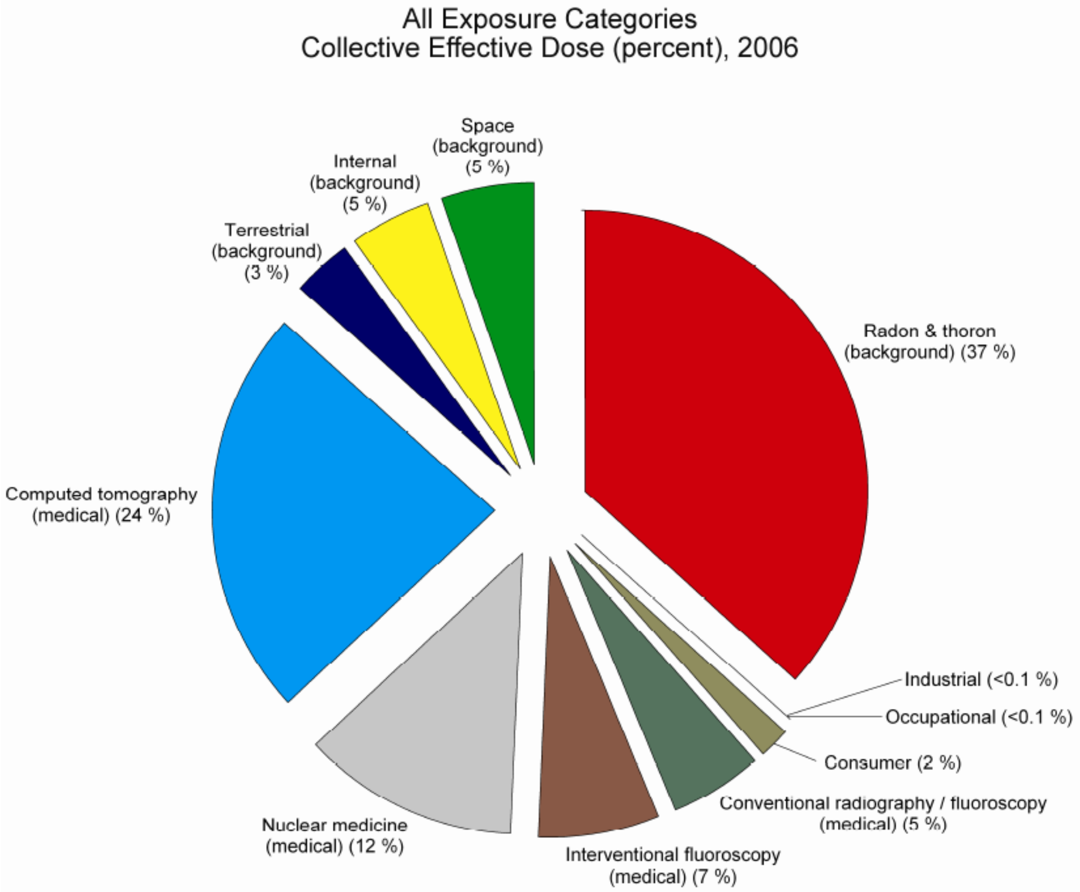

Fig. 1. The distribution of the population dose in the United States. Note that CT is a major contributor to the population dose. In 2006, Americans were exposed to more than seven times as much ionizing radiation from medical procedures as was the case in the early 1980s. Radiation exposure due to medical examinations in the United States is now about 3 mSv per year [Mettler et al.].

Optimization in radiography

X-ray projection radiography is a well known, widely available, and relatively cheap imaging procedure for medical diagnosis; dental X-ray radiography is the most frequently performed examination, next comes chest x-ray radiography. Radiographs of the extremities (legs, arms, hand, feet), of the abdomen, and of the spine are also often made (Figure 2). Radiography of the skull is nowadays rarely performed, the skull is preferably examined with 3D imaging techniques like computed tomography of magnetic resonance imaging.

Radiographs were traditionally recorded on x-ray film. The latent image that is recorded by x-ray film became visible after developing the film. Digital detectors became available about 30 years ago. There were initially high expectations with regard to the improvement of image quality and the reduction of patient dose when the first systems for computed radiography (CR) were introduced. The latent image that is recorded on CR plates is translated into light in readout units by photo-stimulation with a laser. The generated light is then translated into an electrical signal, and the electrical signal into a digital image. Unfortunately in the early years of CR, the high expectations were not fulfilled. Users of CR systems often had to increase dose in order to achieve images with sufficiently good image quality. Advances in detector design made that CR finally evolved into a dose efficient alternative for x-ray radiography on film. Systems for digital radiography (DR) became available somewhat later, these systems use a detector that both records the image, and that translates the image into an electrical signal. DR detectors generally provide the best dose efficiency in radiography, they can be operated at dose levels well below the doses that are used in radiography with film or CR. The opportunity for dose reduction that detectors like CR and DR offer is not the only advantage. It is nowadays generally recognized that, compared to film, CR and DR systems are capable to provide better image quality, and that with these systems less retakes are required. Digital archives and fast networks make it possible to store and distribute digital images much more efficiently compared to film.

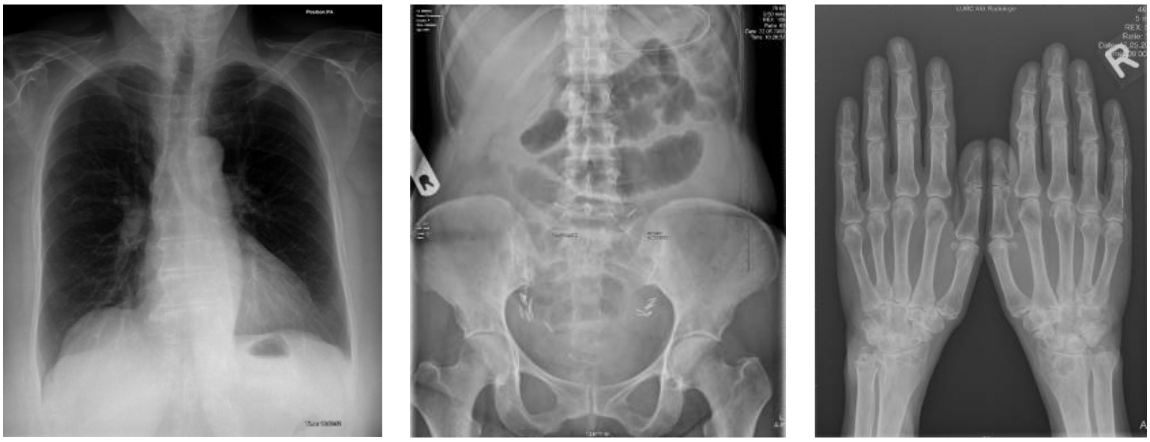

Fig. 2. Three examples of radiography, from the left to the right radiography of the chest (frontal view), radiography of the abdomen (frontal view), and radiography of the hands. Note that radiography yields an superposition in a 2D plane of all tissues within the examined area.

Acquisitions of radiographs have to be optimized; this can be achieved by optimizing the x-ray spectrum. The ‘color’ of the spectrum (or the distribution of photons of different energies in the spectrum) can be optimized by choosing the appropriate tube voltage (kV) and tube filtration (in addition to the inherent filtration by the window of the x-ray tube, additional aluminium and copper filters are being used). Spectra with photons with relative low energies can be used for imaging relatively thin objects (extremities, children), spectra with higher photon energies are used for imaging thicker body parts (abdomen, spine). Very high photon energies are used in chest radiography to reduce the contrast between bone (ribs) and soft tissue (lungs), this is required since the ribs may, at lower photon energies, obscure the lesions in the lung soft tissue. The ‘brightness’ of the spectrum (or the intensity of the spectrum) can be optimized by selecting the most appropriate tube charge (mAs), which is the product of tube current (mA) and exposure time (s). Most x-ray units make use of automatic exposure control, in this case a radiation detector measures the dose that the detector receives, and when the required preset dose level is reached, the system terminates the exposure. It is necessary that the preset dose level is optimized to provide the best balance between the required image quality and at the same time the lowest possible dos for the patient. It is essential that the automatic exposure system of modern CR and DR systems is configured to operate well below the preset dose that would be used for recording images on film. Sometimes it is not possible to use automatic exposure control, for example when radiographs are being made with a mobile x-ray unit at the department of intensive care, in this case a table with appropriate and optimized acquisition techniques that cover all relevant examination should be available for the radiographer. In addition the field size should be optimized, by collimation of the x-ray beam the field size should be just sufficiently large to visualize the body part of interest.

It is also important to realize that for one clinical question; often more that one radiograph is being made. For chest radiography, generally two radiographs are being made; one provides a frontal view, the other a lateral view of the patient. For some clinical referrals, it has been observed that great variety exists in the number of the radiographs that are being taken, and in the views of these radiographs, for example when imaging the cervical spine, the abdomen, or the pelvis. These large variations in clinical practices and acquisition protocols provide a substantial, and until now hardly exploited, opportunity for optimization in x-ray radiography, Figure 3 illustrates that for one and the same clinical question (cervical spine, neck pain), and in only 11 hospitals, eight different views were being used in routine practices, and the numbers of views varied between two and five [Teeuwisse et al.].

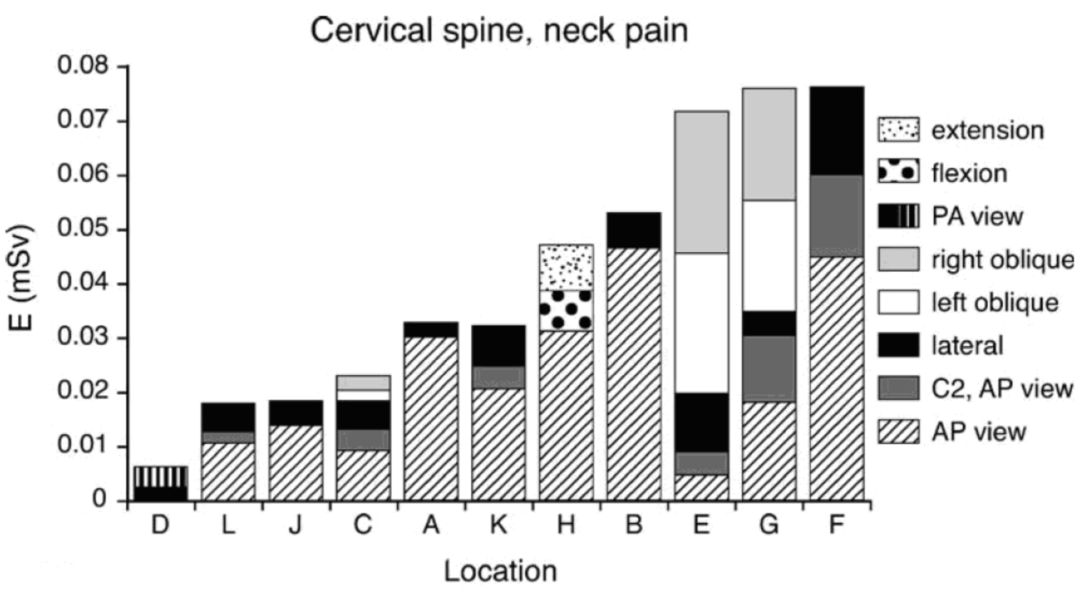

Fig. 3. Cumulative effective dose (mSv) for the clinical radiography indication cervical spine, neck pain illustrates the impact of the clinical protocol (the number and type of views) and the dose for the different views on the spread in integral patient dose [Teeuwisse et al.].

Dosimetry in radiography is usually based on measurement of the dose-area product (DAP, Gy.cm^2), this is a useful quantity, since the DAP incorporates the effect on patient dose of all relevant acquisition parameters like tube voltage, tube charge, and field size. DAP can be converted into effective dose by using appropriate effective dose conversion factors or preferably by dedicated software (PCXMC, Radiation and Nuclear Safety Authority in Finland). The so-called DICOM header that is added as supplementary information to the digital images, generally also stores the dose-area product. Ideally, the dose-area product could be retrieved with a simple query from the digital archive in which the radiographs are being stored; however, unfortunately such a functionality is not always available. European countries defined diagnostic reference levels for selected projections in x-ray radiography of adults; some countries also established diagnostic reference levels for pediatric radiography.

Optimization in computed tomography

Computed tomography (CT) was introduced in 1974. The substantial advances in CT technology for 3D imaging with excellent image quality led to a fast increase of its clinical use. CT is often applied for the diagnosis of tumours, vascular disease, and trauma. CT shows anatomy and pathology in arbitrary planes, or in 3D reconstructions (Figure 4). The excellent image quality of CT comes, compared to radiography, at the price of a substantial higher radiation exposure. CT is nowadays the major source of radiation exposure of patients which makes optimization (and justification) of applications of CT of particular relevance.

CT scanners are equipped with a rotating x-ray tube and detector. During one rotation, of the x-ray tube and detector, the early CT scanners were only capable of scanning one thick axial slice, these scanners were relatively slow, they provided good image quality in the axial plane, but the 3D image quality was poor. Advances in CT technology like multislice CT (CT scanners that were able to scan more than one image in one rotation) and helical CT (CT scanners that combined the rotation of the x-ray tube and the detector with translation of the patient) made CT an excellent imaging modality for fast 3D imaging. With the introduction of volumetric CT (CT scanners that allow for imaging 320 slices during one rotation), entire organs like the brain or the heart can be imaged within a fraction of a second.

Optimisation of CT was mainly enhanced by technical improvements, but can also be achieved by adapting optimal imaging protocols. An important step in the improvement of the dose efficiency of CT scanners was achieved by the transition from air filled CT detectors to solid state CT detectors. Another important step was the introduction of automatic exposure control in CT, this allowed for adapting the tube current for the size of the patient but also for the local attenuation of the patient. To achieve consistent image quality within the scanned range, CT scanners nowadays automatically use higher exposure in thick patients and lower exposure in thin patient. During the CT scan the exposure can also be adapted to the local attenuation, during a CT of the chest for example the tube current is higher when the CT scan passes the shoulder (with its highly attenuating bones) and lower when it scans the lungs. Sophisticated noise reduction algorithms contribute further to the dose efficiency of CT scanners. CT scans can be optimized by carefully selecting the most appropriate preset for the automatic exposure control, which determines the tube charge (mAs) during the scan. Figure 5 shows the differences in image quality within the brain when different presets of the tube current are being used. Experiments to establish the optimal tube current cannot be performed by scanning the patient several times but such experiments have to be performed using low dose simulation algorithms and observer studies

[Joemai et al.]. CT scans can be further optimized by carefully selecting the most appropriate tube voltage, generally a tube voltage of 120 kV is applied in CT, for obese patients a higher tube voltage of for example 140 kV might be more appropriate, but for iodine contrast enhanced vascular studies a lower tube voltage of 100 kV or even 80 kV might be preferred.

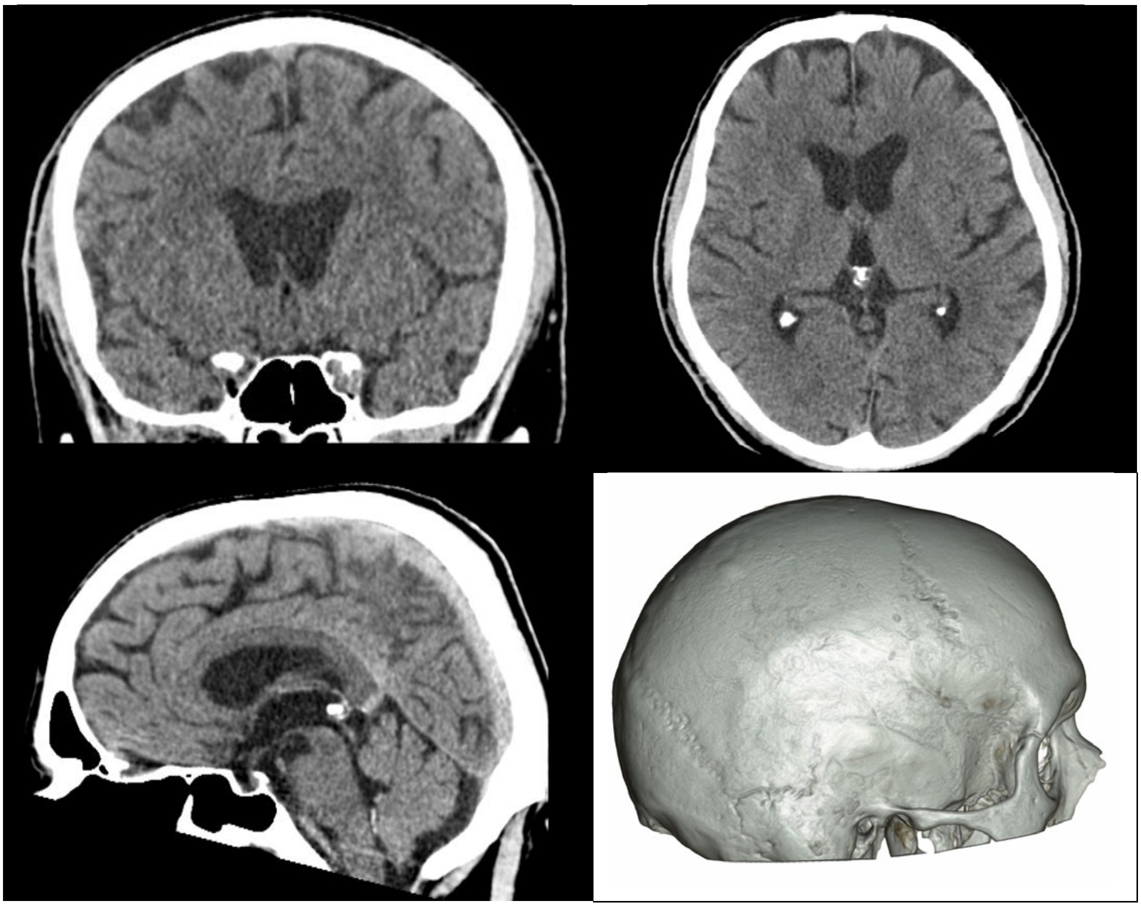

Fig. 4. Computed tomography allows for 3D imaging with superior low contrast resolution compared to radiography. Above respectively reconstructions of a CT head scan in a coronal, axial and sagittal plane; and a 3D surface rendering of the skull.

Dosimetry in CT is based on the concepts of the CT dose index (CTDIvol, mGy) and the dose-length product (DLP, mGy.cm). The CTDIvol represents the average dose that is absorbed during one rotation in the axial plane of either a head phantom or a body phantom (these are cylindrical PMMA phantoms of respectively 16 and 32 cm diameter and 15 cm length). The DLP is the product of the CTDIvol and the actually exposed range. The CTDI takes into account the effects on patient dose of tube voltage, rotation time of the x-ray tube, and tube current, but only represents the local dose within the scanned range. In addition, the DLP takes into account the extent of the exposed range, and therefore DLP represents best the patient dose. DLP can be converted into effective dose by using appropriate effective dose conversion factors or preferably dedicated software like the ImPACT CT dosimetry calculator (Shrimpton et al.).

Like in radiography the DICOM header in CT generally also stores the CTDIvol and DLP. Ideally, these quantities could be retrieved with a simple query from the digital archive in which the CT scans are being stored; however, unfortunately such functionality is not always available. European countries defined diagnostic reference levels for selected acquisitions projections in CT of adults; some countries also established diagnostic reference levels for pediatric CT.

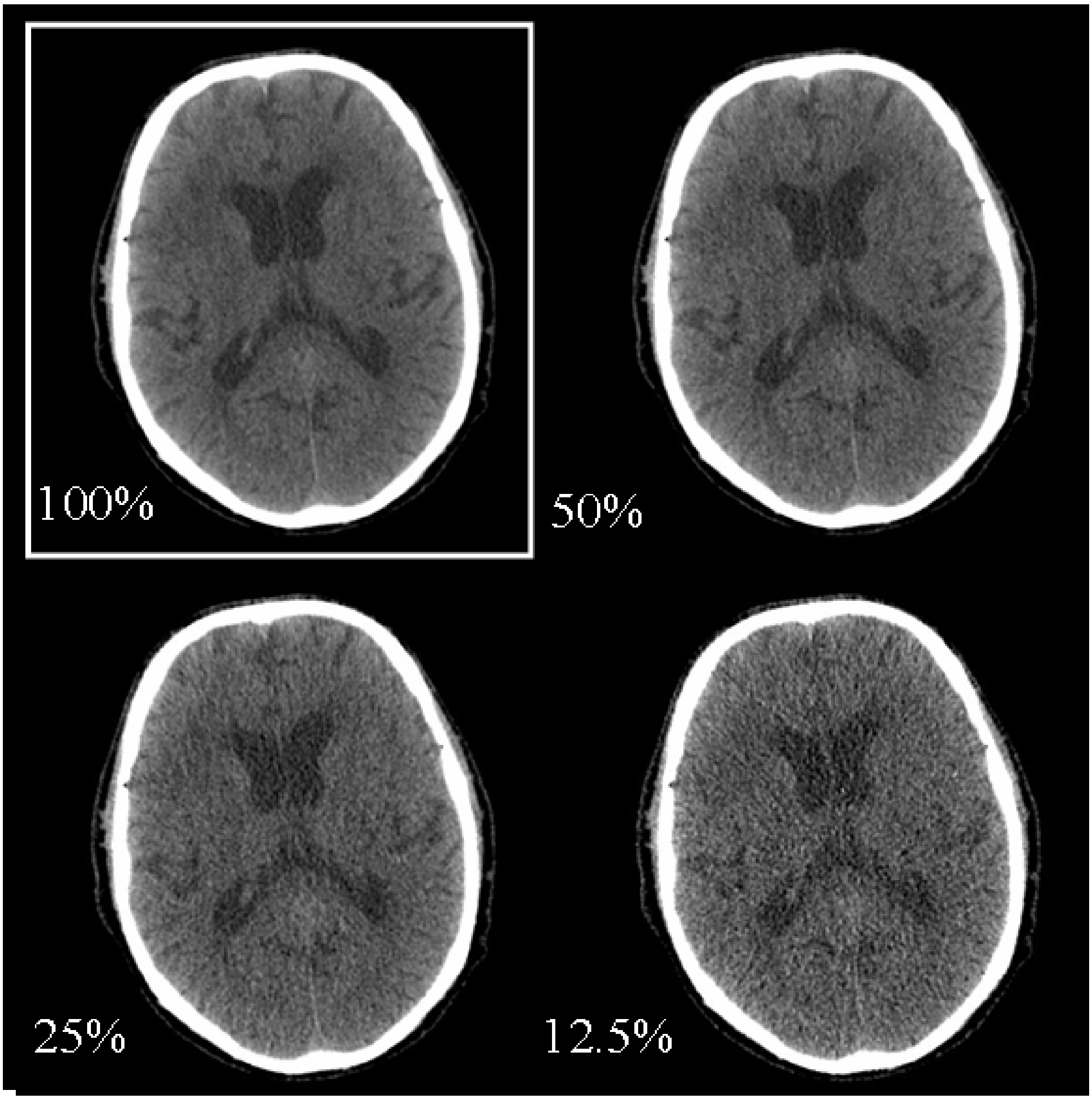

Fig. 5. Four axial images of the brain. The 100% image is acquired as a clinically indicated scan. The other images represent the image quality that would be achieved when a 50%, 25%, and 12.5% lower tube current would have been used. These brain images at reduced dose levels are not scanned images but they are simulated to using a dedicated software algorithm [Joemai et al.]. Note the increased noise in the low dose images. Such low dose simulations allow for studying the image quality of low dose acquisition protocols in observer studies without the need of repeating the CT examination on patients.

Clinical exposures

Table 1 provides an indication of the radiation exposure from natural and medical exposures. The yearly dose limits can be regarded as a frame of reference. For most radiographs of adults, exposures are well below 0.1 mSv; and of the order of magnitude of natural exposures like one week skiing or an intercontinental flight. Exposures from radiographs of extremities are extremely low; radiographs of the abdomen are relatively high and may be several tenths of a millisievert. For CT exposures can be several millisieverts, but remain in general below 10 mSv; this is true for both CT scans of adults and for proper optimized acquisition protocols for pediatric CT.

Table 1. Typical effective doses for exposures to natural and medical sources of ionizing radiation.

Natural exposures and dose limits	
One week skiing	0.015 mSv
Intercontinental flight (London-Los Angeles)	0.08 mSv
Population exposure to natural sources of radiation per year	2.5 mSv/year
Public dose limit per year	1 mSv/year
Occupational dose limit per year	20 mSv/year
Medical exposures radiography, adults	
Dental radiography	< 0.001 mSv
Extremities (e.g. knee, ankle, or elbow)	< 0.001 mSv
Chest, frontal view	0.02 mSv
Chest, lateral view	0.04 mSv
Pelvis, frontal view	0.3 mSv
Abdomen, frontal view	0.5 mSv
Medical exposures CT, adults	
CT head, stroke	1 mSv
CT chest, pulmonary embolism	4 mSv
CT spine, fracture L1	4 mSv
CT abdomen, abscess	8 mSv
Medical exposures, pediatric	
CT head, follow up of hydrocephalus, 4-6 year old child	1 mSv
CT head, battered child or trauma, 1-12 months old child	2 mSv
CT chest, bronchiectasis, 4-6 year old child	3 mSv
CT chest, congenital abnormality, 1-12 months old child	4 mSv

Future developments

Further optimization of diagnostic medical imaging with x-rays can be expected. This will be mainly achieved by technical improvements of current technologies and the development of new technologies. A recent and particular dose efficient technology for detecting x-rays, by photon counting detectors, is already implemented in mammography and may find its way to other imaging modalities. In computed tomography the development of new reconstruction algorithms may allow for lower dose acquisitions for certain, yet to be established, clinical applications. In cardiac CT examinations an enormous reduction of patient dose becomes available for scanners that allow for prospective, ECG triggered, scanning of the heart. Sliding collimators that optimize the beam width at the start and the end of a helical acquisition contribute to the optimization of helical CT acquisitions. Volumetric CT, which allows for scanning entire organs in one single rotation allows for dose reduction for certain acquisition protocols [Kroft et al.]. Tomosynthesis, which is already a clinical application in mammography, overcomes to a certain extend the problem of overprojection in projection mammography, and is expected to be introduced also for chest radiography. Compared to CT tomosynthesis can be performed at relatively low doses, similar to doses in radiography, but does not yield the excellent 3D image quality of CT. However, dedicated ultralow dose acquisitions in CT, combined with dedicated reconstruction and noise reduction algorithms, may provide better performance compared to tomosynthesis. Dedicated low dose CT of the breasts has for example been developed, and it offers the opportunity to scan the breast in real 3D, with superior image quality compared to projection mammography or tomosynthesis of the breast, and at the same dose of projection mammography [Boone et al.]. Low dose cone beam CT scanners that are available for dental applications may also provide opportunities for low dose scanning of other parts of the skull like the inner ear.

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Radiation epidemiology

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Radioecology and environmental exposure pathways

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Abstract
Radioecology ("radiation ecology") is the scientific field encompassing the relationships between ionizing radiation or radioactive materials and the environment or subunits thereof. Radioecology and its study constitutes an important component of radiological protection of both humans and environment mainly, although not solely, through its relevance in understanding and describing environmental exposure pathways and quantifying radionuclide transport along them. Such pathways can be described as the route radioactive substances take from their source to their end point and how humans or biota can be exposed to the substance. This lecture introduces the basics of radioecology and discusses the role of radioecology with respect to environmental exposure pathways for radioactive contaminants. It presents the fundamental concepts of radioecology, the tools used by radioecologists with respect to the study of exposure pathways and future directions of research in the field.

Radioecology and environmental exposure pathways
Radioecology is the term used to describe the scientific discipline concerned with the relationship between ionizing radiation and radioactive substances and the environment or its constituents. These constituents may be populations, communities, ecosystems or biomes and include both humans and biota. Radiobiology on the other hand differs from radioecology with respect to its focus on the effects of ionizing and non-ionizing radiation on biological systems – from the molecular to the individual organism. Three main scientific areas are of interest with respect to radioecology, these being:

1. Radionuclide movement and accumulation in environmental components such as soil, water and biota and within ecosystems;
2. The effects of ionizing radiation upon species, populations, communities and ecosystems;
3. The use of radionuclides and ionizing radiation in understanding the structure, form and function of ecosystems and the constituent parts of those ecosystems.

The pursuit of radioecology as a discipline can be traced back to the early 1940's with the formation of the Applied Fisheries Laboratory in the United States, the purpose of which was to elucidate the effects of releases of radioactive substances to the Columbia River as a result of operation of reactors at the Hanford site (see Whicker and Schultz, 1982). Since then radioecology has matured as a science, that maturation having occurred, somewhat sporadically, during periods such as the early years of nuclear power, the decades of nuclear weapons testing and in the aftermath of accidents such as those at Chernobyl in 1986 and Kyshtym and Windscale in 1957.

In recent years radioecology has addressed new challenges such as those afforded by non-nuclear industries, the nuclear legacies of many countries and the increased focus on protection of the environment as well as man. As radioecology has grown as a discipline, the fundamental set of questions that its pursuit seeks to answer has been distilled to perhaps four that serve to adequately describe the scientific challenges facing radioecology:

1. How, at what rate and to what extent do radionuclides move between parts of an ecosystem and what are the mechanisms and pathways by and along which such transfers occur?
2. What are the concentrations and resultant doses from radionuclides in ecosystem components in relation to the overall contamination levels in the environment?
3. What are the long-term effects and behaviour of these contaminants in the environment?
4. What environmental concentration of radionuclides will result in effects at population level?

Intrinsically linked to these four fundamental questions, and therefore to the study of radioecology is the concept of the *environmental exposure pathway*.

Environmental exposure pathways

An environmental exposure pathway can be defined as the path a substance travels from its source to its environmental endpoint and how people or biota can be exposed to that substance. In essence, environmental exposure pathways can be considered as consisting of five common components:

1. the source of the contamination (e.g. point discharge from a power plant, fallout, abandoned source, etc.);
2. an environmental media and transport mechanism (e.g. soil – soil water, physical transport, bioturbation; water – transport as dissolved substance, particulate transport etc.);
3. an exposure point (i.e. a geographic or temporal point at whicgh the exposure actually occurs);
4. a route of exposure (e.g. eating contaminated food or prey, drinking contaminated water, inhalation of contaminated soil or dust)
5. a receptor population (e.g. rabbits living in contaminated soil, people eating contaminated fish, etc.).

Consideration of these five components of environmental exposure pathways indicates that the nature of the pathway is to a large extent governed by the nature of the environment in which the contaminant is present and the receptor population being considered. Consequently there are a large number of potential exposure pathways (see Figure 1) some of which are generally more dominant than others and some of which are relatively specific to certain isotopes, geographic regions, areas or populations. For the purposes of elaboration it is therefore convenient to breakdown the discussion of environmental exposure pathways with respect to environment "types" – terrestrial, marine and freshwater.

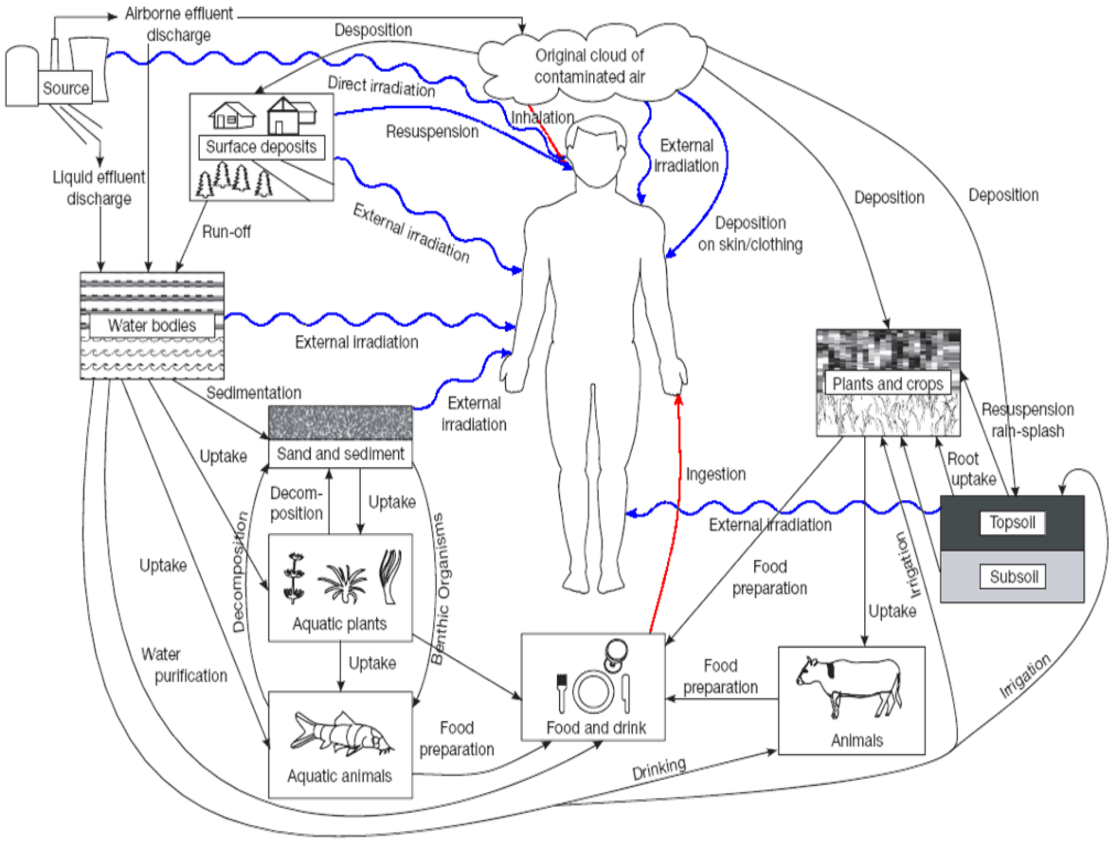

Fig. 1. Potential environmental exposure pathways for radioactive substances.

The isotopes of most interest with respect to radioecology and exposure pathways have varied over time and with respect to which source terms were of most relevance. In general those isotopes produced by fission (controlled or uncontrolled) or nuclear cycle processes and which have a half-life long enough to represent a potential impact on humans or the environment dominate radioecological studies. Typically the suite includes isotopes of plutonium, cesium, strontium, iodine, technetium, uranium, radium, thorium, carbon, hydrogen and cobalt among others.

Terrestrial exposure pathways

Although terrestrial environments can be broken down with respect to climate etc., the main mechanisms and processes involved in governing terrestrial exposure pathways are the same for all although the pathways may differ in their relative importances. As an example, wind resuspension and inhalation of contaminated soils may be an exposure pathway common to both European agricultural ecosystems and tropical deserts but will probably be of greater significance in the latter. Numerous transport processes are of relevance with respect to the terrestrial environment and can include transport by groundwater, particulate transport with soil particles, resuspension of soils and dusts by wind or rain, runoff/snowmelt etc. The most direct exposure pathways for terrestrial environments include direct exposure of the organism to external radiation from deposited radionuclides or ingestion/inhalation of contaminated soils or dusts. Uptake of radionuclides by plant species with subsequent transfer to humans or biota via the food chain is perhaps the exposure pathway that has received most attention over the decades.

Radioecology has played an important role in the study of terrestrial exposure pathways although the years up to the early 1990s were primarily focussed on studies regarding human exposure. This resulted in much information dealing with agricultural systems as the main source of food for large populations with a concomitant neglect of forest, semi-natural and natural and tropical ecosystems. This earlier neglect has been addressed in the last two decades with the radioecology of these ecosystems being studied in greater detail. The heightened vulnerability of Arctic populations to radioactive contaminants such as isotopes of cesium appears to be caused by a short environmental exposure pathway that efficiently transfers cesium to a human endpoint. Deposited cesium is taken up efficiently by Arctic vegetative and lichen species due to the nature of the Arctic climatic regime and adaptations of Arctic biota and is transferred to reindeer which feed almost exclusively on these plants. Arctic populations consuming large amounts of reindeer and foodstuffs such as fungi and berries which also accumulate cesium can be exposed to higher levels of radioactive contaminants that would be expected for a similar amount of environmental contamination in more temperate regions.

Marine and freshwater exposure pathways

Although there can be differences in the radioecological behaviour of radionuclides in freshwater and marine systems, the exposure pathways are generally the same. Direct exposure of humans and biota can occur via external radiation from contaminated water and sediments. Ingestion of contaminated water or sediment or inhalation of spray can also result in exposures although the main exposure pathway is ingestion via the food chain. Benthic biota can be exposed due to living in and on contaminated sediments. The extent to which radionuclides are transferred along aquatic pathways is dependant on a range of factors including organic matter content, suspended sediment content, salinity, isotope type, residence times of waters, nutrient status, species etc. Radioecological studies of freshwater and marine systems have been plentiful although the vast majority of these have been concerned with radiocesium and ^{90}Sr. An important radioecological control in aquatic systems is the salinity of the system – uptake of isotopes such as ^{137}Cs and ^{90}Sr in aquatic fish species has repeatedly shown to be higher in systems with low levels of K^+ and Ca^{2+}.

Marine systems have been the focus of much radioecological study over the years and this work has gone a long way towards a better understanding of exposure pathways in this environment. For marine systems, radionuclides present in the particulate or colloidal form may tend to be accumulated to a greater extent by marine animal species than those present in dissolved form – this being in contrast to freshwater and terrestrial systems where the converse generally tends to be true (hence the predominance of ^{90}Sr and ^{137}Cs in the exposure pathways of the latter two environments). This is primarily due to the chemical nature of the marine environment which features large amounts of chemical analogues for both strontium and cesium and the fact that many marine species on the lower rungs of the food chain (and hence at the early stages of exposure pathways) are filter feeders, a feeding mechanism ensuring high uptake of particulate bound radionuclides.

Quantifying transfer in exposure pathways

An important focus of radioecology regarding environmental exposure pathways has been the means of quantifying radionuclide transfer along these pathways and a number of means of doing so have been developed. The first of these can be encapsulated by three "factors" used within radioecology to describe transfer:

$$\text{Transfer Factor (TF)} = \frac{\text{Nuclide conc. in plant tissue (Bq/kg)}}{\text{Nuclide conc. in soil (Bq/kg)}}$$

$$\text{Concentration Factor (CF)} = \frac{\text{Nuclide conc. in biota tissue (Bq/kg)}}{\text{Nuclide conc. in filtered water (Bq/l)}}$$

$$\text{Aggregated Transfer Coefficient } (T_{ag}) = \frac{\text{Nuclide conc. in plant tissue (Bq/kg)}}{\text{Nuclide deposition on soil (Bq/m}^2\text{)}}$$

Such factors have proved useful in quantifying transfer along exposure pathways for assessment purposes related to regulation, emergency response, rehabilitatory efforts etc. Large scale efforts in the evaluation and collation of empirical data sets regarding radionuclide transfer as expressed by factors such as those above have resulted in compendia published by bodies such as the IAEA (IAEA, 2004; IAEA, 2010). An important and significant common characteristic of such compendia is the high degree of variability in the data sets (often orders of magnitude) for individual isotopes reflecting the complexity of the processes involved in the transfer of radionuclides to plants and animals (see Figure 2). Addressing and accounting for this variability has proven to be a significant driver in radioecological research over the years.

It should also be considered that the transfer of radionuclides to animals or plants is largely a dynamic process and that quantification of transfer by expressions such as those above only represents a temporal "snapshot" and one that is likely to vary considerably depending on the time it was taken if equilibrium had not been achieved.

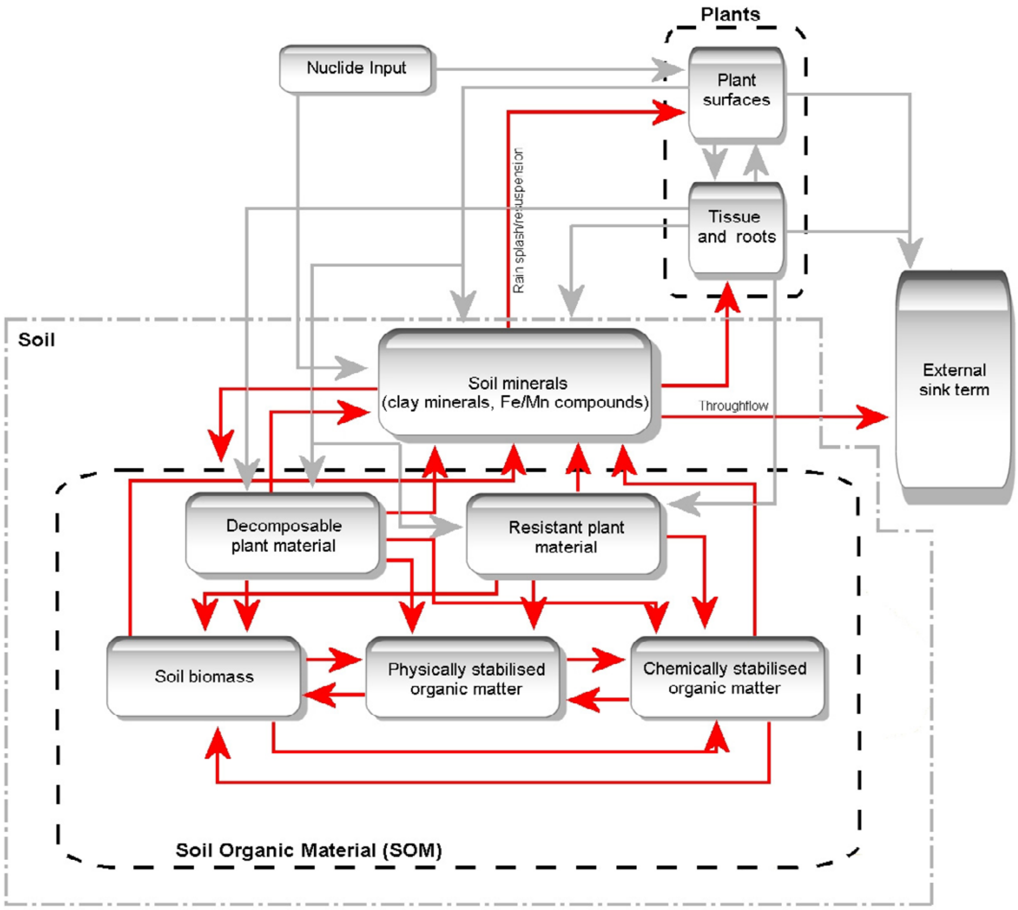

Fig. 2. Schematic of soil processes governing transfer of radionuclides from soil to plant.

For the many situations or radionuclides where data is lacking or the inherent weaknesses of the CF/TF approach are unacceptable (see Green et al., 1997), alternative methods for estimating transfer have been developed. Mechanistic kinetic approaches have been available since the 1980s (see Whicker and Schulz, 1982) and such methods include the advantage that the transfer at any time can be estimated as opposed to the equilibrium assumption inherent in CF/TF approaches. Despite this, mechanistic kinetic approaches have still not fully fulfilled their potential. Allometry, which relates radionuclide uptake to body size, has proved a potentially useful avenue of research in describing transfer (see Higley et al, 2003) as has phylogenetic approaches to extrapolate information on transfer of radionuclides from information derived for one species to another (see Willey and Fawcett, 2006).

Bioavailability

A key concept in the study of environmental exposure pathways is that of bioavailability - bioavailability generally refers to the extent to which humans and ecological receptors are exposed to contaminants in soil or sediment (see Desmet et al, 1991) and with respect to radioecology can be understood as the quantity of a radionuclide available for biological uptake. It is accepted that bioavailability is the factor that ultimately controls the exposure of both humans and biota and therefore the impact of contaminant radionuclides. In studying bioavailability within radioecology, much emphasis has been upon the transfer of radionuclides from soil to plant and an extensive corpus exists in this field although much work remains to be done. Factors such as climate, soil properties and radionuclide speciation have been focussed upon as controllers of terrestrial bioavailability and have been the subject of radioecology research over six decades. As ^{137}Cs came to be recognised as a radionuclide of importance during the period of atmospheric weapons testing and in the aftermath of the Chernobyl accident, significant effort was expended on understanding the bioavailability of this isotope. Cs-137 and study of its behaviour in the environment highlighted the complexity of the problem and although important advances were made in understanding, for example, the role of organic soil materials and potassium in its environmental transfer, much work remains to be done in order to develop mechanistic models describing its uptake by plants and further transfer through ecosystems although some models are currently available. In this respect, bioavailability is a significant area of research in radioecology as models and predictions with respect to environmental exposure pathways are reliant on a thorough understanding of the processes and mechanisms involved.

Current and future directions

Perhaps the most significant development of relevance to radioecology in recent years has been the acceptance that one can not always assume that if humans are protected from the effects of radioactive contaminants, then so is the environment. This acceptance has led to consideration of environmental exposure pathways that are not strictly directed towards the human endpoint but which are of relevance to wildlife species. This expansion in the scope of the pathways and species that must be considered has presented new challenges for radioecology and will continue to be a focus in coming years. In addition, radioactive contaminants occur in the environment as part of a large suite of other contaminants. Radioecology has begun to recognise this and view radioactive contaminants and exposure pathways as part of the wider spectrum of contaminants.

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Malicious events: scenarios, consequences and response

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Abstract

The potential use of radioactive materials by individuals with malicious intent has become an area of increasing international concern in recent years. This refresher course session will cover a number of key areas:

– Consideration of a range of potential scenarios and consequences both in the short and longer term;
– Examination of how lessons identified from past accidents can be used to develop preparedness in this area;
– Discussion of the key elements of planning arrangements required to respond effectively to a scenario of this nature; and
– Reviewing the key international planning guidance that is available to assist in the preparation for such scenarios.

Indoor radon sources, remediation and prevention in new construction

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Abstract
Radon is the second leading cause of lung cancer in the general population, after smoking. Strategies for both radon prevention in new dwellings and mitigation in existing dwellings are needed to achieve an overall risk reduction. The inflow of radon laden soil air into living spaces is the main source of elevated radon concentrations.

The methods of indoor radon remediation are normally based on depressurization of soil under floor construction and decreasing soil air radon concentration. Sealing of entry routes, improvement of air exchange rate and decreasing underpressure level in living spaces has also been used. Sub-slab depressurization (SSD) and radon well have been found the most efficient methods. Typical reduction factors are 70-90%. Ventilation based methods and sealing of entry routes result normally in a clearly lower radon reduction. A SSD system comprises a cavity in the ground immediately under the floor slab. This suction cavity is linked by pipe work to the outside. A radon well sucks air from a well pit constructed outside the building. This air flow reduces the soil air concentration in a wide area round the well. A radon well can be applied on coarse soil types.

Most prevention strategies address steps to limit soil gas infiltration due to air pressure differences between the soil and the indoor occupied space. Both sealing based and sub-slab depressurization based measures have been recommended. The available research results show an average efficiency of 50 % for passive prevention measures. Fan activated sub-slab depressurization reduces radon concentration by up to over 95%. Widespread use of radon resistant construction has resulted in Finland in an average reduction of 30% in radon concentrations of new low rise residential buildings.

1 Introduction
Radon is the second leading cause of lung cancer in the general population, after smoking. Strategies for both radon prevention in new dwellings and mitigation in existing dwellings are needed to achieve an overall risk reduction. In EU countries the typical proportion of low rise residential houses exceeding the national reference value of 400 Bq/m^3 is 0.1% - 5%. Presently the reference value for design and new construction is typically 200 Bq/m^3. In the case of this lower reference value the proportion is typically 0.5% - 15%. New international recommendations have resulted also in a reference value of 100 Bq/m^3 (WHO 2009). The proportion of houses above

100 Bq/m^3 may be 5% - 50%. Respectively the country specific number of houses needing remedial measures can be as high as 400 000 - 2 000 000.

The inflow of radon-bearing soil air into living spaces is the main source of elevated radon concentrations. Radon prevention strategies are therefore based on limitation of soil gas infiltration through reduction of the air pressure difference between soil and indoor spaces and on decreasing the radon concentration in soil air. Prevention goals and strategies are especially important because neglecting the strategy leads to increase of the total number of dwellings with elevated indoor radon.

In this presentation the new WHO radon handbook, several national guides and website material of remedial measures and prevention in new construction has been utilized. Special emphasis has been given also to Finnish experience, especially regarding radon wells and new sample survey results of radon prevention in new construction.

2 Radon sources

Inflow of radon laden soil air is the main entry mechanism increasing radon concentrations in indoor spaces. This inflow is forced by underpressure in rooms above the floor construction compared with soil or spaces beneath the floor. The underpressure is created by indoor-outdoor temperature difference, wind and when in use also by mechanical ventilation. Table 1 shows typical underpressure levels in Finnish low rise residential houses when outdoor temperature is 0 deg C. Mechanical exhaust ventilation creates a high underpressure, depending on the air tightness of the building shell. In the case of a mechanical supply and exhaust ventilation the underpressure can be controlled through adjustment of air flows.

In living spaces building materials are normally only in special cases the reason to elevated indoor radon concentration above 200 Bq/m^3. If the air exchange rate is well below the recommended values (ACH 0,5 1/h), even normal radon emissions can cause elevated indoor radon concentrations.

Figure 1 presents typical entry routes in UK. Figure 2 presents a typical Finnish slab-on-grade foundation. Permeable light-weight concrete blocks promote flow of soil air into living spaces.

Table 1. Typical underpressure levels in Finnish low rise residential houses when outdoor temperature is 0 deg C (STUK 2008).

Ventilation strategy	Typical heating season underpressure, Pa
Natural	1 - 2
Mechanical exhaust	5 - 10
Mechanical supply and exhaust (heat recovery included)	2 - 5

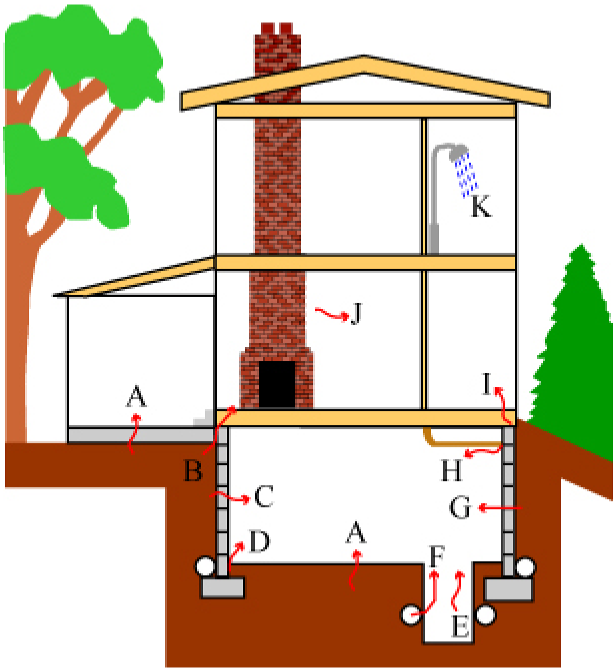


Fig. 1. Major radon entry routes of radon

A. Cracks in concrete slabs.
B. Spaces behind brick veneer walls that rest on uncapped hollow-block foundations.
C. Pores and cracks in concrete blocks.
D. Floor-wall joints.
E. Exposed soil, as in a sump or crawl space.
F. Weeping (drain) tile, if drained to an open sump.
G. Mortar joints.
H. Loose fitting pipe penetrations.
I. Open tops of block walls.
J. Building materials, such as brick, concrete, rock.
K. Well water (not commonly a major source).

Source: http://www.rpwradonwales.co.uk/

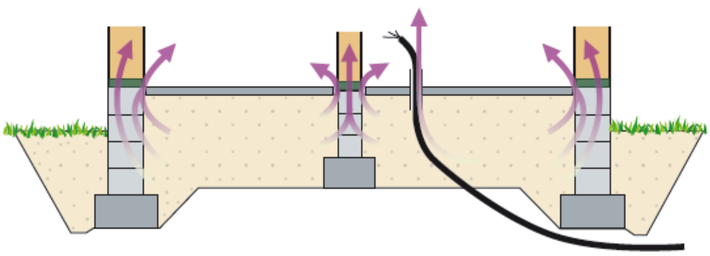


Fig. 2. Entry routes of soil air in a Finnish slab-on-grade foundation. The foundation wall is of permeable light-weight concrete blocks (STUK 2008).

3. National radon prevention and remediation programmes

In the recent WHO Radon Handbook (WHO 2009) the following key elements for successful radon prevention and mitigation actions within the framework of a national radon programme have been given.

1. Radon control actions should consider a combination of building types:
   - new and existing homes, since the greatest amount of radon exposure is generally in homes;
   - buildings where public is likely to be exposed for long periods, such as schools, preschool facilities, state-owned or leased buildings, and lodging facilities.

2. Research on buildings should be used to identify the most cost-effective radon control strategies for prevention and mitigation. Structural, foundation, and ventilation systems as well as construction practices vary from region to region. Specifically, this research should be used to develop:
   - radon prevention standards and regulations such as building codes for new dwelling construction;
   - radon mitigation standards and requirements for remediation of existing dwellings.

3. The contribution of different radon sources varies between countries and even regions. The following mechanisms may be considered:
   - pressure driven soil gas infiltration;
   - emanation of radon from building materials;
   - water transport of radon.

4. Appropriate training and certification of building professionals should be implemented to ensure the efficiency of prevention and mitigation actions.

WHO Radon Handbook gives the following design criteria for the radon control systems for prevention as well as mitigation (WHO 2009):

- able to reduce radon concentrations considerable below the reference level
- safe and not creating back-drafting
- durable and functional for the expected life of the building;
- easy monitoring of the performance
- quiet and unobtrusive
- low cost for installation, operation and maintenance
- easy to install an additional fan when passive soil depression systems (PSD) are used.

4 Remediation in existing dwellings

4.1 Overview on the efficiency of remediation methods

The methods of indoor radon remediation are normally based on the following principles.

1. Depressurization of soil under floor construction and decreasing of soil air radon concentration
2. Sealing of entry routes
3. Improvement of air exchange rate and/or decreasing underpressure level in living spaces
4. Combination of these methods

Table 2 shows the efficiency of different methods in UK and the influence of house characteristics (Naismith et al. 1998). Reduction factors of 1,5 -3 are typical for the methods based on sealing and ventilation corresponding to a percentage reduction of 30-70 %. Figure 3 shows radon reduction factors achieved using various mitigation methods in Finland (STUK 2008). The Finnish results are based on a questionnaire study in 400 houses. Well designed and implemented mitigations result often in reductions which are better than the typical reduction (middle 50 % of results) in figure 3.

Table 2. The influence of house characteristics on radon reduction factors. Reduction factors has been defined as the ratio of radon concentration before and after the remedial measure. A reduction of 50 % corresponds in the table to the reduction factor of 2.0 (Naismith et al. 1998).

Remedial measure	Grouping of data	Number of homes	GM of initial radon level	GM reduction factor	Percentage of homes reduced below the action level
Sump	Installed by major contractor, post-war house	166	540	17	95
	Installed by major contractor, pre-war house	102	630	9	77
	Installed by local builder, post-war house	65	430	11	82
	Installed by local builder, pre-war house	43	530	6	70
	Installed by householder, post-war house	31	550	7	77
	Installed by householder, pre-war house	22	560	4	64
Positive ventilation	Living room and bedroom on different floors	43	500	1.9	42
	Living room and bedroom both on ground floor	51	580	3.7	63
Additional permanent ventilation of the house	Without additional sealing of cracks or service entry points	59	370	1.5	49
	With additional sealing of cracks or service entry points	56	430	2.4	60
Mechanical ventilation of the underfloor space	All data	63	540	2.8	54
Natural ventilation of the underfloor space	All data	171	390	1.9	53
Sealing of cracks or service entry points	All data	71	470	1.7	41

Sub-slab depressurization (SSD) and has been found the most efficient method. Typical reduction factors are 70-90%. Ventilation based methods and sealing of entry routes result normally in clearly lower radon reduction. In the Irish guide to likely effectiveness of remediation techniques (RPI, 2009, Fig. 4) the fan-assisted sump is the only technique with a high likelihood of success above 800 Bq/m^3. The target concentration in Ireland is 200 Bq/m^3 or below. The radon concentration range for good success for other methods is typically 200 - 700 Bq/m^3, for sealing and improving ventilation 200 - 400 Bq/m^3. This guidance is agreement with the Finnish results in figure 3. Radon well technique (chapter 4.3) has been used in Finland and Sweden. The results are on average as good as those for SSD, figure 3.

4.2 Sub-slab depressurization (SSD)

Sub-slab-depressurization is the most common and also most efficient methods in radon remediation. Following names have been used for this system:

- Sub-slab depressurization (SSD)
- Sub-floor depressurization
- Active or passive sub-slab depressurization (ASD, PSD)
- Radon sump (UK, Ireland)
- Sub-slab suction system
- Fan-assisted sump

SSD is based on two processes, first on depressurization of soil beneath the leaking floor structures and second on decreasing the radon concentration of the soil air flowing through the floor. This depressure reverses the pressure differential between soil under floor and the room above. A radon sump comprises a cavity (suction pit) about the size of a bucket (10 - 30 litres) in the ground immediately under the floor slab that is open to the surrounding under-floor hardcore. The sump is linked by pipe work to the outside. A suction pit is needed in order to minimize the flow losses in soil round the sump. The pipe penetrations and possible gaps and cracks in floor construction close to the suction pit should be sealed. Leakage through such gaps can decrease the efficiency remarkably.

Using a sump with an electrical fan is called as an active system. A reduced efficiency can be achieved through a passive sub-slab depressurization. In this case the system works without a fan. Typical radon reduction for an active SSD is 70-90%. In difficult cases additional sealing work is needed in order to achieve a low enough radon concentration.

The best location of the suction pit is in the central area of the floor slab or in a part of floor area which is divided by load bearing foundation walls into separate blocks. In the case this kind of foundation walls prevent the extension to other parts of foundation, two or more suction pits may be needed. In the case of one undivided floor block, one suction pit is normally sufficient for a floor area of 250 m^2 (BRE 2003) or 120 m^2 (STUK 2008), depending on the soil and foundation constructions.

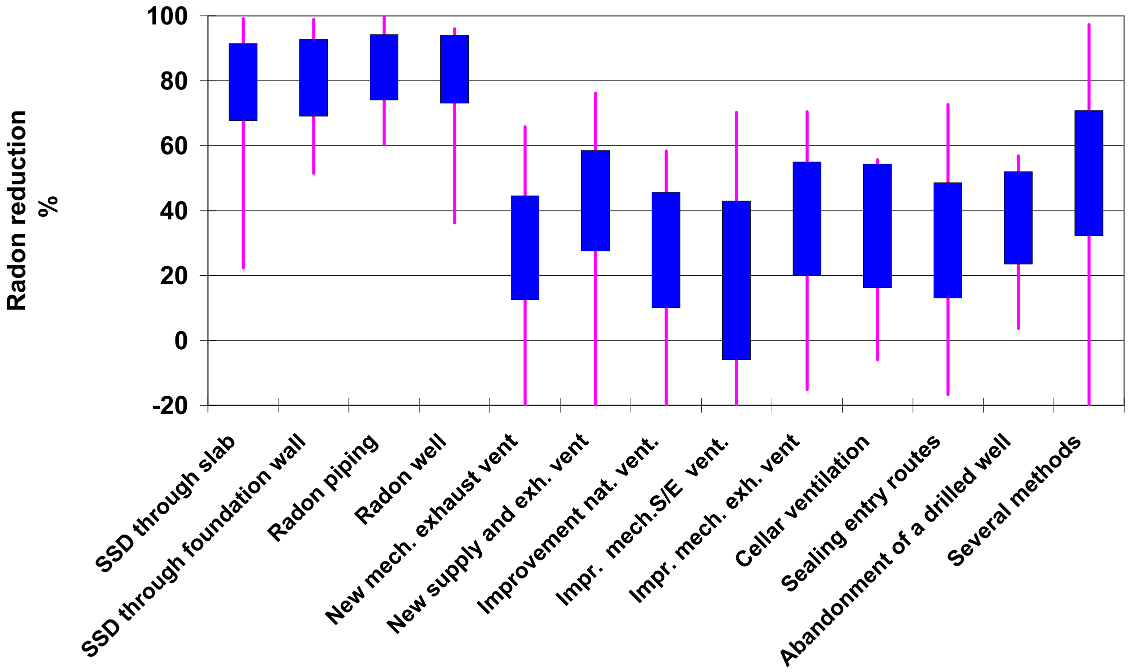


Fig. 3. Radon reduction factors achieved using various mitigation methods, minimum, 25% percentile, 75% percentile and maximum. The results are based on a questionnaire study in 400 Finnish houses. SSD: Sub-Slab Depressurization (STUK 2008).

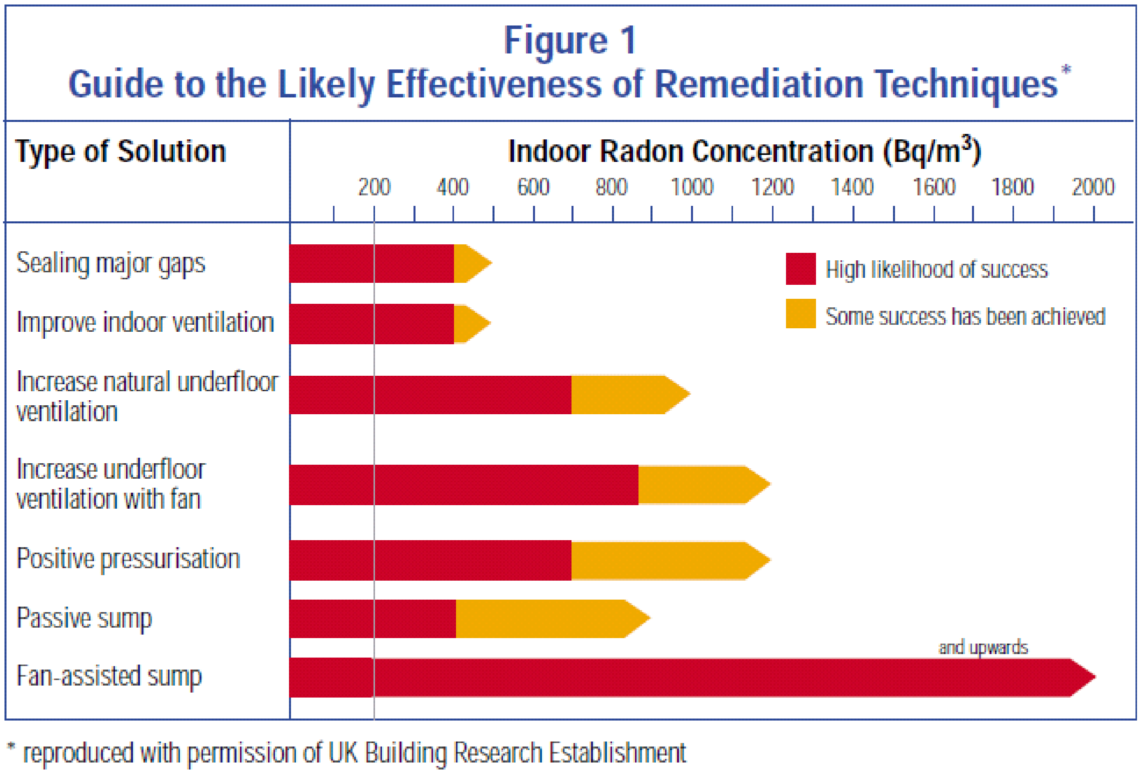


Fig. 4. Guide to likely effectiveness of solutions in Ireland (RPI 2009).

In the basic UK-guide (BRE 2003) **the standard sump** should be located close to the centre of the dwelling. It can be constructed using a paving slab 600*600 mm and a few rows of bricks with air gaps left between them or by using a prefabricated sump (Figure 5A). Typical fan power is approx. 70 watts.

A simple **mini sump** (Figure 5B) can be constructed by breaking out or core drilling a 120 mm diameter hole in the floor slab and excavating about a bucketful (10 -30 litres) of material from below (clearing out a space approximately 200mm in radius). A good volume in the suction pit decreases air flow resistance and increases thus the extension of the air flow.

A SSD can be constructed also from outside the building, According to the BRE guide, an **externally excavated mini sump**, figure 5C can be constructed by breaking out or core drilling a 120mm diameter hole through the external wall just below the floor slab and excavating about a bucketful of material (clearing out a space approximately 200 mm in radius).

Passive sump systems

Experience has shown that in cases where the indoor radon level is relatively low, i.e. just above the action level, it is possible, in the right circumstances, to operate a sump system *passively* – without the need for an electric fan (BRE 2003). If it works, a passive sump system is always preferable to a fan driven one as it is easier and cheaper to install, run and maintain, and will be quieter. However they are not as effective as systems fitted with fans. On average passive sump systems produce a 50% reduction in radon levels, whereas fan powered systems achieve relatively larger reductions – often more than 90%. The important point is that it is a simple task to add an electric fan later if the passive system does not adequately reduce the radon level.

Multiple suction points

Example in figure 6 shows a Czech application (Jiranek 2003) of multiple suction point system in a typical Czech cellar house. The effectiveness of such systems varies between 70 and 98 %, which means that indoor radon concentration decreases to 30 % up to 2 % of the initial values. The effectiveness is mainly influenced by the vertical profile of soil permeability and by the air tightness of the building substructure.

Depressurized hollow floor

Figure 7 shows an application of depressurized hollow floor (Swiss Radon Handbook). The system is appropriate when radon bearing soil air is coming mainly through the floor. In this example a reduction from 1400 Bq/m^3 to 50 Bq/m^3 was achieved.

Installation of radon piping and a SSD system

In cases where floors in old houses have been opened and renewed, it is practical to install a radon piping below the new floor. Figure 8 shows a French example of piping installation (CSTB 2008). Piping has been used also for controlling the leakages through the lower parts of the walls. Piping has been installed into a permeable aggregate layer and has been covered with geotextile and plastic membrane. Radon concentrations before the remedy were 1300 - 2800 Bq/m^3. Radon reductions up to 90% were achieved.

Depressurization of drainage system

In houses with existing drainage system depressurization of the drainage piping can be utilized. According to the Swiss Radon Handbook this system is effective only in 10 % of cases. US EPA (USEPA 2003) gives an average reduction factor of 50 %. The efficiency can be enhanced through installation of sealed valves or drain traps at exits of the drainage system. A sufficient quantity of water opens the valves. In Finland the method has been tested with a success in several houses. Creating a good depressure in the piping is the key problems. Further studies are needed for a good guide material for this method.

4.3 Radon well

Principle of radon well

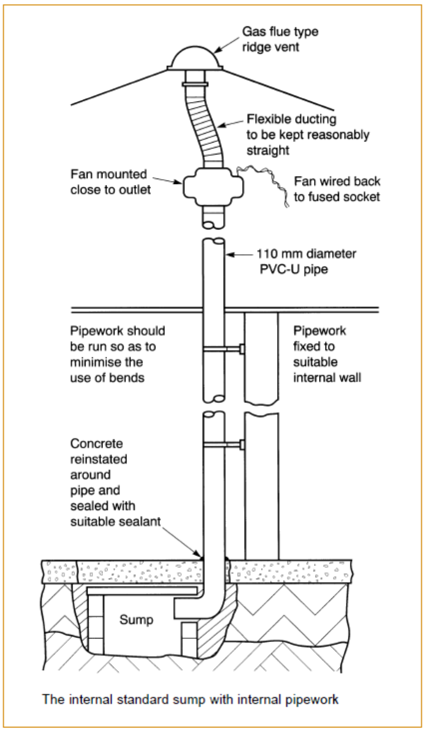
A radon well is constructed outside the house and the well sucks air from the soil from the depth of 3 - 5 meters (STUK 2008). Figure 9 shows the principle of a radon well. Soil ventilation caused by the exhaust fan decreases efficiently the radon concentration of the soil air below the house foundations. Close to the house a radon well can also create underpressure below floor slab. A single radon well can reduce the radon concentration in several dwellings at the distance of up to 20-30 meters.

Building a radon well requires experience and expertise in soil geology. The efficiency of a radon well is affected e.g. by soil permeability, layer structure, depth of the suction pit and the power of the suction fan. Radon well is effective only when built on coarse soils as gravel and coarse sand. If the soil is fine and impermeable, such as clay, silt and moraine, it is not possible to create an air flow field with a good extension. Also fine sands may be problematic. Soil layers with coarse and fine material affect the extension and form of the air flow field.

Figure 10 presents the construction of a radon well. The best depth for the suction pit is normally 4-5 meters, below the footings of the house. The depth of the well promotes the extension of the flow field to a wide area.

The exhaust pipe can be either on the cover of the well ring or at good distance from the well using a horizontal transfer pipe below soil surface. In the case of houses on hillside (split level house), the best location of a radon well is on the upper side of the house.

A



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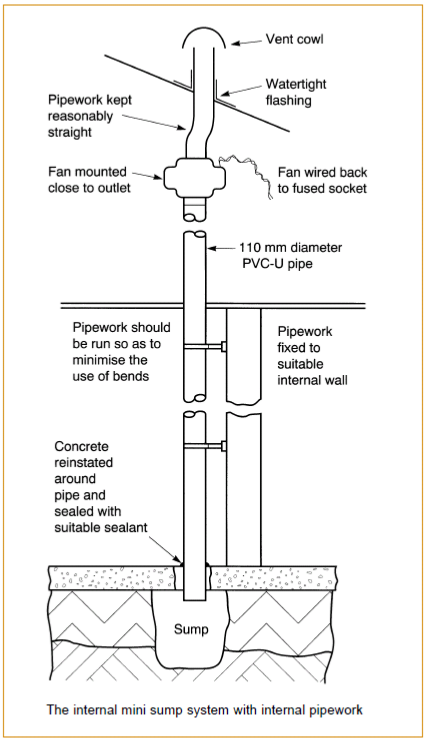

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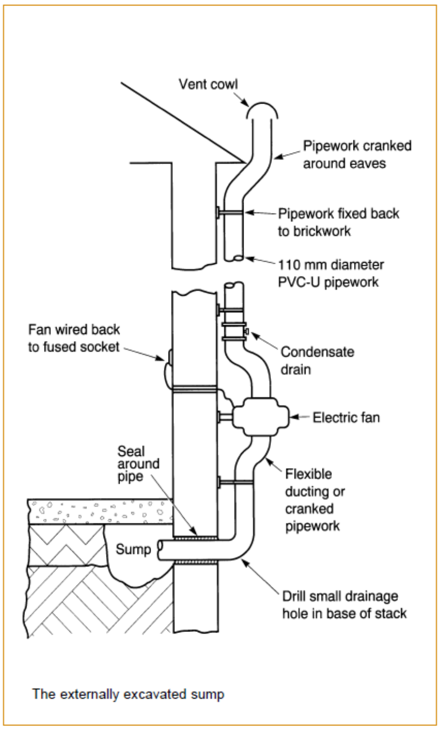

Fig. 5. Sump systems in the UK BRE guide (BRE 2003).
A: Internal standard sump system
B: Internal mini sump system
C: Externally excavated sump system

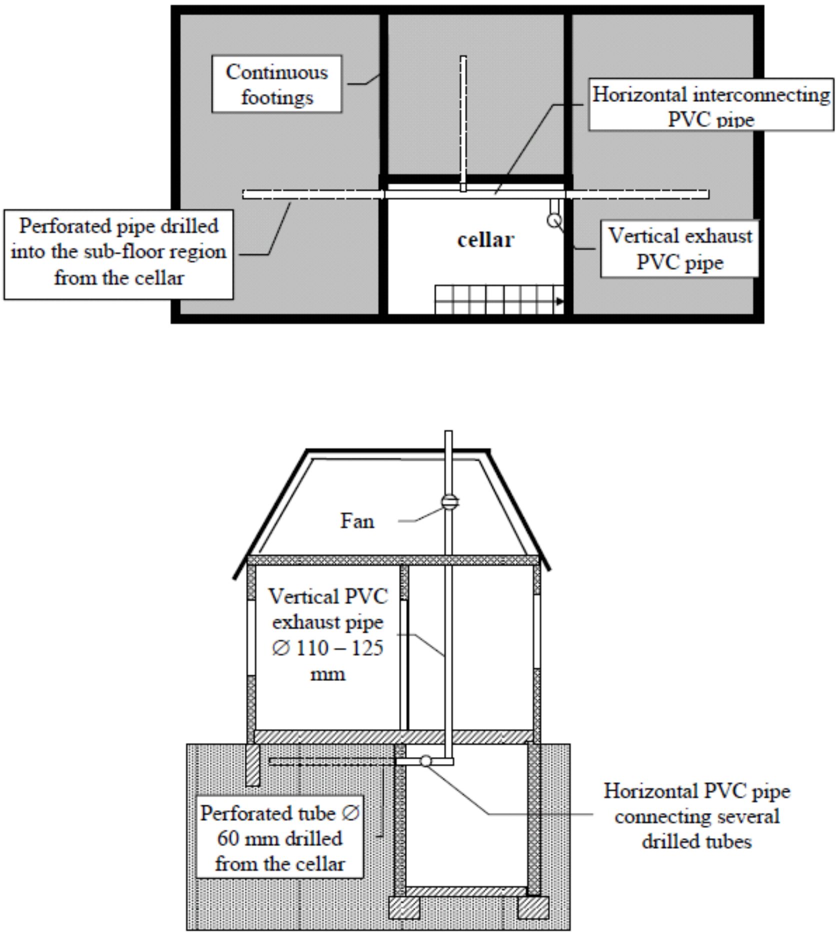

Fig. 6. Sub-slab depressurization based on perforated tubes drilled from cellar according the Czech mitigation standard (Jiranek 2003).

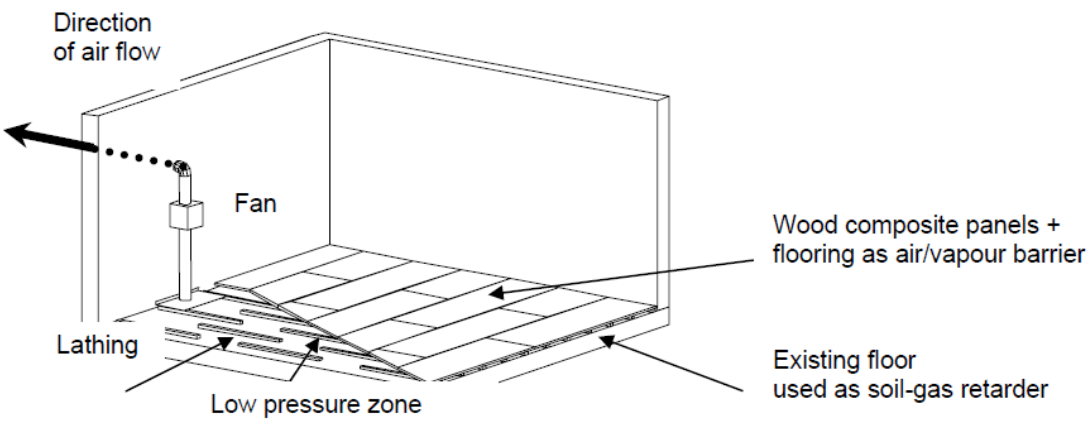

Fig. 7. Depressurized hollow floor, Swiss Radon Handbook.

Fig. 8. Sub-floor piping installation for a sub-slab depressurization system in France (CSTB 2008).

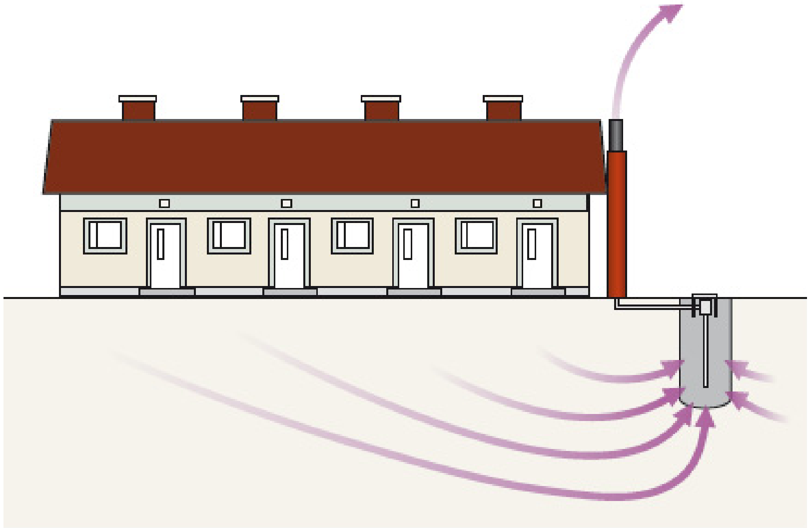

Fig. 9. Principle of a radon well. A radon well ventilates soil air and decreases the soil air radon concentration in a large area (STUK 2008).

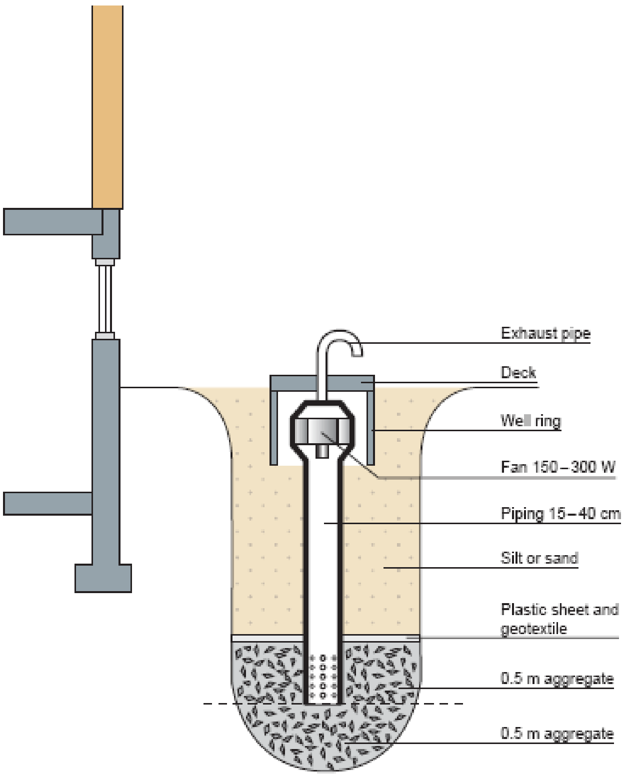

Fig. 10. Construction of a radon well (STUK 2008).

Radon well in Hollola

Figures 11 and 12 present a radon well installation of a row house company with four buildings and 20 dwellings (STUK 2008). The aim was to achieve good results using two radon wells. Fig 11A presents the exhaust fan of 300 W installed in a well ring. A transfer pipe is leaving the well ring towards the gable of the house. Fig. 11B presents the cover of the radon well and the discharge pipe installed above the eave height of the building. The thermally insulated pipe ensures that the pipe does not freeze in the winter. This high exhaust pipe releases the radon bearing exhaust air away from the yard area (including a playground). The depths of the wells are 3.5 - 4 m. The air flows measures were 68 l/s (250 m^3/h, well no 1) and 58 l/s (210 Bq/m^3, well no 2). The average reduction of indoor radon concentration, in 17 dwellings originally above 400 Bq/m^3, was 88% (Fig. 12). The best reduction rate was 99%.

Fig 11. Exhaust fan in a well ring (A) and cover of the well ring as well as the exhaust pipe installed above the eave height of the building (STUK 2008).

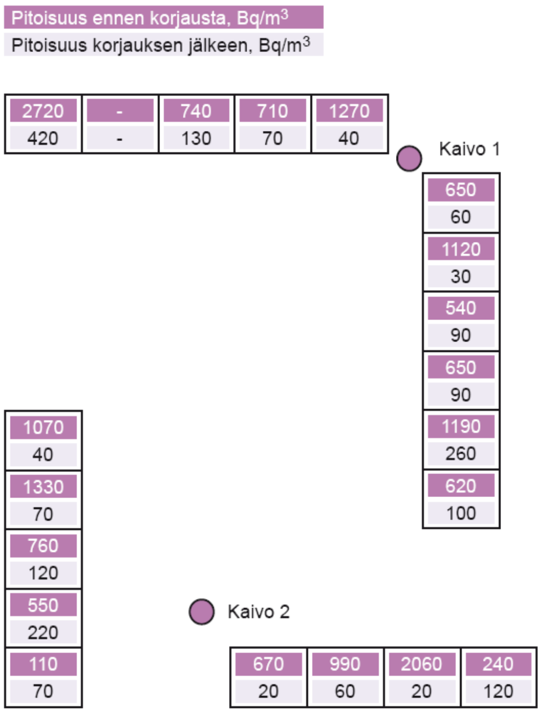

Fig. 12. Reduction of indoor radon concentration in 20 dwellings of a row house company after starting of two radon wells. Radon concentrations before and after the mitigation (STUK 2008).

4.4 Improving ventilation and sealing

Improving ventilation of occupied spaces

The British guide recommends the measures presented in figure 13. The guide emphasizes that it is important to recognise that in most cases the reduction in radon level that can be achieved by changing the way in which you ventilate your home will be small. Generally ventilation approaches are more common in mechanically ventilated schools and other large buildings than in small houses. The systems may be especially useful in the following cases.

- Building materials are the major entry route.
- There are other indoor air quality problems, for example very low air exchange rate.
- In the case of mechanical exhaust ventilation and tight building shell, reduction of underpressure level may be effective.

Radon reduction measures based on ventilation reduce radon concentration either through increased ventilation or decreased underpressure level. Reduction factors above 50% have been achieved only in cases where the original air exchange rate has been defective or when the underpressure level has been high. The typical reduction factors in Finland have been 10-40% (Fig. 3). Increasing the operation time or the power of mechanical ventilation, opening the existing or installing new fresh air vents are typical mitigation measures. Installation of new fresh air vents does not normally result in reduction factors above 50%.

In the Finnish experience the best results have been achieved through installation of a new supply and exhaust ventilation system. The underpressure created can be minimized and controlled in this strategy, which is important in radon control.

Temperature and moisture aspect should always be taken into account when changing the ventilation or depressure conditions.

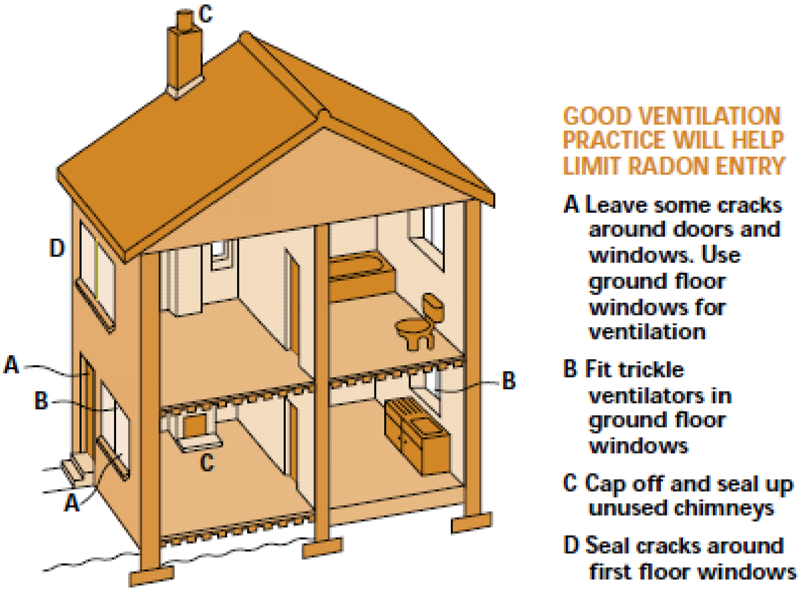

Fig. 13. Good ventilation practices in the UK guidance (BRE 2003).

Sealing entry routes

Sealing entry routes aims at reduction of leakage flows of radon bearing soil air into the living spaces. Sealing may be very demanding. In many cases the results of sealing are sufficient only when the entry routes have been sealed almost completely.

Floor joints with foundation walls of porous light-weight concrete cannot be sealed with normal sealing methods. Porous elements provide new leakage routes although the wall-floor gap has been sealed. Hollow block wall elements need similarly special consideration. Best results in Finland, when sealing the joint of floor and wall have been achieved in houses where the foundation wall is of casting concrete.

5 Radon prevention strategies

Most prevention strategies address steps to limit soil gas infiltration due to air pressure differences between the soil and the indoor occupied space. Radon prevention strategies should consider the specific mix of construction practices, radon sources, and transport mechanisms in the region or country, in order to be cost-effective. Under certain conditions a combination of strategies may be necessary such as in buildings with multiple types of foundations.

Preconstruction site assessment

The WHO Radon handbook (WHO 2009) presents following aspects on preconstruction site assessments. A number of approaches are used to assess the potential for elevated indoor radon concentrations across geographic areas. The most common approach involves mapping of indoor radon concentration in existing houses. This approach can be complemented through utilization of geological information. Areas of permeable sand and gravel e.g. can be classified to the high radon category. Another approach used in some countries, such as the Czech Republic, involves testing individual building sites prior to construction using soil gas measurements to establish a radon index for the site. The index is then used to define the degree of radon protection needed for building on that site. However, in countries including Finland, Ireland, Norway, Sweden, Switzerland, the United Kingdom, and the United States of America, the most cost-effective approach appears to be the use of radon control options in all new homes or in homes in specified areas.

Requirements in Ireland and UK

As an example the requirements in UK and Ireland are considered (BRE 2007, Ireland 2007). The basis for the requirements is the following: It is not possible to accurately predict the concentration of indoor radon likely to occur in a proposed building on the basis of a pre-construction site investigation. Both UK and Ireland have published the results of a national survey of radon levels in existing houses. Based on the results of the survey the countries have identified **High Radon Areas**. In Irish high radon areas, it is predicted that more than 10% of dwellings in the area will have radon gas concentrations above 200Bq/m^3. However, houses with high concentrations of radon gas are not confined to these areas and can occur in individual dwellings in any part of the country. In UK the Health Protection Agency defines Radon Affected Areas as those with 1% chance or more of a house having a radon concentration at or above the Action Level of 200 Bq/m^3.

In Irish high radon areas **full protection** is required. In this case a fully sealed membrane and a radon sump should be provided. In other than high radon areas buildings should be provided with potential means of extracting radon for example a radon sump or sumps with connecting network.

In the radon affected areas of UK sufficient protection will be provided by a well installed damp-proof membrane. This is called the **basic radon protection**. In areas of higher potential **full protection** is required. This means supplementary provisions for radon sump or sub-floor ventilation.

Prevention strategies

Radon prevention practices include both passive and active methods. The passive systems include providing barriers to soil gas entry. Membranes and sealing of openings can be used. Active systems are based on sub-floor depressurization and/or ventilation. Passive systems are to be preferred. However the systems should include provisions for active systems.

Sub-slab depressurization provisions in new construction may be similar with those used in remediation. However in new construction well designed and efficient piping arrangements can be used.

A basic UK approach for a radon-proof membrane and sub-floor depressurization is presented in Figure 14. Figure 15 shows an active soil depressurization approach in USA (USEPA 1993).

In the Finnish guidance (Building Information Ltd 2003) membranes covering the whole floor area are not recommended. Instead the joint of foundation wall and floor slab should be sealed using a strip of reinforced bitumen felt, figure 16. A preparatory radon piping should always be installed, figure17.

The Austrian guide for citizens gives the following general instructions (Lebensministerium, Austria 2005).

- Do not build earth floors.
- Avoid cracks in floor slab.
- Seal the joint of foundation, basement wall in contact with soil and floor slab.
- Seal pipe penetrations.
- Seal pipes if these can serve as routes for soil air.

Radon prevention in new construction, Finnish survey 2009

The building code for radon prevention and the associated practical guidelines were revised in Finland in 2003 to 2004. Thereafter, preventive measures have become more common and prevention practices more effective. Consequently, indoor radon concentrations in new construction have been markedly reduced. In the 2009 study, the indoor radon concentration was measured in 1 500 new low-rise residential houses (Arvela et al. 2010). The houses were randomly selected and represented 7% of houses that received building permission in 2006.

The average radon concentration of all houses measured, which were completed in 2006 to 2008, was 95 Bq/m^3, the median being 58 Bq/m^3. The average was 30% lower than in houses completed in 2000 to 2005. The decrease was 50% in provinces with the highest indoor radon concentration and 20% elsewhere in the country. In houses with a slab-on-ground foundation that had both passive radon piping (discharge pipe uncapped) and sealing measures carried out using a strip of bitumen felt in the joint between the foundation wall and floor slab (Fig. 16), the radon concentration was on average reduced by 55% compared to houses with no preventive measures.

Preventive measures were taken in 50% of single family houses, and in provinces with the highest radon concentration in 90% of houses. Active prevention in areas with high indoor radon concentrations has reduced the regional differences in the radon concentration. Slab on ground is the prevalent type of foundation and necessitates careful radon prevention measures throughout the country. The most serious defects were observed in prevention practices in houses with walls made of lightweight concrete blocks that were in contact with soil. The foundation types with the lowest radon concentrations were those with a crawl space and a reinforced uniform floor slab.

Effectiveness of protective measures

Research results of the effectiveness of protective measures are much more uncommon than results of remedial measures. In order to estimate the effectiveness of protective measures one needs also relevant reference values for houses without preventive measures. For example previous local values can be utilized. In the case of active methods as activation of sub-slab-depressurization, radon concentration measured for the passive system can be used as reference values.

In UK the use of membranes resulted in a reduction of approx. 50 % for both block and beam (under floor ventilation) and in-situ concrete floor (Figure 14) types (Woolliscroft 1994). The results are based on field studies involving over 400 dwellings.

Studies in 44 US homes showed a reduction of 50 % for passive sub-slab depressurization (Dewey and Novak 1994). Two-week tests were conducted in each home with the discharge pipe capped and uncapped.

Results from Czech republic show a reduction of 40% - 80% for membrane and sub-slab depressurization (Jiranek 2003).

Finnish new construction study 2009 in 160 houses (above) gives a reduction of 55% for houses with passive radon piping and sealing using bitumen felt (Arvela et al. 2010).

Activation of radon piping in SSD houses in Finland has resulted in typical reductions of 70 - 90% (STUK 2008).

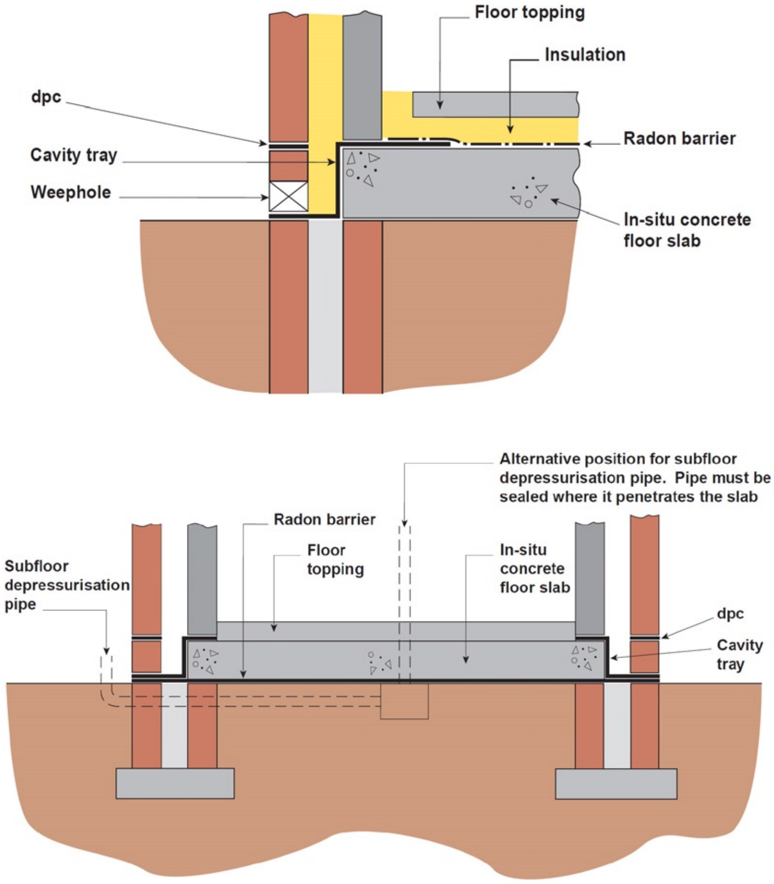

Fig. 14. BRE guidance for radon barrier and sub-floor depressurization in the case of a in-situ concrete ground floor (BRE 2007).

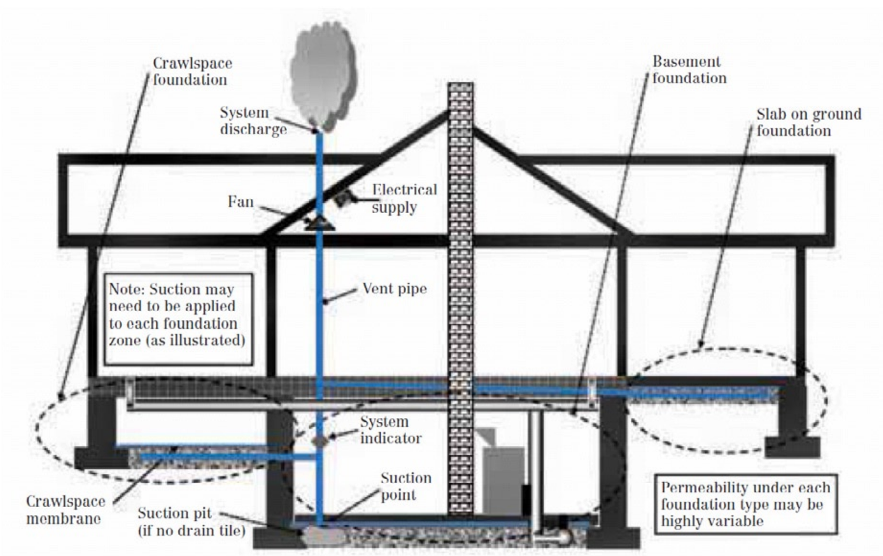

Fig. 15. Active soil depressurization in USA (USEPA 1993).

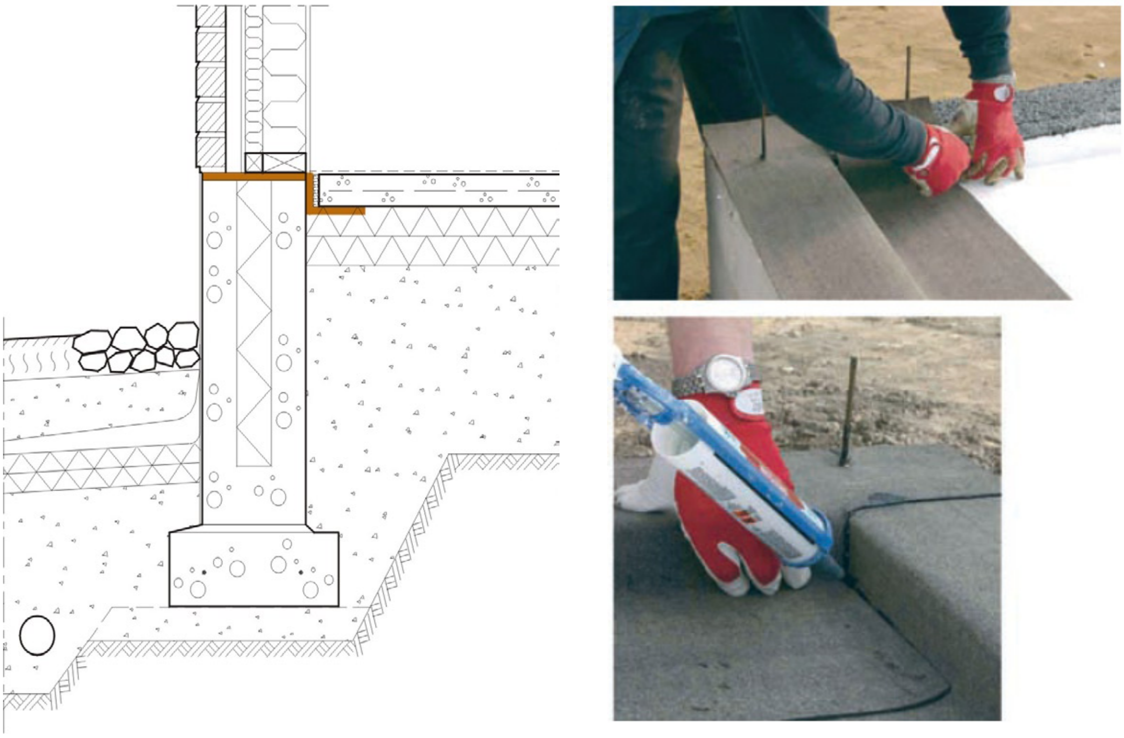

Fig. 16. Sealing of the gap between foundation wall and floor slab in the Finnish guide (Building Information Ltd 2003, Katepal Inc).

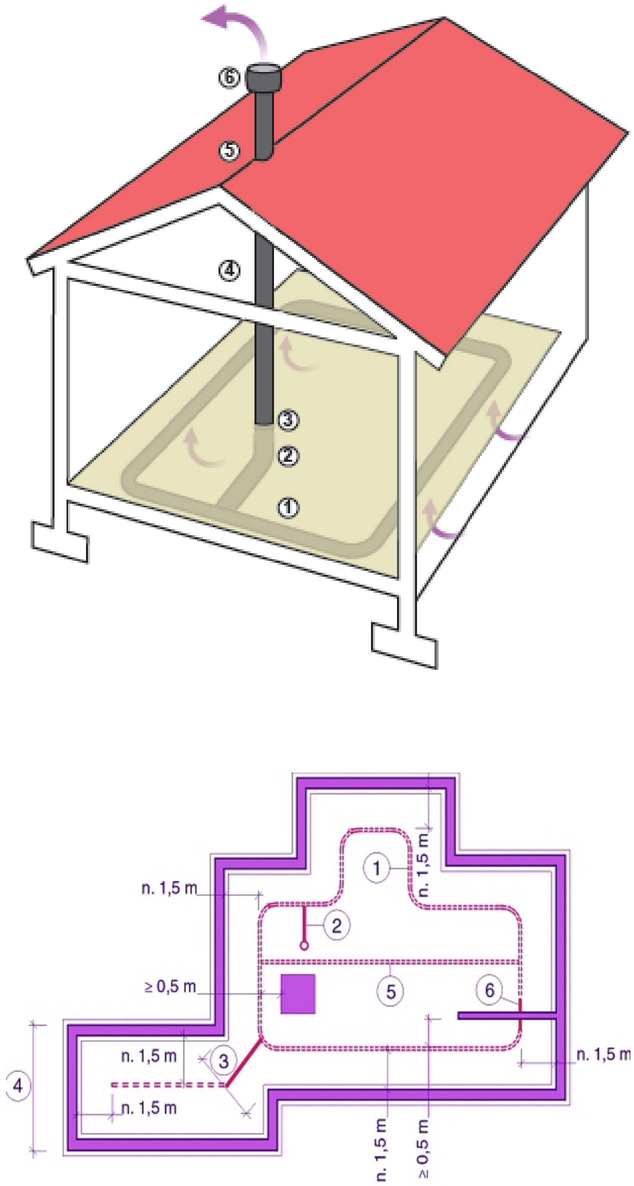

Fig. 17. Radon piping in the Finnish guide (STUK 2008, Building Information Ltd 2003).

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Radiation exposure of space and aircrew

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Abstract

Cosmic radiation and its secondaries created in interactions with planetary atmospheres, shielding structures and the human body constitute one of the most important hazards associated with space and air travel. Crew members are facing exposures to radiation levels that may easily exceed those routinely received by terrestrial radiation workers. To assess the significance of potential biological implications on the health of space and aircrew, it is necessary to discuss the characteristics of the cosmic-ray environment and its dependencies on altitude and geomagnetic latitude. Exposure of space and aircrew to cosmic radiation will be reviewed, and recommended dose limits for astronauts working in low-Earth orbit will be dealt with in comparison with radiation protection guidelines of aircrew personnel.

Introduction

Accomplishments in engineering over the past century have provided unprecedented opportunities for people to become mobile and travel rapidly on or near the surface of the Earth (White and Averner, 2001). Now that new technologies are at our hands to enable us to travel away from our home planet, we are about to become citizens of the universe. For this to happen requires the development of both a new understanding of the risks imposed by the potentially dangerous levels of cosmic radiation, extended weightlessness and psychological stressors, and a more effective means of coping with these hazards to the human organism. Ions of high charge and energy encountered in cosmic radiation have been shown to produce distinct biological damage compared with radiation on ground, leading to large uncertainties in the projection of cancer and other health risks, and obscuring evaluation of the effectiveness of possible countermeasures (Cucinotta and Durante, 2006). On a microscopic scale, it becomes apparent that these particles are likely to deposit their energy in a rather heterogeneous way. Although absorbed doses—averaged over a sufficiently large macroscopic mass element—might be small, there will be microscopic regions of extremely high local doses in close vicinity to the ion path.

Cosmic radiation environment

The radiation environment in space is characterized by a high degree of complexity and dynamics. It is mainly fed by solar and galactic sources (Fig. 1), with additional particles created in interactions with planetary atmospheres, shielding structures and the human body. Among these secondary radiations, neutrons are of foremost importance.

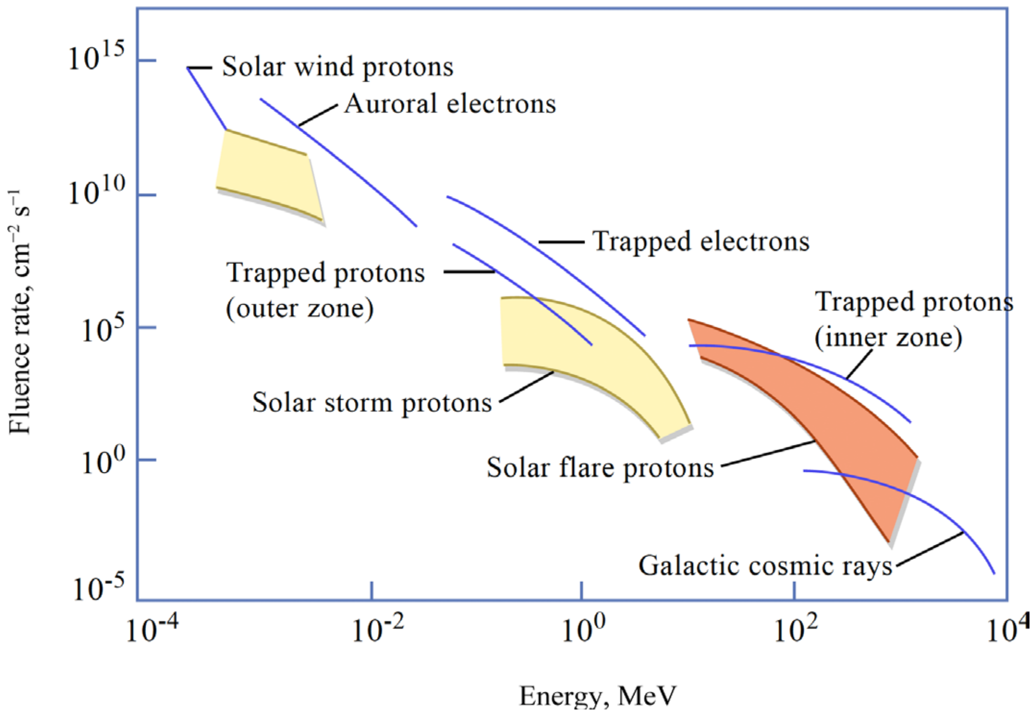

Fig. 1. Energy spectra of cosmic-ray contributors: energetic particles encountered in free space and low-Earth orbit cover a very broad range of energy and fluence rate (MIT OpenCourseWare).

Galactic cosmic radiation (GCR) originates from outside our solar system and is isotropic in distribution, *i.e.*, it arrives at any point in deep space with equal intensity from all directions. The GCR spectrum consists of all naturally occurring chemical elements with energies beyond 10^{20} eV (Lodders, 2003; Mewalt, 1988). Stellar flares, supernova explosions, pulsar spin-offs, or explosions of nascent galactic nuclei were believed to be the sources of GCR acceleration. However, there seems to be no credible mechanism, either inside or outside the galaxy, for accelerating particles to energies above 10^{20} eV (Schwarzschild, 1997). Astrophysicists developed plausible models for how ultra-high-energy cosmic rays might be produced, perhaps even involving new particle-physics phenomena or topological space-time defects left over from the Big Bang, but they still have no definite answers (Cronin *et al.*, 1997). The fluence rate of primary cosmic rays in the galaxy is ~1 cm^{-2} s^{-1}. Low-energy GCR particles consist of 92% protons and 6% helium nuclei, with the remainder being heavier ions with charges of $Z \leq 92$ (^{238}U). The incident fluence rate of cosmic rays with energies above 1 GeV is of the order of 10^{-2} cm^{-2} s^{-1} at the edge of the exosphere.

Solar cosmic radiation (SCR) comprises the flood of low-energy electrons and protons called the solar wind, which increases by factors of the order of 10^{6} during an active sun period to build into a torrential storm. This plasma, streaming out from the Sun's corona at velocities as high as 120 km s^{-1}, creates the interplanetary magnetic field (IMF), which varies according to the 11-year cycle of solar activity. GCR particles entering the heliosphere are scattered by IMF irregularities and undergo convection and adiabatic deceleration in the expanding solar wind (Heber *et al.*, 2009). The GCR intensity is thus anti-correlated with solar activity, which is usually determined from the number of sunspots (Lantos, 1993). Sporadically occurring solar particle events (SPEs) originate from impulsive solar flares, coronal mass ejections or shocks in the interplanetary medium. The emitted protons have energies up to several hundred MeV and, during strong flares, their flux at the Earth's orbit can increase for some hundred percent during hours or days. It is these events, which are of particular concern for the possible manifestation of acute radiation syndrome effects such as nausea, emesis, haemorrhaging or, possibly, even death, since SPEs are still impossible to forecast and might be accompanied by significant dose enhancement (Townsend, 2005).

Energetic particles trapped in the geomagnetic field are confined via magnetic mirroring in two radiation belts, which surround the Earth. The inner belt, which extends from ~1–3 Earth radii in the equatorial plane, was discovered by J. A. Van Allen and co-workers[1] using data taken from Geiger-Müller counters flown on early U.S. satellites. It is mostly populated by protons with energies exceeding 10 MeV. The origin of these protons is thought to be the decay of albedo neutrons from the Earth's atmosphere. The inner belt is fairly quiescent. Particles eventually escape due to collisions with neutral atoms in the upper atmosphere above the Earth's poles. However, such collisions are sufficiently uncommon that the lifetime of particles in the belt range from a few hours to 10 years. Clearly, with such long trapping times only a small input rate of energetic particles is required to produce a region of intense radiation. The outer belt, which extends from ~3–9 Earth radii in the equatorial plane, consists mostly of electrons with energies below 10 MeV. The origin of these electrons is via injection from the outer magnetosphere. Unlike the inner belt, the outer belt is very dynamic, changing on time scales of a few hours in response to perturbations emanating from the outer magnetosphere. In regions not too far distant (*i.e.*, less than 10 Earth radii) from the Earth, the geomagnetic field can be approximated as a dipole field, which is tilted with respect to the Earth's rotational axis by an angle of ~11°. The intersection between the magnetic and rotational axis is located ~500 km more to the North, above the centre of the Earth. Because of this tilt and translation, the radiation belts are closest to the Earth's surface over the South Atlantic Ocean. This region is called the South Atlantic Anomaly (SAA) and is of great significance to space vehicles that orbit the Earth at several hundred kilometres altitude. These orbits take them through the anomaly periodically, each time exposing them for several minutes to increased radiation levels. The high SAA proton fluxes were explained to give rise to light flash phenomena in the eyes of astronauts (Casolino *et al.*, 2003).

Galactic cosmic rays and energetic particles generated in large solar flares finally interact with the Earth's atmosphere to produce in cascade-like reactions hadron, lepton and photon fields at aircraft altitude. The energy spectra of these secondary particles extend from the lowest possible energy to more than 10^{18} eV (O'Brien *et al.*, 1996). The total flux of ionizing particles in the upper atmosphere is fairly constant from 150–50 km altitude. Below 50 km the flux increases due to build-up of cascades and reaches the so-called Pfotzer maximum at about 15–20 km above sea level where absorption starts to dominate. The geomagnetic field deflects the incoming cosmic rays, depending on their rigidity, *i.e.*, momentum per unit charge, and angle of incidence. The vertical critical rigidity is zero at the magentic poles and at its maximum near the magnetic

[1] Van Allen was actually trying to measure the GCR flux in deep space, to see if it was similar to that measured on Earth. However, the flux of energetic particles detected by his instruments so greatly exceeded the expected value that it prompted one of his co-workers to exclaim, "My God, space is radioactive!"

equator. As a consequence, the primary (and secondary) cosmic-ray flux shows a distinct latitude effect. With respect to dose equivalent, atmospheric neutrons are the most important particles at aircraft altitude. They are produced as evaporation products of highly excited nuclei to form a peak around 1 MeV, and in peripheral collisions or charge exchange reactions of high-energy protons with a maximum flux around 100 MeV (Hajek *et al.*, 2004a). The dependence of neutron production on solar activity is most pronounced in polar regions, while the variation around the equator is just about 5%, since low-energy primaries are shielded by the Earth's magnetic field and high-energy particles undergo only slight solar modulation. The higher energy of primary cosmic rays entering the atmosphere around the equator causes the created neutrons to be able to penetrate deeper into the atmosphere, compared with pole-near latitudes. The maximum of the neutron flux is thus found at about 120 g cm^{-2} at the equator and at about 75 g cm^{-2} in polar regions. Considerable fluxes of neutrons are also produced when a strong solar flare hits the Earth.

Space crew exposure and radiation protection

Space travellers are facing exposures to radiation levels that may easily exceed those routinely received by terrestrial radiation workers. Missions in low-Earth orbit (LEO) are not exposed to the full intensities of the GCR and SPE spectra because of the protection afforded by the Earth's atmosphere and magnetic field. Hence, particle fluence rates are much lower than will be encountered in interplanetary missions—about a factor of three from the International Space Station (ISS) to deep space, where no protection from the magnetosphere or planetary bulk exists. The degree of protection is a function of spacecraft orbital inclination and altitude. For the 51.6° orbit of the ISS, typical dose equivalent rates are between 0.5 and 1.2 mSv d^{-1} (Berger, 2008; Hajek *et al.*, 2008), with ~75% coming from GCR and 25% coming from protons encountered in passages through the SAA region of the radiation belts (NCRP, 2006). For high-inclination space missions in LEO, only 25–30% of SPE protons are intercepted due to geomagnetic shielding, while the contribution of SPEs to the radiation load of astronauts is mostly negligible for low-inclination orbits (Benton and Benton, 2001). Outside the protection offered by the geomagnetic field, doses received from a major SPE in less shielded modules might reach lethal levels within a couple of hours. Hence, radiation shelters have to be provided to minimize the health risks for astronauts.

Radiation transport codes, which model the atomic and nuclear interactions of the cosmic-ray particles, are usually applied to describing how the external radiation fields are altered by passage through the spacecraft structure (Sihver, 2008). However, the high degree of complexity of both the shielding distribution and the generation of secondary charged and uncharged radiation makes it virtually impossible to simulate in detail the variation of the resulting particle fluence and energy spectra of the radiation field constituents within a space vessel. Unlike the situation for terrestrial exposures, the high costs of launching materials into space place limitations on spacecraft size and mass and preclude the purely engineering solution of providing as much additional shielding as needed to reduce radiation exposure to some desired level. Some model predictions indicate that some types of shielding materials may even give rise to secondary radiation environments that are more damaging than the unattenuated primary fields, which produced them.

Table 1. 10-year career limits for stochastic radiation effects applicable to missions in low-Earth orbit. Limits are expressed in effective dose (*E*). Recommendations by NASA and JAXA are age and gender specific (male/female).

NASA		Roscosmos	JAXA		CSA
Age, yrs	*E*, Sv	*E*, Sv	Age, yrs	*E*, Sv	*E*, Sv
25	0.7 / 0.4	1.0	25−29	0.6 / 0.6	1.0
35	1.0 / 0.6		30−35	0.9 / 0.8	
45	1.5 / 0.9		36−39	1.0 / 0.9	
55	2.9 / 1.6		≥ 40	1.2 / 1.1	

The development of radiation protection recommendations for astronauts reflects the current knowledge about radiation risks, which is based to a large extent on cancer incidence and cancer mortality in the atomic-bomb survivors of Hiroshima and Nagasaki. Since until now only few experimentally verified data on the biological effectiveness of heavy ions and the dose distribution within the human body exist, the concepts of terrestrial radiation protection are of limited applicability to human spaceflight, except for the principles of justification and optimization (ALARA). Radiation protection limits for astronauts are designed to prevent deterministic or non-cancer hazards and reduce the risk of stochastic effects to an acceptable level. Instead of applying the annual dose limits for workers on ground also to astronauts, whose careers are of comparatively short duration, the overall lifetime risk is used as a measure. The selection of dose limits for stochastic effects are related to the risk for fatal cancers (solid tumours and leukaemia) as well as for genetic effects. While radiation protection in the pre-Apollo era was concerned primarily with the avoidance of exposures, which might deteriorate the operational performance of astronauts, the first genuine radiation protection guidelines of the U.S. Space Science Board (NAS/NRC, 1970) allowed doubling of the spontaneous incidence of malignant tumours[2]. In 1989, the National Council on Radiation Protection and Measurements (NCRP) proposed age and gender specific dose limits for a 10-year career on the basis that a lifetime excess risk of cancer mortality of 3% was acceptable (NCRP, 1989). This risk was comparable with the risk in less safe but ordinary occupations, such as agriculture and construction. However, it is lower than the 5% lifetime risk that a radiation worker on ground would incur if the present annual protection limits were exhausted (20 mSv per year over 50 years). The increase of risk factors for fatal cancers per unit dose by UNSCEAR and BEIR V required reappraisal of the effective dose limits (Table 1), which were published in NCRP Report 132 (NCRP, 2000).

The Russian Federal Space Agency Roscosmos allows an annual limit of 500 mSv, and—in agreement with the Canadian Space Agency (CSA)—a career limit of 1 Sv, both independent of age and gender, since Russian studies yielded an increasing probability of non-cancer radiation effects with age that compensates the decreasing cancer risk (Roscosmos, 2004). The Russian career limit corresponds to an excess risk between 4.6% (at 30 years of age) and 2.4% (at 50 years of age). Like their U.S. analogue, the dose limits defined by the Japan Aerospace Exploration Agency

[2] In industrialized countries, the spontaneous cancer incidence is on average 20–25%.

(JAXA) depend on age and gender, but differ in the tolerated dose values and the age structure (Table 1). The associated excess risk is ~3%, but never exceeds 5%. The European Space Agency (ESA) based its radiation protection concept on the recommendations of the International Commission on Radiological Protection (ICRP, 1991) and the European Council Directive 96/29/Euratom (European Commission, 1996), both of which do not explicitly classify astronauts as radiation workers[3]. The limits applied to European astronauts are thus based on thresholds for deterministic radiation effects in dedicated organs and tissues (Straube *et al.*, 2010).

Dose limits for acute deterministic effects in the bone marrow, lens of the eye and the skin (Table 2) are expressed in gray equivalents (Gy-Eq), in which the organ absorbed dose is weighted by multiplication with the appropriate relative biological effectiveness (RBE) for a specific radiation quality and endpoint. The concept of Gy-Eq became necessary, since the radiation weighting factors used for stochastic effects do not apply to deterministic detriments. Considering the significant uncertainties in assessing RBE at low dose and dose rate, the values on which the dose limit recommendations are based have been determined at the threshold doses for the regarded deterministic effect (ICRP, 1989; Urano *et al.*, 1984).

For long-term missions outside Earth's magnetic field, the acceptable level of risk has not yet been defined, since there is not enough information available to estimate the risk of effects to the central nervous system and of potential non-cancer radiation health hazards (cataracts, cardiovascular diseases, etc.). Available data and pending questions have been compiled in NCRP Report 153 (NCRP, 2006), which will form the basis for developing radiation protection guidelines for missions into deep space.

Table 2. Recommended organ dose limits in Gy-Eq applicable to missions in low-Earth orbit. All limits are independent of age and gender.

Organ		NASA	Roscosmos	JAXA	ESA	CSA
Bone marrow, Gy-Eq	Acute	0.25	0.15	–	–	–
	30 d	0.25	0.25	–	0.25	–
	1 yr	0.5	0.5	0.5	0.5	–
	Career	–	–	–	–	–
Eye, Gy-Eq	Acute	–	–	0.5	–	–
	30 d	1.0	0.5	–	0.5	–
	1 yr	2.0	1.0	1.0	1.0	–
	Career	4.0	2.0	5.0	–	4.0
Skin, Gy-Eq	Acute	–	–	2.0	–	–
	30 d	1.5	1.5	–	1.5	–
	1 yr	3.0	3.0	4.0	3.0	–
	Career	6.0	6.0	20.0	–	6.0

[3] A task group appointed by the ICRP in 2006 shall develop recommendations for human space missions in low-Earth orbit and beyond.

Aircrew exposure and radiation protection

Aircraft passengers and crew are subject to elevated levels of secondary cosmic radiation produced in the atmosphere, the aircraft structure and its contents. From the beginning of the first commercial supersonic Concorde operations, measurements on board passenger aircraft became attractive and contributed to the vast pool of data available today. Total exposure on a given flight depends on the particular air route in terms of altitude (pressure rather than radar altitude) and geomagnetic latitude, as well as on solar activity and the duration of the flight. The dose rate increases with altitude and geomagnetic latitude, reaching a maximum at 15–20 km and a constant level above ~55°, respectively. As a rule of thumb, the effective dose from neutrons in polar regions is enhanced by a factor of ~6 compared with the equator, while the dose delivered by the directly ionizing component increases only by a factor of ~2. Commercial subsonic aircraft generally have cruising altitudes of 7–12 km. The effective dose rate at an altitude of 8 km in temperate latitudes is typically up to ~3 μSv h^{-1}, but decreases to only ~1–1.5 μSv h^{-1} near the equator (Fig. 2). At 12 km, the values are greater by about a factor of two. The dose for a return trans-Atlantic flight is typically 60–70 μSv (Hajek *et al.*, 2004b). The annual hours flown by crew members varies from individual to individual and from airline to airline, depending on policy. The average appears to be 300–900 hours per year. The annual effective doses received by aircraft crew usually lie within 2–4 mSv, with only few crew members receiving higher doses. At aircraft altitude and temperate latitudes, representative values of the main components of effective dose are neutrons 55%, electrons and positrons 20%, protons 15%, photons 5% and muons 5% (Bartlett, 2004). At sea level, the dominant component of effective dose is the muon component.

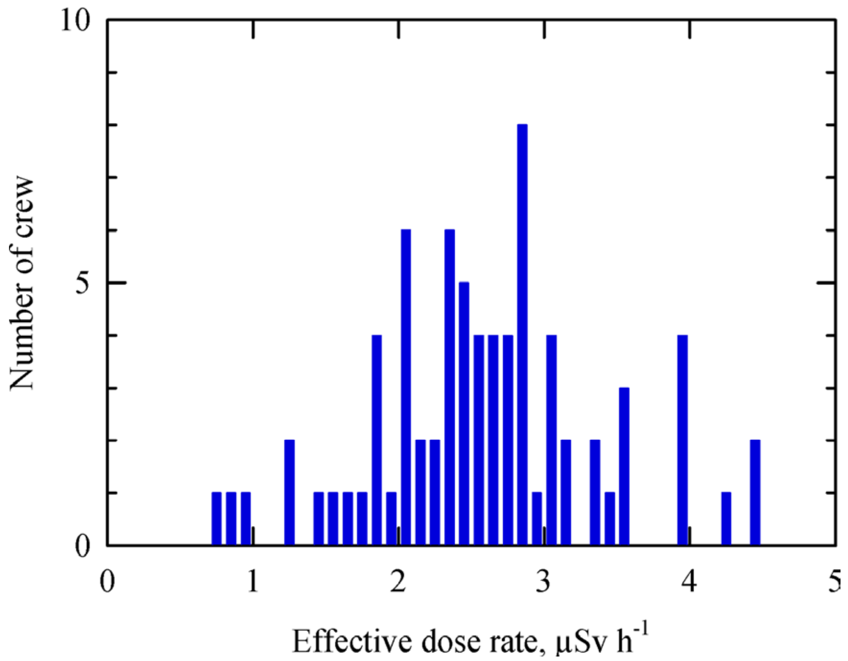

Fig. 2. Histogram of Tyrolean Airways crew radiation exposure on short- and medium-haul flights. The distribution of measured effective dose rates peaks between 2 and 3.5 μSv h^{-1}. For an average of 750 flight hours per year, effective dose can be estimated to result in 1.5–2.6 mSv.

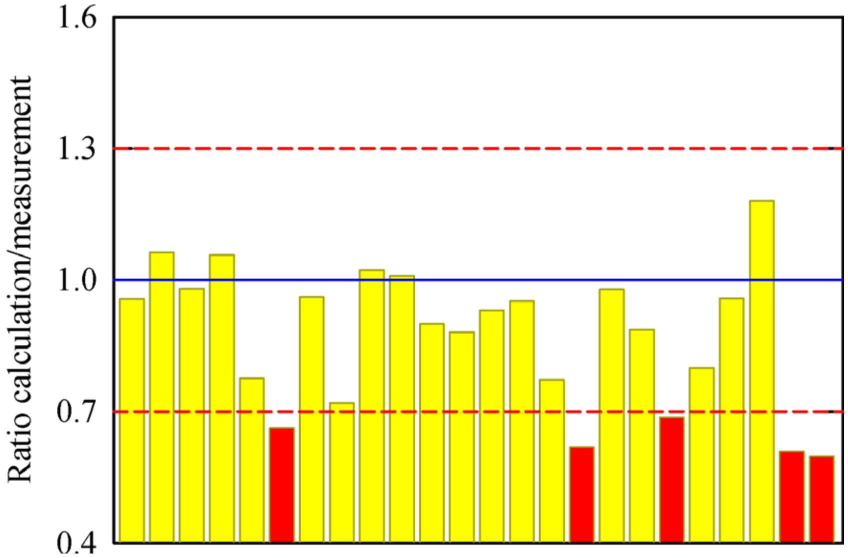

Fig. 3. Comparison of computed and measured route doses for 24 Tyrolean Airways flights. Agreement within ±30% is indicated, as required by Austrian regulations. There is generally good agreement between the results of calculations using CARI-6M and experimental determinations, except for very short-haul flights operated by propellant aircraft at low altitude.

There are a number of radiation transport codes and programmes to calculate dose rates and route doses in current use. The radiation transport codes take as input the cosmic radiation field at the top of the atmosphere and solve, either analytically or by Monte Carlo simulation, the radiation transport equations, which describe the interactions of each particle with the constituents of the atmosphere, in order to calculate the field at a given aircraft altitude and geographic location. The effect on particle trajectories of the Earth's magnetic field is included in approximations using tables of rigidity cut-offs. The programmes take account of the effects of IMF variation by applying an equivalent heliocentric electrostatic field. Generally, there is good agreement (Fig. 3) between the results of calculations and experimental determinations (Lindborg *et al.*, 2004).

Following ICRP recommendations (ICRP, 1991), the European Union (EU) introduced a revised Basic Safety Standards Directive (European Commission, 1996), which, *inter alia*, included the exposure to enhanced levels of cosmic radiation. The Directive requires account to be taken of the exposure of aircrew personnel liable to receive effective doses of more than 1 mSv per year. It further identifies the following protection measures (Bartlett, 2004): (i) to assess the exposure of the crew concerned; (ii) to take into account the assessed exposure when organizing working schedules with a view to reducing the doses of highly exposed crew; (iii) to inform the workers concerned of the health risks their work involves; and (iv) to apply the same special protection during pregnancy to female crew irrespective of the 'child to be born' as to other female workers. The EU Directive has already been incorporated into laws and regulations of the majority of the EU Member States and has been included in the aviation safety standards and procedures of the Joint Aviation Authorities (JAA). The preferred approach, supported by guidance from the European Commission and ICRP Publication 75 (ICRP, 1997), is that where the assessment of the exposure of aircraft crew to cosmic radiation is necessary, doses can be computed from staff roster information, flight profiles and calculations of cosmic radiation dose rates as a function of altitude, geomagnetic latitude and solar modulation. The calculations are to be verified by measurements.

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Stakeholder involvement and engagement

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Abstract
One of the greatest challenges facing radiation protection professionals today is, how to include the society in radiation protection decision making. In response to this issue, internationally accepted guiding principles for radiation protection professionals on stakeholder engagement have been published by the International Radiological Protection Association (IRPA). IRPA goal is to ensure that consensus on directions for improvement of stakeholder involvement programmes is reached among radiation protection professionals, and that these guiding principles are taken into account during the development of future stakeholder involvement programmes.

Introduction
This work has been supported by Finnish Radiation and Nuclear Safety Authority, and parts of it have been published previously by International Radiation Protection Association and Nuclear Energy Agency within the Organisation for Economic Co-Operation and Development.

Stakeholder involvement and engagement are often used interchangeably when referring to public participation. In this article stakeholder involvement will refer to the policy and action taken with the stakeholders, whereas stakeholder engagement will refer to the process of how to engage with the stakeholder.

What is Stakeholder Involvement? Stakeholder involvement comprises of effective interaction between private and public companies, regulators and communities to make sure future activities and plans are discussed and understood by all parties. In decision making the stakeholder involvement is not intended to be stand alone process. Rather, it is designed to be an integral part of the decision-making and management processes, and it only has meaning if all parties have this intent from the outset. Whatever the level of involvement is, the distinction has to be maintained between the extent to which participants are responsible for the decision making and the level of interaction with stakeholders. Once this has been achieved the open and transparent approach of stakeholder engagement can help in establishing and maintaining effective relationships between all stakeholders (IRPA 2007).

History of Stakeholder involvement in Radiation Protection

Stakeholder involvement is not a new idea, but we are learning more and more of its application and benefits. Until the 1950s, the so called "managerial model" defined the relationships between the authorities and the public (Beierle and Crayford, 2002). In the "managerial model" the government administrators and experts in government agencies, are committed to pursuing the common good. In addition, they have been deliberate choices as possible so that the public interest and their policy have been used to produce the greatest benefit to most people in the long run. The signing of the Administrative Procedure Act in the United States in 1946, was a definite sign that the "managerial model" was reaching the end of the road. Soon other developed economies followed this trend with similar legislations. As the economies developed further and became more and more industrialised, the governments were faced with difficult decision making on matters related to environmental and health issues in particular (Beierle and Crayford, 2002, OECD/NEA, 2006).

In the 1970s, pluralism began to replace the "managerial model" as the primary approach to administrative decision making (Reich, 1985). According to this pluralist view, government regulators and politicians were not asked to have the impossible role of objective guardians and decision makers in the public interest, but rather be arbiters among different possible interests within the society. It was recognised that everlasting "common good" has no objective meaning, but rather that public interest has to be debated and arrived at by negotiation among different interested parties (Williams, 1995). This attitude was reflected in the 70s and 80s in many decisions where the role of public industrial (including nuclear) agencies was revised, with a clear separation of agencies having regulatory responsibilities and agencies aiming at promoting industrial development and applications (OECD/NEA, 2006). This separation of regulator and operator functions has since that time been a necessary element of any developed regulatory system (IAEA, 1996).

In the 1990s the "pluralist model" came under pressure as even more intense participatory perspective was expected by the stakeholders; with the aim to have consensus outcome rather than unsatisfying compromise, and from a desire to have long term sustainable decisions rather than immediate decisions from a better funded opposition (Beierle and Crayford, 2002, OECD/NEA, 2006). This "democratic model" stresses the importance of the act of public participation, not only in influencing decisions but also in strengthening public capabilities and social capital (Beierle and Crayford, 2002), emphasises interaction among often adversarial interests, but that interaction is viewed less as competitive negotiation than as a way to identify the common good and subsequently act on shared communal goals (Dryzek, 1997). From the public perspective, the act of participation *"makes people more aware of the linkages between public and private interests, helps them develop a sense of justice, and is a critical part of the process of developing a sense of community"* (Laird, 1993).

In today's regulatory environment stakeholder involvement provides the decision-framing, decision-aiding and decision making processes along the lines of the "popular-pragmatistic" model. The goal of radiological protection is, and has always been, to provide an appropriate standard of protection for man and environment. At the same time the justification principle requires, that consideration is given to the benefits arising practices, which give rise to radiation exposure, and that doses are optimised in accordance to the optimisation principle (ICRP, 2007). These principles are the cornerstones of radiological protection, and they can not be sidelined in any radiation protection decision making. The implementation of these principles was first thought to be possible through semi-quantitative analytical methods, such as cost benefit analyses. While these approaches provide a great guideline it has to be complemented with a genuine public participation.

The result of the evolution of the radiological protection system, towards increasing public participation in environmental and health related decision-making, can be fully appreciated in the current ICRP Recommendations (ICRP, 2007), where a paragraph is devoted to the participation of stakeholders.

"This decision making process may often include the participation of relevant stakeholders rather than radiological specialists alone."

Stakeholders

To understand how to involve the stakeholder, it is first necessary to understand what a stakeholder is and what it will be defined as in this paper. In the simplest form "A stakeholder is a person or other actor with special concern and interest in an issue, and may be considered to be concerned either on the basis of self report or on the basis of observed activities", (Sjoberg, 2003). Significantly, stakeholders bring differing perspectives to the process, given the range of backgrounds and disciplines that stakeholders are drawn from. For instance, some internal and external stakeholders may have a reasonable degree of commonality of interest or a legal obligation within the subject under discussion. These are sometimes referred to as 'true stakeholders'. But other stakeholders become involved because they are affected by the decisions taken, or have a strong view on the conduct of companies and government organisations, even if their interests are very different from those of the organisation in question .

For radiation protection issues the following stakeholders can be identified:

- Regulatory Stakeholders ('true stakeholders'). All practitioners, who require a 'licence to operate' from defined stakeholders, such as radiation protection regulators, health and safety regulators or environmental regulators. In some countries, there are also legal requirements to consult statutory stakeholders with a regulatory or quasi-regulatory function in many contexts (especially planning-related).
- Campaign groups often see themselves as having a 'license to operate' or watchdog role, but they are also significant as opinion formers, able to influence other stakeholders particularly through the media.
- The media are sometimes considered to be stakeholders, but are more often considered separately alongside other opinion formers. There is usually no strong commonality of interest between the media and industry. The media has, almost by definition, the potential to exert considerable influence on other stakeholders, including the public.
- The local community is the main target in many stakeholder programmes. However the community cannot be treated as a single entity, relationships between practitioners and the above are contained within it. The people who live around affected by the decisions and the community are complex and

all the different types of stakeholder described community groups and local authorities that speak for them, have a wide range of inter-relationships and perspectives. There is usually no such a thing as 'the community view'.

- Industry and commerce are often neglected in many stakeholder programmes. Similarly local communities, it is impossible to identify industry a joint commerce view, as there a number of interests and opinions represented in industry and commerce.

To engage with the stakeholders and fully realise the benefits of involving stakeholders in the decision-making processes International Radiological Protection Association (IRPA) have developed Guiding Principles for Radiation Protection Professionals on Stakeholder Engagement.

Guiding Principles for Radiation Protection Professionals on Stakeholder Engagement

The need for Guiding Principles on stakeholder engagement were first identified by IRPA in 2004, when it was agreed that stakeholder involvement, should play an important and integral part in decision-making processes related to radiological protection. The aim of the guidance was to help radiation protection professionals to understand the objectives, requirements and demands of stakeholder involvement, encourage participation and provide a framework for establishing a constructive dialogue with other stakeholders.

According to the IRPA guidance radiological protection professionals should endeavour to:

1. Identify opportunities for engagement and ensure the level of engagement is proportionate to the nature of the radiation protection issues and their context.
2. Initiate the process as early as possible, and develop a sustainable implementation plan.
3. Enable an open, inclusive and transparent stakeholder engagement process.
4. Seek out and involve relevant stakeholders and experts.
5. Ensure that the roles and responsibilities of all participants, and the rules for cooperation are clearly defined
6. Collectively develop objectives for the stakeholder engagement process, based on a shared understanding of issues and boundaries.
7. Develop a culture which values a shared language and understanding, and favours collective learning.
8. Respect and value the expression of different perspectives.
9. Ensure a regular feedback mechanism is in place to inform and improve current and future stakeholder engagement processes.
10. Apply the IRPA Code of Ethics in their actions within these processes to the best of their knowledge.

Case Studies

Radiation protection regulators around the world are in a situation where they are regulating activities at NPPs and other facilities with radiation risks, which were constructed in a previous era without true stakeholder involvement in the associated decision making. However, with the recent developments in the nuclear industry, it is anticipated, that a variety of stakeholders will seek participation in decision making as new plants are developed and old ones decommissioned.

IRPA, ICRP, IAEA and OECD/NEA have all concluded that the expectations of stakeholders of a right to participate in decision making, is something that the radiation protection authorities should address, and in some countries already do. Decisions regarding such matters as policy, regulating radiation practices and siting and construction of NPPs, should be voluntarily opened up to all stakeholders from the closed domain of technical experts and government officials (IAEA, 2006, IRPA, 2007). Below are a few case studies demonstrating the application of stakeholder involvement in radiation protection, full details of these case studies can be seen in Appendix 1, case studies are taken from a OECD/NEA Expert Group on Stakeholder Involvement and Organisational Structures.

Stakeholder Involvement in mobile phones and radiation (Sweden, SSI)

The Swedish Statens strålskyddsinstitut (SSI) has responsibility for ionising and non-ionising radiation. This case study concerned debate around new mobile phone masts. A decision was taken at the European level and licenses issued in Sweden by the post and telecommunications authority for construction of a network for a 'third generation' of mobile phones. However, concerns amongst sections of the population over the health impacts of radiation from the masts associated with the network led to opposition to construction of masts. SSI's risk assessment concluded that the risk from the masts was essentially zero. To understand the views of those concerned by the masts, to explain its own view and to allow dialogue with other stakeholders, SSI organised three seminars around the topic. Although the seminars did not result in consensus, stakeholders discussed with each other and were able to improve their own understanding of other stakeholder roles and views.

Stakeholder Involvement in NPP Siting (Finland, STUK)

The Säteilyturvakeskus (STUK) has a role as an expert body and regulator for the new nuclear power plant at Olkiluoto in Finland. STUK's role was essentially one of providing technical expertise in an open fashion, in a context where local and national elected representatives had to explicitly approve construction of the power plant. STUK provided information on the regulatory process and its work in it, responding to address topical, yet potentially sensitive issues such as aircraft impact. It was also willing to interact with questioning stakeholders. The approach used was open but, broadly speaking, more reactive than proactive and seems to have resulted from a general presumption of openness in STUK. STUK's approach appeared to be effective and 'fit-for-purpose'.

Stakeholder Involvement in post-Chernobyl contamination monitoring (Belarus, CORE)

Generally radiological protection is concerned with 'normality' in 'normal' situations i.e. tasks are entrusted to professionals. However, the situation is somewhat different in the contaminated territories where it becomes even more important to balance radiological protection with other aspects of life. Indeed, 'integrate' is probably a better word, since the question is one of co-working with civil society, rather than arriving at a better informed decision or improving credibility. Thus, the question moves from 'regulation' to 'governance'. A lesson to be learned from experiences in Belarus is that for the public in general to share in governance of a topic, it is necessary for people to fully appreciate all the issues and feel like they have created a real partnership in the decision processes affecting their safety and living conditions.

Stakeholder Involvement in UK Nuclear Decommissioning Authority (UK, NDA)

The Nuclear Decommissioning Authority (NDA) is a public body set up to run decommissioning of United Kingdom civil nuclear facilities. The actual decommissioning is carried out by contractors. Drawing on experience in the nuclear industry and on the findings of a government commissioned report, stakeholder involvement was recognised as an important factor and included in the NDA from its inception. By re-invigorating previous site local liaison groups (run by the operators), the NDA set out to involve stakeholders in its work in a local and national framework. Local liaison groups include representatives from civil society, local municipalities, regulators and contractors as well as the NDA. One of the aims of introducing the national tier was to link local concerns to national, strategic considerations. Through founding legislation and its own policies, the NDA retains a firm commitment to stakeholder involvement in its work.

Stakeholder Involvement in Policy Formulation (USA, USNRC)

The United States Nuclear Regulatory Commission (USNRC) is a federal regulator tasked with nuclear regulation. In developing Below Regulatory Concern (BRC) policy USNRC consulted their stakeholders to achieve a sustainable decision.

This process began in earnest in the early 1990's, but it had limited success. USNRCs most recent effort to consult stakeholders on clearance policy was termed NUREG-1640 rulemaking (NRC, 2002). This programme built on the early BRC stakeholder programme, which started with good intentions, but it polarised the views of the stakeholders further it went along. Consequently four of the eight environmental and consumer groups that had been actively involved from the beginning of the BRC policy formulation refused to enter a proposed consensus building process and the programme ended in failure. The new process centred around a series of public meetings, in which the stakeholders were asked to participate. For their detriment USNRC had lost the trust and confidence of some of the important stakeholders during the first BRC policy formulation and had not done enough in the meantime to engage the public in any effective manner to regain the trust of the stakeholders. This scepticism and lack of trust again led to some national environmental and consumer advocacy groups to boycott the public meetings. Due to the stakeholder feedback USNRC decided to postpone the rulemaking and carry on with case by case clearance.

Conclusions

Stakeholder involvement is a key concept in modern approaches to decision making and it has received considerable attention from IRPA, ICRP IAEA and OECD/NEA (IRPA, 2004, 2008, ICRP, 1991, 2007, IAEA 2006, OECD 2003, OECD/NEA, 2001a, 2001b, 2003, 2004). It is worthwhile to mention the role played by IRPA in identifying the need for the Guiding Principles for Radiation Protection Professionals on Stakeholder Engagement. These principles were produced to help radiation protection professionals to understand the objectives, requirements and demands of stakeholder engagement, encourage participation and provide a framework for establishing a constructive dialogue with other stakeholders.

Taking into account the views of different stakeholders is sometimes considered a time-consuming process leading to solutions technically not optimised. As the case studies have shown stakeholder involvement is often essential in obtaining societal acceptance of decisions. It should be noted, however, that in some cases stakeholders will not wish to accept decision, and that for them the perfect solution is the "do nothing" option sustaining the status quo. Under these circumstances, the regulators will have to decide whether it is best to go forward or not. However, sometimes as highlighted in the case studies this type of arbitrary decisions will result in stakeholders seeking to influence the decisions through courts or even through direct action.

Significant benefits can be gained from the involvement of stakeholders. For radiation protection the most quoted are:

- Responds to shifts in societal attitudes to science, industry and government.
- Offers possibility of resolving tensions between economic and social concerns.
- Helps to prevent disputes and conflicts where it is deployed ex-ante.
- Helps to resolve disputes and conflicts where it is deployed ex-post.
- Increases the substantive quality and sustainability of decisions.
- Builds trust in institutions.
- Educates and informs the public.

To realise these benefits it has to be appreciated different stakeholders have different perspectives, perceptions, beliefs, interests and values. Having clear aims and objectives from the outset will assist in planning a dialogue process without conflict. Also it should be made clear who is responsible for making the final decision. In order to arrive at this point the roles of the regulators and stakeholders, should be well defined. To measure success, evaluation criteria must be developed to evaluate the process with the stakeholders that will be participating.

Even in the most inclusive and transparent stakeholder involvement programmes, there can be conflict between the equal opportunity to participate and influence both programme and outcomes for anyone who affected by the decision, and competent participation, i.e. construction of the most valid, considering both societal and technical aspects. To ensure this seen as an opportunity and a mutual learning process the programme has to be initiated early on, and efforts must be made to build trust and consensus among the stakeholders.

Once the *"optimum solution under the circumstances"* the involvement of local stakeholders must continue with the aim, of demonstrating to all parties the compliance with the objectives of the programme.

As the case studies have shown decision making is not always successful, even with the help of stakeholder involvement, but more than often stakeholder relations and legitimacy of radiation protection professionals can be improved through stakeholder involvement. This is an arduous path to take, but if the field of radiation protection is to involve the stakeholders in future decision making it is the only way to proceed.

To make this process a little simpler, IRPA have published the Guiding Principles for Radiation Protection Professionals on Stakeholder Engagement. These guiding principles should provide a good basis for running any stakeholder involvement programmes in the field of radiation protection.

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Decommissioning and waste management

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Abstract

Decommissioning of nuclear installations gives rise to large quantities of different types of radioactive material. Depending on its type, the overall mass of the controlled area of a large nuclear power plant has a mass of 100,000 to 400,000 Mg. The most important material streams are activated metals from the reactor pressure vessel and its components, activated concrete from the biological shield and the surrounding building structures, contaminated materials from systems, components and other structural elements, contaminated concrete from building structures, electrical installations, insulation material etc.

Decommissioning projects are in progress in a large number of countries around the world. In many countries, decommissioning has developed to be a mature industry. Techniques for dismantling and decontamination have been developed and are now available for all tasks and materials. This is also true for management of radioactive waste originating from decommissioning. Solidification, incineration, melting of metals, encapsulation and supercompaction are examples for techniques for treatment and conditioning of radioactive waste. The final disposal of radioactive waste from decommissioning is treated differently in various countries. Some countries use near-surface repositories for such wastes, which mainly consists of low-level waste, while others rely on deep geological repositories.

Only a small portion of the overall material from decommissioning has to be disposed of as radioactive waste. The vast majority of material can be cleared (that means released from radiological control). A thorough radiological characterisation is the prerequisite for clearance, for which, however, material from decommissioning of nuclear power plants has suitable properties. The radionuclides involved in material from nuclear power plants generally consist of a large portion of gamma emitters, so that the concept of nuclide vectors can be used. The measurements can be based on key nuclides like Co-60 and Cs-137, and the activity of nuclides that are not easy to measure can be derived from these values. Clearance procedures and clearance levels have been introduced by a large number of countries, making clearance are very important option in material management. Various measurement techniques are available for performing clearance.

Introduction

This paper addresses two topics that are closely related:

- decommissioning of nuclear installations and
- management of materials originating from decommissioning.

Decommissioning is the final phase in the lifecycle of nuclear installations after siting, design, construction, commissioning and operation. It is a complex process involving operations such as detailed surveys, decontamination and dismantling of plant, equipment and facilities, demolition of buildings and structures, site remediation, and the management of resulting waste and other materials. Decommissioning gives rise to a large amount of various types of material – a few 100,000 Mg for large NPP blocks – that can be treated and disposed of as radioactive waste or can be decontaminated and cleared (released) as non-radioactive material. This is the reason why the decommissioning phase of nuclear installations poses much larger challenges for the waste management than the operating phase.

This paper tries to address all aspects of this wide range of topics at least briefly: It first describes the decommissioning as a global task and as a process for a particular facility (restricting itself to nuclear power plants as the type of nuclear facilities from which the largest amounts of decommissioning waste arises), then lists the most important material streams originating from decommissioning and the related material quantities, and finally covers both management of radioactive waste (i.e. of material that has to be disposed of because of its activity) and clearance of material (i.e. the release of material that is either not contaminated above clearance levels or that complies with these levels after decontamination). As the topic is very wide, the information given in each section has to be necessarily succinct.

When analysing the development of decommissioning projects over the last two decades, it becomes clear that the vast majority of issues concerning techniques for decommissioning and dismantling, decontamination, waste management including treatment and packaging, measurement and characterisation, as well as clearance and release have been solved. Standard procedures and techniques are available off the shelf and can be deployed in all types of nuclear installations. Decommissioning has become a mature field of work. Nevertheless, each decommissioning projects needs its individual planning and adaptation of the techniques to its distinctive features. Waste management options that are available in a specific country will also influence the choice of techniques and procedures, even the choice of the decommissioning strategy. Therefore, both topics are closely interrelated, and this paper tries to highlight these interrelations.

Generally, it is assumed that the fuel elements would have been removed before start of decommissioning. There are, however, a number of projects where decommissioning has started with fresh and spent fuel on the site. In order to reasonably limit the scope, removal and storage of spent fuel is not dealt with in this paper. Likewise, operational waste that is still present at the beginning of decommissioning can usually be dealt with in the same way as during operation and is therefore also not included in this paper.

Decommissioning

Decommissioning of nuclear installations is not just the opposite of its construction. The aspects of radiation protection, management of partly highly activated and contaminated material and of large quantities of material for which the absence of contamination has to be verified, the necessity of remote operation in some areas where space might be tight and other features make decommissioning of nuclear installations a technically challenging and interesting field. In this chapter, decommissioning strategies that are viable for nuclear power plants and other nuclear installations are presented, followed by an overview of decommissioning projects and is decommissioning techniques.

Decommissioning strategies

There are two basic strategies that are widely used for decommissioning: immediate or early dismantling and deferred dismantling. A third strategy, entombment, leaves the facility and structures in place. This option, however, is rarely used. (IAEA, 2007)

Early Dismantling: Immediate or early dismantling is the strategy in which the equipment, structures, components and parts of a facility containing radioactive material are removed or decontaminated to a level that permits the facility to be released for unrestricted use (clearance) as soon as possible after permanent shutdown. In some cases, where unrestricted release is not feasible, the facility may be released from regulatory control with restrictions imposed by the regulatory body. The implementation of the decommissioning strategy begins shortly after permanent termination of operational activities for which the facility was intended, normally within a few years. Early dismantling involves the prompt removal and processing of all radioactive material from the facility for either long term storage or disposal. Non-radioactive structures may remain on-site. This is the preferred decommissioning strategy.

Deferred Dismantling: Deferred dismantling is the strategy in which the final dismantling of the facility is delayed and the facility is placed into long term storage where it is maintained in a safe condition. This strategy may involve some initial decontamination or dismantling, but a major part of the facility will remain for a certain time period in a caretaker mode. This time period might range from a few years to several decades, after which time the decommissioning process will be completed and the facility can be released from regulatory control. The deferred dismantling option is often used at multi-facility sites when one or more of the facilities are shut down while others continue to operate. This is especially true of facilities that share some common systems.

Entombment: Entombment is the strategy in which the radioactive contaminants are encased in a structurally long lasting material until the radioactivity decays to a level that permits release of the facility from regulatory control. The fact that radioactive material will remain on the site means that the facility will eventually become designated as a near surface waste disposal site and criteria for such a facility will need to be met.

Figure 1 shows an example for the early dismantling option, i.e. the transformation of a nuclear power plant site from the operational status to (nearly) green field conditions. Some buildings may be left in place, in this case buildings housing interim storage facilities that have to remain on site until the waste may be removed to a repository. The rest of the buildings of the controlled area as well as other have been cleared and removed.

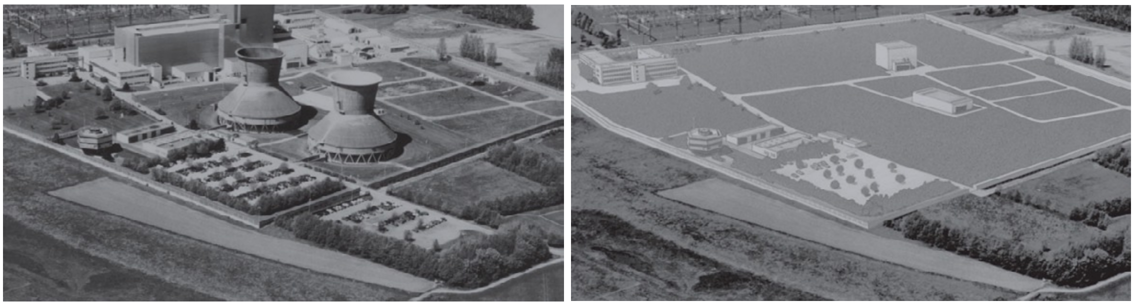

Figure 1. The site of NPP Würgassen (Germany). Left: before start of decommissioning (1995), right: after reaching green field conditions (artist's impression) (E.ON 2008).

Overview of Decommissioning projects

A large number of decommissioning projects exist worldwide. Table 1 gives an overview of shutdown power reactors in various countries that are under decommissioning (either early or deferred dismantling) or are awaiting decommissioning. There is an even larger number of research reactors that have also been permanently shut down or that are undergoing decommissioning. This overview makes it clear that decommissioning in many countries has become an industry handled by specialists rather than an experimental field that could be dealt with by the former operator of the facility alone.

Table 1. Shutdown reactors by country (IAEA 2010).

Country	Number of Units	Total MW(e)
Armenia	1	376
Belgium	1	10
Bulgaria	4	1632
Canada	3	478
France	12	3789
Germany	19	5879
Italy	4	1423
Japan	5	1618
Kazakhstan	1	52
Lithuania, Republic of	2	2370
Netherlands	1	55
Russian Federation	5	786
Slovak Republic	3	909
Spain	2	621
Sweden	3	1210
Ukraine	4	3515
United Kingdom	26	3301
United States of America	28	9764
Total	**124**	**37788**

There are a number of countries where a nuclear decommissioning industry has developed separately from the nuclear industry itself and where there is a good coexistence between operating nuclear installations and those undergoing decommissioning. As an example, Figure 2 shows an overview of NPPs in operation and undergoing decommissioning. The number of operating blocks and the number of blocks under decommissioning already equal each other.

Figure 2. Nuclear power plants in Germany in operation and undergoing decommissioning (Thierfeldt 2009).

Execution of decommissioning projects

Decommissioning projects are often implemented and executed using a phased approach. This means that the entire decommissioning project is structured into phases for which a license, permit or consent of the competent authority would be needed. The advantage of such an approach is that the next phase can be planned and the associated licensing procedure can be carried out while the previous phases are still in execution. Each phase can take advantage of experience gained in the previous phases. Figure 3 shows an example for decommissioning phases and their approximate duration (each phase divided into planning / licensing / implementation), while Figure 4 shows the corresponding illustration of the implementation of these phases for a boiling water reactor.

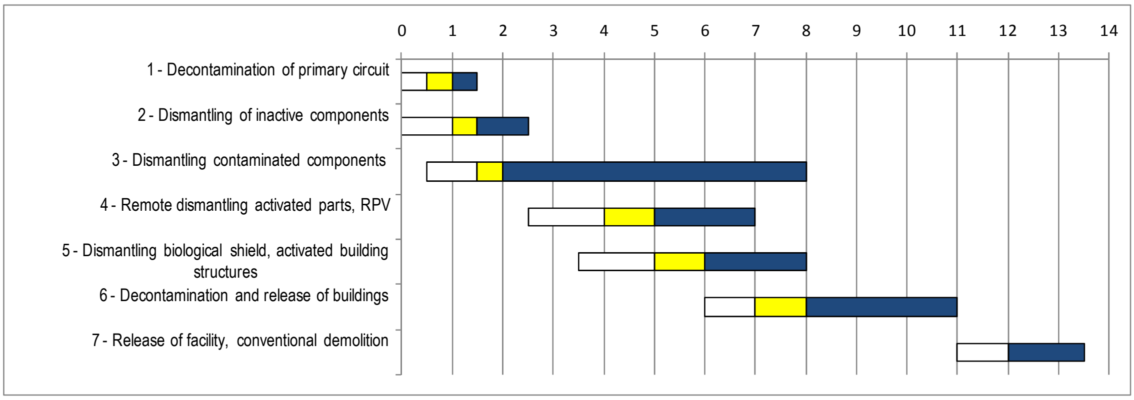

Figure 3. Example for decommissioning phases and their approximate duration for a large NPP.

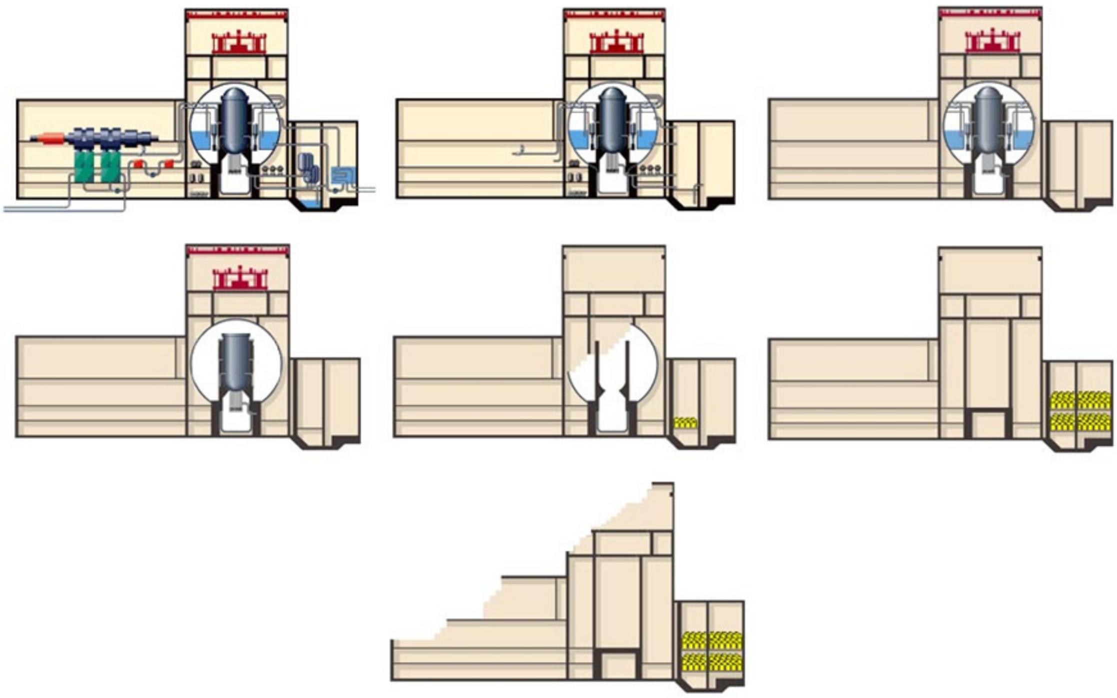

Figure 4. Illustration of decommissioning phases for a boiling water reactor (E.ON, 2008).

Decommissioning Techniques

For all the decommissioning strategies, the dismantling of external, unusable support systems and equipment will generally be performed using standard techniques. Much of this will be uncontaminated material that can be released from regulatory control and therefore needs to be segregated from contaminated material (see chapters on radioactive waste management and clearance).

There are differences in dismantling techniques between early and deferred dismantling. Higher dose rates resulting from high activation or contamination levels may require remote techniques for early dismantling of some facilities. High dose rates from activated components can generally be compensated by using additional shielding, e.g.

by working under water in the reactor spent fuel pool, and/or remotely operated techniques. For deferred dismantling, it is possible that some of the waste will have decayed to low levels and special techniques may not be needed during the dismantling phase. This will depend on the initial level of contamination of the specific structures, equipment or facilities to be decommissioned. It is desirable to plan for deferral periods if this will allow decay to low levels of contamination to simplify decommissioning. - Similar considerations apply to the entombment option: early preparation for entombment will require dealing with material with higher dose rates, while establishing entombment towards the end of a long deferral period will allow use of simpler techniques to prepare the facility as a near surface disposal site.

It can be reasonably assumed that most of the existing facility support equipment (e.g. lifting devices and bridges, utilities, water cleaning and filtration facilities, ventilation, sanitary facilities) will not be available after a prolonged deferred dismantling period without some type of refurbishment or replacement. This means that some advantages can be realised by selecting the immediate dismantling option in preference to deferred dismantling.

Decommissioning of facilities can be carried out with existing technology. There is no significant advantage in waiting for further progress in technology development, but in some particular cases (e.g. treatment/conditioning of special material such as graphite and the remote controlled tools needed for dismantling the reactor internals) some improvements may be expected and this could probably favour deferred dismantling in these particular cases. Nevertheless, there currently exists remote handling technology that allows the performance of this type of activity. (IAEA 2007).

Decommissioning techniques are generally divided into thermal techniques, mechanical techniques and other types. Dismantling and cutting techniques are available for metals, concrete, and all other purposes like cables, insulation etc. There is no single technique which could be used for all purposes. Thermal cutting techniques encompass various principles of heating and melting the material, like arc processes, in particular plasma arc processes, gas processes, laser beam processes, electro discharge machining and other processes, especially combinations of these techniques. An example is shown in Figure 5. Mechanical cutting techniques encompass grinding, various sawing techniques, shears, nibblers, milling cutters, diamond saws and diamond wire cutting etc., but also microwaves for concrete and hydraulic techniques. Examples are shown in Figure 6. A wider overview of decommissioning techniques can be found in (IAEA, 2008a).

Figure 5. Examples for thermal cutting techniques. Plasma cutting.

Figure 6. Examples for mechanical cutting techniques. Left: mechanical shears (BR3, Belgium), Right: diamond wire cutting for large components (HDB, Karlsruhe, Germany).

Materials arising from Decommissioning

This section deals with the various material types arising from decommissioning of nuclear installations. It can only present a short overview of the various material streams and their quantities, and it is necessary to restrict the overview to nuclear power plants, while research reactors and fuel cycle installations present different material compositions and waste streams. The largest part of all material streams will be going to clearance, while only a comparatively small portion will be disposed of as radioactive waste. Because of the completely different management and handling techniques for both parts, management of radioactive waste and clearance of material is dealt with in separate chapters.

Material types from decommissioning

The material types originating from decommissioning of light-water reactors are similar:

- activated and contaminated stainless steel from the reactor pressure vessel and its internals,
- contaminated stainless steel from the circuits, systems and components,
- carbon steel from components and structural elements,
- activated and contaminated concrete from the biological shield and adjacent concrete structures,
- contaminated concrete from building structures,
- electrical installations and cabling,
- insulation material,
- various other materials.

In gas-graphite reactors (GCR), there will also be activated and contaminated graphite.

Material quantities

It is clear that the material quantities that will arise from decommissioning depend on the size of the installation. There is, however, also a dependence on the type of the plant. In nuclear power plants with boiling water reactor (BWR), the secondary circuit and therefore the turbine hall and its components are part of the controlled area. The overall amount of material from BWRs is therefore larger than for a pressurised water reactor (PWR) of similar power rating. Furthermore, the specific type of construction also influences the material quantities. For example, PWRs of Russian type, VVER, have a higher amount of concrete than PWRs of a Western design, as VVERs do not have a full-pressure containment system like Western PWR, but a confinement system which consists of a large number of rooms by which overpressure could be gradually released in the case of an accident. This construction uses a higher amount of concrete per unit power.

Typical quantities of materials are provided in Table 2 for a 900 – 1300 MWe PWR and a 250 MWe GCR. The GCR has a higher mass per unit power than the PWR. In both cases the weight of the materials that are free of contamination and the weight of the building are not shown.

Table 2. Typical quantities of materials from decommissioning for a 900 – 1300 MWe PWR and a 250 MWe GCR (IAEA, 2008b).

Radioactive material	900 – 1300 MWe PWR	250 MWe GCR
activated / irradiated steel	650 Mg	3000 Mg
contaminated steel	3500 Mg	6000 Mg
graphite	-	2500 Mg
activated concrete	300 Mg	600 Mg
contaminated concrete	600 Mg	150 Mg

The weight of the building structures of the controlled area may amount to 200,000 Mg for large PWRs and 400,000 Mg for large BWRs. The structures include a few 10,000 Mg of reinforcement steel and other steel structures that are uncontaminated.

Activity levels and main contaminants

The activity levels, in particular the mass related activity (Bq/g) and the surface related activity (Bq/cm²), vary for the various material types, depending on the origin and history. Activated components of the reactor may reach specific activities of 10^9 Bq/g, the core region of the reactor pressure vessel may reach several 10^6 Bq/g, and activated concrete in the biological shield several 10^3 Bq/g. Surface contamination may range from some 10^5 Bq/cm² on highly contaminated areas down to a few Bq/cm² in less contaminated areas, nearing clearance levels. The main gamma emitting nuclides in LWR materials are Co-60 and Cs-137, causing the main part of the dose rate. Beta emitters like Fe-55, Ni-63 and Ni-59 as well as Sr-90 also contribute to the overall activity but not to the dose rate. Alpha emitters like Am-241 may be present in cases where (smaller or larger) fuel element defects have occurred during operation.

Management of Radioactive Waste

This chapter deals with the management of radioactive waste, while clearance of material is dealt with in the next chapter. First, a short overview of classification schemes for radioactive waste is given. Options for waste management and for decontamination are presented, followed by options for waste disposal.

Classification of radioactive waste

The main categories of decommissioning waste include:

(a) Low level radioactive waste;
(b) Intermediate level radioactive waste;
(c) High level radioactive waste;
(d) Hazardous, non-radioactive waste (chemicals, heavy metals, etc.);
(e) Non-radioactive and non-hazardous waste (cleared material that complies with clearance levels).

Each category of waste has its own unique concerns and specific management requirements.

The classification of radioactive waste is of particular importance when decommissioning nuclear facilities. The existence of guidelines for waste segregation and conditioning, in particular waste acceptance criteria for storage and disposal, can have a significant impact on the planning of decommissioning activities, particularly cost estimates and the selection of decontamination and dismantling activities.

A widely used classification system based on dose rates identifies waste as one of the following: low level waste (LLW), intermediate level waste (ILW), and high level waste (HLW). However, this system serves mainly to support waste handling and storage activities. Other countries may have different classification schemes, like e.g. Germany where the only distinction is made between heat-generating waste (spent fuel, vitrified waste from reprocessing) and waste with negligible heat generation (all other radioactive waste above clearance levels). For the long term management of decommissioning waste, it may, however, be more desirable to employ a classification based on the activity levels and half lives of the radionuclides contained in the waste, as defined by the IAEA. Figure 7 shows the IAEA waste classification scheme together with the German classification scheme.

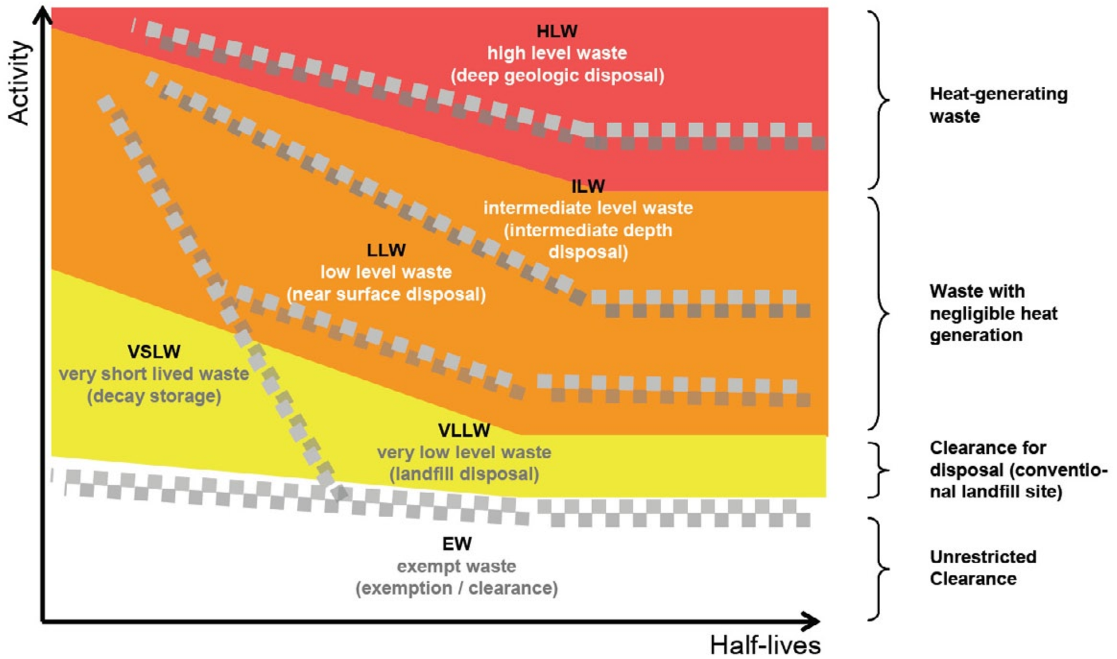

Figure 7. Waste classification according to IAEA and German classification scheme for comparison (German Government 2008).

Waste management options

Waste management starts with thorough characterisation of the material, taking into account the plant history including any incidents that might have led to contamination, activation calculations (for the reactor pressure vessel and the biological shield) and the results of a large number of samples and measurements. Characterisation of the plant is of paramount importance for a smooth implementation of decommissioning. It is, however, not possible to deal with this topic in this paper in any detail.

The basic options for waste management are depicted in Figure 8. In the process on the right starting with radioactive material the first decision is whether any treatment for reduction of activity (segregation of parts with higher and lower activity, decontamination) is technically and economically viable. If not, the material will end as radioactive waste and may need treatment and conditioning before being put into interim storage and eventually into final disposal. If treatment of the initial material is viable, the material has to be checked for compliance with release criteria (clearance levels). It may turn out that the material exceeds these values and has to be treated as radioactive waste, but in general it will be possible to bring the material to one of various release or clearance options (see next chapter). – The process for material for which contamination is only suspected is similar (branch on the left).

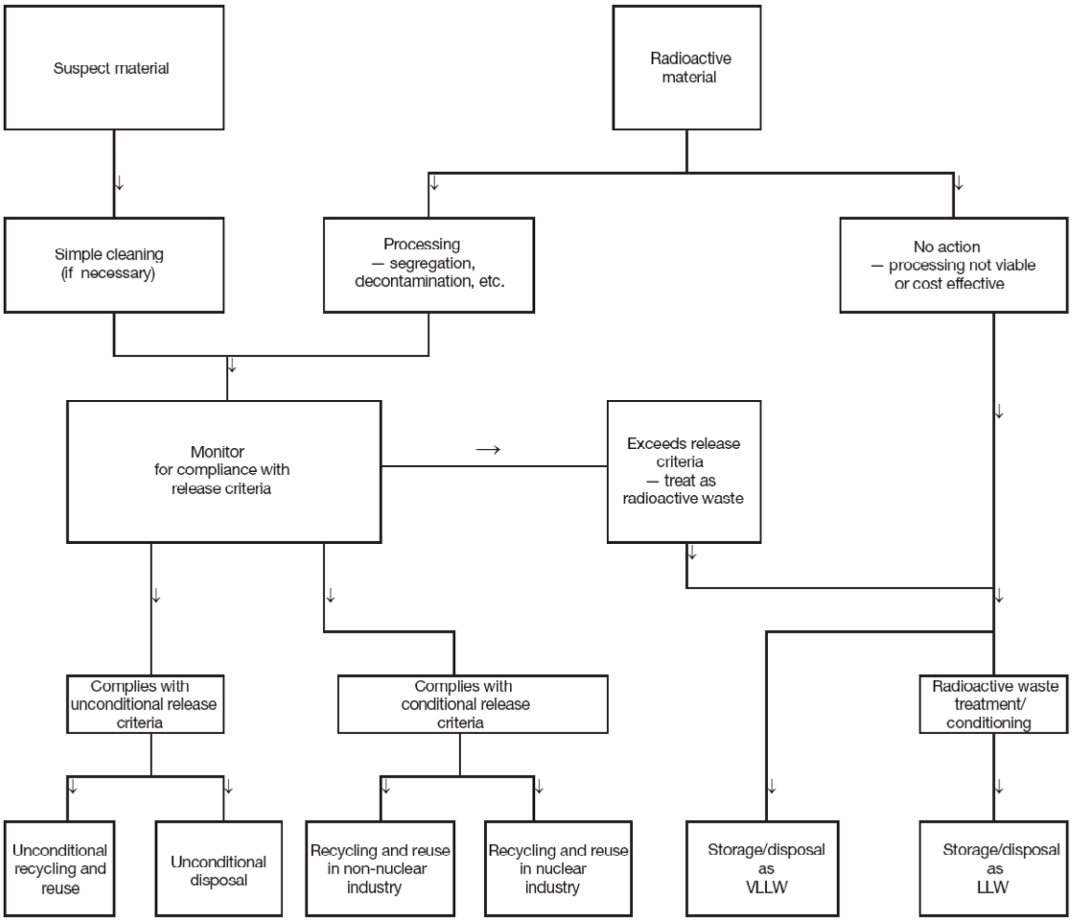

Figure 8. Management and disposition options for the segregation and characterisation of suspect and radioactive material (IAEA, 2008).

Decontamination

Decontamination techniques are based on the removal of surface contamination by mechanical, chemical or other procedures. Activation cannot be removed as it is a property of the entire volume of the material.

In many cases, light decontamination processes may be sufficient, like washing with water or a detergent, high-pressure water jetting or other commonly available processes. However, in many cases a fixed surface layer has to be removed, for which these techniques will not suffice. A widely used procedure for decontamination of metallic surfaces is grit blasting. Figure 9 shows an example of a grit blasting facility in a German nuclear power plant. The mobile facility is placed in the controlled area and forms part of the material management process. Metallic surfaces are monitored prior to and after blasting to verify the effect of the decontamination method and to decide whether the procedure has been effective so that the material complies with clearance levels. If necessary, the procedure has to be repeated. The steel grit is cleaned and reused until it is fully worn and then forms secondary waste (i.e. waste that arises from processes in addition to the primary waste from the material of the nuclear facility itself).

Figure 9. Left: loading of material into the grit blasting caisson. Right: the grit blasting container from the outside (E.ON, 2008).

Table 3. Overview of decontamination techniques for various purposes (IAEA, 2001).

OBJECTIVE/TASK	COMPONENT/SURFACE TO BE DECONTAMINATED	METHOD EMPLOYED
Decontamination before dismantling Reduction of occupational exposure	Pipeline system	Chemical Mechanical
	Pool/tank	Hydro-jet Blast Strippable coating
Decontamination after dismantling Recycle of contaminated metal Reduction of radioactive waste	Pipes, components	Electropolishing Chemical immersion Blast Ultrasonic wave Gel
Decontamination of building Unconditional release of building Reduction of radioactive concrete waste	Concrete surface	Mechanical: • Scabbler • Shaver • Drill and spalling • Steel grit blast Thermal stress: • Microwave irradiation • Flame scarfing

There are numerous other decontamination techniques like chemical baths using acids or bases in combination with electrolytic processes, called electropolishing. By removing a thin surface layer, the contamination is also removed. A rather comprehensive overview is given in Table 3.

Treatment and conditioning techniques

For management of RW from decommissioning, standard technologies can be used which have been applied already during the operational phase. Usually, the technology must be improved and upgraded to be able to manage the decommissioning RW, mainly because of the much larger quantities. The capacity of the RW management technology must be adapted accordingly. In addition, fragmentation and decontamination of solid RW and new measurement tasks usually requires purchasing of special technological equipment. (Thierfeldt, 2006)

The different NPP types produced different RW either from operation or decommissioning. The NPPs with WWER reactors are usually equipped with tanks with big capacities for storage of liquid or semi liquid RW from NPP operations (concentrates, sludges, ion exchangers). This RW must be treated before the decommissioning activities star. Treatment of such kinds of RW could be difficult from the view of meeting the acceptance criteria for disposal and a special methods and equipment must be developed.

There is a wide range of waste treatment and conditioning techniques of which only a few can be presented in the following.

- Solidification refers to a range of processes and which additives are added to a given batch of waste. The waste is then converted to a single, solid form. Before solidification, the waste may take a variety of forms, liquid, slurry, sludge or dried solid particles.
- Incineration may be used if the waste is burnable and does not contain significant amounts of hazardous substances for which the incineration plant is not suitable. The most important property of incineration is its large volume reduction.
- Melting of metals in a specialised licensed facility will transform surface contamination into volume activity, a kind of decontamination process, and will also result in size reduction. The molten metal can be poured into containers with other radioactive waste thus making effective use of any unused space.
- Encapsulation may be used for small, loose materials, e.g. by fixing the material with cement, polyethylene or other material, depending on the acceptance criteria of the interim storage facility and the repository.
- Supercompaction may be used as an effective process for volume reduction of metallic or other compressible waste. Figure 10 shows a volume reduction facility that can be set up in the controlled area of nuclear installations, forming part of the material management process.

Figure 10. Volume reduction of 200 l drums (FAKIR supercompaction facility, GNS).

Disposal options

The large field of disposal options that exist in various countries can only be briefly touched upon in this paper. Various countries like UK, France or USA prefer near-surface disposal of most wastes resulting from decommissioning. The waste is put into waste containers that are filled with concrete and is then emplaced into trenches lined with concrete which are finally covered in concrete, creating a monolithic waste structure. This waste form will remain stable until the activity will have decayed to levels that are below regulatory concern (several decades or a few hundred years). The amount of long-lived nuclides is usually limited in such repositories.

Other countries like Germany prefer deep geological disposal also for decommissioning waste. The characteristics of the waste containers and the mode of emplacement have been derived from operational and safety analysis of the repository. There are usually fewer restrictions on the activity content of the waste, as the isolation of the waste inside a deep geological repository is superior to a near-surface disposal site.

Waste management and selection of decommissioning strategy

Different aspects of waste generation and waste management can have an impact on the selection of a decommissioning strategy. Among the most important aspects are: the overall national waste management strategy, the amount of waste, the types and categories of waste (both radioactive and non-radioactive) and the facilities needed to process, handle, store and dispose of the waste. The absence of a waste management policy for any of these waste categories will introduce uncertainties in the decommissioning strategy selection process. If no disposal route exists for a particular waste category, the only option may be to store the waste on-site in regulated storage facilities.

Lack of a disposal facility is in itself insufficient reason for not performing immediate dismantling. The waste can be placed into an interim storage facility that will also require decommissioning eventually, once a final disposal scheme for the waste is decided upon. If the nuclear programme within a country is very limited and the type of facility to be decommissioned is amenable to entombment, then this may be the preferred option. As stated previously, an entombed facility has to be considered as a near surface waste disposal site and needs to meet the regulatory requirements for such a facility. This means, for example, that no or only limited amounts of long lived radionuclides are allowed in an entombed facility. If the country has not prepared the necessary infrastructure for a low level waste disposal facility, then entombment may not be feasible. However, the facility could be placed into a short deferred dismantling mode until such requirements can be put into place. (IAEA, 2007)

A country specific analysis of the influence of waste management techniques on decommissioning strategy selection among EU member states yielded three basic types of approaches (Thierfeldt, 2007):

- High influence on strategy selection: dismantling is postponed until a viable waste route becomes available, either in a prolonged form of early decommissioning which is drawn out over a suitably period of time or by choosing the deferred decommissioning option. Examples are NPP Barsebäck (Sweden), the four Italian nuclear power plants, in particular the NPP Latina and the NPP Vandellós 1 (Spain).
- Medium influence on strategy selection: dismantling is commenced although no repository is available. However, in order to manage the radioactive waste streams from decommissioning and to provide a storage even after dismantling of the nuclear installation will have been completed, an interim storage facility is created (e.g. in an existing building at the site which is converted into a storage facility or in a building newly constructed for this purpose). In some cases, the characteristics of this interim radwaste storage facility may have influence on the waste management options and the conditioning techniques, e.g. on the use of waste containers because of limited shielding provided by the waste building, on the mass of waste packages because of limited lifting equipment capacity etc. Examples are the NPPs Würgassen and Stade (Germany) and NPPs in the UK.
- Low or negligible influence on strategy selection: disposal routes for radioactive waste are available, either in the form of an existing repository or of centralised radwaste storage facilities, and consequently waste acceptance criteria are also established. In such a case, dismantling can commence without consideration to the creation of new waste disposal facilities or disposal routes. Examples are mainly decommissioning projects from countries with existing repositories like France

Clearance of Material

Clearance is of central importance for the material management in decommissioning projects. The topic is so broad that the analysis in this paper necessarily has to be very short. The vast majority of material from decommissioning of nuclear installations is either free of contamination (especially the large volumes of concrete from building demolition) or can be brought to sufficiently low levels by decontamination. If it can be demonstrated that the material, the buildings or the site of the facility complies with clearance levels, it can be released from regulatory control.

Clearance levels

Clearance is part of the EURATOM Basic Safety Standards (European Communities, 1996). Therefore, a number of countries have implemented clearance regulations with various degrees of complexity and using different clearance levels. The European Commission has also given guidance on clearance levels and the implementation of clearance for metal scrap, buildings, building rubble and unconditional clearance (European Commission, 1998, 2000a, 2000b). The IAEA has issued recommendations on unconditional clearance as well (IAEA, 2004). Table 4 shows a small comparison of values that are used or have been recommended for use in various countries. This table shows that there for some countries there are large differences for certain values which derive from the radiological models used. The values in many international studies that have been explicitly derived for unconditional clearance, however, are generally close together.

Table 4. Comparison of clearance levels for unconditional clearance for key radionuclides from studies and the regulatory framework of various countries and from international recommendations (values rounded to 1 significant digit; "-" means not analysed or no value given) (Thierfeldt, 2005).

Country/ Organis.	Regulatory framework/ study / recommendation	Clearance level in [Bq/g] for				
		Co 60	Cs 137	Sr 90	Am 241	H 3
Germany	Radiation Protection Ordinance Ann. III Tab. 1 Col. 5	0.1	0.5	2	0.05	1,000
Netherlands	report Timmermans et al. (recycling of steel)	100	100	10,000	100	1E7
Sweden	report Elert et al. (minimum of all clearance options for metals)	0.2	8	4	70	-
UK	report DETR	0.04	0.2	3	0.6	600
UK	Radioactive Substances Act	0.4	0.4	0.4	0.4	0.4
USA	report NUREG 1640	0.04	0.04	1	-	-
EU	RP 122 part 1	0.1	1	1	0.1	100
IAEA	Safety Guide RS-G-1.7	0.1	0.1	1	0.1	100
OECD/NEA	report of Task Group (additional surface specific CL apply)	0.2	0.7	94	50	-

A study on EU member states (Thierfeldt, 2006) showed that the existence of regulations for clearance will not necessarily depend on the size of the nuclear programme. Instead, most countries have implemented clearance regulations as soon as there is a substantial amount of material which need not be treated as radioactive waste and which has to be dealt with. A number of countries have followed one or more of the recommendations issued by the European Commission when implementing clearance in national regulations. While clearance regulations have been introduced in all countries except France, substantial differences pertain to the detailed design of these regulations, mainly the clearance options and the clearance levels which are in place. The comparison also reveals that the existence of a repository for radioactive waste is obviously not related to the implementation of clearance regulations in a particular country.

Measurement techniques for clearance

A large number of measurement techniques suitable for various clearance options is available which cannot be enumerated and outlined here. In cases where hard to measure radionuclides (alpha emitters, pure beta emitters etc.) form part of the nuclide vector and have to be taken into account, correlation with easy-to-measure nuclides are used. Usually a correlation between hard gamma emitters which have a high abundance in the nuclide vectors (e.g. ^{60}Co, ^{137}Cs), so-called key nuclides, is used. Correlation factors to other nuclides like ^{90}Sr, ^{241}Am etc. are determined from radiological measurements. The overall radionuclide contents of material can then be calculated from measuring the activity of the key nuclides and calculate the activities of the other nuclides via the correlation factors.

The applicability of direct measurements of gamma emitting nuclides as well as of the correlation method depends to a large extent on the percentage that these gamma emitting nuclides have in the nuclide vector. Because of the radioactive decay and because of the relatively short half-life of ^{60}Co (5.3 a) and the long half-lives of hard-to-measure nuclides (30 a for ^{90}Sr, 432 a for ^{241}Am), this percentage is continuously decreasing. Some installations, especially those with a high abundance of alpha emitting nuclides in the contamination, have taken this fact into account when determining the duration of the safe enclosure period. (Thierfeldt, 2006)

Examples for the three most important measurement techniques for material from NPPs are shown in the following figures. Figure 11 shows measurements with surface contamination monitors on metallic components, a technique which is indispensable for metallic material. These instruments are sensitive for beta and/or alpha radiation, so that the nuclide vector has to be known. Larger quantities (several 100 kg up to 1 Mg) of metallic material and bulk material like building rubble can be measured in bulk monitors of various sizes as shown in Figure 12. These facilities are sensitive for gross gamma radiation, so that the nuclide vector has to be known. Finally, building surface can be monitored using in-situ gamma spectrometry as shown in Figure 13. Here, the gamma emitting nuclides are measured individually.

Figure 11. Measurements with surface contamination monitors on metallic components (EON, 2008).

Figure 12. Measurements with a bulk monitor for large quantities of material (up to 1 Mg) (RADOS).

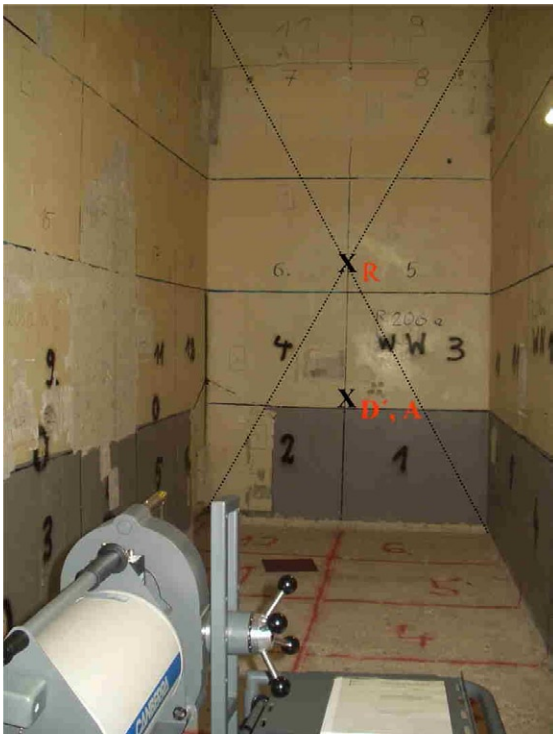

Figure 13. Measurement of a room with in situ gamma spectrometry (Naber, 2007).

In summary, the implementation of clearance or of an equivalent way of dealing with material with extremely low activity is a necessary prerequisite for carrying out active decommissioning. The recommendations on clearance issued by the European Commission (RP 89, 113, 122/I) or the use of similar clearance levels have proven to be viable. International harmonisation of clearance levels between European countries is a goal that has often been called for, but which has been achieved already to a large extent. The recommendations on clearance issued by the European Commission have played an important role as guidance. It is obviously not necessary to have exact harmonisation of the values of all clearance levels for all radionuclides, but to use clearance levels in the same order of magnitude for the most important radionuclides

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Non-ionising radiation

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Abstract
Technological development in modern societies cause an ever increasing exposure of people to non-ionising radiation in all situations of life. Examples include new technologies for diagnosis and treatment in medicine, wireless communication technologies at home and in the office, and high power devices for material processing in industry. The health implications of such exposures have been investigated quite intensively, but remain an issue of public concern. The relevant mechanisms and health effects are different throughout the non-ionising electromagnetic spectrum. Nerve excitation and tissue heating are well established examples. Cancer induction especially in children or from very long chronic exposure is still under investigation. The International Commission on Non-Ionizing Radiation Protection has issued guidance on limiting exposure to static, low, and high frequency electric and magnetic fields, that are regularly reviewed.

Introduction
This refresher course addresses the current scientific evidence of health related risks of exposures to static and low frequency (LF) electric and magnetic fields and high frequency (or radiofrequency (RF)) electromagnetic fields. It illuminates different aspects relevant for setting guidelines on limiting exposure. The following overview is based primarily on reviews conducted by the International Commission on Non-Ionizing Radiation Protection (ICNIRP) (ICNIRP 2003, ICNIRP 2009a) and the World Health Organization (WHO 2006, WHO 2007).

Sources
Exposure to non-ionising radiation can result from both, natural and man made sources. Static electric fields occur naturally in the atmosphere. Beneath thunderclouds values of up to 3 kV m^{-1} can occur. High electric fields from man made sources are the result of either charge separation as a result of friction or the use of high voltages. Common examples are direct current (DC) power transmission that can produce static electric fields of up to 20 kV m^{-1}, rail systems using DC with fields of up to 300 V m^{-1} inside the train, and video display units that can create electric fields of around 10 - 20 kV m^{-1} at a distance of 30 cm. Local fields of up to 500 kV m^{-1} may be generated while walking on non conducting carpets.

The static geomagnetic field varies over the Earth's surface between about 35 - 70 µT. Magnetic flux densities of up to 2 mT have been reported inside electric trains and in developmental magnetic levitation systems. Workers are exposed to larger fields of up to around 60 mT from electrolytic reduction processes of alumina, and electric arc welding produces around 5 mT at 1 cm from the welding cables. Much larger magnetic fields are used in medical diagnosis. The static magnetic field of MRI scanners in routine clinical systems range from 0.2 - 3 T. In research applications, higher magnetic fields up to 9.4 T are used for whole body patient scanning. In terms of exposure, the magnetic flux density at the operator's console is typically about 0.5 mT, and higher. Strong fields are also produced in high-energy technologies such as thermonuclear reactors, magneto hydrodynamic systems and superconducting generators.

The dominant exposure to LF electric and magnetic fields is normally that associated with the generation, transmission and use of electricity at the power frequencies of 50 or 60 Hz. Maximum field strengths of up to several kV m^{-1} might be encountered at ground level below power lines operating at 100 kV or greater. Common sources are the various electrical appliances and tools used in industry and at home. The power frequency electric fields in conventional homes supplied with electricity range up to several hundred volts per metre. Power frequency magnetic fields in homes arise from the flow of current in electrical appliances and circuits both inside and outside the home, and from currents in conducting pipe work and the ground. The strongest LF magnetic fields in homes are usually those generated by electrical appliances. Transmission lines can produce maximum magnetic flux densities of up to a few tens of µT during peak demand, however mean levels are usually no more than a few µT. The magnetic fields encountered in electrified rail systems vary considerably because of the large variety of possible arrangements of power supply and traction. Operators in the vicinity of induction furnaces and equipment heaters may receive some of the highest exposure levels found in industry. Maximum flux densities of 1 - 60 mT have been reported. Incidental exposures to LF electric and magnetic fields can be incurred in a variety of occupational settings. Examples for typically magnetic flux densities that can occur are a few µT from underfloor heating, a few mT from tape erasers, a few mT from motorised equipment, several hundred µT from arc furnaces, and several mT from resistance welding.

At lower radio frequencies, below 30 MHz, the background electromagnetic radiation is mainly due to lightning discharges during thunderstorms. At high radio frequencies, above 30 MHz, the natural EM-fields originate from very broadband blackbody radiation from the warm earth and from extraterrestrial processes, mainly from the sun and the extraterrestrial microwave background radiation from the whole sky. The power density of the radiation component emitted by the warm surface of the ground at 300 K temperature (27 °C) is a few $mW \cdot m^{-2}$.

Man-made high frequency sources produce local field levels many orders of magnitude above the natural background. The antennas of broadcast stations are the most powerful continuous sources of high frequency electromagnetic energy intentionally radiated into free space. The cellular mobile telephone industry has undergone rapid growth resulting in a wide use of wireless communication devices in all parts of modern society today. The mobile phone networks consist of a system of adjoining 'cells', each with its own base station that sends and receives radio signals.

The base stations have power outputs ranging from tens of Watt in macrocells down to a fraction of a Watt in picocells. The exposure levels from these mobile telecommunications base stations are very low. Strong high frequency fields are encountered in medicine. Here high frequency fields are used for example for tissue heating (hyperthermia, diathermia), ablation, telemetry, and diagnostic purposes (MRI). High power electromagnetic sources are also used for material processing in industrial and domestic environments, for electronic article surveillance, for identification, and various RADAR applications. Microwave fields are also considered as carrier for wireless transport of electrical energy.

Mechanisms and effects

The three established physical mechanisms through which static magnetic fields interact with living matter are magnetic induction, magneto-mechanical, and electronic interactions. Magnetic induction originates through electrodynamic interactions with moving electrolytes. Static fields exert Lorentz forces on moving ionic charge carriers and thereby give rise to induced electric fields and currents. In accordance with Faraday's law of induction electric fields and currents within the body may also result from exposure to time varying magnetic fields, or from movement in a static magnetic field. The two types of mechanical effects that a static magnetic field can exert on biological objects are magneto-orientation and magneto-mechanical translation. In a static field, paramagnetic molecules experience a torque that orients them in a configuration that minimizes their free energy within the field. In the presence of gradients, static magnetic fields produce a net translational force on both diamagnetic and paramagnetic materials. Magnetic fields affect the recombination of radical pairs involved in an intermediate state of certain metabolic reactions.

Concerning magnetic field effects, different levels of biological organization at a cellular level have been investigated, including cell-free systems and various cell models using both bacteria and mammalian cells. Only a few in vitro studies on genotoxicity have been performed. Overall there is little convincing evidence from cellular and cell-free models for biologically harmful effects of exposure to static magnetic fields with flux densities up to several T. A large number of animal studies on the effects of static magnetic fields have been conducted (Saunders 2005). The most consistent responses seen in behavioural studies suggest that the movement of laboratory rodents in magnetic fields of about 4 T and higher may lead to aversive responses and conditioned avoidance. A well-established effect of exposure of animals to static magnetic fields greater than about 0.1 T is the induction of blood flow potentials in and around the heart and other major vessels of the central circulatory system. Exposures to static fields of up to 1 T have not been demonstrated to have an effect on foetal growth or postnatal development in mice. It is not possible to draw any conclusions from animal studies on possible genotoxic or carcinogenic effects of static magnetic fields. Detailed physiological studies in humans evaluating various physiological parameters including body (sublingual) temperature, respiratory rate, pulse rate, blood pressure, and finger oxygenation levels have not shown any pronounced effects of exposure to magnetic fields up to 8 T (Chakeres et al. 2003). Behavioural studies carried out on subjects situated in the proximity of MR systems with exposures up to 7 T have suggested that there may be a transient negative influence of exposure on eye-hand coordination and visual contrast sensitivity associated with head movement in the field (de Vocht et al. 2006, 2007a, 2007b). Overall, current information does not indicate any serious health effects resulting from the acute exposure of stationary humans to static magnetic fields up to 8 T. It should be noted, however, that such exposures can lead to potentially unpleasant sensory effects such as vertigo and transient decrements in the performance of some behavioural tasks during head or body movement. The few available epidemiological studies have mostly been conducted on workers exposed to moderate static magnetic fields of up to several tens of mT either working in aluminium smelters or chloralkali plants or as welders. Overall, the few available epidemiological studies on static magnetic fields have methodological limitations and leave a number of issues unresolved concerning the possibility of risk of cancer or other outcomes from long-term exposure. These studies do not indicate strong effects of static magnetic field exposure of the level of tens of mT on the various health outcomes studied, but they would not be able to detect small to moderate effects.

Time varying electric fields external to the body induce a surface charge on the body; this results in induced currents in the body, the distribution of which depends on exposure conditions, on the size and shape of the body, and on the body's position in the field. Surface charges may cause well-defined biological responses, ranging from perception to annoyance. The physical interaction of time-varying magnetic fields with the human body results in induced electric fields and circulating electric currents. Induced electric fields inside the body can be calculated by computational methods using anatomically and electrically realistic models of the body, which have a high degree of anatomical resolution. The responsiveness of electrically excitable nerve and muscle tissue to electric stimuli including those induced by exposure to LF electric and magnetic fields has been well established for many years. Myelinated nerve fibres of the peripheral nervous system have been stimulated by pulsed electric fields induced during volunteer exposure to the switched gradient magnetic fields of magnetic resonance (MR) systems. The most robustly established effect of electric fields below the threshold for direct nerve or muscle excitation is the induction of magnetic phosphenes, a perception of faint flickering light in the periphery of the visual field, in the retinas of volunteers exposed to LF magnetic fields. The retina as part of the CNS is regarded as an appropriate, albeit conservative model for induced electric field effects on CNS neuronal circuitry in general.

The experimental data available so far do not indicate that LF electric and/or magnetic fields affect the neuroendocrine system in a way that would have an adverse impact on human health. It has been hypothesized that exposure to LF fields is associated with several neurodegenerative diseases. For Parkinson's disease and multiple sclerosis the number of studies has been small and there is no evidence for an association between LF exposure and these diseases. For Alzheimer's disease and amyotrophic lateral sclerosis (ALS) more studies have been published. Overall, the evidence for the association between LF exposure and Alzheimer´s disease and ALS is inconclusive. Experimental studies of both short-term and long-term exposure indicate that hazardous cardiovascular effects associated with LF fields are unlikely to occur at exposure levels commonly encountered environmentally or occupationally.

Epidemiological studies have not shown an association between human adverse reproductive outcomes and maternal or paternal exposure to LF fields. There is limited evidence for an increased risk of miscarriage associated with maternal magnetic field exposure, but overall, the evidence for an association between LF fields and developmental and reproductive effects is very weak. A considerable number of epidemiological studies, carried out particularly during the 1980s and 90s, indicated that long term exposure to weak LF magnetic fields, orders of magnitude below the recommended limits, might be associated with childhood leukaemia. In contrast to the epidemiological evidence of an association between childhood leukaemia and prolonged exposure to power frequency magnetic fields, the animal cancer data, particularly those from large-scale lifetime studies, are almost universally negative.

Exposure to electromagnetic fields at frequencies above about 100 kHz can lead to significant absorption of energy and temperature increases. In general, exposure to a uniform (plane-wave) electromagnetic field results in a highly non-uniform deposition and distribution of energy within the body, which must be assessed by dosimetric measurement and calculation. As regards absorption of energy by the human body, electromagnetic fields can be divided into four ranges:

- frequencies from about 100 kHz to less than about 20 MHz, at which absorption in the trunk decreases rapidly with decreasing frequency, and significant absorption may occur in the neck and legs;
- frequencies in the range from about 20 MHz to 300 MHz, at which relatively high absorption can occur in the whole body, and to even higher values if partial body (e.g., head) resonances are considered;
- frequencies in the range from about 300 MHz to several GHz, at which significant local, non uniform absorption occurs; and
- frequencies above about 10 GHz, at which energy absorption occurs primarily at the body surface.

There are several theoretical hypotheses describing potential non-thermal mechanisms for biological effects from low level exposure. Some have been tested experimentally, but so far there has been no compelling evidence that they might plausibly account for any significant health effect.

Over the last 30 years there have been many in vitro studies on potential cellular effects of high frequency electromagnetic exposure. These studies gave insight into the basic mechanisms by which effects might be induced in more complex animal or human organisms. Interpretation of the experimental results is, however, limited by anomalous cell behaviour generated by the culture conditions and other factors which limit the extrapolation to humans. The studies conducted so far have not provided consistent evidence of biological effects under non-thermal electromagnetic exposure conditions. More recently, studies have been carried out using powerful high-throughput screening techniques capable of examining changes in the expression of very large numbers of genes. Such studies often showed a limited number of alterations where some genes were up- and others down-regulated. These advances in molecular studies are promising, but not yet decisive in risk evaluation.

Animal studies further support the conclusion that the most consistent and reproducible responses to acute high frequency exposure result from the induced heating. These studies established that, in general, an increase in body temperature elicits several cardiovascular changes including increased blood flow to the skin, increasing skin thermal conductance, and increased cardiac output, primarily due to an increase in heart rate, in order to maintain arterial pressure within the normal range. Deficits in learned behaviours, particularly the disruption of ongoing operant behaviours, occur mainly when core temperatures are increased by about 1°C or more. Similar rises in body temperature also result in significantly enhanced plasma corticosterone or cortisol levels in rodents and primates and transient changes in immune function and hematology, generally consistent with the acute responses to non-specific stressors. In addition, high frequency radiation can cause increased embryo and foetal losses, increased incidence of foetal malformations and anomalies, reduced foetal weight at term and impair male fertility at exposure levels that are sufficiently high to cause a significant increase in temperature. High frequency induced cataract also remains a well-established thermal effect of exposure in anaesthetised rabbits. However, primates appear less susceptible to cataract induction, and opacities have not been observed following either acute or prolonged exposures. The results of recent carcinogenicity studies in animals are rather consistent and indicate that carcinogenic effects are not likely at specific power absorption rates (SAR) of up to 4 W kg^{-1} averaged over the whole body, even for long-term exposure. To date, there is no consistent evidence of relevant effects at non-thermal exposure levels in animals.

The most consistent effects of acute exposure of human subjects to electromagnetic fields are the thermoregulatory responses to the induced heating. Volunteer studies indicate that exposed subjects can accommodate whole body heat loads of up to several 4 W kg^{-1} with minimal changes in core temperature. Increased skin blood flow and profuse localized sweating minimise skin temperature rises in response to high local peak SAR. Most volunteer studies have investigated the effects of electromagnetic exposures characteristic of mobile phone use, usually to the head, on a number of physiological parameters including brain electrical activity and blood flow, cognition, and more generally on the endocrine and cardiovascular systems. Some evidence suggests that such exposure may affect the spontaneous EEG in volunteers, but results overall are variable and inconsistent. The small changes seen in brain electrical activity and possibly in regional cerebral blood flow may not have any functional significance. Despite the large number of studies concerned with the issue, no consistent effects of electromagnetic fields on cognitive performance have been found so far. There is no robust evidence of any effect of mobile phone type radiation on children or adolescents. A wide range of subjective symptoms including headaches and migraine, fatigue, and skin itches have been attributed to various electromagnetic sources both at home and at work. However, the evidence from double-blind provocation studies suggests that the reported symptoms are not causally related to EMF exposure.

Results of epidemiological studies to date give no consistent or convincing evidence of a causal relation between electromagnetic exposure and any adverse health effect. On the other hand, these studies have too many deficiencies to rule out an association. A key concern across all studies is the quality of assessment of electromagnetic exposure, including the question of whether such exposure was present at all. Another general concern in mobile phone studies is that the time periods that have been examined to date are necessarily short. The implication is that if a longer time period is required for a health effect to occur, the effect could not be detected in these studies. Another gap in the research is children. No study population to date has included children, with the exception of studies of people living near radio and TV antennas.

Recommendations

ICNIRP is currently in the process of reviewing and revising its guidance concerning the limitation of exposure in the whole non-ionising radiation range. Guidelines on static magnetic fields have recently been published, revised guidelines on LF fields (1Hz – 100 kHz) have already undergone broad consultation and guidelines on radiofrequency fields (up to 300 GHz) will be reviewed synchronously to the health risk assessment process of the World Health Organization.

The exposure guidelines developed by ICNIRP are intended to protect against the adverse health effects of NIR exposure. Because adverse consequences of NIR exposure can vary across the entire range from trivial to life threatening, a balanced judgement is required before deciding on exposure guidance. Annoyance or discomfort may not be pathological per se but, if substantiated, can affect the physical and mental well being of a person and the resultant effect is considered to be a potential health hazard. ICNIRP seeks to define what is meant by adverse effects in its specific scientific reviews and guidelines. Examples are provided in Table 1.

Table 1. Relevant mechanisms of interaction, adverse effects, and biologically effective physical quantities used in different parts of the electromagnetic field spectrum.

field type	mechanism, interaction	adverse effect	effective quantity
static electric field	surface electric charge	annoyance of surface effects, shock	external electric field strength
static magnetic field	magnetic induction	annoyance of sensory effects, cardiovascular and CNS effects	external magnetic flux density
LF electric field	surface electric charge; induction of electric fields	annoyance of surface effects, shock; stimulation of PNS and CNS	external electric field strength; internal electric field strength
LF magnetic fields	induction of electric fields	stimulation of PNS and CNS	internal electric field strength
RF electromagnetic field	induction of electric fields; absorption of energy within the body; surface absorption of energy; thermo-acoustic wave propagation.	excessive heating, electric shock and burn; excessive surface heating; annoyance from microwave hearing effect	Specific energy absorption rate; Power density; Specific energy absorption

In general, separate guidance is given for occupational exposures and exposure of the general public. The limits for occupational exposure in the ICNIRP guidelines apply to those individuals who are exposed as a result of performing their regular or assigned job activities.

For static magnetic fields ICNIRP recommended that occupational exposure of the head and trunk should not exceed a spatial peak magnetic flux density of 2 T, to prevent vertigo, nausea and other potentially annoying sensations (ICNIRP 2009b). For work applications for which exposures above 2 T are deemed necessary, exposure up to 8 T can be permitted if the environment is controlled and appropriate work practices are implemented to control movement-induced effects. When restricted to the limbs, maximum exposures of up to 8 T are acceptable. Exposure of the general public should not exceed 400 mT (any part of the body). However, because of potential indirect adverse effects practical policies need to be implemented to prevent inadvertent harmful exposure of people with implanted electronic medical devices and implants containing ferromagnetic materials, and injuries due to flying ferromagnetic objects, and these considerations can lead to much lower restriction levels. The limits recommended for occupational and general public exposures to static magnetic fields are summarized in Table 2.

Table 2. Limits of exposure[a] to static magnetic fields.

Exposure characteristics	Magnetic flux density
Occupational	
Exposure of head and of trunk	2 T
Exposure of limbs	8 T
General public	
Exposure of any part of the body	400 mT

[a] ICNIRP recommends that these limits should be viewed operationally as spatial peak exposure limits.

Concerning time varying electric and magnetic fields, ICNIRP (ICNIRP 1998) recommended two classes of guidance: Basic restrictions that are based directly on established health effects. Depending upon the frequency of the field, the physical quantities used to specify these restrictions are internal electric field strength (E), current density (J), specific energy absorption rate (SAR), and power density (S). Reference levels are provided for practical exposure assessment purposes to determine whether the basic restrictions are likely to be exceeded. The derived quantities are electric field strength (E), magnetic field strength (H), magnetic flux density (B), power density (S), and currents flowing through the limbs (I).

For exposure to LF electric and magnetic fields, basic restrictions are provided on current density to prevent effects on nervous system functions. In the frequency range from a few Hz to 1 kHz, for levels of induced current density above 100 mA m^{-2}, the thresholds for acute changes in central nervous system excitability and other acute effects such as reversal of the visually evoked potential are exceeded. In view of safety considerations, occupational exposure should be limited to fields that induce current densities less than 10 mA m^{-2}. For the general public an additional factor of 5 is applied. ICNIRP is currently revising its guidance on limiting exposure to LF electric and magnetic fields. Different modifications with respect to the 1998 guidelines are under consideration. Protection from effects on central and peripheral nervous system are discussed. Field effects caused in the retina are considered a model for effects in the brain. With regard to the basic restriction, the induced electric field seems to be closer related to the biological effects and thus might replace the induced current density in future recommendations. In deriving reference levels, the most recent advances in numerical dosimetry will have to be considered.

For exposure to high frequency electromagnetic fields, the 1998 guidelines have been reconfirmed recently (ICNIRP 2009c). It is the opinion of ICNIRP that the scientific literature published since the 1998 guidelines has provided no evidence of any adverse effects below the basic restrictions and does not necessitate an immediate revision of its guidance on limiting exposure to high frequency electromagnetic fields. The biological basis of such guidance remains the avoidance of adverse effects such as "work stoppage" caused by mild whole body heat stress and/or tissue damage caused by excessive localized heating. there has been considerable advance in dosimetric investigations in terms of precision and resolution (Lin 2007). A special concern was raised with regard to numerical computations using anatomical models of human bodies which might influence the derivation of reference levels from the basic restrictions. Some published studies showed that in the frequency ranges of body resonance (~100 MHz) and from 1 to 4 GHz for bodies shorter than 1.3 m in height (corresponding approximately to children aged 8 y or younger) at the recommended reference level the induced SARs could be up to 40% higher than the current basic restriction under worst-case conditions. However, this is considered negligible compared with the large reduction factor of 50 (5,000%) for the general public. The basic restrictions recommended for occupational and general public exposures to high frequency electromagnetic fields are summarized in Table 3.

Table 3. Limits of exposure to high frequency electromagnetic fields.

Exposure characteristics	Frequency range	Current density mAm^{-2}	Whole body average SAR W kg^{-1}	Localised SAR (head/trunk) W kg^{-1}	Localised SAR (limbs) W kg^{-1}
Occupational					
	100 kHz-10 MHz	f/100	0,4	10	20
	10 MHz-10 GHz		0,4	10	20
General public					
	100 kHz-10 MHz	f/500	0,08	2	4
	10 MHz-10 GHz		0,08	2	4

Basic restrictions for frequencies between 10 and 300 GHz limit power density to 50 W m^{-2} (occupational exposure) and 10 W m^{-2} (general public).

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International Commission on Radiological Protection – recent publications, current initiatives and future work

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International Commission on Radiological Protection (ICRP)

Abstract
In 2007, after almost a decade of development and worldwide, open consultation, the International Commission on Radiological Protection (ICRP) published its latest fundamental recommendations describing the overall system of radiological protection in the 2007 Recommendations of the International Commission on Radiological Protection. These replaced the 1990 Recommendations of ICRP published almost two decades earlier. Since then ICRP has focused primarily on developing publications that support or further elaborate elements of the system of radiological protection as described in the 2007 Recommendations. This paper describes these publications, as well as ongoing and planned future efforts with the same objective.

Introduction
The International Commission on Radiological Protection (ICRP) is an independent, international organization that advances for the public benefit the science of radiological protection, in particular by providing recommendations and guidance on all aspects of protection against ionizing radiation.

ICRP was established in 1928 by the International Society of Radiology (ISR) to respond to growing concerns about the effects of ionizing radiation being observed in the medical community. At the time it was called the International X-ray and Radium Protection Committee, but was restructured to better take account of uses of radiation outside the medical area and given its present name in 1950.

ICRP is a Registered Charity (a "not-for-profit organisation) in the United Kingdom, and has a Scientific Secretariat in Ottawa, Canada.

In preparing its recommendations, ICRP considers the fundamental principles and quantitative bases upon which appropriate radiation protection measures can be established, while leaving to the various national protection bodies the responsibility of formulating the specific advice, codes of practice, or regulations that are best suited to the needs of their individual countries.

ICRP offers its recommendations to regulatory and advisory agencies and provides advice the intended to be of help to management and professional staff with responsibilities for radiological protection. Although ICRP itself has no formal power to impose its recommendations, in fact legislation in most countries adheres closely to ICRP recommendations. In addition, the International Atomic Energy Agency (IAEA) International Basic Safety Standards for Protection against Ionizing Radiation and for the Safety of Radiation Sources (commonly referred to as "the BSS") is based heavily on ICRP recommendations, and the International Labour Organisation (ILO) Convention 115, Radiation Protection Convention, General Observation 1992, refers specifically to the recommendations of ICRP. Effectively, ICRP recommendations form the basis of radiological protection practice, programmes, regulations, and international standards and guidance worldwide.

Originally, ICRP published its recommendations and advice as papers in various scientific journals in the fields of medicine and physics. Since 1959, ICRP has its own series of publications, since 1977 in the shape of a scientific journal, the Annals of the ICRP, which is published Elsevier Science.

ICRP has published well over one hundred publications on all aspects of radiological protection. Most address a particular area within radiological protection, but a handful of publications, the so-called fundamental recommendations, each describe the overall system of radiological protection. The system of radiological protection has been developed by ICRP based on (i) the current understanding of the science of radiation exposures and effects and (ii) value judgements. These value judgements take into account societal expectations, ethics, and experience gained in application of the system.

The 1990 Recommendations of ICRP form the basis of the current IAEA BSS, and are also the foundation of most radiological protection practices, programmes and regulations worldwide. The 2007 Recommendations recently replaced the 1990 Recommendations. They are the result of nearly a decade of development and several major worldwide public consultations. They update, consolidate, and develop additional guidance on the control of exposure from radiation sources issued since 1990, and they reflect a more up-to-date understanding of the science behind radiological protection and evolving societal expectations.

The 2007 Recommendations of ICRP

Although a complete review of the 2007 Recommendations (ICRP *Publication 103*) is beyond the scope of this paper, it is important to recall that all exposures are divided among three categories of exposure and three exposure situations.

Three categories of exposure described in the system of radiological protection are: occupational exposures, public exposures, and medical exposures of patients (and comforters, carers, and volunteers in research).

The system of radiological protection also recognises three types of exposure situations, intended to cover the entire range of exposure situations. Planned exposure situations involve the planned introduction and operation of sources. Emergency exposure situations are unexpected situations such as those that may occur during the operation of a planned situation, or from a malicious act, requiring urgent attention. Existing exposure situations are situations that already exist when a decision on control has to be taken, such as those caused by natural background radiation. These exposure situations replace the previous categorisation into practices and interventions, although in most cases planned exposure situations are analogous to circumstances in which one is undertaking a practice, and emergency and existing exposure situations are similar to situations requiring an intervention.

For more information on the 2007 Recommendations of ICRP the reader is referred to another paper prepared for this congress by the same author titled "ICRP 103 and Beyond".

Publications since the 2007 Recommendations of ICRP

At the time of writing this paper, ICRP has released seven ICRP publications (formally ten issues) of the Annals of the ICRP, and two additional publications are in press. These are listed in Table 1. In addition, a report titled "Reference Data for the Validation of Doses from Cosmic-Radiation Exposure of Aircraft Crew", jointly prepared by the International Commission on Radiation Units and Measurements (ICRU) and ICRP, is in press and will be published as an ICRU report by Oxford University Press.

Table 1. ICRP Publication to Date since the 2007 Recommendations of ICRP.

ICRP Publication	Title	Ann. ICRP Volume and Issue
104	Scope of Radiological Protection Control Measures	37 (5)
105	Radiological Protection in Medicine	37 (6)
106	Radiation Dose to Patients from Radiopharmaceuticals	38 (1-2)
107	Nuclear Decay Data for Dosimetric Calculations	38 (3)
108	Environmental Protection - the Concept and Use of Reference Animals and Plants	38 (4-6)
109	Application of the Commission's Recommendations for the Protection of People in Emergency Exposure Situations *	39 (1)
110	Adult Reference Computational Phantoms	39 (2)
In press	Application of the Commission's Recommendations to the Protection of People Living in Long Term Contaminated Areas after a Nuclear Accident or a Radiation Emergency	
In press	Avoidance of Unintended Exposure in Radiotherapy with New Technologies	

* ICRP *Publication 109* also includes the article "The History of ICRP and the Evolution of its Policies" by Clarke and Valentin.

These publications support and further elaborate elements of the system of radiological protection as described in the 2007 Recommendations, and most fall into one of two broad categories: those that elaborate on specific exposure categories (occupational, public and medical) or situations (planned, existing, and emergency), and those that provide detailed technical information necessary for the implementation of the system of radiological protection.

Two that do not fit neatly into this categorisation are ICRP *Publication 104* and ICRP *Publication 108*. They are overarching documents which consider, in principle, all exposure categories and situations.

Recent ICRP Publications Elaborating on Specific Exposure Categories and Situations

One of the first publications following the 2007 Recommendations was ICRP *Publication 105* on Radiological Protection in Medicine. This publication describes the application of the system of radiological protection for medical exposures, an exposure category quite different from occupational and public exposures by virtue of the fact that most of the time, most of the benefit and the detriment apply to a single individual: the patient. In the other two exposure categories, typically one group receives most of the detriment (e.g. the workers in a facility), while another group receives most of the benefit (e.g. the public at large receiving electrical power).

Two publications each examine one of the three exposure situations: ICRP *Publication 109* examines emergency exposure situations, and the ICRP publication in press "Application of the Commission's Recommendations to the Protection of People Living in Long Term Contaminated Areas after a Nuclear Accident or a Radiation Emergency" examines the system of radiological protection as applied to existing exposure situations.

These three publications elaborate upon the system of radiological protection as applied in most circumstances. The first covers medical exposures which occur, by definition, only within planned exposure situations. The second covers occupational and public exposures in emergency situations. The third covers public exposures in existing exposure situations, which ICRP considers as the only valid exposure category in that exposure situation. The exposures not within the scope of these three documents are occupational and public exposures in planned exposure situations.

As well, the ICRP Publication in press "Avoidance of Unintended Exposure in Radiotherapy with New Technologies" further elaborates on aspects of radiological protection with respect to medical exposures, focusing specifically on the use of certain new radiotherapy technologies.

Recent ICRP Publications Providing Supporting Information

Within this category is a recent publication specific to radiological protection in medicine. ICRP *Publication 106* is the third amendment to ICRP *Publication 53*, Radiation Dose to Patients from Radiopharmaceuticals. This is part of ongoing work to provide up-to-date biokinetic and dosimetric models for radiopharmaceuticals along with tabulated results giving absorbed doses to the various organs over time per unit activity administered.

Also within this category are two recent publications related to the calculation of protection quantities. Although the concepts of effective and equivalent dose have not changed in the 2007 Recommendations of ICRP, the methods and some of the parameters used to calculate these protection quantities have been updated. Revised radiation and tissue weighting factors were published in the 2007 Recommendations. In addition, the 2007 Recommendations specified that doses should now be calculated using voxel (volume element) phantoms. ICRP *Publication 110* provides the first of these voxel phantoms: those for the reference adult male and female. ICRP *Publication 107* provides updated nuclear decay data for dosimetric calculations, replacing ICRP *Publication 38*, Radionuclide Transformations: Energy and Intensity of Emissions. These publications are necessary steps towards developing dose conversion coefficients compatible with the full implementation of the 2007 Recommendations.

ICRP *Publication 104*: Scope of Radiological Protection Control Measures

In principle the system of radiological protection applies to all exposures to ionizing radiation. However, in practice the measures undertaken to control these exposures must be limited based on practical considerations. ICRP *Publication 104* discusses the scope of radiological protection control measures, and describes certain tools (e.g. exemption, exclusion and clearance) that can be used to manage the scope.

ICRP *Publication 108*: Environmental Protection – the Concept and Use of Reference Animals and Plants

In the past, ICRP concerned itself with the environment primarily with regard to the transfer of radionuclides through it because this directly affects the radiological protection of human beings. In the 1990 Recommendations ICRP stated that "the standard of environmental control needed to protect man ... will ensure that other species are not put at risk. Occasionally, individual members of non-human species might be harmed, but not to the extent of endangering whole species or creating imbalance between species."

However, in its 2007 Recommendations, ICRP acknowledges that that it is now necessary to demonstrate, directly and explicitly, that the environment is being protected. Therefore, it is necessary to develop a clearer framework to assess the relationships between exposure and dose, and between dose and effect, and the consequences of such effects, for non-human species, on a common scientific basis.

To this end, ICRP *Publication 108* sets out some high-level ambitions with regard to environmental protection. To aid in demonstrating whether these ambitions are being achieved, and help optimise the level of effort that might be expended on environmental protection, and ICRP has developed a set of Reference Animals and Plants (RAPs) and derived consideration reference levels (DCRLs). ICRP does not propose the application of dose limits to Reference Animals and Plants.

Current Initiatives and Future Work

The current plan of work for ICRP includes initiatives in several areas. Of significant interest is work on elaborating on the application of the system of radiological protection for radon. ICRP is also continuing efforts to: examine emerging scientific results on radiation effects and how they may impact the system of radiological protection; support calculation of protection quantities (effective and equivalent dose) in accordance with the 2007 Recommendations; and further elaborate on the system of radiological protection in medicine, other specific circumstances, and with respect to protection of the environment.

Radon

In November 2009 ICRP issued a Statement on Radon. The intention is to publish this Statement in the Annals of the ICRP with an accompanying report on assessment of lung cancer risk from radon, after undertaking a public consultation on the two documents together.

The statement reaffirms that, for planned exposure situations, any workers' exposure to radon incurred as a result of their work, however small, shall be considered as occupational exposure. Paragraph 178 of ICRP *Publication 103* defines occupational exposure is as all radiation exposure of workers incurred as a result of their work. It is noted the conventional definition of occupational exposure to any hazardous agent includes all exposures at work, regardless of their source. However, because of the ubiquity of radiation, the direct application of this definition to radiation would mean that all workers should be subject to a regime of radiological protection. Therefore the use of 'occupational exposures' is limited to radiation exposures incurred at work as a result of situations that can reasonably be regarded as being the responsibility of the operating management.

The statement, which will be supported by an accompanying report on assessment of lung cancer risk from radon, introduces a new recommended detriment-adjusted nominal risk coefficient for a population of all ages of 8×10^{-10} per Bq h m^{-3} for exposure to radon-222 gas in equilibrium with its progeny (i.e. 5×10^{-4} WLM^{-1}). The statement notes that this is consistent with other comprehensive estimates including that submitted to the United Nations General Assembly by UNSCEAR.

In addition, ICRP is working on a publication which will provide more specific recommendations for applying the system of radiological protection to radon exposure in various circumstances.

Radiation Effects

The system of radiological protection is based both on science (primarily what is known about the effects of radiation exposure) and value judgements. Therefore, a key component of the work of ICRP is staying current on the latest scientific findings on radiation effects, in particular to evaluate how they might influence the system of radiological protection. Work on assessing the lung cancer risk from radon has already been mentioned. This is only one part of an effort to review the state of knowledge of the health effects of exposure to alpha radiation. Two other groups within ICRP are also examining stem cell biology in relation to radiation effects, and tissue reactions and other non-cancer effects of radiation. The latter group is most interested in effects on the lens of the eye (where evidence seems to point to a threshold for radiation induced cataracts much lower than previously thought), and cardio-vascular effects (which are well documented at doses generally only important for patients undergoing radiation therapy, but for which there may be some emerging evidence of effects at doses approaching the upper end of the occupational dose limits).

Dose Calculation

As mentioned previously the system of radiological protection now recommends that effective and equivalent dose be calculated using more up-to-date radiation and tissue weighting factors, and using voxel (volume element) phantoms. In addition ICRP *Publication 107* provides new radionuclide transformation data, and there are relatively recent modifications to biokinetic models to be taken into account.

ICRP *Publication 110* provides the first voxel phantoms (for the reference adult male and female), but much work remains to develop similar voxel phantoms for paediatric, foetal, and pregnant female voxel phantoms. These phantoms, along with other factors, will then be used to calculate dose coefficients for external and internal exposure for workers and the public.

Radiological Protection in Medicine

When ICRP was created more than eighty years ago, the focus was on radiological protection of medical practitioners and researchers. Today ICRP still has a focus on radiological protection in medicine, and continues to provide recommendations on protection of practitioners, but radiological protection of patients has become a significant issue. Over the last decade this emphasis on radiological protection of patients has grown considerably given the rapid increases in worldwide patient doses from medical imaging.

Currently under way within ICRP are efforts related to the radiological protection of both patients and practitioners related to fluoroscopically guided procedures, including a specific report aimed at cardiologists. Another effort is related to radiological protection in paediatric diagnostic radiology.

ICRP is also working on recommendations on radiological protection education and training for healthcare staff and students, and recommendations that take into account consideration of the possibility of secondary cancers related to some modern radiotherapy techniques.

Further Elaboration of the System of radiological protection

Having broadly described on the system of radiological protection in ICRP *Publication 103*, and elaborated on some general circumstances of exposure in ICRP *Publication 105*, ICRP *Publication 109*, and the ICRP publication in press titled "Application of the Commission's Recommendations to the Protection of People Living in Long Term Contaminated Areas after a Nuclear Accident or a Radiation Emergency", ICRP is now planning a series of publications that further elaborate the system of radiological protection in more specific circumstances. These include reports that will focus on radiological protection related to: aircrew and space crew; naturally occurring radioactive materials; geologic waste disposal; deliberate exposure for security and legal requirements; and, the proper use of effective dose.

Radiological Protection of the Environment

As described previously ICRP *Publication 103* formally introduced explicit protection of the environment into the system of radiological protection, and ICRP *Publication 108* set out some high-level ambitions and a set of Reference Animals and Plants (RAPs) and derived consideration reference levels (DCRLs).

Current efforts in this area are focusing on a broader view of how the protection of humans and non-human biota are integrated into the system of radiological protection, and more specific recommendations on dosimetry for RAPs. The latter includes looking at questions related to radiation weighting factors for non-human biota, transfer values in environmental modelling.

Summary and Conclusion

The system of radiological protection described in the 2007 Recommendations (ICRP *Publication 103*) is an evolutionary change from that described in the 1990 Recommendations (ICRP *Publication 60*). This evolution is necessary in order for the system of radiological protection to remain current with respect to our evolving understanding of the relevant scientific findings, and also to continue to reflect current societal norms. For example, the revised radiation and tissue weighting factors reflect updated scientific knowledge, while a greater emphasis on environmental protection reflects a heightened social awareness of the importance of this area. As well, practical application of the system can point out areas for improvement.

However, as the system of radiological protection in general evolves so must the supporting recommendations and tools. As well, the recommendations must continue to take into account novel uses of radiation in medicine and other fields to help ensure an adequate level of safety under all circumstances.

In the five to ten years leading up to publication of the 2007 Recommendations much of the focus of ICRP was on development of this evolutionary update to the system of radiological protection. Barring revolutionary new findings that demand a rapid overhaul of the system, for the next decade or so ICRP will focus on refining and further elaborating the system of radiological protection as set out in 2007, always with an eye on emerging science, evolving social norms, and novel uses of radiation.

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The Euratom Programme of Research and Training on Low Doses of Radiation

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Abstract
The Euratom programme of research and training on the risks associated with low doses of radiation will be closely aligned to priorities identified in the Strategic Research Agenda (SRA) of MELODI and the Transition Research Agenda (TRA) to be implemented through the project DoReMi funded as part of the 7th Euratom Framework Programme. These research agendas are aimed at integrating the European research effort on low dose, at opening this effort to the wider scientific community and at promoting and facilitating the sustainable integration of key research institutions in European Member States.

Introduction
The mandate of the European Commission to promote and support a programme of research and training in this field arises from Title II Chapter 1 of the Euratom Treaty. This Community programme also includes research on the harmful effects of ionising radiation. The aim of the Euratom Treaty is to promote the safe development of the peaceful use of nuclear energy. However, owing to the link with emission of ionising radiation and the exposure of workers and the public, this development cannot be sustainable without answering the legitimate questions of European citizens on the associated risks. In this regard, low and protracted exposures in the range of 0.1-1 Sv could be expected, but this range of exposures is even more likely to occur during the use of radiation for medical applications, in particular computed tomography. Low doses to normal tissues are also incurred during radiation therapy as a result of exposure to scattered radiation from the main field.

Strategic Research Agendas
The Euratom Framework Programme supported the work of the High Level and Expert Group (HLEG) in 2008 to set-up a Strategic Research Agenda (SRA) on low doses. This agenda promotes the sustainable integration of the research effort of organisations in European Member States, and the European Commission is contributing by providing Framework Programme financial support. This SRA will be the initial deliverable of the "Multidisciplinary European Low Dose Initiative" (MELODI) set up

by the HLEG. Composed so far of 9 European research organisations, MELODI is about to adopt a legal statute which will formalise its existence and allow its involvement in decision making on research strategies on low doses of radiation in Europe. Though it will be some time before the initiative becomes fully operational, this legal process is important to enable sustainability over the long term (20 years). Therefore, in the meantime, the European Commission supports research aimed at ensuring the Transition Research Agenda (TRA). The primary issues to be addressed by the TRA are the identification of priority proposals to be submitted following the Euratom Framework Programme calls in 2011 and 2012. In establishing the TRA, current and recent research findings will be evaluated in order to identify future priorities. This TRA is also, so far, being implemented through the 7th Euratom Framework Programme project DoReMi. More Euratom projects are expected to support the TRA in the future.

Challenge

The field of radiation protection and low dose in particular is a small world. Clear links are already established with other fields of research, especially the medical area. These links are beneficial for both Euratom and medical research. Indeed, through its long history of support to research in this field, which has led to a large number of excellent scientific publications, the Euratom programme contributes to advances in general understanding of the mechanisms of cancer induction and progression. This knowledge is now expanding to other health conditions such as cardiovascular disease. Similarly, the medical sector brings to the table its invaluable cohorts of exposed patients as part of imaging or radiotherapy protocols; these exposures are increasingly well recorded and contain reliable biological parameters. However, more efforts are needed to expand the synergy with other research fields. In particular, more and more knowledge is expected to be gained from the identification of gene expression, protein synthesis and their feedback on the gene expression, which are among the effects observed after exposure to ionising radiation. At low doses of radiation, the statistical power of epidemiology becomes less evident and needs to be substantiated by the observation of identified biochemical processes. Most of the other disciplines in the health sector are using similar approaches and similar tools. It is therefore time to carefully identify the synergies that can help Euratom to make the significant breakthroughs needed in the quantification of the risks arising from low doses of radiation and to open the radiation protection community to the wider scientific community.

In doing so, Euratom will attract new talents to radiation protection and this will also contribute to fulfilling the mandate from article 4 or the Euratom Treaty to develop training in this field. In this regard, Euratom research projects are not intended as a means to finance fellowships per se, rather they offer the possibility to engage graduate researchers in the research activities covered by a Euratom grant. The European Commission recommends that at least 5% of the overall budget of research projects is dedicated to such training activities.

Opening to the wider scientific community is also supported through large research projects such as DoReMi, which is a 6-year project with a total budget of €21M (of which €13M is from the Euratom programme). A significant part of this budget is foreseen for imbedded calls for proposals open to scientific teams which are not yet part of the project consortium. Such a provision is expected to give flexibility to the research, which is likely to need competences that may not necessarily be available among the teams at the origin of the project, in order to address specific issues in the strategic agenda. This also demonstrates that large projects such as DoReMi, which are crucial in terms integration of research, visibility, optimisation of management resources and follow-up, are also making funding available to a wider community than is represented simply by the actual project consortium.

Conclusions

The SRA of MELODI and the TRA of DoReMi will provide crucial orientations for the Euratom programme of research and training on low doses of radiation. This approach is in line with the overall strategy of the Euratom Framework Programme in the area of fission and radiation protection, which is also becoming more closely aligned with the two other SRAs, one drawn up for research in nuclear safety and technology by the Sustainable Nuclear Energy Technology Platform (SNE-TP) and the other being established by the Implementing Geological Disposal Technology Platform (IGD-TP). With this strategy, the European Commission relies on the key nuclear research and industrial organisations in Europe to establish strategic priorities and ways of collaborating to implement agreed agendas. This in turn commits Member States' national research funding and thereby contributes to the joint programming with the Community in priority areas, such as energy research. The success of this strategy relies on the endorsement by the key players of a common vision for the future, their commitment to realising this vision through an agreed strategic research agenda, and the openness towards the wider scientific community and other socio-political stakeholders.

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New Build, environment and waste – application challenges for the ‘new’ RP

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