5th European IRPA Congress
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Encouraging Sustainability in Radiation Protection

Proceedings

IRPA 2018 Proceedings

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Introduction

International Radiation Protection Association (IRPA)

IRPA is the international society for radiation protection with as main purpose to provide a medium whereby those engaged in radiation protection can easily communicate with each other and through this process improve radiation protection in many parts of the world. This includes branches of knowledge such as science, medicine, engineering, technology and legislation, to protect mankind and its environment against the hazards caused by ionising radiation, and thereby implicitly facilitating the safe application of medical, scientific, and industrial radiological practices for the benefit of mankind.

IRPA comprises about 18 000 individual members representing 52 national and regional societies in 67 countries. Through these associate societies, benchmarks of good practice are provided and professional competence and networking is enhanced.

One of the major tasks for IRPA is to provide and support international (regional) meetings for the discussion of radiation protection. Ever since 1966 international congresses have been organised for radiation protection practitioners to gather and exchange achievements, scientific knowledge and operational experience in radiation protection.

5th European Regional IRPA Congress

The Dutch Society for Radiation Protection (NVS) was pleased to host the 5th European Regional IRPA Congress, that took place from 4th to 8th June, 2018 in the historical city of The Hague, The Netherlands.

With the theme “Encouraging Sustainability in Radiation Protection”, the congress focused on aspects needed to make sure that we have, and will continue to have, adequate equipment, staff and resources to protect human health and our environment against the adverse effects of ionising and non-ionising radiation.

More information about the 5th European IRPA Congress can be found in the next contribution, written by the chair of the congress, Hielke Freerk Boersma. In the following sections, we present 65 full papers contributions for this congress, grouped in the principal subjects of interest.

5th European Regional IRPA Congress:

Encouraging Sustainability in Radiation Protection

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KEYWORDS: IRPA, Radiation Protection, Sustainability

Introduction

It has been a great pleasure for me to welcome about 650 colleagues and friends in The Hague at the 5th European IRPA Congress. With this congress we continued a tradition that started in 2002 in Florence although many participants will not have realized that it indeed was the start of a tradition. The first two European regional congresses (Florence, 2002 and Paris, 2006) were 'surrounded' by various sub regional congresses, most of which co-organized with IRPA, the last one being held in 2007. Every four years since then RP professionals from all over Europe and beyond gather and thus fill the gap between the International IRPA congresses.

The Dutch Society for Radiation Protection (NVS) was involved in the organization of three sub regional radiation protection congresses. The most recent one, in June 2003, will certainly be remembered by of some you. About 200 professionals gathered in Utrecht for a congress of the 'North-West European countries' [1] – although it cannot be found in the historical overview on the IRPA website [2], most likely due to the fact that IRPA was not formally involved. Much longer ago, in March 1985, a Western European Regional IRPA congress was held in Maastricht on the theme 'Enhanced Radiation and its Regulatory Implications'. However, historical documents of the NVS do not refer to this congress as being hosted by the NVS. The most successful IRPA congress in the Netherlands – at least according to the number of delegates – was held in 1975 in Amsterdam when the NVS hosted the 2nd Western European Regional IRPA congress. About 500 participants gathered under the presidency of mrs. Zwanette Beekman, the later IRPA president [3,4].

Encouraging sustainability in radiation protection

Preparations for this congress started as early as 2013. In 2015, when we decided on the theme of this congress, we became aware of the United Nations Goals for Sustainable Development [5]. We immediately realized that both radiation research and radiation protection significantly contribute to achieving many of these goals. I merely want to mention the fourth goal: to ensure inclusive and quality education for all and to promote lifelong learning. It will come as no surprise that the UN note that in fact this is a key goal for achieving many other sustainable development goals. You only have to think

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about the goal to promote healthy lives. The past years have also shown a strong interest in attracting young people to the field of radiation protection as a result of the wish to prevent the loss of expertise due to the retirement of many experienced radiation protection professionals. The abovementioned facts supported the decision for the central theme of this congress: Encouraging Sustainability in Radiation Protection.

The prominent place of refresher courses and the Young Professional Award sessions in the scientific programme of this congress reflects our intentions. We were glad to offer the refreshers without an additional fee. Furthermore the Launch event of the IRPA Young Generation Network as well as the IRPA session on Public Understanding of Radiation Risk bear witness of our combined efforts to promote education and lifelong learning in our field of expertise. And then I did not even mention the fact that we also hosted the first school event for secondary schools in The Netherlands, nor that we had the pleasure to host a Workshop of the Foundation for Education and Training in Radiation Protection (EUTERP).

I am very proud on what both the Scientific Programme Committee (SPC) and the Local Organizing Committee (LOC) have achieved in preparation of the congress. I hope you all participants had a great congress and hope they have been able to exchange experiences, to meet old friends and make new friends, and, above all, and that they will remember this congress for its contribution to sustainability in radiation protection. I'd like to conclude this section by thanking all participants symbolically with presenting the special art work that artist and NVS member Arie van't Riet created with the theme of our congress in mind.

Figure 1: X-ray art work 'Stages of Tulips – I' inspired by 'Encouraging Sustainability in Radiation Protection' (*photo: Arie van't Riet*)

Acknowledgements

We owe many thanks to the members of (both the core and the extended) SPC as well as the LOC for their outstanding work! It has been heart-warming to see so many members of the NVS working as volunteers on the success of The Hague 2018. It has also been a great pleasure to work together with IRPA on many parts of the programme. The cooperation with the Society for Radiological Protection as well as EUTERP has been appreciated a lot. We furthermore acknowledge the sponsorship of the WHO, IAEA, ICRP, ICNIRP and the European Commission, as well as the large contributions of the Dutch Authority for Nuclear Safety and Radiation Protection, and the University of Groningen. The financial success of the congress along with the attractiveness for its participants is largely determined by the attendance of many exhibitors of which Mirion Technologies and the Nuclear Research Group (NRG) deserve special attention. The opportunity to host an X-ray art exhibition of Arie van't Riet was gratefully accepted. Finally the work of our congress organizer, A Solution, cannot be overestimated. Thank you all so much for contributing to the success of 'IRPA2018'!

References

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[5] https://www.un.org/sustainabledevelopment/sustainable-development-goals/

Decommissioning

RADIOLOGICAL CHARACTERIZATION OF ACTIVATED MATERIAL AT ACCELERATORS

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Abstract. Materials inside and around accelerators are activated by the irradiation fields caused by the primary beam particles and their interactions with matter. Because of the complexity of these irradiation fields, the radiological characterization of materials for clearance from regulatory control or disposal becomes a challenging procedure. Different particle types contribute. Their energies may vary by orders of magnitude from energies of thermal neutrons to Tera electron volts. The intensities of the fields may change rapidly over relative short distances. Furthermore, the activation products depend strongly on the chemical composition of the objects. Hundreds of different radioactive nuclides may be produced, while typically a significant part is not easily detectable by simple spectroscopy or count rate measurements. In view of these challenges and the change of the Swiss clearance limits in 2018, the radiation protection groups of the European Organization for Nuclear Research (CERN) and the institute of radiation physics of Lausanne University Hospital (IRA) are developing new characterization techniques, which shall allow for efficient radiological controls. The basic ideas of the approach will be illustrated in this contribution by using the example of two accelerators, a proton cyclotron for the production of radio pharmaceutics in the environment of a hospital and a high-energy proton-synchrotron accelerator for physics research at CERN. The benefits as well as the limitations will be discussed and summarized.

KEYWORDS: Accelerators, Characterization, Decommissioning, Clearance, Simulation, Spectroscopy

1. INTRODUCTION

Beside research, hadron accelerators are also widely used in medical applications. While their benefits in research and medicine are undoubted, the operation of such accelerators may lead to the activation of materials, in other words radioactive waste is accumulated. The regulations of radiation protection should be applied to such material based on a detailed radiological characterization. For instance, about 3400 tons were leaving CERN's accelerator tunnels or experimental areas during the last long shutdown at CERN in 2013-2014 and more than 40 000 radiological controls had been performed by CERN's radiation protection group. The radiological classification of the material is required for CERN's internal transport, the choice of workshops and storage, for handling, for export and for elimination as conventional or radioactive waste. As an example for a classification scheme, the requirements for the clearance of solid materials in Switzerland as demanded by the ordinance will be discussed [1]. The radioprotection ordinance applies to (solid) substances or articles (in the following called material) when at least one of the criteria below is fulfilled:

1. The surface contamination exceeds the guidance values specified in Annex 3 Column 12.
2. The ambient dose rate at 10 cm from the surface after subtraction of background exceeds 0.1 µSv/h.
3. The specific and absolute activity of the material is above the clearance limits given in Annex3.

The surface contamination of criterion 1 is a concern only for a small part of the radiation zones at CERN, where high beam intensities interact to a large fraction with massive materials. Typical examples are beam dumps, secondary targets or splitter elements. Criterion 2 can be controlled by relatively trivial measurements and will not be discussed here.

This article will focus on the characterization of materials at accelerators as required by criterion 3. Such materials usually contain from a few up to hundreds of radioactive nuclides. In order to decide, if a material containing a mixture of nuclides can be cleared, the summation rule is applied. Here, the

nuclides are weighted according to the hazard they pose [1]:

$$S = \sum_{i=1}^{N} \frac{a_i}{LL_i} \tag{1}$$

Here, a_i is the specific activity of radionuclides inside the material and LL_i denotes the clearance limits given by the ordinance. A material with $S < 1$ can be cleared. The task of characterization is equivalent to determine the summation rule or, in other words, supplying the list of relevant radionuclides and their activities. The list of nuclides can be extracted from simulations, from measurements or from a combination of both. Each of the different methods have their pros and cons. Such as, it is difficult to prove that simulation assumptions are justified without any measurements. Count- and dose-rate measurements will not give any information about the origin of the detected radiation and therefore clearance limits cannot be applied as necessary. HPGe spectroscopy measurements will give detailed information about high energetic gamma emitters but will not identify alpha, beta or low energetic gamma emitters like Fe-55, Ni-63 or Cl-36.

In order to facilitate the characterization of materials at CERN and to overcome some of these limitations, beside others a new software tool, the Radiological Workstation (RAW) has been developed, which is based on software package ActiWiz [2]. RAW should give easy access to simulation results and allow for a combined analysis of simulation results with measurements.

2. THE SIMULATION TOOLS

The simulation of the activation products in a material is done in several steps. First, the irradiation fields have to be determined at the location of the material. For this purpose, the interaction of the primary beam particles of the accelerator with matter has to be described by a simulation program. Subsequently, the differential fluence $\phi_{ps}^{E}(E) = d\phi_{ps}(E)/dE$ of the secondary particles p_s is scored as a function of energy. The scoring can be limited to particles, which are relevant for activation. For high energy hadron accelerators, the secondary particles, which are important for the activation calculations, are neutrons, protons and charged pions. The calculation of the secondary particle fields is implemented with FLUKA, which is a fully integrated particle physics Monte Carlo simulation package. It has many applications in high energy experimental physics and engineering, shielding, detector and telescope design, cosmic ray studies, dosimetry, medical physics and radio-biology [3],[4].

Once the $\phi_{ps}^{E}(E)$ of a secondary particle type p_s is known, the specific production rate $P_{e_k \to n_n}^{p_s}$ of a radio nuclide n_n for one gram of a chemical element/isotope e_k can be calculated by a convolution of cross-sections with the fluence spectra:

$$P_{e_k \to n_n}^{p_s} = \frac{N_A}{M(e_k)} \int \sigma_{\sigma_{e_k \to n_n}}^{p_s}(E) \cdot \dot{\phi}_{ps}^{E}(E)\, dE \tag{2}$$

Here, $\sigma_{e_k \to n_n}(E)$ is the energy dependent cross section of $p_s + e_k \to n_n + X$, N_A the Avogadro constant and $M(e_k)$ the molar mass of the element/isotope. With the help of the production rates, the number of radio nuclides inside that sample can be calculated using the Bateman equation [5], [6]:

$$Q_n^{p_s,e_k}(t) = \sum_{i=1}^{n} \left[\prod_{j=i}^{n-1} k_{j,j+1} \times \sum_{j=i}^{n} \left(\frac{Q_i(0)e^{-k_j t}}{\prod_{\substack{p=i \\ p \neq j}}^{n} (k_p - k_j)} + \frac{S_i\left(1 - e^{-k_j t}\right)}{k_j \prod_{\substack{p=i \\ p \neq j}}^{n} (k_p - k_j)} \right) \right] \tag{3}$$

Here, Q_n is the number of atoms of species n present at time t, k_n is the decay constant for species n (k = ln2/τ = 0.69315/τ), $k_{n,n+1}$ is the partial decay constant (partial removal constant) and is related to the branching ratio $BR_{n,n+1}$ through the relation $k_{n,n+1} = BR_{n,n+1} \times k_n$. As source terms, $S_i = P_{e_k \to n_n}^{p_s}$ are used in the equation (3).The total number of radionuclides of species n of a material sample composed by K different elements/isotopes and being irradiated by the irradiation fields composed of S different particle types can be determined with equation:

$$Q_n(t) = \sum_{s=1}^{S} \sum_{k=1}^{K} w_k Q_n^{p_s,e_k}(t) \tag{4}$$

The variable w_k denotes the weight fraction of the element/isotope inside the material. The specific activity needed for the summation rule in equation 1 is obtained by:

$$a_n(t) = k_n Q_n(t) \tag{5}$$

If $\dot{\phi}_{ps}^{E}(E)$ is normalized to the rate of interacting beam particles R, we obtain the specific activation rate f_n instead with:

$$a_n(t) = f_n(t)R \tag{6}$$

The equation (3) gives the correct solution for irradiation fields with constant fluence rates. Irradiation fields, which vary in time, so-called irradiation patterns, can be approximated by sequential calculations of periods with virtually constant rates and by combining the different results. Cooling periods can be considered by setting the source terms S_i in (3) to zero.
The determination of production rates (equation 2) and the subsequent calculation of activities of composite materials for irradiation fields (equation 3 and equation 4) with complex time patterns is implemented in a very efficient way in a new software package 'ActiWiz 3.3' [8].
This package was recently developed at CERN by Helmut Vincke and Chris Theis. Presently, irradiation fields of neutrons, protons, charged pions and photons can be handled. The production rates for 85 different chemical elements and isotopes can be calculated in an energy range from thermal neutrons to 100 TeV.
As already mentioned, the irradiation fields have to be calculated by particle physics simulation packages like Fluka. This is still a difficult and time-consuming task. In order to organize the characterization of material from accelerators more efficient, it may be advantageous to calculate a set of irradiation fields for strategically expedient positions and combine them with sets of irradiation patterns and cooling times. One irradiation field, in combination with an irradiation pattern and a cooling time will be called in the following an irradiation scenario. A set of irradiation scenarios will be called a grid. An example for a generic grid as being used frequently at CERN is given in Table 1. Here, 'a' represents the index for the accelerator denoted by the accelerator energy, 'p' the index for the radial position of the irradiation field relative to the beam axis. The pair of indices (a,p) represents an irradiation field (Fluka), which can typically be found in accelerator 'a' at position 'p'. 'b' is the index of the irradiation pattern and 'c' the index of the cooling time. In this example, a trivial beam pattern is assumed, where the interacting beam particle rate R is constant for the given time period followed by a cooling time c.

Table1: Generic grid for accelerators at CERN.

Accelerator Index: a		*Position Index: p*		*Irradiation time b*		*cooling time c*	
0	*7 TeV (LHC)*	*0*	*beam impact*	*0*	*1 w*	*0*	*1 d*
1	*400 GeV (SPS)*	*1*	*bulky material*	*1*	*1 m*	*1*	*1 w*
2	*14 GeV (PS)*	*2*	*adjacent to bulky*	*2*	*2 m*	*2*	*2 w*
3	*1.4 GeV (Booster)*	*3*	*close to the concrete tunnel*	*3*	*4 m*	*3*	*1 m*
4	*800 MeV (MED)*	*4*	*Behind massive concrete shielding*	*4*	*8 m*	*4*	*2 m*
5	*160 MeV (LINAC)*	*5*	*10 cm lateral distance to target*	*5*	*1 y*	*5*	*4 m*
		6	*close to the concrete tunnel-wall*	*6*	*2 y*	*6*	*8 m*
				7	*5 y*	*7*	*1 y*
				8	*10 y*	*8*	*2 y*
				9	*20 y*	*9*	*5 y*
						10	*10 y*
						11	*20 y*
						12	*40 y*

For this grid, Actiwiz will return for each of the 69 chemical elements and each of the 5460 irradiation scenarios a tuple containing the nuclides n_i and specific activities a_i (or activation rates f_i) as expected by the simulation:

$$\mathrm{T}_{\mathrm{a,p,b,c}}^{\mathrm{elem}} = \{(\mathrm{n}_1, \mathrm{a}_1), (\mathrm{n}_2, \mathrm{a}_2) \ldots \ldots (\mathrm{n}_\mathrm{N}, \mathrm{a}_\mathrm{N})\} \tag{5}$$

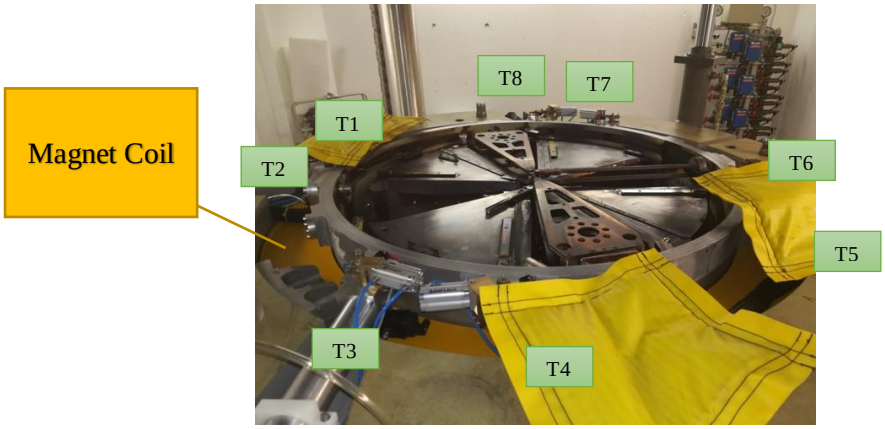
The nuclide content of compound materials can be calculated easily from the 69 chemical elements.

One of the main advantages using ActiWiz is having fast access to the radio nuclide content of arbitrary materials being irradiated by the irradiation fields inside CERN accelerator tunnels. RAW will use the results of the ActiWiz calculations and stores them optionally for fast access in a data base. From here, the data is available for further analysis by RAW as will be demonstrated below by two examples. The grid of irradiation scenarios can be adapted to the needs of the user.

3. EXAMPLE 1: CHARACTERIZATION OF MAGNET COILS OF A CYCLOTRON

A cyclotron of type IBA Cyclone 18/9 is used for the manufacturing of radiopharmaceuticals at the Geneva University Hospital (HUG). The machine is shown in Figure1. Being installed in the year 2000, the magnetic coils of the machine – about two tons of copper- had to be exchanged after 15 years of operation. The material will be declared as waste and has to be characterized in respect to the summation rule in equation 1. The main reason for the activation of the machine are neutrons which originate from the different production targets for pharmaceuticals (typically O-18 enriched water). They are irradiated with the 18 MeV primary proton beam of the cyclotron. The cyclotron delivers beam currents up to 150 µA (~10^{15} p/s). Typically, 100 GBq of F-18 are produced per day. The activation of the magnet coils is in first approximation a superposition of the contributions of the individual target cells (other beam losses inside the machine neglected).

Figure 1 Open cyclotron with iron yoke, vacuum chamber and magnetic coil during maintenance. The position of the production targets are labelled with T1- T8.



The integrated beam current of each target cell is shown in Figure 2 (left) for the relevant time period. The right picture shows exemplarily the irradiation pattern (beam current as function of time) for target cell 1. For the characterization of the magnet coils a grid was defined (see Table 2). The triple of indices (a, p, b) defines the irradiation field at position (a, p) inside the magnetic coil emitted by target cell 'b'. The underlying coordinate system is explained by Figure 3. The parameter 'b' indicates simultaneously the irradiation patterns being assumed for the calculations. The index 'c' represents the cooling time after the removal of the coil in 2015. In a first characterization step: the irradiation fields for the 32 positions (a, p, b=0) were simulated using Fluka. Secondly, the specific activities $a_n(t)$ of the activation products in copper are calculated for the 2048 predefined irradiation scenarios of the analysis grid (< 30 min computing time), using the corresponding irradiation patterns of each target cell. As a last step, the results of each target cell were combined:

$$T_{a,p,c}^{copper}\{(\mathrm{n}_1,\mathrm{a}_1),(\mathrm{n}_2,\mathrm{a}_2)\ldots\ldots(\mathrm{n}_\mathrm{N},\mathrm{a}_\mathrm{N})\} = \sum_{b=0}^{B} T_{a,p,b,c}^{copper}\{(\mathrm{n}_1,\mathrm{a}_1),(\mathrm{n}_2,\mathrm{a}_2)\ldots\ldots(\mathrm{n}_\mathrm{N},\mathrm{a}_\mathrm{N})\} \quad (6)$$

Here, B are the number of target cells. In June 2017 (roughly two years of cooling), samples were taken from the copper coils and analyzed by gamma spectroscopy. For that date, the spatial distribution of activities within the copper coil is given by $T_{a,p,c=3}^{copper}$ from RAW. The result for Co-60 is shown in Figure 4. Where available, the spectroscopy results for Co-60 were added to the plot (squared markers). The agreement between the measurement and Raw is typically within a factor 2-3. The predictions of the activities themselves vary by more than a factor of 10000. After two years of

cooling, the second relevant activity expected by RAW is Ni-63. Its activity distribution is shown as well for position P1. Due to its β-decay, it is not detectable by gamma spectroscopy. At this date, the largest part of the coil is radioactive because of Co-60 exceeding the clearance limit of 0.1 Bq/g. Near the hotspot close to the target cell one also Ni-63 exceeds its limit of 100 Bq/g.

Figure 2: Left: Total number of protons delivered on targets between 2006 and 2015. Right: Mean Bam intensity per year for Target cell 1.

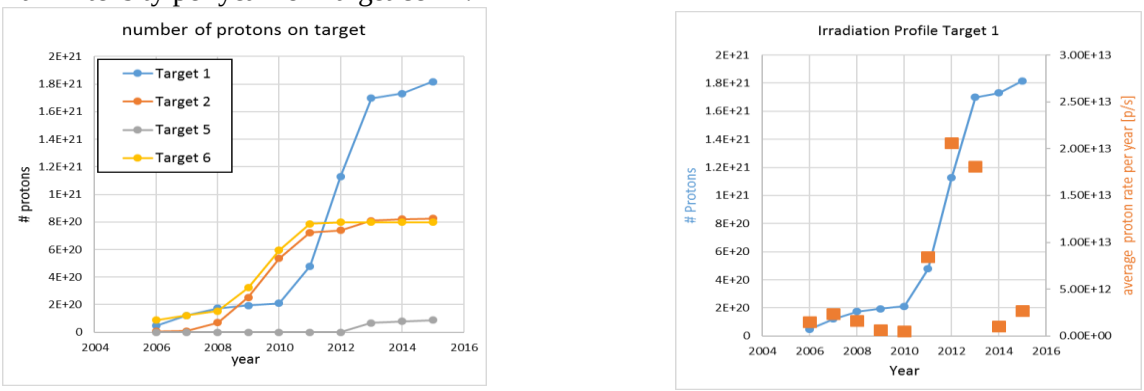


Figure 3: Geometrical model being used in FLUKA. The spatial distribution of the activities within the magnet coils is given for four positions P1-P4 (grid parameter 'p') as a function of the azimuthal angle φ (parameter 'a'). φ is defined relative to target cell 1 with φ=0 being in its centre.

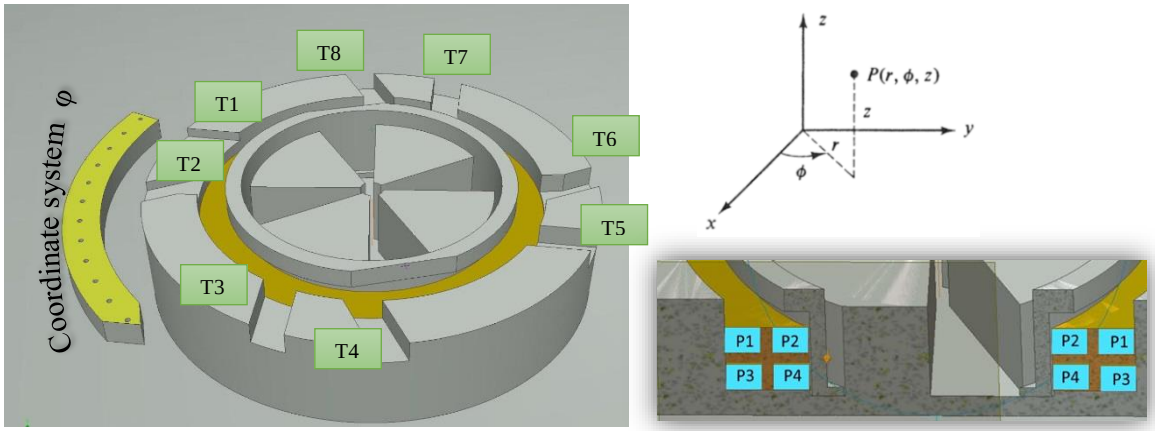


Table 2: Grid for the characterization of the magnet coils of the cyclotron.

Index: a		*Position Index: p*		*Irradiation pattern b*		*cooling time c*	
0	*φ(0-15)*	*0*	*Pos.1*	*0*	*Target 1 , 2006-2015*	*0*	*1 w*
1	*φ(15-45)*	*1*	*Pos.2*	*1*	*Target 2 , 2006-2015*	*1*	*1 m*
2	*φ(45-60)*	*2*	*Pos.3*	*2*	*Target 3 , 2006-2015*	*2*	*1 y*
3	*φ(60-75)*	*3*	*Pos.4*	*3*	*Target 4 , 2006-2015*	*3*	*2 y*
4	*φ(75-90)*			*4*	*Target 5 , 2006-2015*	*4*	*5 y*
5	*φ(90-135)*			*5*	*Target 6 , 2006-2015*	*5*	*10 y*
6	*φ(135-165*			*6*	*Target 7 , 2006-2015*	*6*	*20 y*
7	*φ(165-180)*			*7*	*Target 8 , 2006-2015*	*7*	*30 y*

In Switzerland, a cooling down period of 30 years is allowed and requested. Figure 5 show the distribution of S extrapolated to this scenario by RAW. More than 75% of the spatial points are now non-radioactive and could be cleared in future. The data shown here come from a first test analysis

and are still preliminary. If the characterization of the magnet coil is successful, RAW could be extended to other elements of the cyclotron and the bunker where it is installed.

Figure 4: Spatial distribution of the specific activity of Co-60 and Ni-63 inside the magnet coils.

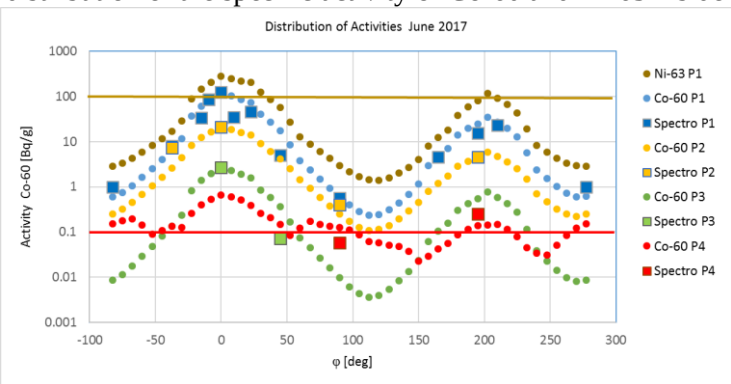
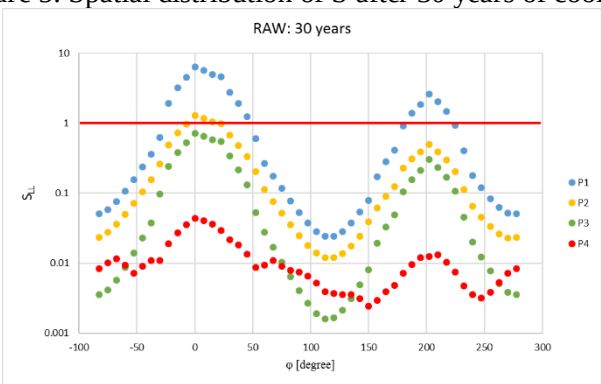


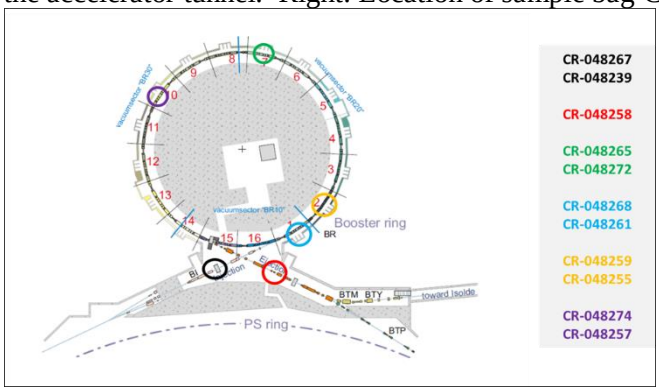
Figure 5: Spatial distribution of S after 30 years of cooling.



4. EXAMPLE 2: CHARACTERIZATION OF IRRADIATION FIELDS.

In contrary to the example above, the interaction rate of beam particles *R* in terms of beam losses are often badly defined at accelerators. The properties of the irradiation field at a given location can still be extracted by combining the information of simulations with the results of spectroscopy measurements on material samples. This will be illustrated on the example of the Proton Synchrotron Booster at CERN. This machine is made up of four superimposed synchrotron rings that receive beams of protons from the linear accelerator Linac 2 at 50 MeV and accelerate them to 1.4 GeV for injection into the proton synchrotron (PS). Several bags with copper, steel ST304 and aluminium 6082 samples, were installed inside the Booster tunnel next to the machine (see figure 6).

Figure 6: Booster ring. The coloured circles symbolizes locations of samples (CR-xxx) installed inside the accelerator tunnel. Right: Location of sample bag CR-048274.



Raw allows for an automatic comparison between spectroscopy results and the simulation predictions of Actiwiz. This is illustrated here by the example of copper CR-048274 (see figure 6). The sample was irradiated during 4 month and measured after 1 week cooling by spectroscopy. For the comparison, the generic grid of the accelerators at CERN is taken as already defined in table 1. The corresponding irradiation scenario on the grid is: a=3, p=3, b=3 and c=1. The result of the comparison is summarized in table 3. By a quantitative analysis of the detected activities a_i^{meas} and the expected activation rates f_i, the strength of the irradiation field at the location of the sample can be determined using equation 6. No additional information about the beam loss rate in the vicinity of this sample is necessary. The result of this analysis for the spectral flux density is given in figure 7. Subsequently, the activities a_i^{calc} of all nuclides expected by Actiwiz can be calculated. S_i^{mix} is the best estimate for the terms of the summation rule including difficult to measure radio nuclides. The quality of agreement between simulation and measurement can be expressed with

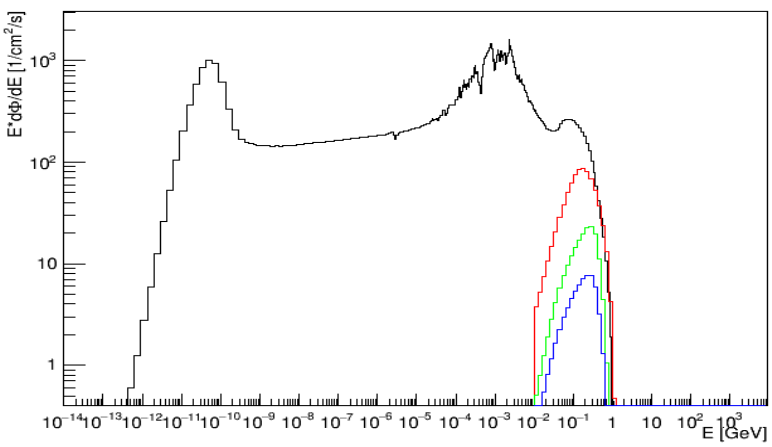
$$\chi^2 = \sum_{i=1}^{n} \chi_i^2 = \sum_{i=1}^{n} \frac{\left(a_i^{meas} - a_i^{calc}\right)^2}{\sigma_i^2} \tag{8}$$

The lower the χ^2-value the better the agreement. Empirically, a χ^2/n-value below 50 suggests that an irradiation scenario may be compatible with the measurements, while larger values indicate a significant incompatibility between the model predictions and the measurements. For completeness, $Ratio_i$ in Table 3 denotes the proportion between a_i^{meas} and a_i^{Calc}.

Table 3: result of the comparison between spectroscopy and ActiWiz.

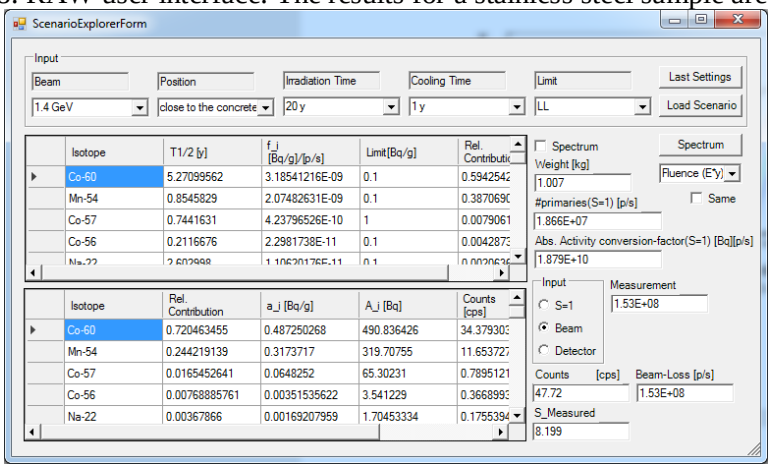
N_i	$T_{1/2}$	LL_i	a_i^{meas}	MDA	R_i	a_i^{calc}	$Ratio_i$	χ^2	S_i^{mix}
	[years]	[Bq/g]	[Bq/g]	[Bq/g]	[p/s]	[Bq/g]			
Co-58	1.94e-001	1.0	0.420	-	2.72E8	0.331	0.788	1.638	0.4199
Mn-54	8.55e-001	0.1	0.040	-	3.75E8	0.023	0.572	11.540	0.4042
Co-56	2.12e-001	0.1	0.030	-	1.50E8	0.043	1.425	2.204	0.3047
Co-60	5.27E0	0.1	0.023	-	4.50E8	0.011	0.476	26.915	0.2257
Co-57	7.45e-001	1.0	0.108	-	2.51E8	0.092	0.855	0.606	0.1079
Sc-46	2.30e-001	0.1	0.004	-	4.00E8	0.002	0.536	11.700	0.0372
Fe-59	1.22e-001	1.0	0.034	-	4.89E8	0.015	0.439	35.141	0.0336
Zn-65	6.69e-001	0.1	-	-	-	0.001	-	-	0.0072
V-48	4.38e-002	1.0	0.006	-	2.26E8	0.006	0.948	0.070	0.0061
Mn-52	1.53e-002	1.0	0.005	-	3.19E8	0.004	0.672	5.138	0.0055
Cr-51	7.59e-002	100.0	0.053	-	3.78E8	0.030	0.567	7.263	0.0005
Na-22	2.60E0	0.1	-	-	-	0.000	-	-	0.0003
Be-7	1.46e-001	10.0	-	-	-	0.002	-	-	0.0002
Cu-64	1.45e-003	100.0	-	0.251	-	0.018	-	-	0.0002
Fe-55	2.74E0	1000.0	-	-	-	0.014	-	-	0.0000
result					2.14E8			102/10	1.5533

Figure 7: Particle Flux at the location of the copper sample in the booster determined by RAW with neutron (black), proton (red), π^+ (green) and π^- (blue).



Once the particle flux is known at the location of the sample and the correctness of the model assumption is validated in terms of χ^2. The activation products for practically any material and for arbitrary irradiation and cooling times can be predicted by RAW – of course under the assumption that the beam loss mechanism does not significantly change. In figure 8, the results are shown for a stainless steel sample as obtained by RAW. Twenty years of irradiation were assumed with one year of cooling down with an irradiation field as in Figure 7. The Relative contribution of each nuclide to S is listed beside the total value of the summation rule.

Figure 8: RAW user interface. The results for a stainless steel sample are shown.



5. SUMMARY

Rate measurements and spectroscopy results are not able to deliver complete information for the radiological characterization of materials. The missing information can either be complemented by time or cost expensive measurements like radio chemical analysis, or from simulation. The software tool RAW goes the second way. RAW together with Actiwiz and Fluka was applied for the characterization of material from a cyclotron and the results were presented. First results look promising. Beside the characterization of material, RAW can also deliver essential information about the irradiation fields at accelerators in situations, when beam loss mechanisms and rates are badly defined.

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Education

Engaging with School Children - Experiences from the SRP Schools Outreach Programme

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Abstract

Forging links with schools and opening up conduits to educate and inspire young persons is a key route to engaging the public with topics related to radiation and its safe use. In the author's opinion it is the duty of each radiation protection professional society around the world to do what it can to ensure that our 'adults of the future' receive robust and balanced information about radiation risks and benefits in a proactive, interactive and stimulating way.
This paper describes the efforts made by the UK's Society for Radiological Protection (SRP) over the last 6 years to realise its strategic commitment to "deliver a strong and effective outreach programme" with particular focus on school-aged students. The methods and resources that have been developed together with the experience gained are shared and may be of use to other IRPA Associate Societies that are planning to attempt similar engagement projects.

Introduction

The UK's Society for Radiological Protection (SRP), founded in 1963 and granted a Royal Charter in 2007, is Europe's largest learned body for radiological protection professionals. The Society has a Charter duty to "promote, advance and disseminate, to the public advantage, knowledge of radiation protection and allied fields". This duty has been an integral part of SRP's strategic plan for the last six years. Indeed its current Strategic Plan (2018 – 2022) contains a commitment to "deliver a strong and effective outreach programme" over the next five years and a number of goals to engage with school students in particular.

The aspiration is two-fold:

1. To inspire young persons with robust and balanced information about radiation science and safety in a fun, interactive and stimulating way which may facilitate them to become the knowledgeable and rational adults of the future, and
2. To encourage school students to consider further studies of scientific topics whilst still in school, in higher education college, or at university, and to contemplate a science-based career particularly in radiation protection.

Engaging the interests of school students before they make educational choices that lead to a particular career path has long been a way of influencing the young people to follow a career in science. The UK Government's STEM (Science Technology Engineering and Mathematics) programme, established in 2006, provides an example of how industry and the professions are responding to the skills gap shortage through the medium of education. This initiative is supported through a network of regional centres providing curriculum related resources and activities. A national body SCORE (Science Community Representing Education) includes partners from the Royal Society, the Institute of Physics, the Royal Society of Chemistry, the Society of Biology and The Association for Science Education. The Engineering Council's 'Inspiring the Future' programme brings professionals into schools to talk directly to students about the work experience. Popular 'Big Bang Fairs' already highlight the exciting possibilities that exist for young people with science, technology, engineering and maths backgrounds.

Following these examples, and as part of its charitable duties to serve the public, the SRP has recognised the need to take a proactive outreach role to engage with young people with respect to the science of radiation protection.

With reference to The British Science Association's Audience Segmentation Map (BSA, 2016) shown in Figure 1, the author believes that the 'Interested' category - reportedly containing about 50% of the UK population and the one that most school students might fall into - is where SRP's engagement efforts will be most effective, have the strongest impact, and thus lead to the best possible long-term benefits to the public.

The SRP's first major foray into schools' outreach came at the IRPA 13 Congress in Glasgow in 2012 with a large exhibition, and 'schools lecture' that focussed on the medical aspects of radiation. The event attracted approximately 1200 students plus their teachers from all around Scotland. One of the legacies from this event

was a suite of 40 large posters covering radiation science and some of the ways radiation sources are routinely used in industry, research and medicine. These posters have been updated and are freely available for use, typically by schools.

Building upon the success of this event, in 2013 the established a dedicated Schools Outreach Working Group with the responsibility of providing information to schools. One initial objective for the group was to develop a central resource of material that would be available for members and teachers to use for engaging with school students and stimulating their interest in science, specifically in the field of radiological protection. Another objective was to establish an annual Schools' Event for secondary school students and their teachers. Over the next few years this group evolved into the current Outreach Committee which sits within the SRP's Engagement Directorate together with the Rising Generations Group, the Communications Committee, Journal Board, the Committee for Liaison with IRPA and Partner Societies, and the Events Committee.

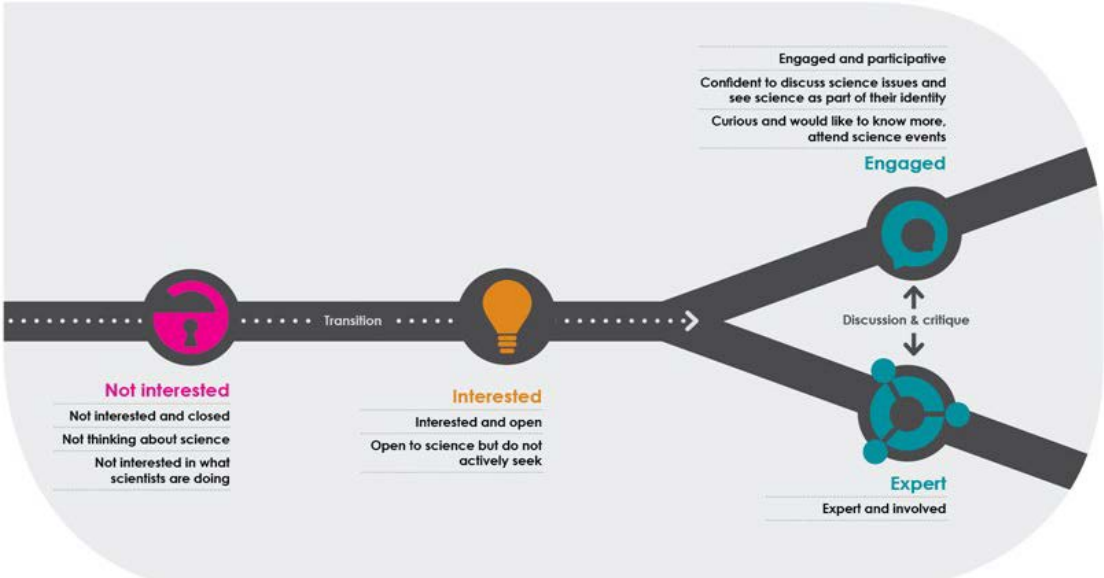

Figure 1: The BSA Audience Segmentation Map ("Transforming Science Engagement", BSA, 2016)

The SRP Schools Events

The SRP Schools Event targets students who are typically aged 14 to 15. The event aims to excite, enthuse and inspire these students using a variety of interactive talks, demonstrations, exercises, games, and quizzes.

The second Schools Event was held in Harrogate at SRP's 50th Anniversary annual meeting in 2013 (SRP, 2013). It attracted just over 400 school students. Harrogate, in North Yorkshire, has fewer secondary schools within easy travelling distance, which partly explains the reduced student numbers from Glasgow. The lecture was entitled "What's the Risk? – Putting numbers on the chances in your life". This was delivered by Professor Sir David Spiegelhalter, Winton Professor of the Public Understanding of Risk at the University of Cambridge. The third staging of the Schools Event was in Southport, at the SRP 2014 annual conference and attracted a similar attendance as in the previous year. Once again it involved a large exhibition of fun and interactive activities from over 30 exhibitors from a variety of sectors including Sellafield Ltd, the National Institute for Medical Research, the Royal Navy, Public Health England, and the emergency services. This time, rather than inviting an external speaker, the on-stage lecture was developed and delivered by a young team from the SRP's Rising Generations Group (Young Generation Network of the UK SRP). This allowed the students to see young radiation protection professionals as roles models. It also supported the SRP's policy of developing its younger members to become more proficient communicators and facilitate future schools engagement events by increasing the 'pool' of young and enthusiastic speakers. The 'lecture' became an interactive 'show' featuring three on-stage demonstrations using audience members and posing questions to which the audience could respond 'YES' by waving a green-coloured card or 'NO' with a yellow-coloured card as seen in Figure 2.

Figure 2: The Schools Event on-stage interactive show

In 2015, at Eastbourne, attendance numbers dropped to 200 and for 2016 in Llandudno the event was hosted by a local high school thereby palliating the problems encountered in getting students out of the school and transported to a central conference venue. Although numbers recovered to 400, a strategic decision was made to concentrate SRP's schools engagement efforts into participation in large national STEM events in order to maximise the impact of the limited outreach budget.

The National STEM Big Bang Fair

Held annually at the National Exhibition Centre in Birmingham this schools and public engagement event attracts approximately 70,000 students over its four days.

The SRP stand, measuring 4 x 12 metres, was divided into three main sections, and each section accommodated an interactive demonstration that had been developed out of the three demonstrations used in the 'on-stage' shows of the previously described Schools Events run at SRP conferences.

A. Fighting the Photon:

This shielding demonstration used Nerf guns to simulate a collimated beam of photos which were plastic 'darts' with blunt Velcro tips. Three different shields were constructed from wooden boards as shown in Figure 3. The two different sizes of holes represented two different materials with two different photon interaction cross-sections whereas the two different numbers of holes represented two different physical densities. So that shield 1 (green) and shield 2 (yellow) are the same material but with different densities. Shield 3 (red) represents the same density as shield 1 but is a different material.

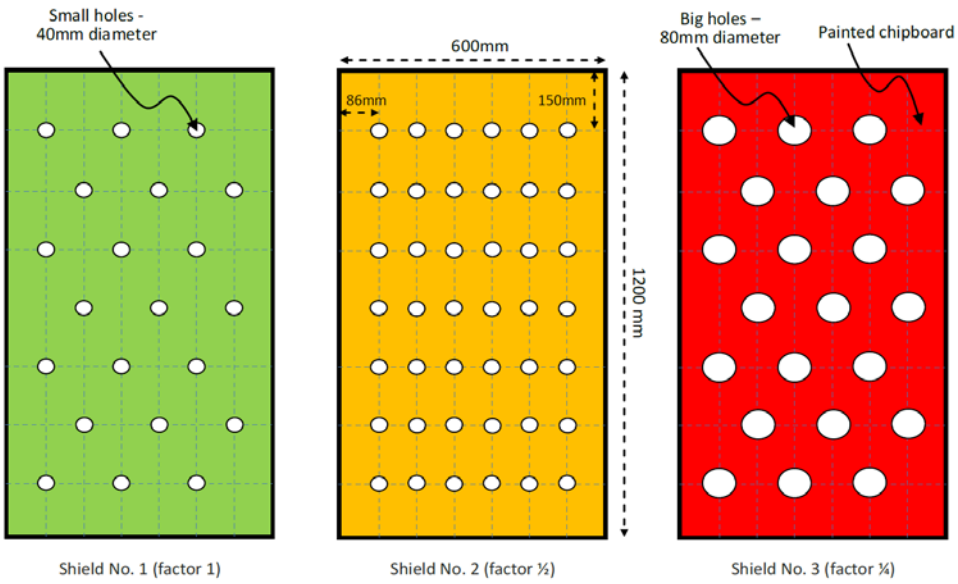

Figure 3: 'Fighting the Photon' shields

As shown in Figure 4, an SRP volunteer stands behind each shield in turn whilst a school student fires 'photon' darts at the shield for a fixed time e.g. 30 seconds. For each shield the numbers of darts that penetrate through the holes and stick to the volunteer are counted. The counts are compared to each

other and to a count with no shield in front of the volunteer. Students are asked to comment on the counts and explain the differences.

Figure 4: The green shield (number 1) in use at The STEM Big Bang Fair

B. Everyday Objects:

Students, usually a group of 5 to 6, play a game where they have to rank a number of 'household' objects from most radioactive to least (see Figure 5). The objects are typically, and in order:

- Smoke detector
- Luminous dial wrist watch
- Welding rods
- Lo-salt
- Brazil nuts
- Bananas

Students get to use both a GM and a scintillation counter to help them make their ranking decisions, and following their attempt they can engage in a discussion with SRP volunteers regarding the activities of which radioisotopes are present in each object. This highlights that radioactive substances in small concentrations are all around us in the world.

Figure 5: Radioactivity in everyday objects

C. Half-Life Paddling Pool

In this exercise, 64 plastic discs are divided up into 8 buckets. Each disc has a trefoil symbol on one side and the other side is blank. The students throw the discs into an inflatable paddling pool as shown in Figure 6. The discs that land blank side up are removed. The remaining trefoil side up discs are counted and returned to the buckets for another throw. Throws continue until no trefoil discs remain. Each throw represents a half-life and students plot the number of 'trefoils up' discs versus throw number to generate an 'exponential' decay curve. A discussion about the random nature of decay and half-lives follows the exercise.

Figure 6: The Half-Life Paddling Pool

D. Posters and Quiz

To add to the interactions available through the above three demonstrations, a bank of over 40 posters of colourful radiation protection information have been produced, and about 15 of these are displayed on the walls around the inside of the stand.

In order to persuade the students to read the posters, a quiz of ten simple questions has been developed and each student visitor to the stand is given a quiz sheet, a free SRP pen and encouraged to do the quiz. The answers to the questions are all contained somewhere within the text of the 15 posters. As an added incentive, the winner on each day, drawn from a 'pot' of '10 correct answer' quiz sheets, wins a prize such as a radio-controlled quadcopter.

New demonstrations have already been developed and implemented at the STEM Big Bang Fair in 2018. These include 'Fission Chips' where an array of toppling dominoes is used to simulate nuclear fission chain reactions and wooden blocks act like control rods.

Resources

It is important to back up these engagement events with educational resources that can be obtained by students and teachers when they have returned to their schools and colleges. SRP have set up a 'Radiation Protection Schools Resources' webpage. Available on this page for free download are currently:

- The 40+ posters referred to above.
- Six lesson plans
- Videos of school lectures

The intention is to grow these resources over the next few years – see 'Future Plans' below.

Other Initiatives

Over the last four years the SRP Outreach Committee with assistance from the Rising Generations Group has participated in a number of other initiatives and events including:

- Liverpool Youth Centre Project: 2015 – to engage with under-privileged young persons in Merseyside who have fallen outside the mainstream education system.
- Get Girls into Physics: 2016, Vale of Leven Academy near Glasgow.
- New Scientist Live: 2017, ExCel, London – a 4-day national public engagement event.
- Manchester Science Festival: 2015 and 2016, Media City, Salford – a 2-day regional public engagement event.
- The Geological Society: 2018 – SRP begins a collaboration by producing information leaflets for high school students on radioactivity and radioactive waste management.

Developing Impact Metrics

Attempts have been to measure the 'impact' of SRP's school engagement work. Simple metrics have so far included the number of 'visitors' to the SRP stand at engagement events. This has been estimated by:

- Performing a 'head count' of persons on the stand at various times during the day.
- A count of the number of completed quiz sheets
- A count of the number of 'further information' request slips
- A count of the number of post-event Twitter 'likes' @SRP_UK or @SRP_RGG

The "Today on this stand I learnt …" poster shown in Figure 7 was displayed on the SRP stand for each day of the STEM Big Bang Fair in 2018. Students were provided with a yellow sticky dot and encouraged to place the spot on the particular roundel that indicated what they felt they had learnt – from '10 – Lots' down to '1 – Nothing'. The numbers of dots in each section were used to produce an impact 'Figure of Merit' (FoM) as follows:

$$\text{Impact FoM} = \sum \text{(Number of dots in each category * category weighting)}$$

For example:

Category	10	9	8	7	6	5	4	3	2	1
I learnt …	Lots					Something				'Nothing'
Weighting	1.0	0.9	0.8	0.7	0.6	0.5	0.4	0.3	0.2	0.1
Number of Dots	117	70	60	43	23	16	7	7	4	10
Weighted Dots	117	63	48	30.1	13.8	8	2.8	2.1	0.8	1
Summation of Dots			**357**							
Summation of Weighted Dots (Impact FoM)			**286.6**							
Percentage in Categories '8' to '10'			**69.2**							

In this case the resulting impact FoM (scaled to Category 10 'I learnt lots') = 286.6 and this number can be used to gauge overall impact and make comparisons with other events.

Figure 7: Measuring impact with a "Today on this stand I learnt …" poster.

At the UK Radiology Congress in 2015 a qualitative attempt was made to assess the impact on student thinking of a lecture on "The Importance of Radiation in Medicine". Before the lecture students were asked to think of the first three words that come to mind when they heard the word 'radiation'. This was repeated following the lecture. The collected data was used to create 'before' and 'after' 'word clouds' which are shown in Figure 8. In such 'clouds' the more often a word is contributed the larger it appears. The change in opinions is evident and it appears that students are a lot more positive about radiation having attended the lecture.

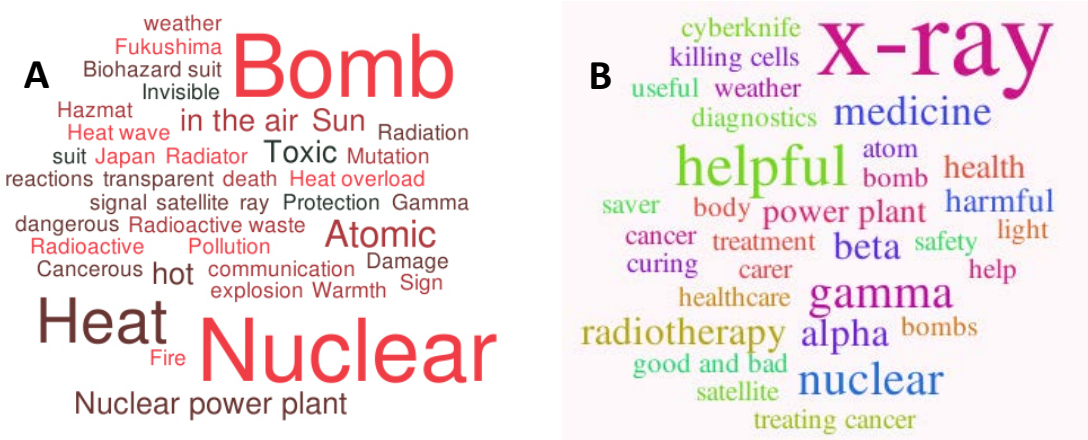

Figure 8: Word clouds showing student feelings (A) before and (B) after a lecture on "The Importance of Radiation in Medicine"

Costs

Outreach work is expensive. Over the last five years it is estimated that SRP has spent approximately £107,000 on schools engagement work. Over the next five years SRP has pledged to continue to support these projects with a budget of £20,000 per annum. With this level support there is a limited scope of work that is achievable per year as seen in Figure 9.

EVENT	Potential Timing	Potential Location	Potential Impact*	Estimated Cost (£)	Efficiency# (£)
STEM Big Bang	March	Midlands	7000	14000	2
SRP Schools Event with AGM (just 'stage show')	May	South England	600	3000	5
Small Event 1	June or July	Wales or North England	300	1500	5
Small Event 2	October	Scotland	300	1500	5
TOTAL				20000	

* Number of persons engaged by SRP stand/volunteers
\# Cost per 'head' figures

Figure 9: Proposed annual schedule of schools engagement work based on current budget

This level of support by SRP may not be sustainable in the long-term beyond five years, and if the work is to continue, indeed if the scope is to expand, then there must be a concerted effort to generate alternative funding streams e.g. donations from industry, grant applications, develop collaborations with other 'outreaching' organisations, employing the services of professional fundraisers, and 'crowd funding'.

Future Plans

It is SRP's aim to expand the amount and variety of its free web-based resources for school children and teachers. As well as adding to the existing library of information posters, lesson plans, and videos of lectures, the intention is to explore the development of a series of radiation science and safety quizzes for various school age-groups, set up on-line competitions, create 'talking head' vlogs, encourage the use of Twitter groups (e.g. #SRPinschools) and dedicated Facebook pages, provide free webinars for schools, as well as the production of a newsletter aimed at young persons. The whole arena of social media requires a thorough exploration.

To capitalise on a large potential pool of SRP member volunteers across the UK plans are afoot to develop standardised 'in-school' resource packs to include: slide sets on a variety of radiation protection topics consistent with the school syllabus, practical equipment for fun demonstration(s) or a lab lesson, information

leaflets relating to topical radiation protection subjects to support the teacher with projects or debates, plus quizzes, competitions, and 'freebies'. Such packs can be couriered out to any volunteer member anywhere in the UK thereby allowing them to go into schools in their local area and engage to teachers and students. The pack can then be returned to a central location for restock and subsequent re-use by another SRP volunteer member in another part of the UK.

In addition, there are plans to exploit the SRP's Open grade of membership for schools to join the Society and thereby optimise opportunities for better schools engagement.

Conclusions

There can be little doubt that engaging with school-aged students on topics relating to radiation science and radiation protection can be nothing but beneficial to the radiation protection profession and to society in the longer term future. It's an investment that the profession and its learned societies must be prepared to make now.

For each outreach project that it has organised and/or participated in the SRP have documented a full report. These reports together with a paper on "strategies for engaging with future radiation protection professionals" (Cole P, 2015) now form the basis of a resource for information exchange with other worldwide IRPA Associate Societies. In this way, the knowledge and experience gained from these activities can be shared throughout the international radiation protection community and may help further schools engagement projects to be established and run successfully in other countries.

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RADIATION PROTECTION TRAINING IN UPDATED SLOVENIAN LEGISLATION: WHAT IS IMPROVEMENT AND WHAT IS NOT

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Abstract: System of radiation protection training in Slovenia has been designed in legislation approved during harmonisation process prior to joining EU in 2004. Rules related to radiation protection training from the year 2004 were detailed and particular. Courses for exposed workers and two types of radiation protection officers were defined. The first type of radiation protection officers in Slovenia are members of radiation protection units in nuclear and radiation facilities. The second type, which was introduced at the time, was a person responsible for radiation protection in other facilities.
Rules related to radiation protection training were changed last year. Changes were not related to training of workers and radiation protection officers in nuclear and radiation facilities, but to other practices. Majority of courses from previous Rules were shortened, and have become more specific and adjusted to changes in technology, but the implementation of numerous courses for limited number of participants will be complicated and expensive, The most important and, in our opinion, most questionable change is removal of additional training for persons responsible for radiation protection from legislation. Considering the duties and responsibilities of these persons, basic course for exposed workers cannot provide necessary knowledge and skills for their work. Consequently, we can expect that radiation protection safety in facilities will be reduced. It is up to legislator to correct this deficiency in a new version of Rules, which is expected in near future.

KEYWORDS: radiation protection training, programmes of training, exposed workers, radiation protection officers, workers, who manage radiation sources, exposed workers who work under supervision.

1 INTRODUCTION

1.1 Introduction

Before joining EU in the year 2004 Slovenia had to harmonise legislation, also in the field of protection of the health of workers and the general public against the dangers arising from ionising radiation. Council Directive 96/29/EURATOM [1] was implemented in Ionising Radiation Protection and Nuclear Safety Act [2] in the year 2002. In the following years, second-level legislation was approved, also legislation regarding required education and training of exposed workers and other persons involved in the implementation of radiation protection in practice. These are persons responsible for radiation protection, which are appointed by employers and are in fact radiation protection officers. Staff members of the radiation protection units in NPPs, nuclear reactors, and other nuclear and radiation facilities also perform tasks of radiation protection officer. While persons responsible for radiation protection were introduced with the new legislation, radiation protection units were already operational in nuclear facilities for more than two decades at the time.

Legislation from the year 2002 has also introduced authorised radiation protection experts, authorised medical physics experts and authorised dosimetric services according to the definitions of relevant qualified expert, medical physics expert, and approved dosimetric service in Council Directives 96/29/EURATOM [1] and 97/43/EURATOM [3].

Programmes and requirements for radiation protection training of exposed workers and both “types” of radiation protection officers in relevant Rules from the year 2004 [4] were elaborated and particular. There were seventeen different courses defined for exposed workers, while training for persons responsible for radiation protection consisted from a relevant course for exposed workers and additional training dedicated mostly to familiarisation with legislation. Required training of staff members of

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radiation protection units in nuclear facilities was particularly demanding and extensive. Authorised radiation protection experts in authorised institutions should perform training for all these groups.

In the year 2017, the new Rules related to radiation protection training were approved [5]. The Rules were formally still based on the old Act from 2002, and consequently on Council Directive from 1996, but since the legislation based on the new Directive [6] was to a great extent under development at the time (the deadline for national approval of updated legislation was 6th February this year), we can consider these updated Rules as a revision zero of the updated training requirements.

In December 2017, the new Ionising Radiation Protection and Nuclear Safety Act [7] based on the new Council Directive 2013/59/EURATOM [1] was approved in Slovenia. Consequently, all second-level legislation must be updated in the near future. While some Decrees and Rules updated and approved in last two years were already compatible with the new Directive [6] in many details, we think that Rules related to radiation protection training should be considerably updated to provide conditions for effective training of all persons responsible for radiation protection.

2 RADIATION PROTECTION COURSES IN THE NEW RULES

2.1 Overview of courses

Seventeen different types of radiation protection courses for exposed workers and two additional courses radiation protection officers in nuclear and radiation facilities are defined in Rules [5]. Courses are defined for exposed workers and exposed workers, who work under supervision (Table 1), and staff members of radiation protection units in nuclear and radiation facilities (Table 2). In fact, courses are defined also for all other radiation protection officers, i.e. for persons responsible for radiation protection, since for this category, training for “regular” exposed worker training is only required. The difference in relation to a exposed worker is just more demanding conditions regarding prior formal education for persons responsible for radiation protection.

Categories of courses and required duration of training for exposed workers, and workers, who work under supervision are listed in Table 1. In the table, a column related to training of an additional group of workers is integrated. These are so-called workers, who manage radiation sources. Radiation practitioners, who are not categorised as radiation workers, also belong to this group. While it is not expected that annual doses of these workers will exceed annual dose limits for members of the public, since they work with sources (or with the patients), authorities require some kind of “radiation protection awareness” training or training on radiation protection of patients (e.g. for dentistry and bone densitometry in medicine).

Among courses listed in Table 1, there are also two categories for which duration (and also contents) are not prescribed in such details as for other courses. These are courses for “Other practices in medicine and veterinary medicine” (i.e. for practices not contained in other medical courses, including radiology in veterinary medicine), and “Other practices” in industry, research, education, etc. These categories are left “open” and it is up to the authorised radiation protection expert, or authorised medical physics expert to decide on the suitable content and duration of courses. This should be done during the licencing process when assessment of radiation protection and programme of radiological procedures are prepared.

Duration of courses for exposed workers and other mentioned groups of workers ranges from 4 hours (courses for exposure to natural sources) to 40 hours (course for exposed workers in all nuclear and radiation facilities.

Table 1: Duration of radiation protection courses for Exposed workers in Slovenia according to Rules from 2017 [5]

Type of facility/practice		Exposed workers (hours)	Workers, who manage radiation sources	Exposed workers who work under supervision	Change with regard to Rules from 2004.
Nuclear and radiation facilities	NPPs and nuclear reactors	40 h	/	8 h	No change
	Other nuclear and radiation facilities	40 h	/	8 h	No change
Medicine and veterinary medicine	Dentistry and bone densitometry	/	8 h	/	-4 h/0, instead of RP training courses are related only to protection of patients
	Diagnostic radiology	16 h	/	8 h	-8 h
	Intervention radiology	20 h	/	8 h	New (separated from diagnostic radiology)
	Nuclear medicine	24 h	/	8 h	-8 h
	Tele radiotherapy	24 h	/	8 h	No change
	Brachytherapy	24 h	/	8 h	-8 h
	Other practices in medicine and veterinary medicine	*	*	*	Course for veterinary medicine was 16 hours, now depends on RPE opinion.
Exposure from natural sources	Exposure to radon	4 h	/	/	-2 h
	Exposure of air crew	4 h	/	/	-2 h
Industry and other activities	Industrial radiography	36 h	/	/	No change
	Use of unsealed sources (Categories I and II)	36 h	/	8 h	-4 h/0
	Baggage and packages X-ray screening	8 h	/	/	No change
	Field measurement of density and humidity, portable XRF spectrometry, use of HAAS	20 h	/	/	No change
	Use of unsealed sources (Category III)	12 h	/	/	-12
	Other practices	* (at least 8 hours)	8 h	/	-12 or less

* Contents and duration are defined by Authorised radiation protection expert during the licencing process as a part of the document Assessment of radiation protection

In Table 2, categories of courses for radiation protection officers in NPPs, nuclear reactors, and other nuclear and radiation facilities are listed. These are members of radiation protection units. Apart from extensive radiation protection training (200 hours or 80 hours), they should also have the appropriate formal education (they should have first level university degree). As we have mentioned earlier, requirements for training of other radiation protection officers (i.e. persons responsible for radiation protection) are the same as requirements for exposed workers, except that they should also a have first level university degree or equivalent education.

For all groups of exposed workers (also workers, who work under supervision, and workers who manage radiation sources) and both types of radiation protection officers, re-examination is required every five years. For workers involved in the implementation of radiological procedures complete training must be repeated based on the same period.

Table 2: Duration of radiation protection courses for Radiation protection officers (RPOs) in Slovenia according to Rules from 2017 [5]

Type of facility/practice			Change with regard to Rules from 2004.
Nuclear and radiation facilities	Staff members of the radiation protection units in NPPs and reactors	200 h	No changes
	Staff members of the radiation protection units in other nuclear and radiation facilities	80	No changes
All other practices (including outside undertaking in nuclear facility)	Person responsible for radiation protection	Basic course only	-4 hours

2.2 Differences between programmes in new and old Rules

As we have mentioned in the introduction, programmes and requirements for radiation protection training of exposed workers and radiation protection officers in relevant Rules from the year 2004 were elaborated and particular. All courses had to be developed and implemented by just two organisations, who are authorised for radiation protection training in Slovenia. Variety of courses presented a considerable problem for implementation since the number of fresh workers for many practices is limited to a just few per year. Exemptions were courses for some medical practices and courses for exposed workers, and those, who work under supervision in NPP. The situation was better if re-training courses are considered, but the problem with numerous programs was still present. A consequence of these problems was the complicated implementation of courses that required careful coordination of lecturers, and also not the optimal sequence of lectures. In addition, consequence was also higher price of a course for the participant.

Programmes in the new Rules (see Table 1) did not simplify implementation of radiation protection courses. It is already clear that implementation of new programmes will be even more complicated than

the implementation of old ones. The number of programs is similar as in old Rules, but this is only apparent since specific courses for particular practices have been introduced without fixed duration and content. Duration of these courses is marked with an asterisk in Table 1, and, as it is explained in the previous chapter, these courses are left "open" and it is up to the authorised radiation protection expert, or authorised medical physics expert to decide on the suitable content and duration of courses. Although these "specialised courses" must be approved during the registration or licencing process, there is no formal assurance that different courses will be introduced for particular "other practice" without the prior agreement of different authorised radiation protection experts and authorities.

Differences between new and old courses for all groups of workers are marked in the rightmost columns of Table 1 and Table 2. As can be seen from comments, duration of most courses has been reduced except for courses in nuclear and radiation facilities. This is true for all groups of workers in nuclear and radiation facilities, including staff members of radiation protection unit.

Apart from duration of the courses, the most significant change in courses for exposed workers are emphasis on radiation protection of patients in courses for medicine, and emphasis of practical training in courses for industrial radiography, use of HASS, gauges for field measurement of density and humidity, portable XRF spectrometry, and use of unsealed sources (Categories I and II).

The most important change in Rules, in our opinion, is "elimination" of additional training for persons responsible for radiation protection. Required training for this group of radiation protection officers in old Rules consisted of basic training for exposed workers, and additional training (4 hours) related to legislation. This additional training was usually implemented as a short course separate from other courses and did not interfere with the implementation of regular courses for exposed workers.

3 IMPACT OF CHANGES TO EFFECTIVNES OF TRAINING AND RADIATION PROTECTION

As we have seen in the previous chapter, the new Rules have introduced numerous changes in radiation protection training in Slovenia. Although some changes can be identified as related to the introduction of new Council Directive [6], it seems that the majority of changes are the consequence of other reasons.

For example, advances in technology, computer support and availability of different safety systems that could be built in different research equipment (e.g. XRD or XRF devices) has significantly changed requirements for radiation protection training of workers. New devices are inherently safe, doses of workers are well below annual dose limit for members of the public, and numerous interlocks prevent accidental exposure. Therefore, it makes sense to reduce required training and concentrate on essential information regarding the particular device. What information should be delivered depends on the opinion of authorised radiation protection expert. This approach has been chosen for "Other practices" in Industry and other activities type of practices. In this case, courses were reduced to less than half of previous duration (from 20 hours to 8 hours). However, as we have explained before, it is not certain that this approach will be optimal when more participants using different equipment will attend the same course. Separating them will increase costs of training while joining them in one group in this "unified course" will decrease the effectiveness of training. This approach is could be useful in environments where more workers use the same (or similar) equipment, but not in our country, where the number of users is limited.

The same arguments apply to course for unsealed sources (Category III), where the duration of the course has been reduced from 24 hours to 12 hours.

The required length of training has been reduced also for other categories, especially for courses in medicine. Considering the requirement for increased training regarding radiation protection of patients, some new approaches to training of protection of workers must be introduced if we want to ensure the same effectiveness of training in considerably shorter time.

Requirements for training of all categories of workers in NPPs, nuclear reactors, and nuclear and radiation facilities have not changed, which we think is the proper approach. Training is well established and any changes could introduce unnecessary problems regarding compliance with international requirements for nuclear and radiation safety.

Reduction of training of persons responsible for radiation protection to basic training for exposed workers will definitely have the undesirable effect to radiation protection safety in facilities. Although this training is not required in Directive [6] (Article 14, section 3: "*The* M*ember States* ***may*** *make arrangements for the establishment of education, training and retraining to allow the recognition of radiation protection officers, if such recognition is provided for in national legislation.")*, our experience from the past years confirms that training of persons responsible for radiation protection is very important. Since they are always appointed by undertaking (usually by facility manager) and chosen among existing facility workers, their knowledge on radiation protection is consequently limited to knowledge acquired on basic radiation protection courses. However, duties of a person responsible for radiation protection related to the implementation of protection are far beyond duties of any other exposed worker. For example, he/she is *in charge of high safety culture and good condition of radiation protection* [5]. Practically, he is responsible for a number of administrative and operational aspects of radiation protection implementation, among others also for planning and implementation of radiation protection measures, making of written procedures and instruction, and also for monitoring of workplaces (in some facilities). As we can see, knowledge and skills required for the position of a person responsible for radiation protection are not something common among many professions and could be acquired only through specialised training.

Implementation of radiation protection could be simple in some cases, but there is number of facilities and practices where it is highly complex and demanding. If we want to ensure a high level of radiation safety in all these cases, we need to support undertakings with the appropriate and sufficient training of all persons involved in the implementation of a practice, including persons responsible for radiation protection.. Unfortunately, current discontinuation of training of persons could have lasting consequences.

4 CONCLUSIONS

The new Rules on the obligations of the person carrying out a radiation practice and person possessing an ionizing radiation source [5] from the year 2017 have considerably changed radiation protection training in Slovenia. While requirements for training of workers in nuclear and radiation facilities were not changed, duration and contents of required training for other practices were modified. Duration of these courses was shortened, which we have found justified in some cases, but not in all. One of the consequences of this modification is increased number of different courses, which is not beneficial for a small country like Slovenia, which has a small number of exposed workers.

Slovenia has two categories of radiation protection officers. Training of staff members of radiation protection units in nuclear and radiation facilities was not changed, while required additional training for persons responsible for radiation protection in other facilities was removed from the legislation. Considering the duties and responsibilities of a person responsible for radiation protection, we think that this change has not been properly justified and will have negative consequences on radiation safety in facilities. Since new Rules are expected in near future, we hope that this deficiency will be corrected, but current discontinuation could have longer consequences.

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Developing and supporting the next generation of radiation protection professionals in Ireland

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Abstract
Ireland's Environmental Protection Agency, EPA, is the national competent authority for the protection of workers, members of the public and the environment against the hazards associated with ionising radiation and has a role in maintaining, growing and building national capacity in radiation science. Evidence indicates that the current capacity nationally in terms of the availability of skilled radiation scientists is insufficient to meet future staffing requirements for EPA in this field. It is acknowledged that a programme to build radiation research in Ireland is a strategic priority for the EPA in its 2016 – 2020 Strategic Plan under the heading "Implement the EPA Research Strategy and leverage national co-funding and EU funding opportunities to help build environmental and radiological protection research capacity in Ireland and improve the dissemination of research outputs". This paper presents the vision, approach and successes towards reinvigoration of radiation research in Ireland to attract the next generation into this field of science and to achieve the following objectives :

- To stimulate the Irish radiation research community so as to develop national radiation research capacity
- To support high standards and broad horizons in radiation research by facilitating engagement with national and international research groups.
- To address knowledge gaps on radiation matters relevant in Ireland and internationally and aligned with the EPA Corporate Strategy.
- To build expertise and facilitate knowledge sharing by engaging with a network of stakeholders.
- To inform EPA and national policy by addressing the knowledge needs of governmental and non-governmental stakeholders, both nationally and internationally and providing evidence based solutions with an emphasis on continually improving nuclear safety and radiation protection

Keywords ***radiation protection, radiation research, radiaiton protection professionals, next generation***

1. Introduction

Recent efforts by the EPA to recruit new staff for positions relating to radiation protection have shown that the current capacity nationally in terms of the availability of skilled radiation scientists is insufficient to meet future staffing requirements. This concern was acknowledged during the development of the EPA's 2016 – 2020 Strategic Plan [1]and a programme to build radiation research in Ireland was identified as a strategic priority in the plan as a way of developing the next generation of radiation protection professionals. In particular, the plan includes an action to "*implement the EPA Research Strategy and leverage national co-funding and EU funding opportunities to help build environmental and radiological protection research capacity in Ireland and improve the dissemination of research outputs*".

Additionally, 2018 sees the transposition of the Euratom Basic Safety Standards (BSS) Directive (Council Directive 2013/59/EURATOM) [2]into Irish legislation.

The new BSS Directive applies to any planned, existing or emergency exposure situation which involves a risk from exposure to ionising radiation (artificial or naturally-occurring). It covers protection of workers, public and patients. A radiation research programme offers an opportunity to use research to improve regulatory decision making and the development of policies, keep abreast of new

technologies for example, in medicine and supports filling gaps in knowledge in regulatory requirements.

2. EPA participation in research

The EPA has been assigned a statutory role to co-ordinate and support national environmental and radiation research following its merger with the Radiological Protection Institute of Ireland (RPII) in 2014. As a result of the merger, the EPA is mandated under the Radiological Protection Act, 1991 [3] '*to carry out or to arrange for the carrying out of and to co-ordinate or assist in arrangements for the carrying out of research into any matter'* relating to its functions or activities.

The fulfilment of these functions points towards the need to be involved in radiation research. Following the Chernobyl accident in 1986, there was a request by Government for the then RPII to be involved in international research into agricultural countermeasures and to be involved in projects that supported the development of technical expertise in radiation monitoring and assessment. This involvement in research has been clearly beneficial in many ways. For example, in the years following the Chernobyl accident, the RPII participated in several international collaborative research projects. These were hugely important in expanding and developing Ireland's skills base in the areas of environmental behaviour of radionuclides and their transfer through food systems, a key area of interest for Ireland given the emphasis on the agricultural sector. The RPII's, and more recently the EPA's, involvement in research to underpin its environmental monitoring and assessment roles has also allowed it to establish the capability to undertake nationally important projects such as the assessment of new nuclear build in the UK, and to provide credible and high quality advice to Government in this area. It has also provided a solid base to allow it to fulfil its role in relation to Ireland's preparedness for a nuclear emergency and its capacity to respond to such an event.

In addition, the RPII/EPA's work on radon established the scale and nature of the problem in Ireland and developed expertise that was crucial to formulating the advice to Government and the public on this issue.

Collaborative radiation research is particularly important to EPA due to the fact that Ireland is a non-nuclear country and the pool of radiation expertise and the radiation research community is quite small. Participation in research opened up staff access to the wider research community, and allowed EPA to maintain and develop links with colleagues in other agencies and third level institutes both within Ireland and abroad.

With the passage of time since Chernobyl, the radiation research community in Ireland, particularly in environmental aspects of radiation research, has dwindled. The impact of this decline is seen when experienced recruits are sought; this has been the EPA's experience and more recently succession was identified as an issue by the reviewers during the IAEA International Regulatory Review Service [4] peer review carried out in 2015. This decline was also illustrated in the 2017 EPA call for radiation research tenders where no submissions were received in response to calls for two projects, one on radioactivity of Irish building material and a second on radioactivity in the coastal environment.

Research provides a means of keeping abreast of latest developments, provides a basis for providing up to date and sound advice to Government, and sustains capacity to respond as needed in the event of a nuclear emergency. Supporting radiation research in Ireland will nurture a pool of scientists as a national resource in radiation protection. Currently the EPA is funding a number radiation research projects:

- UNVEIL: UNderstanding VEntilation and radon in energy efficient buildings in IreLand(commenced in 2015);
- Investigation into radon mitigation techniques (commenced in 2015);
- Development and Application of Monte Carlo models for HPGe gamma spectrometry (commenced in 2016)

- European Joint Programme CONCERT co-funded: VERIDIC: Validation and Estimation of Radiation skIn Dose in Interventional Cardiology (commenced 2018)
- The European Joint Programme CONCERT co-funded: PODIUM: Personal Online DosImetry Using computational Methods (commenced 2018)

The current EPA corporate strategy recognises the importance to the future of radiation protection in Ireland of maintaining a commitment to radiation research. Being able to attract the next generation of talent is crucial to the future success of any organisation. This is particularly important in the context of the current age profile of the radiation protection experts within the EPA, with many the senior managers due to retire within the next 10 -15 years.

Looking forward, it is clear that research will continue to play an important role in underpinning delivery of the EPA's mandate. The following vision has been developed for radiation research in Ireland:

Ireland will have a vibrant, well-resourced and sustainable radiation research community, with high quality outputs actively addressing knowledge gaps and working towards enhanced radiation safety, and understanding of environmental and health aspects of radiation science.

The EPA will be a key player in delivering on this vision and will do this by stimulating, facilitating and supporting the development of radiation research in Ireland:

- through engagement with the research community and funding authorities nationally and internationally,
- by supporting the co-ordination of access to EU funds, and
- by directly funding a research programme.

3. Delivering on the vision

National research capacity can be understood as a country's ability to produce, debate and use research knowledge and products relevant to their needs. Research capacity building, is thus the long-term, complex processes aiming to enhance these abilities. A wide variety of approaches and interventions can be employed to build capacity.

A three stranded approach to realising the vision at national and European/International level is proposed as follows, Figure 1

Figure 1 Approach to realising radiation research vision

3.1 Strand 1: National Level

The EPA recognises the value of engagement and networking with other sectors and organisations involved in research. Its research programme has formed strong linkages with national and international partners over the past number of years and the research it funds is of significant value to other government departments and state agencies. The objective under this strand is to build on the current suite of radiation research EPA funded projects between 2016 and 2020 in order to grow the radiation research programme under the Sustainability Pillar. This will involve a modest incremental allocation of resources to radiation research.

In order to deliver this objective an EPA-based Radiation Research Co-ordinator has been appointed to work with the EPA Research team to:

- Be a point of contact on radiation research and integrate radiation in existing EPA research activities e.g. dissemination of information, information days, research conferences, updating EPA website.
- Liaise and collaborate with other national funding agencies to identify synergies and ensure a coordinated approach to Irish research funding
- Consult with colleagues in EPA radiation teams to identify knowledge gaps as a basis for future research calls and work with the EPA Research Team to encourage research calls
- Support the EPA Research Team in budget negotiations to secure additional radiation research funding
- Engage and work with other sectors and organisations involved in radiation research.

The EPA's research programme has formed strong linkages with national and international partners over the past number of years and the research it funds is of significant value to other government departments and state agencies.

3.2 Strand 2: European & International Level

Under this strand the ambition was to have the necessary arrangements in place by 2018 to support the Horizon 2020 National Contact Point (Euratom) thereby enabling Irish researchers access to EU research funds.

The EPA has worked with the Irish National Contact Point (Euratom) for European research funding to:

- Identify actions needed to open access to Euratom funding for Irish radiation researchers;
- Develop a collaborative work programme to deliver on these actions;
- Continue participation as a National Delegate to the Euratom Programme Committee - Fission (complementing the Horizon 2020 – The Framework Programme for Research and Innovation) representing Irish views at Programme Committee meetings.

A number of international linkages have been established to promote Irish radiation research in the European research area. By ensuring that Ireland is represented in significant European initiatives such as Horizon 2020, working towards participation Joint Programming Initiatives e.g. CONCERT[1], the EPA will aim to increase the critical mass, reach and impact of Irish radiation research.

3.3 Strand 3: Strategic

[1] http://www.concert-h2020.eu/

The actions under the *Strand 1: National Level* and *Strand 2: European & International Level* will be delivered within the framework of the existing EPA research strategy which currently does not have a dedicated radiation pillar. However, in 2018 it is envisaged the EPA will commence work on the development of a new research strategy for 2020 and beyond. As part of this development work, it is intended to fully integrate radiation research in the new strategy.

As part of this work the EPA Radiation Research Coordinator will:

- Ensure that radiation research is included in the next EPA Research Strategy;
- Consult with Ireland's research community on the strategic direction for radiation research;
- Work to ensure that the necessary funding is made available to support radiation research.

4. Educational Programmes

There are a small number of taught postgraduate courses in Ireland related to radiation protection. University College Dublin (UCD) offers a four-month Professional Certificate in Radiation Safety, National University of Ireland Galway (NUIG) runs a one-year MSc in Medical Physics and Trinity College Dublin (TCD) runs a one-year MSc in Physical Sciences in Medicine, though this latter course is currently under review. While the NUIG and TCD MSc programmes include modules on radiation physics and radiation protection, they are primarily aimed at those working in a hospital environment. A number of staff currently working in the EPA's Radiation Licensing and Inspection Services have successfully completed these programmes and can apply their knowledge and skills to the regulation of the medical sector.

Notwithstanding the existing postgraduate courses referred to above, it is clear that there is no postgraduate training programmes available in Ireland that focuses explicitly on radiation protection. There is also a concern in relation to where the next generation of Radiation Protection Advisers (RPA)[2] will come from. For licensees in the medical sector, the career development frameworks within hospitals, and associated training programmes, will naturally provide for the next generation of medical RPAs. However, for licensees involved in non-medical applications, such as those in the industrial, educational or veterinary sectors, it is not clear where the future generation of RPAs will come from. The establishment of a new postgraduate educational programme would go some way towards providing future potential RPAs with appropriate training in radiation protection. With this in mind EPA has recently initiated discussions with a third level college on the possibility of developing a viable MSc programme in Radiation Sciences.

EPA has an active undergraduate student internship programme in collaboration with the School of Physics, Clinical & Optometric Sciences at Dublin Institute of Technology (DIT), the aim of which is to enrich students' education through hands-on experience, provide a pathway to develop graduate attributes and enhance employability. From an EPA perspective, it offers an opportunity to connect with enthusiastic students who can bring specialist skills, energy and new ideas to the work place. EPA also gives second level students an opportunity to participate in short duration placements.

Nationally the radiation research community is small (less than 100) with research currently being undertaken in EPA, DIT Centre for Radiation and Environmental Science, Focas Institute, Trinity College Dublin's Applied Radiation Therapy and Translational Medicine Institute Research groups and

[2] Licensees holding/using sources of ionising radiation are required to appoint/consult with an EPA-approved Radiation Protection Adviser on all issues relating to radiation protection and complying with the conditions of their licence and relevant legislation.

the National University of Ireland Galway. Although EPA maintains a list of organisations and universities it collaborates with, there is no network of stakeholders in radiation research in Ireland. Groups outside EPA rely on national sources of funding (Science Foundation Ireland, for example) and international (European Union) sources to fund research.

EPA staff are active participants in the Irish Radiation Research Society (IRRS)[3]. IRRS is an all-Ireland non-profit organisation for researchers in the field of ionising and non-ionising radiation. The IRRS aims to promote and advance learning and education in the field of radiation protection and radiation research throughout the island of Ireland.

5. Achievements to date

The EPA successfully completed a number of actions in 2017 aimed at addressing the skills deficit in radiation protection within Ireland, primarily through radiation research initiatives. These include:

- Inviting proposals for two new radiation research projects in the 2017 EPA's Annual Research Call;
- Initiating exploratory discussions with a third level college on the setting up of a taught Masters course in Radiation Protection;
- Visiting third level colleges to explore possibilities for the establishment/expansion of radiation research programmes;
- Continuing to host a PhD research student in the EPA Radiation Monitoring laboratory;
- In collaboration with the Euratom National Contact Point, nominating EPA as the Programme Owner/Manager (POM) for the European Joint Programme Co-fund project CONCERT;
- Hosting the 2017 annual Irish Radiation Research Society's annual scientific meeting in the EPA's HQ, Johnstown Castle, Wexford

The work undertaken to sign Ireland up to the CONCERT programme is already realising benefits, with the successful inclusion of Irish researchers in two co-funded CONCERT projects.

Other actions planned for 2018 include:

- Bringing all stakeholders with an interest in radiation research activities to a workshop to explore potential synergies, supports and future activities.
- Continued engagement with third level colleges to establish and or expand radiation protection education and research capabilities
- Taking opportunities to 'get the message out' e.g. attendance at third level college Careers Day, EPA Research Roadshows

6. Conclusions

As the national competent authority, EPA has a role in maintaining, growing and building national capacity in radiation science. Evidence indicates that the current capacity nationally in terms of the

[3] www.irrs.eu

availability of skilled radiation scientists is insufficient to meet future staffing requirements for EPA. To address this concern, the EPA's current corporate strategy contains an action to implement the EPA Research Strategy and leverage national co-funding and EU funding opportunities. It is further intended to help build capacity by other means such as student placements and developing relevant educational programmes.

Early evidence suggests some success with a past intern joining EPAs Office of Radiation Protection and Environmental Monitoring as an employee and second undertaking a PhD in the radiation research area. A recent review of past second-level student placements at the EPA has shown that a considerable number of these students have gone on to study science, technology, engineering and mathematics (STEM) courses at third level.

It is believed that this approach will go some way in enriching the pool of knowledge and expertise available for addressing Ireland's current and future radiation protection capacity requirements.

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Learning outcomes for the RPO responsible for Dispersible Radioactive Material

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ABSTRACT

In 2015, the Authority for Nuclear Safety and Radiation Protection (ANVS) requested stakeholders to revise the training system for Radiation Protection Officers in The Netherlands. The core of these revisions which derive from the European Basic Safety Standards (EU-BSS) is that the training for Radiation Protection Officers should be application specific. The University of Groningen chaired a workgroup whose task was to formulate the qualification descriptors for the training of Radiation Protection Officers of Dispersible Radioactive Material (RPO-DRM).

The workgroup formulated a graded approach, demanding three different levels of RPO-DRM, the required knowledge and competences increasing as the risk of the application increases. For pragmatic reasons it was proposed to hold to the limits of the former Dutch Guideline for Radionuclide Laboratories (laboratory on B, C or D level), which in any case adheres to the graded approach for regular applications. The three proposed levels were: RPO-DRM B for radionuclide laboratories at B-level, RPO-DRM C for radionuclide laboratories at C-level, and RPO-DRM D for radionuclide laboratories at D level.

Because the required level of knowledge of the RPE is comparable with the level of knowledge needed for the responsibility for a laboratory on level B it was suggested that the RPE training is adequate for the RPO-DRM B and no separate training is needed. Complying with the decreasing risk level, the learning outcomes for RPO-DRM C and D are simplified learning outcomes of RPO-DRM B. The RPO-DRM D learning outcomes are compatible with the former Dutch `level 5B` training, apart from the supervising tasks.

The workgroup recommends using the course for RPO-DRM D as training for the workers in radionuclide laboratories as used to be the case with the former 5B course. Using a well-defined set of learning outcomes like the RPO-DRM D learning outcomes for the instruction of workers prevents uncertainties in the level of knowledge when moving from one institute to another. Besides the graded level of knowledge, the workgroup also strongly suggested that the RPO-DRM could supervise encapsulated radioactive sources for e.g. calibration purposes.

In this contribution, the process of establishing learning outcomes for RPO-DRM in The Netherlands will be discussed along with the main results.

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1. Introduction

The Dutch Authority for Nuclear Safety and Radiation Protection (ANVS) requested the field to revise the training system for Radiation Protection Officers. The core of these revisions which derive from the European Basic Safety Standards (EU-BSS) [1] is that the training for Radiation Protection Officers should be application specific. During the past years, a start has been made on these revisions [2]. As an outcome, the University of Groningen has decided to form a workgroup whose task is to formulate the qualification descriptors for the training of Radiation Protection Officers responsible for Dispersible Radioactive Materials, abbreviated as RPO- DRM. The workgroup consisted of 20 members from 15 organizations and met twice in 2016.

2. Scope of the Qualification Descriptors

The qualification descriptors are meant for the tasks of the Radiation Protection Officers responsible for Radioactive Materials in dispersible form in unlimited quantities. Unlimited here refers to “all permits that relate to radioactive materials in dispersible form, regardless of the licensed activity”. The definition and tasks of the Radiation Protection Officer are given in the Radiation Protection Decree [3]. The qualification descriptors are primarily meant for

- Research, analysis and material research
- Production of radioactive materials in dispersible form
- Human radio diagnostics, radiotherapy and nuclear medicine
- Performance of leakage tests

on the understanding that a RPO-DRM can supervise in the medical sector as long as radioactive materials are not applied to the patient (no direct patient contact). Should this be the case, then the supervisor should have successfully completed a training for Radiation Protection Officer for Medical Applications. The qualification descriptors for RPO-DRM should also be sufficient to function as a Radiation Protection Officer for small calibration sources.

The RPO-DRM can be the responsible party for releasing material, waste, equipment and the performance of control measurements on any residual contamination in the laboratory. The *RPE* is actually responsible for the release of the entire laboratory, including technical facilities outside of the lab such as sewer pipes and ventilation systems. The release or dismantling of rooms and technical facilities (during decommissioning) where there is a risk of activated material also falls under the responsibility of a RPE.

The ANVS has worked on an adjusted system of permits, registrations and notifications as part of the new national Decree on Basic Safety Standards for Radiation Protection (Bbs) and the implementation of the new EU-BSS. The implementation of this project strives for a graded approach, which is to say that the requirements increase as the risk of the application becomes greater.

In light of this, the workgroup is of the opinion that a two- to three-fold division in the level of RPO-DRM is desirable, and for pragmatic reasons it is proposed to hold to the limits of the Directive Radionuclide Laboratories, which in any case adheres to the graded approach for regular applications:

- RPO-DRM B for radionuclide laboratories at B-level (A_{max} = 2000 Re_{inh}*)
- RPO-DRM C for radionuclide laboratories at C-level under the direct repsonsibility of a RPE (A_{max} = 20 Re_{inh}*)
- RPO-DRM D for radionuclide laboratories at D-level under the direct repsonsibility of a RPE (A_{max} = 0,2 Re_{inh}*)

(*: In the Netherlands the quantity Re_{inh} is used for the amount of activity A that leads to an effective committed dose of 1 Sv upon inhalation)

A RPO-DRM will, in many situations regarding radiation protection, work under the direct "responsibility" of a RPE. A RPE generally possesses a broad expertise in the area of radiation protection and functions as the first contact point for the RPO-DRM for incidents, etc. Some RPO-DRMs work alone and occasionally must quickly make a decision based on the relevant radiation risks. In such a situation, the RPE is mostly hired in and has limited tasks as minimally defined by law in the Radiation Protection Decree. More is expected from the RPO-DRM, such as quickly making decisions during incidents. The workgroup believes that the difference between these two situations is mainly a distinction in the basic knowledge of a RPO-DRM B with respect to a RPO-DRM C and D. A solitarily operating RPO-DRM should thus be trained to the RPO-DRM B level.

The EU-BSS states that a Radiation Protection Expert can perform the tasks of a Radiation Protection Officer. Beginning with the assumption that this implies that in the Bbs the tasks from a RPO may be performed by a RPE, there is no reason to formulate separate qualification descriptors for an RPO-DRM Level B – this person should successfully complete the training for a RPE. The workgroup recommends to state explicitly in regulations that the application-specific portion of the training for a RPO-DRM C counts as appropriate (refresher) training in radiation protection for a RPO-DRM B. Summarizing, we assume for the qualification descriptors given here that the RPO-DRM works under the substantive responsibility of the RPE within the organization.

3. Qualification Descriptors / Core Competencies

Two separate documents were produced presenting the qualification descriptors for RPO- DRM C and RPO-DRM D respectively. Both documents summarize the main assignments of RPOs along with the required skills.

The training for a RPO-DRM C is on EQF-level 6. The prerequisites for a course participant will in many cases be a BSc (or just below) with a profile in the exact sciences (physics and health, or physics and technical) from secondary school. The training for a RPO-DRM D is on EQF- level 4 to 5.

The draft qualification descriptors for the basic competencies of an RPO-DRM are grouped in four clusters:

- Core competency 1: The RPO-DRM supervises and enforces (for the applications for which he is responsible) the relevant laws and regulations in the area of ionizing radiation and gives content appropriate advice to the workers and the organization in consultation with the RPE.
- Core competency 2: De RPO-DRM contributes to the appropriate management of an unintentional event or (imminent) incident for the applications for which he is responsible.
- Core competency 3: The RPO-DRM actively works on furthering his own expertise and those of others for whom he is responsible.
- Core competency 4: The RPO-DRM possesses knowledge, skills, attitudes and competencies that specifically apply to radioactive materials in dispersible form.

The core competencies have each been worked out in detailed learning outcomes including a table of keywords for the E&T programs. Learning outcomes for the practical have also been formulated along with recommendations for the assignment procedure.

The nominal training period can vary per educational institute according to the didactic interpretation (schedule, contact hours versus self-study, contact hours versus e- learning/blended-learning, the use of web lectures, etc.), the combination with other courses for RPOs, the entry level of the participants (prerequisites), and the extra packets offered in addition to the minimally required packet. Indicative figures for the training period are given below:

	Indicative length (incl practicals)	Practicals	Professional attitude
RPO-DRM C	10-12 days	2-3 days	1-1,5 days
RPO-DRM D	3-5 days	1-2 days	Not specified

In September 2016, the documents containing the draft learning outcomes for RPO-DRM C and D have been approved by the Advisory Committee on Radiation Protection for inclusion in the new Dutch regulations. The English and Dutch version of the draft learning outcomes will be available through our website http://www.rug.nl/radiationprotection . The learning outcomes have been included in the Regulation Basic Safety Standards Radiation Protection [5] and came into force on the 6th of February 2018.

4. Relation with the old Dutch system of Education & Training

When drafting the qualification descriptors, the workgroup realized that the former Level 4B [6] training is from origin the training for workers who in large part may work independently in radionuclide laboratories. The former Level 5B training had been used by many employers the past decade to train workers who may in large part work independently in radionuclide laboratories. Both Level 4B and 5B experts may even be deployed occasionally as an RPO (currently for sealed sources of limited risk). Consequently there is a large overlap with the old qualification descriptors of the training Radiation Expert Level 4B and 5B [7].

In order to provide employers the possibility to use an acknowledged E&T program for radiation workers (RWs) in the future, the workgroup explicitly recommends the application of the qualification descriptors for the RPO-DRM D to those exposed workers working with radioactive material in dispersible form.

5. Comparison to the Germen Equivalent

The results of the study on the comparison between the Dutch learning outcomes for the RPO-DRM and the German equivalent are presented by J.W. Vahlbruch of the Leibniz University Hanover, Germany, in a separate publication [8].

6. Conclusions

The formulation of qualification descriptors for RPOs responsible for radioactive material in dispersible form contributes to the implementation of the European BSS. Simultaneously, the fact that the qualification descriptors for the RPO-DRM D can also be used as adequate instruction for RWs, facilitates bi- or multilateral comparison of training programs not only for RPOs, but also for RWs.

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Learning outcomes for E&T programs for RPOs responsible for open radioactive sources – A German Dutch Comparison

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Abstract. Building on earlier work the Universities of Hannover and Groningen collaborated in comparing the new Dutch learning outcomes with the current and possible future German requirements for RPOs for open radioactive sources. This bilateral project aimed at providing advice to the Dutch Authority for Nuclear Security and Radiation Protection (ANVS) and the German Bundesamt für Strahlenschutz (BfS) to formulate the final learning outcomes for or harmonizing E&T programs for these RPOs. Furthermore – as the lowest level of these programs will also be suitable for radiation workers (RWs) – the project also aimed at facilitating employers in both countries in mutually recognizing the instruction programs for RWs. The results of the German-Dutch comparison show that the learning outcomes differ primarily with respect to the national legislation. The analysis clarifies especially the impact of German legislation references and explains the thematic equalization of the German-Dutch subjects. Significant expertise discrepancies occur e.g. for the various waste management systems for radioactive material. Furthermore, the comparison of keywords identifies missing topics. All relevant expertise discrepancies are summarized by an additional training advice.

KEYWORDS: Education & Training, harmonization, bilateral comparison, open radioactive sources

1. INTRODUCTION

Already in January 2012 a "Comparison of the lowest level radiation protection courses in Germany and The Netherlands" was published based on a collaboration between the University of Groningen (The Netherlands) and University of Hannover (Germany) [1]. The aim of that report was to identify gaps and accordances between the E&T of persons responsible of RP on a lower level in both countries in order to support a mutual recognition of qualifications in RP and to make a step forward to a harmonisation of E&T in RP. In 2014 the EU-BSS were published [2] and definitions of Radiation Protection Officers (RPOs) and Radiation Protection Experts (RPEs) were introduced. That started a process of reviewing existing national regulations and rules concerning the organisation of RP and E&T of persons responsible for RP. One consequence is that in The Netherlands the E&T of RPOs has to be modified to a more application specific system. In Germany the implementation of the EU-BBS lead to a fundamental restructuring of the system of legislation by introducing a new law, the Radiation Protection Law [3]. Consequently, the ordinances have to be renewed and restructured completely and the Radiation Protection Ordinance [4] and X-Ray-Ordinance [5] will be merged into one new comprehensive Radiation Protection Ordinance. Proposals for this new Radiation Protection Ordinance are not yet published[1] but the process of implementation has to be and will be finished at the end of 2018. Following that process a discussion will start in Germany of how to modify existing expert-knowledge-groups for RPOs and RPEs. Whether the outcome of this discussion will lead to a significant change of the German system cannot be foreseen now. But as the organisation of RP in Germany will not change significantly (the system of so called Strahlenschutzbeauftragte (RPOs

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[1] 10.05.2018

and/or RPEs) responsible for RP in a designated specific area will be preserved), it is likely, that the system of E&T will stay more or less unchanged[2]. In that context a comparison between the learning-outcomes concerning the handling of open radioactive material with lower activity between The Netherlands and Germany is useful in many ways. First, it might help to develop in The Netherlands an E&T-system that considers more the specific applications and as long as the German system does not change significantly it will secondly provide a valuable comparison of learning outcomes to support the mutual recognition of qualifications in RP between both nations.

To support the demands described, the project aimed to reach the following objectives

1. A translation into English of the draft learning outcomes for RPO-DRM in The Netherlands.
2. A description of the expected changes in the current learning outcomes for these RPOs in Germany.
3. Identify gaps between both learning outcomes and formulate advice how to bridge these gaps. This advice will be offered to the competent authorities in relation to mutual recognition of these courses.
4. To make the results available to the whole EUTERP-community as well as to employers interested in mutual recognition of E&T for RWs working with open radioactive sources.

2. METHODS

In The Netherlands a working group with the task to formulate the qualification descriptors for the training of Radiation Protection Officers responsible for Dispersible Radioactive Materials has defined four basic core-competencies:

1. Core competency: The RPO-DRM supervises and enforces (for the applications for which he is responsible) the relevant laws and regulations in the area of ionizing radiation and gives content appropriate advice to the workers and the organization in consultation with the RPE.

2. Core competency: The RPO-DRM contributes to the appropriate management of an unintentional event or (imminent) incident for the applications for which he is responsible.

3. Core competency: The RPO-DRM actively works on furthering his own expertise and those of others for whom he is responsible.

4. Core competency: The RPO-DRM possesses knowledge, skills, attitudes and competencies that specifically apply to radioactive materials in dispersible form.

The core competencies have each been worked out in detailed learning outcomes including a table of keywords for the E&T programs. Learning outcomes for the practical have also been formulated along with recommendations for the assignment procedure.

In September 2016, the documents containing the draft learning outcomes for RPO-DRM D have been approved by the Advisory Committee on Radiation Protection for inclusion in the new Dutch regulations. Learning outcomes were formulated in terms of “knowledge”, “skills” and “competencies” according to the European Qualification Frame (EQF) [6]. The English and Dutch version of the draft learning outcomes are available through [7]. An extended description of the realization of these learning outcomes is given in the contribution of Zandvoort et al. [8]. The learning

[2] At least there will be new Expert-Knowledge Groups and RPOs concerning transportation of radioactive material, the handling of NORM and for member of aircrews exposed to cosmic radiation.

outcomes have been published in the new Dutch Regulation BSS Radiation Protection [9] and came into force 6 February 2018.

In Germany the Guidelines for the requisite qualification for the handling of radioactive material for technical application [10] defines the learning content for the S4.1 Module. However, the definition of the content is only realized with key-words including an estimated amount of time that the lecturer should spent to communicate that learning content. To clarify these learning goals the Working Group E&T of the German-Swiss Association for Radiation Protection (AKA-FS) developed a catalogue with detailed learning outcomes using operators like “to mention”, “to describe”, “to explain” and more [11]. By using these operators the categories “knowledge”, “skills” and “competencies” are taken into account indirectly.

To identify the gaps and the conformities between the learning outcomes of the RPO-DRM D in the Netherlands and the S4.1 Module in Germany a comprehensive table was generated and published in the report [12]. Keywords, teaching extend and learning outcomes connected to an experiment have been coloured differently in that comprehensive table. Learning goals linked to regulations have been investigated more in detail, as differences are expected to be more likely in this subject.

2.1. Analysis of Keywords

As a first step, the learning objectives were compared by focusing on the keywords. If the content differs partially, the differences are marked and integrated as supplements. In general, the German learning outcomes are more detailed, which causes an assignment of several German subjects to one Dutch learning objective. Learning outcomes, which are content-wise identical, are contrasted, too [12]. Table 1 explains shows as an example how differences in the keywords are highlighted in the report:

Table 1: Differences in the keywords describing a learning-content are highlighted in the report

Dutch subject	German subject	Explanation
radiotoxicity equivalent, Re*	-	The Dutch learning objective, marked by a red star “*”, is not included in the German objectives catalogue.
Physics: electromagnetic radiation, (duality)*, wave/particle	Nr. 137 GH Nr. 135 GH	The Dutch learning outcome is more comprehensive than the German counterpart. The content variation is marked by red brackets “()” and a red star “*”.
practical skills in sampling	Nr. 249 OG	To present which German subjects are not included in the Dutch RPO-DRM D, the content of the German learning outcome is presented in the column “Dutch subject” and is coloured in red.
characteristic X-ray radiation with the aid of an example	Nr. 151 GH	The German learning outcome is more comprehensive than the Dutch counterpart. The content variation is supplemented to the Dutch subject. The addition is coloured in red.
-	Nr. 160 OG (1*)	The German subject is a volunteer excursus.

2.2. Analysis of Teaching Extend

In the catalogue developed by the FS-AKA the extend of teaching is defined by the use of two parameters: the qualification level (described by the use of “operators”) and the number of significance. The number of significance rates the importance of a learning-outcome from “1” (more

or less insignificant but nice to have) to “3” (most significant). The qualification level describes how comprehensive the learning outcome has to be implemented and is specified by the use of didactic operators as, for example: “to mention”, “to comment on”, “to estimate” etc. The qualification level in combination with the number of significance is compared with the Dutch Qualification Descriptors that uses the teaching categories “knowledge”, “skills” and “competences” according to the EQF [6] to specify the Dutch teaching extent.

2.3. Analysis of Learning Outcomes combined with an experiment

To compare the practical elements implemented in a RP-course learning outcomes that can be combined with an experimental set-up are coloured in the table, too. In both countries, the course providers are responsible for the application and the arrangement of experiments [10]. However, the comparison in this report is limited and based on the information of the Dep. of Health, Safety and Environment / Radiation Protection Unit of the University in Groningen and the Institute for Radioecology and Radiation Protection of the Leibniz Universität in Hannover.

3. RESULTS

A **comprehensive table** was developed to compare all learning outcomes of the German S4.1 Module and the Dutch RPO-DRM D course. In table 2 an extract of that comprehensive comparison is shown. The complete table is published in the report [12].

Table 2: Comparison of the learning outcomes of the German S4.1 module and the Dutch RPO-DRM D. Differences in the learning outcomes are marked in different colours to highlight gaps, overlaps etc. For more details check the report [12].

Dutch learning outcome	*German learning outcome*	*German didactic operator*	*German RP Module*	*Number of significance*	*Dutch teaching category*		
General					K	S	C
Structure and content of the nuclide chart	Nr. 144	to explain	GH,OG	(1),(2)			
The functional principle of accelerators and plasma equipment	Nr. 136	to explain	GH	(1)			
atom structure	Nr. 134	to describe	GH	(2)	X		
ionization, excitation	Nr. 135	to explain	GH	(2)	X		
proton / neutron ratio	Nr. 134	to describe	GH	(2)	X		
radioactive decay, half-life	Nr. 141	to explain	GH	(2)	X	X	
	Nr. 139	to explain	GH	(2)			
decay formulas and decay constant	Nr. 143	to calculate	GH, OG	(2), (3)	X	X	
mother-daughter connections	Nr. 142	to explain	GH, OG	(2)	X		
specific activity	Nr 140	to explain	GH	(1)	X		
α-, β, γ-decay, electron capture, internal conversion	Nr 142	to explain	GH, OG	(2)	X		
characteristic X-ray radiation with the help of an example	Nr. 151	to explain	GH	(1)	X		
Bremsstrahlung radiation with the help of an example	Nr. 151	to explain	GH	(1)	X		
decay schemes	Nr 142	to explain	GH, OG	(2)	X		

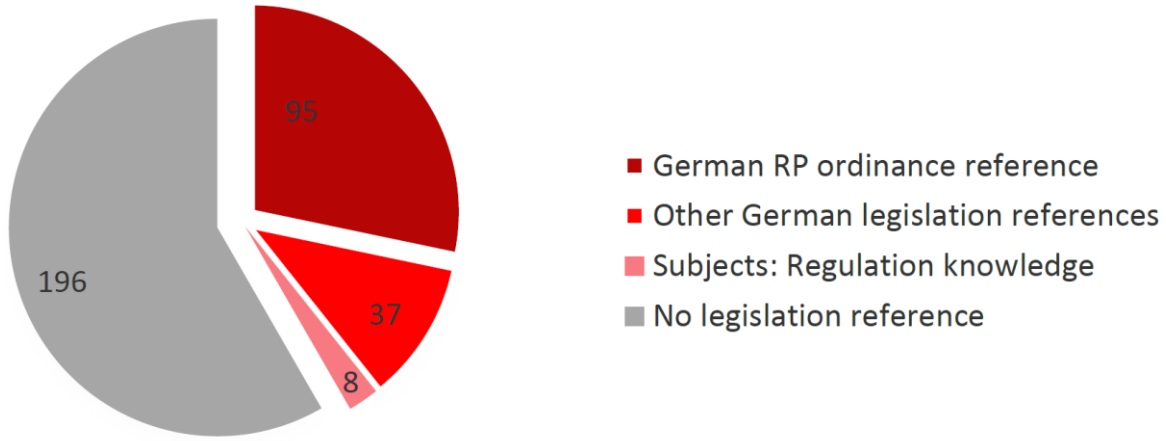
In summary, a comprehensive conformity between the Dutch and German learning outcomes can be observed if only key words are compared [12]. The three topics “maintenance of equipment”, “neutron dosimetry” and “radiation passport” are essential parts of the education in Germany and are not included in the Dutch syllabus. In contrast, “calculation of radiation scattering by objects”, ”p,q,r-formula”, “general ICRP transport model for internal contamination” and “rules of thumb for both β- and γ-dosimetry” as well as the “average energy of β-emitters” are not addressed directly in German learning-outcomes. The “p,q,r-formula” is used to perform a specific Dutch risk estimate when handling open radioactive material and is not generally used or even known in Germany. The

calculation determines the maximum manageable activity of radioactive material inside a laboratory. The formula includes the available ventilation, the laboratory classification, the type of practice and the dose coefficient [13].
The analysis of identical learning outcomes with respect to the extent and to the depth of teaching reveals no noticeable gaps, too. Therefore, the compared learning outcomes are more or less equivalent. However, it is necessary to comment on the correlation between the lecture times and the extent of teaching.

Approximately one-third of all German learning outcomes refer directly to the German RP ordinance or to further regulations. Most German learning outcomes affected by the legislative text do have a Dutch counterpart with respect to the general topic. Though the topics are identical, the content might vary. Concerning the implementation of experiments, at a first glance, the University of Groningen includes more experiments than the Institute for Radioecology and Radiation Protection in Hannover.

As shown in Fig. 1, a considerable amount of German learning outcomes refer directly to the radiation protection ordinance or to other German regulations. As a consequence of the implementation of the EU-BBS the German Radiation Protection Law was adapted and the Ordinances will be renewed completely. Therefore, unfortunately, all the references to the national regulations have to be updated once the process of renewing is completed.

Figure 1: Total distribution of German legislation references respecting all subjects of the modules GH and OG (Module S4.1)



As expected, the content of the learning goals dealing with legislation differs more strongly compared to the others. Especially the German hierarchical structure of RPOs and RPEs have to be explained in detail if the Dutch qualification has to be recognized by the German authorities. The authority of an RPO and the administrative aspects like the specific licensing-procedures need to be exercised, too. The general German waste handling principles are not comparable with the Dutch arrangements. In addition, the official documentation and marking requirements differ considerably, what is intensified by the language differences. An additional instruction program should also compensate these discrepancies [12]. As mentioned, the German subjects “maintenance of equipment”, “neutron dosimetry and radiation passport” are not included in the RPO-DRM D. An additional training that clarifies the fundamental German RP legislation is necessary, too. The implementation of a detailed excursus that focuses on the German legal framework of RPO operations is advisable

The official recognition as a German RPO depends not only on the acceptation of the Dutch certificate by the German competent authority. If the Dutch certificate is accepted to be equivalent to the S4.1 Module, the requirements for sufficient professional education and the sufficient practical experience are necessary to be verified independently [10]. Most of the Dutch RPOs reach the required sufficient

practical experience as well as the sufficient professional education in the same way as German course participants [12].

The RPO-DRM D deals with the p,q,r-formula, the calculation of radiation scattering by objects and various rules of thumb. These contents are not included in the S4.1 Module. An additional training is necessary to clarify these differences together with legislative aspects. Furthermore, a German RPO needs to be instructed by the local Dutch RPE to become familiar with the local established procedures and requirements for the disposal of radioactive waste. The lack of German experiments is roughly compensated by the sufficient practical experience [12].

4. CONCLUSION

The report about "Learning outcomes for E&T programs for RPOs responsible for open radioactive sources – A German Dutch Comparison" compares comprehensively and in detail learning-outcomes between The Netherlands and Germany. It shows that there is a wide overlap in learning-outcomes in these RP-courses. Significant expertise discrepancies occur e.g. for the various waste management systems for radioactive material. Furthermore, the comparison of keywords identifies missing topics. All relevant expertise discrepancies are summarized by an additional training advice. For more details download the full report via www.strahlenschutzkurse.de or via www.rug.nl/radiationprotection.

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Measurements

An analysis of the reliability and validity of smartphones as dosimeters

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Abstract. Researchers have shown dose reductions of up to 81 % when professionals are provided with real-time dose feedback during fluoroscopy. However, the equipment used is often expensive and not widely available. Smartphones have already shown their potential as dosimeters for measuring isotopes and therefore could provide a cheap and accessible alternative. Yet little is known about their reliability and validity in combination with the lower energies and higher dose rates associated with fluoroscopy. This study was set up to analyze this reliability and validity. Tests were performed at three tube voltages, three dose rates, involving five smartphone models, two apps and two professional devices to assess the reliability and validity of smartphones as dosimeters. The results predominantly show percentile deviations of < 5% with relative standard deviations of < 2 % in dose-rates of roughly 0,03 – 1.2 mGy/min. Larger deviations where noted in the directional sensitivity with deviations up to 50 %. The results show the feasibility of smartphones as dosimeters if there is no need for precise measurements such as real-time dose feedback.

Keywords: Dosimetry, radiation detection, real-time feedback, radiation awareness, smartphones,

1 Introduction

Radiation detection has come a long way since the advent of x-ray imaging at the end of the 19th century. This has become more evident since the turn of the 20th century during which digitalization has made medical imaging of ever increasing importance. With this advancement has emerged an emphasis to keep the ALARA principle in high regard to reduce the dose not only for patient and professional alike.

One way to reduce the radiation dose for professionals is increasing dose awareness, for instance by using is real-time dose feedback. Although approaches may vary, real-time feedback increases the professional's dose awareness which in turn influences his or her behavior. During fluoroscopy this can lead to a decreased radiation dose up to 81 %, in turn decreasing the risk of stochastic effects [1 – 8].

Real-time dose feedback is an important opportunity for professional dose reduction, however the devices used are expensive and not widely available. Cheaper devices could lead to a breakthrough in professional dose reduction.

Several researchers have shown that smartphones are able to detect radiation produced by radioactive decay [9 – 10]. This can be achieved either by a peripheral device or the smartphone camera in combination with an app. Because of the worldwide availability of smartphones these could possibly provide breakthrough in real-time dose feedback.

The energy and dose-rate during fluoroscopy however are different than those commonly found in clinical use of isotopes [11]. Little is known about the reliability and validity of smartphones in the detection of scatter radiation during fluoroscopy.

This study examines the reliability and validity of smartphones as a (real-time) dosimeter for detecting scatter radiation from fluoroscopy. The focus lies on differences between smartphone models, apps and professional equipment at different tube voltages and dose rates. The aim is not to replace existing professional equipment but to provide a novel supplement in radiation detection. The results from this study can be implemented for the use of smartphones by (specialized) professionals looking for a cheap and readily accessible dosimeter used for instance to increase dose awareness through real-time dosimetry.

2 Method

2.1 Materials

To determine the percentage deviation a RaySafe X2 with R/F detector was used as a gold standard. During the measurements the R/F detector was clamped just above the smartphone camera.

Philips Dose Aware was included as an comparative instrument to previous publications and its proven implementation as a real-time feedback device[1, 5 – 8].

To assess the reliability and validity of smartphones as dosimeters five models were examined. The models mentioned in table 1 were included. Due to budgetary limitations all models needed to be easily obtained without any costs.

Table 1: Overview of included smartphones as described by the manufacturer

Smartphone	Series number	Rear camera sensor
HTC Desire 500	SH38SWE02601	HTC ImageChip
LG G2	LG-D802	Sony IMX135 Exmor RS
LG G3	LG-D855	Sony IMX135 Exmor RS
Samsung Galaxy S4	GT-I9506	Sony IMX135 Exmor RS
Samsung Galaxy S3 mini	GT-I8190	Unknown

Both GammaGuard (3.0.3) and RadioactivityCounter (1.8) apps were included because of results in previous literature, preliminary tests and independency of peripheral devices.

2.2 Setup and validation

All exposures were performed using a Siemens Axiom Artis DMP system as the source of primary radiation. Scatter radiation was created by using a 15 cm high square stack of Perspex as body equivalent material (hereafter referred to as phantom). This phantom was placed at the center of the headboard of the table with the tube overhead with a focus detector distance of 120 cm and table height of -30 cm.

Validation measurements were performed at the distances described in the paragraphs 2.3 and 2.5. The goal of this validation was to ensure a consistent rebuild and thus dose rate at all distances making comparisons between rebuilds valid.

At the start of the measuring day the smartphone and apps due for that day were calibrated to zero to limit the influence of background radiation and electronic noise. These zero calibrations were performed in the fluoroscopy room after the lens was covered with three layers of black electrical tape and smartphones were set to flight mode. The room itself had no external windows, this ensured a constant natural light level during the experiment which is found to be of possible influence on dose measurements [10]. All zero calibrations were checked by pointing the covered sensor towards a light source. If the lens was correctly covered no photons were detected. In all situations the shielding proved adequate hence no improvements or repeated zero-calibrations were needed.

Both RadioactivityCounter and GammaGuard offer the possibility to adjust the conversion factor used to translate the number of counts to dose. This makes it possible to perform calibrations in addition to the initial zero calibration increasing the validity of the measurements. These conversion factors are relative simple calculations done by the apps and were performed *after* the measurements. This resulted in a non-calibrated measurement and a calibrated measurement originating from the non-calibrated measurement. The calibrations were performed for all three tube voltages used to determine an optimal calibration voltage. Only the calibrated data was used in this publication. Additionally, the

calibration data related to RadioactivityCounter were used as an approximation of the dose rate dependency as four doses were used for calibration of one tube voltage.

2.3 Energy dependency and dose rate setup

The influence of energy dependency and dose rate measurements were measured at distances from the phantom edge of 30 cm, 100 cm and 150 cm. The 30 cm and 100 cm distances represented a close working distance and the common measuring distance for scatter radiation at the University Medical Centre Groningen (UMCG). This increase in distance caused a change in dose rate and scatter radiation pattern thus creating a lower dose rate measurement. To simulate positions of staff at a larger distance and lower dose rate the 150 cm distance was included.

At each of these distances, three sets of measurements were performed at tube voltages of 70 kV, 96 kV and 109 kV to ascertain the energy dependency at a voltage range commonly during fluoroscopy at the UMCG. These energies and distances resulted in a total of nine measurement cases. All exposures used a tube current of 20 mA without additional filtering to minimize tube load over extended periods of use.

Every app measurement ran for a minute and averaged independent measurements over roughly six frames per second. Measurements with professional equipment were timed and averaged to one minute as they do not possess a timer ability. All measurements were performed fivefold (referred to as a set) to determine a mean and standard deviation (SD). This set size was chosen as the minimum amount to ascertain a somewhat reliable average. Larger sets were not feasible to realize within the available time frame. Smartphones were not moved during these sets as this could influence measurements.

2.4 Energy dependency and directional sensitivity setup

Directional sensitivity was assessed using the same measurement setup as before at 100 cm. The measurements were done in fivefold over a total of six angles (0°, 45°, 90°, 180°, 270° and 315°) using all smartphones and Philips Dose Aware. These angles are chosen to represent the possibility of smartphone being turned away by while measuring or being worn as a dosimeter with the sensor initially pointed to the radiation source. The 180° was included to assess the smartphone being pointed away from the source. Energy dependency was also assessed using the before mentioned tube voltages of 70 kV, 96 kV and 109 kV.

The 0° measurement was also performed during the energy sensitivity and dose rate measurements. The two resulting set populations were significantly tested to determine the effect of repositioning.

Due to time constraints, only one app was used for the energy dependency and directional sensitivity measurements. RadioactivityCounter was used as it showed the lowest standard deviation in the energy dependency and dose rate measurements, this ensured the most reliable app was used.

2.5 Data

Reliability and validity were identified through the percentage deviation, SD and relative standard deviation (RSD) from the measurements. Statistical significance for repositioning was determined using SPSS 23 to perform a paired sample t-test. Due to the novel approach and limited size of the sets more in-depth statistical analysis was not relevant.

3 Results

3.1 Total measurements

Both setups resulted in a total of 2493 measurements included in this publication.

3.2 Setup validation and repositioning

The validation measurements performed with RaySafe X2 showed a standard deviation between 0,000 mGy/min to 0,034 mGy/min. The RSD ranged from 0,00 % to 5,05 % as shown in table 2.

Table 2: Mean dose, standard deviation and relative standard deviation resulting from the validation measurements done on at the start of each day's measurements.

	30 cm			100 cm			150 cm		
	70 kV	96 kV	109 kV	70 kV	96 kV	109 kV	70 kV	96 kV	109 kV
Mean dose (mGy/min)	0,46	1,00	1,32	0,07	0,15	0,20	0,03	0,07	0,09
SD (mGy/min)	0,016	0,027	0,034	0,000	0,005	0,005	0,000	0,000	0,005
RSD (%)	3,41	2,68	2,59	0,00	3,07	2,32	0,00	0,00	5,05

Comparison of the 100 cm 0° measurements from both setups showed no significant difference between the measurements in regard of repositioning (p=0,586).

3.3 Smartphone and app dependency

The RSD resulting from the measurement sets shows values between 0,11 % and 13 %, the lower the dose rate the higher the resulting RSD's. Mean RSD per smartphone are shown in table 3 for both apps.

Table 3: Mean RSD (%) (SD) for RadioactivityCounter and GammaGuard per smartphone model

	RadioactivityCounter	GammaGuard
HTC 500	1,47 (0,58)	4,98 (3,71)
LG G2	1,12 (0,95)	2,76 (2,02)
Samsung Galaxy S4	1,51 (0,64)	1,56 (0,82)
LG G3	2,35 (1,10)	2,69 (1,40)
Samsung Galaxy S3 mini	0,45 (0,19)	2,47 (0,98)

[α] These values are averaged over all distances and energies.

The mean percentage deviations per set ranges from 0,38 % to 53 %. In many cases the calibrated measurements showed comparative results to Philips Dose Aware as can be derived from tables 4 - 6. It should be noted that these tables contain overall mean data from all tube voltages and dose rates. This doubly processed data shows relative high deviations rarely found in the initial processed data. A sample of the original data shown in table 7 shows that the large deviations in tables 5 and 6 result from mismatching the calibration energy with the exposure energy.

The data from the 96kV calibration energy show the lowest overall mean percentage deviations in most cases as shown in tables 5 - 7.

Table 4: Mean measurements (mGy/min) and standard deviations from professional equipment for all measurement circumstances. [α]

Raysafe X2 (RF-detector)		Philips Dose Aware		
Overall mean (SD)	Mean RSD (SD)	Overall mean (SD)	Mean RSD (SD)	Mean %dev (SD)
0,40 (0,46)	0,81 (1,06)	0,48 (0,58)	1,29 (0,48)	17,59 (7,13)

[α] These values are averaged over all distances and energies.

Table 5: Overall mean percentage deviations and corresponding SD from Raysafe X2 (RF) for RadioactivityCounter per smartphone model. [α]

	RadioactivityCounter (%)		
Calibration energy	70 kV	96 kV	109 kV
HTC 500	10,88 (8,19)	-2,48 (8,05)	-6,40 (8,86)
LG G2	51,75 (33,86)	-5,51 (18,87)	-14,22 (16,25)
Samsung Galaxy S4	1,54 (0,68)	-3,97 (18,07)	-12,14 (15,77)
LG G3	47,28 (32,02)	-7,03 (21,24)	-16,76 (17,05)
Samsung Galaxy S3 mini	9,58 (6,02)	-1,28 (7,07)	-3,35 (6,89)

[α] These values are averaged over all distances and energies.

Table 6: Overall mean percentage deviations and corresponding SD from Raysafe X2 (RF) for GammaGuard per smartphone model. [α]

	GammaGuard (%)		
Calibration energy	70 kV	96 kV	109 kV
HTC 500	4,68 (9,06)	- 2,84 (8,41)	-6,62 (8,08)
LG G2	37,90 (36,71)	-12,48 (23,30)	-23,38 (20,40)
Samsung Galaxy S4	32,02 (20,81)	-4,31 (15,08)	-11,87 (13,89)
LG G3	53,74 (31,25)	12,17 (22,80)	6,5 (21,67)
Samsung Galaxy S3 mini	8,70 (5,92)	0,38 (5,46)	-1,95 (5,34)

[α] These values are averaged over all distances and energies.

Table 7: Mean percentage deviation resulting from RadioactivityCounter measurements calibrated using different tube voltages (shown per rows) using an HTC 500.

	30 cm			100 cm			150 cm		
Calibration energy	70 kV	96 kV	109 kV	70 kV	96 kV	109 kV	70 kV	96 kV	109 kV
70 kV	0,01	11,68	14,15	-0,62	13,29	16,50	1,33	17,84	23,74
96 kV	-10,75	-0,03	2,24	-13,82	0,25	3,46	-15,17	2,76	8,74
109 kV	-13,06	-2,30	-0,03	-18,52	-3,17	0,29	-22,86	-2,25	4,28

3.4 Energy dependency

The linear graph slopes vary from 4 to 11 with differences between smartphones and professional equipment. Only minimal differences were noted between apps. The lowest slope is resulting from the Raysafe X2 (RF) while comparable results are seen between the HTC 500, Samsung Galaxy S3 mini and Philips Dose Aware.

3.5 Dose rate dependency

The conversion graphs used to calibrate RadioactivityCounter show a coefficient of determination of R^2 = 1,00 showing a linear link between the measurements and dose rate.

3.6 Directional sensitivity

The directional sensitivity measurements resulted in large deviations from the moment when the smartphones were turned away from the scatter source as shown in table 8. The 45° shows a large difference between all smartphone models in regard to Philips Dose Aware which only shows a minor deviation at this point.

Two possible errors were noted for the Samsung Galaxy S3 mini at 315° and the Samsung Galaxy S4 at 45°, shown in table 8 by the large SD. Due to time constraint these could not be retested.

Table 8: Average percentage deviation and corresponding standard deviation relative to the 0° measurement averaged for 70 kV, 96 kV and 109 kV.

	Angle					
Model	0°	45°	90°	180°	270°	315°
HTC 500	100 (0)	27,4 (4,8)	14,2 (2,9)	61,0 (2,4)	11,0 (1,5)	26,7 (5,3)
LG G2	100 (0)	32,5 (5,2)	16,9 (2,7)	81,1 (9,5)	10,6 (1,8)	16,7 (2,9)
Samsung Galaxy S4	100 (0)	61,0 (15,7)	13,7 (3,9)	86,1 (7,9)	13,1 (3,3)	31,1 (7,7)
LG G3	100 (0)	40,6 (6,0)	13,0 (3,4)	76,5 (8,3)	7,8 (1,7)	31,3 (4,2)
Samsung Galaxy S3 mini	100 (0)	21,5 (7,0)	9,6 (2,5)	52,0 (5,9)	9,4 (2,6)	36,3 (29,7)
Philips Dose Aware	100 (0)	109,4 (1,7)	9,9 (3,6)	83,8 (19,9)	10,8 (11,6)	96,6 (1,1)

4 Discussion

This study shows the potential use of apps as real-time dosimeters. Results show an optimal percentage deviation in comparison with the golden standard of roughly 10 % with a RSD of roughly 5 %. The percentage deviation is comparable with results from previous research which have a of deviation roughly 10 % - 20 %. However, the RSD is clearly lower than the 15 - 300 % reported by previous research, which could be explained by the higher dose rate used in this research [9,10].

The RSD ranges between 0,11 % and 13 % with a mean of 2,20 %. The higher RSD are, as expected, measured at the lower dose rates. This does surpass Philips Dose Aware, which has a maximum RSD of 2,37 % and the gold standard with a maximum of 3,67 %, respectively 1,29 % and 0,81 % mean RSD's.

Both the LG G2 and LG G3 seem to be the least valid measuring instruments when looking at the overall mean percentage deviations in table 5 and 6 both resulting in a roughly 23 % overall mean percentile deviation compared to roughly 5 % for the other models. However, looking at the initial means this is somewhat nuanced. The height of the overall means in table 5 and 6 is a result of the larger deviations when mismatching the calibration voltage with the tube potential. This is most apparent for both the LG G2, LG G3 and, in a minor degree, the Samsung Galaxy S4 implying an overall higher energy sensitivity than other smartphone models and the gold standard. This is further implied by looking at the graph slopes, where the above-mentioned models show the highest slopes. Even though these models use the same image sensor, deviations between these measurements show that a universal calibration seems a non-feasible option as mentioned by others [12].

It is difficult to exclude any influence from dose rate because the gold standard is also influenced. However, when comparing the measurements per smartphone with the gold standard, as done in the calibration of RadioactivityCounter, all models show a linear effect within the tested dose rates. It should be kept in mind that if higher dose rates were used it is possible that a ceiling effect could be reached even though this research does not show this effect for RadioactivityCounter and GammaGuard. This ceiling effect could be partially circumvented by adjusting the framerate of the apps if the user is able to anticipate the dose rate. Increasing the framerate does however influence processing power needed, which generates heat and decreases battery life.

One of the biggest drawbacks of using apps as dosimeter is the clear directional sensitivity. Even though all smartphone models show a roughly comparable directional sensitivity they have a large deviation from Philips Dose Aware. When using smartphones as a replacement for Philips Dose Aware this aspect should be considered as turning 45° from the radiation source itself leads to a roughly 50 % decrease in registered dose rate. The primary bundle energy does have an influence, it is however negligible to the influence of the angle from the source. Other research has also shown that directional sensitivity is an issue with smartphones with a possible difference between Android and Apple models [9, 12]. The results in these studies show lower deviations than was found by this research with a directional sensitivity between 10 % - 30 % from the 0° measurement [9,12].

5 Conclusion

This research shows differences between smartphones when measuring scatter radiation in a fluoroscopy setting. These differences are nonetheless easily overcome by calibrating the applications used which is a clear advantage over both RadioactivityCounter and Gammaguard.

After calibration, the differences in measurements between smartphone models and apps becomes minimal. Both the energy sensitivity and dose rate sensitivity differ between smartphones and professional equipment alike, this does however also have a minimal impact.

Smartphones in combination with apps can be used as a novel supplement for real-time dosimetry with an acceptable to good reliability and validity. However, the directional sensitivity, differences between smartphones and between apps should be considered. Specialized professionals with enough knowledge and understanding of these drawbacks should be the primary user group. Use by laymen or professionals with limited knowledge could easily lead to misconceptions which could either lead to unnecessary worries or a false sense of security.

RadioactivityCounter is the most reliable and valid application as shown by the results, with minimal difference from GammaGuard. Looking at the smartphones the Samsung Galaxy S3 mini has the best results. Differences in measured doses between smartphone models exist but these seem to be minimal.

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A Rapid, Robust and Sensitive Monitoring System for the Quantification of Pulsed Ionizing Radiation at CERN

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Abstract. CERN designs and operates large powerful accelerators to probe matter and to understand the fundamental laws of the universe. Their operation generates stray radiation in local areas due to the interaction of high energy particles with surrounding stable matter. Accurately measuring the dose rate of such generated mixed radiation behind shielding requires very specific instruments capable to detect various particles (neutrons, pions, muons, electrons, photons, protons …) through wide ranges of energies and temporal dynamics. This paper gives a general description of the newly designed Cern RadiatiOn Monitoring Electronic (CROME) system and focuses on the detection part. Conceived and manufactured for radiation protection, CROME is a novel solution for radiation assessment capable of measuring very low dose rates down to the natural background, whilst being able to measure radiation over an extensive range of nine decades without auto scaling. This paper further concentrates on the analytical analysis of the CROME systems front-end response and the way used to establish the equivalent ambient dose rates. On the one hand, it addresses the modeling of the electro-physical phenomenon used to measure mixed ionizing fields and on the other hand, it describes the transfer function used to quantify pulsed ionizing radiation. Finally, we demonstrate the measurement capabilities of the system in response to both static and dynamic stimulus. In order to qualify the CROME system, we performed two characterization campaigns. The first focuses on static tests with the injection of continuous current while the second focuses on the dynamic response of the monitor.

KEYWORDS: Radiation monitoring system, Pulsed ionizing radiation, Modeling of linear system, Pseudo Random Binary Sequence.

1 INTRODUCTION

CERN (European Organization for Nuclear Research) uses the world's largest and most complex particle accelerators chain to study the basic constituents of matter. The accelerators boost beams of particles to high energies before they are made to collide with each other or with stationary targets at almost the speed of light. This process gives physicists information about how the particles interact and as a side effect generates various types of mixed radiation. This secondary field, known as stray radiation, differs significantly from the ionizing radiation produced by the nuclear industry in terms of:

- composition (coexistence of charged hadrons, neutrons, leptons and photons)
- energies (~eV to GeV)
- temporal dynamic (pulsed radiation fields)

Hence, since the very beginning, it was necessary for the CERN radiation protection group to take such specificities into account and to upgrade its instruments continuously and especially in the light of the construction and the operation of the LHC. This upgrade phase led to RAMSES (Radiation Monitoring System for the Environment and Safety) system, which was commissioned for the start of the LHC in 2008 and complements the previous ARCON (Areas Controllers) system which is in operation since 1989. In this context, the CERN radiation protection group launched the in-house development of a new generation of ionizing radiation monitoring system four years ago. This system would be capable of fulfilling the required performances in terms of measurement, reliability and

safety as well maintainability for the next couple of decades. This new system is called CROME, standing for CERN Radiation Monitoring Electronics. In its most basic configuration, the system is composed of an ionization chamber and a novel electronics measuring the output charges, generating alarms and interlock signals, local alarm units, all powered by uninterruptible power supplies to guarantee a continuous monitoring in case of loss of power for up to two hours [1].

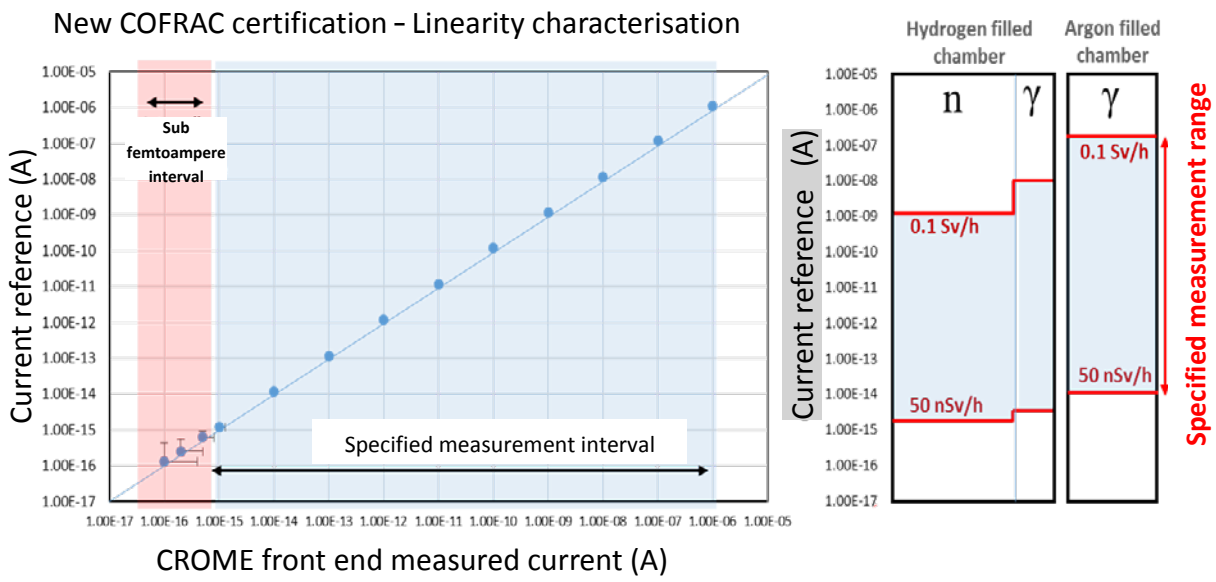
2 STATIC DETECTION PERFORMANCES

Amongst other capabilities, the CROME system is able to measure both ultra-low doses rates, down to 50nSv/h, whilst also being able to track high and fast pulsed ionizing radiation without any change in gain. Any auto scaling could imply a loss of charges and therefore result in an underestimation of the dose rate. The detection principle is based on an ionization chamber [1] that delivers electric charges when exposed to ionizing particles. Hence, for a dose rate interval of [50nSv/h 0.1Sv/h], the conversion of the delivered charges to electrical current gives:

- [80 fA – 160nA] for an Argon filled chamber in a gamma field (Cs-137)
- [5.7 fA – 11.3nA] for a Hydrogen filled chamber in a gamma field (Cs-137)
- [2fA – 4nA] for a Hydrogen filled chamber in a neutron field (AmBe)

In this paper, the static characterization of the CROME monitor focuses specifically on the read-out electronics. The ionization chamber has been extensively characterized in previous work. Its response to various ionizing particles has been both simulated [2][3] and tested throughout wide range of energies.

Figure 1. CROME front-end electronic static characterization results. (a) : Front-end linearity over the measurement interval. (b) : Conversion of the current into dose rate on the measurement interval

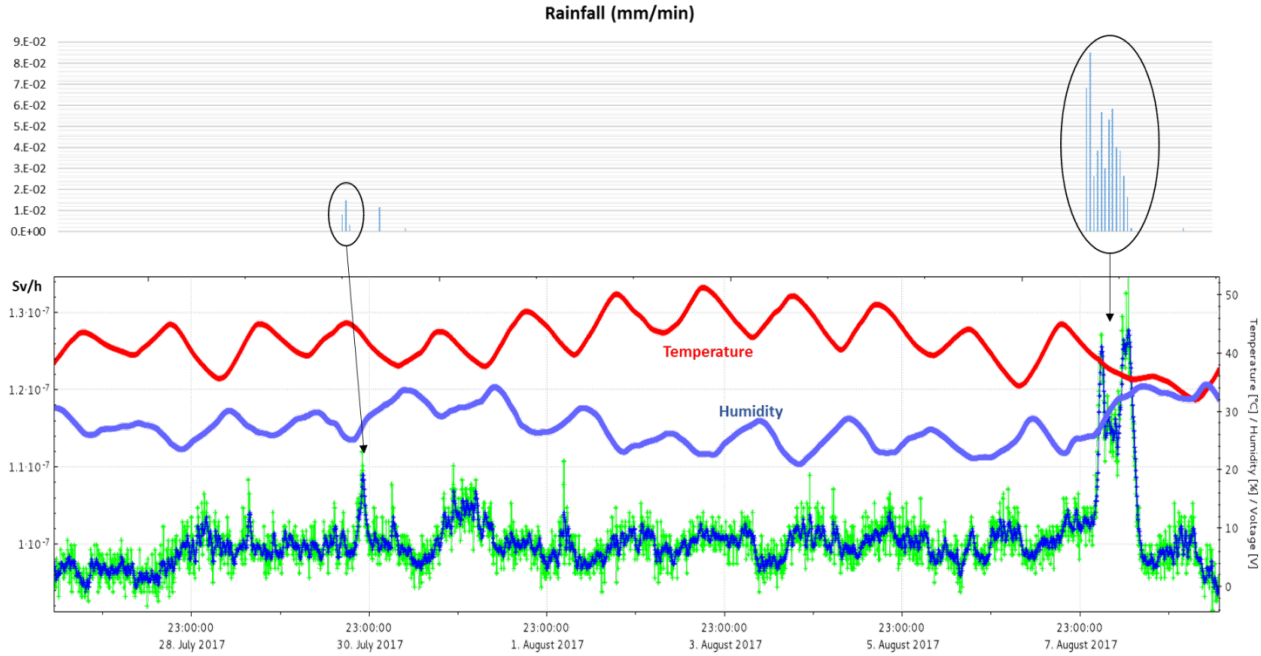


(a) (b)

The new characterization campaign of the analogue front-end has been performed by a certified laboratory [4] in order to validate the performances of the electronic design. Injection of multiple calibrated continuous currents has validated the CROME front-end measurement precision over a range of nine decades: from the femto-amperes to the micro-amperes. As shown in Fig. 1, the system has demonstrated a very good linearity on the specified measurement interval. When it is used for environmental monitoring, a calibrated CROME monitor has shown, under real climatic conditions, very good performances in terms of both thermal stability and measurement accuracy (Cf. Fig. 2). For

example, the system was able to accurately measure the contribution due to the deposit of radionuclides present in the air to the ground during rainfall. As a consequence, the radiation background raised as shown in Fig. 2 from 90nSv/h up to 128nSv/h.

Figure 2. Measurement of ambient dose rate in the environment correlated with rainfall



3 DYNAMIC DETECTION PERFORMANCES

The main challenge for the front-end electronics is to manage the collection of some thousands of electrons per second whilst being capable to measure and track high dynamic currents without any change in gain. In order to estimate the dynamic behavior of the CROME monitor, we built an experiment aimed at modulating the radiation level seen by the detector (Cf. Fig.3 (b)). To correctly model the studied system, we need to stimulate the entire frequency bandwidth that could contain the system time constants. Mathematically speaking, the best approach would be the use of the Dirac delta function as a stimulus. This theoretical approach is however impractical. An alternative method consists of the generation of a pseudo random binary sequence (PRBS). Composed of N bits with a length $L = 2^N - 1$, it could be generated using various algorithms [5][6]. The simplest one uses a shift register on N bits and a feedback XOR (Exclusive OR) gate. The spectrum of the generated sequence is calculated using a Fourier transform of its auto-correlation function:

$$C_{xx}(t) = \frac{-a^2}{L} \tag{1}$$

where a is the amplitude of the signal.
The analysis of the spectrum (Cf. Fig. 4 (c)) shows that the signal can be considered as white noise in the interval between $[0\ \frac{1}{3T_e}]$. Where $T_e$is the sampling time of the PRBS.
Hence, in order to stimulate the system's smallest time constant τ_{min}, we should generate a PRBS with a frequency equal to :

$$\frac{f_e}{3} = \frac{1}{2\pi\tau_{min}} \tag{2}$$

For a good identification of the static gain, the length of the maximum pulse duration should satisfy the condition:

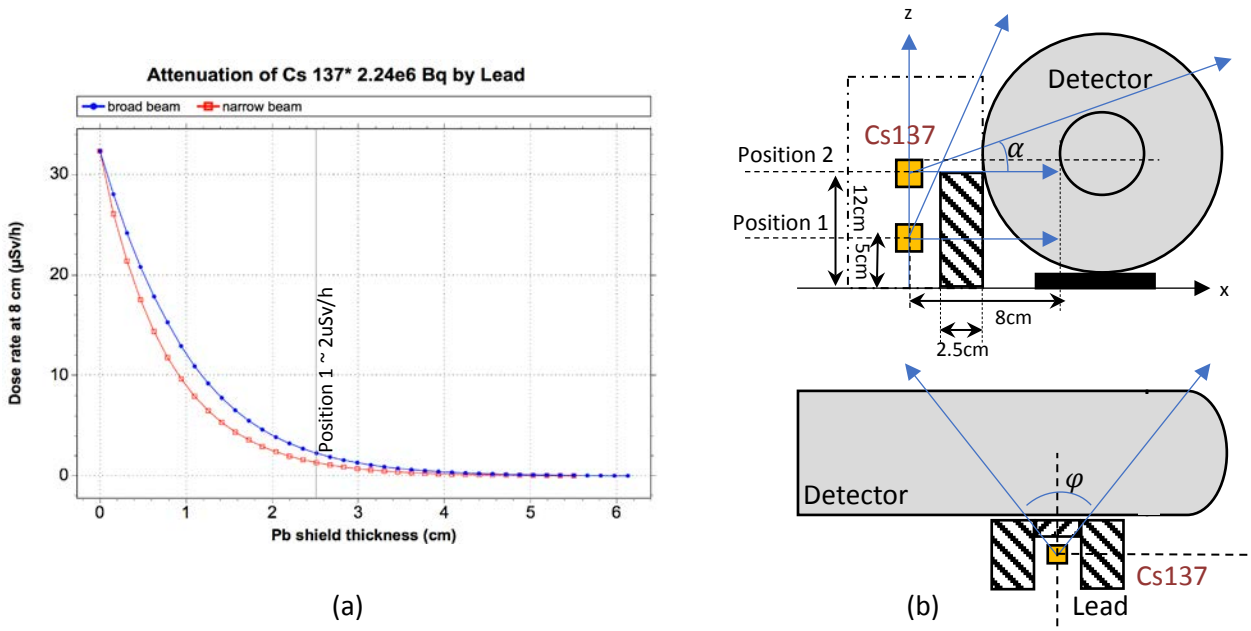
$$\frac{N}{f_e} > 3\,\tau_{max} \qquad (3)$$

Where τ_{max} is the largest time constant of the system.
Finally, the amplitude of the excitation signal should be 2 or 3 times higher than the intrinsic noise of the system.

3.1 Experimental setup

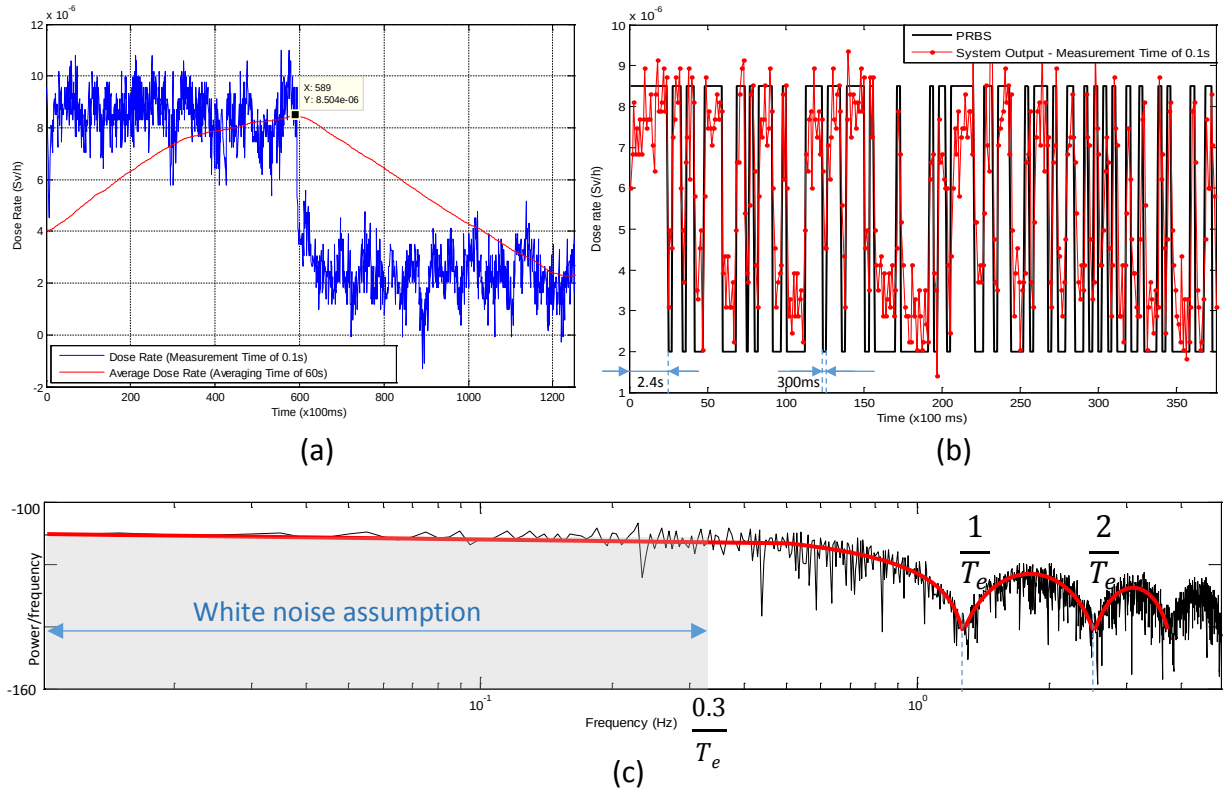
Using a Cs137 source attached to an electro-mechanic actuator, we created a linear translation motion controlled by a computer in order to alternate the radioactive source between position 1 and 2 as presented in Fig. 3 (b). When the source is in position 1, the effective active zone of the detector is protected in the X axis by a lead shielding of 2.5cm. The calculation [7] with a source of 2.24MBq gives a dose rate of ~2.3μSv/h (broad beam – Cf. Fig. 3 (a)) at 8cm with: $\dot{K}_{air} = 1.928\,\mu Gy/h$, $\dot{X} = 5.677.10^{-8}\,C/Kg/h$, $B = 1.705$ and $\mu.d = 3.169$. This dose rate value has been chosen because it is bigger enough than the intrinsic noise of the "open" front-end electronics when it is configured to a measurement period of 0.1second (Cf. Fig. 4.a). When the Cesium source is in position 2, the PRBS amplitude "a" is equal to 6.5μSv/h.

Figure 3. Experimental setup. (a): Attenuation of Cs137* 2.24MBq by Lead at a distance of 8cm. (b): Principle of the experimental setup



The response time of this experiment is limited by the reaction time of the electro-mechanic actuator which is at the order of 200ms. In order to avoid any nonlinear effect due to the setup limitation, we fixed the period of the PRBS to 300ms. According to equation (2), this value limits the modeling of the studied system to its time constants that are slower than 143ms. To optimize the time duration of the experiment we limited N to eight bits. With this configuration, the maximum alternation period of the Cesium source between the two positions is equal to $8 \times 300ms$ ($2.4s$). In the other hand, the minimal period is equal to 300ms (Cf. Figure 4(b))

Figure 4. Pseudo Random Binary Sequence. (a) : Amplitude of the generated PRBS. (b) : CROME monitor response to a complete PRBS of Te = 300ms presenting a total length of 127 states. (c) : Spectral response of a PRBS – Te = 800ms



3.2 Experimental results

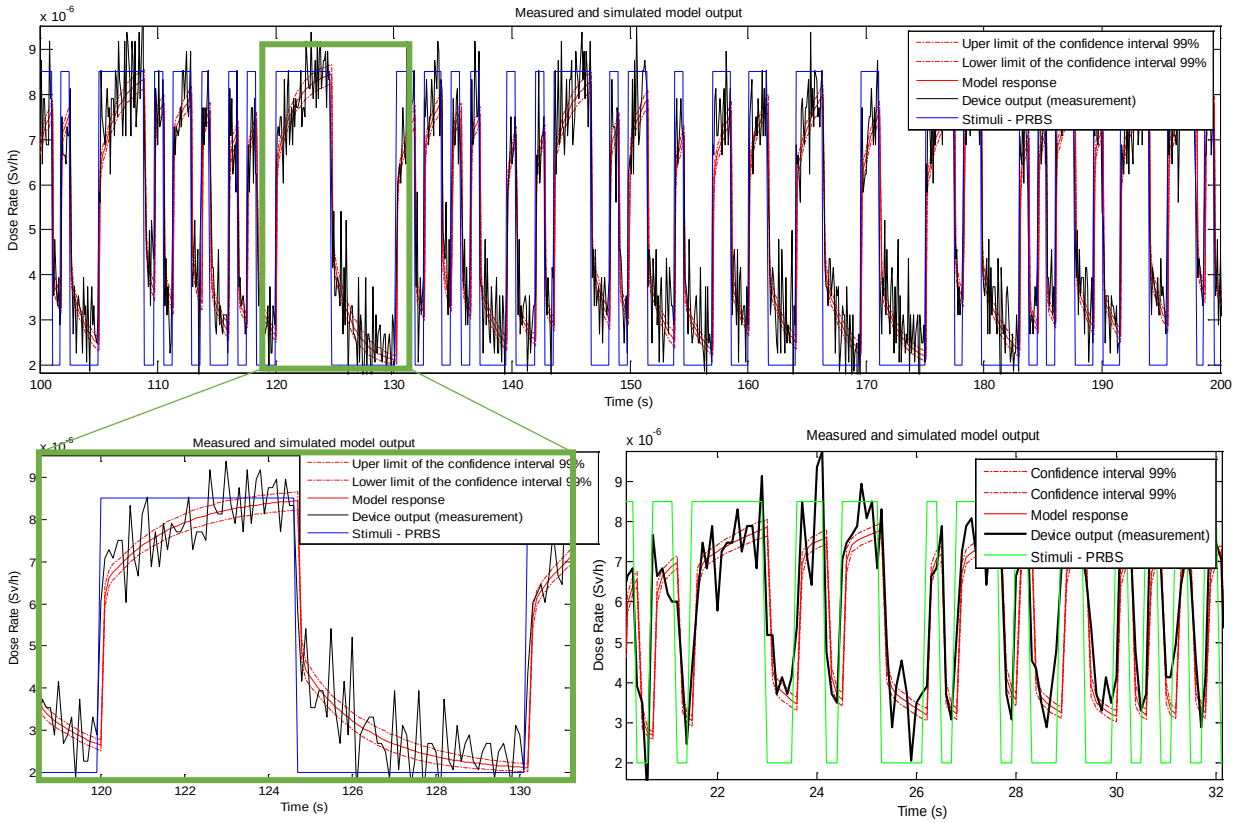
As shown in Fig.4 (b), the response of the monitor to the excitation is quite fast. Various tests and analysis with different sequences lengths and frequencies (Cf. Fig 5) have led us to a second order space state model described by the following equations:

$$\begin{cases} \dot{x}(t) = \begin{bmatrix} -2.973 & 8.815 \\ 5.58 & -21.41 \end{bmatrix} x(t) + \begin{bmatrix} 1.06\ 10^5 \\ -2.39\ 10^5 \end{bmatrix} u(t) + ke(t) \\ y(t) = [6.67\ 10^{-5} \quad -2.87\ 10^{-5}]\, x(t) + e(t) \end{cases}$$
(4)

Where x is the state vector, u is the control input and corresponds to the ionizing radiation, y the measured dose rate, k the disturbance matrix and e the noise. We notice that the system is asymptotically stable with two strictly negative poles given by the eigen values of the state matrix. Their influence can clearly be seen in the temporal response of the monitor. We have then two time constants:

- ~0.3s which corresponds exactly to the period of the first harmonic contained into the excitation signal. This value corresponds then to the lower limitation value of our experience. In other words, we cannot do better
- a slow time constant of ~3s

Figure 5. Model response vs measurement. (a) : Response of the model correlated to the measured one for a PRBS excitation signal (L=254 and Te=800ms) . (b) : Response of the model Vs measurement for an PRBS excitation signal (L=128 and Te=300ms)



(a) (b)

For a slow PRBS presenting a minimal period of 0.8s, the model has a fit ratio of 71%. When a new PRBS is generated with the minimal timing of 300ms, the model fits only at 50% (Cf. Fig. 5 (b)). Due to the limitation of the mechanical setup and the refresh rate of the electronic it is very hard to clearly characterise the fast pole of the system. It seems actually clear from Fig. 5 (b) that the system is faster than 0.3s.

On the other hand, the slow pole is well defined and we can clearly see in the zoom of Fig. 5 (a), its effect on the stabilisation time (steady state). When the output reaches 60% of the final value, the second pole becomes dominant and slows down the reaction time of the monitor. At this stage it is very difficult to determine if this behaviour is either due to the detector or to the front-end electronics.

4 CONCLUSION

The CROME System is a complex radiation instrumentation equipment capable to autonomously monitor levels of ionizing radiation in order to either trigger alarms, interlock signals or both. Thanks to an embedded intelligence, the system does operate autonomously based on various configurations and parameters including the beam status, the access system status or based on other sensors. Depending on the operational constraints around the accelerator complex, the CROME system can be used to monitor either low levels of radiation in the environment or high level pulsed radiation in accelerators facilities and experimental areas. In order to fully understand the dynamic behaviour of the CROME detection system, complementary tests are planned using a different setup in order to fully characterize its fastest time constant.

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Investigation of Bayesian Statistical Techniques at the Decision Threshold

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Abstract. Statistical analysis of a radiation measurement traditionally relies upon the use of the frequentist (Classical) statistical test. In the case of radiation measurements, a statistical analysis is carried out on the assumption that the calculated background distribution comprised of a series of measurements is accurately representative of the true background distribution. Use of the frequentist statistical test for radiation detection and measurement arises from the necessity to separate signal from background. ISO 11929 defined the decision threshold, y^*, which is considered an investigatory level and does not provide any information regarding detection capabilities; the detection decision is purely based on this value. However, problems arise when applying traditional decision rules to very low count rate data due to the assumption that the experimenter has an extremely accurate estimate of the mean and standard deviation of the background. Accounting for the uncertainty in background fluctuations while simultaneously improving the decision threshold is a considerable undertaking in frequentist statistics. Bayesian data analysis could supply the tools to approach this problem. The study presented investigates applications of Bayesian data analyses to the decision threshold. The posterior distribution of a Bayesian model contains all available information and uncertainty about the true value of a parameter of interest. The challenge lies in developing a model such that decisions can be made on sample measurements similar to frequentist decision theory. The model is tested on measurements taken in controlled settings in a laboratory using both static and variable background. Ultimately, a successful decision threshold developed through Bayesian data analysis will be easy to implement and improve true detection rates under lower false positive rates when compared to the traditional statistical tests used for radiation detection.

KEYWORDS: Bayesian statistics, radiation detection, decision threshold

1 INTRODUCTION

1.1 Bayesian Applications in Radiation Protection

Applications of Bayesian statistical techniques to radiation protection include source localization and identification [1], characterization of radioactive samples [2, 3], and uncertainty analysis [4]. However, limited literature exists regarding the application of a Bayesian decision threshold for simple gross count measurements. The difficulty lies in employing the Bayesian interpretation of probability to the characteristic limits defined in ISO11929 [5]. Bayesians use Bayes Theorem to update prior beliefs in light of new information. The parameters described in a Bayesian model can be considered as a target of learning, which are characterized by the posterior distribution the model produces. In this sense, the Bayesian modeling procedure provides the means to obtain abstract statistical knowledge about the data in question. The most widely applied statistical model to learning from data is a regression analysis. When used in a Bayesian sense, regression analysis provides an intuitive manner to finding the relationship between parameters of interest.

1.2 Bayesian Linear Regression

This section utilizes language and terminology for describing and coding statistical models commonly found in statistical text and journals, and is general to both Bayesian and non-Bayesian modeling. In a linear regression study, a set of measurements that the experimenter wishes to predict or understand is recognized as the outcome variable or variables, outcome_i. A likelihood function/distribution is defined for each of these outcome variables, providing the probability of observing the specified data under distinct parameter values. These distributions are always assigned normal distributions in linear regression. Separately, the experimenter identifies another set of data, the predictor variables predictor_i, to use to predict or understand the outcome variables. Linear regression computes the relationship between outcome_i and predictor_i by linking the shape of the likelihood (i.e., for mean, variance, etc.) to the predictor variables. Under this experiment structure,

the experimenter must determine and define all of the parameters for the model. In a Bayesian context, this includes appropriately selecting prior distributions for all of the parameters in the linear regression model. Selection of these priors defines the initial state of the model. The basic structure for the linear regression model is then

$$\begin{gathered}\text{outcome}_i \sim \text{Normal}(\mu_i, \sigma)\\ \mu_i = \alpha + \beta \times \text{predictor}_i\\ \alpha \sim \text{Normal}(\mu_\alpha, \sigma_\alpha)\\ \beta \sim \text{Normal}(\mu_\beta, \sigma_\beta)\\ \sigma \sim \text{Uniform}(\text{min}_\sigma, \text{max}_\sigma)\end{gathered} \tag{1}$$

This simple model can be extended to a multivariate regression where multiple predictors are used.

1.3 Bayesian Interaction Model

Basic linear consider each outcome, outcome_i, to be conditional on a set of predictors for each case i. Embedded in these models is the assumption that each predictor has an independent association with the mean of the outcome. Such an assumption is not always correct, as it is conceivable that associations among predictors are conditional. For example, suppose that a relationship between outcome and predictor used in a multivariate model varies according to whether the measurement is from background or with a source present. Simple linear models cannot account for the required conditioning. Modeling conditionality where one predictor depends upon another predictor is achieved by using an interaction [6]. A linear interaction model is built by creating a linear model where the relationship between outcome_i and predictor_1 is made to vary as a function of predictor_2. Within the linear model, this relationship is measured by the slope parameter β_1. To accomplish the desired interaction, β_1 is constrained to be dependent upon predictor_2 by defining β_1 as a linear model itself and including predictor_2 in the definition. Using the previous multivariate regression equations, the linear interaction model takes the following form:

$$\begin{gathered}\text{outcome}_i \sim \text{Normal}(\mu_i, \sigma)\\ \mu_i = \alpha + \gamma_i \times \text{predictor}_{1,i} + \beta_2 \times \text{predictor}_{2,i}\\ \gamma_i = \beta_1 + \beta_{1,2} \times \text{predictor}_{2,i}.\end{gathered} \tag{2}$$

Prior distributions are placed on all applicable parameters, but for simplicity have been omitted here. The parameter $\beta_{1,2}$ defines the strength of dependency of outcome_i and predictor_1 on predictor_2; and γ_i is a placeholder that defines the linear function of the slope between outcome_i and predictor_1. This equation defines the linear interaction effect between outcome_i and predictor_1. The interaction model creates posterior distributions that are conditional on those aspects of the data that posterior distributions from the simpler linear models cannot resolve. This modeling approach allows the relationship between the predictor variable and outcome to change depending upon another predictor variable, and provides the ability to estimate the aspects of a distribution of those changes. This paper presents the results of an investigation using a multivariate linear regression analysis with Bayesian modeling applied to radiation count measurement data.

2 METHODS AND MATERIALS

2.1 Application to Radiation Count Measurement Data

The scenario tested in this analysis is one in which outcome data either originated from measurements of ambient background or measurements with a radioactive source present. The data analyzed were gross counts in fixed 1 s intervals, such that the variable $\text{outcome}_i = Count_i$. Suppose the experimenter wishes to study the relationship between a gross count measurement and the standard deviation of the gross counts obtained in the current measurement and the previous four measurements. This predictor can be designated as $SD5$, and $\text{predictor}_{1,i} = SD5_i$. The condition of whether the analyzed data are from a background measurement or a measurement with a source present requires the linear model to incorporate a categorical predictor, such that $\text{predictor}_{2,i} = Bkgd\_Sample_i$. In this categorical predictor, $Bkgd\_Sample = 1$ if $Count_i$ is attributed to a

background measurement; 0 otherwise. The simplified multivariate linear model for this scenario takes the following form:

$$\begin{gathered} Count_i \sim \text{Normal}(\mu_i, \sigma) \\ \mu_i = \alpha + \beta_{SD5} \times SD5_i + \beta_{Bkgd\_Sample} \times Bkgd\_Sample_i. \end{gathered} \tag{3}$$

Given this model and the strategy discussed previously regarding the components of an interaction model, we want $Count_i$ and $SD5_i$ to vary as a function of $Bkgd\_Sample_i$. This relationship is measured by the slope β_{SD5}. To accomplish the desired interaction, β_{SD5} is constrained to being dependent upon $Bkgd\_Sample$ by defining a linear model for β_{SD5} that includes $Bkgd\_Sample$. The linear interaction model takes the following form:

$$\begin{gathered} Count_i \sim \text{Normal}(\mu_i, \sigma) \\ \mu_i = \alpha + \gamma_i \times SD5_i + \beta_{Bkgd\_Sample} \times Bkgd\_Sample_i \\ \gamma_i = \beta_{SD5} + \beta_{Bkgd\_Sample,SD5} \times Bkgd\_Sample_i. \end{gathered} \tag{4}$$

By defining the relationship between $Count$ and $SD5$ with a linear interaction model, any change in μ_i resulting from a unit change in $SD5_i$ is given by γ_i. Now, to compute the relationship between $SD5_i$ and $Count_i$, incorporation of β_{SD5}, $\beta_{Bkgd\_Sample,SD5}$, and $Bkgd\_Sample_i$ is required. The model explicitly addresses the hypothesis that the slope between $Count$ and $SD5$ is conditional upon whether or not a measurement is from background, and the parameter $\beta_{Bkgd\_Sample,SD5}$ defines the strength of the dependency. Even though the parameters $SD5$ and γ are constructs of this experiment, in the Bayesian framework they can be thought of as targets of learning that can be characterized by a posterior density. This point highlights a valuable aspect of Bayesian statistical modeling: when an experimenter wishes to obtain abstract statistical knowledge about the data, a Bayesian procedure provides a model that can describe non-physical parameters.

Under the investigative nature of the experiment, appropriate selection of prior distributions and parameter values for the model is based on partial knowledge of the data to be analyzed and a preferred state of ignorance, premised on information theory and maximum entropy. The model used to generate the results in this paper takes the form:

$$\begin{gathered} Count_i \sim \text{Normal}(\mu_i, \sigma) \\ \mu_i = \alpha + \gamma_i \times SD5_i + \beta_{Bkgd\_Sample} \times Bkgd\_Sample_i \\ \gamma_i = \beta_{SD5} + \beta_{Bkgd\_Sample,SD5} \times Bkgd\_Sample_i \\ \alpha \sim \text{Normal}(600,10) \\ \beta_{SD5} \sim \text{Normal}(0,1) \\ \beta_{Bkgd\_Sample} \sim \text{Normal}(0,1) \\ \beta_{Bkgd\_Sample,SD5} \sim \text{Normal}(0,1) \\ \sigma \sim \text{Uniform}(0,50). \end{gathered} \tag{5}$$

Here, α is the expected value for $Count_i$ when all of the predictors are equal to 0. Based on knowledge from past observations, acceptable values for the expected gross counts, μ_i, from background measurements in the data analyzed in this investigation can be centered at 600 counts and assumed to be normally distributed over a relatively wide range. This allows the model to start at a reasonable approximation, as well as maintaining a partial state of ignorance. The values for the Gaussian priors of β_{SD5}, β_{Bkgd_Sample}, and $\beta_{Bkgd\_Sample,SD5}$ were selected because equal probability exists above and below 0, where a value of 0 suggests that the predictors have no relationship to the observed gross counts. This is a conservative assumption, especially considering that most of the probability mass is around 0. A flat prior for σ constructs a posterior distribution proportional to the likelihood, with the advantage that the data will drive the model towards an appropriate approximation.

The model operates by examining a set of previously recorded background gross count measurements, the training data, and resulting $SD5$ with a set of unknown sample gross count measurements and resulting $SD5$. These two sets of data make up the arrays for $Count_i$ and $SD5_i$. Included in the data array is $Bkgd\_Sample_i$, such that each index is correctly categorized as a known background measurement or an unknown sample measurement. This setup allows the model to work in a way that is intuitive to operational measurement technique: known background data and resulting estimates are used to create a relationship that is expected to be consistent across all measurements with no source present, and this relationship is compared to the samples in question to judge if a source is present. All statistical calculations and analyses were performed in the R statistical programming language.

2.2 Experimental Laboratory Setup

The measurement data used in the analyses were collected in a laboratory setup utilizing a 2×2 NaI scintillation detector (Model 902, Canberra Industries Inc., Meriden, CT) with a resolution of 8.5% at the ^{137}Cs photopeak energy of 662 keV. Acquisitions took place at times of the day when the background was expected to be stable. To simulate a low-fidelity system for initial algorithm evaluation and testing, the data were acquired in the open window of the NaI detector system. The data were collected as full spectral data; however, all channel entries were summed in the subsequent data analysis to provide simple gross count data. The same laboratory setup was used for data acquisition with and without a source present, representing the presence of an exposed ^{137}Cs source and an evaluation of the background, respectively. Different source "strengths" were simulated by varying the distance between the (point) source and the face of the NaI detector. Measurements were conducted for source-detector distances of 200 cm and 400 cm, and two separate background measurements were conducted. Data were collected in measurement sequences to obtain 1800 spectra per background measurement and 1800 source spectra at each source to detector distance.

3 RESULTS AND DISCUSSION

Due to the partially non-informed prior distributions and parameter values in the model presented, it is important to assess the hypothesis in a manner that is driven completely by the data. The clearest way to visualize the hypothesis is to analyze all 3600 (1800 from background and 1800 from unknown sample) measurement indices per comparison, plot the linear regression with and without the interaction model, and overlay the regression line of expected gross counts for a given $SD5$. These plots are shown in Figs. 1 and 2.

Figure 1: $Counts$ vs $SD5$ for training background measurements with corresponding 200 cm (top) and 400cm (bottom) source distance measurements without (left) and with (right) interaction

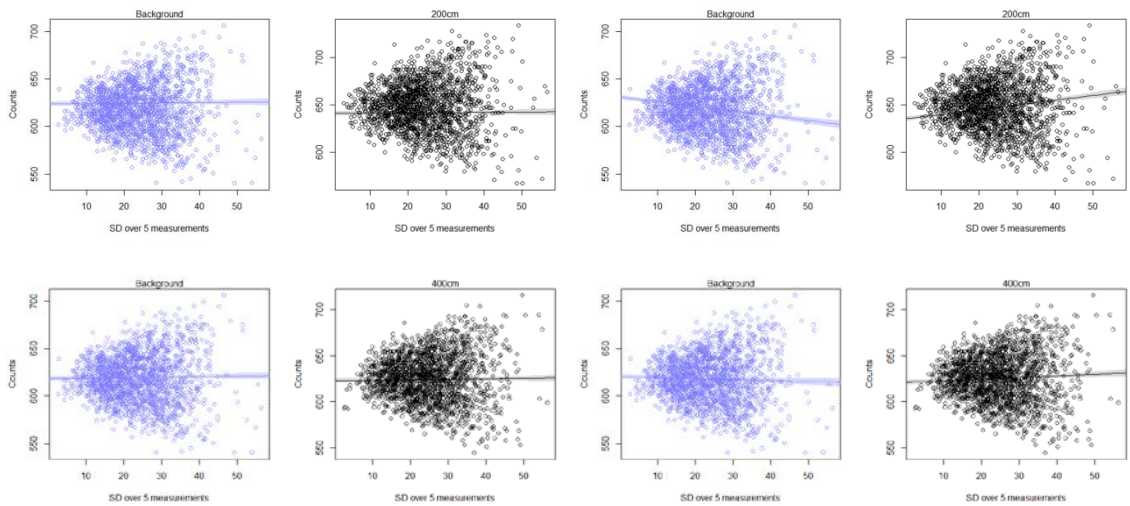
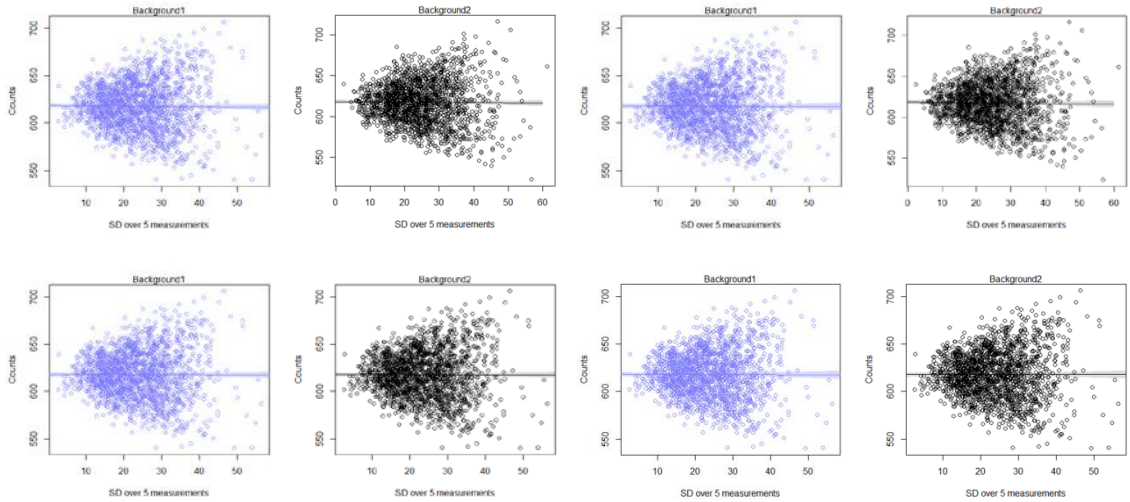


Figure 2: $Counts$ vs $SD5$ for training background measurements with corresponding second background set (top) and self-test (bottom) measurements without (left) and with (right) interaction



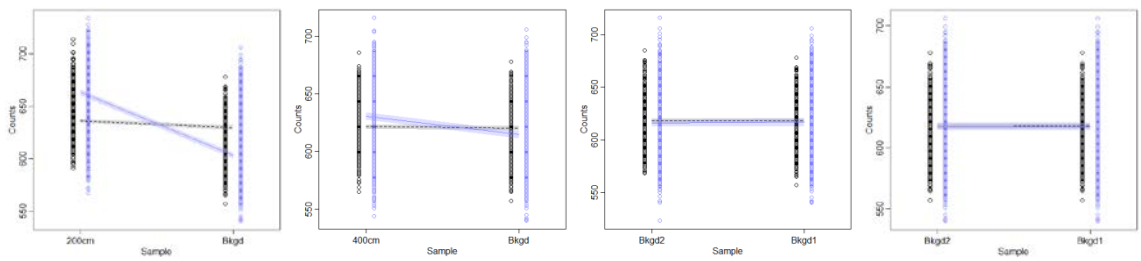
The positive slope seen in the 200 cm and 400 cm interaction plots suggests that a relationship between $Count$ and $SD5$ is conditional upon whether or not the sample is from background. To further substantiate this observation, it is apparent that when no source is present, the change in slope, if any, is minimal. This indicates that a relationship exists when comparing background measurements to sample measurements with a source present.

Additional support for the interaction model is obtained by taking advantage of the fact that linear interactions are symmetrical [7]. We change the model such that we are investigating if the relationship between $Count$ and $Bkgd\_Sample$ is dependent upon $SD5$. To accomplish this, we still maintain every parameter and value from the original linear interaction model with the exception that we define a linear model for β_{Bkgd_Sample} that includes $SD5$. The model now takes the following form:

$$\begin{aligned} Count_i &\sim \text{Normal}(\mu_i, \sigma) \\ \mu_i &= \alpha + \gamma_i \times Bkgd\_Sample_i + \beta_{SD5} \times SD5_i \\ \gamma_i &= \beta_{Bkgd\_Sample} + \beta_{Bkgd\_Sample,SD5} \times SD5_i. \end{aligned} \tag{6}$$

Essentially, we are asking if the gross counts obtained in background measurements versus measurements with a source present depend on the standard deviation of the gross counts obtained in the current measurement and the previous four measurements. The results for all four scenarios are provided in Fig. 3.

Figure 3: Training background (Bkgd/Bkgd1) versus 200cm source distance, 400cm source distance, a second background set, and a self-test, respectively, for the symmetry interaction model



Measurements from the training background data set are plotted on the right and the corresponding unknown sample measurements are plotted on the left. Within each category on the plots, the entries are split such that counts below the median $SD5$ value for that category are plotted on the left, and

counts above the median $SD5$ value are plotted on the right. The dashed line and shaded region is the expected reduction in counts when we take an imagined sample with the minimum $SD5$ value and change between source present and no source present. It is apparent that across all scenarios, the slope of this line is nearly zero. This suggests that for any measurement, from background or otherwise, having a relatively low $SD5$ has a negligible effect on the gross counts for a particular measurement. What is more important is the comparison of this line to the solid line within each plot. The solid line and shaded region represent the expected change in gross counts for an imaginary sample with a maximum $SD5$ value. This line shows a clear reduction in expected counts when you compare a measurement with a source present versus no source present. As the comparison changes from stronger to weaker to no source present versus the background measurement data, this reduction decreases to a slope similar to the dashed line. These results indicate that for measurements associated with a high $SD5$, there is a positive effect (increase in expected gross counts) on the observed gross counts when a source is present. The original perspective presented, that the relationship between $SD5$ and gross counts in a measurement depends upon whether or not a source is present, is simultaneously true with this second perspective, that the relationship between whether or not a source is present and gross counts in a measurement is dependent upon $SD5$. This result establishes that the interaction model is valid. It is important to note however that these statements are only credible under the model assumptions and the data used. To further confirm the linear interaction model, the same analyses were performed using two additional background measurement datasets as the training dataset. The outcomes were similar to the previous results presented here.

Given the confirmation of the model assumptions, the next step is to utilize Bayesian data analysis to approximate the posterior distributions of the relevant parameters to help understand the previous observations. Maximum a posteriori (MAP) estimates of the mean and standard deviation for the parameters α, β_{SD5}, β_{Bkgd_Sample}, $\beta_{Bkgd\_Sample,SD5}$, and σ were obtained using the quadratic approximation technique and are reported in Tab. 1

Table 1: MAP estimates of 200 cm source and 400 cm source distances, a separate background (Background 2), and against an identical dataset (Self)

	Unknown Sample Dataset							
	200cm		400cm		Background 2		Self	
parameter θ	μ_θ	σ_θ	μ_θ	σ_θ	μ_θ	σ_θ	μ_θ	σ_θ
α	635.02	1.34	621.15	1.34	618.19	1.32	617.75	1.3
β_{SD5}	0.5	0.05	0.17	0.05	-0.04	0.05	0	0.05
β_{Bkgd_Sample}	-4.04	0.93	-0.75	0.93	-0.02	0.93	0.08	0.92
$\beta_{Bkgd\_Sample,SD5}$	-0.99	0.05	-0.27	0.05	0.02	0.05	0	0.05
σ	26.1	0.31	25.72	0.3	25.82	0.3	25.36	0.3

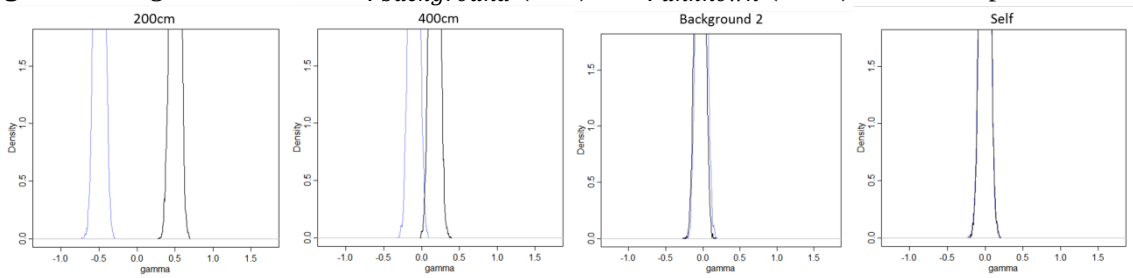
The parameter γ can be calculated using the relationship from Eq. 4. From this calculation, we can see that the relationship between $SD5$ and observed gross counts is reversed when comparing measurements from background and measurements with a source present. The calculated γ for each experiment is shown in Tab. 2. The table shows that the relationship changes sign when the background dataset is compared to a dataset with a source present, whereas no reversal occurs when comparing the background dataset to another background dataset or itself.

Table 2: Calculated γ for each experiment

Source Type	γ	
	No Source	Source
200cm	-0.49	0.5
400cm	-0.1	0.17
Background 2	-0.02	-0.04
Self	0	0

To gain a better understanding of the relationship between $SD5$ and observed gross counts depending upon the presence of a source, we need to find the uncertainty associated with γ. This uncertainty can be obtained by computing the posterior distributions for the parameters in the model. These distributions were approximated by taking 10000 random samples from the linear interaction model. The resulting marginal distributions are shown in Fig. 4.

Figure 4: Marginal densities for $\gamma_{background}$ (blue) and $\gamma_{unknown}$ (black) for each experiment



The marginal distributions allow us to ask the most useful question from the interaction model: what is the probability that the relationship between $SD5$ and gross count measurements from background is less than the relationship between $SD5$ and gross count measurements from a sample? This question can be answered by computing the difference between the marginal distributions, γ_{diff}, for each sample from the posterior, and find the proportion of differences below 0. The results of these computations for each of the source strengths are shown in Tab. 3.

Table 3: Calculated γ_{diff} for each experiment

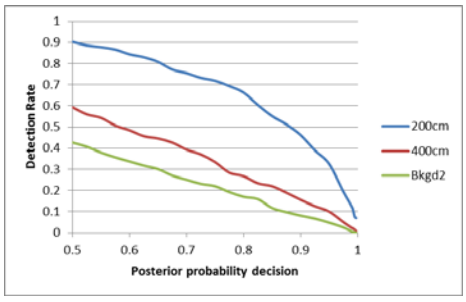
Source Type	γ_{diff}
200cm	1
400cm	1
Background 2	0.3201
Self	0.5284

Detection using γ_{diff} is accomplished by defining a decision such that a set of measurements, i, analyzed in the model are considered to trigger an alarm if

$$\Pr^* < \gamma_{diff_i} \tag{7}$$

where $\Pr^*$ is a pre-specified, fixed probability and γ_{diff_i} is the probability that the relationship between observed gross counts and $SD5$ from measurements of an unknown sample are greater than the relationship between observed gross counts and $SD5$ from measurements of a background sample. Fig. 5 displays the results from analyzing 360 independent trials using $\Pr^*$ values ranging from 0.5 to 0.9975. The same data used to generate the 1800 individual 1 s measurement results were used, but the data were partitioned into 5 1 s measurement sequences, such that 5 measurements were used per trial and each trial was equivalent to 5 s per acquisition.

Figure 5: Alarms triggered (Detection Rate) for a selected $\Pr^*$ (Posterior probability decision)



Detections for the 200 cm and 400 cm source distances can be equated to a true detection rate (true positives), and detections on Background 2 are equivalent to false alarms (false positives). Setting $\Pr^* = 0.5$ is the least conservative decision, evident from the amount of alarms triggered on the Background 2 data. At an approximate "false positive" rate of 0.05 (when the detection rate of Background 2 ≈ 0.05), the true detection rate for the 200 cm source distance is$\sim$0.33, and $\sim$0.10 for the 400 cm source distance. While these detection rates are suboptimal, the fact that partially non-informed priors, conservative estimates, and broad normal distributions were used in the interaction model must be considered.

4 CONCLUSIONS

Applications of a Bayesian decision threshold for simple gross count measurements have been limited by the difficulty of employing the Bayesian interpretation of probability to the characteristic limits defined in ISO11929. Bayesian inference and modeling provides the means to obtain abstract statistical knowledge about the data in question. Parameters described in a Bayesian model can be considered as a target of learning, which are characterized by the posterior distribution the model produces.

The Bayesian linear interaction model allows the experimentor to conditionally model a parameter of interest using predictor variables. The model presented explicitly addresses the hypothesis that the relationship (slope) between $Count$ and $SD5$ is conditional upon whether or not a measurement is from background. The model was confirmed by comparing multiple datasets, and by showing that a symmetrical relationship was present among the variables tested.

The model was then extended into utilizing posterior distributions using known background data and resulting estimates to create a relationship that is expected to be consistent across all measurements with no source present, and this relationship is compared to the samples in question to judge if a source is present. This relationship, γ_{diff}, was extended into a detection method requiring a fixed posterior probability Pr^*. The detection results were promising considering partially non-informed priors, conservative estimates, and broad normal distributions were used in the interaction model. These possible shortcomings can be addressed with model specification and optimization using information criterion, and incorporating Markov Chain Monte Carlo to approximate posterior distributions constructed from a wide range of distributions.

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Utilizing Data Patterns in String Measurements to Enhance Radiological Detection Capabilities

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Abstract. The detection of a signal embedded in background in counting experiments customarily utilizes a fixed decision threshold determined from previous measurements of background data to distinguish signal from background events. Standardized methods to calculate this decision threshold assume that the distribution of the background data follows a normal distribution, which can be characterized by the mean and the standard deviation of the data. For the analysis of data at low signal-to-background ratios, the assumption of a normal background distribution may not always be valid. The approach to calculating the decision threshold has been generalized and the statistics of string measurements have been evaluated. Binomial statistics can be utilized to describe data patterns in string measurements which frequently occur in a variety of operational radiation protection applications, such as continuous laboratory or environmental contamination surveys or in the search for orphan sources. The analysis of those data patterns allows for the identification and detection of a signal at a lower signal-to-background ratio, resulting in higher detector sensitivity or, conversely, in a lower false positive detection rate. Results are presented for data pattern analyses of gross count / count rate instruments as well as of spectroscopic measurements. Enhancements in radiological detection capabilities for these instruments have been observed in laboratory measurements for string measurement gross count data and sequential spectroscopic data. The binomial discriminator approach has also been tested in low-fidelity spectral data analysis and provided similarly enhanced detection capabilities. Statistical algorithms have been developed which utilize the analysis of patterns in the data strings to enhance the test statistic for the decision on the absence or presence of a radiological source. Null hypothesis test performance and source detection efficacy have been shown to improve compared to the traditional method of achieving a detection decision by the comparison of a measured value to a fixed decision threshold.

KEYWORDS: radiation detection, decision threshold, environmental monitoring, cargo inspection

7 INTRODUCTION

The detection of ionizing radiation often relies on the identification of an electronic signal (e.g., current, voltage above a discriminator threshold, charge on a capacitor, etc.) resulting from interactions of the radiation with the detection medium. In simple counting experiments, these signals may not be unique in their identifiable characteristics, however, but might be mimicked by particles from other interactions, emitted from natural background sources, or simulated by electronic noise in the signal processing. Any contribution to the signal count which does not result from an interaction of the ionizing radiation of concern is considered background in this paper. Ionizing radiation is detected when a statistical analysis determines that the total number of events registered in the detector exceeds the number of background events expected in the measurement interval.

In a frequentist (classical) statistical approach to the data analysis of a radiation measurement, the experimenter assumes that the calculated background distribution derived from a "paired blank" count [1,2] or a series of measurements is accurately representative of the true background distribution. A distinction between background and signal as the basis of a detection decision is then obtained by the appropriate utilization of a set of characteristic limits [3]. The characteristic limit at which the sample is investigated further whenever a set of measurements contains a count rate data point exceeding that limit, or otherwise indicates an activity greater than it is called the decision threshold, y^*.

If a measurement value exceeds y^*, the measured sample is considered to emit radiation above the established background. The experimenter decides that a radiological source is present, and accepts a probability that this decision is incorrect of magnitude α, the error of the 1st kind [3] or false positive detection rate. Conversely, if a measurement is below the decision threshold, the decision is that the

sample does not contain a radiological source but rather reflects a measurement of the background. The decision threshold, y^*, is a purely investigatory level which does not provide information regarding detection capabilities; however, the initial detection decision is based on this value. Standard Operating Procedures for the experimenter would then determine the further course of action after a positive detection event, such as e.g., prolonged confirmatory measurements, additional sampling, or other experimental techniques to confirm the detection and enhance the identification of the radiological source.

In various health physics and radiological protection applications, radiation measurements may be conducted on a continuous basis, such as during contamination surveys in a laboratory, during land or site surveys for decommissioning of a facility, in the search for orphan sources or clandestine radiological material, or in cargo and passenger inspection at border crossings or ports of entry. Many such continuous measurements actually comprise of a set of individual measurements within a fixed integration interval. Instruments generally do not retain measurement information from the previous integration interval, such that individual measurements can be considered independent. An alarm is triggered when a measurement exceeds a pre-set decision threshold in the instrument which is derived from operational necessities (low incidence of false or nuisance alarms, minimum activity level required for the measurement for regulatory compliance, etc.).

The information contained in the data string acquired in the course of continuous measurements can be utilized to enhance conventional detection capabilities by examination of the data patterns generated in a (time) sequence of independent measurements. This paper presents the statistical basis for a string data analysis by means of a binomial discriminator which quantifies the joint probabilities for specific patterns to be observed in background. If those joint probabilities are too low, the string data pattern is attributed to a signal event with a corresponding detection decision.

The (frequentist) statistical approach to string data analysis has been tested in the laboratory for a variety of operational settings, such as different source strengths, unshielded and shielded gamma sources, and for applications in gamma spectrometric analyses. Utilization of the binomial discriminator was also extended from sole use in time series applications to the investigation of a low-fidelity gamma spectrum. Results of those laboratory investigations are presented here.

8 METHODS AND MATERIALS

8.1 Statistical Basis

The definitions and symbols used follow the international standard literature for the measurement of ionizing radiation [3]. A general expression for the decision threshold derived from the Neyman-Pearson criterion [4] is of the form:

$$P(y > y^*) = \int_{y^*}^{\infty} \phi(y')\ dy' = 1 - \int_{-\infty}^{y^*} \phi(y')\ dy' = \alpha, \tag{1}$$

where $P(y > y^*)$ is the probability that a measurement value exceeds the decision threshold, y' denotes the parameter space of measurement values, and ϕ is the background distribution.

In simple counting experiments, as they often present themselves in radiological measurements, experimenters usually consider Poisson or normal distributions for the variation in measurement values from a fixed background, depending on the observed counts in a measurement / integration interval. The statistical basis for a pattern analysis in string data has been developed for both possible background distributions in the course of this study.

The general approach to computing the decision threshold for Poisson and normally distributed background is explained in Table1. Naturally, the Poisson distribution is a discrete distribution, in contrast to the continuous normal distribution, such that the computational implementation is slightly different.

Table 1: Decision threshold calculated for Poisson and normal background distributions at $\alpha = 0.05$

Distribution	Decision Threshold for $\alpha = 0.05$
Poisson	iterative solution to: $1 + \hat{y} + \frac{\hat{y}^2}{2!} + \frac{\hat{y}^3}{3!} + \cdots + \frac{\hat{y}^{y^*}}{y^*!} = 0.95\, e^{\hat{y}}$
Normal	inversion of: $\int_{y^*}^{\infty} \frac{1}{\sqrt{2\pi}\; u(\hat{y})} e^{-\frac{(y'-\hat{y})^2}{2\,u^2(\hat{y})}}\, dy' = 0.05$

For a sequence of N independent measurements, the joint probability for all N results to exceed y^*, $P(y_1, \cdots, y_N > y^*) = \alpha^N$. If that sequence also includes individual measurements not exceeding y^*, then the joint probability $P(y_1, \cdots, y_n > y^*;\ y_{n+1}, \cdots, y_N \leq y^*)$ follows the binomial distribution with α representing the probability of a successful binomial trial outcome (exceeding y^*) and $(1-\alpha)$ quantifying the probability of an unsuccessful trial outcome (not exceeding y^*) [5]. For an unordered set of sequential independent measurements sampled from the same background distribution in which of the N sequential measurements, n measurements would provide a result exceeding y^*, irrespective of their exact position within the sequence, the joint probability can be expressed as:

$$P(y_1, \cdots, y_n > y^*;\ y_{n+1}, \cdots, y_N \leq y^*) = \binom{N}{n}\alpha^n(1-\alpha)^{N-n} = \frac{N!}{n!\,(N-n)!}\alpha^n(1-\alpha)^{N-n}. \quad (2)$$

For the practical implementation of a systematic approach to evaluating the decision threshold from a binomial discriminator in a data string, rather than requiring exactly n measurements to exceed y^* within N sequential measurements, a more inclusive approach for any value of y would compute the joint probability that at least n measurements exceed y^* in that sequence. For an unordered set, such a joint probability would be described by:

$$P(y_1, \cdots, y_i > y^*;\ y_{i+1}, \cdots, y_N \leq y^* | i \geq n) = \sum_{i=n}^{N} P(y_1, \cdots, y_i > y^*;\ y_{i+1}, \cdots, y_N \leq y^*) = \sum_{i=n}^{N} \frac{N!}{i!\,(N-i)!}\alpha^i(1-\alpha)^{N-i}. \quad (3)$$

The data strings are analyzed for the patterns in which individual measurements exceed the decision criterion of the binomial discriminator. The data string metric denotes the pattern in a data string of length N, and enumerates the expected number of individual readings exceeding the binomial discriminator level, i.e., the data string metric is given by all n/N with $n = 0,1,\cdots,N$ and n being the number of readings exceeding the discriminator level. As an example, the results presented for gross count measurements in this paper utilize a string length $N = 5$. However, many of the conclusions from those examples can be applied to other string lengths as well.

8.2 Experimental Laboratory Setup

Data for this string data analysis were collected in a laboratory setting using a 2×2 NaI scintillation detector (Model 902, Canberra Industries Inc., Meriden, CT), a tube base (Model 2007P, Canberra Industries Inc., Meriden, CT), and a preamplifier. The preamplifier signal is further processed by a Lynx Digital Signal Processor (Canberra Industries Inc., Meriden, CT). The detector resolution is 8.5% at the ^{137}Cs photopeak energy of 662 keV. A low-fidelity system for algorithm evaluation and testing is simulated by data which were acquired in the open window of the NaI detector system. All channel entries were summed in the subsequent data analysis to provide gross count data, or were suitably combined for spectral analyses.

Prior to commencement of the actual data acquisition, the background in the laboratory was characterized for variations throughout the day. The subsequent data acquisition was timed such that background was expected to be stable. Individual measurements were recorded at 1 s integration times. Data were acquired in the same experimental setup without a source present, representing

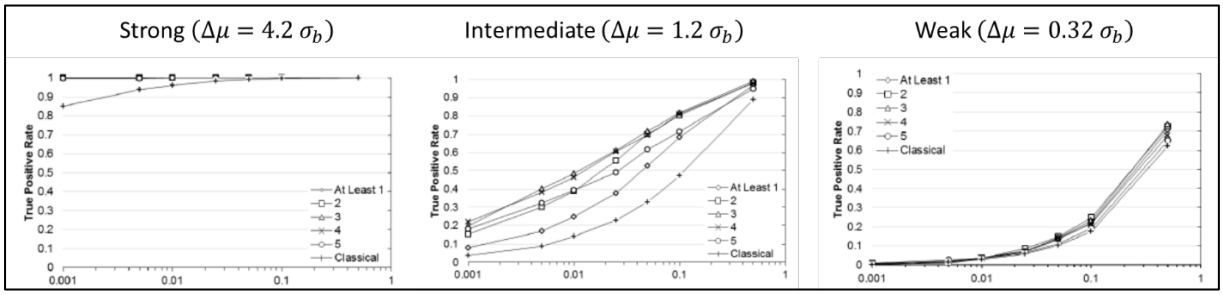
background only in the evaluation of the false positive rate, and in the presence of an exposed ^{137}Cs source. Varying the distance between the (point) source and the face of the detector allowed for the simulation of different source "strengths." Measurements were obtained for source-detector distances of 100 cm, 200 cm, and 400 cm. For each background or exposure scenario, 1800 background or 1800 source spectra were collected.

A "strong" source is modeled by a source-detector distance of 100 cm and exhibits an average number of counts of 108. above background, or 4.23 σ_b, in the 1 s integration interval. For an "intermediate" source at 200 cm, the average number of counts is 30.7 above background, or 1.17 σ_b. And a "weak": source utilizes a source-detector distance of 400 cm with an average number of counts of 8.01 above background, or 0.32 σ_b. False positive rates were verified by checking the performance of the string data analysis on background measurements only. The analysis detection capabilities were tested by finding the true positive detection rate on the different source "strengths" and comparing them to source detection rates utilizing the traditional decision threshold for a single measurement.

9 RESULTS AND DISCUSSION

Receiver Operator Characteristics (ROC) curves for different source strengths are shown in Fig. 1. These ROC curves plot true detection rates on the ordinate against false positive detection rates on the abscissa. Detection system performance is considered more optimal the larger the number of true detections for a given level of false positive detections.

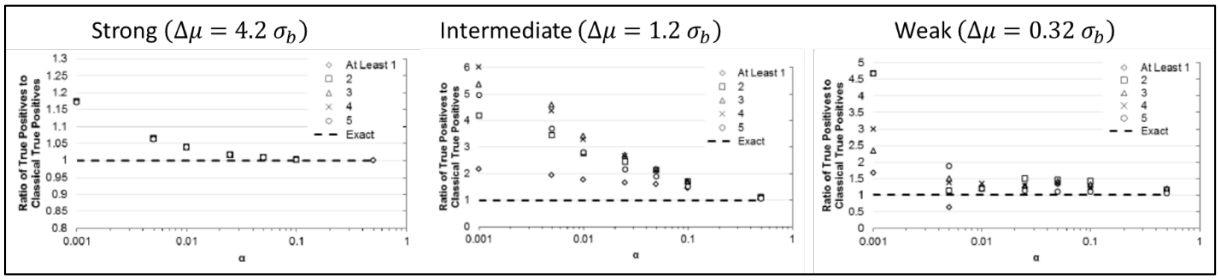
Figure 1: Receiver Operator Characteristics for a string length of $N = 5$ and different data string metrics



A comparison of the true positive detection rates of any given data string metric relative to the true detection rate when the traditional decision threshold is used further aids in the visualization of the actual improvement in radiological source detection. The ratios of the true detection rates of the various data string metrics and the traditional radiological detection decision are provided in Fig. 2. Analyses based on the binomial discriminator to quantify patterns in measurement strings outperform the traditional ("classical") decision process. As Fig. 2 shows, the majority of the data string metrics exceed the true positive rate by a factor of 1.5 in the range of operational false positive values (α~0.05). More importantly, some data string metrics perform better at rates 2 to 3 × higher for $\alpha < 0.01$.

An important result with respect to the data string metric is that fixing a specific value of the data string metric in the general case where the unknown source could be weak or strong does not provide optimal performance. The different data string metrics perform better at different fixed false positive rates as shown by the ratio of string metric true positives to traditional true positives. The dashed line along "Exact" corresponds to a ratio of one, and any point above this line signifies greater detection efficiency than the traditional decision threshold method.

Figure 2: Ratios of true detection rates for different data string metrics and the traditional detection decision



A similar approach to utilizing the binomial discriminator has been tested for spectroscopic data. The same data sets were analyzed for source detection in a time series of independent 1 s spectra. The ROC curves for different data string metrics using the "intermediate" source strength are shown in Fig. 3 with particular emphasis on plate b) which represents an optimization of the energy Region(s) of Interest (ROI) for an unshielded source. A suitable parameter for ROI optimization was found to be the value of $\Delta\mu/\sigma$ for any given ROI as compared to the same parameter for the combined ("gross count") spectrum. The optimization parameter is plotted in Fig. 4 in support of the optimized ROI in plate b) of Fig. 3.

Figure 3: Receiver Operator Characteristics for time series spectral analyses in a) for gross count measurements using the combined spectral data, and in b) for the optimized Region(s) of Interest

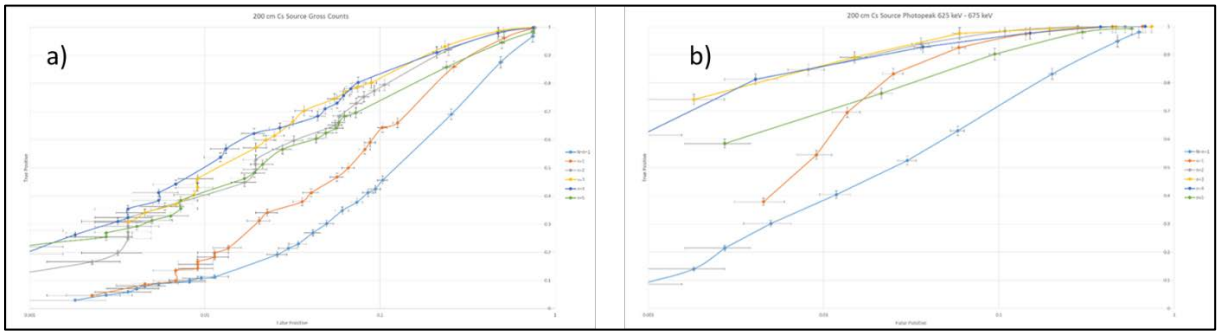
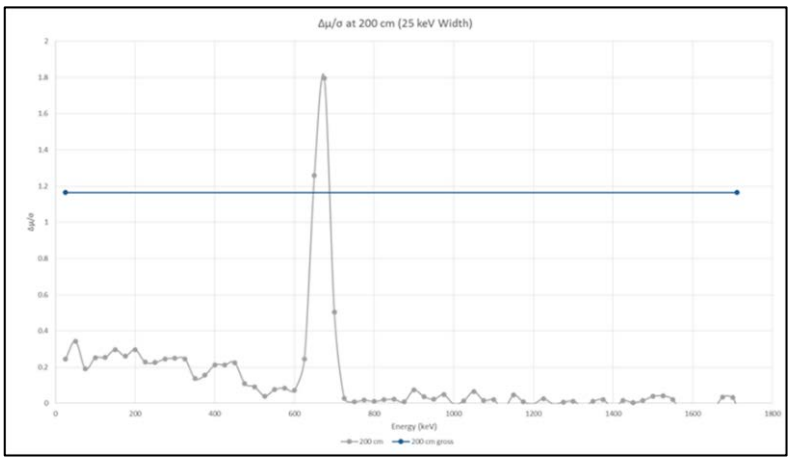


Figure 4: Optimization criterion for selection of Region(s) of Interest in spectral data



The application of the binomial discriminator approach has also been extended to the analysis of low-fidelity spectra. Each spectrum in the laboratory data set was divided into three distinct energy regions: the (Compton) background region at low energies, a suitable ROI around the photopeak

energy, and all energies beyond the photopeak. The ROC curves for the "intermediate" source strength are shown in Fig. 5 for comparison between an unshielded and a shielded source. The corresponding true detection ratios are presented in Fig. 6.

Figure 5: Receiver Operator Characteristics for a low-fidelity spectral analysis using a) an unshielded, and b) a shielded source

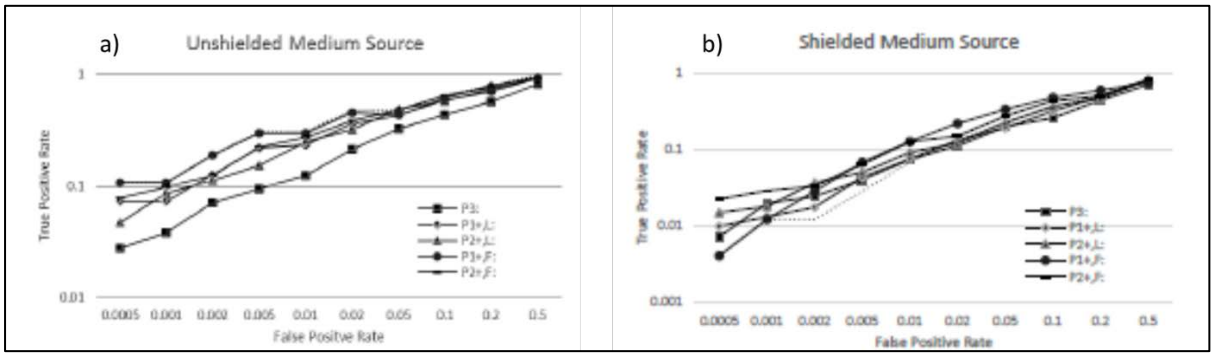
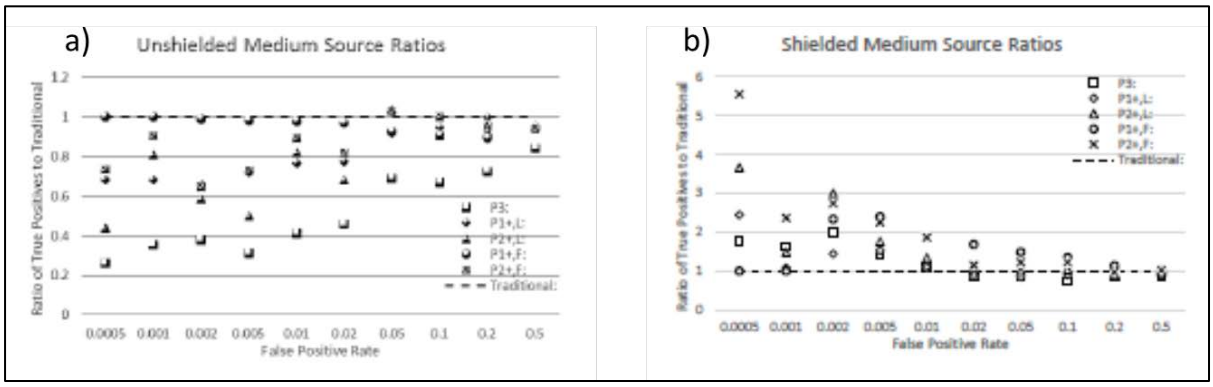


Figure 6: Ratios of true detection rates for different data string metrics and the traditional detection decision for a low-fidelity spectral analysis using a) an unshielded, and b) a shielded source



In the low-fidelity spectral analysis, the binomial discriminator was not efficient in utilizing data beyond the energy of the highest energy photon emission by the source, as those energies do not contribute to the spectrum in a meaningful manner. Requiring all ROIs to increase in the number of entries becomes particularly inefficient. However, all binomial discriminators which did not explicitly depend on increasing numbers of entries beyond the highest energy photopeak performed well. For the unshielded source, unscattered signal contributions allowed the traditional detection decision to perform as well as or better than the binomial discriminator. When Compton scattering contributes measurably to the signal, such as for the shielded source, the binomial discriminator algorithms perform better, and outperform the traditional detection decision in many instances.

The binomial discriminator analysis initially did not suitably account for the low number of entries in the photopeak region. It did not perform as strongly as expected at first due to the Poisson nature of the background and signal distributions in that ROI. A region-specific binomial discriminator value between two integers may not provide a detection advantage over the traditional detection decision when the binomial discriminator triggers between the same two integers as the traditional decision threshold. For larger numbers of entries in a spectral region with background and signal being more closely represented by continuous probability distribution functions, the binomial discriminator algorithms are expected to perform even better.

10 CONCLUSIONS

Data analysis utilizing binomial discriminators to quantify patterns in string data from independent measurements in (time) sequences of simple counting measurements have been shown to achieve performance improvements compared to the traditional approach to the decision of whether a weak source is present in a set of measurements. This approach can be employed in the analysis of data from instrumentation generating sequences of independent individual measurements, such as for example, from hand-held or portable instrumentation for laboratory or site surveys, from stationary portal monitors at border crossings or ports of entry, and from other low-fidelity instrumentation.

The binomial discriminator approach initially developed for simple counting measurements has been extended to the analysis of spectral data where it also allows for improvements in performance for spectroscopy instumentation. Spectra have been analyzed in time series and optimized for different ROIs, and a criterion has been established which provides for a rapid ROI optimization process.

In addition, low-fidelity spectra have been divided into distinct ROIs with the string metric denoting the number of partitioned spectra segments instead of sequential counting measurements. The binomial discriminator approach performs reasonably well in low-fidelity spectral analysis. For an unshielded ^{137}Cs source, the performance approaches that of the traditional detection decision at higher false positive rates. In the detection of a shielded ^{137}Cs source with additional contributions from Compton scattered photons, this approach performs increasingly well even at lower false positive rates, and outperforms the traditional detection decision in many instances.

11 ACKNOWLEDGEMENTS

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The views and conclusions contained in this document are those of the authors and should not be interpreted as necessarily representing the official policies, either expressed or implied, of the U.S. Department of Homeland Security.

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Medical

Applications of a solid state dose profile meter in a fluoroscopic system

Esperanza Pérez Alvarez

INTRODUCTION

The AAPM published a task group report on the Role of the Clinical Medical Physicist in Diagnostic Radiology in 1994, were points out that the primary responsibility of the medical physicist in an imaging program, is the development and supervision of a quantitative quality assurance program, where it is included choosing the most suitable type of equipment for making this quality assurance.

Equipment has become more complex with the maturation of digital radiography and diagnostic radiology has undergone fundamental changes. Because of that we have witnessed that other technologies as CR (computed radiography) has become obsolete. This advances has directly influenced the requirements for equipment needed for quality control. At the same time, our ability to test radiographic systems without invasive measurement has developed along with the computer industry, making possible the capture of test data directly into a database running on a laptop computer.

Medical physicist must be knowledgeable in current control quality equipment designs, intended use, and the appropriateness of the various test instruments that may be used in performance evaluation. Summarizing, medical physicist needs to be able to provide a well-designed QC program that evolves with the technology.

The intent is for this report is to be used by other medical physicist for open their minds and try to explore the different possibilities their actual equipment for control quality offers. That way they can resolve problems derived from the evolution of the equipment they have to check.

Equipment must be appropriate in terms of its ability to deliver the quality necessary for a particular imaging task at a cost to both patient and hospital that is reasonable in terms of dose, money, and downtime. The medical physicist must be an integral component of the equipment selection process since quality begins with proper equipment selection. The diagnostic medical physicist, having been educated in the administrative, technical, and clinical aspects of equipment performance, possesses a unique vantage point from which to assess appropriateness of imaging equipment.

The main purpose of QC testing is to detect changes that may result in a clinically significant degradation in image quality or a significant increase in radiation exposure. But the test results should be recorded in a database for analysis and after that should be used to inform the user of the equipment that have been evaluated.

To assist in the selection of appropriate instrumentation, medical physicist can refer to AAPM Report No. 60. Where contains a compilation of instrumentation requirements for use in evaluation of fluoroscopic equipment. But it does not represent a mandatory fact, because of that we wanted to point out a new choice for the measurement of the field size of x ray equipment.

In Spain, the quality control of field size of x ray in fluoroscopic equipment is a minimum performance criteria that should be met in order to achieve acceptable image quality because a X-ray beam misalignment with the II may resulting in increased radiation exposure to the patient. Until now, computed radiography system and radiographic films have been considered as the gold standard for field size determination. But as digital systems are here to stay, in the next few years CR will become obsolete and radiographic film is very costly in a long term.

Because of that a challenging area in the field of quality control of radiodiagnostic is to find a new methodology for measuring field size of a X-ray machine. In this paper we present a new approach to measuring it.

METHOD

We opted to compare our new method named Dose Profiler method, against CR (Computed Radiography) method. Tests would be performed in all II (Imager Intensifier) magnification modes.

DOSE PROFILER METHOD

The instrument we have used for measuring field size is a dose profiler called CT Dose Profiler belonging to RTI. It is based on solid-state technology. This device mounted on Black Piranha multimeter belonging to RTI can achieve sensitivity for dose rate of 40nGy/s-760mGy/s and a spatial resolution of 0,25mm.

As a preliminary study, a multifunction X-ray machine was chosen. This allows us to control the position of dose profiler moving the table remotely. That way we are allowed to scan the x-ray field size completely with our dose profiler.

The very first time we measure it, we have to characterize the speed of the table. First we measure the speed in its longitudinal and after that in lateral direction. For doing it we put a radiopaque marker at one end of the field attached to the equipment table. After a single shot, we move the radiopaque marker a distance of 20 cm along with the table remotely. Finally, we counted the time of fluoroscopy, helping us with the images taken, since the radiopaque mark was moved until it moved 20 cm. In the following measurements we could use this characterization of the speed table for our measurements.

Once we have made the characterization of the speed table we introduce in our data base “Ocean” the first time we use this method in one equipment. The next time we will do this test we will not have to repeat this process.

For knowing the exact localization of the maximum diameter of the x ray field, it is required to perform one measurement where we obtain a chord of the circumference. The following measurements will be made on the bisector of this chord. Therefore we assure the dose profile is passing through the center of the radiation field.

Once we placed our dose profiler in the maximum diameter of the x ray field we measure the necessary time we take to displace our dose profiler from one end to the other of the x ray field. This measure it will be done automatically by our dose profiler which is connected to the data base of “Ocean”.

The software “Ocean” will perform the analysis of the measured parameter comparing automatically with the field size.

Finally, we can measure all the field sizes of our multifunction X-ray machine, repeating the last step.

CR METHOD
We place the CR on top of the II.
We make one shoot for each field size with the same CR.
We reveal the CR and download the image for processing it.

RESULTS

DOSE PROFILER METHOD
For analyze the results obtained with the dose profiler method, we have used the specific software of RTI “Ocean”. This analysis will indicate the length of each diameter from every field size as can be seen on figure 1.

This method allow us to measure in real time the size of the radiation field. In addition, it immediately performs the analysis of the results without the need for any type of processing. Another advantage that we obtain with this method is that it does not depend on the observer, since it is the software itself that automatically determines the radiation field size.

Finally, another advantage that we find is that the data is archived together with all the data obtained during the quality control, avoiding any loss of information.

CR METHOD
After reveal the CR, we obtain the resulting image and use the software Imagej for processing it.
The software Imagej allow us te measure the length of the diameter of each field. But this measure will depend on the observer who makes the measurement as long as it is not perform the FWHM to determine the exact point from which we should start to measure the size of the radiation field.

Finally, we introduce the results in our data base to be analyzed and attach the image to the final report.

The results of both measurements can be seen on table 1.
The differences between diameter field measured with CR and measurements with dose profiler are not higher to 4mm.

CONCLUSIONS

The results of this study demonstrate that the dose profile method is comparable to the CR method.

The main advantages are that outdated and more expensive methods requiring read-out after each scan can be avoided. In addition, the results are obtained immediately without the need to move from the room in which quality control is being carried out and the results are implemented in the database in which the other data are stored automatically.

The main disadvantage is careful placement of the dose profiler is required and it only gives diameter information instead of the whole field.

In conclusion, as the role of the medical physicist has expanded and the communication of the results of medical physics surveys has become a much more important part of medical physics practice. The ability of the medical physicist to properly communicate the results of his/her work is an important part of the professional practice of medical physics. For this reason we consider that the dose profiler method provide a results that are easier to implement in the final report and represent a lower risk of loss of information.

Figure 1:

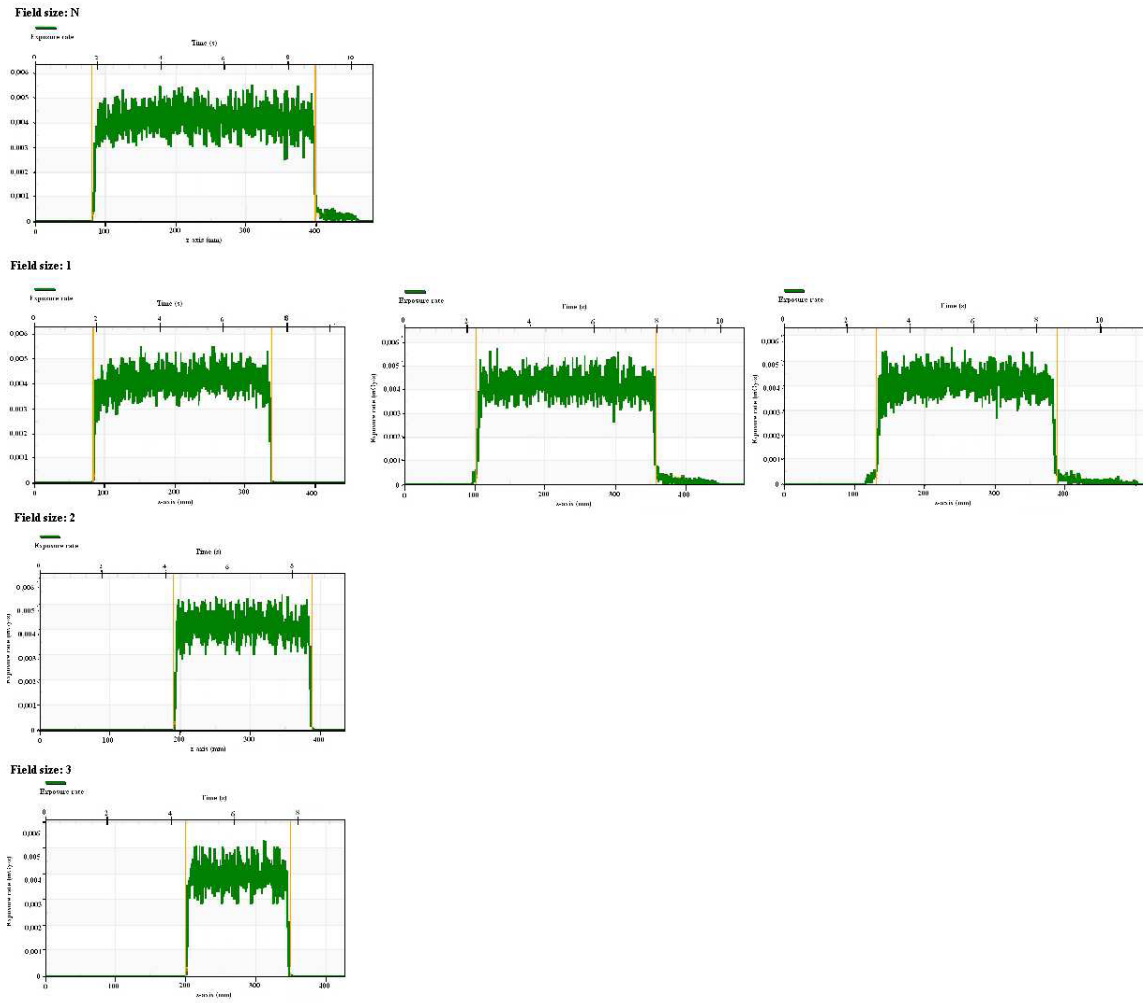

Table1:

Speed of table (mm/s)	Field size	Light field size	CR Field size (mm)	Measured field size (mm)	Diference CR-measurement (mm)
45,08	N	320	321	317	-4
45,08	1	260	257	257	0
45,08	1	260	257	257	0
45,08	1	260	257	255	-2
45,08	2	195	196	196	0
45,08	3	150	145	148	3

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Implementation of Regulation for Quality Control of Medical Uses of Ionizing Radiation in Korea

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Abstract. In this study, regulatory practices and approaches for quality control (QC) of radiotherapy and nuclear medicine procedures were introduced. Korean radiation safety regulation requires licensees, who use ionizing radiation in medical treatment purpose, to establish and implement comprehensive QC program, in order to ensure that radiation doses or admitted activities to patients are maintained as prescribed by physicians. As part of QC, the regulations require medical centers of radiotherapy to have independent external audit. Based on these provisions, the Nuclear Safety and Security Commission (NSSC) and the Korea Institute of Nuclear Safety (KINS) have established external audit teams and conducted the onsite audit for a total of 27 medical centers from 2015 to 2017. For items regarding dosimetry and mechanical accuracy of medial linear accelerators (LINACs), it has been confirmed that most of centers were within tolerance, for example, beam output constancy was within ±2%. In parallel with onsite audit, postal audit has been started from 2017. It was designed to verify output and check mechanical isocenter of linacs using glass dosimeter and film. In nuclear medicine practices, the accurate assay of activity of radiopharmaceuticals prior to administration is important process to assure that patients receive the correct prescribed dose. Dose calibrators have been used as the principal instruments to assay activity before administration. In 2017, the KINS surveyed status of usage and QC of dose calibrators to design appropriate regulatory approach. Totally 154 licensees participated in the survey voluntarily. According to the survey, only 46 licensees has conducted regular basis QC. From 2019, medical centers that use unsealed sources for treatment will be required to conduct day to day constancy test to verify operational performance of the instruments. KINS plans to check compliance through regulatory inspections.

KEYWORDS: regulation, quality control, radiotherapy, nuclear medicine, linac, dose calibrator

1 INTRODUCTION

Medical exposures have been dominant part of individuals' exposure against ionizing radiation. In view of radiological protection, principles of justification and optimization have to be applied to this exposure category. Generally medical uses of ionizing radiation are justified by weighing benefits of exposure against likelihood of the radiation detriment and considering alternative techniques that do not involve medical exposure. For each justified procedure, its exposure also should be optimized by keeping the exposure of patients to the minimum necessary to achieve the required medical objectives [1]. Among the medical uses of radiation, radiotherapy and nuclear medicine procedures deliver relatively high doses to patients, whole body or target organs (or tissues) as well. Therefore, for these procedures, optimization process is very essential in view of radiation protection and safety. As a part of optimization, comprehensive quality assurance (QA) for medical exposure is required to keep patients' doses as prescribed and to avoid unintended accidental exposure to patients and workers.
Based on the international recommendations and standards for radiation protection and safety of radiation sources, Korean regulatory framework for radiation safety and control introduced provisions to regulate medical exposures by principles of justification and optimization. According to the regulation on quality control (QC) of medical radiation, licensees who use radiation (sources) on human body in medical treatment purpose shall maintain the patient's exposure dose or administered radioactivity as prescribed by medical doctors [2]. This provision is mainly aimed to radiotherapy or nuclear medicine procedures.

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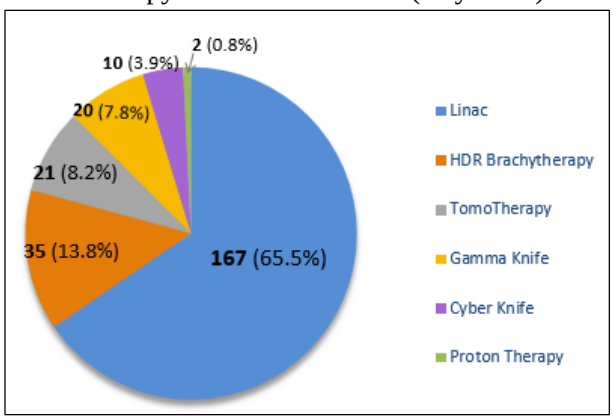
In this study, implementation of regulations on QC of radiotherapy was introduced, mainly focused on independent external audit for radiotherapy machines. Also regulatory approach to QC for nuclear medicine procedures was introduced. Efforts were being made to design effective regulatory approach. The main target was appropriate use of measurement instruments used to assay activity of radiopharmaceutical before administration.

2 IMPLEMENTATION OF REGULATION ON QUALITY CONTROL OF RADIOTHERAPY

2.1 Status of Radiotherapy Machines in Korea

As of July 2017, a total of 255 radiotherapy machines were used in 91 medical centers in Korea. There were six types of machines, such as linear accelerator (LINAC), high dose rate (HDR) brachytherapy, TomoTherapy, Gamma Knife, Cyber Knife and Proton Therapy. LINACs (167 units) accounted for about 65% of all the machines. Current status of radiotherapy machines was summarized in figure 1.

Figure 1: Current status of radiotherapy machines in Korea (July 2017)



For medical centers, which use radiotherapy machines, it is important to obtain qualified expert capable of comprehensive quality control of the machines maintenance and treatment planning. Currently (as of July 2017), 232 experts were registered in 91 medical centers. For linacs, newly introduced treatment methods using advanced and sophisticated technologies such as intensity modulated radiation therapy (IMRT), volumetric modulated arc therapy (VMAT), respiratory gating, could give higher doses to smaller target volume. Also, considering the increase of older machines (over 10 years) and higher energy machines (over 10 MV), quality control of linacs have been focused on regulations of medical radiation.

2.2 Independent External Audit of Linacs

For radiotherapy procedures, maintenance of radiotherapy machine, such as linacs, is required by the regulation on QC of medical radiation [2]. The relevant technical standards applied to radiation safety and control in medical uses require licensee to establish comprehensive QC program to ensure that all necessary procedures are developed and implemented to comply with the regulation [3]. The standards also specify items which should be included in the QC program and they are listed in table 1. These items are recommended with consideration of design and operational aspects of treatment machines and medical facilities.

For the implementation of the QC audit by independent external organizations recommended by the standards, regulatory bodies, Nuclear Safety and Security Commission (NSSC) and Korea Institute of Nuclear Safety (KINS) have been operating onsite external audit program since 2015. On a long term perspective, remote audit program was also considered for the sustainability of the independent external audit program. This program has been launched in 2017.

Table 1: Items should be included in the QC program [3]

(a) QC organization and their duties
(b) Equipment for QC
(c) Protocol, frequency, tolerance with action level for QC items
(d) Beam output calibration and evaluation of the radiotherapy machine
(e) Treatment planning and verification
(f) Education and training on QC personnel
(g) Independent external audit (every 3 years)

2.2.1 Onsite audit

For the onsite audit, regulatory bodies (NSSC and KINS) have organized the onsite audit team consisting of six qualified experts, mainly experienced health physicists. The team visited the designated medical centers accompanied with staffs of regulatory body and conducted QC procedures by field measurement using independent methods and tools, such as ion chambers and physical phantoms. They reviewed QC programs of centers and checked basic performance items such as mechanical/radiation rotation isocenter, light/radiation field coincidence, beam profile constancy, output constancy, physical wedge transmission factor constancy and photon energy constancy. Those items were selected by considering operation of linacs in three-dimensional conformal radiation therapy (3D CRT) mode. Organizing the team, operating the program, measurement methods and tolerance values of items were referred from recommended standards and protocols by international organizations and professional societies [4, 5]
From 2015 to 2017, the team had conducted onsite audits of a total of 27 medical centers (2015 : 2 centers, 2016 : 10 centers and 2017 : 15 centers) and confirmed that most of centers maintained their linacs within tolerance values. There were three cases exceeding tolerance values of light/radiation field coincidence (tolerance : ± 2mm), physical wedge transmission factor constancy (tolerance : ± 2%) and radiation rotation isocenter (tolerance : ± 1mm). By re-conducting of onsite audit and corrective actions by centers, those cases were also adjusted to be within the tolerance values.

2.2.2 Postal audit

It seemed impossible to conduct onsite audit for 91 medical centers in three year, even the centers using linacs have been increased. So paralleled with onsite audit, nuclear bodies considered to develop remote QC audit program using postal service (so called as postal audit). The postal audit was designed to evaluate photon beam output dose and mechanical accuracy of linacs using glass dosimeter, Gafchromic film and physical phantoms. There were efforts to set up secondary standards dosimetry laboratories (SSDL), including development of standard procedures for remote QC audit, design and manufacture of physical phantoms, establishment of dosimetry and mechanical accuracy measurement system traceable to the national standard and providing manuals for participants (field users in medical centers) [6].
In 2017, regulatory bodies conducted the postal audit for a total of 47 medical centers with 58 linacs. For output dose evaluation for photon energies (4, 6, 10 and 15 MV), all cases were confirmed within ± 5% of tolerance value. For evaluation of the mechanical accuracy, there were 4 cases (2 medical centers) exceeding tolerance values of collimator and gantry isocenter (tolerance : 2mm). For these medical centers, onsite audit has been planned in 2018. Results of the postal audit conducted in 2017 were summarized in table 2 and 3. It has been planned to conduct the postal audit for 46 medical centers with 109 linacs in 2018.

Table 2: Results of photon beam output dose evaluation of 45 centers (58 linacs, 114 beams)

	6 MV	10 MV	15 MV
Average (diff. %)	0.4	-0.5	-1.3
Min.value (diff. %)	-2.9	-3.4	-4.6
Max. value(diff. %)	3.9	3.4	1.1
Standard deviation	1.6	1.7	1.5

Table 3: Results of mechanical accuracy evaluation of 45 centers

	Field size (Y axis)	Field size (X axis)	Couch isocenter	Collimator isocenter	Gantry isocenter
Average (diff. mm)	-0.4	0.2	0.6	0.6	1.0
Min.value (diff. mm)	2.0	1.8	1.8	2.8	3.1
Max. value(diff. mm)	-1.9	-1.4	0.1	0.1	0.2
Standard deviation	0.70	0.74	0.31	0.5	0.6

3 REGULATORY APPROACH TO QUALITY CONTROL OF DOSE CALIBRATOR

3.1 Survey on Licensees Which Possess Dose Calibrators to Assay Radiopharmaceuticals

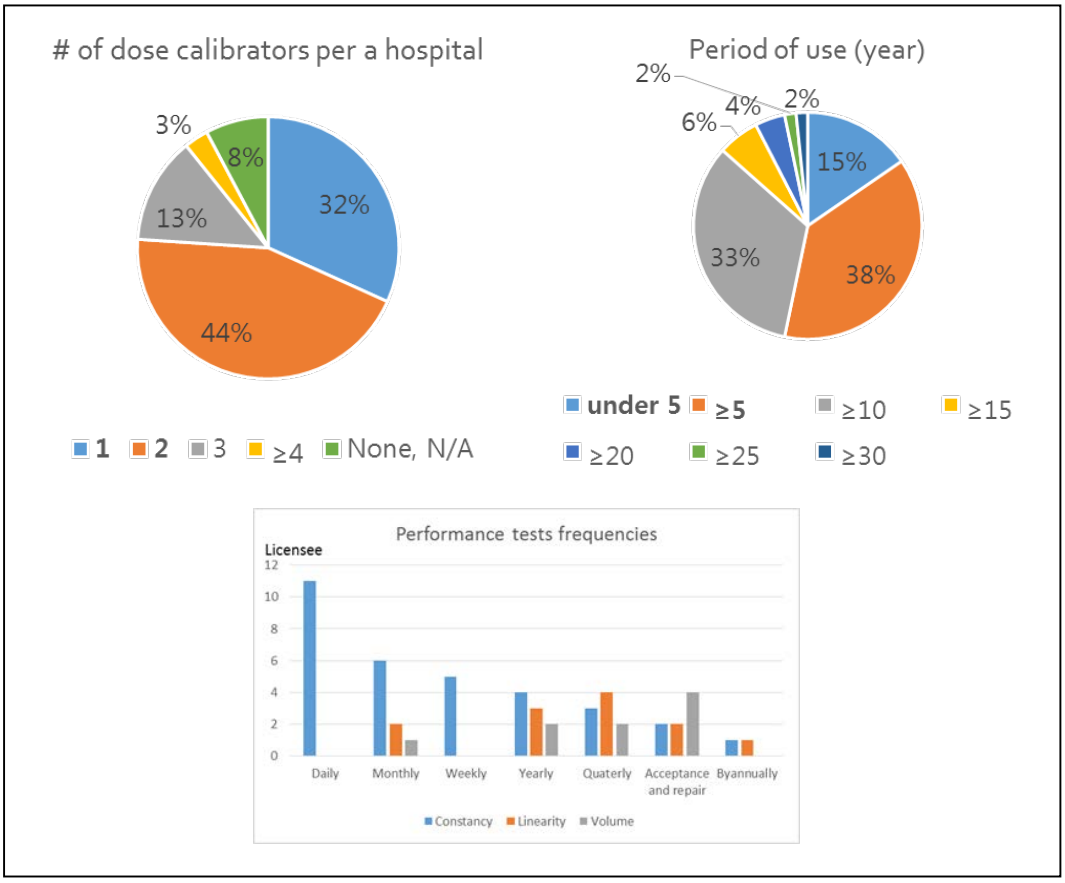
As mentioned above, domestic regulation on QC of medical radiation requires licensees to maintain the patients' administered radioactivity as prescribed by medical doctors [2]. The accurate assay of activity of radiopharmaceuticals prior to administration is important process to assure that patients receive the correct prescribed dosage [7]. Dose calibrators (or radionuclide activity calibrators) have been used as the principal instruments to assay activity before administration. To measure the activity, the devices utilize ionization chambers of the well-type directly coupled to an appropriate electronic circuitry and a direct readout in units of activity [8]. With regard to activity measurements in nuclear medicine, it is recommended that verification of the appropriate calibration and conditions of operation of the dose calibrators should be included in framework of quality assurance (QA) program [9].
Korea Institute of Nuclear Safety (KINS), surveyed condition of usage and QC of dose calibrators (2017.4.13 ~ 5.30). Among all registered 211 licensees of using or producing unsealed sources, 204 licensees (185 medical centers and hospitals with nuclear medicine practices and 26 organizations of production of unsealed sources with totally 336 dose calibrators) participated in the survey voluntarily. Main items of survey questionnaire were information of dose calibrators (numbers, manufacturer, model type, period of use, radioisotopes to measure) and QC program (possession of appropriate calibration sources, testing items and testing frequencies). The purpose of the survey was to estimate regulatory cost and to design appropriate regulatory approach in advance. The survey results were summarized in figure 2.

3.2 Status of Usage and QC of Dose Calibrators

According to the survey results, almost licensees had more than one dose calibrator. For only 21% of licensees (33 medical centers and hospitals and 13 licensees of production), QC of dose calibrators have been conducted on a regular basis. Also only 52 licensees (37 medical centers and hospitals and 15 manufacturers) possessed calibration sources which have metrological traceability to national standard. The calibration sources were mainly Ba-133, Co-57 and Cs-137. For Cs-137, 52 % of among them, the residual activities were not within the recommended activity range for daily QC (i.e., 3.7 - 7.4 MBq) [10]. Though 46 licensees answered they had conducted QC of dose calibrators, some routine performance test items, such as linearity, volume and activity correction, had been done by only 8 to 12 licensees. These results implied that regulatory intervention was required for appropriate use and QC of dose calibrators in nuclear medicine practices. Also it was needed to supports licensees with education

and training to enhance their understanding and to ensure appropriate level of QC for field instruments. Inter-laboratory comparison program of activity measurement would be a solution to resolve this problem [11].

Figure 2: Survey results on QC of dose calibrators (upper left : number of dose calibrators per a hospital, upper right : period of use (year), bottom : QC tests frequencies)



3.3 Proposed Regulatory Approach on QC of Dose Calibrators

From 2019, licensees that use unsealed sources for treatment to human body would be required to conduct day to day constancy test to verify operational performance of the instruments. For the test, licensees shall secure check sources, at least one or more. Test results shall be recorded with actions in case of exceeding tolerance level. In Korea, there were a technical standard by Korean Society of Nuclear Medicine (KSNM) and a guideline QC of dose calibrators by Korea Food and Drug Administration (KFDA) [10, 12]. The technical standard recommended to test the daily basis constancy (reproducibility) of performance of dose calibrators and endorsed QC procedures specified in the guideline. In the guideline, procedures applied to QC items of dose calibrator, such as constancy test, linearity of activity response, source geometry (volume) and accuracy (calibration) with correction of those factors, were provided in accordance with internationally recognized standards. The guideline listed recommended check sources with relevant activity ranges appropriate for the constancy test. They were characterized as long-lived sealed source emitting medium energy gamma ray. Also actions related with exceeding tolerance level of 5% and 10% were recommended. These standard and guideline would be supportive to the compliance with the regulation on QC of dose calibrators. Before the implementation of the regulation, there will be notice and guide by regulatory body. KINS plans to check the compliance of licensees through regulatory inspections.

4 CONCLUSION

In Korea, for last three years, independent external QC audit of radiotherapy machines, mainly linacs has been conducted as part of regulatory implementation. Originally it was started as onsite audit but remote QC audit program using postal service was also launched for more effective implementation and sustainability of the external audit program in long term perspective. Through operating the external audit programs, it was confirmed that linacs were maintained with appropriate accuracy.
From 2019 regulatory intervention would be expected to improve QC of nuclear medicine practices. The main target would be appropriate use of dose calibrators, the main instruments used to measure activity of radiopharmaceutical before administration to a patient. Implementation of new regulation would be noticed in advance and compliance of licensees would be included in regulatory inspection items.
QC has been essential part to help licensees ensure successful optimization of protection and safety in medical uses of radiation. It was expected that regulatory intervention on QC of radiotherapy and nuclear medicine would promote a prospective and iterative process of the optimization in those practices.

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Eye lens dosimetry for interventional procedures in cardiology

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Abstract. The eye lens is one of the most sensitive organs for radiation injury and exposure might lead to radiation induced cataract. The International Commission on Radiological Protection (ICRP) recommended a reduction in the annual dose limit for occupational exposure for the lens of the eye from 150 to 20 mSv, averaged over a period of 5 years. This prospective study investigated eye lens dosimetry in two medical institutions in Warsaw during a routine year of professional activity. The radiation exposure measured in a normal working schedule of the intervention caradiologists during 3 months and this cumulative eye lens dose was extrapolated to 1 year. Body and skin dose measurements were also performed, estimating $H_p(10)$ and $H_p(0.07)$ values. The eye lens doses were measured in terms of the dose equivalent $H_p(3)$ with the eye dosimeter with thermoluminescent detectors, close to the left eye (placed on the temple). The estimated annual eye lens doses for about 50 persons , range from a minimum of 0.1 mSv to a maximum of 3.0 mSv with an average dose of 1.0 mSv.

This study demonstrated that the estimated annual eye lens dose is well below the revised ICRP's limit of 20 mSv/year. There is probably the correlation between doses on the skin (measured by the ring termoluminescent dosimeter) and doses on the eye lenses of the staff (numerical numbers not captured due to too few results), but further research is required. With radiation induced cataract being explained as a possible stochastic effect, without a threshold dose, radiologists and anesthesiologists who regularly work in a radiological environment should remain care and vigilant. This persons should also maintain radiation safety standards at all times, which includes adequately protective equipment, keeping distance, routine monitoring and periodic training in occupational safety.

KEYWORDS: Eye-lens dosimetry, Hp(3), intervention cardiology

1. INTRODUCTION

The lens of the eye is one of the most radiant tissues in the human body.

Detectable lesions in the lens can be seen at doses of at least an order of magnitude smaller than observed changes in other eye tissues. Initially, these changes do not cause visual impairment, but their severity tends to increase gradually with dose and time. Consequently, they can lead to partial or total opacity of the lens. Recent reports and publications show an increased risk of cataracts in workers exposed to ionizing radiation. Previously it was thought that the threshold limit is 2-5 Gy, the recent results indicate that radiation cataract occurrence after exceeding the dose of 0.5 Gy, and the most radical suggests even a non-threshold induction of radiation cataract.

Studies on the occurrence and induction of eye lens cataracts have clear consequences primarily for staff performing interventional radiology treatments. Probably the source of the highest doses to which medical staff are exposed remains radiology procedures, despite the widespread use of radiation shielding. These are all therapeutic and diagnostic procedures performed through the skin of a patient under local anesthesia or general anesthesia, and using fluoroscopic imaging for localization of lesions and for monitoring radiological medical procedures, as well as for monitoring and documentation of therapy.

The characteristic of radiotherapy is the relatively high doses of radiation that both patients and medical personnel are exposed to. Statistical evaluations show that 5 to 15% of patients undergoing such treatment have markedly different skin lesions.

The staff performing the treatments remains close to the source of X-rays and in the high range of scattered radiation, for several hours a day during the treatment which causes a high risk of occupational exposure. It is important that the primary source of human exposure is diffuse radiation produced in the patient.

The $H_p(3)$ value was not used from the very beginning in radiological protection, as opposed to $H_p(10)$ and $H_p(0.07)$ values , which were used widely for many years. One of the reasons for the lack of dedicated the $H_p(3)$ dosimeters was the high limit of the annual dose limit, so there was no need for monitoring and no reliable KAIR/$H_p(3)$ conversion factors.

The ICRU publications state that "monitoring of $H_p(3)$ will be required only in exceptional circumstances", because it is assumed, that the dose limit equivalent to the eye level will not be exceeded when the limit of personal dose equivalent dose limits (20 mSv per year) and the dose equivalent to the skin (500 mSv per year) are not exceeded.

2. METHODS

The International Commission on Radiation Protection ICRP has recommended reducing the current annual eye dose limit from 150 mSv to 20 mSv [ICRP 118, 2012].
This limit was approved and entered in the new European Union Directive [Euratom, 2013].
As a consequence of the reduction of this limit, it was necessary to monitor the dose equivalent on the lens of the eye, using the operating value $H_p(3)$, routinely not measured so far [ICRP 103, 2007; IAEA-TECDOC-1731, 2013].

In response to recommendations regarding the need to measure doses on the lens and to lower the dose limit, work has begun on the development of eye dosimetry methods, involving the search for a suitable dosimeter, phantom, and calibration procedures. The first work in this direction has been undertaken within the framework of the European ORAMED (Optimization of Radiation Protection of Medical Staff) project [www.oramed-fp7.eu/]. Designed in the ORAMED project in 2011, dosimeter is named EYE-DTM and consists of a plastic housing with a polyamide capsule housed in a high-detector MCP-N (LiF: Mg, Cu, P) with a diameter of 4.5 mm and a thickness of 0.9 mm (Fig. 1).

Figure 1: The EYE-DTM eye dosimeter (also in version with headband)

3. RESULTS

In 2015, have been carried out in CLOR the works to determine the energy characteristics of thermoluminescent detectors used for eye lens dosimetry and to develop the method for estimating exposure of the eye lens to ionizing radiation using an individual dose equivalent $H_p(3)$.

The aim of the following pilot studies was implementation and testing new methods and dosimeters to measure eye doses. Within the framework of the study, detailed measurements of ionizing radiation doses and exposure assessment of workers in interventional radiology were performed - with particular attention to eye lens doses. Body and skin dose measurements were also performed, estimating $H_p(10)$ and H_p (0.07).

The measurements were performed in two medical institutions in Warsaw: the Mazovia Brodnowski Hospital (Hemodynamic Laboratory) and the Angiography Laboratory (Department of Diagnostic Imaging) of the Institute-Memorial Children's Health Center.

The Hemodynamics Laboratory of the Mazovian Brodnowski Hospital was established in November 2004. It is equipped with a modern digital a angiograph Philips Allura type (70 kV, 480 mA).

There are about 1200 coronary angiography and approx. 700 coronary angioplasty procedures annually.

The laboratory performs, among others: coronary angiography, coronary artery disease, coronary stent implantation.

Table 1: The evaluation of effective whole-body (Hp (10)), skin (Hp (0,07)) and eye lens (Hp (3)) doses for the staff the Hemodynamics Laboratory of Mazowian Brodno Hospital in Warsaw

Employee	$H_p(3)$ [mSv]	$H_p(10)$ [mSv]	$H_p(0.07)$ [mSv]
1	0,21	X[(a)]	X
2	Y[(b)]	X	X
3	0,02	X	0,44
4	Y	X	X
5	Y	X	X
6	0,64	X	X
7	Y	X	X
8	Y	X	X
9	2,26	X	6,2
10	Y	X	X
11	0,46	0,31	1,24
12	0,28	X	X
13	Y	X	X
14	0,08	X	X
15	0,73	x	0,59
16	Y	X	0,29
17	Y	X	X
18	Y	X	X
19	0,41	0,20	X
20	0,39	X	X
21	0,41	0,20	X

[(a)] X=limit of detection (0,1 mSv) [(b)] Y=limit of detection (0,02 mSv)

Table 2: The evaluation of effective whole-body (Hp (10)), skin (Hp (0,07)) and eye lens (Hp (3)) doses for the staff the Center for Cardiac Catheterization and Angiography in the Memorial of the Children's Health Center (IPCZD)l in Warsaw

Employee	$H_p(3)$ [mSv]	$H_p(10)$ [mSv]	$H_p(0,07)$ [mSv]
1	Y[(a)]	X[(b)]	X[(b)]
2	Y	X	X
3	Y	X	X
4	Y	X	X
5	0,33	X	X
6	Y	X	X
7	Y	X	X
8	Y	X	X
9	Y	X	X
10	Y	X	X
11	Y	X	X
12	Y	X	X
13	0,19	X	X
14	Y	X	X
15	Y	X	X
16	Y	X	X
17	Y	X	X
18	Y	X	X
19	Y	X	X
20	0,03	X	X
21	Y	X	X
22	0,13	X	X
23	Y	X	X

[(a)] Y=limit of detection (0,02 mSv) [(b)] X=limit of detection (0,1 mSv)

The Center for Cardiac Catheterization and Angiography is located in the Department of Diagnostic Imaging of the Institute - Memorial of the Children's Health Center (IPCZD), where under the control of X-rays are performed so-called invasive cardiovascular diagnostic tests and therapeutic treatments in the field of cardiology and interventional radiology in children, patients in clinical departments of IPCZD. The diagnostic examinations (so called angiographies) are intended to illustrate the vascular architecture of the pathological organs (e.g. central nervous system, kidney) and, in the case of the heart, so-called cardioangiography, visualization of individual heart cavities and abnormalities of the internal organs.

The results are presented in Tables 1,2.

4. CONCLUSIONS

The interventional cardiology causes the radiation risk exposure of eye lens (doses have been registered both in persons directly performing treatments and the accompanying persons too. In the group of surveyed employees, no dose limit for the exposed workers was exceeded, the value did not exceed 3 mSv per year).

There is probably the correlation between doses on the skin (measured by the ring dosimeter) and doses on the eye lenses of the staff (numerical numbers not captured due to too few results), but further research is required.

With radiation induced cataract being explained as a possible stochastic effect, without a threshold dose, radiologists and anesthesiologists who regularly work in a radiological environment should remain care and vigilant. This persons should also maintain radiation safety standards at all times, which includes adequately protective equipment, keeping distance, routine monitoring and periodic training in occupational safety.

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Dose reduction in proton therapy from accurate proton energy loss prediction using dual energy CT.

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Abstract. Proton therapy is part of radiotherapy and increasingly applied in treatment of cancer, especially for children and patients with tumours in the head and neck region. With proton therapy the tumour can be irradiated with less damage to the surrounding healthy tissues and critical structures compared to irradiation with photons. To optimally exploit this benefit of protons, the energy transferred by the protons to the tissues (the dose distribution) must be calculated very accurately. For this, the specific energy loss of the protons for each tissue is determined based on x-ray computed tomography (CT) imaging. In clinical practice, a phenomenological model is used based on an image obtained with a single x-ray spectrum (single energy CT, SECT). The predictions of this model are not patient specific and very inaccurate for materials which differ in composition and density from the materials used for determination of the model parameters. We have developed a method using two x-ray spectra (dual energy CT, DECT). With this method the electron densities and effective atomic numbers, which determine the specific energy loss of protons in a material, are derived from two images on basis of fundamental theory of the interactions of x-rays. This method provides patient specific predictions with an accuracy better than 2%. This is a large improvement in accuracy and stability of the method with respect to the clinically applied SECT method and can reduce the risk of high doses to healthy tissue or low doses to the tumour due to inaccurate prediction of proton energy loss in the tissues on the beam path.

KEYWORDS: dose reduction, proton therapy, dual energy CT, proton stopping power

1 INTRODUCTION

In treatment planning for proton therapy, single energy computed tomography (SECT) stoichiometric calibration methods are used to determine stopping powers of tissues relative to that of water (RSPs). In analytical treatment planning systems this specific energy loss of protons is derived from the CT number with a calibration curve based on tissue substitute measurements or calculations for average tissue compositions [1, 2]. Monte Carlo dose calculation requires the mass density and elemental mass fractions which can be assigned to CT number bins based on modelling of average tissue compositions and densities [3]. The linear attenuation coefficient measured in CT depends on the electron density and electronic cross section which is dependent on the elemental composition. Due to this twofold dependence, materials with different composition and density can have the same CT number in a single energy CT scan. This shows the main disadvantage of SECT based methods, namely their lack of specificity. In addition, beam hardening affects the measured CT numbers which can cause deviations from the SECT calibration curve.

DECT provides linear attenuation coefficients for two photon spectra and thus results in extra information compared to SECT using only one photon spectrum. This advantage of DECT can be exploited by using a theoretical framework, including an accurate parameterization of the photon-electron interaction cross section, which describes the dependence of the linear attenuation coefficient

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on the electron density and atomic number as a function of photon energy. Using spectral weightings specific for the tube potentials applied for imaging with a particular CT system, the theoretical framework allows solving for the effective atomic number and electron density from the two measured linear attenuation coefficients. This is a natural parameterization of this twofold dependence of the linear attenuation coefficient. By implementing an iterative optimization procedure which changes the spectral weighting of the parameterization of the cross section using the obtained effective atomic number and electron density, electron densities are determined with high accuracy (~1%) [4]. The derived electron density has a clear physical meaning and is directly relevant for proton dose calculations. The effective atomic number is more difficult to interpret. This number incorporates the *Z* dependence of the photon interaction processes. A systematic deviation between measured and calculated effective atomic numbers has been observed for materials of known composition. This deviation is attributed to a phenomenological beam hardening correction for water in the CT reconstruction process [4]. The tissue electron density provides the largest contribution to the proton stopping power. In addition, the proton stopping power depends on the mean excitation energy of the tissue. The effective atomic number corrected for the water beam hardening correction correlates well with the mean excitation energy. With an empirical relation effective atomic numbers from DECT can be converted into mean excitation energies [5]. The electron density and this relation between the effective atomic number and mean excitation energy can be used to calculate the proton stopping powers needed for dose calculations from DECT measurements.

In this study we present a comparison of relative proton stopping power predictions from SECT and DECT with experimental data for a large set of well characterized materials and bovine tissues.

2 METHODS

The RSPs derived from SECT and DECT have been compared to RSPs measured with 149 MeV protons (at the entrance of the sample) at the AGOR cyclotron [6]. The RSPs have been experimentally determined for 32 materials with known composition and density and for 17 bovine tissues.

2.1 Single energy CT calibration for RSP calculation

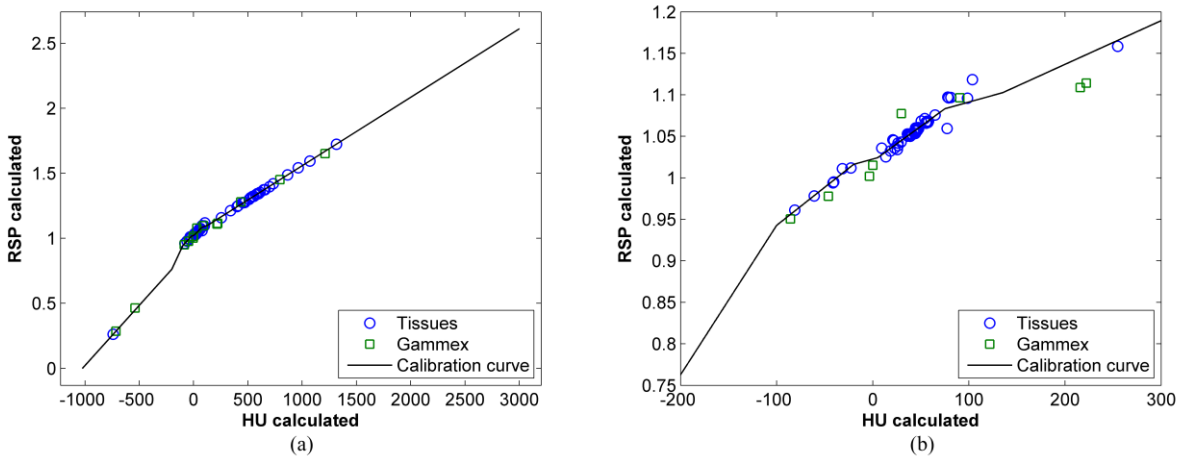
The SECT based approach described by Schneider *et al.* [3] has been followed to calculate CT numbers for 81 average tissues [7, 8]. Values for the scanner and energy specific calibration coefficients k_1 and k_2 have been determined from SECT data of the Gammex 467 tissue characterization phantom (Gammex Inc., Middleton, WI, USA) measured on a dual source CT (DSCT) system (SOMATOM Force, Siemens Medical Solutions, Forchheim, Germany). The SECT data has been acquired at 120 kV (147 mAs) ($CTDI_{vol}$ = 9.9 mGy) using an abdomen protocol in spiral mode with a collimation of 192x0.6 mm. The data has been reconstructed with a Qr40 strength 5 advanced modeled iterative reconstruction (ADMIRE) kernel with a slice thickness of 1 mm. From the measured average CT numbers of the 16 Gammex materials the coefficients k_1 and k_2 have been determined using the *lsqnonlin* function in MATLAB (MATLAB 8.3, The MathWorks Inc., Natick, MA, USA). A least squares solution has been found where $k_1 = 1.0377\text{x}10^{-5}$ and $k_2 = 2.4736\text{x}10^{-5}$ ($R^2 = 0.9998$). With these coefficients the calculated CT numbers for the 16 Gammex materials were found to be within 13 Hounsfield units (HU) of the measured CT numbers. Subsequently, CT numbers for the average tissues have been calculated using the derived coefficients. Relative proton stopping powers (RSPs) for the average tissues have been calculated using the approximation

$$\frac{S_m}{S_w} = \frac{\rho_e^m}{\rho_e^w}\left(\frac{\ln\left(\frac{2m_e c^2\beta^2}{1-\beta^2}\right) - \beta^2 - \ln\langle I_m\rangle}{\ln\left(\frac{2m_e c^2\beta^2}{1-\beta^2}\right) - \beta^2 - \ln\langle I_w\rangle}\right) \tag{1}$$

where ρ_e^m/ρ_e^w is the relative electron density, $m_e c^2$ the electron rest mass with c the speed of light in vacuum, $\beta = v/c$ with v the proton velocity, $\langle I_m\rangle$ the mean excitation energy of the material calculated with Bragg's additivity rule and $\langle I_w\rangle$ the mean excitation energy of water for which a value of 78 eV

has been adopted from Sigmund *et al.*[9]. The RSPs of the average tissues have been calculated at a proton energy of 143.5 MeV which corresponds to the average energy in our samples with a water equivalent thickness of about 2 cm at an entrance energy of 149 MeV. Fig. 1 presents the calibration curve used for the SECT analysis which has been composed of 4 linear fits of the tissue data in HU intervals of [-1024, -200] lung, [-100, -20] adipose, [5, 75] organs and muscle and [135, 3000] bone [1]. The HU intervals have been connected by linear interpolation of the corresponding RSPs. CT images of the samples acquired at 120 kV have been converted to RSP images using this CT number to RSP calibration curve.

Figure 1: (a) Single energy CT calibration curve based on the average tissues [7,8] which relates calculated CT numbers (HU) [3] to calculated relative stopping powers (RSPs). (b) Enlarged view of the fits to the adipose, organs and muscle and bony tissues.



2.2 Dual energy CT for RSP calculation

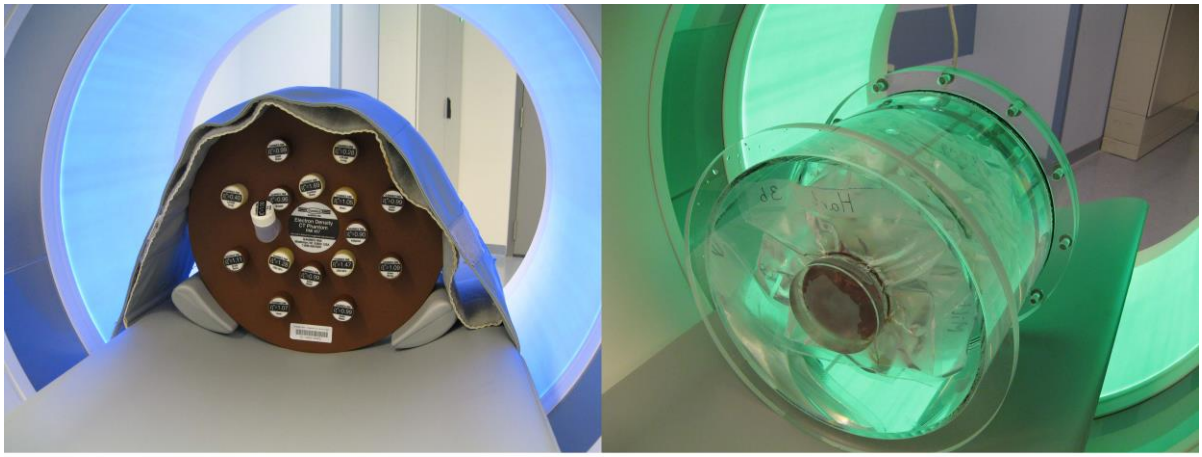
Dual energy CT provides linear attenuation coefficients derived from measured CT numbers at two different spectral energy distributions (high and low kV setting) for each voxel. Using the method presented by van Abbema *et al.* [4] effective atomic number and electron density images have been derived from DECT data. The effective atomic number images have been calibrated for not adequately corrected beam hardening in the CT reconstruction process. A relation has been established between this effective atomic number and the logarithm of the mean excitation energy [5]. Relative stopping power images have been determined from the images of the relative electron density and the mean excitation energy derived from the effective atomic number at a proton energy of 143.5 MeV and with an ICRU recommended value for $\langle I_w \rangle$ of 78 eV [9]. Alternatively, a value for $\langle I_w \rangle$ of 69 eV could have been selected which corresponds to the value obtained with Bragg's additivity rule used by the SECT and DECT methods. This would lead to a reduction in the RSPs predicted by SECT and DECT of 1.5%.

2.3 Comparison SECT and DECT

The SECT and DECT data of the sample materials and bovine tissues has been acquired on a DSCT system (SOMATOM Force). Inside a Gammex 467 tissue characterization phantom (diameter 33 cm) the different sample materials (diameter 2.85 cm) have been distributed over the phantom in different configurations (Fig. 2a). From a variety of bovine organs and tissues 23 different tissue samples have been prepared. The two bone samples have been obtained from pulverizing a bovine shoulder blade. A cortical bone sample contains the bone cortex fragments while a mixed bone sample includes cortical and cancellous bone. The mass densities of these bone samples are smaller than tabulated densities for bone [8]. The tissues have been pressed in cylindrical dishes (diameter 8.7 cm, thickness around 2 cm) after which they were vacuum sealed in microchannel bags to conserve and preserve their thickness and

shape. A CT water phantom (diameter 30 cm) has been developed which centrally aligns the tissues (Fig. 2b).

Figure 2: CT scan setup of the (a) Gammex phantom with different insert configurations (shown is the default configuration) and (b) CT water phantom with tissues vacuum sealed in cylindrical dishes and microchannel bags.



(a) (b)

The phantoms have been scanned with SECT at 120 kV (147 mAs) using the same settings as applied for the calibration scan. With DECT the phantoms have been scanned using a clinical 90 kV / 150 kV Sn dual energy abdomen virtual noncontrast (VNC) protocol in spiral mode with a collimation of 64x0.6 mm with 266 mAs (90 kV) and 166 mAs (150 kV Sn) ($CTDI_{vol}$ = 15.5 mGy). For reconstruction of both SECT and DECT data, a Qr40 strength 5 ADMIRE kernel and a slice thickness of 1 mm have been used for a field of view of 35 cm.

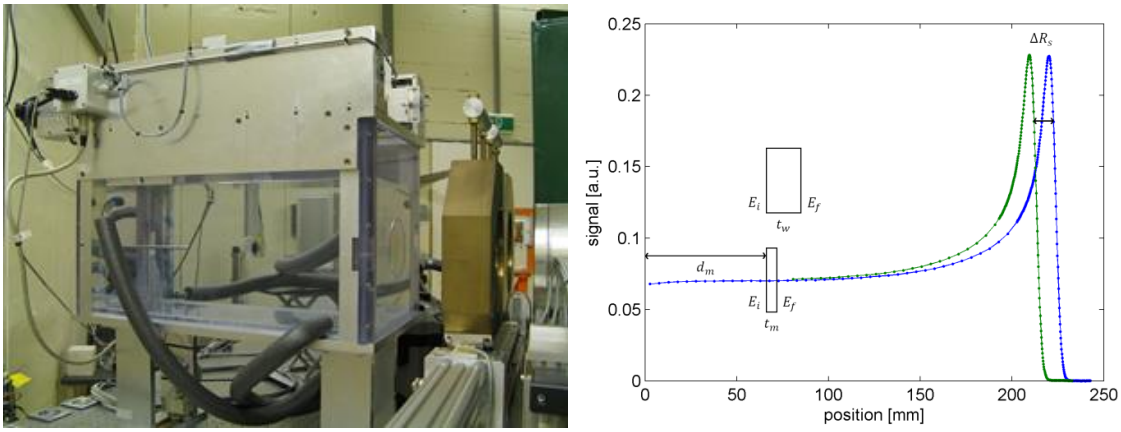
From the reconstructed SECT and DECT data central slices of the sample materials have been selected for which RSP images have been calculated. The mean and standard deviation in the RSPs have been determined by drawing circular regions of interest (ROIs) in the RSP images derived from SECT and DECT, respectively.

For the bovine tissues, circular ROIs have been selected with a diameter of 5.3 mm at the center of the sample in each slice of the tissue. The diameter of 5.3 mm corresponds to the diameter of the electrode in the Markus ionisation chamber used for the proton measurements. The ROIs of all slices in the tissue have been combined to obtain a cylindrical volume of interest (VOI) with a thickness around 20 mm (depending on the tissue thickness). For each voxel in the VOI, RSPs have been calculated for SECT and DECT after which for each slice the mean and standard deviation have been calculated. The mean RSP of the tissue has been derived from the RSPs in all slices of the tissue. Based on analysis of the CT images and CT number profiles with depth in the tissue, 6 of the 23 bovine tissues have been excluded from further analysis due to their inhomogeneity.

2.4 Proton relative stopping power measurements

Experimental RSPs for the sample materials and bovine tissues have been determined from measured residual ranges relative to water and sample thicknesses (Fig. 3). The proton measurements of the bovine tissues have been performed within 36 hours of the CT measurements and within 48 hours from slaughter. During the proton and CT experiments the temperature of the water in which the tissue samples have been measured has been stabilized at 21ºC.

Figure 3: Experimental setup and concept of the water equivalent thickness t_w (WET) derived from the measured range difference ΔR_s and the material thickness t_m. ΔR_s is the difference in $R_{80\%}$ (distal 80% of the dose) between the measurement in water only (blue) and the measurement with a material positioned at depth d_m in the water (green). t_w is the thickness of water that results in the same energy loss $(E_f - E_i)$ as in the material with thickness t_m. The relative stopping power (RSP) is given by the ratio t_w/t_m which equals $(t_m - \Delta R_s)/t_m$.



3 RESULTS AND DISCUSSION

High accuracy measurements of proton depth dose distributions have been performed with a water phantom. The measured range in water has been determined with an uncertainty < 0.03 mm. Relative proton stopping powers have been determined from depth dose measurements with an uncertainty < 0.4% for the 32 materials. For this study, the achieved accuracy (< 0.4% and for most materials 0.2%) is sufficient to provide sub percent ground truth relative stopping powers for comparison with values derived from DECT and SECT. The measured relative stopping powers for the 32 materials and the differences with the predicted values based on the DECT and SECT images are presented in Table 1.

For the proton experiment with the bovine tissues the uncertainty in the $R_{80\%}$ in water is < 0.04 mm and is only a minor contribution to the uncertainty in the experimental RSPs of the tissues. This uncertainty is dominated by inhomogeneities in the tissues, in particular small air cavities. Therefore, inhomogeneous tissues with air cavities recognized on the CT images have been excluded from the analysis. Rietzel *et al.*[10] corrected their experimental data for air mainly on the borders of the tissue samples. We consider correcting for air cavities doubtful since the RSP measurement is highly sensitive for air and the method for air correction can strongly influence the resulting RSPs. The relative stopping powers predicted by DECT and SECT for the selected 17 bovine tissues and the differences with the measured values are given in Table 2.

In this study the ICRU recommended value for the mean excitation energy of water of 78 eV [9] is used. For mean excitation energies of the elements the ICRU recommended values of ICRU report 37 [11] are used. Bragg's additivity rule with these elemental mean excitation energies gives a mean excitation energy for water of 69 eV. This is not consistent with the ICRU recommended value of 78 eV. A decrease from the applied 78 eV for water to 69 eV yields a reduction of 1.5% in relative stopping powers. Applying this 69 eV as the mean excitation energy for water would be consistent with the approach used in the SECT and DECT methods. Consequently, an improved correspondence of the DECT predicted relative stopping powers with the experimental values can be observed (Fig. 4). For more accurate relative stopping power prediction, an accurate determination of elemental mean excitation energies and verification of Bragg's additivity rule for compounds (including water) is required.

Table 1: Mass densities ρ_m, relative electron densities $\rho_{e,m}{}'/\rho_{e,w}$ measured with DECT and relative differences with calculated values for 32 materials. Calibrated effective atomic numbers Z'_{cal} from DECT determined values for Z'. Relative stopping powers (RSPs) derived from SECT and DECT (with $\langle I_w \rangle$ of 78 eV) compared to the experimental RSPs determined with uncertainty < 0.4% at a proton energy of 143.5 MeV (average in sample).

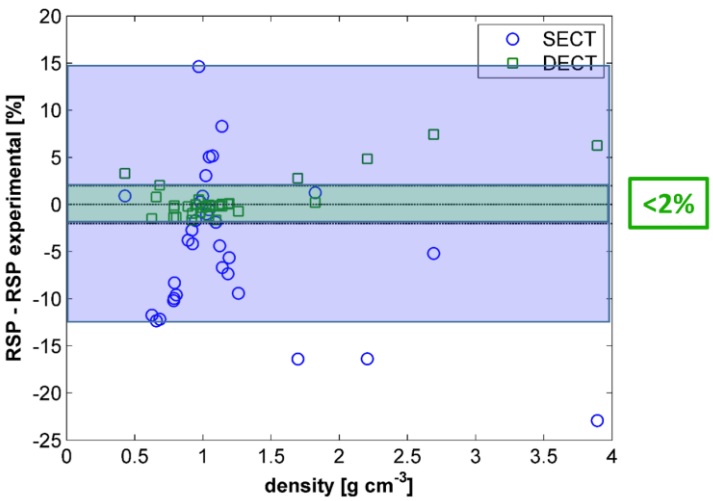
No.	Material	ρ_m [g cm^{-3}]	$\rho_{e,m}'/\rho_{e,w}$ meas	$\rho_{e,m}'/\rho_{e,w}$ meas-calc [%]	Z'_{cal}	RSP SECT	RSP DECT	RSP Exp	RSP SECT-Exp [%]	RSP DECT-Exp [%]
1	LN-450 Lung	0.428	0.431	3.2	7.52	0.431	0.441	0.421	2.4	4.8
2	AP6 Adipose	0.946	0.934	0.5	6.23	0.949	0.965	0.951	-0.2	1.5
3	BR-12 Breast	0.981	0.965	0.5	6.96	0.996	0.986	0.979	1.7	0.7
4	Solid Water M457	1.045	1.019	0.2	7.38	1.043	1.036	1.033	1.0	0.3
5	LV1 Liver	1.095	1.067	0.1	7.55	1.080	1.083	1.084	-0.4	-0.1
6	SB3 Cortical Bone	1.823	1.712	0.7	14.01	1.666	1.649	1.622	2.7	1.7
7	n-Pentane	0.626	0.654	-0.6	6.01	0.614	0.684	0.684	-10.2	0.0
8	n-Hexane	0.659	0.689	-0.2	5.67	0.639	0.733	0.717	-10.9	2.2
9	n-Heptane	0.683	0.720	0.8	5.37	0.661	0.766	0.740	-10.7	3.5
10	Methanol	0.791	0.803	0.1	6.54	0.760	0.827	0.816	-6.9	1.3
11	Ethanol	0.788	0.804	0.1	6.24	0.752	0.832	0.822	-8.5	1.2
12	Propan-1-ol	0.805	0.817	-0.6	6.39	0.773	0.842	0.841	-8.1	0.1
13	Propan-2-ol	0.785	0.798	-0.5	6.36	0.750	0.823	0.822	-8.8	0.1
14	Oleic acid	0.892	0.900	-0.1	6.10	0.899	0.932	0.920	-2.3	1.3
15	Ethyl acetoacetate	1.026	0.992	-0.4	6.67	1.008	1.016	1.003	0.5	1.3
16	Water	0.998	0.996	-0.4	7.39	1.024	1.011	1.000	2.4	1.1
17	Polyethylene glycol 200	1.123	1.100	-0.4	6.56	1.082	1.130	1.115	-3.0	1.3
18	Glycerol	1.260	1.221	-1.1	6.80	1.140	1.248	1.238	-7.9	0.8
19	Silicone oil Siluron 5000	0.970	0.948	0.4	10.75	1.076	0.945	0.926	16.2	2.1
20	Potassium Chloride 4.01%	1.021	1.017	-0.1	8.26	1.060	1.027	1.014	4.5	1.3
21	Potassium Chloride 7.71%	1.046	1.036	-0.2	9.31	1.093	1.040	1.025	6.6	1.5
22	Potassium Chloride 11.13%	1.070	1.056	0.0	10.11	1.113	1.057	1.043	6.7	1.3
23	Potassium Chloride 20.03%	1.139	1.114	0.2	11.85	1.191	1.101	1.084	9.9	1.6
24	Carbon graphite	1.696	1.536	0.4	6.21	1.292	1.583	1.518	-14.9	4.3
25	UHMWPE	0.923	0.958	0.8	5.74	0.968	1.000	0.994	-2.6	0.6
26	Polypropylene	0.919	0.946	0.0	6.37	0.967	0.977	0.979	-1.2	-0.2
27	Nylon 6.6-101	1.142	1.124	-0.4	6.14	1.089	1.163	1.148	-5.1	1.3
28	PMMA	1.183	1.151	0.0	6.42	1.099	1.186	1.168	-5.9	1.5
29	Polycarbonate	1.192	1.128	-0.5	6.50	1.093	1.159	1.140	-4.1	1.7
30	Teflon	2.205	1.885	-1.3	8.30	1.522	1.902	1.788	-14.9	6.4
31	Aluminium $AlMgSi_1$	2.691	2.383	1.8	13.27	2.050	2.319	2.129	-3.7	8.9
32	Al_2O_3 99.7%	3.892	3.452	0.2	10.61	2.513	3.446	3.198	-21.4	7.8

The stoichiometric single energy CT calibration method relates a CT number measured at a single x-ray tube voltage to a relative stopping power by deriving a composition and density or directly using a calibration curve. This is not specific since different tissues can have the same CT numbers on a SECT image but differ in composition, density and relative stopping power. The validity of the SECT method is limited to materials and tissues with a composition as well as density very similar to the materials used for the calibration. Dual energy CT tissue characterization allows a physics based determination of the electron density the accuracy of which is independent of the materials. With these electron densities and mean excitation energies derived from the calibrated effective atomic numbers, relative stopping powers have been determined which are close to experimental values. Larger differences have only been found for materials without hydrogen in their composition. The deviation for LN-450 is due to excluded air voxels in the image contributing to the density. The relative differences with respect to experimental relative stopping powers are between -21 and 16% for SECT while for DECT the differences are mainly within 2% for the 32 assessed materials (with $\langle I_w \rangle$ of 69 eV, Fig. 4). The differences for 17 bovine tissues are within 3.5% for SECT and 1.8% for DECT of experimental values (with $\langle I_w \rangle$ of 69 eV), except for the bone samples. SECT proved to be more susceptible for partial volume averaging effects between e.g. air and bone in the bone samples and yields large deviations (~20%) in these cases.

Table 2: Estimated mass densities ρ_m, DECT measured relative electron densities $\rho_{e,m}{}'/\rho_{e,w}$ and calibrated effective atomic numbers Z'_{cal} of 17 bovine tissues. Relative stopping powers (RSPs) derived from SECT and DECT (with $\langle I_w \rangle$ of 78 eV) are compared to experimentally determined RSPs at a proton energy of 143.5 MeV (average in tissue).

No.	Tissue	ρ_m [g cm^{-3}]	$\rho_{e,m}'/\rho_{e,w}$ meas	Z'_{cal}	RSP SECT	RSP DECT	RSP Exp	RSP SECT-Exp [%]	RSP DECT-Exp [%]
1	liver (1)	1.151	1.063	7.24	1.070	1.082	1.051	1.8	2.9
2	kidney	1.096	1.035	7.26	1.057	1.054	1.020	3.6	3.3
3	heart (1)	1.042	1.043	7.28	1.059	1.061	1.041	1.7	1.9
4	lung	0.628	0.452	7.18	0.435	0.457	0.444	-2.0	2.9
5	muscle	1.035	1.052	7.09	1.061	1.073	1.046	1.4	2.6
6	thymus	1.058	1.036	7.06	1.051	1.057	1.030	2.0	2.6
7	heart thymus	0.997	0.979	6.34	0.995	1.009	0.985	1.0	2.4
8	tongue	1.047	1.010	6.79	1.026	1.033	1.010	1.6	2.3
9	brain	1.083	1.030	7.27	1.047	1.048	1.031	1.6	1.6
10	spleen	1.112	1.051	7.55	1.074	1.067	1.049	2.4	1.7
11	adipose	1.009	0.953	5.69	0.959	0.996	0.970	-1.1	2.7
12	bladder	1.093	1.026	7.18	1.044	1.047	1.030	1.4	1.7
13	bone cortical	1.068	1.068	13.67	1.222	1.034	0.982	24.4	5.3
14	bone mixed	1.065	1.149	13.22	1.261	1.119	1.062	18.7	5.4
15	liver (2)	1.043	1.069	7.26	1.078	1.088	1.063	1.4	2.4
16	heart (2)	1.218	1.048	7.17	1.059	1.068	1.047	1.1	2.0
17	thyroid	1.035	1.029	7.20	1.043	1.048	1.029	1.4	1.8

Figure 4: Relative differences between relative stopping powers (RSPs) determined with single energy CT (SECT) and dual energy CT (DECT) compared to experimental RSPs as a function of mass density for 32 materials. The RSPs are derived with a mean excitation energy for water of 69 eV, consistent with the SECT and DECT methods. The highlighted areas represent the differences for materials mainly composed of C, H and O with regular tissue densities for SECT (blue) and DECT (green), respectively.



Schaffner and Pedroni estimated the uncertainty associated with the conversion of patient CT data to relative proton stopping powers at 1.8% for bone and 1.1% for soft tissue [1]. The uncertainty due to the calibration curve alone was estimated at 1% [1]. This uncertainty combined with an uncertainty in measured CT numbers of ~2% leads to the widely adopted range uncertainty in the order of 3.5% [12]. We have measured differences for the tissue substitutes and soft tissues up to 3.5%. For other materials that have tissue-like compositions based on C, H or C, H, O, larger differences between measured and predicted relative stopping powers (~10%) have been found for the stoichiometric SECT method. Materials like ceramics used in prostheses and silicone oil used in treatment of ocular tumours deviate

even more while the results from the DECT method do not show these large differences. This confirms that SECT is not material specific and cannot provide patient specific tissue relative stopping powers.

4 CONCLUSION

With the DECT method relative proton stopping powers can be determined on an image basis with accuracy better than 2% when inconsistensies in mean excitation energies are solved. The validity of the DECT method for materials without hydrogen in their composition has to be individually verified. Due to the small validity bandwidth and non-patient specificity of the stoichiometric SECT method no single range uncertainty value can be given for all human tissues. The DECT method is based on the physics interactions of photons with tissue and is more robust to patient specific variations in composition and density of the tissues. The improvement in the accuracy of tissue relative proton stopping powers which can be accomplished with the DECT method is likely to be of great benefit for the accuracy of dose calculations in proton therapy.

5 ACKNOWLEDGEMENTS

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Case-based practical training in radiation safety for medical interventional teams

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Abstract. At the cardiac catheterization laboratory of the Rijnstate Hospital (Arnhem, the Netherlands) the collective team dose (Hp10) of the cardiologists and the technicians increased more than could be explained from the increase in procedures. After initial investigation a radiation safety training seemed the best solution. Each topic was presented by a small case from daily practice. The team was then asked to indicate the best solution. Next, their proposed solution was tested in practice on a phantom, with real-time feedback provided by active dosimeters. This enabled them to reflect on their expectations and the concrete experience. In the last stage, the team members were facilitated by the medical physicist in understanding the underlying radiation physics concepts. To also learn by repetition the training ended with more complex cases where all the concepts could be applied and tested again. The collective team effective dose decreased by 26%, whereas the number of patients treated stayed stable. This indicates a strong dose reducing effect from the training. Long term effects as reported by the team: a better cooperation within the team, higher awareness on radiation dose, and improved contact with the medical physicist for advice. These are all elements of a good safety culture, indicating that this practical training also has a positive effect on the radiation safety culture.

1. INTRODUCTION

Rijnstate Hospital (Arnhem, the Netherlands) is a large general hospital in the eastern part of the country. The cardiac catheterization laboratory delivers 24/7 cardiac care for all patients of the region (450,000 inhabitants), with over 3,000 procedures per year. It is equipped with two interventional x-ray systems. The team consists of 5 interventional cardiologists, 3 electrophysiologists, and 21 specialist nurses and technicians. Over the years the collective team effective dose, as measured in Hp(10) by the personal dosimeters, increased more than could be explained from the increase in procedures. In three years (2010-2012) the collective dose increased by 53%, and number of procedures only 16%. After initial observation, in the team there was confusion about different radiation physics concepts, and there was no consensus on the best method. There had not been any specific continuing education on radiation physics and safety. Therefor a radiation safety training seemed the best solution. This paper describes the development of this radiation safety training program and the initial experiences.

2. METHOD

The setup of this radiation safety training is primarily based on Kolb's experiential learning theory [1]. Kolb's theory describes an experiential learning cycle that consists of four stages: 1. active experimentation, 2. concrete experience, 3. reflective observation, and 4. abstract conceptualization. Learning is most effective when the learner goes through all stages: by trying out what they know (1), experience the outcome (2), reflect on inconsistencies between knowledge and outcome (3), and modification of their concepts (4). See Fig.1 for a visual presentation of Kolb's learning cycle.
Individual differences mainly determine where people prefer to enter the learning cycle. For radiation safety experts with a background in physics this is typically at stage 3 with observing and reflecting. Most healthcare professionals with a 'hands-on' attitude prefer to enter the circle at stage 1 by trying and experiencing. Therefor to build a learner-centered training, rather than a teacher-centered one, a case-based approach was chosen. With cases from daily practice people are allowed to first apply their existing knowledge by indicating the best solution (1). Next, their proposed solution was tested in practice (2), with feedback provided by active dosimeters (DoseAware, Philips Medical). This enabled them to reflect on their expectations and the concrete experience (3). In the last stage, the underlying radiation physics concepts were presented (4). This application is also shown in Fig. 1.

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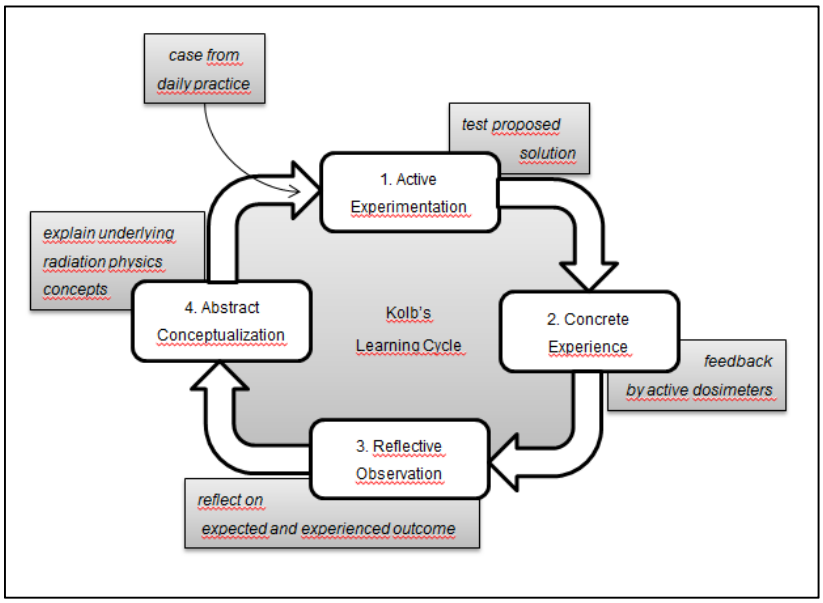
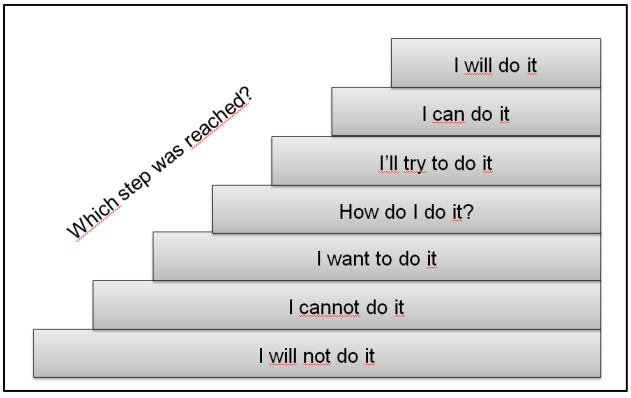
Figure 1: Kolb's learning cycle



To accommodate all types of learners, it is best to work with short cycles going multiple times through all learning stages. This resulted in small cases, each based on one radiation concept at a time. And with each case going through all learning stages before continuing with the next case. The universal concept of learning by repetition was implemented in the second half of the training. Here more complex cases were presented where all the treated concepts could be applied and tested again.
A safe learning environment is created by replacing the humans for phantoms, so that no one is harmed while experimenting with radiation. To facilitate active learning, the groups should be small. A mixed group, consisting of physicians, nurses and technicians, will facilitate team discussions of how to implement the newly learned lessons into practice.

One of the key elements to measure changes in radiation safety behavior is the collective team dose [2]. The collective team dose (effective dose by Hp(10)) is reported from three years before educational intervention until two years after to also measure long term effect. The collective team dose is reported together with number of patients treated to correct for changes in workload.
Efficacy of the training can also be measured by reported self-efficacy of the participants. To perform certain behavior people need both knowledge and the confidence that they can perform a task [3]. The degree of confidence can be measured on a scale from 'cannot do it' to 'will do it' [4]. An example of a self-efficacy scale is given in Fig.2. However, as this training was not setup in a scientific study, only a subjective report of participants reactions can be given in this case.

Figure 2: Example self-efficacy scale

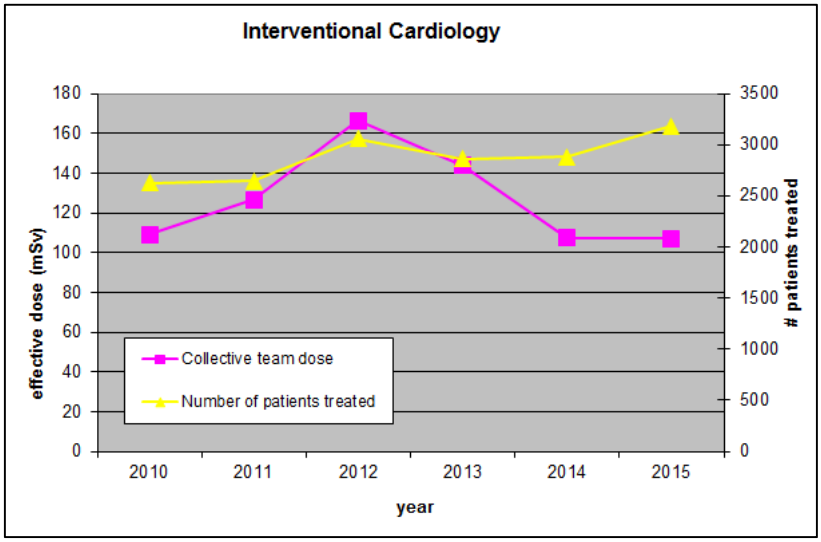


3. RESULTS

Radiation safety topics that needed to be addressed were identified from observation in practice and on indication by a specialized technician. All topics were split into cases addressing one physics principle at a time. This resulted in cases like 'Should the patient table be positioned as close as possible or as far as possible from the x-ray tube', and 'Estimate the dose saving effect of collimation possible in this image'. Repetition was implemented using more complex cases based on multiple principles. For example, participants were asked to build a worst-case stray radiation situation for a person standing in a certain position. In total the training contained five simple cases and three complex cases. In addition, due to the interactive nature of the training participants often came up with other questions/cases. These were be taken along immediately.

The collective team effective dose and number of patients treated from 2010 until 2015 is shown in Fig.2. From 2010 till 2012 the collective team dose increased more than could be explained from the increase in workload. In 2013 we developed the case-based practical training. The training started in September 2013 and all team members were trained by the end of January 2014. In 2014 the full effect of the training is visible in a strong reduction of the collective dose. The collective dose from 2015, almost two years after completion of the training program, the team dose was stable whereas workload increased. This indicates a long lasting effect from the training. Comparing team dose over years 2010-2012 with 2014-2015 normalized for the number of patients treated, a reduction of 26% is measured.

Figure 3: Collective team dose before and after training in 2013



Efficacy of the training was also gauged by the initial reactions of the participants in the training. On the self-efficacy scale from 'cannot do it' to 'will do it', there were many reactions starting with 'From now on I will do …'. This means participants were able to translate the newly learned lessons directly into their clinical practice. They scored very high on the scale of self-efficacy, which is a key-component in the application of new knowledge in practice. As was confirmed by the dosimeters.
Long term effects as seen from the dosimeters were also confirmed by the team. They reported a better cooperation within the team, a better awareness on radiation dose, and a better contact with the medical physicist for advice on advanced topics. These are all elements of a good safety culture, indicating that this practical training also has a long lasting effect on the radiation safety culture.

4. CONCLUSIONS

A case-based practical training was setup to accommodate an efficient learning cycle adapted to the dominating learning style of health care professionals. Effects were measured by the collective team dose showing a long lasting 26% reduction in effective dose. The training was very well received by the team of cardiologists, specialized nurses and technicians. They were able to transfer the new knowledge directly to their clinical practice. This resulted in a better safety culture.

5. ACKNOWLEDGEMENTS

Special thanks to Tamara van der Beek, cath-lab technician, for her help with identifying the topics that needed to be addressed in the training.

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Technology to reduce or even prevent harmful radiation effects on the skin.

Ad den Boer

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Abstract. With the development of pulsed fluoroscopy in 1992 in the Thoraxcenter Rotterdam we got the opportunity to develop new cardiological intervention techniques, leading to prolonged treatment- and fluoroscopy times. A real-time entrance dose monitoring system, made 1999 in Rotterdam, enabled us to follow up patients who received a high local skin dose. We followed various patients who received a pulsed X-ray entrance dose over two gray, those patients showed no deleterious effects to the skin. Hereafter we performed a pig-skin study. Using continuous X-ray fluoroscopy skin reactions were seen as expected. However, no harm was seen after using pulsed radiation. These findings cannot be explained by differences in dose, dose-rates and (extra) beam filters, these were identical, continuous X-ray: 480 nGy/s, (110kV-12mA), pulsed X-ray: 475 nGy/s, 38 nGy/p (110kV-120mA at-12.5 pulses/s and-8 ms/pulse). An in vitro (skin)cell survival study with 6 gray, using continuous fluoroscopy, showed LQ curves with a survival rate of <1%, (this fit with data from radiobiologic literature). On the other hand, pulsed (12.5 per second) radiation showed a cell survival of more than 80%. It was concluded that a cellular adaptive response was found: Cells can repair or prevent the expression of radiation damage within milliseconds if pulsed radiation is applied. This is very reassuring for patients and for operators, occupationally exposed to high levels of X-ray radiation. While pulsed X-ray radiation does not exist in nature, (it is man-made since 1992), until now this phenomenon has never been examined at cellular level. The presentation focus on various questions about the (absence of) harmful biological response to pulsed X-ray, as seen in practice in interventional radiology and cardiology. It is stressed that further research should be done to (skin)dose response relationship and to the optimization of the pulsed X-ray technique.

KEYWORDS; Pulsed Radiation, Biological Effect, Cell Survival, Patient Dose, Intervention.

1 INTRODUCTION

The author, active in the radiological and cardiological field for 45 years, gives an overview of milestones concerning his specialism: X-ray technology, imaging, dose reduction and safety. Together with radiobiological studies he puts the last 25 years to glance. Based on his own experience he addresses two questions: can cells repair radiation damage and can proper X-ray technology save worldwide 15,000 lives annually?

2 A HISTORICAL OVERVIEW ABOUT RADIATION SAFETY AND SKIN REACTION

1900 Skin lesion due to radiation.

Figure 1. *Hands used as test object.*

Detailed description of dr. Kassabians own hand lesions lead to use of beam filters.

1915 Aluminum (Al) beam filter legally obligated**.** Open tube =100% photons at object, with a 2 mm Al beam filter only 35% photons are entering the object.

1950 The invention of X-ray image intensifier, viewing to the output-screen by mirrors or optics, increased the use of fluoroscopy and a cine film camera could document the image.

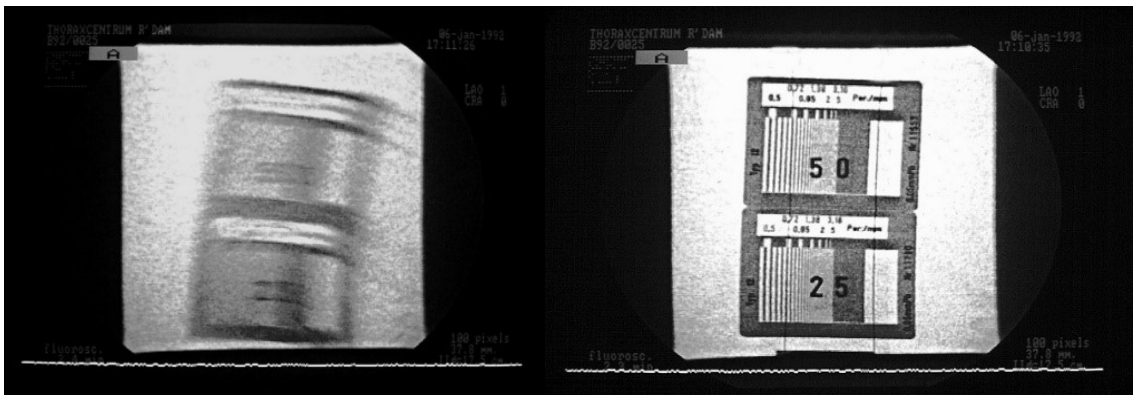
1962 For cinematography a pulse technique was developed, high voltage switching with vacuum tubes, no radiation when the camera shutter is closed during film transport, the first major tube load and dose reduction and biplane imaging became possible.

1965 Introduction of video systems increased the application of fluoroscopy significantly, the investigator could share the image with others.

1980 Beginning of interventional fluoroscopy, balloon dilatation, stenting and electro physiologic investigations demanding long investigation- and extended fluoroscopy times, causing erythema and other radiation skin effects to patients.

1992 World première clinically used pulsed fluoroscopy in Rotterdam, a Thoraxcenter/Siemens project with grid switched X-ray tubes. Electronic gap filling (no flickering images anymore), gave us continuous monitor display a major image quality improvement [1] and a dose reduction during investigation for patient and operator while less images per second could be used.

Figure 2. *Image quality Continuous fluoroscopy versus Pulsed fluoroscopy*

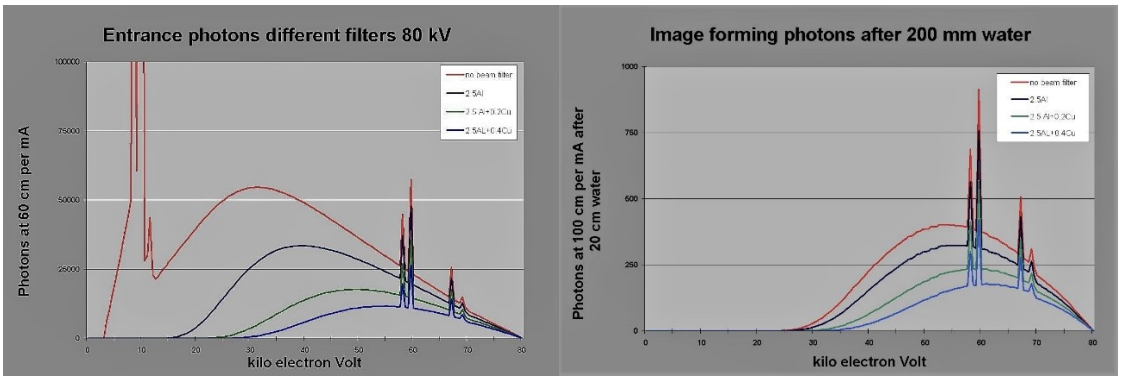


Frozen images from a rotational test device, pulsed technique made moving guidewires visible.

1993 X-ray tubes with liquid metal bearing improved the anode heat exchange and gave us silent fluoroscopy, no irritating noise anymore from the rotating anode, this made patient treatment much more relaxing. Due to electronic enhancement the image quality improved, double guide-wires became visible in moving vessels and the kissing ballon technique was introduced.

1995 Improved high output X-ray tubes with liquid metal bearing made extra beam filtration feasable, with the same image quality the skin dose reduced with 50%.[2]

Figure 3. *Patient Entrance Radiation Spectrum* **Figure 4.** *Detector Entrance Radiation Spectrum*



Adding 0.2 mm Copper (Cu) to the legally obligated 2.5 mm Al filter halves the entrance dose (from bleu[2nd] to green[3rd] curve) while less non-image forming photons are entering the patient.

1996 Digitalization of video images and digital storage replaced the cine film camera, this silent

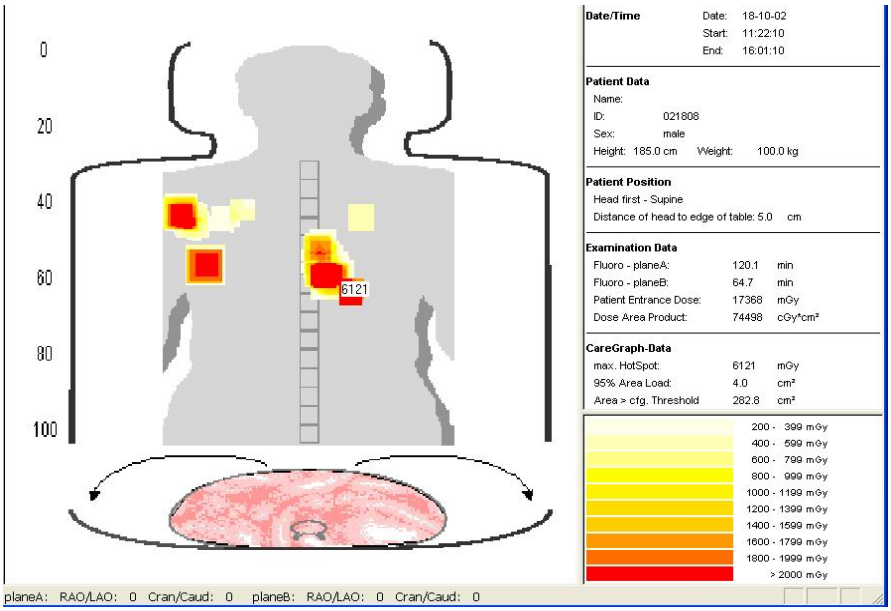
digital cine mode (DCM), no noise making camera above the patient made better communication possible, a big improvement in treatment in the cathterisation laboratories.

1997 The introduction of digital X-ray solid state flat panel tecnology made image quantification and recontruction easier, no geometrical distortion from the vacuum imagage intensifier anymore.

1998 Because more and more alarming skin lesions are published, we developed in the Thoraxcenter Rotterdam a real time entrance dose monitoring system, showing patient skin per cm^2 [2] which,

1999 Resulted in the first industrial dose monitoring system, the Siemens Caregraph [3]

Figure 5. *Real time entrance dose monitoring during cardiac intervention.*



Skin dose alarm (>2 Gy) suggest to use other gantry settings, reports of entrance dose per cm^2

2000 Until 2007 more than 580 skin lesion cases are world-wide published in different papers.

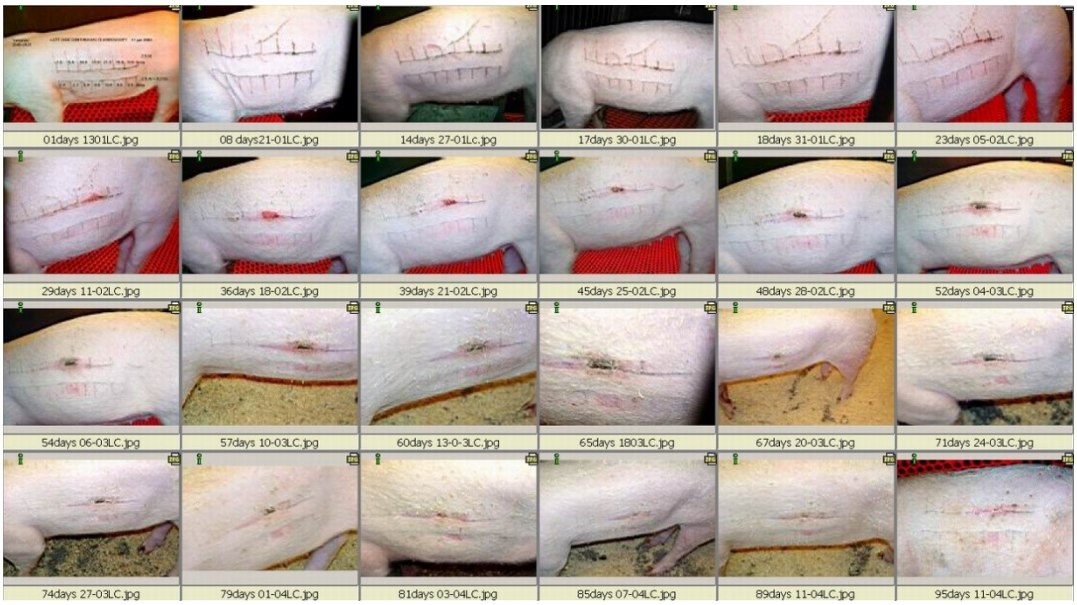
Figure 6. *Pictures of skin lesions as reported in multiple scientific magazines.*

Among other reasons, these lesions led to the obligation to educate European medical specialists in radiation safety, protection and hygiene. This decreased malpractice due to no knowledge or ignorance.

2003 We followed dozens of patients, who should have developed deteministic skin effects for a year, but no harm was observed, this was the reason to perform a skin study on pigs.

2005 Pig skin study, dose effect with different fluoroscopic techniques, continuous, pulsed with- and without extra filtration.

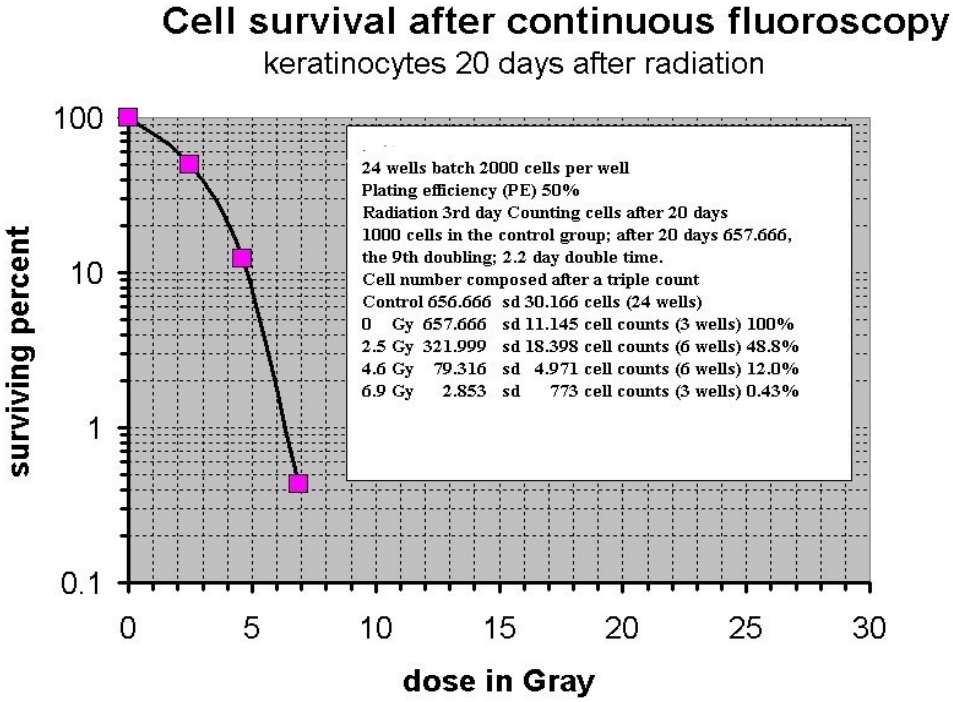
Figure 7. *Pig skin reaction followed until100 days after radiation.*



Result of continuous radiation (upper part of radiated skin field): skin lesions were as expected. Result of pulsed technique almost no harm visible. We studied 6 pigs, the difference cannot be explained by the used dose-rates and (extra) beam filtering, they were identical (continuous radiation 110 mGy/min (110kV-12 mA), pulsed radiation 110 mGy/min (110kV-120mA-12.5 pulses/s-8ms/pulse). We continued with a dose/survival study using in vivo skin cells.

2006 In vivo cell investigations using Caucasian keratinocytes

Figure 8. *Linear Quadratic (LQ) survival curve of human skin cells*



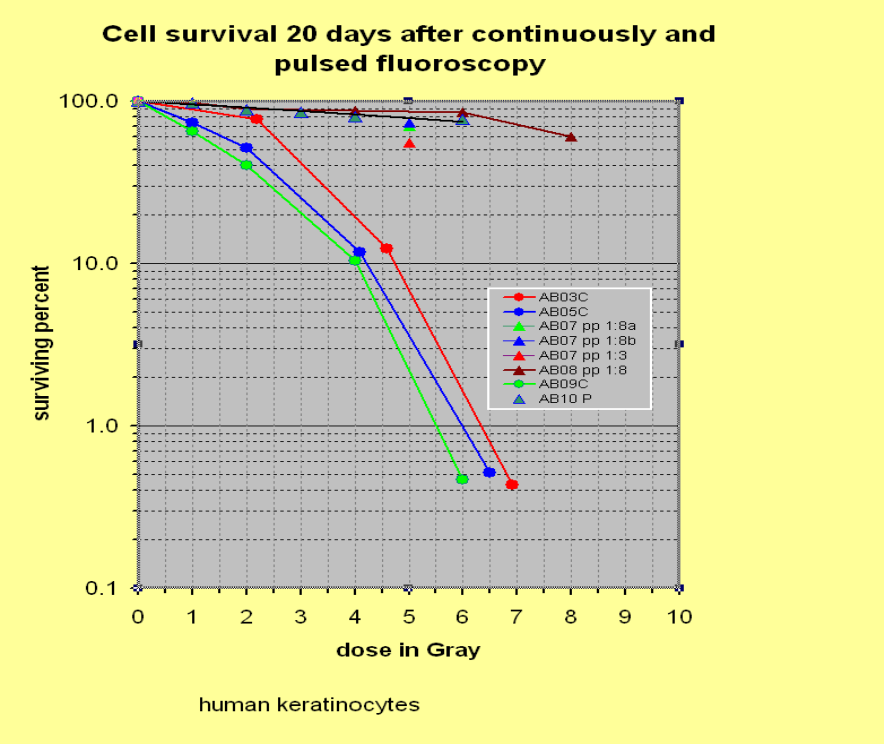
Result as expected, just like in radiobiology textbooks. Dose administered simulating investigation circumstances in blocks of 30 sec fluoroscopy and 30 sec pauses, 1 Gray dose after

20 minutes. Dose measured with 3 systems, a semiconductor system an ionization chamber and thermoluminescence dosimeters, they all matched ± 5%.

2007 Cell survival study skin cells human Caucasian keratinocytes
Simulating practice, prolonged investigation times (up to 240 minutes), prolonged fluoroscopy times (up to 160 minutes). Entrance dose to the cells ≥6 Gray. Survival continuously fluoroscopy was <1%, survival pulsed fluoroscopy was>50% with 25 pulses/s even <80% when 12.5 pulses/s was used. This explained why we could not find lesions with high dose monitored interventional patients as well as the results of our pig skin study.

3 RESULTS CELL SURVIVAL STUDY

Figure 9. *L.Q. curves of 8 cell cultures, survival in % versus dose level in Gy.*



Cells radiated simulating practice, prolonged investigation times (up to 240 minutes), prolonged fluoroscopy times (up to 160 minutes). Entrance dose to the cells ≥6 Gray. Survival continuously fluoroscopy 6 Gy <1%, pulsed fluoroscopy 6 Gy >50%. Our results differ from all other radiobiological and cell dose/survival investigations, because these were all performed with continuous radiation. In radiobiological studies one should specify the type and origin of radiation, the quality, quantity, flux and timing. [4] This study was done with a pulsed fluoroscopic technique as used in daily practice, simulating investigation time and fluoroscopy time settings and the usual geometrical settings. Cumulative biological effect of radiation seems to be absent or is much less when pulsed radiation is applied. The cell survival differs using other pulse frequencies, 12.5 pulses/s (p/s) = >80%, 25 p/s = >50%, continuous <1%. If pulsed technology (≤8 ms) is used, (skin)cells can survive radiation, they repair or even prevent damage within a very short time. We measured a cell survival mechanism, not described until now. Cumulative biological effect of radiation is shown in numerous studies. Our study showed that these effects are much less or even seem to be absent when pulsed X-ray fluoroscopy is applied. Pulsed radiation in this format does not exist in nature, it is man-made since 1992, hereafter radiobiological studies with this kind of radiation are never performed. An earlier pilot entrance dose study from 15 patients showed that high skin doses (>4 Gy) caused by digital cine mode (DCM), 25 and 12.5 p/s with 63nGy/p, did not show skin effects. In chinese-hamster ovaria cell studies (too few data to publish) a 6 Gy DCM dose (12 nGy/p) showed with 8.3 p/s >90%-, 12.5 p/s >80%- and 25 p/s >50% survival.

4 DISCUSSION

Population dose.
The Linear No Threshold (LNT) model is used by the ICRP and the World Health Organisation as radiation risk model for the population dose.[6] One millisievert extra per 100,000 persons, 5 of those persons will develop a fatal, radiation inducted, cancer (FRIC) within 40 years. Worldwide 2,5 billion persons receive >0,8 mSv dose annually from medical diagnostically per year, total about 2 MSv [7]. Since the introduction of the (Multi Slice) Computed Tomography (MSCT) and interventional fluoroscopy in 2000, the world-wide usage of medical radiation increased yearly with 7 percent. Fluoroscopy, Angiography and (Multi Sliced) Computed Tomography (MSCT) is a major part (80%) of the contribution to the public medical radiation exposure [5, 6, 7]. World-wide use of continuous fluoroscopy, 2,5% of total [6], is >50,000 Gy, although most patients are elderly (less stochastic effects), replacing continuous fluoroscopy by a pulsed technique could save about 15,000 FRIC, according to the current used LNT risk model. If the described pulsed dose effects are identical to all human cells, a pulsed technique used in (MS)CT) machines for children and young adults could save much more lives annually.
Image quality.
For moving objects, pulsed fluoroscopy shows a superior image quality compared to continuous fluoroscopy, this is why most hospitals are using the pulsed technique these days for interventional work, confirmed by the absence of publications with deterministic skin problems the last 10 years.
Occupational dose.
Twenty-three million people worldwide are exposed professionally to ionizing radiation [7]. Does the pulsed technique scattered radiation also show a less destructive or non-cumulative effect?
There is no reason to suppose the pulsed scattering radiation should have another biological effect than pulsed fluoroscopy and DCM. However, the level of scattering is so low (100 µGy/h at 1 m) that an exposure of 6 Gray for a cell survival study would ask 60,000 hours radiation. This is why proof has to come from other approaches. Publications like," Cellular adaptive response to chronic radiation exposure in interventional cardiologists "and "Surgeons may be adapting to 'safe' X-ray doses" [8], (the search of biomarkers in blood), could be one approach.

5 CONCLUSION

We found a cellular adaptive response which only exist with pulsed radiation. Radiobiological effects with pulsed radiation are never examined. (Skin)cells can repair radiation damage if pulsed technology is used. Continuous fluoroscopy should not be used anymore and must be forbidden. According to the used risk model, worldwide changing continuous conventional fluoroscopy with pulsed technology can safe annually about 15,000 fatal radiation induced cancers. For the MSCT investigation of children and young-adults, pulsed technique should be developed. The biological effect of picture creation with multiple short intense pulses should be investigated. Using a technology with less biological effect of (scattered) radiation is reassuring for radiological workers, they could benefit. It is not correct to use the Linear No Threshold Risk Model for pulsed X-ray technology. All radiation, used for diagnostics and interventional procedures, should be administered in such a way that it shows the least biological damaging effects. The advantage of diagnostically radiation is much more than the disadvantage, when pulsed radiation is used, diagnostically equipment is even more beneficial and save more lives.
It is stressed that further research should be done to (skin)dose response relationship and to the optimization of the pulsed X-ray technique in radiology and cardiology.

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Guidelines for radiation protection and dosimetry of the eye lens in the Netherlands

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Abstract. In the past, the dose limit for occupational exposure of the eye lens to ionising radiation was rarely exceeded. Early 2018, this limit was reduced from 150 to 20 mSv per year. When category A workers are liable to receive a significant exposure of the lens of the eye, an adequate system for monitoring shall be set up to ensure that the dose remains under the dose limit. A committee of the NCS is preparing guidelines to support radiation protection experts in the Netherlands with the practical implementation of this legislation.

If the calculated annual eye lens dose in the prospective risk assessment is higher than 15 mSv, a worker is to be classified as a category A worker. The largest group of those workers is found in the medical field, in particular in interventional radiology and cardiology. It should be noted that some of the workers in nuclear medicine, veterinary practice, non-destructive testing and isotope production may be exposed to such high dose levels as well. The type of eye lens dose monitoring depends on the exposure conditions. It is expected that whole body dosimeters give a reasonable estimate of the eye lens dose for workers who are exposed to uniform photon beams. The committee recommends using a dosimeter near the eye lens for fluoroscopic procedures, nuclear medicine, veterinary medicine, nuclear industry and cyclotron maintenance for workers with an expected dose of more than 15 mSv per year . The use of a whole body dosimeter at the chest is recommended for industrial radiography

General protection principles can be used to restrain the eye lens dose such as limiting time, increasing distance and applying shielding of the source. The use of personal protective measures should be considered as a last, but sometimes inevitable step in this process.

KEYWORDS: eye lens, radiation protection, dosimetry

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1 INTRODUCTION

The sensitivity of the eye lens to ionising radiation has been gaining interest in the last decade. As a result of various investigations, the International Commission on Radiological Protection (ICRP) recommended in its statement on tissue reactions in 2011 [1] a reduction of the equivalent dose limit for the lens of the eye for occupational exposure in planned exposure situations from 150 mSv to 20 mSv per year (averaged over a period of 5 years, with no annual dose in a single year exceeding 50 mSv) . This resulted in adaptation of the dose limit for the eye lens in the EU BSS in 2013 [2]. Early 2018, this EU dose limit was implemented in the national legislation in all European member states. When putting this legislation into practice, certain terminology requires interpretation and a number of issues need to be addressed in relation to the protection of the eye lens:

1) What level of exposure of the eye lens is considered to be "significant" and requires monitoring?

2) What is an "adequate system for monitoring"?

3) How is the dose to the eye lens that a worker is "liable to receive" determined?

4) Which worker groups are expected to be involved?

5) Which measures can be applied for protection of the eye lens?

The NCS (Netherlands Commission on Radiation Dosimetry) started a subcommittee in 2016 to provide guidance and answers with respect to these questions. This paper contains the most important topics of the NCS report. The questions related to the interpretation of the legal framework will be addressed in Chapter 2. Typical worker groups will be discussed in Chapter 3. Recommendations with respect to protection measures are presented in summary of the NCS recommendations (Chapter 4).

2 INTERPRETATION OF THE LEGAL FRAMEWORK

2.1 What (significant) exposure of the eye lens requires monitoring?

According to the EU directive, all exposed workers with an equivalent eye lens dose >15 mSv are categorised as category A workers for which an adequate system for monitoring needs to be set up. On the other hand, publications from both the IAEA [3] and ISO [4] recommend monitoring routinely above 5 or 6 mSv. As the (equivalent) dose limit for the eye lens is 15 mSv for the public, the NCS committee proposes to consider an exceedance of 15 mSv as a significant exposure of the eye lens. Below this level, there is no legal context for eye lens monitoring. We recommend that for those workers with a calculated equivalent dose between 10-15 mSv a survey shall be performed to demonstrate that the 15 mSv dose level is not exceeded.

2.2 What is an adequate system for monitoring?

The answer to this question depends on the type and quality of radiation, the exposure pathway and if the radiation field is distributed homogeneously. Not all exposure conditions require dedicated eye lens dosimetry. IAEA TECDOC 1731 [3] contains an extensive scheme describing the different circumstances and the adequate systems for monitoring.

2.3 How is the (eye lens) dose that a worker is liable to receive determined?

For those employees exposed to ionising radiation, a prior risk assessment (or risk analysis) should be performed by the radiation protection expert. This results in an estimation of the eye lens dose for the

individual employee. This prior risk assessment should cover the worker exposure resulting from regular and potential exposure situations.

3 TYPICAL WORKER GROUPS

3.1 Population overview based on Dutch national dose registry

An overview of the number of workers in the Dutch national dose registry (NDRIS) as a function of different dose groups and work categories is presented in Table 1 for the year 2015. Because in the Netherlands the worker dose is assessed on the outside of protective aprons, the protective effect is not included in this overview.

Table 1 Number of workers in different work categories as a function of $H_p(10)$ values in the Dutch national dose registry for the year 2015.

Risk class		**$H_p(10)$ National Dose Registry (in mSv)**					
(H/M/L)	**Work category**	**≥ 0**	**> 1**	**> 6**	**> 10**	**> 15**	**> 20**
H	Radiology	10323	824	175	94	53	34
H	Interventional proc., MD	3694	1084	262	135	67	30
M	Other industrial applications	1261	81	5	4	3	2
M	Other applications	1001	50	14	3	3	2
M	Industrial radiography (mobile equipment)	990	164	6	3	3	2
M	Interventional proc., other	840	121	18	7	3	
M	Veterinary medicine	4762	192	23	7	1	
M	Industrial irradiation	608	19	5	3		
M	Isotope production	374	99	4	1		
M	Radiotherapy	1549	6	3	1		
M	Other medical applications	642	30	1	1		
M	Nuclear medicine (excl. PET)	1458	118	2			
L	Education, R&D	1451	15	1			
L	Transport (excl. nuclear fuel)	78	2	1			
L	Aircrew	14462	12157				
L	Nuclear energy	1045	73				
L	PET applications	402	53				
L	Maintenance and repair	686	10				
L	Transport of nuclear fuel	346	10				
L	Nuclear fuel enrichment	189	2				
L	Waste treatment and storage	30	2				
L	Dental radiology	2030	1				
L	Security, Safety & Inspection	269	1				
L	Industrial radiography (fixed)	208					
L	Other nuclear applications	47					
L	Mining, oil & gas extraction	32					
L	Automatic control engineering	14					
L	Decommissioning	5					
	Total	47721	15111	520	259	133	70

3.2 Risk categories for exposed workers groups

Based on the data from the Dutch national dose registry and input from professionals involved, the following three risk categories are proposed:

1. Significant eye lens doses to be expected (marked as H in left column of Table 1)

Staff working in close proximity to patients in:

- fluoroscopically-guided procedures (including interventional cardiology, interventional radiology, other interventional fluoroscopically-guided procedures)

- CT-guided interventions

2. Moderate eye lens doses to be expected (marked as M in left column of Table 1)

Staff working in:

- nuclear medicine (typically those preparing radiopharmaceuticals)

- veterinary medicine (especially those involved in imaging of horses, both by X-ray and by scintigraphy)

- industrial radiography (especially those performing non-destructive testing with sealed sources)

- nuclear industry and isotope production (replacement of cyclotron targets)

Workers handling beta sources require special attention, because the measured $H_p(10)$ whole body dose may underestimate the eye lens dose.

3. Only limited eye lens doses to be expected (marked as L in left column of Table 1)

In general, the groups marked with an L in Table 1 and some of the categories marked with a M are expected to receive limited eye lens dose. This category comprises aircrew, staff involved in radiation therapy (manual brachytherapy with high energetic sources is very rare nowadays) and workers involved in veterinary medicine applications for pets, including scintigraphy.

4 SUMMARY OF THE NCS RECOMMENDATIONS

Workers receiving a high dose are often working in close proximity of the radiation (scatter) source. Because of this, the distance and angle under which the eye lens is exposed may differ from the trunk and a measurement near the eye lens can be considered the only accurate way to determine the eye lens dose. For this reason, the committee recommends using a dosimeter near the eye lens for fluoroscopic procedures [5], nuclear medicine, veterinary medicine, nuclear industry and cyclotron maintenance for workers with an expected dose of more than 15 mSv per year . The use of a whole body dosimeter at the chest is recommended for industrial radiography. For workers with an expected dose below or equal to 15 mSv per year, this overview provides a recommendation for an adequate system for monitoring, but the monitoring itself is not mandatory. Under specific circumstances where the ratio between eye lens dose and whole body dose is constant or the whole body dose is consistently higher than the eye lens dose, monitoring near the collar or on the chest may prove to be adequate.

The committee recommends that the radiation protection expert verifies the assumptions underlying the prior risk assessment with the results of whole body and/or eye lens dosimetry. Where the calculated eye lens dose for workers lies between 10 and 15 mSv per year and no adequate monitoring data is available to validate the risk assessment, the committee recommends performing a survey demonstrating that the 15 mSv dose level is not exceeded.

With respect to the protection measures the committee recommends the following:

- While choosing protection measures, the preferred approach would be to shield the source itself, to apply dose reduction on the system, to decrease the exposure time, to increase the distance to the source (with or without tools) and to use room protective measures. The use of personal protective measures (such as radiation safety glasses) should be considered a last step in this process.

- Radiation safety glasses should contain a minimum of at least 0.5 mm lead equivalent and only be applied with careful consideration to proper fitting to the face, front and side shielding and orientation of the head with respect to the radiation source. When taking these conditions into account properly, a dose reduction factor of 2 is considered a conservative approach for medical and veterinary X-ray procedures [4,5].

- The eye lens dose should preferably be measured on the outside of personal protective equipment near the most exposed eye. This does require the use of a suitable dose reduction factor to estimate the actual eye lens dose. In specific situations it may be more practicable to measure under the personal protective equipment (for instance in a lead cabinet).

The following recommendations are provided for the medical surveillance of category A workers with an expected eye lens dose of more than 15 mSv per year:

- Yearly questionnaires should be extended with questions on visual acuity, other eye-related complaints and questions on problems in the use of and coping with personal protection of the eyes.

- A visual acuity examination should be offered at least biannually.

- A slit-lamp examination should be offered to workers, periodically, every 5 or 10 years.

The committee of the NCS intends to publish the guidelines shortly on the NCS website (radiationdosimetry.org).

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Evaluation of Structural Radiation Protection in Radiotherapy by Monte Carlo Methods

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Abstract. In the current regulations and standards for safety for radiation protection we find general information about techniques to build an adequate structural radiation protection in radiotherapy facilities. But ducts, gaps or geometry overlaps may lead to weak spots, which are usually only noticed in the late construction phase by real dose measurements and then are costly to correct. We address this problem by observing different geometries to identify constructions, which guarantee a sufficient radiation protection. The general-purpose Monte Carlo code MCNP was used to model the head of a clinically used linear accelerator and its surrounding radiation protection construction. The verification was carried out with the help of depth-dose datasets. The structural radiation protection was adjusted for the three acceleration voltages and not only gamma radiation, but also neutron radiation was considered. Following, different geometries and their radiation protection abilities were investigated. Variation of the duct diameter and its geometry showed that for ideal bent ducts the photon dose does only have small variations as a function of diameter. For neutrons, the difference between the duct types is very small. A reduction of the gap between the radiation protection door and the floor does not have a big effect on the gamma dose behind the door for high energies. The lacking overlap between the radiation protection door and its adjoining wall is best cured by simple blocks of concrete. With the help of MCNP different geometries were successfully modelled, so the sufficiency of different radiation protection geometries could be proven. We recommend the usage of bent ducts, so the gamma dose behind the walls is as low as possible. For neutrons, it can still be a source of leakage. Adjustments of the doors to reduce clearances show only small improvements at high costs.

KEYWORDS: shielding, linear accelerator, Monte Carlo, structural radiation protection, radiation facility

1 INTRODUCTION

Linear accelerator facilities are used in medical cases for radiation therapy, for industrial non-destructive testing and for research of structural analysis. The most frequent use of linear accelerators is medical application. To achieve adequate shielding for high energy gamma radiation and in some cases also neutron radiation, shielding calculations must be done to evaluate the structural radiation protection. Today, we have international standards for many applications regarding radiation protection [2] and regulations on national level [1].

In general, these regulations describe standard geometries. Moreover, they offer good first approaches for even more complex issues. For example, a standard way for constructing ducts is integrated within the German standard DIN 6871 [6]. Still, the more complex the problems get, the less amount of standard solutions for structural radiation protection is offered. Often modelling with Monte Carlo methods give the only reliable answers on complex questions in structural radiation protection. Furthermore, Monte Carlo methods may help improving facilities with known or unknown radiation leakage. Prior to real measurements in a facility, a Monte Carlo calculation may help to identify weak spots of shielding and prevents costly corrections. In the following, these weak spots like ducts, gaps and a lack of overlaps on areas with material variation were examined. Ideal construction before building or simple and cost-efficient improvements after building were examined.

2 MODEL GENERATION

To generate an accelerator model for shielding publicly available information of technical specifications [4, 5] were used. Furthermore, several information was gained by own simulations and comparison with results of real measurements. 6.7 MeV, 10 MeV and 15 MeV accelerators were simulated. MCNPX [3] was used for modelling the geometry, the source and the detectors. Following we present the composition of the electron accelerator head and its validation. To evaluate the structural radiation protection the physical structure of a radiotherapy facility and several challenging, potential weak spots were modelled.

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Due to energy dependence of attenuation we modelled the structural radiation protection for a radiotherapy facility in three implementations. One for each energy limit of 6.7 MeV, 10 MeV and 15 MeV. In addition to gamma shielding also shielding of neutrons caused by activation processes at high energies (above 8 MeV [7]) for the 10 MeV and 15 MeV accelerators was simulated.

2.1 Accelerator Head

The accelerator head was modelled based on information publicly available [4, 5]. Before using the accelerator head, an evaluation with real data sets was performed.

2.1.1 Modelling

The main components of the accelerator head were considered for our modelling. We modelled a tungsten-rhenium alloy target [4, 5] with density of 19.2 g/cm³ and a corresponding copper target cover for heat diffusion. Water cooling was not contemplated. Subsequent to the target the primary collimator for gamma rays in useful beam direction was modelled. The primary collimator is a rectangular block with a conus-like opening and consists out of a tungsten, iron and nickel alloy. The gamma rays leaving the collimator create a circular radiation field. For balancing the irregularities between the inner and outer part of the radiation field a flattening filter is modelled. The flattening filter varies in dependence on the accelerator energy. Finally, apertures were used to model the contour collimator. With the help of apertures, we simulated a radiation field of the size 40 cm x 40 cm. To complete the accelerator head, a shielding with a leakage rate of 0.1 % was modelled.

2.1.2 Validation

Due to the usage of different literature sources and own adaptions, a validation to ensure its feasibility for shielding calculation was carried out, before using the calculated radiation beam. To do so, datasets for depth dose measurements and dose profiles in a water phantom were used. The measurement steps were 2 mm for the dose profiles and 1.5 mm to 5 mm for the depth dose profile. Concluding, we compared our simulations to datasets for all 3 energies, one depth dose profile and two dose profiles in 10 cm and 20 cm depth.

Analogously a water phantom was modelled in MCNPX [3]. For the detector modelling we used 42 positions for depth dose simulation and 15 positions for the dose profiles in 10 cm and 20 cm depth. These results were compared with the above mentioned real depth dose and dose profile measurements.

2.2 Physical Shielding Structure

The focus of this work is on the structural radiation protection. Therefore, with the help of Monte Carlo methods [3] radiation therapy facilities were modelled, and the validated radiation field was inserted. Within the German standard DIN 6847 [1] there are several structures, which are not described, but may lead to unpredictably high dose rates behind the thick wall shielding. Often these weak spots are only noticed at final inspection by a measurement. At this point, the retrofitting of the shielding is laboriously and costly. This includes covering procedures for the linear accelerator due to particulate matter, which arises while supplementary shielding is installed. The potential weak spots are evaluated in the following and ways to compensate them pre- or post-building are shown. Furthermore, one of the utilized structural radiation protection designs is presented.

2.2.1 Facility

The shown model in figure 1 is based on a real facility, which is used for 15 MeV electron radiation and 15 MeV gamma radiation. We only considered gamma radiation and the potentially generated neutron radiation. For shielding it is not relevant to use the electrons because the gamma radiation shielding is the conservative case. The simulated radiation therapy shielding is shown in figure 1. It contains a maze to prevent direct exposure of the door by the primary beam. The door construction is determined by the chosen maximum energy of the radiotherapy device. For energies above 8 MeV [7] neutron generation is indicated, so a layer of neutron shielding polyethylene (PE) is used. If a labyrinth is absent, direct exposure by the primary beam may occur and the door shielding characteristics must be more efficient.

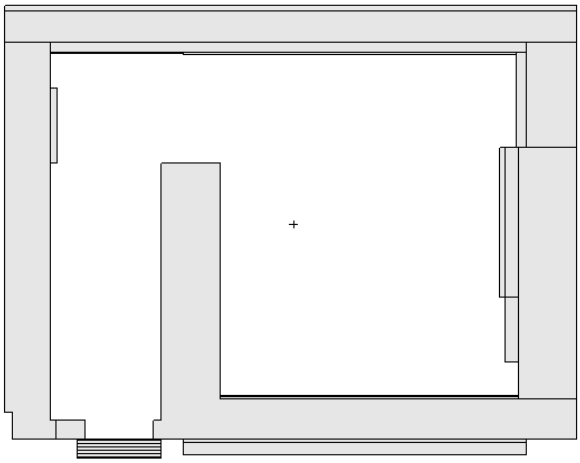

Figure 1 Modelled Radiation Therapy Facility

2.2.2 Ducts

The supply of electrical power, cooling water and fresh air should be guaranteed within a radiation therapy room and might cause problems for the structural shielding protection. For conventional straight ducts a huge amount of leakage radiation may occur, so special duct geometries should be used. These ducts are bent in contrast to the usual ducts in conventional surroundings. First, we evaluated the ideal bent ducts in radiation therapy in contrast to conventional ducts. To do so, a spot with only scattered radiation was used to examine the different duct geometries. We modelled different duct diameters for analysis of the impact of bent and straight ducts in dependence of their diameter. This was done for the two energies 6.7 MeV and 15 MeV. Also, neutron radiation and its dispersion within the facility and ducts was evaluated. For this case a detector in the model was placed behind the duct.

2.2.3 Gaps

In general, the radiation protection doors have a relatively high weight due to their incorporated layers of PE and lead or iron for neutron and gamma radiation shielding. Still, they need to be handled easily for daily use in radiation therapy facilities. So usually they have gaps between the floor and the door itself, so tires or rails have enough space to work. These gaps are problematic for radiation protection because scattered radiation may leak trough the gap. In short distance, only small part of a person's body is radiated. But in wider distances the radiation spreads and the small gap results in a great radiation field of gamma and neutron radiation. This is especially a problem for medical staff, working or waiting in front of the doors. So, for reduction of radiation exposure of medical staff the leakage radiation should be low. Many constructors of radiation doors offer costly lowering of the radiation protection doors. We evaluated how effective this lowering is and in which cases the gap reduction may not be the most efficient way of reducing the leakage radiation.

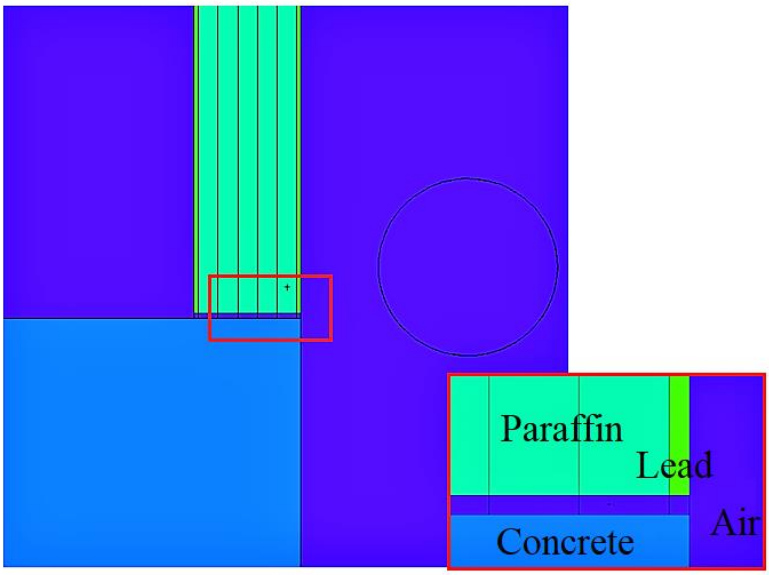

Figure 2 Simulated gap between door and floor

2.2.4 Overlaps

Where different materials for shielding are used, there may occur leakage radiation due to a lack of shielding material. This event may occur, if a zone of material variation is radiated by oblique radiation. Due to its oblique nature, the radiation is attenuated in one material in an adequate way. A second material does have different shielding characteristics and leakage may occur. This is a phenomenon which for example occurs next to doors due to a lack of overlap.

We evaluated the impact of different materials at a door and how overlaps may reduce the radiation leakage. The efficiency of three different actions was evaluated (see figure 3) with 15 MeV. First the addition of concrete blocks (1), second the relocation or extension of the door (2) and third the minimization of the gap between door and wall (3) was considered.

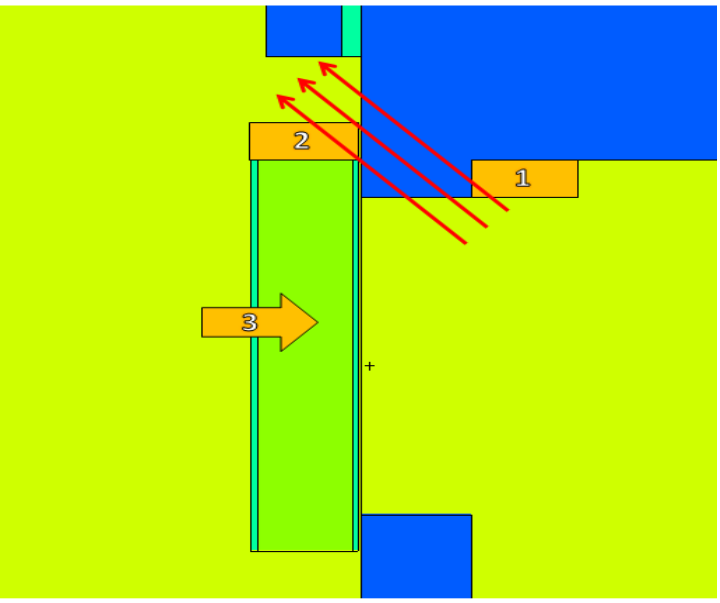

Figure 3 Three possible actions to reduce the leakage radiation due to oblique radiation at the facility door

3 SIMULATION RESULTS

All the mentioned weak spots and structural changes in radiation shielding on structural radiation protection are evaluated in simulated radiation facilities. Due to the great amount of calculation time for the evaluation of the weak spots, not the whole head was used, but the calculated radiation beam. If the whole accelerator head is simulated the losses due to apertures, shielding and flattening filter are high. Therefore, we use the resulting beam with its energy and direction and other variance reduction for increasing the calculation efficiency. Still, the relative number of particles colliding with the simulated detector is small.

3.1 Accelerator Head

As mentioned above, the modelled accelerator head was evaluated with dose-depth measurements and dose profiles. This was done for the accelerator energies of 6.7 MeV, 10 MeV and 15 MeV. Exemplary for all energies you can see in figure 4 the depth-dose profile for 15 MeV. The dose is normalized. The curve has its maximum in 25 mm depth due to build up effects. The maximum of depth dose curve is specified by the average range of secondary electrons in water. Beneath the maximum the relative dose in water decreases exponentially.

In general, high energetic radiation penetrates in greater depth, which leads to relative dose of 30 % in 300 m depth for 15 MeV. In contrast to that we reach 18 % relative dose for 6.7 MeV. As seen in figure 4 the measurement and the calculated depth-dose curve are in good correspondence. The depth dose curve provides indirect information about spectral energy distribution. Similar curves mean similar spectral dose distribution. All in all, we have maximal deviation of ca. 4-5 % in greater depth. Due to deviations at the water surface we get an average deviation of ca. 4 % for all three evaluated energies. This deviation is adequate for our field of application ‘shielding calculations’.

The dose profile is in good correspondence with the measurements. Therefore, we see an average deviation of 2.5 % to 5.5 %. For the application in shielding calculation it is an acceptable deviation, so no further improvements for the accelerator head setting were done.

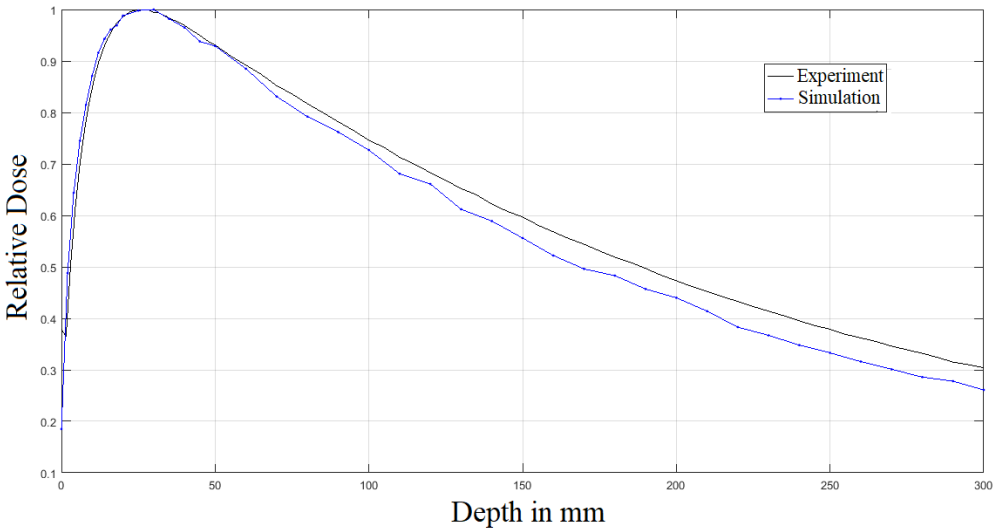

Figure 4 Depth dose profile in a water phantom for a 15 MeV accelerator

3.2 Physical Shielding Structure

With the shielding structure different dose spectra and radiation dispersion were evaluated. To do so photons with 6.7 MeV and 15 MeV maximum energy were used and neutrons due to their production in 15 MeV photon mode for the evaluation of soft spots. This was done to evaluate the range of typical energies in radiation therapy facilities. The 10 MeV maximum energy was considered for radiation dispersion within the shielding room.

3.2.1 Facility

To evaluate the worst case of accelerator head position for leakages due to weak spots, the dispersion in

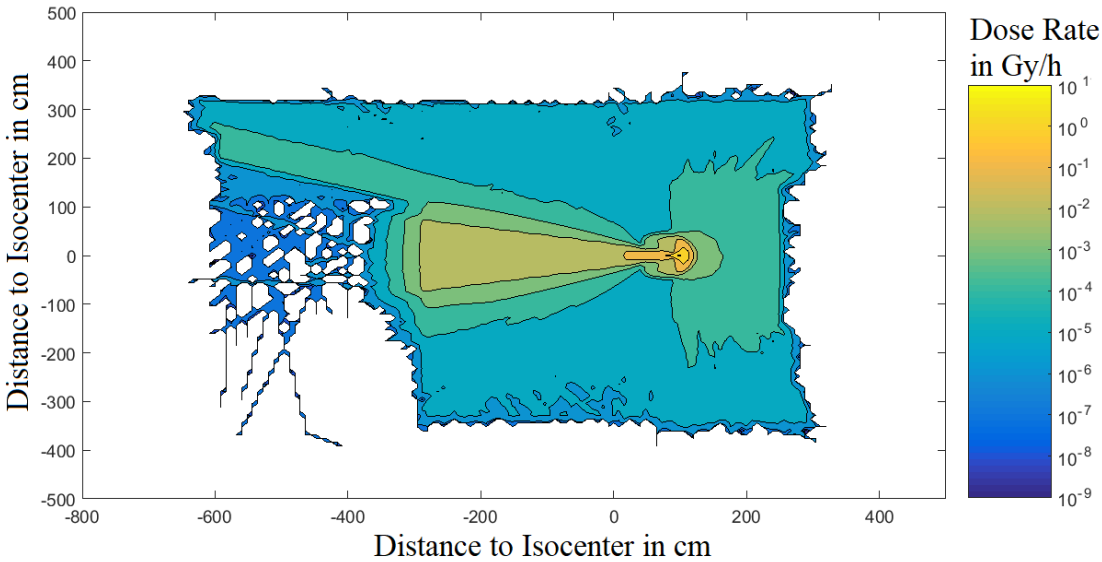

Figure 5 Gamma dose rate distribution in a treatment room with 10 MeV accelerator and 270° head rotation

the radiation facility must be evaluated. The maze in the facility prevents primary radiation to collide directly with the door. The door and potential ducts do not have such high conformance issues due to the maze. But the accelerator head can be rotated. Hence, it is possible to reach the door with the help of scattered radiation. Due to its intensity we evaluated two main directions (90° and 270°), where the accelerator head might have big impact on the radiation in the maze. This also includes the dose in front

of the door due to the gap between floor and door. The results in figure 5 show the dispersion of photons created with 10 MeV and an accelerator head position of 270°. The direct radiation on the maze wall creates a dose in the maze. So, in the following we will use the 270° accelerator head position for evaluation at the facility door. Because neutrons are scattered on heavy elements without a great loss of energy, the accelerator head position is not as relevant for the neutron radiation.

3.2.2 Ducts

Based on the most effective radiation protection, the evaluated ducts were placed in spots with scattered radiation only. The results of simulation with straight and bent ducts are shown in figure 6. In general, we see that the dose rate for gamma radiation is decreasing analogously to a decreasing duct diameter. For 6.7 MeV the difference in dose rate between a diameter of 7,5 cm and 5 cm is quite small for a straight duct. We expected the ideal duct to have better shielding characteristics than a straight duct. These problems can partially be explained by relatively high statistical uncertainties for the 6.7 MeV accelerator, which are seen as error bars in figure 6. Especially for the ideal ducts calculated at 6.7 MeV we have statistical uncertainties, which do overlap with the calculated values for straight ducts.

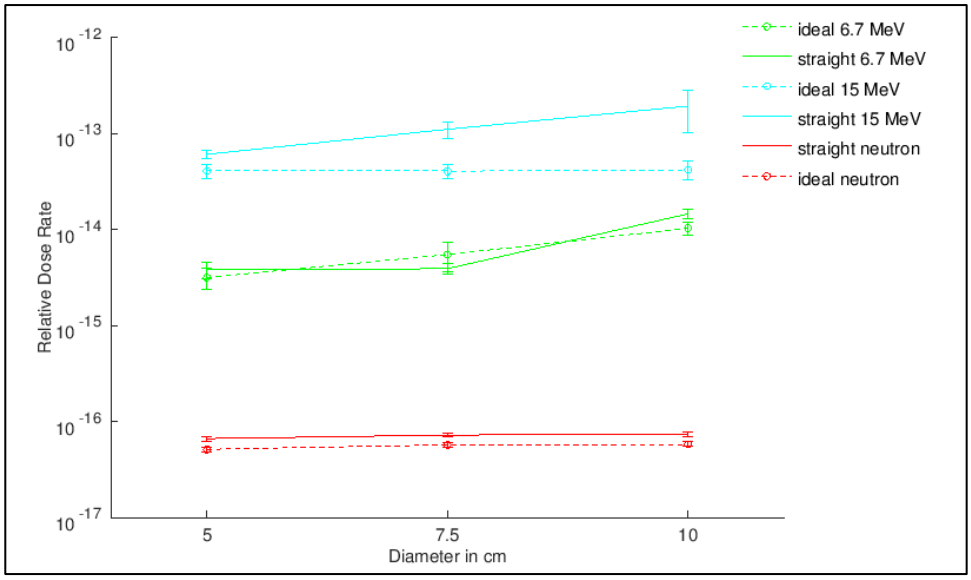

Figure 6 Comparison of dose rates for straight and bent ducts in dependence of the diameter

The expected behaviour is seen with maximum energy of 15 MeV. The dose rate behind the straight duct is analogue to the diameter decreasing. For the ideal duct, at first, we see a decrease with its diameter but then only small changes between 7.5 cm and 5 cm occur. This is what we expected, because due to the high energy the photons might not only scatter through the duct but also directly go through the wall.

For neutrons we see a better shielding when ideal bent ducts are used. The difference in diameter is not as important as it is for photons. The neutrons just scatter through the duct with heavy elements in the wall without a huge loss of energy. So, for neutrons the most important fact for ducts is its bending but not the diameter.

Some of the results are afflicted with relatively high statistical uncertainties. While the neutron deviations do not exceed the 10 %, the results for photons are afflicted with higher deviations. This can also be seen in figure 6. After all, the results are largely as we expected and give a first hint on general shielding characteristics of different duct constructions.

3.2.3 Gaps

The gap between door and floor was evaluated regarding its shielding characteristics. The dose rate was calculated with a simulated detector behind the gap. The results are seen in figure 7. The differences for dose rate in dependence of the gap size are seen easily. For the 6.7 MeV maximum energy the dose rate

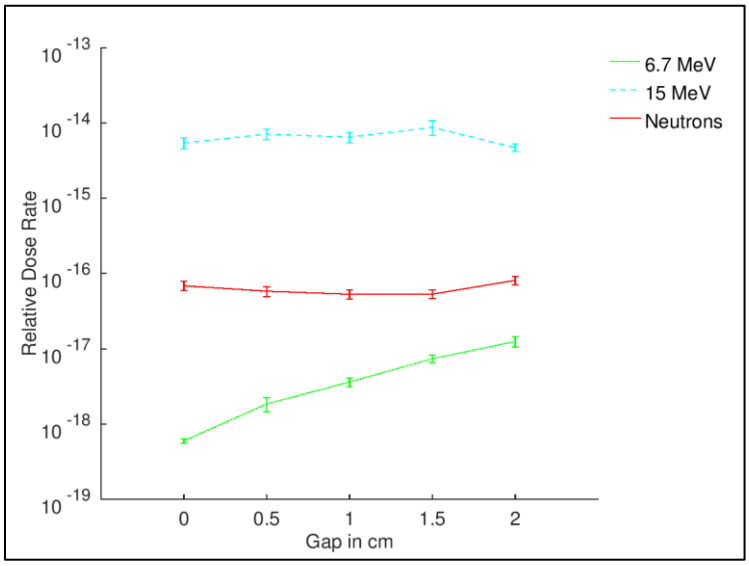

Figure 7 *Relative Dose Rate behind the door in dependence to the gap size*

is doubled with each 0.5 cm extra gap size between 0.5 cm and 2 cm. For the maximum energy of 15 MeV we see different results. In general, the variability in dependence to the gap size is way smaller. The variability expresses itself with a magnitude of two to three. We observe a similar behaviour for the neutrons. That is, for higher energy rates and neutrons the evaluated gap size up to 2 cm has no influence on the dose rate behind the door. In all cases the variability is about a factor of two.

3.2.4 Overlaps

For the examination of overlaps and oblique radiation we evaluated three different approaches to decrease the dose rate behind the overlap. On the one hand, more material like concrete was modelled, which can easily be used to upgrade the shielding, if enough space is given. Also, an extension of the modelled door was evaluated and the closing of the gap between wall and door. The result can be seen in figure 8. In the figure the attenuation for each approach is shown. For comparison the result of the original construction without improvements is shown.

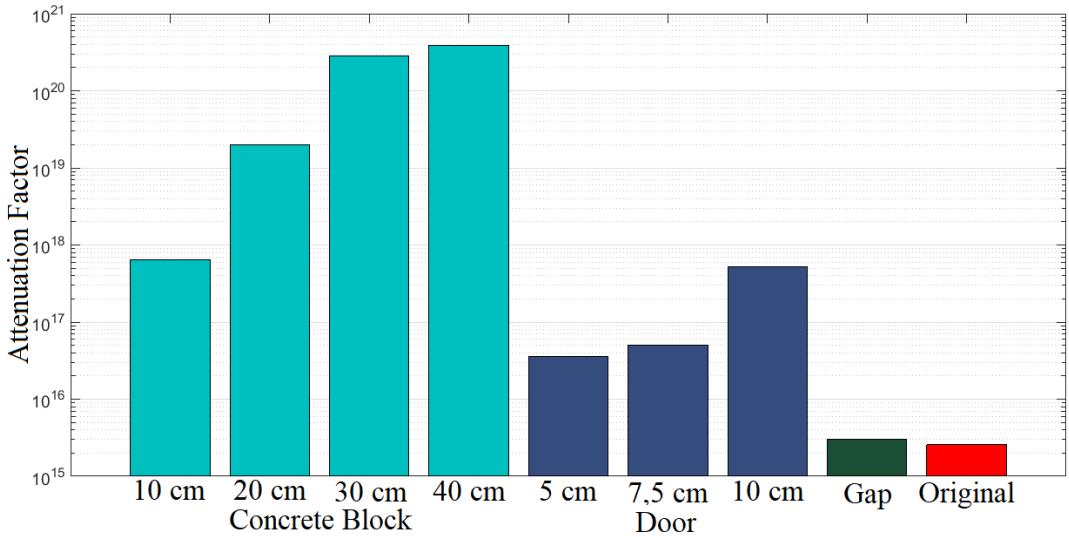

Figure 8 *Attenuation Factor by different approaches to improve radiation protection at doors*

We can conclude that the avoidance of a gap between wall and door is not an efficient method to reduce the dose rate behind the overlap. Extending the door reaches way better attenuation. The attenuation is increased 200 times if the door is extended by 10 cm. In general, you can see that the shielding is more effective when the overlap is bigger. Similar effects are reached when using concrete blocks. A 10 cm block already increases the attenuation more than 200 times. Therefore, the usage of concrete blocks for retrospective improvement of radiation shielding at overlaps seems to be a practicable and cost-efficient method if enough space is given.

Due to the high attenuation we get a maximum statistical uncertainty of 28 %. Because of the great differences between the approaches it does not have a big impact on our general conclusion.

4 CONCLUSION

4.1 Validation Calculations

The validation calculations have in general small statistical uncertainties and only relatively small deviation in comparison to the experimental determined data. Especially considering the usage of accelerator head information of different publicly available sources [4, 5] for our calculation, we created a practical model of an accelerator head. For a more realistic model the accelerator head components and the detectors could be improved. However, for the area of application 'shielding calculations' the precision is good enough, to make general prediction of shielding behaviour.

4.2 Structural Radiation Protection Calculations

For shielding calculation, it is in general a challenge to get statistical valid results. In our simulations we had relatively high statistical uncertainties. This might be acceptable when considering calibration errors of real measuring systems. Also, the global tendencies of shielding can be evaluated.

For all cases we see that extension of gaps and ducts is important for lower maximum energies. Especially for 6.7 MeV we could see that a decrease of diameter or gap also leads to a decrease of dose rate behind these structures. So, for accelerators with lower maximum energies the procedures of decreasing door gaps and using small duct diameters decrease the dose rate perceptible and should be used.

For accelerators with higher energies, in this case 15 MeV, neither for gamma radiation nor for neutron radiation we could see a significant decrease of the dose rate. For high energies, where neutron creation is involved, we recommend bent ducts, because they decrease the dose rate considerably.

For problems with leakage radiation due to a lack of overlapping materials, we recommend the retrospective installation of concrete blocks. Also, the extension (which can also be expressed as a relocation) has a good impact on the reduction of leakage radiation.

In summary, we could make general prediction about shielding construction, but further calculations should be done to confirm the evaluations made with smaller uncertainties and to evaluate more geometries.

5 ACKNOWLEDGEMENT

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Dissemination of Materials Regarding Exposure to Ionising Radiation in Diagnostic and Interventional Radiology by Using the Web Platform

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Abstract. The objective of this paper is to present the implementation of the Romanian version of some materials published by international bodies in the nuclear field within the dissemination module of the web platform. The posters and leaflets developed by the International Atomic Energy Agency (IAEA) for dissemination of practical information on radiation protection in medical exposures were translated in Romanian, with IAEA acceptance. An analysis regarding the impact of publication of materials on the platform's audience was achieved. Hence, the web platform's dissemination component has been enriched with topical materials and more capabilities have been acquired to monitor and analyse the impact of the published materials.

KEYWORDS: medical exposure, dissemination, web platform

1 INTRODUCTION

Nuclear Training Centre (CPSDN), department within the "Horia Hulubei" National Institute of Physics and Nuclear Engineering IFIN-HH, develops training programmes on radiation protection and radiation safety in medical, industrial and research practices. The CPSDN's web platform is a module of a complex strategy that involves a coherent integration of classical methods and online information management systems.

2 OBJECTIVES

The objective of this paper is to present the implementation of the Romanian version of some materials published by international bodies in the nuclear field within the dissemination module of the web platform (Fig.1). The aim is to increase the level of awareness of the visitors with regard to basic knowledge of radiation physics and radiation protection, maximizing the impact of information and dissemination actions carried out by CPSDN through training programmes.

Figure 1: Web platform of the training centre

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The results of our previous studies show a lack of knowledge about exposure levels in medical examinations with ionising radiation, even for a relatively knowledgeable audience in the field, like the visitors of the CPSDN's web platform. Based on these results, it has been decided to achieve local versions of some materials published by international bodies (IAEA) in order to increase the attractiveness of the information and dissemination module.

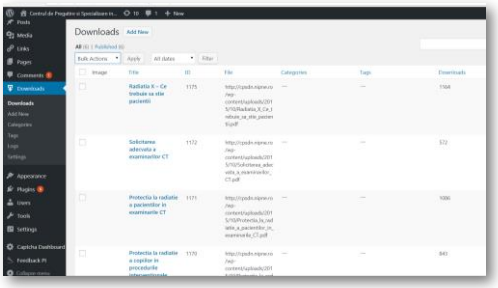
3 RESULTS

The posters and leaflets (Fig.2) developed by the International Atomic Energy Agency (IAEA) for dissemination of practical information on radiation protection in medical exposures were translated in Romanian, with IAEA acceptance. All these informational materials were then published within the platform in order to be disseminated among workers and the general public.

Figure 2: Translated IAEA's 10 Pearls Posters and the Leaflet

Because an analysis regarding the impact of publication of materials on the platform's audience has been considered useful, a tool to monitor downloads was implemented (Fig.3).

Figure 3: Monitoring tool



The recorded results (2 years) indicate the clear interest of visitors for quality information on patient protection in widespread procedures with high level of radiation (Table 1). To be noted the low interest on protection in pediatric interventional procedures, much rarer and less known.

Table 1: Results on downloads

Published material	No. of downloads
Radiation protection of patients in fluoroscopy	648
Radiation protection of staff in fluoroscopy	1398
Radiation protection for children in interventional procedures	843
Radiation protection of patients in CT	1086
Appropriate referral for CT examinations	572
Leaflet “X rays - What patients need to know”	1164

4 CONCLUSIONS

It can be considered that the web platform's dissemination component has been enriched with topical materials in the nuclear field. By adding new features, more capabilities have been acquired to monitor and analyse the impact of the published materials.

5 ACKNOWLEDGEMENTS

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EYE LENS RADIATION EXPOSURE OF WORKERS DURING MEDICAL INTERVENTIONAL PROCEDURES AND SURGERY

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To evaluates the eye-lens radiation exposure of workers during medical interventional procedures and surgery in a military hospital as well as of the equine veterinarians. The measures represent the exposure in a normal workload schedule of ninety randomly selected workers over a 3-month period, extrapolated to 1 year. The eye-lens dosemeters were placed near the eye closest to the radiation source (Carinou, E., Ferrari, P., Bjelac, O. C., Gingaume, M., Merce, M. S. and O'Connor, U. Eye lens monitoring for interventional radiology personnel: dosemeters, calibration and practical aspects of H p (3) monitoring. A 2015 review. J. Radiol. Prot. 2015;35(3): R17–R34). Three models of eye-lens dosemeters (Dosilab, Landauer and IRSN) were assessed in term of ergonomics. The annual estimation of eye-lens doses did not reach the annual dose limit of 20 mSv revised by the ICRP, ranged from 0.00 to 18.12 mSv with a mean of 0.96 ± 2.28 mSv. However, these results cannot be representative of a heavy workload or incident situations for which radiation exposure to the eye-lens could exceed this limit. The IRSN dosemeter model was considered the most convenient.

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Radiation Protection Dosimetry (2018), pp. 1–6 doi:10.1093/rpd/ncy068

Monitoring

Keeper: An Integrated System for the management of the Spanish Radiological Environmental Monitoring Data

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Keywords: Radiological Environmental Monitoring, Public Radiation Protection, Keeper, Radiological Environmental Monitoring Data Management

Abstract. In order to protect the population and the environment from the harmful effects of ionizing radiation, Spain has a comprehensive and consolidated system of radiological environmental monitoring that, through a structure of networks and programs, oversees the radiological quality of the environment in the entire country.

The results obtained by these networks and programs are stored in an integrated data management system called Keeper, which is based on five modules that provide service to different types of users and needs. The fifth and last of these modules has been developed and launched by the CSN in February 2017. It's an application hosted on the website of this organization, which, on a cartographic database and in easy to use way, gives public access to the radiological environmental monitoring data within Keeper.

With this new release, the CSN advances in the performance of the functions entrusted to this body in terms of information to the public and transparency, being one of the first European regulatory agencies, which have a tool of this kind.

1 SPANISH RADIOLOGICAL ENVIRONMENTAL MONITORING SYSTEM

The Spanish environmental monitoring system is composed of a series of networks and monitoring programs whose basic objectives are to detect and monitor the levels of radioactivity in the environment of our territory in order to determine the causes of possible increases, estimate the potential radiological risk for the population, and establish, where appropriate, the need to take preventive or corrective measures.

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The *Integrated system for the management of the environmental radiological monitoring data*, «Keeper», contains the information provided by the majority of these programs and networks, which can be divided into two main groups depending on their purpose. On one hand, the monitoring environmental radiation programs carried out by the licensee holders in the environs of nuclear power plants and nuclear fuel cycle facilities (PVRA), and on the other hand the national sampling stations network (REM), which together with the network of automatic stations (REA), constitute the national monitoring network not associated to facilities (REVIRA). Some other special monitoring programs also have their room in Keeper, such as those that track unusual events or existing exposure situations. The common goal of REM and PVRA is to collect samples in the field, which will be taken to laboratories, specialized in low-level measures, where they are analyzed. The different types of samples and determinations carried out, aim to represent the different potential routes of radiological exposure to the population caused by different sources.

1.1. Radiological environmental monitoring programs in the environs of the facilities (PVRA)

Currently, there are five nuclear power plants operating in Spain, one in a definitive shutdown situation, and two in a dismantling stage. There is also a fuel assembly manufacturing facility, a solid radioactive waste disposal facility, a nuclear research facility under decommissioning, two former uranium concentrate manufacturing plants under closure, one under decommissioning and some special existing exposure situations. All of them have a PVRA in their environs.

According to the Safety Guide 4.1 of the CSN [1], a PVRA has a number of sampling stations around facilities (see Fig. 1) which will include samples of air (dust particles and radioiodine), gamma radiation (in 16 sectors of the wind rose from the center of the installation), rain or dry deposition depending on if it rained or not in the sampling period, soil, water (drinking, surface and groundwater), sediment, fauna, flora and food (milk, vegetables, meat, eggs, fish and seafood). The results of these samples are the ones, which, since 1981, have been sent to the Keeper database in the first quarter of each year following the completion of the annual campaign of the previous year.

Figure 1: PVRA sampling stations

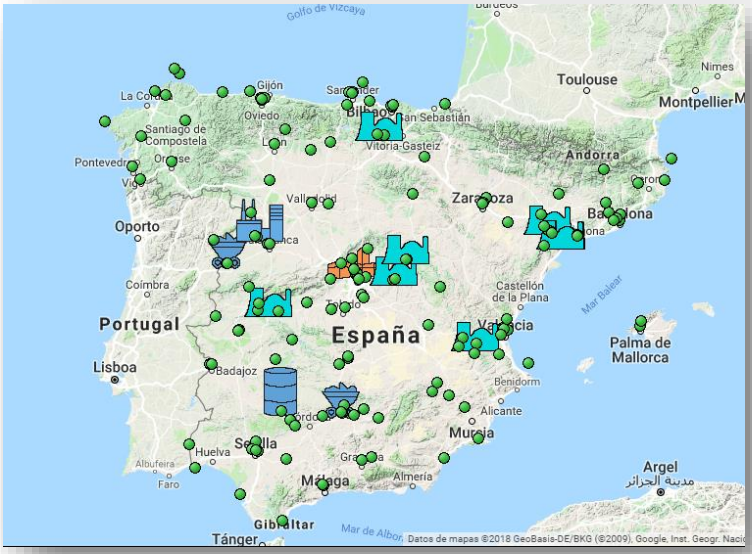
Keeper also contains data from the quality control programs (CC) of the facilities: These programs are also carried out by the licensees, following regulatory requirement, by taking 5%-10% of duplicated samples and analyzing them in a different laboratory with the objective of confirming the analytical PVRA results. Finally, Keeper also has data from the independent monitoring programs (PVRAIN) carried out by the CSN under agreement with universities nearby the facilities, which are responsible for the independent sampling and analysis of the 5%-10% of the PVRA samples, with the objective of verifying the data reported by the licensee.

1.2. National Radiological Monitoring Network, not associated to facilities (REM)

Nationwide and not associated to facilities, the REM network includes two types of surveillance programs: the surveillance of the earth and atmosphere and the surveillance of the continental and coastal waters. The former, was initiated by the CSN in 1992, under agreements with several laboratories of universities and research centers distributed throughout the country. Currently, it is carried out by 20 laboratories that sample and analyze air, soil, water, milk, and mixed diet (dense network). The latter, includes data supplied by the Ministry of the Environment (Cedex), as a result of the sampling and analysis of the main river waters since 1986 and coastal waters since 1992 (see Figure 2).

In the framework of articles 35 and 36 of the Euratom Treaty, these programs were expanded in 2000 on the basis of a recommendation from the European Union [2], by the development of a sparse monitoring network, additionally to the dense one, which is made up of 6 laboratories distributed throughout the country, which are able to carry out high sensitivity measurements.

Figure 2: REM sampling stations

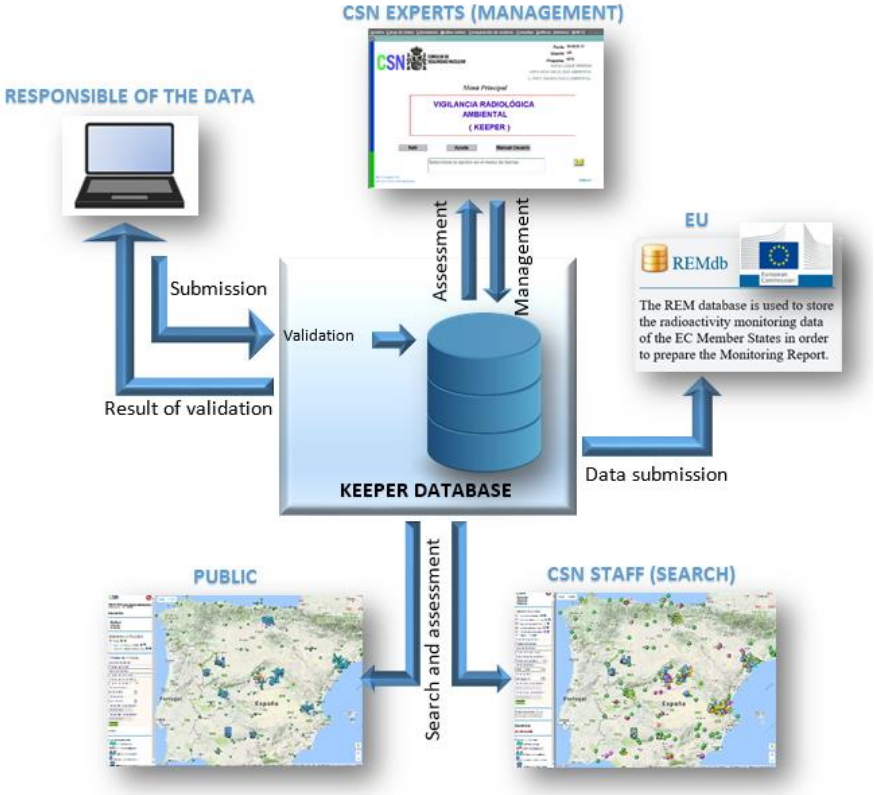


2. KEEPER, THE CSN INTEGRATED SYSTEM FOR THE MANAGEMENT OF THE RADIOLOGICAL ENVIRONMENTAL MONITORING DATA

Keeper was developed in 1993 with the data stored in the computer program ANATHEMA of the CSN, in which the results of the programs, as well as historical data from the former Nuclear Energy Joint (JEN) were kept. Since then, this database has been under continuous development, having currently more than three and a half million records (each record includes information such as the facility code, sampling date, sampling point, sample, analysis, radionuclide, activity, uncertainty and minimum detectable activity, among others) with the radiological quality of the environment of the past 40 years.

The main goal of this system is to process a large amount of data in a quick and reliable manner, in order to ease their assessment by the experts, adapting its architecture to the needs of the CSN for the accomplishment of its obligations. For this purpose, the CSN's Deputy Direction for Environmental Radiation Protection together with the Deputy Direction for Information Technologies have developed along the years this System, which nowadays is a very complete tool, launching its last module in February 2017. As a result, Keeper is supported by five modules, which are shown in Figure 3.

Figure 3: Concept map of Keeper.



Telematic delivery: Hosted on the electronic site of the CSN, this tool is used by the owners of facilities, laboratories and bodies responsible for the data to load directly into Keeper results making a first validation of their own data that allows them to detect format errors, among others. Once validated, the CSN staff completes the loading receiving the feedback of qualitative and quantitative anomalies. If a value exceeds certain levels of activity, uncertainty, LID (lower detection limit) or certain historical ranges of measure, Keeper will provide a warning.

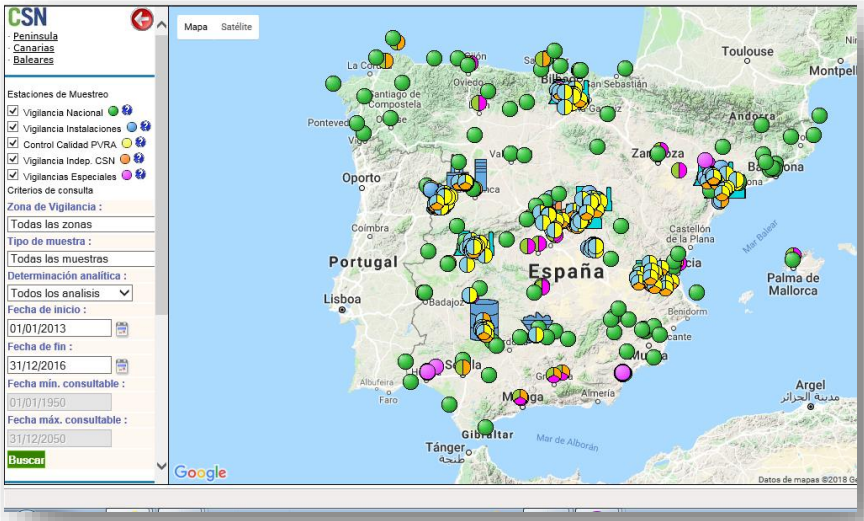
Keeper database: It is the core of the system, a powerful tool for the management and evaluation of the data that can only be accessed by the experts of the Radiological Environmental Monitoring Department through a username and password, who have full clearance. Currently, Keeper has a series of menus from which you can check the stored information, including, additionally to the results of the environmental measures, extra information such as the satellite view in Google Earth of the sampling stations, a library with reference values of different radionuclides in various types of samples or the yearly sampling calendar with a weekly schedule of samples to be taken by each installation. Keeper has tools to carry out statistical analysis of the data, filtering them by installation, sampling station or type of sample, among others, with the aim of identifying trends. Furthermore, it can compare activities and AMD, with different reference levels that are constantly updated according to the regulations, with different life stages of the facilities (pre-operational, operational, decommissioning, and post-closure), and with the results of the different quality control or independent programs, to analyze their consistency, coherence or overlap.

Completing the multiple possibilities of Keeper is the Oracle Business Intelligence Discoverer module, which is a powerful package of tools that enables ad-hoc consultations and report issuing, and a module for the graphical analysis of data, which can display their temporal evolution by installation, station or sample, among others.

Submission of data to the European Union: In the framework of articles 35 and 36 of the Euratom Treaty, the EU member states shall send their environmental monitoring data annually to the European Commission *Radioactivity European Monitoring database* [3] in a certain format. For this purpose, Keeper provides a module that generates the required files that can be easily loaded to the REM database.

Corporative Keeper: Based on Google Maps, this module is hosted since 2016 on the internal network of the CSN, enabling the whole staff of the regulatory body to check, analyze, graph and export all data from the Keeper database in a user-friendly way, in support of their technical work (see figure 4).

Figure 4: Cartographic corporative application



New module for public access: This new tool, also based on Google Maps, has been launched in February 2017, and is accessible from the website of the CSN, allowing the public to consult most of the PVRA and REM data through the internet. This module is described in detail in the following section.

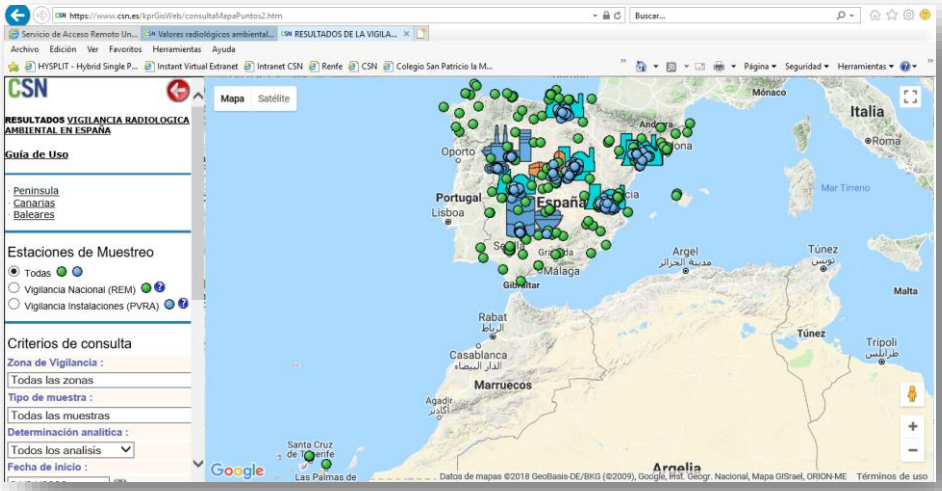
3. KEEPER's NEW APPLICATION FOR PUBLIC ACCESS TO THE DATA

Article 2 of the law of creation of the CSN [4] establishes among the functions entrusted to this body the "Control and monitoring of the radiological quality of the environment" and

"Information to the public on matters of its competence". With this aim, the CSN presents annually to the Chamber of Deputies and the Senate, a report [5] that includes a summary of the results obtained by the environmental radiation monitoring system in Spain. In addition, it also issues an annual technical report with the results of the radiological environmental [6] monitoring programs and networks of the previous year. Law 27/2006 of 18 July regulates the right of access of the public to environmental information [7], making public authorities ensure its dissemination by adopting the necessary measures, easily through public telecommunications networks. In this context, the CSN, in its continuous process of improvement in transparency and public information, launched in September 2012, the Keeper-Web project, whose objective was the development of an application that would give public access to the environmental radiological data contained in Keeper.

The project was carried out by the Radiological Environmental Monitoring Department (AVRA) in collaboration with the Department of Development of Applications (DESA), both of the CSN, who have worked to make an in-depth review of logs and encodings contained in the database, giving them the suitable format for their publication. In February 2017, it was opened to the public, through the following link: https://www.csn.es/valores-radiologicos-ambientales-pvra-rem, or through the "Operating States and environmental data" section of the homepage of the CSN.

Figure 5: New tool for public access to Keeper.



The new application has a friendly interface, whose initial screen is divided in two parts: map and selection panel. On both sides of the screen, there are active buttons that connect us with different contents of the CSN website, with the description of installations that develop PVRA, or with the user's guide. The map allows different views (satellite images, maps, or changing the zoom level) and can focus on the peninsula, Balearic or Canary Islands. The dots on the map are the REM and

PVRA sampling stations, and on the left side of the display panel, different consultation criteria can be defined (surveillance zone, sample type, radionuclide, sampling period), showing on the map, as a result, the stations that meet them. Currently, the range of available dates is from 2006 to 2016. This period will be extended annually upon receipt and review of the data of the previous year.

By clicking on the icon of each of the stations, radioactivity values are displayed, showing information about the time interval of data, station, surveillance (dense network, sparse network), sample type, isotope determination or the number of records that exist on that query. Finally, by clicking on any of the isotopes the results are shown in graphs and tables. The AMD values are shown in white dots and the activity values are shown in green ones. By positioning the mouse over any of them, the final sampling date is shown, as well as the activity and uncertainty values. Both, graphs and tables, can be printed and exported to Excel.

4. CONCLUSION

Keeper is a mature and solid system for the management of the Spanish radiological environmental monitoring data, which is under constant improvement and development in accordance with the state of the art and the times we're living in. The development of the new public access application has been a major effort for the Spanish Nuclear Safety Council, which puts our system of environmental radiological monitoring among the most advanced ones, strengthening the CSN's commitment with public information and transparency.

5. REFERENCES

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[2] Commission Recommendation of 8 June 2000 on the application of Article 36 of the Euratom Treaty concerning the monitoring of the levels of radioactivity in the environment for assessing the exposure of the population as a whole. Official Journal of the European Communities, L 191: 37-46, 27.7.2000. http://eur-lex.europa.eu/legal-content/EN/TXT/?uri=CELEX%3A32000H0473

[3] European Radioactivity Environmental Monitoring (REM) Web Site: https://rem.jrc.ec.europa.eu/RemWeb/

[4] Act 15/1980 of 22 April, of creation of the Nuclear Safety Council https://www.csn.es/normativa-del-csn/ley-del-consejo-de-seguridad-nuclear

[5] Annual report of the Nuclear Safety Council to the Congress of Deputies and the Senate https://www.csn.es/documents/10182/13529/Informe+anual+2016

[6] Annual report of results of radiological environmental monitoring programs https://www.csn.es/documents/10182/27786/INT-04-38+Programas+de+vigilancia+radiol%C3%B3gica+ambiental.+Resultados+2015

[7] Law 27/2006 of 18 July by which regulate the rights of access to information, public participation and access to justice in environmental matters

Naturally-Occurring Radioactive Materials (NORM)

NORM: EU Directive Implications beyond EU Borders

N. Tsurikov (Keynote speaker)

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Abstract. The lecture provides general overview of the management of Naturally Occurring Radioactive Materials (NORM) that is required in accordance with the Council Directive 2013/59/EURATOM of 5 December 2013.
Following a brief introduction to the issue of NORM, particular attention is drawn to the implications of the adoption of the document beyond the borders of the European Union:

- The potential problems associated with trade in commodities containing NORM, their transport and entry into the EU through the international borders; and
- The implication of the adoption of the principles and limiting values listed in the Directive in countries outside the EU.

The examples of the following instances are provided –

- Verbatim adoption of the limiting values in developing countries and associated implications for the management of the overall health of the population and the state of the environment in these countries; and
- Advice provided in the past by some EU experts to developing countries, mostly through community organisations, with limited understanding of the local culture, on the basis of incomplete (or incorrect) data and without consideration of long-term implications of the information and advice provided.

1. DEFINITIONS

Many different definitions are used to describe 'Naturally Occurring Radioactive Material', NORM. The 'natural source' is defined in both 1996 [1] and 2013 [2] EURATOM Directives, but not 'NORM'.

The Safety Glossary of the International Atomic Energy Agency (IAEA) [3] defines NORM as –
Radioactive material containing no significant amounts of radionuclides other than naturally occurring radionuclides
To which we can add:
Material designated in national law or by a regulatory body as being subject to regulatory control because of its radioactivity.

Effectively, almost everything is "NORM" and a qualifier above is definitely needed. For the material to come within the scope of regulations it has to be disturbed or altered from natural settings, or contain technologically enhanced concentrations of natural radionuclides due to human activities; which may result in an increase in relatively significant radiation exposures and risks to the public above background radiation levels [4].

2. INTRODUCTION

Article 23 of the 2013 Directive [2], on NORM, states –
Member States shall ensure the identification of classes or types of practice involving naturally–occurring radioactive material and leading to exposure of workers or members of the public, which cannot be disregarded from a radiation protection point of view. Such identification shall be carried out by appropriate means taking into account industrial sectors listed in Annex VI.

What we need to consider is that several such lists exist and, whilst they are generally similar, there are some important differences. Let us compare the lists of industries of interest from the 2013 EC Directive [2], International Atomic Energy Agency (IAEA) [5], and Australian Radiation Protection and Nuclear Safety Agency (ARPANSA) [6], in Tables 1 and 2.

Table 1: Similarities in the lists of industries

Industry	EU [2]	IAEA [5]	Australia [6]
Thorium compounds and products	X	X	X
Niobium	X	X	X
Oil and gas production	X	X	X
Titanium pigment	X	X	X
Thermal phosphorus	X	X	X
Phosphate fertilisers	X	X	X
Phosphoric acid production	X	X	X
Zircon and zirconia	X	X	X
Coal fired power plants	X	X	X
Iron and steel	X	X	X
Tin, lead, copper	X	X	X
Mining ores other than U	X	X	X

Table 2: Differences in the lists of industries

Industry	EU [2]	IAEA [5]	Australia [6]
Rare earths	From monazite only	From all minerals	From all minerals
Tantalum	X	-	X
Geothermal energy	X	-	X
Cement, clinker ovens	X	-	-
Water treatment	Only ground water	All water	All water
Aluminium	-	X	X
Zinc, lead	-	X	-
Mineral sands	-(1)	-	X
Scrap metal recycling	-	-	X
Tunnelling	-	-	X
Building industry	X(2)	-	X
Missing from all three lists			
Paper and pulp production(3)	-	-	-
Hydraulic fracturing(4)	-	-	-

Notes:
(1) Mining and processing of heavy mineral sands occurs in Norway and Ukraine.
(2) Building industry is listed separately, in Annexes 8 and 13 of the 2013 EC Directive [2].
(3) The pulp and paper industry in Europe accounts for approximately a quarter of an overall world production and is a major employer [7]; it is not known if any investigation of NORM-associated issues were carried out in the EU.
(4) It is not clear if hydraulic fracturing would be included into 'oil and gas production' listed in Table 1.

3. WHAT IS IN THE SCOPE OF REGULATIONS

Prior to 2013 there were significant differences between the levels applicable in the EU and those used internationally, and a detailed discussion was provided in the Comparative study of EC and IAEA guidance on exemption and clearance levels, in 2010 [8]. Two documents were compared – IAEA Safety Guide on the Application of the concepts of exclusion, exemption and clearance [9] and EU Guidance on the practical use of the concepts of clearance and exemption [10].

It has been shown that –
…the overall correspondence of both sets of values is poor. The ratios span a large range between 0.01 and 100, with many nuclides being two orders of magnitude less restrictive in RS-G-1.7 than in RP 122 part I, including the important nuclide Pa-231 and many other radiologically important nuclides.
The illustration is given in Table 3, reproduced from page 10 of EU RP-157 [8].

Table 3: List of radionuclides of natural origin grouped according to the ratios of values between IAEA RS-G-1.7 [9] and EU RP-122 Part I [10]

Ratio up to…	Nuclides
0.01	Th-231
0.1	Bi-210, Th-234
1	Ra-223, Ra-224, Th-227, U-234, U-235, U-238
10	K-40, Th-228, Th-230
100	Pb-210, Po-210, Ra-226, Ra-228, Ac-227, Th-232, Pa-231

It should be noted that the differences above no longer exist; the values in the 2013 Directive [2] are the same as in the IAEA Basic Safety Standards [11]. Table A Part 2 of the Annex VII of the 2013 Directive [2] and Table I.3 of the IAEA BSS [11] give the same values for exemption or clearance for naturally occurring radionuclides in secular equilibrium with their progeny:

- Each radionuclide in the uranium decay chain or the thorium decay chain – 1 Bq/g
- Potassium-40 – 10 Bq/g

Or course, it does not mean that the full scope of regulations needs to apply as soon as these concentrations are exceeded, the graded approach should be used:
IAEA BSS [11], paragraph 2.18: *The government shall ensure that a graded approach is taken to the regulatory control of radiation exposure, so that the application of regulatory requirements is commensurate with the radiation risks associated with the exposure situation.*
Article 24.1 of 2013 EU Directive [2]: *Member States shall require practices to be subject to regulatory control for the purpose of radiation protection, by way of notification, authorisation and appropriate inspections, commensurate with the magnitude and likelihood of exposures resulting from the practice, and commensurate with the impact that regulatory control may have in reducing such exposures or improving radiological safety.*

4. NORM – NOT WELL-KNOWN ISSUES AND PROBLEMS

4.1. Relatively high levels of radiation exposure

Typically, NORM is not associated with levels of radiation exposure that may result in a worker approaching even 25% of the annual exposure limit. However, this is not always the case, as illustrated by the following exposure situations – all three observed personally by the author.

4.1.1. Thorium-containing mineral monazite

A rare earth phosphate mineral, monazite, is a typical component of heavy mineral sands deposits around the world and contains around 7% of thorium. It may be present in enhanced concentrations in a deposit itself, or may be concentrated during separation of different minerals.
The gamma dose rates in order of 140 – 180 µSv/hour are not uncommon, thus a worker may reach an annual limit of 20 mSv/year in approximately two weeks, not even accounting for the exposure due to the inhalation of dust and radon/thoron progeny.

4.1.2. Abandoned 'legacy' sites

There are many so-called 'legacy' sites around the world, where the mining/processing operations were simply abandoned without any remediation. Most of them are associated with uranium mining and production, but there is also a significant number related to NORM in general.
For example, at one of the legacy sites in Ukraine (abandoned thorium and uranium production), gamma dose rates may reach 1,300 µSv/hour; therefore a worker may reach an annual limit of 20 mSv/year in only two days, only from gamma radiation. Additional information about this site has been presented at the NORM-VIII Symposium in 2016 [12].

4.1.3. Build-up of radon in enclosed environments

A significant amount of minerals and concentrates are transported around the world inside sea containers and in bulk, in hermetically sealed hulls of bulk carriers.
It is not a well-known fact that in these situations concentrations of radon emanating from the material containing only 1 – 1.5 Bq/g of uranium may reach 8,000 or more Bq/m^3, within approximately 48 hours. Therefore, the workers opening sea containers or unloading bulk mineral shipments may reach an annual limit of 20 mSv/year within 4-5 weeks.

4.2. Additional health, safety and environmental hazards

Additional, and very serious, health and safety hazards are very often observed in situations where NORM is present. As a rule, these hazards are much more severe than any presented by potential radiation exposures and must be addressed first. The examples, also personally observed by the author, include:

- Nests of feral bees in a rusted drum with NORM in Australia (approximately 20 bites are likely to result in a fatality),
- Abandoned NORM waste mixed with chemicals, asbestos and mercury in Asia (representing an immediate health hazard),
- Wall cracks in underground mines in Europe, Australia and Africa (possibly resulting in multiple fatalities),
- Significant problems with acid mine drainage in Africa, Asia and Australia (resulting in major environmental problems).

4.3. Detection of radioactivity at EU borders

Even if a material is exempt from the transport safety regulations and the associated signposting, the concentrations of radionuclides may cause elevated gamma radiation levels outside the packages (e.g. sea containers). The equipment that is used at border crossings and in ports worldwide easily detects these levels.

The complexity of regulations dealing with transport of potentially radioactive materials and very small differences that may or may not qualify NORM for exemption are typically hard to understand – even for a regulator. A full understanding of the regulations can hardly be expected from a customs official, normally dealing with many other (and very different) matters on a day-to-day basis.

Therefore, the transport documentation for NORM needs to contain detailed information about the concentrations of naturally occurring radionuclides in this material, irrespective of the applicability of transport regulations. All necessary information, such as gamma spectrum for the material, may be provided in the document that is accompanying every shipment – Material Safety Data Sheet (MSDS). Whilst not absolutely necessary, this information would assist in the process of clearing a particular NORM through the radiation detection equipment at international border crossings.

5. POTENTIAL ADOPTION OF THE 2013 DIRECTIVE INTO THE REGULATION OF A NON-EU COUNTRY

5.1. Potential issue

It is quite possible that the text of the EU 2013 Directive [2] (or some parts of it) will be adopted into the regulations in some developing countries, especially in Africa, for the following reasons:

- Regulators in the francophone countries tend to rely more on the documents from the EU, which are immediately available in French, and not on the IAEA ones;
- As a rule, only Safety Standards of the IAEA are translated into six official UN languages;
- The IAEA documents are not translated into the languages such as Portuguese.

There are two paragraphs that specify the limits of radiation exposure:

1. Article 7.2 discussing reference levels and referring the user to an annex.
2. Article 12.2 telling the user exactly what the limit should be.

It is, therefore, very likely that the Article 12.2 will be used – as it gives clear and exact instructions. The Article 7.2 is likely to be ignored, as the concept of reference levels is typically not well understood, and the Article 12.2 seems to over-write what Annex I may be suggesting, in any case.

Article 7.2: *The choices of reference levels shall take into account both radiological protection requirements and societal criteria. For public exposure the establishment of reference levels shall take into account the range of reference levels set out in Annex I.*
Annex I: *Without prejudice to reference levels set for equivalent doses, reference levels expressed in effective doses shall be set in the range of 1 to 20 mSv per year for existing exposure situations.*

However, irrelevant of what Article 7.2 and Annex I say, Article 12.2 provides clear instruction:
Member States shall set the limit on the effective dose for public exposure at 1 mSv in a year.

Even if there is a discussion in a country about planned or existing exposure situations – in each and every case (except, possibly, abandoned 'legacy' sites), the same 1 mSv/year will still apply.
Article 100.3 on existing exposure states –
Existing exposure situations which are of concern from a radiation protection point of view and for which legal responsibility can be assigned shall be subject to the relevant requirements for planned exposure situations and accordingly such exposure situations shall be required to be notified as specified in Article 25(2).
It should be noted that §3.4 of the IAEA 2014 BSS contains similar requirements.

The main problem with the adoption of 2013 Directive in non-EU countries is associated with so-called "precautionary principle", which usually interpreted in a way that in dealing with potentially hazardous technologies the benefit of the doubt must go to the public and not to technologies. The combination of this principle with the uncertainty about health effects of low level ionising radiation means that a theoretical possibility "a small dose may cause harm" is transformed into an axiom "a small dose most definitely will cause harm".

The implementation (basically, copying) of Article 12.2 of 2013 EU Directive into the regulations in developing countries could (and most likely will) lead to the diversion of limited funds from other more important health problems of the population as a whole.

It should be noted that over-regulation typically results in huge costs, despite the fact Linear-No-Threshold dose response model still being just a hypothesis, not a conclusively proven fact.
Each human life hypothetically saved by implementing the US Nuclear Regulatory Commission's regulations costs about $2.5 billion. Such costs are absurd and immoral when compared to the costs of saving lives by immunisation against measles, diphtheria and pertussis, which in developing countries range between $50 and $99 per one life saved. [13]

Of course, radiation is not the only low-level risk that is over-regulated. There appears to be an obsession with regulating low risks and an overall blindness to the diseases that are often fatal (measles, malaria and tuberculosis), and to other dangers that kill tens of thousands of people every year, such as prescription opioids and alcohol.
Developing and applying regulations intended to reduce risk from minor or hypothetical hazards (such as low-level radiation):

- Gives elected officials an opportunity to say "we are here to protect you";
- Provides support for the scientific research that may not be needed, and for the government departments that, in some cases, have much more staff that is necessary; and

- Appeases BANANA's – people of the following opinion: "Build Absolutely Nothing Anywhere Near Anything" [14].

5.2. An argument and a possible solution

J-F Lecomte from ICRP said at NORM-VIII symposium in 2016 –
NORM are existing exposure situations, because the source is not deliberately introduced, it already exists when a decision on control is taken; concentration and dissemination of radionuclides are incidental. …some control is needed and should be provided; the level of protection should be commensurate with the risk [15].

This may not be entirely correct. If the decision is made to open a new mine, bring up the ore that is rich in uranium and thorium and process it, concentrating radionuclides significantly in the process – the source appears to be deliberately (although incidentally) introduced. Each and every new operation dealing with NORM appears to deliberately introduce the source – we could interpret this in a way that this "introduction" is associated with the regulations. The source may have existed at 0.1 Bq/g of thorium in the ground, but if after processing the concentrations of thorium reach 100 Bq/g – the regulation of the material becomes necessary.

Linking the planned exposure situation with the possible use of material for its radioactive properties appears to be a re-introduction of the concept that was present in the transport safety regulations prior to 2012, where the exemption was given or not given to the same copper concentrate depending on where it was transported – to a copper smelter or to uranium processing plant... Which was a cause of a continuous confusion, both for the industry and for the regulators.

Following a similar logic it could be argued that nuclear fuel is also NORM, as both ^{238}U and ^{235}U already existed – just the ratio has changed during enrichment… Yes, ^{235}U may have been introduced for its radioactive properties, but ^{238}U in depleted uranium – wasn't.

- So – is depleted uranium a NORM residue/waste then?

Let us consider an example where a new operation is proposing to process copper ore containing uranium and due to some objective conditions (economic, political, etc) the ore with relatively high concentrations of uranium is "put aside for the future":

- At the commencement the operation would be an existing exposure situation.
- However, when in several years uranium processing circuit is constructed and uranium concentrate is produced – the situation will become 'planned'. Or will it be "partially planned'...?
- A single worker operating half of the shift in copper flotation circuit and another half – in uranium extraction circuit will be in two exposure situations at the same time.

It is obvious that the situation will be absurd, as different radiation protection approaches and limits may need to be applied to the same worker at the same time.

Another ridiculous example would be a situation of two ^{226}Ra atoms floating in a river side by side, one being from a farmer's field (due to fertiliser use), another one – from a 'nuclear' facility. Then:

- The one from the farmer's field would be "harmless" (NORM, existing exposure situation), but –
- Another one (whilst exactly the same) acquires some magic powers and must be managed properly (nuclear, source is deliberately introduced, planned exposure situation).

A possible solution may be similar to the three "exposure bands" for workers:

- Less than 1 mSv/y
- From 1 mSv/y to a few mSv/y
- From a few mSv/y to 20 mSv/y

Or similar to the ICRP approach to the reference levels for the exposure to radon in workplaces, described by JF Lecomte at the NORM-VII Symposium in 2013 [16], that is based on the reference level of 10 mSv/year.

Could a recommendation be made on the application of the reference levels to the industries dealing with NORM?

A typical operator would always be of the opinion of "just give me the number that I should not exceed". The "practice – intervention" concept was not well understood, and the introduction of variable reference levels will, undoubtedly, result in a confusion for many regulators, who would simply use "the lowest denominator" of 1 mSv/year…

6. VERBATIM ADOPTION OF LIMITING VALUES

A verbatim adoption of the limiting values in developing countries may lead to the implications for the overall health of the population and for the state of the environment in these countries, due to potential over-regulation in a situation where resources are very limited.
The adoption of limiting values from the EU or from similar regulations in developing countries is a well-known fact and two examples can be given, without specifying the countries themselves. Both of the examples are associated with gross alpha activity concentrations in the drinking water. It should be noted that these examples are from the countries where much more serious health problems exist.

6.1. Adopting the US EPA limit after its fifteen-fold reduction

US EPA limit for gross alpha concentration in the drinking water in the document that was used in the development of "water quality regulations" was 15 pCi/L (0.555 Bq/L) [17]. In one of the African countries this limit was reduced fifteen times, to 1 pCi/L (0.037 Bq/L) and is legally applicable – despite the fact that there appears to be no laboratory in this country that could possibly measure gross alpha activity to such low levels. It seems that the values for Ra-226 and Ra-228 from the table describing minimum detection limits in the same document [17] were confused with limiting values.

6.2. Adopting the EU limit instead of WHO one

Guidelines for Drinking-water Quality of the World Health Organisation [18] state –
Screening levels for drinking-water below which no further action is required are 0.5 Bq/litre for gross alpha activity and 1 Bq/litre for gross beta activity. …The screening level for gross alpha activity is 0.5 Bq/litre (instead of the former 0.1 Bq/litre).
However, draft regulations in one of the countries in West Africa, to be adopted in 2018 say –
Le seuil de contrôle recommandé pour l'activité alpha globale est de 0,1 Bq/l. Le seuil de contrôle recommandé pour l'activité bêta globale est de 1,0 Bq/l.
The text was copied directly from Annex III of another 2013 EC Directive: 2013/51/Euratom of 22 October 2013, on radioactivity in drinking water [19].

7. ADVICE BY EU EXPERTS TO DEVELOPING COUNTRIES

Unfortunately, an advice provided by some EU experts to developing countries was, in some cases, not entirely correct. The work was often carried out with limited understanding of the local culture, on the basis of incomplete (or incorrect) data and without consideration of long-term implications of the information and advice provided. The local regulatory authorities were, in many cases, not even aware of the fact that EU experts were present in the country and it was a big disappointment for them to discover that reports about their countries have been published without their knowledge.

7.1. Namibia

The Environmental Justice Organisations, Liabilities and Trade (EJOLT) is a project that was supported by the European Commission in 2011-2015. The project was to "support the work of Environmental Justice Organisations, uniting scientists, activist organisations, think-tanks, policy-

makers from the fields of environmental law, environmental health, political ecology, ecological economics, to talk about issues related to Ecological Distribution".

Two EJOLT reports were published in 2014: *Radiological Impact of Rössing Rio Tinto Uranium Mine* [20] and *Study on low-level radiation of Rio Tinto's Rössing Uranium mine workers* [21].
Unfortunately, the researchers forgot altogether to consult with the National Radiation Protection Authority of Namibia and even to tell them about the project. The possible effects of radiation were significantly over-exaggerated; for example, one of the reports stated that –
In the case of the Rössing parking, spending 5 minutes per day during 200 working days gives an additional exposure in excess of 10 microSieverts. This is considered a "significant exposure" [20].
Of course, 10 microSieverts per year cannot possibly apply to the exposures from NORM due to the variability of natural background. Furthermore, the radiation levels at this parking spot, which the author surveyed personally, appear to be simply an elevated natural background in the area – which one would expect in the vicinity of a uranium ore body…

7.2. Zambia

Another example is the radiation protection report funded by Norwegian Church Aid [22], prepared by experts who appear to be opposed to uranium mining anywhere in the world. The copy was obtained by the author during the visit to the Council of Churches in Zambia, and neither National Institute for Scientific and Industrial Research nor Environmental Management Agency of Zambia were aware of a visit to the country and of the contents of this report. It is understood that a significant number of people lost their jobs as a direct or indirect result of the publication of this document.

7.3. Gabon

The third illustration of the problem is the EU report on the potential use of contaminated materials in the construction of residential homes in Gabon and Niger [23]. The European Parliament's Committee on Development requested this study to be undertaken.
The appropriate regulatory authority in Gabon was not contacted (and thus was not aware of the existence of the document), and the authors of the study did not even take any radiation monitoring equipment for the visit to Gabon, therefore no conclusions of the report could be verified. For Niger only a desktop study was conducted.
In addition, photographs presented in the document illustrated that there were still remnants of the old processing plant present at some location on the site.
But during the extensive audit undertaken by the author in 2011 including visits to all accessible areas of the site and during the numerous discussions with the site personnel and with the members of the public it was concluded that these photographs are likely to have been taken 7-8 years before the report was published, when the remediation of the site was still in progress…

7.4. European Committee on Radiation Risk (ECRR)

ECRR is an *informal* committee formed in 1997 following a meeting by the European Green Party at the European Parliament to review the 96/29/EURATOM Directive [1].
However –
Relatively often EU experts and local environmental organisations in Africa and Asia present ECRR documents [24, 25] as the last and definitive word of European Community on the issue of radiation protection.
This approach results in severe anxiety of local population, as both the local appropriate authorities and the companies seeking to establish a mining/processing operation are seen as providing incorrect information – despite the fact that this information is based on the latest ICRP and IAEA publications.

8. ANOTHER POTENTIAL IMPLICATION – "OUTSOURCING" THE INDUSTRY

As limiting values, both for the concentrations of radionuclides and for the radiation exposure, seem to continue to decrease – more and more companies make an effort to relocate their operations out of developed countries. The costs are, of course, a significant factor – but a regulatory burden that industries face is as significant as costs, and sometimes it is even more important.

When some industry is faced with excessive regulations – of course the preferred option is to close the plant in the country where this happens and open it in another country, where regulatory regime is not that stringent. The problem is not with the regulations themselves, as the IAEA Basic Safety Standards are accepted worldwide, – it is with their application.

Unfortunately, in a developed country it is not uncommon to hear an argument based on "an interpretation of an appendix of a guideline for a procedure that describes a regulation relevant to a section of an Act".
It would be much more practical to set performance standards for the industry in the form of radiation exposure or 'release' limits and then leave it to the industry to develop systems to meet these standards in its specific circumstances. As it was correctly pointed out about 35 years ago –
When a regulatory agency gets into writing detailed and compulsory specifications on how to meet the performance standards, there is a danger that the system of radiation protection will degenerate into a continuing effort to comply with ever more complicated regulations, procedures and guidelines – completely losing sight of the basic goal of safe operation. [26].

The long-term consequences of 'outsourcing' the industry are currently not known, but one example can be provided:
For each and every wind turbine erected in the EU about 300-400 tonnes of radioactive ore needs to be mined, crushed, leached and processed to get enough material for one neodymium magnet. Generating in the process approximately 200-250 tonnes of radioactive waste. A typical mobile phone would need about 3 kilograms of radioactive ore for the vibrating magnet, speaker, screen… Which results in serious pollution of the environment and, possibly, in some health effects for the local population – but not in the EU…
It should be noted that the issue with rare earth supply appears to be understood in the EU and different options (both recycling and mining/processing) are being discussed [27].

9. CONCLUSIONS AND QUESTIONS FOR CONSIDERATION

9.1. Health

There are areas of high natural radiation background in Africa and Asia, where annual exposure of the public may reach 7-8 mSv/year. But –

- How many African and Asian children will not have malaria or other badly needed vaccinations because the health budget would be re-directed to keep everyone under 1 mSv/year?

9.2. Environment

Having a predominantly service economy, importing everything may not be as good as some people think… As the former Australian Liberal Party Leader J. Hewson said in 2015:
With an economy that is 68 per cent services, the entire country is basically sitting around serving each other cups of coffee… [28] Thus –

- What are the <u>world-wide</u> and, subsequently, EU local health and environmental implications of production of nice, clean and green products for the EU elsewhere, where the controls over radiation are either at a low level or may not exist at all?

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Natural radioactivity of building materials in Belgium: current situation and regulatory approach

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Abstract. As all other EU member states, Belgium had to implement the relevant requirements of the EU BSS regarding natural radioactivity of building material. The Belgian radiation protection authority (FANC) organized in 2016-2017 a measurement campaign of 70 building materials. These building materials were selected taking into account the indicative list of annex XIII of the EU BSS; for imported natural stones, the selection was guided by the portal monitor measurements performed by the Customs on all containers in Belgian harbours. Based on the results of this and previous campaigns performed by other institutions, no building material of concern could be identified. The implementation in Belgian regulations of building materials aspects of EU BSS will thus focus on regular surveys of the natural radioactivity in building materials used in Belgium rather than on systematic measurements of specific categories of building materials. For the control of imported building materials, it will be striven for an increased collaboration with the Customs in order to more closely follow portal monitor data on building material containers in the harbours.

KEYWORDS: NORM, building material, regulations.

1. INTRODUCTION

The 2013/59/euratom BSS directive [1] sets a reference level of 1 mSv/a for external exposure to the natural radioactivity of building materials. It asks member states to identify building materials of concern from radiation protection point of view taking into account an indicative list published in Annex XIII of the directive. For that purpose, the activity index I may be used as a conservative screening tool in the identification of the building materials which could induce an external exposure exceeding the reference level:

$$I = C_{Ra\text{-}226}/ 300 + C_{Th\text{-}232}/ 200 + C_{K\text{-}40}/ 3000 \qquad (1)$$

If this index is higher than 1, a specific dose-assessment must be performed in order to compare with the reference level. A stepwise method for calculating this external dose has been proposed by the European Centre for Normalization (CEN) in a recently published technical report [8].

Studies on the radioactivity of Belgian building materials have already been carried out in the 1980s [2]. More recently, the University of Hasselt made a survey of more than 120 building materials on the Belgian market; this survey included natural stones, tiles, cement, concrete, bricks and gypsum. No one of the samples exceeded the reference level of 1 mSv/a [3][4]. To complement these results, the Federal Agency for Nuclear Control (FANC), the Belgian radiation protection authority, performed another survey in 2016-2017.

2. NATURAL RADIOACTIVITY OF BUILDING MATERIAL PRODUCED OR IMPORTED IN BELGIUM: RESULTS OF FANC SURVEY

2.1 Overview of the campaign

The FANC survey included 73 samples which were analysed by gamma spectrometry in the Belgian laboratories of SCK-CEN and IRE-Elit. The samples were collected in collaboration with professional associations and companies from the building sector. They covered the categories listed in Table 1. Cement and natural stones were the categories for which the most samples were analysed. Cement samples included Portland CEM I cement (made essentially purely of clinker), composite cement CEM II and CEM V, blast furnace cement CEM III.

Imported natural stones were selected on basis of the portal monitor measurements performed by the customs in the Belgian harbours of Antwerp and Zeebrugge.

Table 1 shows an overview of the results of the laboratory measurements for the activity index I.

Table 1: activity index I results for the different categories of building material

Categories	# samples	I min	I max	I average
CEM I (Portland)	13	0.15	0.36	0.27
CEM II + V (composite)	11	0.31	0.61	0.45
CEM III (blast furnace)	14	0.41	0.69	0.55
aggregates	4	0.09	0.67	0.37
bricks	4	0.52	0.67	0.61
gypsum	6	0.05	0.46	0.24
Imported natural stones	21	0.59	1.47	1.08

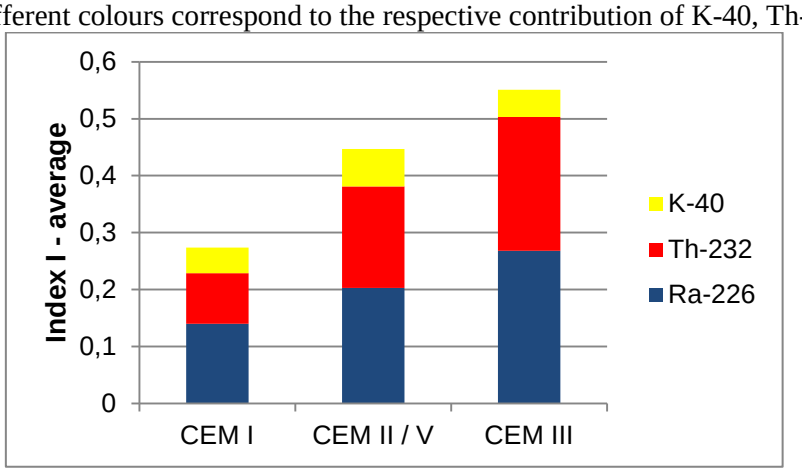
2.2 Results for cement samples

Annex XIII of the EU BSS lists materials containing fly ashes or blast furnace slag among the indicative list of building material potentially of concern. Fig. 1 shows the results for the average value of the activity index I for the three cement categories. It shows also the relative contribution of K-40, Th-232 and Ra-226. Cement incorporating fly ashes (CEM II and CEM V) or blast furnace slag (CEM III) have indeed a higher activity concentration in natural radioactive substances compared to CEM I cement but the activity index I stays lower than 1 for all cement categories.

2.3 Bricks, aggregates, gypsum

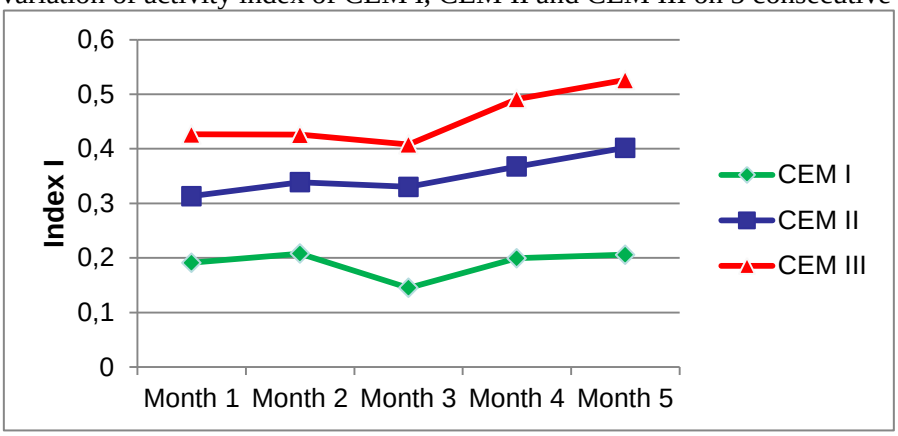
In Table 1, bricks have a higher activity index among the building materials used in bulk – essentially due to the contribution of K-40. The index I however does not exceed 0.67 - well below the reference level. The same is true for the aggregates, which included two samples of limestone and one of porphyry from Belgian quarries, as well as an expanded clay aggregate. Porphyry is cited in the indicative list of Annex XIII of the EU BSS but the Belgian sample has an activity index of 0.5, slightly higher than the limestones (I = 0.1 and 0.2) and slightly lower than the expanded clay (I = 0.67).

Fig. 1: average activity index for the three main categories of cement, CEM I, CEM II and V, and CEM III. The different colours correspond to the respective contribution of K-40, Th-232 and Ra-226



In one factory, one sample of each cement category CEM I, CEM II and CEM III has been analysed for 5 consecutive months in order to check the temporal variation of the activity concentration. Results are displayed in Fig. 2. The index of CEM I is essentially constant on the 5 months period but there is some more variability (25 – 30%) in the activity index of CEM II and CEM III cement.

Fig. 2: variation of activity index of CEM I, CEM II and CEM III on 5 consecutive months

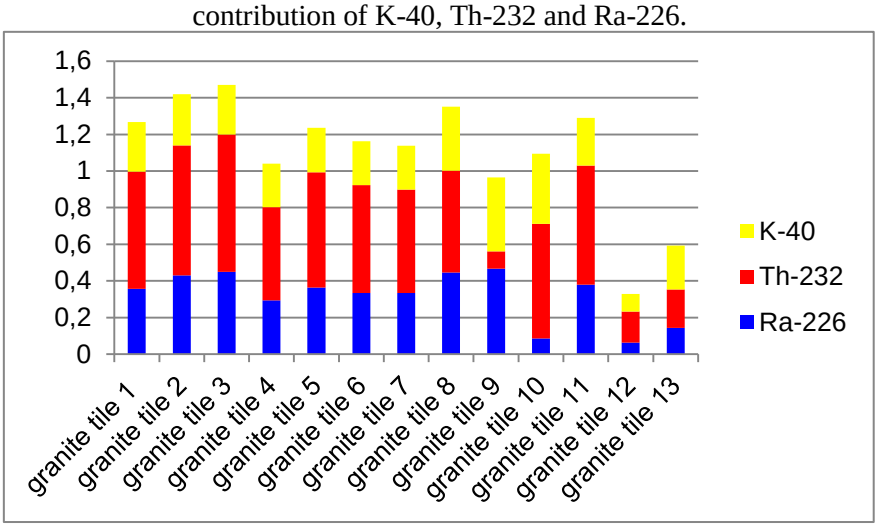


In Belgium, phosphogypsum from phosphate production is used as a building material. The activity concentration of this phosphogypsum is measured by the producer and is quite low due to the use of magmatic phosphate ore in the production process [5]. The results on the building material made of that phosphogypsum confirm these low values of radioactivity with an activity index which does not exceed 0.46.

2.4 Results for natural stones

The samples have been selected on basis of the level of gamma radiation measured on the portal monitor in the harbours of Antwerp and Zeebrugge [6]. Shipments of granite systematically presented the highest values on the portal monitor: 13 samples of granite tiles have been analysed. 2 slate tiles and one sandstone tile have also been selected. Fig. 3 shows the results for the activity index for the 13 granite samples. Although the majority of the samples have an activity index higher than 1, we will see in section 3 that the dose-impact stays largely below 1 mSv/a – these materials being devoted to superficial applications in buildings.

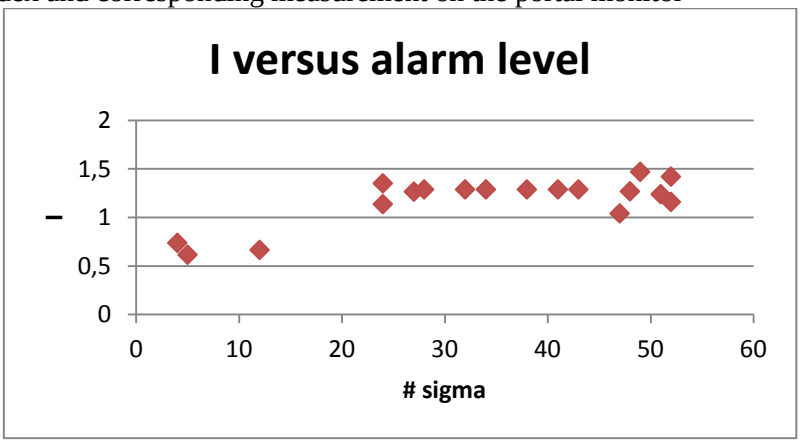
Fig.3: activity index of the 13 samples granite tiles. The different colours correspond to the respective contribution of K-40, Th-232 and Ra-226.



2.5 Correlation with portal monitor measurement

The alarm level on the portal monitor is expressed as a number of « sigma » (standard deviation of natural background). Fig. 4 shows the relation between the activity index of the material and the number of « sigma » on the portal monitor for the corresponding container. Materials with an activity index lower than 1 induce a lower signal on the portal monitor but the relation between the measurement on the portal monitor and the index I is not linear. For a same activity index (I ~ 1.3), the number of "sigma" may vary between 25 and 50. This is probably due to the numerous factors influencing the measure on the portal monitor: geometry of the shipment, density of the materials, speed of the truck, etc.

Fig.4: activity index and corresponding measurement on the portal monitor



2.6 "Inter-comparison" between laboratories

Some of the materials have also been analysed by the producer in another laboratory. Although it can not really be considered as an inter-comparison (the different laboratories analysed the same material but not necessarily the same sample), it still give an idea of the robustness of the results. The results are summarized in Table 2. The agreement between the measures performed by the different laboratories is satisfactory as the difference in the results didn't exceed 10%.

Table 2: activity index of some building materials as measured by three different laboratories

Sample	Laboratory 1	Laboratory 2	Laboratory 3
Sample 1	0.65	0.62	0.65
Sample 2	0.67	0.66	0.66
Sample 3	0.59	0.54	0.55
Sample 4	0.52	0.51	0.52
Sample 5	0.72	0.7	-
Sample 6	0.41	0.46	-
Sample 7	0.41	0.39	-

3. DOSE-ASSESSMENT

Section 2 showed that only imported granite tiles have an activity index higher than 1. These granite tiles are superficial material and the activity index is obviously far too conservative to assess their effective dose-impact. The European Committee of Normalization (CEN) developed a formula [8] which takes into account the density and thickness of the material to assess the dose. Using this formula, the dose induced by the granite tile with the highest activity index (I=1.47) can already be shown as being lower than 1. This formula still is quite conservative as it assumes that the room is fully covered by the material. More realistic calculations can be performed e.g. using the tables of the CEN technical report and calculating the contribution of each building product and each structure of the room to the external dose. Alternatively, a specific calculation code such as RESRAD-BUILD may be applied. These calculations have been detailed in [7]. One assumes the standard CEN room with four walls made of bricks, a ceiling and a floor made of concrete; the floor is covered with the

granite tiles. The activity concentration for each of the building material is taken from the present measurements. The resulting external dose is 0.35 mSv/a (resp. 0.27 mSv/a) using the CEN method (resp. RESRAD-BUILD code) with inclusion of the granite floor tiles. Without granite tiles, the external dose will be 0.315 (resp. 0.21) mSv/a. The incremental dose due to the use of granite floor tiles is thus only of 30 – 60 µSv/a depending on the calculation method. In any case, the external dose in a room made of building materials typical for the Belgian market is only a fraction of the reference level.

4. FUTURE REGULATIONS

The results of FANC survey and of previous studies didn't allow identifying any building material of concern in Belgium. Consequently, in its transposition of art. 75 of the EU BSS, Belgium didn't define any specific categories of building material and choose not to implement a general obligation of measurements for the producers or importers of building materials. However, FANC has defined the mechanisms needed to take action in case some building material of concern would nonetheless be identified. In particular, FANC developed mechanisms to insure a long-term follow-up of the issue of natural radioactivity in building material: it has integrated the radioactivity of building material in its radiological surveillance program. FANC intends to analyse around 40 samples of building materials each year in order to follow the evolution of natural radioactivity in building materials and of the corresponding public exposure. For the selection of these samples, FANC will collaborate with the ministry of Economy which is in charge of the control on the application of the Construction Products Regulations (CPR). FANC will also continue to work together with the customs which are in charge of the portal monitors installed in Belgian harbours. Anomalies on the radiation level of incoming shipments will be controlled.

5. CONCLUSIONS

Literature data and survey carried out by FANC didn't allow identifying any building material of concern in Belgium. Based on a standard room model, external exposure due to the investigated building material may be estimated to approximately 0.3 mSv/a – well below the reference level of 1 mSv/a. Consequently, FANC will not impose a general obligation of measuring radioactivity for given categories of building materials. The follow-up of the exposure from building materials will rather be integrated in the national radiological surveillance program of FANC. These and additional controls and surveys will be undertaken in collaboration with other authorities, such as customs, which monitors the radioactivity of incoming sea shipments, and the ministry of Economy, which controls the application of the Construction Product Regulations.

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Assessing the public exposure related to the use of NORM in new types of building materials

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Abstract. For a safe reuse of Naturally Occurring Radioactive Materials (NORMs) in construction, it is of great importance to evaluate the radiological aspects of the reuse in addition to chemical, environmental, economic… aspects before the construction materials are introduced on the market. This is of particular importance for new types of construction materials, such as alkali activated materials, that allow the reuse of a large faction (wt%.) of residues. The Euratom BSS (basic safety standards) sets the requirement of the radiological evaluation of building materials that incorporate specific residues from NORM related industries. In the period 2014-2017, the COST Action Tu1301 NORM4Building initiated a lot of research on the radiological evaluation of new types of construction material that are currently in the research state. In the course of the NORM4Building project a radiological database on NORM & building materials was developed. In addition, new dosimetrical tools were developed for a more realistic evaluate of the gamma dose related to the reuse of NORM in construction. These dosimetrical tools provide a more realistic radiological screening of the reuse of building materials in addition to the Activity Concentration Index (ACI) that is proposed by the EU-BSS as screening tool. In the current paper and linked presentation, the contents of the NORM4Building database will be presented next to the newly developed dosimetrical tools for the evaluation of the public exposure to gamma radiation from different types of building materials. The NORM4Building database is available via www.norm4building.org.

KEYWORDS Natural occurring radioactive materials, building materials, database, concrete, by-products, Euratom Basic Safety Standards

1. INTRODUCTION

Turning waste into resources is a key step on the Roadmap to a Resource Efficient Europe [1]. The recycled materials can however contain a measurable amount of natural occurring radionuclides such as ^{238}U, ^{232}Th and their decay products and ^{40}K and this aspect needs to be considered, particularly when the residues are included in building materials. Several industries that need to consider the precences of naturally occurring radioactive materials (NORMs) are listed in Annex VI of Council Directive 2013/59/Euratom [2]. An enhanced content of natural occuring radionuclides can be an issue for by-products such as fly ash from coal, peat and heavy oil fired power plants, phosphogypsum from phosphate industry, phosphorous slag from thermal phosphorus production, copper and tin slags from primary and secondary production, red mud from aluminium production and some residues from steel

production. For the use of these by-products in building materials, the Council Directive 2013/59/Euratom (Euratom Basic Safety Standards; EU-BSS) [2] sets the requirement of the radiological evaluation of the produced building materials. In the EU-BSS, a screening parameter, the activity concentration index (ACI), is defined for the initial screening of the building materials incorporating NORM residues however the real criterion that determines if the use of the considered residues in building materials is acceptable or not is the reference level of 1 mSv/year.

In the concrete industries, the considered residues are used in increased amounts as supplementary cementitious materials (SCMs) (as partial cement replacement or as mineral additions in concrete) and as aggregates [3]. In the ceramic industries metal smelting slags can be used as aggregates in clay-based ceramics [4]. In the bond system of clay ceramics residues, such as red mud, can be used [4]. Alternatives for cement and concrete using Alkali-Activated Materials (AAMs) are being developed. AAMs contain calcium silicate or a more aluminosilicate-rich precursor such as a fly or bottam ash, metallurgical slag or natural pozzolan, as solid aluminosilicate source [5].

In the period 2014-2017, the COST Action Tu1301 NORM4Building initiated a lot of research on the radiological evaluation of new types of construction material that are currently in the research state. In the course of the NORM4Building project a radiological database on NORM & building materials was developed. In addition, new dosimetrical tools were developed for the evaluation of the gamma dose related to the reuse of NORM in construction. These dosimetrical tools provide a more realistic radiological screening of the reuse of building materials in addition to the ACI that is proposed by the EU-BSS as screening tool. In the current paper (and linked presentation) the contents of the NORM4Building database will be presented with the scenarios used for the simulation of building materials incorporating NORM residues. In addition, newly developed more realistic dosimetrical tools for the evaluation of the public exposure to gamma radiation from different types of building materials are discussed.

2. SCENARIOS FOR INCORPORATION OF A BY-PRODUCT IN CONCRETE

Table 1 lists the compositions which were used to model the use of by-products in specific types of concrete.

Table 1 Description of concrete compositions used in the model compositions

Scenario ID	Construction Material	Composition (kg/m^3)			
		Cement	By-product	Aggregates	Water
1	Reference concrete	400		1850	150
2	High volume fly ash (HVFA) concrete	160	220 (fly ash (FA))	1700	140
3	Concrete with FA as partial replacement of cement and sand’	320	130 (FA)	1750	150
4	Concrete with FA as partial replacement of sand	360	90 (FA)	1800	150
5	Concrete with slag as partial replacement of cement and aggregates’	80	720 (slag)	1850	150
6	Concrete with slag as partial replacement of cement	80	320 (slag)	1850	150
7	Concrete with slag as partial replacement of aggregates’	400	400 (slag)	1450	150
8	Alkali activated concrete containing red mud as partial replacement of cement and aggregates		1800 (red mud)	450	150

The activity concentration index (equation 1) [6] was calculated for several types of concrete using the compositions listed in table 1.

$$I - index = \frac{Ac_{226Ra}}{300\ Bq/kg} + \frac{Ac_{232Th}}{200\ Bq/kg} + \frac{Ac_{40K}}{3000\ Bq/kg} \quad (1)$$

With Ac the activity concentration of the mentioned radionuclide expressed in Bq/kg. The average values of 0.38 and 0.45 were used in the calculations as I-indexes for respectively cement and soil/aggregates [7]. The results of the I-index calculation using the concentrations listed in Table 1 are described in detail in [8].

3. DATABASES TO ASSESS THE USE OF NORM IN CONSTRUCTION

During the course of the COST Action NORM4Building several stategies to data collection and the verification of the collected were explored and efforts were initiated to merge databases that contain data on NORM and the use of NORM in construction materials.

A lot of data on natural radioactivity in European building materials was collected by Trevisi et al. [7]. Recently, this dataset that mainly contained data on ^{226}Ra, ^{232}Th and ^{40}K activity concentrations of building materials in Europe was further enlarged and expanded with radon emanation and exhalation rates [9]. The data collection for the construction of this database and the verification of the collected results involved an extremely labour intensive process.

A new approach to data collection was developed in order to semi-automatically collect data from scientific publications. This appoach that relies on automated data mining via natural language processing and text link analysis is further described in [8]. This approach has important advantages: (1) hundreds of publications can be processed automatically at a monthly basis; (2) the approach allows a continuous (automated) search for newly published literature and is therefore very useful for keeping an inventory up to date; (3) the data search can be expanded or modified using different key-words which allows the construction of a more detailed and expanded database. In its current form, the approach has several limitations: (1) data from graphical images (eg.: histograms) is not collected; (2) the licence for datamining software is very expensive; (3) the reliability of the collected data is strongly dependent of the reliability of the included publications, an aspect that cannot be assessed by the automated datamining program, and therefore the validation of the results requires a labour intensive verification step; (4) the included publications enclose both papers that reported averaged results and papers that reported individual measurements in more detail. A drawback of this fact is that only a limited amount of statistical analysis can be applied on the collected data. On the basis of this approach the NORM4Building database was constructed and this database is available via www.norm4building.org. A large set of the data that is collected in this database is described in detail in [8] and [10].

To allow a more in-depth statistical analysis, a database that is purely based on individual measurement entries and not on average results reported in literature was constructed by Sas et al. [11] ('By-BM database'). This database allows many interesting analysis and visualisation options. A future aim is to step by step investigate the data reported in the previously mentioned databases and to track the underlying 'individual measurement results' (if they can be found) and to incorporated these, after a verification step, in the By-BM database. The combination of the described approaches and databases can provide important added value especially if automated data collection can be combined with more in-depth statistical analysis options. The By-BM database is accessible online via http://bybmproject.com/ and the data included was discussed in [11].

4. EXPANDED SET OF SCREENING TOOLS FOR GAMMA DOSE ASSESSMENT

For the assessment of gamma ray exposure from building materials several methods have been developed ranging from simple indices to more sophisticated Monte Carlo simulations [2][12][13][14]. In the dose assessment calculations based on gamma ray attenuation and build-up factors the density and wall thickness were identified as very critical parameters [15][16][17]. The approach implemented by the EU-BSS uses an Activity Concentration Index (ACI) [2] that does not include the density and wall thickness as modifiable parameters. Technical guide Radiation Protection (RP)-112 [6] describes the index, originally developed by Markkanen [18], in more detail. The index described in RP-112 assumes a standard room with dimensions 400 cm x 500 cm x 280 cm, uses the density of concrete (2350 kg/m^3) and assumes a thickness of 20 cm for walls, floors and ceilings. A screening method that takes into consideration density an thickness via a density and thickness corrected index I(ρd) was proposed by Nuccetelli et al. [19] Complementary to the methodology proposed by the EU-BSS, the technical report CEN/TR 17113:2017, potentially a precursor for the development of a harmonized European Standard, also included a more elaborate index that allows modifying the density and the thickness [20].

A new study described by Croymans et al. [21] provided a dose calculation assessment using the original dose calculation of Markkanen with an expanded set of gamma lines and a higher total gamma intensity. The developed model by Croymans et al. [21] that uses an expanded set of gamma lines is complementary to the existing ACI model, proposed in the EU-BSS and the density and thickness corrected assessments proposed by Nuccetelli et al. [19] and CEN/TR 17113:2017 [20]. An initial screening can be based on the ACI proposed by the EU-BSS, especially useful in case that the building materials are thinner than 20 cm or lighter than 2350 kg/m^3. For building materials thicker than 20 cm or heavier than 2350 kg/m^3 it is advisable to use a density corrected assessment tool, especially useful for standard room sizes. The expanded gamma dose assessment method, allowing the assessment of non-standard rooms, can be used in specific cases. This model also allows considering the presence of doors and windows in the considered model.

5. CONCLUSSION & OUTLOOK

A semi-automated database for screening, identifying materials of concern from a radiological perspective was set-up by the COST-Action NORM4Building. More realistic scenarios are proposed for assessing the impact of the use of NORM in building materials. Complementary tools for the evaluation of the gamma dose related to the use of NORM in building materials were developed.

The NORM4Building network joint forces with the EAN-NORM & EU-NORM networks to form the European NORM Association (ENA). Future aims involve the further integration of the developed databases and implementing the developed tools for gamma dose assessment in the online database.

6. ACKNOWLEDGEMENT

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Remediation of a basin for mine drainage water in Germany using an artificial cover

Concept development of an ecologically reasonable remediation strategy and compliance with long-term safety

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Abstract. This paper provides a case study in terms of a concept development for the remediation of a basin for mine drainage water in Germany using an artificial cover. The basin covers an area of about 3,000 m². The sediments show increased levels of naturally occurring radionuclides, e. g. specific activities of > 10 Bq/g for the main nuclide Ra-226 locally. Therefore, the sediments must be handled adequately taking into account the German Radiation Protection Ordinance including a good radiological practice. Extensive radiological and (geo-) technical investigations in terms of measurements and sampling have been performed. Based on the results the requirements for the cover with respect to radiological safety, (geo-) technical issues and long-term safety are defined and a specific solution for the artificial cover is prepared on a conceptual basis. The concept has been presented to the respective regulatory authorities and is accepted to be carried out.

KEYWORDS: NORM, remediation, mine drainage water, radiological measurements and sampling, effective dose assessment, long-term security

1. INTRODUCTION

In the course of the remediation of a former mining site in Germany, it is also planned to remediate a drainage water basin (DWB). The DWB served as a sedimentation and retention basin for the mine water pumped during operation of the mine before the water was discharged into the receiving channel. Mining waters at this site show increased levels of naturally occurring radioactive materials (NORM) from the decay series of uranium and thorium. In the area of dwellings, mine water pipes, receiving channels and mine water basins, this can lead to precipitation and incrustations. Preliminary radiological investigations in the area of the basin show increased values of dose rates as a result of sediments and incrustations with increased levels of natural radionuclides, in particular for Ra-226.
The DWB covers an area of approx. 3,000 m² and is bordered by an encircling rampart with a walk-on dam crown (approx. width: 3.5 m). The interior of the basin is accessible via two ramps in the northwestern and southwestern part. The bottom elevation of the basin is about 25.8 m above sea level (a.s.l.), the crown of the wall at about 28.8 m a.s.l.. The surrounding area is located at about 25.9 m a.s.l. in the south, about 28.2 m a.s.l. in the west and 27.7 m a.s.l. in the north of the basin. The basin is divided by a metallic sheet pile wall. The sheet pile wall penetrates into the subsoil down to a depth of 19.8 m a.s.l., with a concrete structure with a thickness of approx. 0.7 m at the bottom of the DWB (from 25.1 m a.s.l. to 25.8 m a.s.l.). The bottom of the pit basin is completely covered with bentonite membranes, which were nailed to the foundations of the sheet piling at a depth of 25.3 m a.s.l. The coverage of the bentonite layers with other material at the bottom of the basin is about 0.5 m.
Due to radiological (namely the difficulties with respect to landfilling of NORM residues as a consequence of the loss of acceptance of radiological residues in Germany) and chemical reasons (namely the presence of other chemical contaminants at the site) the client asked Brenk Systemplanung GmbH (BS) to work out in detail the variant "safe encapsulation in-situ with suitable cover" (SESC).

2. REGULATORY FRAMEWORK AND PRACTICE

The legal regulations for the disposal and recycling of residues from industry and mining with increased specific activities of natural radionuclides are regulated in §§ 97 to 102 and Appendix XII of

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the German Radiation Protection Ordinance (RPO). "Residues" are defined as materials which are produced in the industrial and mining processes as specified in Appendix XII Part A (recycling and disposal of residues requiring surveillance), and which fulfill the conditions specified therein. The sediments, precipitations and incrustations in the area of the DWB do not correspond to any of the "list of residues to be considered" from that Appendix XII Part A of RPO. Therefore, §§ 97 to 101 RPO are not applicable to the DWB in the first instance.
Since improper disposal of residues in the area of the DWB due to the increased levels of specific activity can cause increased radiation exposure of the population, the radiation protection principles according to §§ 4 to 6 RPO and good radiation protection practice must be performed. Pursuant to § 102 RPO (surveillance of other materials), the competent authority may order protective measures and compliance with good radiation protection standards, in particular by analogy with the provisions of §§ 7 to 101 RPO. In the present case of NORM from mine drainage activities, this enables the competent authority to require analogous measures compared to those for residues listed in Annex XII Part A of the German RPO.
BS therefore recommends a good radiation protection practice in analogy to § 101 RPO (removal of radioactive contamination from real estates). According to § 101 RPO, the reference value applied to an unrestricted use of real estate is that the radiation exposure of members of the public through the residues not removed shall not exceed an effective dose of 1 mSv per calendar year. On a case-by-case basis the competent authority may grant full or partial exemption from the duty to remove the residuals when circumstances are present or protective measures are applied that prevent a radiation exposure of more than 1 mSv effective dose per calendar year for members of the public even without removal of the contamination. Accordingly, the implementation of the preferred option SESC is in general possible. Whether this option can also be reliably implemented under aspects of radiation protection technology and under geo-technical or structural-technical aspects was the subject of an investigation conducted by BS and which will be discussed in the following sections.

3. RADIOLOGICAL INVESTIGATIONS

The objective of the radiological investigations includes

- the initial radiological characterization of the metallic sheet pile wall on which NORM incrustations are present by sampling,
- the radiological initial characterization of the basin sediments by sampling as well as
- the concentration of the radiological measurements in the bank area of the DWB by sampling and gamma dose rate measurements.

Due to its comparable chemical behavior, the radium contained in the mine water is incorporated into the crystal lattice of barium sulfate (BaSO4) instead of barium. Ra-226 and its daughter nuclides is the main radionuclide responsible for radiological exposure.

3.1 Sampling

3.1.1 Metallic sheet pile wall

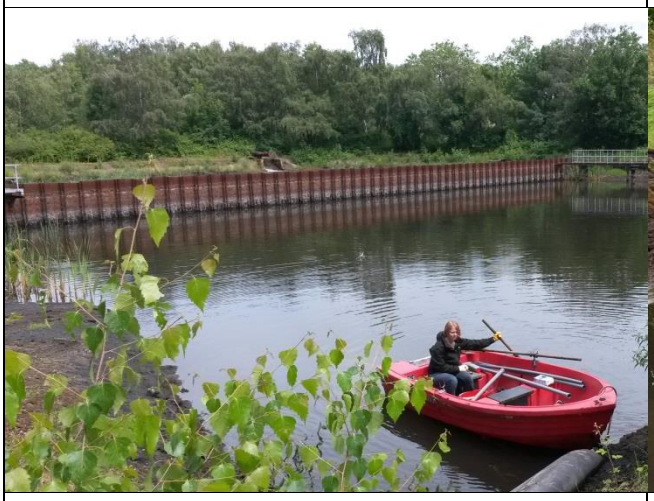
The measurements on the sheet pile wall made of steel should provide information on whether it can remain at place in the course of the realization of the remediation from a radiological point of view or whether this is to be included in the remediation concept for radiological reasons. Its length is about 50 m, its height is about 3.0 m above the foundation base and its thickness is about 1 cm. Furthermore, there are metallic structures such as the overflow or inlets in the western and northern part of the basin. Due to the operation practice, these are treated equivalent to the sheet piling from the radiological point of view. The total mass of the metallic components is about 60 Mg. On the sheet pile wall incrustations were formed in the course of time. They consist mainly of barium sulfate (BaSO4) with a fraction of iron hydroxides, which cause the reddish-brown color, see Fig. 1. The sampling of the sheet pile wall has been performed from the water side at the southern part of the DWB after measurement of the gamma dose rate with the gamma dose rate meter FH 40G-L10 with probe FHZ 672 E-10 (containing 750 cm^3 of organic scintillation material and a NaI crystal) from Thermo Scientific Inc. At four representative positions a sample above the water surface has been taken from the incrustations.

The recovered material has a thickness of up to 1 cm. The locations of the samples are shown in Fig. 2.

Figure 1: On-site impressions from the measurements and the sampling

DWB with metallic sheet pile wall (winter season)

Gamma dose rate measurement at shore



Inspection by rowing boat

Sediment sampling

Sampling of metallic incrustations

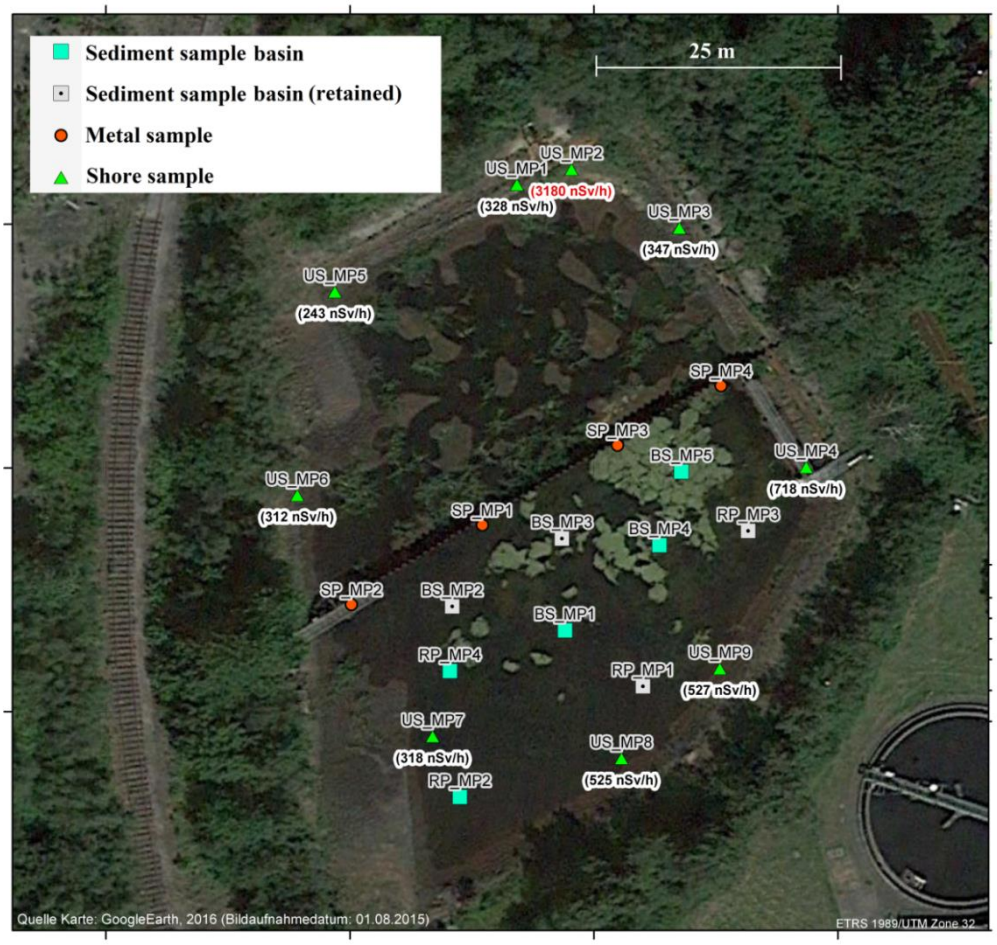
Gamma dose rate measurement at sheet pile wall

3.1.2 Basin sediments

The sampling of the basin sediments served to determine the specific activities of the relevant radionuclides. Furthermore, the consistence of the sediment has been checked. The sampling has been

conducted by means of ram pipes in the southern part of the DWB. Here, sediments were taken at five representative sites with one sample each. In addition, four reference samples were taken, but these were not evaluated by gamma spectrometry. The thickness of the sediments deposited in the DWB was determined in the course of sampling.

Figure 2: Radiological measurements and points of sampling in the basin area



3.1.3 Shore sediments

The sampling of the shore sediments served primarily to determine the specific activities of the relevant radionuclides. As with the basin sediments, the consistence of the sediments depending on the water content was also checked. In addition, measurement results of the gamma dose rate with the gamma dose rate meter FH 40G-L10 with probe FHZ 672 E-10, were taken approximately 20 cm above the sediment in order to complete the data.
Shore sediments were collected at nine representative sites with one sample each. Two of these sampling sites were suitable for taking a depth profile consisting of two samples, since there was a sediment thickness of approximately 80 cm (at US_MP2) and 60 cm (at US_MP3).

3.1.4 Basin water

Until recent times the water of the DWB was refilled within regular intervals, so that the sediments were always covered with water. Accordingly, a mixing due to the raising of sediments took place. In order to estimate the radionuclide content of this squirl mixing, several water samples with a high fraction of suspended matter from the basin and bank were further investigated. This serves to estimate whether the basin water may be drained off directly via the receiving channel in the course of the remediation option SESC. After a settling time of about two days, the sediment content of selected

water samples was separated in the laboratory. The remaining water with a small amount of suspended matter was filled as a partial sample and its specific activity was measured. Furthermore, the filtrate of selected samples of this aqueous fraction was examined.

3.1.5 Sediment elution behavior

In order to investigate a possible radiological influence on the groundwater by radium discharge from remaining sediments, the elution behavior of radium was investigated in more detail. Investigations concerning the leaching behavior according to DIN EN 12457-4 (S4 "shaking" elution test) were carried out on three representative samples.

3.2 Results

The laboratory measurements have been carried out by a subcontractor on behalf of BS. The radiological examinations (direct dose rate measurements, laboratory measurements) show the following results, see Tab. 1:
In comparison to the natural background of approx. 0.07 µSv/h, the local gamma dose rate is increased up to a range of 0.2 µSv/h and 0.8 µSv/h. All gamma dose rate measurements have been taken at the distance of 1 m to the residues. At the inflow of the DWB (US_MP2 in Fig. 2), a bank area of about 10 m shows an ODL in the range of 0.8 µSv/h to 7 µSv/h.
The key nuclide is Ra-226. Typical specific activities are 2 - 4 Bq/g for the sediment in the DWB and on the shore of the DWB. In the area of the inflow local specific activities for Ra-226 of more than 30 Bq/g were determined. The activities of the sediments are above the surveillance limit of 1 Bq/g according to Annex XII Part B RPO. At the inflow of the DWB in the north, the precipitation and sedimentation are particularly pronounced. The layer thickness of these sediments amounts to 80 cm (area about 20 m²), while in the rest of DWB sediment the thickness is between 10 and 40 cm. The average thickness of the sediment layer can is approximately 30 cm averaged over the entire area of the DWB.

Table 1: Main results from the radiological investigations

Parameter	**Values**
Gamma dose rate	< 1 µSv/h, at inflow 3-6 µSv/h
Radionuclides	mainly Ra-226, 10% Ra-228; [Pb-210]/[Ra-226] ~ 0.2 - 0.4
Specific activity of Ra-226 in sediments and incrustations	typically 1-10 Bq/g, at inflow ~35 Bq/g; sediment thickness 0.15 m - 1.0 m (~0.3 m)
Specific activity of Ra-226 in water with suspended matter	< 50 Bq/l, typically ~10 Bq/l
Specific activity of Ra-226 from elution tests	~ 5 mBq/l

The mass of the radiological residues in terms of the sediments can be estimated to 3,000 m² × 0.3 m × 1.2 Mg/m³ = 1,080 Mg. With a conservative value of 4 Bq/g for the specific activity of the main nuclide Ra-226, the total activity is calculated to 4,320 MBq. Both total activity and mass are considered to be low compared to the maximum storage levels of a special waste disposal site suitable for disposal of NORM residues.
The metallic sheet pile wall shows incrustations with increased radionuclide contents. For the main nuclide Ra-226, activities of approx. 1 Bq/g were measured, the sum of the maximum activities of the thorium and the uranium decay series is on average above the surveillance limit of 1 Bq/g according to Annex XII Part B RPO for residues under surveillance. Therefore, the metallic structures should be part of the radiological remediation concept.
Based on investigations of the aqueous portions of the sediment samples and their filtrate, it can be assumed that the radium activity is mainly located in the sediments. About 1 % of the mass-specific sediment activity is mainly accumulated in the aqueous fraction and 1 per thousand of the mass-specific sediment activity is accumulated in the filtrate of the aqueous fraction. Based on the sampling

procedure and the laboratory performance, the results here are to be understood as guideline values. In reality, due to the much longer periods of sedimentation or filtration in the soil on the one hand and avoidance of a DWB refilling, which would result in a re-suspension of sediments, on the other hand, in the unlikely event of a low outlet of pool water, the activity level in the suspended phase or in the filtrate should be significantly lower again. The results also show that a discharge of the mine water (e.g. using a submersible pump) into the flood waters from a radiological point of view is possible without restriction in case the raising of sediments during the implementation is avoided.
During the investigation the elution behavior of Ra-226 was thoroughly investigated. The measured value of 5.4 mBq/l on average for the eluate or the value of 73 mBq/l calculated from the solubility of BaSO4 and the maximum specific activity at the inflow of the DWB via the solubility are well below the reference activity concentration of 500 mBq/l according to the German drinking water regulation. The result shows that, due to the significantly lower specific activities of the eluate compared to the aqueous phase, an increased discharge of radium is highly unlikely and could only be related to colloidal transport mechanisms. Furthermore, discharges of radium through any seepage water and transport via suspended matter are also very low. Relevant exposure to drinking water in the vicinity of the DWB can thus be ruled out since the activity of the relevant nuclide Ra-226 is fixed in the sediments.

4. GEOTECHNICAL INVESTIGATIONS

The geotechnical investigations serve to clarify, whether the area of the DWB for the variant SESC from geotechnical and technical view is suitable. The investigations were carried out by a subcontractor on behalf of BS. The geotechnical on-site investigations included

- subsoil investigations by means of driving core sampling according to DIN EN ISO 22475-1 and heavy dynamic probing as well as
- geotechnical laboratory investigations (particle size analyzes, determination of water content as well as state limits, organic content and ignition loss).

The key results of the geotechnical investigations are summarized below:

- The subsoil conditions of the DWB are to be regarded as typical of the local environment with respect to the geological structure. Anthropogenic soil remodeling in the course of mining in the 20th century led to a raise of the terrain by up to 4 m. The underlying former floodplain deposits show a thickness up to 2 m.
- The basin floor of the DWB is located at 25.8 m a.s.l. The bottom of the receiving water is between 23.0 and 23.4 m a.s.l. Due to the bentonite seal of the DWB on the one hand and the concrete cover of the bottom of the receiving water on the other hand, both are not hydraulically linked. The distance from the bottom of the basin to the quaternary aquifer is approx. 6.5 m. Due to the attachment of the brookhole, the quaternary aquifer is not hydraulically linked to the receiving channel.
- The geotechnical laboratory investigations show different results of consistence and water contents of the subsoil. Soft consistence and high water contents in the peaty layers can cause irregular settling behavior. In peaty subsoil total organic carbon (TOC) values of 10 to 30% have been determined. The sediment of the DWB also has high levels of organic matter, so that large amounts of water can be bound and thus an irregular settlement behavior can result.
- Distinctive organoleptic features in the form of coke-typical odor were found in some samples.

5. RADIOLOGICAL REQUIREMENTS FOR THE VARIANT SESC

The radiological studies show that leaving the residues in the DWB without further measures (especially without cover) can result in increased exposures for the general public, since the activities are above the value of 1 Bq/g (surveillance limit due to Annex XII Part B RPO for the respective residues due to Annex XII Part A RPO). Increased exposure could result from external radiation, inhalation and ingestion. For radiation protection reasons, it must be ensured that the guideline value of the effective dose for the general public of 1 mSv per year is not exceeded. The option SESC ensures that a value well below the guideline value is achieved with regard to the expected effective dose for persons of the general population. This is justified as follows:

- The external irradiation is attenuated by the thickness of the cover of one meter by at least a factor of 1,000. The remaining exposure after appropriate coverage with the thickness of at least one meter is therefore negligible.
- Inhalation of aerosol-borne radionuclides of the sediment is not possible after adequate coverage. The inhalation of radon and/or radon-derived products is negligible because of the experience of low concentrations outdoors.
- Ingestion of radionuclides of the sediment is excluded after appropriate coverage.
- The option SESC is also viable with regard to long-term safety issues if the requirements described in the section below are met.

6. REQUIREMENTS OF AFTERCARE AND LONG-TERM SAFETY

After the successful implementation of the remediation option SESC, it must be ensured that the effective dose for the general public is below 1 mSv/a. This is ensured if a subsequent proliferation of the sediments and their use as subsoil or garden/arable grounds are effectively prevented. After proper implementation of the measure, no local and temporal restrictions regarding the abidance on the remediated DWB are necessary. These points are ensured by the combination of technical and organizational measures:

- The dismantling of the metallic sheet pile wall and the subsequent appropriate disposal has to be performed. This serves the purpose of minimizing the risk of disclosure of the deposits on the sheet pile wall and the sediments by improper removal of the sheet piling at a later date, by facilitating the maintenance of the area (e. g. in the form of mowing).
- The appropriate cover of the sediments with a thickness of at least one meter and concomitant solidification incl. appropriate greening of the surface minimizes the risk of gradual disclosure of the sediment due to wind and weather. The introduction of a sealing layer (e. g. bentonite) as an additional barrier reduces the (in any case very small) discharge of contaminants into the groundwater and the likelihood of subsequent exposure of the sediment as a result of human activities.
- The implementation of the security measure is to be accompanied and documented in a qualified manner. This ensures the correct execution.
- The area is to be maintained regularly. It must be prevented that e. g. as a result of wild growth or improper reuse an undefined state arises in which the gradual disclosure of the sediments can no longer be ruled out.
- Proper documentation of the fate of the (NORM) residues must be provided (e. g in the form of an appropriate entry of the residues in the register for contaminated sites, etc.) in order to exclude the use as a subsoil, garden or agricultural area after completion of the inspection. As a follow-up use of the area of the DWB, a green area for recreation is planned, against which there are no restrictions from the point of view of radiation protection.

7. DETAILLED CONCEPT OF IN-SITU COVERAGE AND GEOTECHNICAL ASPECTS

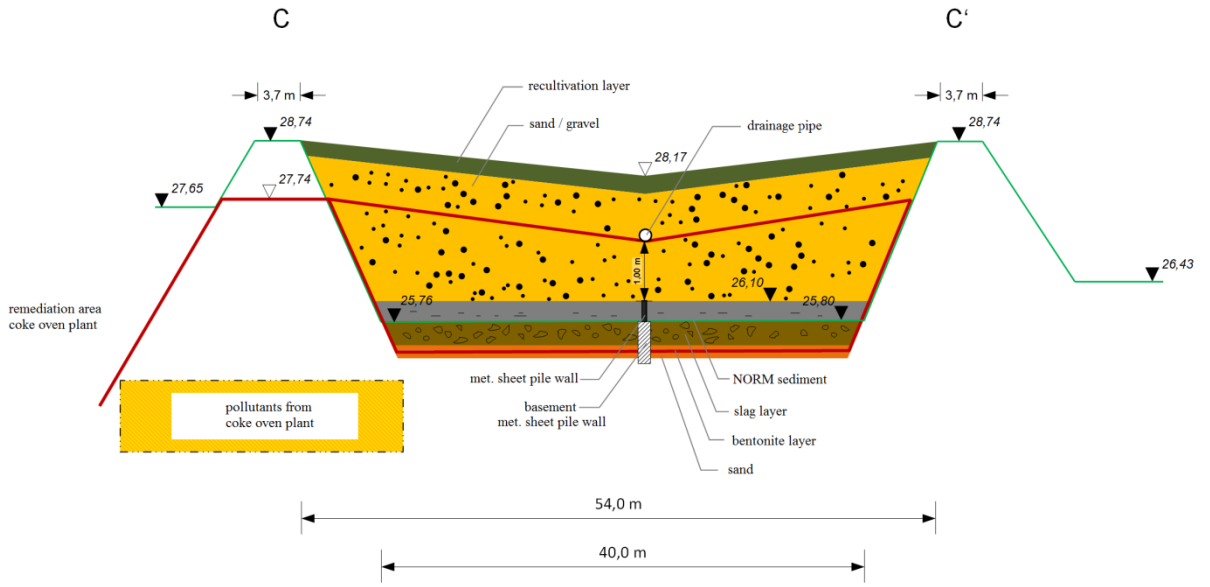
In the following, the preferred option SESC is explained and geotechnical aspects are considered in more detail, see also Fig. 3:

- The current form of the DWB is essentially retained. The contouring of the northern, western and southern area remains in the form of the current dam crown. The current design of the existing bentonite layer is essentially retained.
- In the eastern area next to the receiving water, the dam crown is flattened towards the middle of the dam. This measure serves for suitable drainage of the surface (see below).
- After suitable pumping of the water from the DWB, a compaction layer of coarse-grained material (sand/gravel mixture) with a thickness of approx. 0.3 m is applied to the sediment. This measure is used to produce the drainage or stability for further work. Depending on the consistence of the sediment this application can be supported by additional measures prior to the implementation of the compaction layer, e. g. by suitable application of geogrids or geotextile. A

desirable side effect of this initial coverage is the fact that external irradiation of workers for further action is mitigated and inhalation and unintentional ingestion are minimized.

- Subsequently, the metal sheet pile is separated on the level of the sediments.
- The sediments are then covered with a first full layer of again coarse-grained material (sand/gravel mixture) with a total thickness of at least one meter (including the first compaction layer). The design of the contouring is shown in Fig. 3. This layer meets the radiological necessity for a shielding of external radiation and to exclude inhalation or ingestion of contaminated material. The layer serves as a drainage or compaction layer from a geotechnical point of view.
- After the first full layer, the entire surface of the DWB is sealed using a bentonite layer, which are suitable for installation. This minimizes the entry of rainwater. The bentonite layers serve as an encapsulation of the sediments and which must not be re-opened after the remediation. This encapsulation is a desirable side effect for preventing (unintentional) exposure of the sediment at later times. With a suitable gradient of at least 1 %, rainfall water can be diverted in the overlying layer of sand/gravel towards the receiving channel. The already existing bentonite seal at the edge of the DWB is thereby connected to the newly applied bentonite mats, so that the sediments are included.
- On top on the bentonite layer, a drainage pipe with a drain to the receiving channel in the east of the DWB is installed.
- After installation of the drainage, the bentonite layer is covered with a layer of sand/gravel with a thickness of 0.7 m. This layer serves to protect the bentonite mats and fulfills the purpose of drainage or hinders the rooting of the bentonite mats, in conjunction with suitable shallow-rooted growth (e. g. grass).
- Contouring is completed by covering with a reclamation layer of thickness of about 30 cm. This layer allows greening with shallow rooting (e.g. grass) and thus serves as an erosion protection.

Figure 3: Concept of in-situ coverage (section C-C' from north to south)



Corresponding recommendations for the suitable implementation of the option and for occupational safety during the implementation of the measures were also developed.

7. CONCLUSIONS

Extensive radiological and (geo-)technical investigations in terms of measurements and sampling have been performed. From the respective results the requirements for the cover with respect to radiological safety, (geo-)technical issues and long-term safety were defined and a specific solution for the artificial cover is prepared conceptually. The concept has been presented to the respective regulatory authorities and is accepted to be carried out.

Discharges from NORM industries in Germany: estimate of doses to members of the public

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Abstract. Council Directive 2013/59/Euratom (the European Basic Safety Standards, or BSS) has removed the previous distinction between practices and work activities. Henceforth the dose limits for public exposure shall apply for the sum of annual exposure for members of the public resulting from all authorized practices. From this it follows that exposure levels being significantly smaller than the dose limits have to be considered, where appropriate, and the NORM industries, in particular the discharges of radionuclides from this industries into the environment have to be re-assessed. In transposing the BSS into national law, Art. 23 requires Member States, *inter alia*, to identify classes or types of practice involving NORM that may need to be regulated because they lead to exposures of the public that cannot be disregarded from a radiation protection point of view. Discharges are part of a practice and, therefore, may be subject to regulatory control. A comprehensive investigation of discharges in Germany included the 16 sectors of NORM industries listed in Annex VI of the BSS, and additional sectors that have been known in Germany for potentially being of radiological concern. This paper provides an overview of the approach, methods and results of the research project. A conservative screening tool was developed and applied to select NORM industries that may safely be excluded from a more detailed analysis. NORM industries that passed the screening stage were then analysed in detail with respect to the source term of radioactive discharges. The dispersion of radioactivity in the environment was described using advanced models such as air quality code ARTM, and all relevant pathways were considered for dose estimates. It is important to note that the project deliberately avoided a site-specific analysis but adopted a "generic" approach covering a wide range of sources terms and environmental dispersion conditions encountered in Germany.

KEYWORDS: NORM; discharges; atmospheric dispersion; public dose estimate.

1 INTRODUCTION

Council Directive 2013/59/Euratom (the European Basic Safety Standards, or BSS) [1] have removed the previous distinction between practices and work activities. This follows the concept adopted by the International Commission on Radiation Protection (ICRP) in its Recommendation No. 103 [2], which distinguishes planned, existing and emergency exposure situations. Henceforth handling and processing NORM is covered by the concept of practices.

Henceforth the dose limits for public exposure shall apply for the sum of annual exposure for members of the public resulting from all authorized practices, according to Art. 12 of the BSS. Discharges that are intrinsically part of a practice and may then be subject to regulatory control. Therefore practices involving naturally-occurring material (commonly but not entirely correctly known as "NORM industries") may need to be taken into consideration if exposure of members of the public resulting from discharges cannot be disregarded from a radiation protection point of view. Art. 12 (1) of the European BSS specifies that "the dose limits for public exposure shall apply to the sum of annual exposures of a member of the public resulting from all authorised practices". Hence, exposure of the public due to multiple sources must be taken into consideration. Consequently, in transposing the BSS into national law, Article 23 requires Member States *inter alia* to identify classes or types of practice involving naturally-occurring material that may need to be regulated. As guidance which practices may be relevant, Annex VI provides a list of industrial sectors that shall be taken into account. However, Member States may need to include other industrial sectors over and above the Annex VI of the BSS, if they are suspected of significant contributions to public exposures.

In Germany, the Federal Office of Radiation Protection (BfS) is responsible for investigating such issues and contracted IAF-Radioökologie GmbH (IAF), an accredited radiological laboratory and

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radiation safety consultancy, to implement a research project entitled "Estimation of potential public exposure due to discharges from NORM industries". IAF investigated the 16 industrial sectors of Annex VI of the BSS, and additional sectors that have been known in Germany for potentially being of radiological concern. In total, water- and airborne discharges from 26 industrial sectors were analysed with respect to the public exposures caused.

A preliminary "threshold of relevance" less than the dose limit for members of the public is required to account for the summation of multiple exposures. Within this project, an annual effective dose of 100 µSv/a received by the representative person as defined in Article 4 (89) of the BSS was used as a threshold for what is considered radiologically relevant (preliminary "relevance threshold" or working hypothesis for the purposes of this study). If doses resulting from multiple sources are added, this relatively low threshold of 100 µSv/a provides a sufficiently conservative framework. It is important to note that the project deliberately avoided a site-specific analysis but rather adopted a "generic" approach, i.e., it covered a wide range of situations that may typically be encountered in Germany, in terms of sources terms and environmental dispersion conditions of air- and waterborne discharges. While the research project covered discharges of dust, radon and water, this paper focuses on dust-borne radioactive discharges. Waterborne radioactive discharges and discharges of radon from the industrial sectors considered within this project have been found to be less relevant.

This paper provides an overview of the approach, methods and results of the research project. It largely follows the outline of the final report that will be available on the official BfS website [3]. Due to limited space, it provides little information on intermediate steps, and largely omits discussion of peculiarities of certain industrial sectors or processes. Readers who are interested in this in-depth information are referred to the final report of the project [3], or are encouraged to contact the authors.

2 IDENTIFICATION OF "NORM INDUSTRIES" THAT ARE CURRENTLY ACTIVE IN GERMANY

Annex VI of the EU BSS [1] provides some guidance on industrial sectors involving NORM, including research and relevant secondary processes that shall be taken into account when identifying sectors with discharges that may lead to exposure which cannot be disregarded from a radiation protection point of view. Since obviously not all of the sectors listed in Annex VI are currently active in Germany, those that are active had to be identified under the project in a first step. In addition to the above 16 sectors, another 10 sectors known to have existed, or are still existing, in Germany were deemed to be worth considering by BfS, based on historic evidence and experience. These additional sectors include, *inter alia*, the optical industry, the production of abrasives, ceramics and refractories, the use of thoriated welding electrodes, aluminium production including bauxite processing.

Whether these sectors still today are using materials with elevated levels of natural radioactivity, and may therefore have relevant discharges of natural radionuclides, was a question that had to be answered under this project. Apart from the mere existence of active operations in Germany belonging to the above sectors, it also had to be determined whether there are any relevant discharges from the operations. In answering this second question, a broad range of sources were used, including company information available in the public domain and interviews with industry representatives at both company and industry association level, best practice reference (BREF) documents of industrial sectors (e.g., the series of BREF documents released by the European Commission's Joint Research Centre on various industries), and finally the experience accumulated within the project team from more than two decades of working with NORM industries. It was found that with the exception of extraction of rare earths from monazite, thermal phosphorus production, and phosphoric acid production, all sectors listed in Annex VI of the BSS are active in Germany and are associated with discharges that may, in principle, be radiologically relevant.

In order to identify those sectors that can be safely excluded from a more detailed analysis, a series of steps were taken that are briefly described in the next section.

3 SCREENING OF NORM INDUSTRIES REGARDING THEIR RADIOLOGICAL RELEVANCE

The most relevant exposure pathways and radionuclides were identified that would lead to the highest effective doses in any of the age groups <1, 1-2, 2-7, 7-12, 12-17, and >17 years, respectively [4]. It was concluded that dust settling on agricultural products and the subsequent consumption chain "mother – breast milk – infant" leads to the highest effective doses, and infants (age bracket <1 year) are the representative person as defined by Art. 2 (89) of the BSS [1]. Unsurprisingly the nuclide with the highest contribution is Po-210 due to its high ingestion dose conversion factor. Due to its volatility, Po-210 is also predominantly present in dust from all typical high temperature processes, e.g., from cement kilns, furnaces, or smelters.

A deposition rate of 1 Bq/(m² a) of Po-210 on agricultural products leads to an effective dose of 39 µSv/a in infants (age group "<1 year"). For later reference (see Section 6) it is noted in passing that the same settling rate of 1 Bq/m²/a of Pb-210 on agricultural areas leads to an effective dose of 7 µSv/a in infants. Other exposure pathways such as inhalation of dust, or continued accumulation of radioactive dust on soil, contribute significantly less to the total effective dose and can be dismissed for the purposes of this study. A detailed breakdown of dose contribution from the various exposure pathways can be found in the full study [3] and is omitted here for brevity.

For airborne discharges, a simple, but sufficiently conservative, Gaussian plume dispersion model from [5] was used to calculate upper bounds of the dust settling rate. The model contains several parameters such as height above ground level of the point of emission, dust particle settling velocity that is closely correlated with the dust particle size, and angular distribution of wind direction. In order to span a sufficiently wide but conservative range of dispersion conditions, chimney heights of 20 and 100 meters, respectively, were used for the initial screening step. For dose estimates, maximum dust deposition rates were used, which for chimney heights of 20 and 100 m, respectively, occur at distances of 100 and 600 m from the source, respectively. It was also conservatively assumed that dust was only transported within a single angular element of 30 degrees, and a particle settling velocity of 0.01 m/s was used, which is at the upper end of settling velocities used in the literature for dust particles in the industry sectors in question, and in applicable technical standards [6].

The maximum sum of dry fall-out and wet wash-out of Po-210 from a standard source strength of 1 GBq/a is of the order of 4 Bq/(m² a) and 200 Bq/(m² a), respectively, for the above chimney heights of 100 and 20 meters, respectively. This allows, in turn, to define conservative screening thresholds for airborne dust discharges: A discharge of 15 MBq/a of Po-210 from a chimney of a height of 20 m, or 700 MBq/a of Po-210 from a chimney of a height of 100 m, respectively, lead to an annual effective dose of approximately 100 µSv in infants (representative person).

4 ESTIMATE OF DISCHARGES FROM POTENTIALLY RELEVANT INDUSTRY SECTORS

4.1 Data sources

In order to *quantitatively* estimate radioactive discharges, a broad range of information sources were used, including but not limited to environmental permits of operations including permit conditions that limit airborne and/or waterborne discharges, emission data from the Pollutant Release and Transfer Register (PRTR) set up under European Directive 2006/166/EC [7], information provided by plant operators or industry associations, process technology textbooks and good practice handbooks of some industrial sectors both in Europe and overseas, data on the specific activity of raw materials, wastes, and discharges, that can be found, *inter alia*, in articles and presentations of NORM and IAEA conferences.

Information on non-radiological parameters of discharges such as dust streams (tons per year, particle size distribution) or wastewater streams (m³/h, concentration of total suspended solids), is publicly available to a limited extent for some but not all industries. However, information on radiological

parameters such as specific activity of dust particles or dissolved and/or particulate-bound radionuclides in waste water was extremely difficult to come by for some industries. Estimates using analogues and simplified radioactivity balances of the industrial process in question based on assumed radiological properties of raw materials were needed to close data gaps. If no data on airborne discharges (source term expressed in activity per unit time) of natural radionuclides was forthcoming, the discharged activity was estimated using one, or a combination, of the following methods depending on the data available:

- Maximum mass flow of dust particles (kg/h) allowed under the environmental permit of an operation, or obtained from published data such as the PRTR [7], multiplied by the specific activity of the dust (Bq/g). The specific activity of the dust may be estimated using the specific activity of the raw material and the typical mass reduction factor of the process.
- Reverse modelling of (non-radioactive) dust deposition data (g/m²a) available from environmental monitoring programs to obtain the mass flow (t/a), e.g., environmental monitoring data of conventional lead deposition around a lead smelter. The radioactive discharge was then obtained from multiplying the mass flow by the specific activity of the dust particles (Bq/g),
- Consumption rate of raw materials (t/a) were multiplied by the specific activity of the raw material (Bq/g), resulting in an activity flow entering the operation. It is then conservatively assumed that the nuclides of volatile elements such as Po-210 and Pb-210 leave the process attached to dust particles. An industry-specific retention rate of dust filters (see Section 4.3) is applied to estimate the actual discharge through the chimney.

Discharges from some sectors are constrained by general environmental protection or emission control regulations, and/or best environmental protection practice adopted by most industrial operators in Germany. Such considerations that are not related to radioactivity in the first place are very useful in determining upper bounds to emissions in general. For example, according to the Technical Instructions on Clean Air [8], the limit of the average annual concentration of PM10 in ambient air is 40 µg/m³, and total dust emissions (including the PM10 fraction) must not exceed either a mass stream of 200 g/h or a concentration of 20 mg/m³. As it was assumed in the study that industry operators comply with these limits as part of their environmental permits, these constraints indirectly put upper limits on radioactive discharges. In turn, this fact can be used to determine the radiological relevance of an industry sector.

4.2 Industry sectors that are subject to more detailed investigation

A combination of all information sources and approaches described in Section 4.1 have enabled the study team to draw relatively robust and plausible conclusions on the discharges from most industries listed in Table 1, expressed in GBq/a, at least up to the order of magnitude. Details of the discharges from each industry sector can be found in the study [3] and are not reproduced here for limitations on space. Upon application of the screening method described in Section 3, Table 1 provides an overview of those industry sectors with potentially relevant discharges that exceed the conservative screening threshold and are therefore subject to more detailed investigation. Note that the data in Table 1 describe a base case that is complemented by a sensitivity analysis in Section 4.3.

Table 1: Industry sectors that exceed the screening threshold of radioactive discharges and therefore are subject to more detailed investigation, base case of production figures

Potentially relevant sectors	Potentially relevant processes and associated discharges	Typical production figures	Typical discharges of relevant radionuclides (GBq/a)
Cement production	Dust from clinker kilns	0.7 Mt/a	Pb-210: 0.014 Po-210: 0.28
Coal-fired power plants (hard coal)	Dust from coal combustion	1 GW_{el}	Pb-210: 0.67 Po-210: 1.35
Primary iron production	Sinter and smelter dust	5 Mt/a	Pb-210: 8.5 Po-210: 8.5
Lead smelting	Dust from smelting of ore concentrates	0.1 Mt/a	Pb-210: 0.3 Po-210: 0.3

4.3 Parameter variations for sensitivity analyses

It must be emphasised that the data collected in the study only reflect orders of magnitude of an industry as a whole covering typical ranges of operating facilities. In order to arrive at a realistic picture of the radiological impact of discharges from a NORM industry *sector* on public exposure, a careful sensitivity analysis is required. While Table 1 contains a "base case" scenario for each industry sector with regard to production figures that reflect the size of a *typical* plant in Germany, parameters for conservative scenarios were developed for the purposes of a sensitivity analysis, as shown in Table 2. Over and above the variations of the parameters shown in Table 2, the atmospheric dispersion conditions are subject to uncertainties that must be taken into consideration in a sensitivity analysis, too. Variations of atmospheric dispersion parameters are considered in Section 5.4 below.

Table 2: Variations of emission parameters for industry sectors from Table 1

Industry sector	**Conservative production figures**	**Discharges of relevant radionuclides (GBq/a)**	**Variation factor to account for uncertainties of the specific activity of dust particles**	**Dust filter retention rate (%)[a]**
Cement production	2 Mt/a	Pb-210: 0.04 Po-210: 0.8	2	95-**99**
Coal-fired power plants (hard coal)	2 GW_{el}	Pb-210: 1.3 Po-210: 2.7		95-**98**
Primary iron production	10 Mt/a	Pb-210: 17 Po-210: 17		97-**99**
Lead smelting	0.15 Mt/a	Pb-210: 0.45 Po-210: 0.45		95-**99**

(a) Base case retention rates on which figures in Table 1 are based are printed bold face

5 DETERMINATION OF GENERIC DISPERSION CONDITIONS FOR DUST

5.1 Topographical and meteorological parameters

As stated in the Introduction (Section 1), the objective of this study is *not* a site-specific calculation of public exposure due to discharges from a particular facility, but an investigation of which NORM industries would warrant regulatory control because discharges of a facility of "typical" size and technological parameters under "typical" meteorological and topographical conditions may lead to significant public exposure. Following this objective the challenge of this study was to develop a range of characteristic dispersion conditions that are typical for a sufficiently large number of NORM industry facilities operating in Germany. This task has therefore been tackled from various directions, as outlined in the following:

- *Geographical distribution of facilities:* The geographical distribution of facilities in the potentially relevant industrial sectors listed in Table 1 is analysed in order to determine whether there are regions in which these facilities are concentrated. If this is the case, the meteorological and topographical conditions of those regions should be considered "typical".
- *Meteorological parameters:* For geographical regions in which a large number of NORM facilities with discharges are concentrated, the statistical distributions of meteorological parameters such as wind speed and direction, turbulence (stability) class, and precipitation patterns must be obtained. However, even more important than the statistical data for each region is the question how sensitive the atmospheric dispersion depends on the meteorological conditions. In relation to this question it is important to remember that the source terms are known only up to an order of magnitude, so that the requirements on the precision of the dispersion modelling are limited. The German Meteorological Service (DWD) provides so-called TRY (Test Reference Year) model datasets that represent average weather conditions over the course of one year in one-hour time steps [9]. They were available for 15 characteristic regions[1]. From an evaluation of the geographical concentration of the locations

[1] It should be noted that from 2018 the datasets for 15 TRY regions that were used for the study, have been replaced by datasets for each square kilometre of Germany.

of NORM facilities the following can be concluded that 70% of all sites of potentially relevant NORM industries listed in Table 1 are located in only 4 TRY regions (Nos. 3, 5, 6, and 12). TRY regions no. 3, 5, and 6 are very similar with respect to key meteorological parameters such as wind speed and direction, and have been treated with one common meteorological dataset (that of TRY region no. 6). TRY region no. 12 is characterised by a bimodal angular distribution of wind directions. Therefore two distinct datasets were used to cover a wide range of meteorological conditions typical for NORM facilities in Germany. Between the various locations, precipitation varies typically between 500 and 1000 mm/a.

- *Topographical relief:* The overwhelming majority of sites of potentially relevant NORM industries listed in Table 1 are located in lowlands and rolling country with relatively flat hills. This fact significantly simplifies the atmospheric dispersion modelling.

For the numeric modelling of atmospheric transport of dust and radon the code ARTM (Atmospheric Radionuclide Transport Model [10]) was used. ARTM is based on Lagrange trajectories of unit air volume elements. It includes dry and wet deposition and takes into account radioactive decay along the trajectory.

5.2 Other relevant parameters, and parameter variations

To test the sensitivity of the deposition rates (and hence, the effective dose) with respect to variations of the atmospheric dispersion conditions, the following sensitivity analyses were carried out:

- The parameter that is ultimately relevant for the dose calculations under this project is the deposition rate of dust-borne radioactivity on agricultural areas. While the wet deposition rate depends on the precipitation rate and its temporal distribution (short, intense rainfall events vs. long periods of drizzle), the dry deposition rate depends on the atmospheric stability class and on the deposition velocity of the dust particles, which in turn depends on the particle size distribution.
- Dispersion modelling started with a complete initial parameter set for a reference site; the parameter sets for precipitation and stability class distribution were then varied to test the sensitivity of the model results with respect to these parameters.
- Different settling velocities were tested in the dust dispersion model.
- In order to test the sensitivity of the radionuclide deposition rate with respect to variations the atmospheric stability class, variable statistics were used, ranging from stable highly stable to unstable, and the resulting deposition rates compared.
- Atmospheric dispersion patterns also strongly depend on the height of the source. A survey carried out under this project has identified typical ranges of chimney heights in each of the industry sector of interest.
- Another important parameter for the dust deposition and dose model is the lateral distance between source and point of impact. Based on empirical evaluation (using Google Earth) of randomly selected sites of the first four industries listed in Table 1 it was found that typically the lateral distance between the point of emission and areas of agricultural use is at least four times the chimney height. As a, possibly oversimplified, working hypothesis it can be assumed that plants with large emissions (and higher chimneys) are part of industrial zones that are generally farther away from agricultural areas than smaller plants. The model base case, therefore, assumes that the point of impact is four times the chimney height, whereas in a conservative variation this distance is reduced to twice the chimney height.
- As was stated in Section 5.1, most facilities of the industry sectors in question are located in areas of a relatively flat topography or in mildly rolling hills. In order to test the model sensitivity with respect to special topographical conditions, dust dispersion calculations were also carried out using the digital surface model of a relatively deep and narrow valley.

5.3 Results of dust dispersion and deposition rate modelling, base case

For the base case parameters of atmospheric dispersion conditions the model results of the deposition rate for Pb-210 and Po-210 are summarised in Table 3. The deposition rates are normalised to a source strength of 1 GBq/a. Deposition rates that were obtained using the ARTM code for TRY regions 6 and

12 and for two sites with high (1000 mm/a) and low (559 mm/a) precipitation, respectively, were averaged and the result rounded up to the next full integer of the base. It is worth noting that wet deposition dominates total deposition in the near field of distances up to around 10 times the chimney height.

Table 3: Deposition rates of Pb-210 and Po-210, normalised to a source strength of 1 GBq/a, at a lateral distance from the source of four times the chimney height

Chimney height (m)	Average total (wet and dry) deposition rate (Bq/m^2s), rounded up to full integer of base	Standard deviation of modelling results used for averaging (% of average)
50	1E-7	46%
100	3E-8	27%
200	1E-8	19%

5.4 Sensitivity analyses of the dust deposition rate modelling

Sensitivity analyses were carried out to cover a sufficiently broad range of dispersion conditions. Apart from the dispersion conditions, the source strength of discharges may vary depending on the size and technical parameters of the industrial facility, as shown in Table 2. A complete account of the model results obtained for a large number of scenarios would go beyond the limited space available in this paper but can be found in the original study [3]. An important conclusion is that combining only a few parameter variations from Table 2 and Section 5.2 may lead to deposition rates and effective doses that are higher by a factor of 10 and more compared to those of the base case.

6 ESTIMATE OF EFFECTIVE DOSES DUE TO THE DISCHARGE OF DUST

Using the base case scenario for both source strength of dust discharge and atmospheric dispersion parameters, and the relationship between deposition rate of Pb-210 and Po-210 and the effective dose incurred by the most sensitive age group (infants) established in Section 3, the following results have been obtained (doses exceeding the threshold of 100 µSv/a are printed bold face).

Table 4: Effective annual dose in infants (µSv/a) due to dust discharges from potentially relevant industry sectors

Industry sector, base case	Chimney height (m)		
	50	**100**	**200**
Cement production	3.3E+01	9.9E+00	3.3E+00
Coal-fired power plants (hard coal)	n.a.	5.2E+01	1.7E+01
Primary iron production	n.a.	**3.5E+02**	**1.2E+02**
Lead smelting	4.2E+01	1.2E+01	n.a.

In the base case, only primary iron production leads to effective doses in infants that exceed the threshold of relevance of 100 µSv/a. However, as was mentioned in Section 5.3, variations of a combination of only a few parameters within a sensitivity analysis may easily lead to significant increases of the activity deposition rate by a factor of 10 and more, and hence of the effective dose. Therefore, it is obvious that discharges of virtually all four industries listed in Table 4 may be radiologically relevant in the sense of the “relevance threshold” used in this study. It must be emphasised again that these considerations are based on a generic approach; the discharge from each site must be evaluated individually.

7 SUMMARY OF RESULTS

From the 16 industry sectors enumerated in Annex VI of the European Basic Safety Standards [1] and the 10 additional sectors that were investigated within the scope of this study, only a relatively small number has turned out to be potentially relevant from a radiation protection point of view. More specifically, only dust discharges from primary iron production are likely to exceed the threshold of

100 µSv/a in the most vulnerable age group (infants). Within an extensive sensitivity analysis it was found that under more conservative assumptions regarding the source strength of the discharges and atmospheric dispersion conditions, cement production, hard coal fired power plants, and lead smelting may also lead to effective doses of more than 100 µSv/a in infants, if several conservative parameter variations are combined.

One of the reasons for the relatively low effective doses resulting from discharges of dust is the industry's compliance to strict regulatory limits on air- and water-borne discharges that have been introduced independent of radio-ecological considerations in the first place.

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CSN's Pilot Inspection Programme as a tool for achieving sustainable compliance in NORM industries

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Abstract. This paper describes the main aspects of an ongoing pilot programme carried out by CSN on NORM industries. Besides providing the basis to design the systematic inspection programme for NORM, the two main aims of the Programme are: (a) to allow inspectors to learn on the ground about the obstacles preventing firms from complying with regulation; (b) to identify synergies with ongoing health & safety and environmental controls. A flexible, educative approach is adopted, with a focus on progressing towards efficient implementation at the facility level. The Programme's outcomes are discussed in view of their effectiveness to provide sustainable compliance solutions in industries that have traditionally operated out of regulatory radiological control.

KEYWORDS: NORM, regulatory compliance, inspection

1 INTRODUCTION

For a few of decades now, NORM processing or generating industries have been subject to regulatory radiological control in several countries. At European level, the 1996 Basic Safety Standards (BSS) Directive included some general-scope provisions on work activities involving NORM. These provisions, however, had a very uneven implementation across Member states. The new BSS – published as Directive 2013/59 – reflect a clear commitment of the Commission to consolidate the regulatory control of NORM industries, requiring they be managed in the same framework as other practices.

In Spain, regulation specific to the control of NORM industries was passed in 2011 [1], expanding on the requirements earlier imposed in transposition of the 1996 BSS [2]. Nevertheless, those legal mandates translated into a poor practical implementation for several diverse factors. Thus, in a parallel effort to the BSS transposition works, a strategy to foster compliance was designed and led by CSN (Spain's radiation protection authority).

This compliance strategy includes four core lines of action, one of them being a two-year Pilot Inspection Programme for NORM industries. The strategy accounts for the fact that NORM industrial sectors in Spain, as across the whole EU, are highly regulated in the fields of protection of the environment and worker health. Within this specific context, a pragmatic approach to practical implementation of the BSS is required in order to avoid unnecessary burdens while ensuring an adequate level of radiation safety.

Unavoidably, this entails collaboration among the various authorities involved (and, desirably, coordination memorandum of understanding among their inspectorates), as well as a permanent dialogue between industry and regulator, to which this Programme searches to contribute.

2 REGULATORY FRAMEWORK

NORM industries are regulated in Spain under Title VII of Royal Decree 783/2001 (transposing directive 96/29/Euratom) and CSN Safety Instruction IS-33 (issued 2011).

Following IS-33, all industries included on a positive list have to register with the regional industry authority and to conduct a study on their radiological impact to workers and public. If the results of the study show that the dose criteria (1 mSv/y for workers and 0.3 mSv/y for public) are exceeded, the facility has to notify it, and it will remain subject to regulatory control.

The level of control imposed varies in terms of risk and plausibility of protection measures, with additional requirements applying if doses to workers are liable to exceed 6 mSv/y.

Complementary to this legislation, Ministerial Order IET/1946/2013 [3] regulates the control of NORM residues. Exemption-clearance levels in Radiation Protection 122 Part 2 are used as a screening tool in order to determine which residues need further consideration from the radiation protection standpoint. For residues exceeding those levels, a case-by-case analysis needs to be performed. Conventional disposal routes are approved provided that requirements in IS-33 (including the application of the optimisation principle) are observed and that specified dose limits are met. These are 6 mSv/y for workers, and 1 mSv/y for public. NORM residues above these limits need to be managed as radioactive waste.

Additionally, CSN has issued two Safety Guides on the content and methodology of the mandatory radiological studies. These are GS 11.2 [4] and GS 11.3 [5].

Some amendments to this framework will be enacted as a result of the BSS transposition – still in progress – although the fundamental of this regulatory control system will remain unchanged.

3 PROGRAMME SCOPE AND DEVELOPMENT

Out of each NORM-regulated sector currently operating in the country, one facility was selected based on a two-stage sampling: first, if applicable, the highest-risk type of facility in the sector was identified; second, the installation was selected by random sampling among these. Following this procedure, the following facilities were chosen for inspection:

- A titanium dioxide production facility
- A coal-fired power plant
- A gas extraction site
- A Zr-milling plant
- A primary-production steelworks
- A dicalcium phosphate production plant
- A cement plant
- A non-hazardous landfill receiving NORM waste

Inspections are announced to the operator at least one moth in advance. Their scope is defined by radiation protection regulation (as specified in section 2 of this paper). They are conducted according to the following scheme:

- Opening meeting
- Discussion and interviews with personnel of the operator
- Examination of procedures and equipment, records and documentation
- Walk-through inspection. Includes in-situ measurements (such as of gamma dose rates, radon in air concentration, or surface activity) and/or collection of samples for subsequent analysis
- Closing meeting

Within 15 days after the inspection, an inspection report is prepared and sent to the operator, allowing them to file written submissions. Taking those into consideration, a final version of the inspection report is elaborated and made available to the public.

4 PROGRAMME OUTCOMES

The first most evident effect of the Pilot Programme was an improvement in compliance across all regulated sectors. Prior to the star of the Programme, CSN publicized it among the industry, although not disclosing the particular types of facilities selected. This resulted in an increase in the number of operators submitting a registration to the corresponding regional authority.

Despite failures to comply with radiation protection regulations, all the facilities inspected operate under high standards of environmental and worker health protection standards, thus resulting in an indirect control of radiation doses. Most of them have adopted self-regulation schemes, such as ISO 14001 or OHSAS 18001 certifications. These achievements pave the way for developing a radiation safety culture. Nevertheless, the situation in small firms operating in some NORM sectors might be different.

A common complaint was a lack of clarity of the regulation at the operational level. Guidance on specific technical issues or operation surveillance and control methods was provided on site by the inspectors. Poor technical advice was not infrequently identified as the cause of deficiencies. Indeed, education, training and certification of radiation protection experts and officers specialized in NORM had been already identified as one of the priorities for CSN, and much progress has been achieved in this area in 2018.

With regard worker protection, the proper classification of radiation controlled areas and the protection of external workers operating therein are the most urgent needs for improvement. From an economic standpoint, meeting the requirement for the characterization and disposal of NORM waste has the most impact. In particular, better solutions need to be provided for the management of NORM-contaminated scrap metal at the country level.

The accumulated experience during the Pilot Programme will also be the basis to develop CSN's inspection procedures specific to NORM, as well as to design the long-term inspection strategy for NORM industries, including setting strategic objectives and risk-based frequencies of inspections and allocating enough inspection resources.

5 CONCLUSION

In a newly regulated field, inspections are key to filling the gap between written regulation and practical implementation. At all the installations inspected, both the existing workplace health and safety culture, and the environmental programs in place, largely facilitate the introduction of a radiation safety culture. However, compliance is hampered by a lack of regulation clarity at the operational level and, often, poor external expert advice. Difficulties raised by operators or identified by inspectors in the frame of the Programme, allow CSN to feed the dialogue with individual facilities and industry associations, in search for optimal sector-specific solutions. On the other hand, economic or technical issues identified on NORM waste management call for a strengthening of the national strategy, in order to provide producers and waste managers with more effective compliance solutions.

6 REFERENCES

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Integrated approach for workers protection in industries involving NORM

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Abstract. Many industries involve naturally occurring radioactive materials (NORM) in their processes. For many of these industries, exposure to radioactive materials is not tackled as a specific issue. As a result of the transposition of the European directive 2013/59/Euratom, the French labour code is going to be amended to foster a comprehensive approach in the field of occupational risks (fire, explosion, chemistry…). In this manner, all the hazards are considered in the Health, Safety and Environment (HSE) policy of the plant. One of the aspects of this integrated approach is that all the hazards due to Uranium are taken into account. Indeed, among the naturally occurring radionuclides, Uranium 238 (and its daughter product Uranium 234) and Uranium 235 are found in variable amounts in ores. Natural Uranium presents a chemical toxicity and a radiological toxicity. However, currently in France, only the radiological toxicity of Uranium is considered by the operators. To optimize the workers protection, it may be necessary to take into account the chemical hazard arising from uranium, in addition to the radiological hazard. HSE policy shall encompass all the provisions against all the risks and check that there is no conflict between them: i) optimization of the workplace, taking into account the collective protection equipment and the monitoring, ii) zoning plan, iii) individual protection equipment. Moreover the issue of Radon 220, Radon 222, and their daughters cannot be disregarded. Generally, this aspect in industries involving NORM is considered apart from the other hazards, even the radiological hazard. The management of radon or thoron is a current open question.

KEYWORDS: NORM, workers protection, risks management

1 CONTEXT

Many industries involve naturally occurring radioactive materials (NORM) in their processes.
Due to the wide range of raw materials and that of processes, the occupational radiation exposure is very variable for the different workplaces of a plant, for the different plants of an industrial sector. Furthermore, for many of these industries, exposure to radioactive materials is not tackled as a specific issue.

2 REGULATIONS

2.1 Former regulation for industries involving NORM

According to the European directive 96/29/Euratom [1], France sat up a specific regulation for the protection against the natural occurring radiative sources. The Public Health Code and the Labour Code were implemented and a ministerial Order published on 25 May 2005 [2]. This ministerial Order sets a list of industrial sectors involving NORM and requires dose assessments for workers and public members.

The operator of plant in which the occupational exposure cannot be ignored from a radiation protection point of view must apply the principles of radiation protection that are **justification**, **optimisation** and **limitation** and put in place radiation protection provisions.

2.2 Evolution of the regulation

The review on enforcement of the ministerial order of 25 May 2005 [3] has proposed to have a new list of occupational activity categories endorsed by the regulation: the existing list completed by some activity categories as “oil and gas production”, “paper mills”…

The new French positive list, presented in the Environment Code, is based on the list presented in the annex VI of the European directive 2013/59/Euratom [4], excluding “mining of ores other than uranium ore”. This list is completed by “Extraction of natural materials of magmatic origin such as

granitoides, porphyries, tuff, pozzolana and lava when they are intended to be used as a construction product". So the final French positive list of industrial sectors involving NORM is:

- Extraction of rare earths from monazite,
- Production of thorium compounds and manufacture of thorium-containing products,
- Processing of niobium/tantalum ore,
- Oil and gas production,
- Geothermal energy production,
- TiO_2 pigment production,
- Thermal phosphorus production,
- Zircon and zirconium industry,
- Production of phosphate fertilisers,
- Cement production, maintenance of clinker ovens,
- Coal-fired power plants, maintenance of boilers,
- Phosphoric acid production,
- Primary iron production,
- Tin/lead/copper smelting,
- Ground water filtration facilities,
- Extraction of natural materials of magmatic origin such as granitoides, porphyries, tuff, pozzolana and lava when they are intended to be used as a construction product.

One observation of the enforcement of the ministerial order of 25 May 2005 is the difficulty in enforcing this ministerial order. Indeed, the authorities have only received about 100 studies by the operators whereas there are many more facilities concerned by the list of activity categories mentioned in the order. Furthermore, some operators are aware of having to carry out studies consistent with the ministerial order but they have not wished to do it. Indeed, they feared that the image of their society would be tarnished due to the notion of radioactivity.

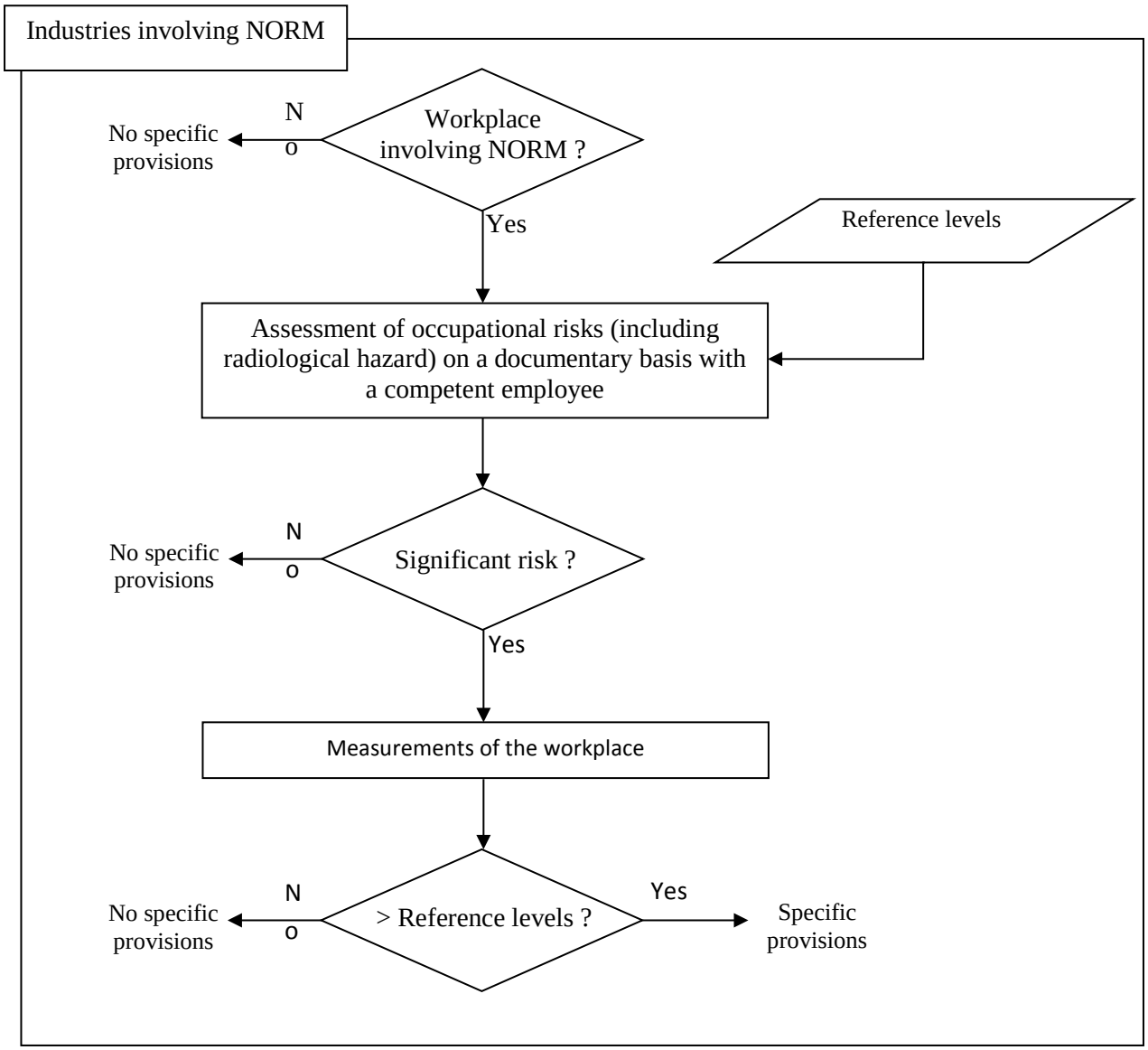
So the regulators have decided to take advantage of the transposition of the European directive 2013/59/Euratom [4] to strengthen their position. As a result, the industries involving NORM will no longer be considered in a specific manner, but as "practices", i.e. like the other nuclear activities. But in order to have a graded approach, the regulators will introduce exemption thresholds for NORM (especially for administrative aspects and for waste management).
Likewise, the French labour code has been amended to foster a comprehensive approach in the field of occupational risks (fire, explosion, chemistry, ionising radiations…). Indeed, industries involving NORM often give rise to multiple hazards, and the radiological hazard is not necessarily the priority. An integrated approach to safety and protection has been identified as necessary by the French regulators, bearing in mind that the radiological protection system is not necessarily the driving force. The starting point is protection strategies already implemented by industries involving NORM to manage other workplace hazards and then assessing, after characterisation, the need for any further action for protection against radiation.

The figure 1 is proposed to illustrate the graded approach for implementing provisions against all the risks.

For example, the reference levels may be occupational exposure level for the chemical risk, dose constraint (1mSv/year or else) for the radiological risk, equivalent dose rate for the radiological zoning…

All the hazards are considered in the Health, Safety and Environment (HSE) policy of the plant.
Furthermore, this approach is consistent with the graded approach recommended by the European directive 2013/59/Euratom [4], ICRP and IAEA.

Figure 1: Graded approach for implementing the risks provisions



3 RISKS MANAGEMENT IN FRANCE

Industries involving NORM often give rise to multiple hazards:

- Risks related to products, emissions and waste (including chemical risks),
- Risks of fire and / or explosive (notion of ATEX zones),
- Risks of tripping, bumping or other disruption of movement,
- Risks of falling from height,
- Risks related to work equipment,
- Risks related to mechanical handling,
- Risks related to the internal circulation of vehicles,
- Risks related to electricity,
- Risks related to the thermal environment,
- Risks related to collapses and falling objects,
- Risks related to the physical workload,
- Risks related to ionizing radiation,
- Road risks on mission,
- Risks related to lighting environments,
- Risks and nuisances related to noise.

3.1 Main principles of risks management

The main principles for occupational risks management are the following:

- **Avoid risks**: eliminate danger or exposure to danger.
- **Evaluate the risks that can't be avoided**: assess the exposure to danger and the importance of the risk in order to prioritize the preventive actions to be carried out.
- **Fight risks at the source**: integrate prevention as early as possible, especially when designing workplaces, equipment or operating procedures.
- **Adapt the work to humans**, in particular as regards the design of workplaces and the choice of work equipment and working and production methods, with the aim of reducing the effects of work on health. For this, it is necessary to also take into account differences between individuals.
- **Take into account the evolution of the technique**: adapt the prevention to the technical and organizational evolutions.
- **Substitute what is dangerous for what is less hazardous**: avoid the use of hazardous processes or products when the same result can be achieved with a less hazardous method.
- **Plan prevention** by integrating, in a coherent whole, the technique, the organization and the working conditions, the social relations, the environment (influence of the ambient factors).
- **Take collective protective measures** giving them priority over personal protection measures. Use personal protective equipment only in addition to collective protection if it proves to be insufficient.
- **Giving the right instructions to workers**: train and inform workers so that they know the risks and the prevention measures.

These principles are consistent with the principles of radiation protection (justification, optimisation and limitation).

3.2 Generalities about chemical risk management

The prevention of chemical risks is based on the general principles of prevention defined in the French Labour Code.

Chemicals that come into contact with the human body (through the respiratory tract, skin and mouth) can disrupt the body's functioning. They can cause:

- Acute intoxications, with more or less serious effects,
- Chronic intoxications: repeated contact with certain chemical agents, even at low doses, can damage the lungs, nerves, brain, kidneys.

Inhalation is the most common pathway of occupational exposure. The second occupational exposure pathway is the percutaneous route, by contact and / or by passing through the skin: the effects are then either local (irritation, burn, necrosis ...) or general. The mode of exposure by ingestion is also very important for certain products (powders, metals) in the workplace either because of swallowing ingestion of previously inhaled substances (by swallowing its saliva) or because of hygiene problems (dirty hands). Ingestion can also be accidental (deconditioning of mixtures used for example).
The quantitative risk assessment for workers consists of verifying that the exposure of workers does not exceed the reference values known as Occupational Exposure Limits (OELs).

Currently, in France, there are OELs only for inhalation: it is the concentration in the air of a chemical compound that can breathe a person for a determined time without risk of deterioration of health, even if reversible physiological changes are sometimes tolerated. It should be kept in mind that OELs meet minimum prevention objectives and that exposure should be as low as reasonably achievable. OELs are expressed as mg/m^3, ppm, $fibre/cm^3$ or fibre/litre.

For the specific case of carcinogens, there is rarely a threshold of exposure below which the risk can be considered zero. For these substances, OEL must be considered as a useful value for risk characterization and is not protective against possible carcinogenic effects.
There are different types of OELs:

- "Average Exposure Limit Value" intended to protect against long-term effects. The pollutant concentration in the breathing zone of workers is measured or estimated over the duration of an 8-hour workplace.
- "Short Term Exposure Limit Value" to protect against the effects of exposure peaks. The pollutant concentration in the breathing zone of workers is measured over a maximum duration of 15 minutes.

4 SOME REFLECTIONS ON THE RADIATION PROTECTION PROVISIONS

The integrated approach has some consequences on the radiation protection provisions as the radiological zoning, the collective/individual protection equipment, radon...

4.1 Chemical toxicity of uranium

Among the naturally occurring radionuclides, Uranium 238 (and its daughter product Uranium 234) and Uranium 235 are found in variable amounts in ores. Natural Uranium presents a chemical toxicity **and** a radiological toxicity. However, currently, only the radiological toxicity of Uranium is considered by the operators in many countries including France. To optimize the workers protection, it may be necessary to take into account the chemical hazard arising from uranium, in addition to the radiological hazard.
Unlike exposure to ionizing radiation, no regulatory value for uranium is defined in the Labour Code for chemical impact. It is therefore necessary to rely on values of references defined by French or international organizations. However the approach for chemical protection is based on an airborne concentration control (*cf.* 3.2) whereas the approach for the radiation protection is based on a dose assessment taking into account an airborne concentration control among other things.

IRSN (Institute of radiation protection and nuclear safety, France) performed a survey of the existing values to determine the value that it will retain for its studies.

Like many other heavy metals, the target organ to consider for the toxicity of uranium is the kidney. Chemical toxicity depends on the mass of incorporated uranium, its solubility and the duration of exposure. In contrast to radiological toxicity, chemical toxicity does not depend on the isotopic composition of uranium, two isotopes of the same element having the same chemical properties.
Exposure pathways for the chemical toxicity of uranium are ingestion, inhalation, and skin contact. However, there is very little information at present for ingestion and skin contact exposure routes, so only the inhalation route, which is generally predominant in industrial settings, is retained.
The type of dominant toxicity depends to a large extent on the solubility of the element under consideration:

- The insoluble elements of Uranium tend to stagnate in the lungs after inhalation and do not allow sufficient renal deposition to induce toxic chemical effects. The insoluble forms of uranium are mainly the oxides type (type of slow absorption S) [5]. The predominant risk in the presence of uranium in insoluble form will then be the radiological risk.
- Conversely, a soluble element will be easily transferred to the blood after inhalation and then to the kidneys, the target organ of chemical toxicity. Soluble forms of uranium are mainly hexavalent compounds (rapid absorption type F) [5]. Some less soluble chemical forms are also to be considered as nephrotoxic (average absorption type M). The predominant toxicity in the presence of uranium in soluble form F and M will then be the chemical toxicity.

The French bodies in charge of the regulation of chemicals have not yet defined an OEL for uranium, hence IRSN needed to rely on international reference values.
For soluble compounds, OSHA (Occupational Safety and Health Administration, USA) recommends not to exceed, in the workplace and for chronic inhalation exposure, an atmospheric concentration of 0.05 mg/m^3 (8 hours of work per day) [6]. WHO (World Health Organization) has defined two values [7], one for classified workers and the other for the public. The limit for respiratory exposure to the soluble forms of uranium is 0.05 mg/m^3 for duration of 8 hours per day. The recommended limit for

public exposure is 0.001 mg/m^3 (for natural uranium and depleted uranium). These values take into account both the radiological toxicity and the chemical toxicity.

IRSN retains [8] that the reference value for workers, which can be considered as an occupational exposure limit value (OEL), is 0.05 mg/m^3 (chronic exposure, 8 hours per day) for soluble compounds of uranium. IRSN notes, however, that this value may lead to exceeding the value of 1 mSv/year taking into account the radiological risk.

4.2 Radiological zoning

In some cases, it is necessary to implement zoning for radiation protection. These zones can be put in place because of the risk of irradiation or the risk of contamination. These are supervised areas or even controlled areas.
Access to these areas, especially controlled areas, is subject to conditions. Therefore, for installations using ionizing radiation sources, the installation and even the workplaces are designed taking into account radiation protection provisions, including radiological zoning.
However, the industrial plants involving NORM have not been designed taking into account the radiological risk. In this case, the radiological zoning is of the "leopard spots" type. All workers in an installation are not "classified" due to exposure to ionizing radiation. Their workplace may however require punctual access or regular crossing of a supervised or controlled area. Changing their workplace to avoid regular crossing of a controlled area could expose them to other risks. It is therefore necessary to determine, on a case-by-case basis, and with all the persons concerned (the concerned worker(s), the radiation protection expert, the competent employee for the other risks, the occupational physician, etc.) changes to take into account all risks.

4.3 Collective and individual protection equipment

One of the principles of radioprotection is the optimization of the workplace. This optimization is done by setting up collective and/or individual protection equipment. Collective protection equipment is to be preferred over individual protective equipment.
It is necessary to ensure that the implementation of collective protective equipment against the radiological risk will not degrade the protection against other risks on the workplace concerned or on neighbouring workplaces.

Likewise, it is necessary to ensure that all the individual protective equipment that a worker must wear to protect himself against all risks does not hinder the work to be done, leading de facto to a deterioration of the protection of the worker against some other risks (such as the risk of disruption of movement, the risk associated with mechanical handling, the risk related to thermal environments, etc.). In order to limit this problem, a reflection must be carried out on the compatibility of some equipment against several risks. For example, the choice of gloves or clothing material can help protect against chemical risk and radiological risk. In the same way, it is possible to consider that the mask has a synergistic character against dust, radioactive aerosols and particulate radon progeny.

4.4 Radon (Rn222) and Thoron (Rn220)

It should be noted that in France only some workplaces (underground and ground floor places) and some departments are covered by the French regulations on Radon 220 (Thoron), Radon 222 (Radon) and their daughters.

Since the transposition of the European directive, the Labour Code recommends a reference level for the annual average activity concentration in air of 300 Bq/m^3. If the annual average activity concentration in air is higher than this reference level then operator must implement provisions to reduce the activity concentration in air.

The notion of radon/thoron in industries involving NORM is considered apart from the other hazards, even the radiological hazard. However, the workplace study should consider radon and thoron as well

as other exposure pathways, even if the annual average activity concentration in air is lower than 300 Bq/m^3.

A graded approach for the radon/thoron management in workplaces is discussed in France (a ministerial order is under development). According to the estimated impact, the operator implements necessary provisions:

- Identification of any area where workers are likely to be exposed to levels exceeding, for the concentration of activity of radon in the air (expressed as an effective dose), the value of 6 mSv/year. These areas are called "radon zone".
  The assessment of the exposure levels used to identify these areas is carried out taking into account a workplace permanently occupied and the following aspects:
  - The nature of the sources,
  - The duration of the exposure,
  - Information about radon emanation level,
  - The existence of protective equipment such as ventilation or catchment installations to reduce the level of exposure,
  - Reasonably foreseeable incidents inherent to the work process or to the work performed.
- An individual monitoring of the workers that are exposed to levels exceeding the value of 6 mSv/year due to radon and thoron. In this case, the concerned workers might wear a radon dosimeter and have an adapted medical follow-up.

5 CONCLUSIONS

Due to the transposition of the European directive 2013/59/Euratom, the French Labour Code has been modified. These modifications impact the risks management for the operators of industries involving NORM. Considering the questions of these operators for the implementation of the Labour Code in the Health, Safety and Environment (HSE) policy of their plants, it would be necessary to develop some guidelines documents, and even regulatory documents (as ministerial order), to help them.

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Monitoring of NORM in secondary raw materials from the non-ferrous metallurgy in Belgium

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Abstract. Non-ferrous metallurgy in Belgium uses a large range of secondary raw materials, some of which may show an enhanced concentration in natural nuclides. Identifying and characterizing the raw materials which may be of concern from a radiation protection point of view is often a challenge and requires in most cases the development of a cost-efficient screening approach.
Non-ferrous metallurgy has been included since 2012 in the "positive list" of NORM sectors subject to notification to the Belgian radiation protection authority (FANC): a short overview is given of the different types of non-ferrous metallurgy in Belgium (production of Sn, Pb, Co, Zn, noble metals,… through both pyro- and hydrometallurgical processes), of the type of secondary raw materials involved and of their radiological characteristics. Challenges regarding radiological screening and analysis (such as detection and characterization of enhanced Pb-210 and Po-210 activity concentration) will be discussed.
From a regulatory perspective, this large range of exposure circumstances exemplifies the need of a graded-approach to the issue.

KEYWORDS: NORM industry, non-ferrous metal, secondary raw materials, screening measurement.

1. INTRODUCTION

Extraction of the non-ferrous metals tin, lead and copper is cited as a potential NORM industry in the indicative list of annex VI of the 2013/59/euratom directive (EU BSS) [1] as well as in IAEA Safety Report N°49 [2] (which also explicitly includes zinc and aluminium). These activities however cover a wide range of practices ranging from direct metal extraction from ores or concentrates to recycling or metal extraction from a large set of secondary raw materials. An overview of the non-ferrous metallurgy in Europe, its processes, its raw materials, etc. has recently been published by the Joint Research Centre of the European Commission [3].

This variety of processes and raw materials constitutes challenges for the identification of NORM issues in these sectors, as it is often difficult to identify beforehand which processes or raw material may or may not be of concern. In the processing of secondary raw materials, an additional challenge is due to the general absence of secular equilibrium in the materials, making also the identification of the most relevant nuclides challenging.

2. OVERVIEW OF NON-FERROUS METAL INDUSTRY IN BELGIUM

Since 2012, FANC has included all non-ferrous metal production activities in its "positive" list of NORM industries. These industries must introduce a declaration to FANC which allows identifying potential NORM issue and assessing the risk of exposure for the workers and the public. In case risks of exposure to NORM are identified, FANC may impose "corrective measures" in order to make sure that the exposure stays below the level of 1 mSv/a. Details on the Belgian regulatory framework for NORM activities and residues may be found e.g. in [4].

Nine non-ferrous metal companies have introduced a declaration to FANC. The range of metals produced is large; when some of them only produces a few main metals, some other produces a large spectrum of metals: next to the most common, such as zinc, lead, copper or tin, other substances like chrome, bismuth, antimony, molybdenum, cobalt, germanium or noble metals are also produced. Different processes are being used, both pyro-metallurgical and hydrometallurgical. Several companies combine pyro- and hydro-metallurgical processes depending on the type of metal produced. The raw material may be processed thermally in different steps, including e.g. roasting or smelting, followed by chemical or electrolytic separation and purification. In hydrometallurgical

processes, the materials are essentially chemically processed through reaction with acid and a set of other chemical reactions.

After review of their raw materials, processes and residues, NORM issues were deemed to be significant in three of these companies and FANC imposed them some corrective measures. These measures consisted essentially in the obligation of implementing a measurement procedure for their raw materials and relevant residues and to report the results to FANC. How these companies tackle the measurements procedure will be described in the next section.

Table 1 gives an overview of some raw materials presenting an enhanced concentration of natural nuclides. The main nuclides of concern in these materials and the range of their activity concentration are also indicated. It must be noted that these examples are not necessarily representative for the majority of raw materials used in the process. Generally, these specific raw materials are mixed with other materials and the impact of their processing on the end-products or on the residues is in most cases limited.

Table 1: some raw materials used in non-ferrous metal extraction and their activity concentration

	Raw material	**Main nuclide of concern**	**Range of activity concentration (kBq/kg)**
Sn production	Cassiterite	U-238sec Th-232sec	Up to 50
Sn/Pb production	Sn/Pb ingot from primay tin extraction	Pb-210 Po-210	Up to 600
Co production	Co concentrate	U-238 (without progenies)	0.1 up to 10
Cu production	Copper cement (from Zn or Co production)	U-238 (without progenies)	1 up to 50
Zn and Pb production	Flue dust from primary Zn production	(Pb-210)	Up to 0.6
		Po-210	Up to 4
	Residues from flue dust leaching	Pb-210	15
		Po-210	80

In many cases, the residues of one process will be the raw material of another: e.g. copper cement may be produced by zinc or cobalt production factories as a result of the cementation of a leaching solution and will be sold as a raw material for copper production.

In the case of cobalt production, uranium is present in the raw material coming from mining operation. Due to the pretreatment of the ores, the progenies of uranium are not present anymore. Moreover, the concentration of uranium displays a significant variability, from less than 10 ppm up to 600 ppm. In the production process, the uranium precipitates into an iron residue with an activity concentration which in some cases may reach 25 Bq/g U-238. This residue needs then to be disposed of on a landfill authorized by FANC for the acceptance of NORM residues.

Some of the non-ferrous metal production companies also face issues regarding orphan sources in their raw materials. Scrap non-ferrous metals may constitute a significant part of their raw materials or they may be confronted to the consequences of the melting of an orphan source by one of their suppliers of secondary raw materials (contaminating the resulting flue dust). Several cases of contamination of secondary raw material, such as flue dust, with Cs-137 have been reported.

3. LEGACIES RELATED TO NON-FERROUS METAL EXTRACTION

Metal extraction in the past also led to NORM legacies. Radium extraction is an obvious example but is generally not considered as a NORM activity. The issue of legacies from radium industry has been thoroughly described elsewhere (e.g. in [6]). Ferro-niobium extraction can also lead to an enhanced concentration of Th-232 and to a lesser extent of U-238 in the resulting slags [7]. Slags from zinc production or so-called zinc ashes have been landfilled or used as backfilling material in road

construction. In one sample analysed by FANC, the activity concentration amounted to 120 Bq/kg of U-238sec. In the Netherlands, radioactivity had been used as a proxy to identify the presence of zinc slags in the basement of some roads through on-field gamma radiation measurement [8].

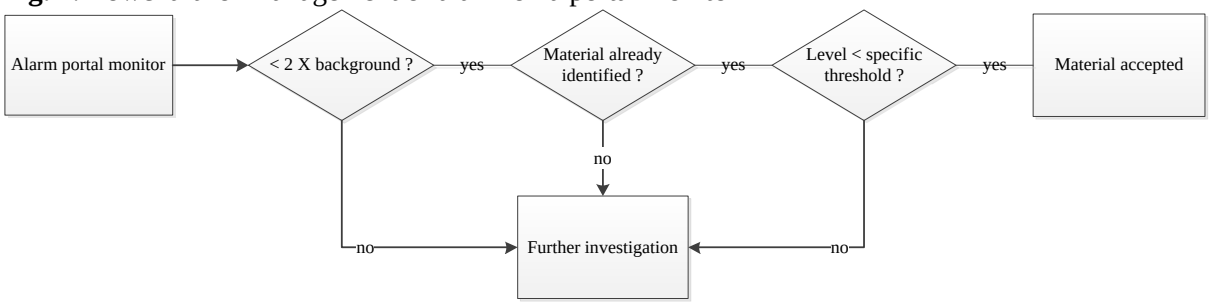
4. MONITORING OF RAW MATERIALS

4.2 Use of portal monitors

Most of non-ferrous metal production companies also recycle scrap metal. They are generally well aware of the risks related to orphan sources and have installed portal monitors to screen their raw materials. The installation of a portal monitor is compulsory in Belgium for all companies recycling at least 25 000 tons of scrap metal. Procedures have been published by FANC regarding the actions which have to be undertaken in case of detection of orphan sources.

Although the primary goal of a portal monitor is the detection of sources, they may also be used as a screening instrument for the monitoring of NORM in raw materials. The alarm level of most portal monitors is generally set to a few times the standard deviation of background level. This alarm level is exceeded for most common NORM material, including many materials which are out of regulatory concern (i.e. materials for which the activity concentration, although higher than background levels in Belgian soil, are below exemption levels: for instance alumina refractories or potassium compounds). For raw materials unlikely to contain any orphan source, an action level is defined which is generally equal to two times the normal background. A sample will be taken and analysed only if this action level is exceeded. Some companies also defined material-specific thresholds derived from experience. Exceeding this threshold for the material in question is considered as an anomaly; sampling and further characterization will be undertaken before processing the material. This procedure of routine control is summarized in Fig. 1.

Fig. 1: flowchart for management of alarm on a portal monitor



4.3 Screening measurement in the prospection phase

Portal monitors give an indication of the radiation level on materials which have already been accepted for production by the metal processing company. That information is not available when the company makes the initial characterization in the phase of prospection and selection of their raw materials. The procedures for controlling radioactivity in samples during the prospective phase vary from one operator to another. In one of the company, all samples are measured with hand-held instruments (contamination monitor). When the counts in beta/gamma or alpha signal exceed two times the background, the sample is often analysed in laboratory with gamma or/and alpha spectrometry. A simple gamma counter is generally not sufficient to identify materials possibly of concern as in several cases Pb-210 and Po-210 are the most relevant contaminants. For such material where Pb-210 and Po-210 are the dominant nuclides, a gamma spectrometry will not provide a complete characterization of the material: as can be seen from Table 1, Po-210 may sometimes be one order of magnitude higher than Pb-210 activity concentration.

4.4 Additional issues

The significant time necessary to carry on radiometric analysis often conflicts with commercial imperatives where the decision to buy a given material generally needs to be taken quite quickly.

The world-wide character of the trade of (secondary) raw materials is an additional challenge: suppliers of raw materials are located all over the world, from Mexico to Europe, from Africa to Australia. Standards, regulations, measurement protocols, awareness of NORM issues greatly vary and communication and transfer of information between processing company and supplier may be challenging.

5. CONCLUSIONS

Most of non-ferrous metal extraction factories process a large range of secondary raw materials, the residue of one being often the raw material of the other. Due to the variety of processes and raw materials involved, it is generally difficult to predict beforehand which specific raw material will be of concern from radiation protection point of view and which nuclide from the natural decay chains will be the most prominent. As a complete and systematic radiometric characterization of each raw material is not cost-efficient in the current circumstances, screening methods are generally applied to identify potential material of concern. They generally involve a combination of portal monitor, hand-held and laboratory measurement. Although progress has been made e.g. in the context of the recent MetroNORM project [9], screening identification of enhanced Pb-210 and Po-210 activity concentration in raw materials remains in many cases a challenge.

6. REFERENCES

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Application of an Artificial Neural Network for evaluation of activity concentration exemption limits in NORM industry by gamma-ray spectrometry

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Abstract. Naturally occurring radionuclides like ^{40}K and the decay products of the primordial radionuclides ^{232}Th, ^{235}U and ^{238}U are present in many natural resources. Naturally occurring radioactive materials (NORM) containing these radionuclides are exploited industrially and often exceed the exemption limits of the activity concentration. These industrial activities generate a significant portion of waste, possibly enhanceing the potential of exposure of workers and the public and the management and deposition of material above the exemption limit is very costly. The European Metrology Research Project MetroNORM focuses on creating traceable, accurate, and standardised measurement methods, reference materials and systems for application in the concerned industries. The main problem with NORM lies in the variety of densities and compositions of the materials. NORM emits many (interfereing) gamma-rays of different energies that have to be analysed by an expert. An alternative way to approach this problem is the use of artificial neural networks (ANNs). ANNs are mathematical software tools that emulate the way the human brain works. They are trained, tested and validated using sample datasets and can generalise the "knowledge" gained from the content of the training set, applying it to new problems. This can be viewed as a new calibration tool where no expert knowledge of gamma-ray spectrometry is needed by the end-user. In this work an ANN was created in the frame of MetroNORM that is able to decide from the input data of a raw gamma-ray spectrum if the activity concentrations in a sample are above or below the exemption limits. Six NORM reference materials have been analysed. To widen the applicability of the algorithm, a set of artificial gamma-ray spectra with varying densities and activity concentrations and material compositions have been created by Monte Carlo simulation and used in the training, testing and validation of the ANN.

KEYWORDS: artificial neural network, exemption limit, NORM, Monte Carlo, gamma-spectrometry

1 INTRODUCTION

One of the main applications and challenges of gamma-ray spectrometry and especially when dealing with natural radioactivity, is to determine if a certain material is above or below the exemption limit stated in the national legislation. With the latest Basic Safety Standards Directive entered into force on 6 February 2014 these limits are 1 Bq/g for all natural radionuclides with the exception of ^{40}K (10 Bq/g) (Council of the European Union, 2013). Naturally occurring radioactive materials (NORM) containing these radionuclides are exploited by industrial endeavours and often exceed the exemption limits of the activity concentration for radionuclides of the U and Th series, depending on the mineral composition and geological origin. Industrial activities are generating a significant portion of waste and can enhance the potential of exposure of workers and the public.

The aim of this work was to create an artificial neural network (ANN) that is able to decide from the data of a raw gamma spectrum if a NORM material is above or below the exemption limit, avoiding the rather complex analysis associated to spectral deconvolution. The main problem with analysing NORM lies in the variety of densities and compositions of the materials. NORM emits many (interfering) gamma-rays of different energies that have to be analysed by an expert.

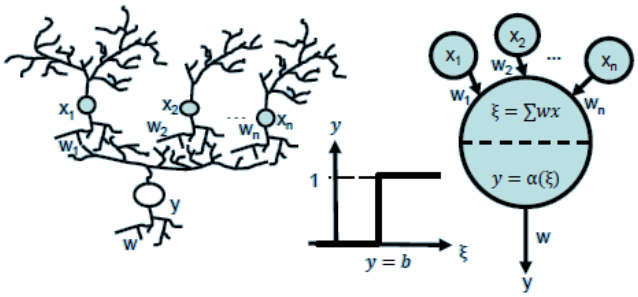
The evaluation of activity content in a sample also requires the detector to be properly calibrated in terms of energy and efficiency response. The detection efficiency highly depends on source-to-detector and sample geometry, amount and composition of sample material and energy of the gamma-rays to be measured. Self-attenuation of the gamma-rays due to the sample material has to be

considered. The detection efficiency for each sample has to be determined. This can be done with a standard source – a source with the same geometry and material as the sample that is to be measured, but with a traceably known activity concentration. This is a time-consuming process that in most cases requires a large number of standard sources or is only feasible for a small number of sample situations. Therefore, in many cases, mathematical models are used to calculate the efficiency of the detector. With mathematical models a large variety of sample situations can be used. The use of Artificial Neural Networks can be seen as an alternative way of calibrating the detector.

2 ARTIFICIAL NEURAL NETWORKS (ANNS)

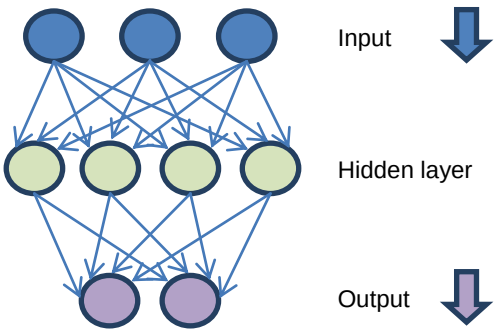
ANNs try to imitate the way the human brain works. In the human brain a number of biological neurons each generate a signal of an intensity x and a synaptic feeding strength *w*. These signals feed into a neuron with a threshold b using dendrites and axons. If the product of *x* and *w* is below the threshold, the neuron does not recognize the input. If the product is above the threshold, the neuron computes the inputs and generates a signal *y* that can again be an input to another neuron. Fig. 1 illustrates the process [1].

Figure 1: Basic processes in human brain.



An ANN consists of a number of nodes that represent the computing units (or neurons), connections (axons and dendrites) between those nodes, connection weights (synapses) and thresholds (activity in the soma). A non-linear transfer function, often a sigmoidal function, is applied to the weighted sums of each neuron. Generally speaking, an ANN consists of an input and an output layer and can also contain one or more hidden layers. Networks with no hidden layer are only able to perform linear tasks [2] while a second hidden layer is only necessary for discontinuous problems [3]. Depending on the problem, an even greater number of hidden layers may be necessary. Fig. 2 shows a schematic of an ANN.

Figure 2: ANN schematic.



The number of input neurons is defined by the number of input parameters and the number of output neurons reflects the number of output variables. The optimal number of neurons in the hidden layer has to be obtained. This is usually done by employing a trial-and-error method. Each connection comes with a connection weight that signifies the importance of the input. The network is trained by providing it with a number of inputs and the corresponding outputs. Unless reliable a-priori

information is available, the network algorithm starts out with random connection weights that are changed after each training cycle to reflect the wanted output. Literature research shows that the most commonly used algorithm to adapt the weights is the backpropagation algorithm. This algorithm uses the difference between the ANN solution and the actual solution of the training example that is provided for the training process to change the connection weights. ANNs trained in that style are capable of learning and can apply their "knowledge" to unknown situations. The main advantage of the network is its ability to generalize and handle imprecise and noisy information. This process can be considered as a robust alternative to a classical calibration method.

ANNs come in different shapes and many different kinds of ANNs exist. They are classified by their purpose (e.g. solving classification problems, forecasting, etc.), learning algorithm and other characteristics of their use but for working with all kinds of ANNs there are three major steps to follow:

- *Training:* The training phase consists of designing and building the neural network as well as providing it the relevant training data. A set of training examples consists of the input data as well as the respective output or target data. With the use of the chosen training algorithm the connection weights and biases are adjusted until the desired output is reached.
- *Validation:* This phase is used to evaluate if the network is correctly trained and working properly. Additionally, it is used to minimize overfitting. An over fitted network loses the ability to generalize the data. If the accuracy over the training data set increases, but the accuracy over the validation data set stays the same or decreases, the network is overfitting and training should stop.
- *Testing:* This phase is to check if the output is correct and to evaluate predictive power of the ANN.

3 MATERIALS & METHODS

3.1 ANN specifications

After examining the alternatives found in literature, it was decided to create an ANN able to predict if a material is below or above the exemption limit for a predefined list of radionuclides and materials relevant to NORM. This restriction is necessary as the number of materials available is limited and the aim of this work is to study the feasibility of using ANNs in this context. A number of seven materials available at CIEMAT where used to train the ANN. Five of those materials come from the JRP MetroNORM and two are reference materials available at CIEMAT (Table 1). The spectra are analysed for six radionuclides typically found in NORM materials and representing the three naturally occurring decay chains. Although the nuclides considered for analysis emit several gamma lines, it was decided to analyse a subset of those shown in Table 2 as they are free from interferences with other lines.

Table 1: List of NORM reference materials used to create the sample materials.

CIEMAT	**MetroNORM**
Phosphogypsum Huelva	Phosphogypsum (D1.2.2)
Ilmenite Huelva	Tuff 1 (D1.2.2)
	Tuff 2 (D1.3.2)
	Sand (D1.3.2)
	TiO_2 (D1.3.2)

Table 2: List of radionuclides and gamma lines analysed by the ANN.

Radionuclide	Gamma-ray energies (keV)		
^{210}Pb	46,65		
^{234}Th	63,31	92,59 *	
^{235}U	143,77	163,36	
^{212}Pb	238,63	300,09	
^{214}Pb	242,00	295,22	351,93
^{228}Ac	911,20	968,96	

*combination of 2 lines that are not well separated:
92.38 keV and 92.80 keV

All materials have been analysed at CIEMAT's chemical laboratories. The information provided by the chemical analysis was used to produce artificial spectra of the available materials with the Monte Carlo code PENELOPE v. 2014 and CIEMAT's add-on PENNUC. The artificial spectra differ from the original gamma-ray spectra in density of the sample, radionuclide composition and activity concentration. These spectra are used in order to cover a wider range of experimental conditions and to provide the network with more training examples. In order to create an ANN that correctly predicts the output, representative samples have to be used as training input. This requires intensive study of equilibrium and disequilibrium situations and careful sample preparation.

3.2 Measurement setup

The measurement system used for this project is based on a Canberra Industries GX4020 extended-range coaxial detector with 45.4 % relative efficiency and a carbon-epoxy window. This kind of detectors can extend the usual energy range of Ge detectors down to a few keV. The electronic setup includes a high-voltage power supply from BERTAN, preamplifier, spectroscopy amplifier and pulse generator from CANBERRA and a successive-approximation analog-to-digital converter from SILENA driven by a PC. A prismatic shielding structure of about 80 x 80 x 80 cm surrounds the detector. It is composed by layers of Pb (5 cm), Cd (3 mm) and Cu (1.5 mm) (V. Peyres, E. García-Toraño, 2007). Fig. 3 shows the detector system. The software used for spectrum acquisition is SILENA International SpA EMCA2000 MCA Emulation Software (2000). The software used to calculate the peak area is a non-linear code (GRILS) included in the GANAAS [4] package, which is freely distributed by the International Atomic Energy Agency.

Figure 3: Detector and cooling system at CIEMAT gamma laboratory.

Once the gamma spectrum has been acquired, the activity of a sample can be calculated using the following formula:

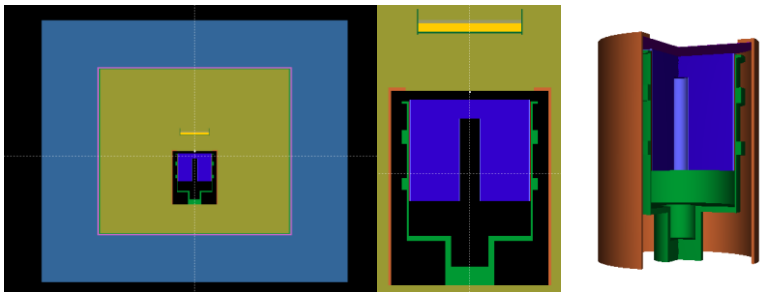
$$A = \frac{N}{t\,\varepsilon(E)\,p_\gamma}\,f_c\,f_a \tag{1}$$

A	sample activity	$\varepsilon(E)$	efficiency at energy E
N	counts in the peak area	p_γ	photon emission probability
t	measurement time	f_a	pile up correction factor
f_c	coincidence summing correction factor		

3.3 Monte Carlo calculations

In order to cover a wider range of real conditions, make up for the limited number of real sample material and make the ANN applicable to a greater number of situations, a large number of artificial spectra have been created using the Monte Carlo code PENELOPE v.2014 [5] with CIEMAT's add-on PENNUC [6]. In accordance with literature values, these spectra vary significantly in density and activity concentration of the analysed radionuclides and are used as training material for the ANN. With PENNUC the simulation process involves all particles in a cascade, whereas PENELOPE on its own simulates each particle separately. Therefore, coincidence summing is not separately calculated, but is an integral part of the whole detection efficiency calculation. Fig. 4 shows the detector model.

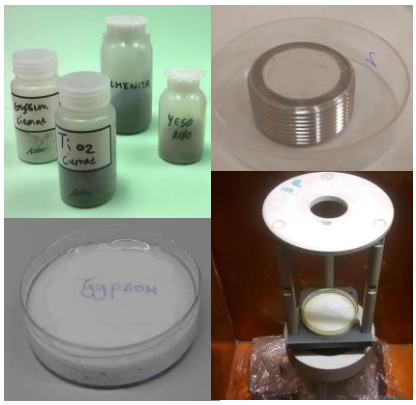
Figure 4: 2D and 3D virtual detector images of CIEMAT detector (PENELOPE).



3.4 Samples

The samples used as input data for the ANN are samples from MetroNORM Work Package 1 as well as NORM samples available at CIEMAT. The samples were carefully weighted and prepared using polypropylene containers as can be seen below. After equilibrium had been reached (21 days) the samples were measured on CIEMAT's GX4020 detector. Fig. 5 shows the samples at different stages in the preparation and measurement cycle.

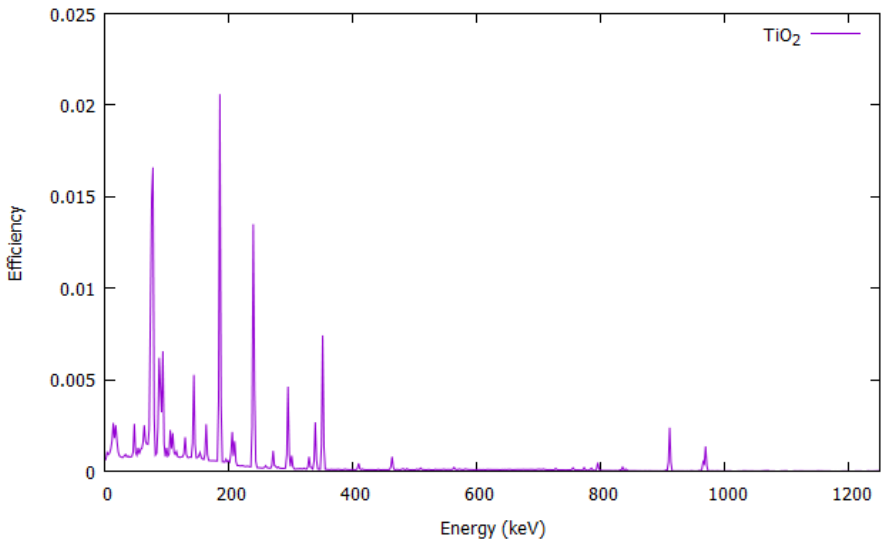
Figure 5: Samples at various stages.



The samples were analysed for major elements and trace elements by Wavelength dispersive X-ray fluorescence spectrometry (XRF) using a PANAlytical AXIOS automated XRF spectrometer with Rh radiation. The x-ray diffraction (XRD) data were collected using a PANalyticalX'Pert PRO diffractometer operating in θ-θ configuration, with Cu Kα radiation. The data was collected from 20-120º 2θ (θ…Bragg angle). Elements were determined by simultaneous Inductively Coupled Plasma

Optical Emission Spectrometer (ICP-OES), model Varian (now Agilent) 735-ES in a radially viewed configuration and a VistaChip image-mapped Charged Coupled Device (CCD) detector. The determination of concentration was performed using an Inductively Coupled Plasma Mass Spectrometer (ICP-MS) Thermo i CAP Q (Thermo Scientific, Bremen, Germany) equipped with a high-performance quadrupole analyser, He KED collision cell, and electron multiplier detector. A LECO CS-244 elemental analyser was used for determining carbon content. The amount of carbon dioxide was measured by an infrared detection method (Method ASTM 415.1, EPA-600/4-79-020). Total uranium analysis was performed by laser kinetic phosphorimetry. The kinetic phosphorimetry measurements were carried out using the kinetic phosphorescence analyser KPA-11 (Chemchek).
To gain significant results, the real materials have been measured for 200000 s on the detector while a number of 5.00E+08 showers have been simulated for each radionuclide and density, amounting to a total number of 126 simulations. After separately calculating the efficiency for each radionuclide, material and density using PENELOPE and PENNUC, the spectra are convoluted. This means, that the calculated data is convoluted with a Gaussian curve whose width is a function of energy in order to reproduce the system's response and include the electronic noise. Figure 6 shows the simulated spectrum after convolution.

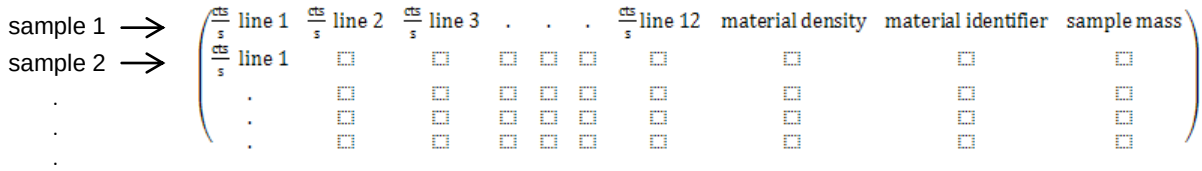
Figure 6: Exemplary PENELOPE/PENNUC simulation data for TiO_2.



Afterwards, the calculated data of each material and radionuclide is multiplied by the target activity and combined in order to gain one synthetic spectrum per material, density and activity concentration that includes all radionuclides as in the real spectrum. This gives a spectrum containing the counts per second for each energy bin according to equation (1). Correction factors are included in the efficiency calculated by PENELOPE and PENNUC. In this way a total number of 635 artificial spectra with varying distributions of activity concentration have been generated as training input to the ANN. To generate the artificial spectra and train the ANN, the software package Matlab has been used.

The structure of the input is a matrix (635 x 15) where the rows represent the sample and the columns represent the individual lines to be analysed, followed by information on density, material and sample mass.
Figure Fig. 7 shows a schematic of the input matrix.

Figure 7: Schematic of input matrix.



The ANN's output layer consists of 12 nodes, each one corresponding to one analysed line. After studying possible scenarios of disequilibrium, the artificial samples for both equilibrium and disequilibrium state were classified using five categories of activity content. This means, that samples were produced to correspond to the activities shown in Table 3, therefore minimizing the total number of samples and relying on the power of the ANN to intrapolate to the missing data. These categories are also used to classify the output of the ANN. Category 1 corresponds to an activity well below the exemption limit, while categories 2 and 3 correspond to an activity a little below or of 1 Bq/g (the exemption limit), respectively. Categories 4 and 5 correspond to activities a little above and well above the exemption limit.

Table 3: Activity categories assigned in the sample preparation stage.

Activity	Category
0,1 Bq/g	1
0,7 Bq/g	2
1 Bq/g	3
1,2 Bq/g	4
20 Bq/g	5

4 RESULTS

The best results were obtained using an ANN with 15 input nodes, one hidden layer, 31 hidden neurons and 12 output nodes using a backpropagation algorithm and a sigmoidal transfer function based on a logarithm. Introduction of the sample weight as an input parameter caused regression to get significantly better. The output of the ANN agrees with the manually assigned activity categories. Training the ANN for 91 epochs and then retraining with the same training data resulted in an ANN with an overall regression factor of 0.9975. Good convergence of the output can be observed. Fig. 8 shows the training plots.

Figure 8: Performance, regression and training state plots after retraining.

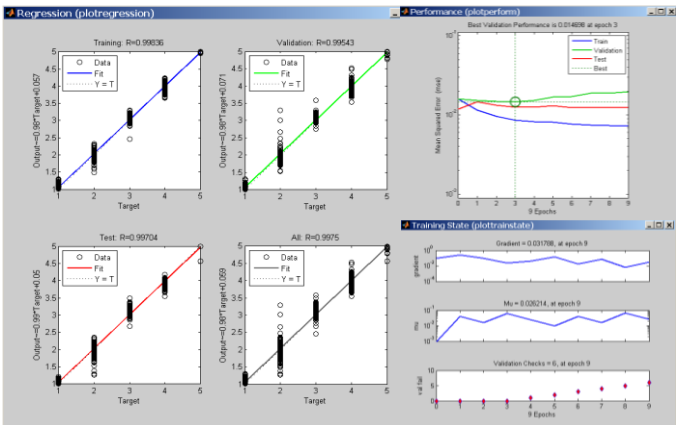


Table 4 shows the results of the final ANN when presented with the unfamiliar testing samples.

Table 4: ANN output for each of the six testing materials (exact ANN output).

Radio-nuclide	γ-ray energy (keV)	TiO_2	Sand	Phosphog. Huelva	Phosphog. MetroNORM	Ilmenite	Tuff
^{210}Pb	46.65	4.129	1.1106	2.3224	1.1028	1.0873	1.4027
^{234}Th	63.31	1.0005	1.1162	1.2007	1.0976	1.0721	1.576
	92.59*	1.0006	1.1165	1.1992	1.0967	1.072	1.5747
^{235}U	143.77	1	1.0843	1.2183	1.0696	1.0565	1.1353
	163.36	1	1.0841	1.2213	1.0701	1.0562	1.1353

Radio-nuclide	γ-ray energy (keV)	TiO_2	Sand	Phosphog. Huelva	Phosphog. MetroNORM	Ilmenite	Tuff
^{212}Pb	238.63	5	1.0487	1.0569	1.0966	1.2138	1.2004
	300.09	5	1.1069	4.1687	1.1735	1.0686	1.2905
^{214}Pb	242.00	5	1.1106	4.1404	1.2009	1.0744	1.3819
	295.22	5	1.0478	1.0586	1.1147	1.2061	1.2637
	351.93	5	1.1073	4.1697	1.1724	1.0694	1.2908
^{228}Ac	911.20	5	1.0458	1.0774	1.1386	1.2256	1.2389
	968.96	5	1.0459	1.0774	1.1386	1.2243	1.2399

*combination of 2 lines that are not well separated: 92.38 keV and 92.80 keV

5 CONCLUSION

This study was undertaken to find out if ANNs can be applied to the problem of determining if the activity concentration in a sample is above or below the exemption limit of 1 Bq/g. In the course of this work it was shown, that the ANN was able to correctly classify all of the testing materials and ANNs are well-suited to carry out this task.

For specialized industries where only one material has to be analysed and a large number of sample spectra are available, the authors propose the use of an ANN specifically trained to that purpose without the use of other materials.

The biggest limiting factor for the use of ANNs is the availability of real sample material. In the course of this work this problem has been sidestepped by calculating artificial spectra from Monte Carlo simulations but this, in turn, necessitates the complicated and time-consuming study of disequilibrium situations. It is necessary to note that this constraint only applies to the creation of training material and of the ANN itself, not the usability of the ANN.

For the end-user only a gamma spectrum in ASCII format and no specialized knowledge whatsoever in the field of gamma-ray spectrometry is required.

6 ACKNOWLEDGEMENTS

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A study of behavior of radon-222 and carbon dioxide (CO_2) in soil air (II)

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Abstract. Using formulae derived on the assumption that the flow of CO_2 is accompanied by ^{222}Rn in a region of high soil CO_2 production (present assumption), concentration profiles of ^{222}Rn in soil air are calculated for various values of tortuosity and depth. Flux densities of CO_2 at the soil surface (soil surface CO_2 fluxes) are calculated for values of the tortuosity obtained from the concentration profiles of ^{222}Rn in soil air. Flux density of ^{222}Rn at the soil surface (soil surface ^{222}Rn flux) and soil surface CO_2 fluxes are also calculated based on the assumption that ^{222}Rn and CO_2 behave separately in soil air in all unsaturated soil zone (conventional assumption). Since the above-mentioned formulae contain the terms representing the measured soil surface ^{222}Rn flux, and since the soil surface ^{222}Rn flux calculated on the conventional assumption is about 78 % of the soil surface ^{222}Rn flux measured by another investigator, the present assumption might be more reasonable than the conventional assumption. The soil surface CO_2 fluxes calculated on the present assumption are closer to that measured by the investigator than those calculated on the conventional assumption. Furthermore, when the flow of CO_2 is accompanied by all ^{222}Rn atoms in soil air, it is found that soil surface ^{222}Rn flux is about 14 % more than that based on the conventional assumption, and that soil surface CO_2 flux is about 30 % more than that based on the conventional assumption. From the results presented here, it is suggested that the flow of CO_2 might be accompanied by ^{222}Rn in a region of high soil CO_2 production. Experimentally verified, this may contribute to making more exact estimates of the flux density of ^{222}Rn due to the NORM residues and the soil surface CO_2 flux by ^{222}Rn calibrated method.

KEYWORDS: Flux density of ^{222}Rn; Flux density of CO_2; Diffusion of ^{222}Rn in soil air; CO_2 flow in soil air; Concentration profile of ^{222}Rn in soil air.

1 INTRODUCTION

Considering new findings regarding radon [e.g.: 1-3] and the results of the IAEA's projects on improving environmental assessment and remediation, in 2013 the IAEA published Technical Reports Series No. 474 [4], Measurement and Calculation of Radon Releases from NORM Residues, which is a revision of Technical Reports Series No. 333 [5]. In this report, radon releases from repositories of NORM residues are described in more detail. On the other hand, carbon dioxide (CO_2) is one of the main greenhouse gases and soils store two to three times as much carbon as the atmosphere. Therefore, many researchers have been interested in the behavior of CO_2 under the ground in view of global warming [e.g.: 6-8]. Being an almost chemically inert gas with a convenient half-life (3.82 days), ^{222}Rn has been used as a tracer for studying production and transport of CO_2 in soil [e.g.: 9-11]. In both cases, it is assumed that ^{222}Rn and CO_2 behave independently in soil air in all unsaturated soil zone. Since a large amount of CO_2 is produced in a region where soil microorganisms and plant roots exist densely, the flow of CO_2 might be accompanied by ^{222}Rn there. The objective of this study is to examine in detail the assumption that the flow of CO_2 is accompanied by ^{222}Rn in such a region.

We use the following terminology. Flux density of ^{222}Rn in soil, flux density of ^{222}Rn at the soil surface, flux density of CO_2 in soil, and flux density of CO_2 at the soil surface will hereafter be designated as, in order, ^{222}Rn flux, soil surface ^{222}Rn flux, CO_2 flux, and soil surface CO_2 flux. Also, concentration profile

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of ^{222}Rn in soil air, concentration profile of CO_2 in soil air, and flux density profile of CO_2 in soil will hereafter be designated as, in order, concentration profile of ^{222}Rn, concentration profile of CO_2, and flux profile of CO_2. Unless otherwise noted, concentration of ^{222}Rn and concentration of CO_2 will hereafter denote concentration of ^{222}Rn in soil air and concentration of CO_2 in soil air, respectively.

2 CONCENTRATIONS OF CO_2 AND ^{222}Rn UNDER THE GROUND

It is assumed that soil properties such as bulk dry density ϱ, air ratio n_a, tortuosity k, concentration of ^{226}Ra in solid materials A_{Ra}, and escape-to-production ratio of ^{222}Rn δ, which is the ratio of the amount of ^{222}Rn that escapes into air-filled pore space relative to the amount produced in soil at equilibrium state are homogeneous under the ground.

In horizontal homogeneity and steady state, using the boundary conditions $C_{CO_2,a}(z=0) = C_{CO_2,a,0}$ and $C_{CO_2,a}(-L > z) = C_{CO_2,a}(z = -L)$, concentration of CO_2 $C_{CO_2,a}$ (molecules m^{-3} of soil air) for uniformly distributed source strength of soil CO_2 is expressed as

$$C_{CO_2,a} = -\frac{C_{CO_2,a,uni,-\infty}}{L^2}\{(z+L)^2 - L^2\} + C_{CO_2,a,0}\,, \tag{1}$$

and using the boundary conditions $C_{CO_2,a}(z=0) = C_{CO_2,a,0}$ and $C_{CO_2,a}(z = -\infty) = C_{CO_2,a,exp,-\infty}$, $C_{CO_2,a}$ for exponentially distributed source strength of soil CO_2 is expressed as

$$C_{CO_2,a} = C_{CO_2,a,exp,-\infty}\{1 - \exp(\frac{z}{d_{CO_2}})\} + C_{CO_2,a,0}\,, \tag{2}$$

where $C_{CO_2,a,uni,-\infty}$ (molecules m^{-3} of soil air) is the concentration of CO_2 at $-L \geq z$ only due to the source strength of soil CO_2, L (m, L>0) is the depth to which soil microorganisms and plant roots exist densely, $C_{CO_2,a,exp,-\infty}$ (molecules m^{-3} of soil air) is the concentration of CO_2 at $z = -\infty$ only due to the source strength of soil CO_2, and $C_{CO_2,a,0}$ (molecules m^{-3} of soil air) is the concentration of CO_2 at the soil surface, which is equal to the concentration of CO_2 in the atmosphere at the soil surface, d_{CO_2} (m, $d_{CO_2} > 0$) is the relaxation depth for the exponentially distributed source strength of soil CO_2, and z (m) is the depth from the earth-air interface and taken as negative in the downward direction.

Using the boundary conditions $C_{Rn,a}(z = 0) = C_{Rn,a,0}$ and $C_{Rn,a}(z = -\infty) = \delta\varrho A_{Ra}/\lambda n_a$ in horizontal homogeneity and steady state, concentration of ^{222}Rn $C_{Rn,a}$ (atoms m^{-3} of soil air) for the conventional assumption is expressed as

$$C_{Rn,a} = \frac{\delta\varrho A_{Ra}}{\lambda n_a}\{1 - \exp(\sqrt{\frac{k\lambda}{D_0^{Rn}}}z)\} + C_{Rn,a,0}\exp\left(\sqrt{\frac{k\lambda}{D_0^{Rn}}}z\right). \tag{3}$$

and that for the present assumption is expressed as

$$C_{Rn,a} = \frac{\sqrt{D_0^{Rn}}\left(\frac{\delta\varrho A_{Ra}}{\lambda n_a} - C_{Rn,a,0}\right) - \frac{1}{n_a}\sqrt{\frac{k}{D_0^{CO_2}\lambda}}F_{Rn,s,0}\left\{\left(\sqrt{D_0^{Rn}} + \sqrt{D_0^{CO_2}}\right)\exp\left(\sqrt{\frac{k\lambda}{D_0^{CO_2}}}L_{CO_2}\right) - \sqrt{D_0^{Rn}}\right\}}{\sqrt{D_0^{Rn}}\left\{\exp\left(-\sqrt{\frac{k\lambda}{D_0^{CO_2}}}L_{CO_2}\right) + \exp\left(\sqrt{\frac{k\lambda}{D_0^{CO_2}}}L_{CO_2}\right) - 2\right\} - \sqrt{D_0^{CO_2}}\left\{\exp\left(-\sqrt{\frac{k\lambda}{D_0^{CO_2}}}L_{CO_2}\right) - \exp\left(\sqrt{\frac{k\lambda}{D_0^{CO_2}}}L_{CO_2}\right)\right\}}$$

$$\times\left\{\exp\left(\sqrt{\frac{k\lambda}{D_0^{CO_2}}}z\right) + \exp\left(-\sqrt{\frac{k\lambda}{D_0^{CO_2}}}z\right) - 2\right\} + \frac{1}{n_a}\sqrt{\frac{k}{D_0^{CO_2}\lambda}}F_{Rn,s,0}\left\{\exp\left(-\sqrt{\frac{k\lambda}{D_0^{CO_2}}}z\right) - 1\right\} + C_{Rn,a,0} \quad (0 \geq z \geq -L_{CO_2}), \tag{4}$$

and

$$C_{Rn,a} = [\frac{\sqrt{D_0^{Rn}}\left(\frac{\delta\varrho A_{Ra}}{\lambda n_a} - C_{Rn,a,0}\right) - \frac{1}{n_a}\sqrt{\frac{k}{D_0^{CO_2}\lambda}}F_{Rn,s,0}\left\{\left(\sqrt{D_0^{Rn}} + \sqrt{D_0^{CO_2}}\right)\exp\left(\sqrt{\frac{k\lambda}{D_0^{CO_2}}}L_{CO_2}\right) - \sqrt{D_0^{Rn}}\right\}}{\sqrt{D_0^{Rn}}\left\{\exp\left(-\sqrt{\frac{k\lambda}{D_0^{CO_2}}}L_{CO_2}\right) + \exp\left(\sqrt{\frac{k\lambda}{D_0^{CO_2}}}L_{CO_2}\right) - 2\right\} - \sqrt{D_0^{CO_2}}\left\{\exp\left(-\sqrt{\frac{k\lambda}{D_0^{CO_2}}}L_{CO_2}\right) - \exp\left(\sqrt{\frac{k\lambda}{D_0^{CO_2}}}L_{CO_2}\right)\right\}}$$

$$\times\{\exp\left(-\sqrt{\frac{k\lambda}{D_0^{CO_2}}}L_{CO_2}\right) + \exp\left(\sqrt{\frac{k\lambda}{D_0^{CO_2}}}L_{CO_2}\right) - 2\} \quad + \frac{1}{n_a}\sqrt{\frac{k}{D_0^{CO_2}\lambda}}F_{Rn,s,0}\{\exp\left(\sqrt{\frac{k\lambda}{D_0^{CO_2}}}L_{CO_2}\right) - 1\}$$

$$+ C_{Rn,a,0} - \frac{\delta\varrho A_{Ra}}{\lambda n_a}]\exp\left(\sqrt{\frac{k\lambda}{D_0^{Rn}}}L_{CO_2}\right)\exp(\sqrt{\frac{k\lambda}{D_0^{Rn}}}z) + \frac{\delta\varrho A_{Ra}}{\lambda n_a} \qquad (-L_{CO_2} \geq z \geq -\infty), \tag{5}$$

where λ (s^{-1}) is the decay constant of ^{222}Rn, $D_0^{CO_2}$ ($m^2\ s^{-1}$) is the molecular diffusion coefficient of CO_2 in air, D_0^{Rn} ($m^2\ s^{-1}$) is the molecular diffusion coefficient of ^{222}Rn in air, L_{CO_2} (m, L_{CO_2}>0) is the depth from which the flow of CO_2 is accompanied by ^{222}Rn, $C_{Rn,a,0}$ (atoms $m^{-3}$of soil air) is the concentration of ^{222}Rn at the soil surface, which is equal to the concentration of ^{222}Rn in the atmosphere at the soil surface, and $F_{Rn,s,0}$ (atoms m^{-2} of soil s^{-1}) is the soil surface ^{222}Rn flux [12].

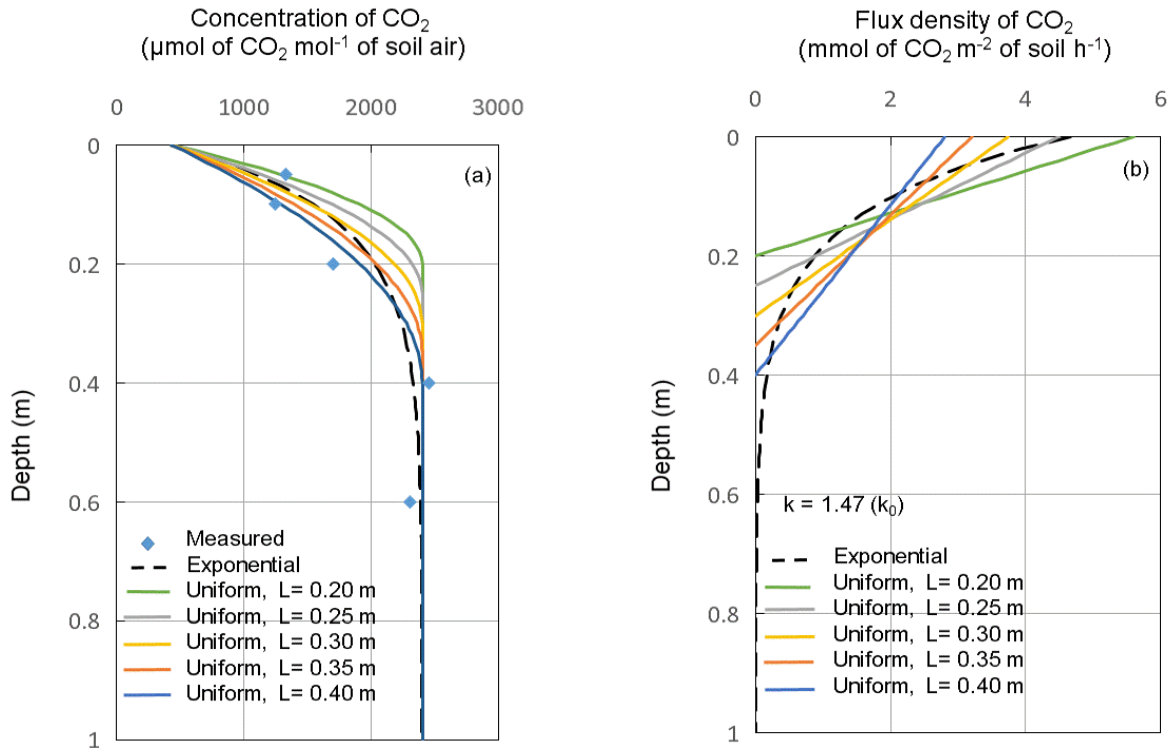
3 RESULTS AND DISCUSSION

Measured data presented here are all from Dörr and Münnich [9].

Fig. 1(a) shows concentrations of CO_2 calculated using Equation (1) and Equation (2), and measured concentrations of CO_2. From Fig. 1(a), d_{CO_2} and $C_{CO_2,a,exp,-\infty}$ are estimated to be 0.12 m and 1,973 ppm (= 88.1mmol of CO_2 m^{-3} of soil air), respectively. They are almost the same as those obtained by Dörr and Münnich [9]. Depth of uniformly distributed soil CO_2 source strength L is thought to be between 0.2 and 0.4 m.

Standard tortuosity k_0 is estimated to be 1.47 by calculating concentration of ^{222}Rn using the conventional assumption and by fitting the concentration profile of ^{222}Rn to the measured concentration profile of ^{222}Rn (see Fig. 2). This tortuosity is also almost the same as that obtained by Dörr and Münnich [9]. Fig. 1(b) shows flux profiles of CO_2 calculated using the estimated standard tortuosity and assuming the uniformly distributed source strength of soil CO_2. In this calculation, depths of 0.2, 0.25, 0.30, 0.35, and 0.40 are selected for L. A flux profile of CO_2 calculated using the estimated standard tortuosity and

Figure 1: (a) Concentration profile of CO_2 and (b) flux profile of CO_2.

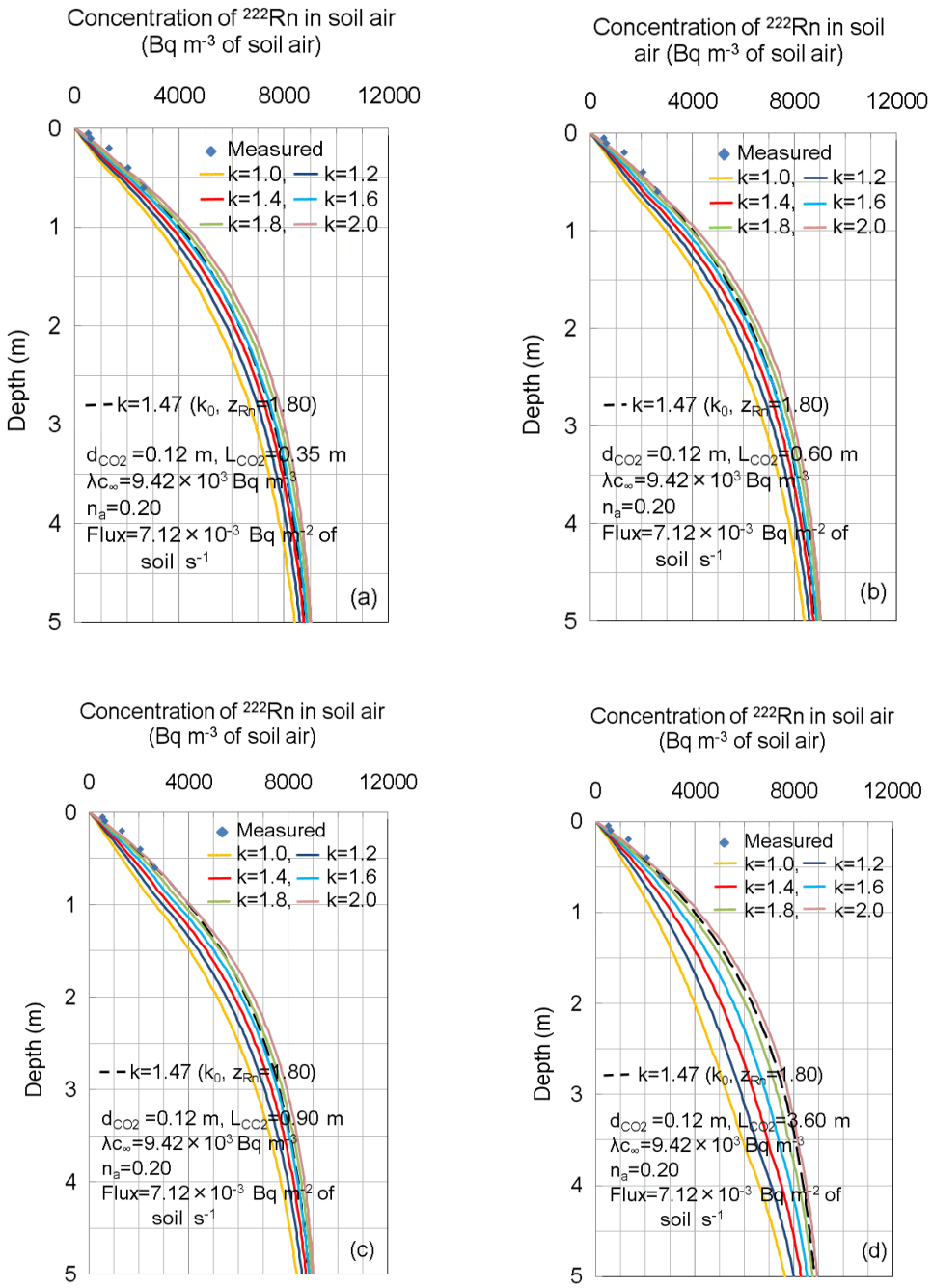


assuming exponentially distributed source strength of soil CO_2 is also shown in Fig. 1(b). We notice that the flux profiles of CO_2 calculated assuming the uniformly and exponentially distributed source strength of soil CO_2 have the same trend. It is assumed that the flow of CO_2 is accompanied by ^{222}Rn from the soil surface to the depth (L_{CO2}) where the CO_2 flux is 10 times or larger than the soil surface ^{222}Rn flux. Depths of 0.122 m (L = 0.2 m), 0.165 m (L = 0.25 m), 0.227 m (L = 0.3 m), 0.292 m (L = 0.35 m), 0.356 (L =0.4 m), 0.6 m, 0.9 m, 2.0 m, and 3.6 m are selected for L_{CO2}.

Fig. 2 shows concentration profiles of ^{222}Rn calculated assuming that the flow of CO_2 is accompanied by ^{222}Rn between the soil surface and L_{CO2}. The measured concentration profile of ^{222}Rn is also shown in each figure. In addition, a concentration profile of ^{222}Rn calculated using the conventional assumption and $\delta \varrho A_{Ra} / \lambda n_a$ estimated by Dörr and Münnich [9] is shown in each figure. In Fig. 2(a)-(d), we obtain the values of k from the concentration profile of ^{222}Rn calculated using the present assumption well fitted to that calculated using the conventional assumption. They are shown in Table 1. We find from this table that k increases with increase in L_{CO2}. This suggests that, when depth (L_{CO2}) from which $D_0^{CO_2}$ is used instead of D_0^{Rn} increases, k becomes apparently larger.

Table 1 also shows soil surface CO_2 fluxes calculated using the values of L_{CO2} and k. In the case of the exponentially distributed soil CO_2 source strength, when L_{CO2} is between 0.122 and 3.6 m, the soil surface CO_2 fluxes calculated on the present assumption are closer to the measured soil surface CO_2 flux than those calculated on the conventional assumption and estimated by Dörr and Münnich [9]. In the case of the uniformly distributed soil CO_2 source strength, when L_{CO2} is between 0.165 m (L = 0.25 m) and 0.292 m (L = 0.35 m), the soil surface CO_2 flux calculated on the present assumption are closer to the measured soil surface CO_2 flux than those calculated on the conventional assumption and estimated by Dörr and Münnich [9].

Figure 2: Concentration profile of ^{222}Rn. (a) L_{CO2} = 0.35 m, (b) L_{CO2} = 0.6 m, (c) L_{CO2} = 0.9 m, and (d) L_{CO2} = 3.6 m.



If the flow of CO_2 is accompanied by all ^{222}Rn atoms in soil air, since $(D_0^{CO_2}/D_0^{Rn})^{1/2}$ is equal to 1.14 ($D_0^{CO_2}/D_0^{Rn}$ = 1.3 at 15 ^{O}C [13]), flux densities of ^{222}Rn from the surface of bare NORM residues (e.g.: uranium mining tailings) and covered NORM residues (e.g.: uranium mill tailings) in TRS No.474 may

Table 1: Soil surface CO_2 fluxex calculated using k and L_{CO2}. Figures in parentheses indicate values relative to the measured soil surface CO_2 flux set as 100.

Measured soil surface CO_2 flux (mmol of CO_2 m^{-2} of soil h^{-1})	Calculated									
	Conventional: ^{222}Rn diffuses independently of the flow of CO_2.					Present: The flow of CO_2 is accompanied by ^{222}Rn				
	No consideration of n_a and use of measured soil surface ^{222}Rn flux	Assumed n_a and no use of measured soil surface ^{222}Rn flux						Exponentially for CO_2 n_a=0.20 for Dörr and Münnich [9]	Uniformlly for CO_2 n_a=0.20 for Dörr and Münnich [9]	
			Soil Surface CO_2 flux (mmol of CO_2 m^{-2} of soil h^{-1})							
	Soil surface CO_2 flux (mmol of CO_2 m^{-2} of soil h^{-1})	k	Exponentially for CO_2 n_a=0.2 for Dörr and Münnich [9]	Uniformly for CO_2 n_a=0.20 for Dörr and Münnich [9]	L_{CO2} (m)	k	Soil surface CO_2 flux (mmol of CO_2 m^{-2} of soil h^{-1})	L (m)	Soil surface CO_2 flux (mmol of CO_2 m^{-2} of soil h^{-1})
3.76[(a)] (100)	5.99[(b)] (159)	1.47	4.67 (124)	3.20 (85)	0.122	1.5	4.58 (122)	0.20	5.50 (146)
					0.165	1.6	4.29 (114)	0.25	4.12 (110)
					0.227	1.6	4.29 (114)	0.30	3.44 (91)
					0.292	1.6	4.29 (114)	0.35	2.95 (78)
					0.356	1.6	4.29 (114)	0.40	2.58 (69)
					0.6	1.7	4.04 (108)	- - -	- - - - -
					0.9	1.8	3.82 (102)	- - -	- - - - -
					2.0	1.9	3.62 (96)	- - -	- - - - -
					3.6	1.9	3.62 (96)	- - -	- - - - -

[(a)]After Dörr and Münnich [9]. [(b)]This value calculated using the data of the soil surface ^{222}Rn flux measured by Dörr and Münnich [9] is 6.8 % larger than that calculated by them due to rounding errors of the data.

be about 14 % larger than those calculated on the conventional assumption. Therefore, root respiration and microbial decomposition of soil organic matter which increase concentration of CO_2 may need to be added as one of factors that affect soil surface ^{222}Rn fluxes as well as soil moisture content, size and shape of soil particles and so on. Since Equation (4) contains the terms representing the measured soil surface ^{222}Rn flux, and since the soil surface ^{222}Rn flux calculated on the conventional assumption is about 78 % of the measured soil surface ^{222}Rn flux, the present assumption might be more reasonable than the conventional assumption.

Considering $D_0^{CO_2}/D_0^{Rn}$ = 1.3, soil surface CO_2 fluxes calculated using the present assumption are about 30 % larger than those calculated using the conventional assumption.

In order to verify experimentally the present assumption that the flow of CO_2 is accompanied by ^{222}Rn in a region of high soil CO_2 production, soil must be homogeneous between the soil surface and an underground depth of at least 15 m, and soil surface ^{222}Rn and CO_2 fluxes, concentration profiles of ^{222}Rn and CO_2, soil moisture content, porosity, and ^{222}Rn escape-to-production ratio of soil for estimation of saturated concentration of ^{222}Rn must be measured precisely. However, since the errors are large as to measurements of concentration of ^{222}Rn and ^{222}Rn fluxes, and since concentration of CO_2 has seasonal variation because of its dependence on plant roots and microorganisms, and since precipitation affects soil moisture content, field exercises will lack reproducibility. Therefore, we recommend laboratory exercises (e.g.: [14-16]) in order to verify experimentally the present assumption.

4 CONCLUSIONS

From the results presented here, it is suggested that the flow of CO_2 might be accompanied by ^{222}Rn in a region of high soil CO_2 production. In addition, it becomes clear that the increases in the soil surface ^{222}Rn flux and in the soil surface CO_2 flux due to this present assumption amount to 14 % of the soil surface ^{222}Rn flux and 30 % of the soil surface CO_2 flux calculated on the conventional assumption, respectively. Experimentally verified, this may contribute to making more exact estimates of flux densities of ^{222}Rn from the surface of NORM residues and soil surface CO_2 fluxes by ^{222}Rn calibrated method.

5 ACKNOWLEDGEMENT

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Nuclear Emergencies

Emergency preparedness: advancement and still open gaps in Europe

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Abstract. Notwithstanding the tremendous progress made following the Chernobyl accident, the response of Europe as a whole was far from optimal in the beginning of the Fukushima disaster. European countries decided by their own on supporting their citizens in Japan and did similar with their embassy staff members, instead of looking for a common European assessment. Information sharing of a potential source term took much time and was not coordinated in the beginning. On the other side, improvements were made in international agreements (e.g. HERCA-WENRA scheme) and important documents were published by ICRP, IAEA and EU strengthen emergency preparedness and response. Research on the other hand is one of the key drivers for advances at the European level. In particular the development of advanced decision support systems such as ARGOS and JRODOS provides harmonisation at the operational level as both systems use the same or similar simulation models. As one of both systems is installed in nearly all European countries, assessment results are now much more comparable and consistent compared to the aftermath of the Chernobyl event. To advance research, the NERIS Platform (http://www.eu-neris.net/) with now 59 participating organisations defines research needs in their strategic research agenda and identifies gaps that can be closed at the European level. In this respect, the operational community together with academia and other stakeholders define research needed to further advance emergency preparedness and response in all phases including long term rehabilitation. Finally, the presentation will highlight the results of European research projects and to which extend they were taken over by the operational community and which gaps need further attention such as the question of optimisation and proper use of the residual dose concept.

Keywords: emergency preparedness, response, decision support, gap analysis, NERIS Platform, CONFIDENCE

1 INTRODUCTION

Following the Chernobyl accident, the response in Europe was far from being optimal and partly confusing [1]. Individual countries started their own research activities to improve the situation. In parallel, the European Commission set up work in their Framework programmes to improve emergency management and response at the European level. This resulted in many projects that finally contributed to improved emergency preparedness in Europe. At present, the NERIS Platform supports this development in setting up a platform to harmonise and improve preparedness for nuclear and radiological emergency response and recovery as well as defining further research needs at the European level.

This paper will discuss some national and international initiatives demonstrating improvements in preparedness and tries to identify still open gaps. It is written more from a “research” perspective and not necessarily reflect all the operational needs. It also is limited in number of topics addressed, mainly dealing with European aspects. Chapters of this paper deal with decision support systems, national and international initiatives to improve preparedness, research needs identified by European Platforms and finally response to Fukushima and the recent Ruthenium case.

2 DECISION SUPPORT SYSTEMS

In the beginning of the 1990th, a European Framework project with the acronym RODOS “**R**eal time, **O**nline **D**ecisi**O**n **S**upport for nuclear emergencies” was funded [2] aiming to harmonise decision support in Europe. The main objectives of the project were:

- to provide a comprehensive and integrated decision support system that is generally applicable across Europe;

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- to provide a common platform or framework for incorporating the best features of existing systems and future developments;
- to provide greater transparency in the decision process as one input to improving public understanding and acceptance of emergency measures;
- to facilitate improved communication between countries of monitoring data, predictions of consequences, etc.;
- to promote, through the development and use of the system, a more coherent, consistent and harmonised response to any future accident that may affect Europe.

The system on the one hand side was developed successfully but on the other hand, its last objective was not reached. However, cooperating with the ARGOS decision support system in common model development has resulted that in particular for the late phase, the same simulation model is implemented in each system. This results in a "bottom up" harmonisation as most countries in Europe have installed either RODOS with its newest version named JRodos) and ARGOS.

3 PREPAREDNESS AT INTERNATIONAL LEVEL

In 2007, a new general guidance document was issued by the International Commission on Radiological Protection (ICRP) [4], followed by ICRP 109 and 111, dealing with the "Protection of people in emergency exposure situations" and "Radiation protection principles for the protection of individuals living in contaminated territories", respectively. The latter two documents are under revision and the consultation process will start soon. Important to note here, that the term "optimisation" plays an important role and requires further clarification, in particular with respect to its operational implementation.

The International Atomic Energy Agency (IAEA) developed also many important guidance documents that can be used for member states to improve preparedness in their countries (e.g. [5]).

The Association of the Heads of the European Radiological Protection Competent Authorities (HERCA) and the Western European Nuclear Regulators' Association published in 2014 a document aiming to harmonise emergency preparedness and response in Europe [6]. This document stressed the importance of cross-border harmonisation, in particular "adopt the principle "We do the same as the accident country" in the first hours of the accident". In addition the so called HERCA-WENRA scheme was proposed to initiate protective actions in the pre-release phase. The scheme is based on 4 questions and their related answers triggering response activities (see Table. 1). It is intended in particular for pre- and early release situations.

Table 1: The HERCA-WENRA scheme

Attribute	Value
Risk of core melt?	No, yes, unknown
Maintaining of containment integrity	No, yes, unknown
Wind direction	Steady, variable, unknown
Estimated release time	Known, unknown

With that information the following actions can be initiated

- Evacuation should be prepared up to 5 km around nuclear power plants and sheltering with Iodine Thyroid Blocking (ITB) up to 20 km
- A general strategy should be defined in order to extend evacuation up to 20 km and sheltering with ITB up to 100 km

4 NATIONAL INITIATIVES

In France, the Nuclear Safety Authority (ASN) established in 2005 a steering committee for the management of the post-accident phase of a nuclear accident or a radiological emergency (CODIRPA). From 2005 to 2012, work was performed to develop policy elements for post-accident management in the event of a nuclear accident of medium scale. A document summarising the framework was published end of 2012 [7]. This framework defines measures that might be considered in post-accidental situations all over France. Important to note here is that these doctrines were developed integrating all local and national stakeholders, international regulations and also European research projects. This document harmonises response and is key for preparedness at all levels.

In Sweden and Germany, the Swedish Radiation Safety Authority (SSM) and the German Federal Office for Radiation Protection (BfS) have developed new zones to prepare for early protective actions. Several thousands of different calculations for different source terms, different sites and different weather conditions were performed and analysed. As a consequence the German areas for pre-planning were considerably changed (see Table 2) [8]. Feedback from the population and in the mass media were generally positive stating that people felt better protected.

Table 2: Revised German Planning areas

Old	New
Central zone up to 2 km distance	**Central zone up to 5 km distance**
Middle zone up to **10** km distance	Middle zone up to **20** km distance
Outer zone up to **25** km distance	Outer zone up to **100** km distance
Far zone up to **100** km distance	**Whole Germany**

5 NERIS PLATFORM

NERIS is the European platform on preparedness for nuclear and radiological emergency response and recovery. Founded in 2010, its number of members has reached 60 in the beginning of 2018. The platform aims to

- Improving the effectiveness and coherency of current approaches to preparedness
- Identifying further development needs and prioritise them
- Improving know-how and technical expertise
- Establish a forum for dialogue and methodological development

With partners from operational emergency management, research, industry and non-governmental organisations, NERIS is an ideal platform to discuss topics that are of interest at the European level. Consequently, a Strategic Research Agenda (SRA) was developed considering all the research needs identified so far. The current one is available via the web page of the platform [9].

The SRA defines three challenge areas and 10 key topics with 27 research areas. The challenge areas are:

- Challenges in radiological impact assessment during all phases of nuclear and radiological events
- Challenges in countermeasures and countermeasure strategies in emergency and recovery, decision support and disaster informatics
- Challenges in setting-up a trans-disciplinary and inclusive framework for preparedness for emergency response and recovery

Roughly, on a yearly basis, the SRA will be updated and research priorities defined. The most recent prioritisation (2015/16) showed the following research needs

- Assessment of and communication of uncertainties (SRA 5.1)
- Robust decision-making (SRA 5.3)
- Countermeasure strategy preparedness (SRA 5.7)
- Atmospheric dispersion modelling (SRA topic 1, subtopic not specified)
- Local radio-ecological models (SRA 3.4)
- Monitoring strategies (SRA 5.9)

Some of these research needs are addressed in European research projects that started in 2017. This will be briefly discussed in the next chapter.

Under the H2020 project CONCERT "European Joint Programme for the Integration of Radiation Protection Research" [10] a gap analysis was performed beginning of 2018. All European research platforms such as MELODI (low dose), ALLIANCE (radioecology), EURADOS (dosimetry), NERIS (emergency response and recovery), EURAMED (medical) and SSH (society) participated in that activity. The objective was to identify either individual or common gaps of two or more platforms. NERIS identified common gaps with ALLIANCE on improved modelling, methods and guidance for optimisation and decision making under uncertainties and with EURADOS on improved strategies for monitoring including information from non-professional community and drones. A gap related to most platforms addressed improved modelling for internal doses after accidental situations based on environmental monitoring data and personal monitoring data.

This exercise, together with the research needs identified by each platform – other platforms mentioned above have their own SRA – help to identify gaps and provide the basis for European research calls at the Euratom level. It also helps to narrow down the research and focus it on the most important topics.

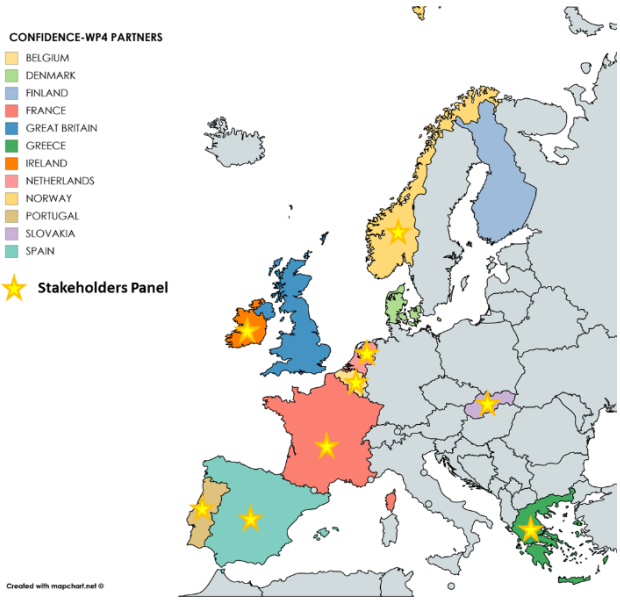
6 ONGOING EUROPEAN RESEARCH PROJECTS

In the frame of CONCERT, internal calls were initiated in 2016 resulting in the funding of two research projects dealing with decision making under uncertainties, thus addressing important research needs identified within NERIS. CONFIDENCE addresses the early and transition phase, whereas TERRITORIES is more related to the late phase (see [11] and [12]). Main activities of CONFIDENCE relate to the following topics

- Identifying and ranking the main sources of uncertainties in the early phase, and characterising and quantifying their effect on simulation results
- Developing a prototype software for the quick and efficient assessment of cancer risk to affected population to be used as an input in the overall decision making process
- Developing approaches and tools to integrate external and internal dosimetric monitoring data with simulation results to improve understanding of the radiological situation
- Improving the capabilities of radioecological models used to predict activity concentrations in foodstuffs and to better characterise, and where possible, reduce uncertainties (the element of our work programme being the focus of the next section)
- Improve the methodology to develop sensible countermeasure strategies in the transition phase by engaging national stakeholders and considering uncertainties when defining the strategies
- Identifying social and ethical issues related to uncertainty management in emergency and post-accident situations, and clarify how stakeholders at various levels deal with uncertainty in their decision making processes
- Support and improve communication of uncertainties and facilitate robust decision making taking into account the variability of the radiological situation
- Develop training courses and educational material for professionals and students related to the issues and activities addressed in CONFIDENCE

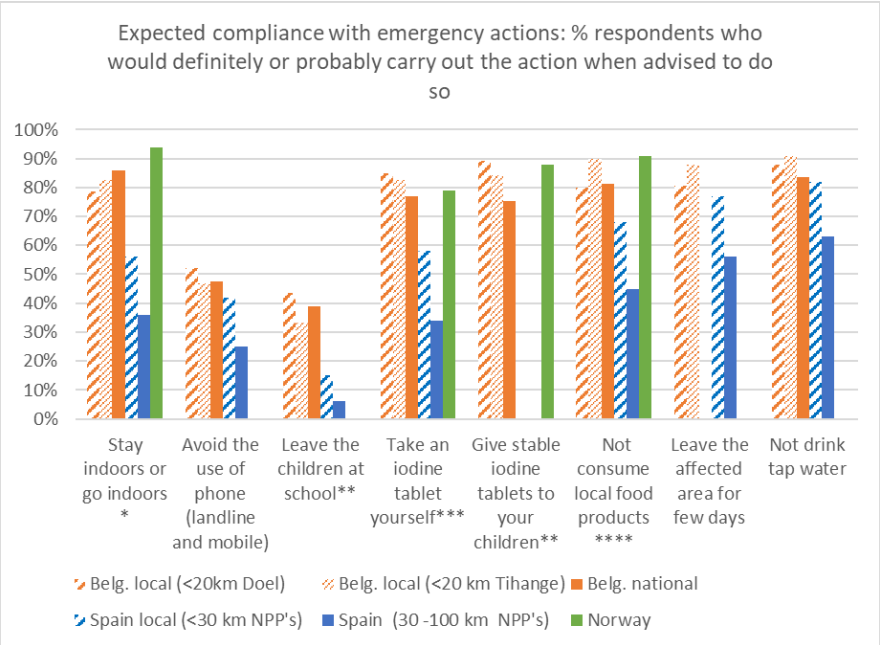
With 32 partner from 18 countries CONFIDENCE surely reaches European dimension. For example stakeholder panels on countermeasure strategies will be carried out in nine countries as shown in Figure 1.

Figure 1: Stakeholder panels on countermeasure strategies



One aspect that might require also more attention in the operational community is communication. Within CONFIDENCE, About 3000 people have been interviewed in Belgium, Spain and Norway related to compliance with advice provided by authorities in case of a nuclear accident. The participants in Spain and Belgium were further subdivided into those living close (< 20km or < 30km for Belgium and Spain, respectively) or further away from the site. Even if in most questions the compliance to the recommendations is high, some topics such as “leave children at school” or “use of phones” showed deviations. Differences between those living close and further away could be also observed (Figure 2).

Figure 2: Expected compliance with emergency actions in Belgium, Norway and Spain

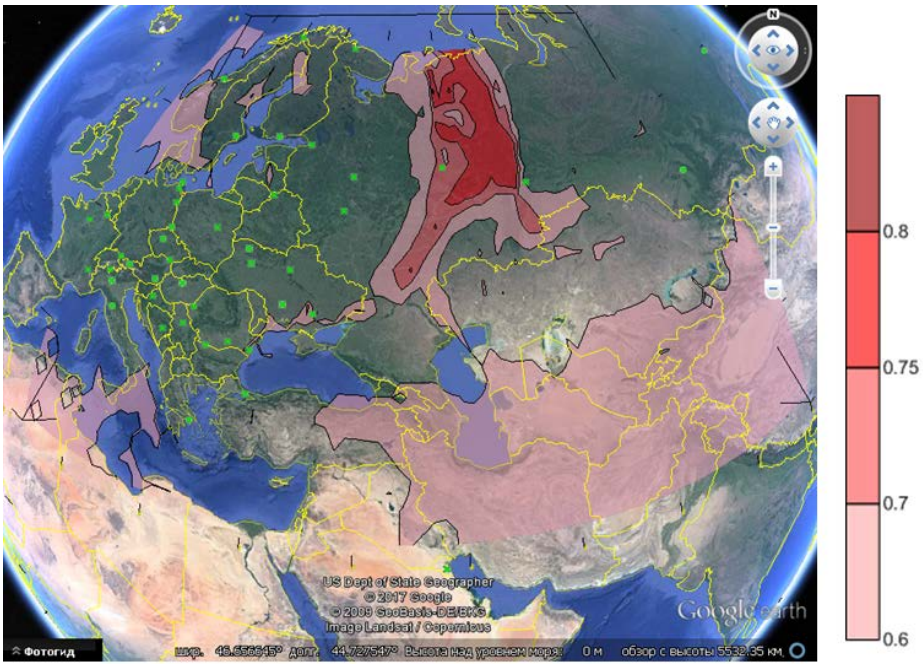


7 RESPONSE

The Fukushima accident of March 11, 2011, triggered many activities of European countries, mainly related to their nationals living abroad, requests from industry and even aviation. However, the response was not at all harmonised in Europe. Each country announced recommendations for their own citizens living abroad. Source terms were available in each European member state, but not shared officially. The first "official" source terms was published by IRSN in March 22 [13]. As one consequence, the European PREPARE project performed research to investigate the technical means for a so called "Analytical Platform" to allow exchange of technical information at a European level, however it is so far not really applied [14]

In late September 2017, RU-106 was detected in several European aerosol-monitoring stations. The level were very low and there was no risk for European citizens. The origin of the Ruthenium, however, was not known and no country reported an incident. As with Fukushima, no harmonised response in Europe could be observed. Some organisations such as BfS and IRSN exchanged information and issued press releases referring to each other.

Figure 3: Source location using JRodos; spatial distribution of the maximum correlation coefficient of calculated vs. measured values that correspond to the location of the source from I. Kovalets, Ukraine



Moreover, operational aspects or gaps should be further discussed. Using the agreed range of possible Ru-106 released (between 100 – 300 TBq), the INES 5 level is an appropriate characterisation of that event ((INES 5 = hundreds to thousands of TBq of 131I) - Ru-106 has multiplication factor of 6). The IAEA could not react as this requires a trigger from the country where the incident is located. As Russia did not acknowledge this, coordination work of IAEA was very limited and mainly related to data collection and storage. Furthermore, aerosol stations (e.g. ring of five - informal network of experts) are not part of an early notification system and information from these stations are therefore not "official" and might be difficult to refer to.

Summarising observations from Fukushima and Ruthenium, an exchange platform at the European level to inform each other at the technical level (e.g. source terms) might become more and more important to strengthen harmonised response in Europe. The NERIS platform might be one promising candidate as many organisations are members and the technical means could be established there.

8 CONCLUSIONS

Reviewing the decades, technical tools have significantly improved but gaps were identified in e.g. monitoring, countermeasure strategy preparedness, optimisation, communication and decision making partly addressed at present within ongoing European research projects such as CONFIDENCE and TERRITORIES. Also international guidance is available from many international organisations including but not exclusively from IAEA and ICRP. Preparedness at national level is progressing, however with different speed. European Platforms such as NERIS on preparedness for nuclear and radiological emergency response and recovery help to identify gaps and propose research activities closing them. In this respect, improved cooperation between the operational and research community surely will help to minimise gaps further and produce those tools that are needed.

Finally, harmonisation of preparedness and response can be improved in Europe as demonstrated in this paper. Again, the NERIS platform may play a role in this, which – for sure – requires future discussions and negotiations between NERIS members and organisations responsible for these areas in European member States.

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Development of a dynamic food chain model for assessment of the radiological impact from radioactive releases to the aquatic environment

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Abstract. The software tool POSEIDON-R, used for modelling the concentration of radionuclides in water and sediments and the uptake and fate of radioactive content in the aquatic environment and marine organisms, is recently extended to incorporate the uptake in the benthic food chain. This work will summarise the developed food chain model and present the findings of the model when applied to the Fukushima accident. ^{137}Cs concentrations in water, sediment and sea organisms are computed and compared with experimental data. The results generally hold good agreement with experimental findings for the area 15 km around the Fukushima NPP. Subsequently, the absorbed dose rate for marine organisms from ^{137}Cs in this area is estimated at around 2 to 4 $\mu Gy \cdot d^{-1}$.

KEYWORDS: Aquatic food chain, Fukushima accident, Caesium-137.

1 INTRODUCTION

For the uptake and fate of radioactive content in the aquatic environment and marine organisms, a software tool that consists of the compartment model POSEIDON-R linked with a dynamic food chain model is developed in the early 2000´s[1]. This tool is recently extended to accommodate for more advanced aquatic ecosystems. The tool together with its new functionality is validated using available data from the Fukushima accident and has now been integrated in the European decision support system JRODOS.

Traditionally activity concentrations in marine organisms are estimated using a steady-state biological concentration factor (BCF) approach based on computed activity concentrations in water. A more suitable approach, particularly for modelling the effects from accidental releases, is the use of a dynamic time dependent food chain model. Such model computes the uptake and accumulation of radioactive content in the marine organisms using a generalisation of the predator prey relationships found in the aquatic environment. This includes a food web of the most dominant marine organisms. In addition, the POSEIDON-R tool is recently extended with a benthic food web which addresses the uptake of radioactivity from the sediment layer to the marine organisms found at the interface of the sediment and aquatic layer. The importance of such model extension results from deposition and accumulation in the sediment. Consequently, these organisms are at risk from higher uptake of radioactivity, leading to a higher radiation dose to the organisms themselves as well as potential fish consumers. In addition the uptake in benthic fish leads to a transfer of activity to the pelagic food chain located in the upper part of the water column.

The presented work will focus on: i. the developed food chain model, ii. its application to the Fukushima accident, and iii. validation of the computed activity concentrations in water, sediment and fish against experimental data. Furthermore, estimated dose rates for the marine organisms are also reported.

2 MODEL DESCRIPTION AND SETUP

2.1 Model description

In this study, the dynamic food chain model that is part of POSEIDON-R was extended to describe transfer pathways of ^{137}Cs from bottom sediments to marine organisms. The model was developed to assess doses to human from marine products in the decision-support system JRODOS for off-site nuclear emergencies[1]. For such aim it was necessary to use a robust and generic model requiring a minimal number of parameters. Therefore, in the model, marine organisms are grouped into a few classes based on trophic levels and types of species. The radionuclides are also grouped in several classes depending on the type of tissues in which a specific radionuclide accumulates preferentially. These simplifications allow for a limited number of standard input parameters. The transfer scheme of radionuclides through the marine food web is shown in Figure 1 where transfer of radionuclides through the food web is shown by arrows whereas the direct transfer from water is depicted by the shadowed rectangle surrounding 11 biotic compartments (*i*=1, ..., 11). Pelagic organisms are divided into primary producer, phytoplankton (*i*=1), and consumers which consist of zooplankton (*i*=2), foraging (non-piscivorous) fish (*i*=3), and piscivorous fish (*i*=4). The benthic food web includes three primary pathways for radionuclides: (I) transfer from water to macroalgae (*i*=5), then to grazing invertebrates (*i*=6, 7, 8); (II) transfer through the vertical flux of detritus and zooplankton faeces[2] to detritus-feeding invertebrates (*i*=8); and (III) transfer through contaminated bottom sediments to deposit-feeding invertebrates (*i*=6).

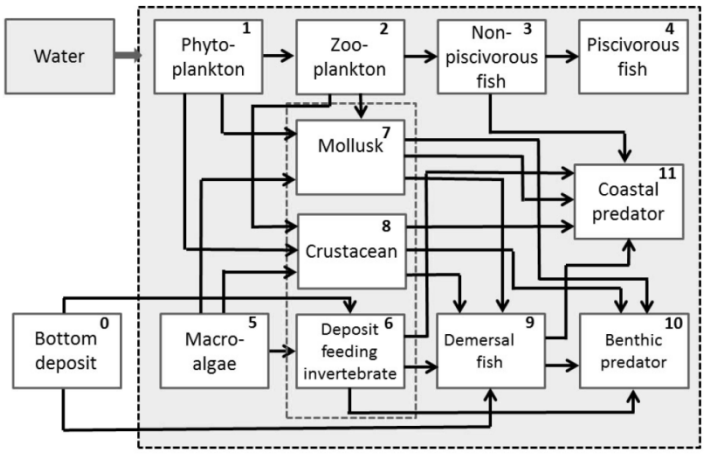


Figure 1 Biological compartments with corresponding numbers (reported in the text and tables) and radionuclide transfer (arrows) for extended dynamic food chain. Compartments within the dashed rectangle also exchange directly with the surrounding water.

The radionuclides adsorbed in the organic matter in the sediments are bioavailable for benthic organisms but the mineral component of sediments is not[3]. However, Koyanagi et al.[4] found relatively rapid and more intensive transfer of several sediment adsorbed radionuclides (^{54}Mn, ^{60}Co, ^{65}Zn) to particular organs of the demersal fishes in contrast to flesh. We assume that: (i) radioactivity concentrations in organic and mineral fractions of bottom deposit are in mutual equilibrium, (ii) radioactivity concentrations in microbial biota and non-living organic matter also are in equilibrium, and (iii) only organic matter in the bottom deposit is bioavailable. The benthic invertebrate group includes mollusks (e.g. filter feeders) (*i*=7), crustaceans (e.g. detritus feeders) (*i*=8), and subsurface and surface deposit feeders (e.g. annelid) (*i*=6). In the model, radioactivity is transferred from benthic invertebrates to demersal fishes (*i*=9) that feed on them, and on to omnivorous bottom predators (*i*=10) (Figure 1). The marine food web also includes "coastal predators" (*i*=11) feeding in the whole water column in shallow waters. In the extended model utilized in this study, the concentration of radioactivity in phytoplankton C_1 is calculated using the biological concentration factor (BCF) approach due to the rapid uptake from water and the short retention time of radioactivity:

$$C_1 = CF_{ph} C_w, \quad (1)$$

where C_w is concentration of radioactivity in water and CF_{ph} the BCF for phytoplankton. For the macroalgae, a dynamic model is used to describe radionuclide concentrations due to the longer retention times

$$\frac{dC_5}{dt} = (CF_{ma}C_w - C_5)ln2T_{0.5,5}^{-1}, \quad (2)$$

where C_5 is the concentration of radioactivity in the macroalgae and CF_{ma} the corresponding BCF, $T_{0.5,5}$ is the biological half-life of the radionuclide in the macroalgae, and t is the time. The concentrations of a given radionuclide in the zooplankton (i=2), invertebrates (i=6, 7, 8), and fish (i=3,4, 9, 10, 11; see Table 1 for a description of the different fish groups in the model) are described by the following differential equation:

$$\frac{dC_i}{dt} = a_iK_{f,i}C_{f,i} + b_iK_{w,i}C_{f,i} - ln2T_{0.5,i}^{-1}C_i, \quad (3)$$

where C_i and $C_{f,i}$ are the concentrations of radioactivity in the marine organisms and their food, respectively, a_i is the assimilation efficiency, b_i is the water extraction coefficient, $K_{f,i}$ is the food uptake rate, $K_{w,i}$ is the water uptake rate, and $T_{0.5,i}$ is the biological half-life of the radionuclide in the organism. The activity concentration in the food of a predator $C_{f,i}$ is expressed by the following equation, summing up for a total of n prey types

$$C_{f,1} = \sum_{j=0}^{n} C_{prey,j}P_{i,j}\frac{drw_{pred,i}}{drw_{prey,j}}, \quad (4)$$

where $C_{prey,j}$ is the activity concentration in prey of type j, $P_{i,j}$ is preference factor for prey of type j, $drw_{pred,i}$ is the dry weight fraction of predator of type of i, and $drw_{prey,j}$ is the dry weight fraction of prey of type j. The index "0" corresponds to the bottom deposit in sediment (Figure 1). The concentration of assimilated radioactivity from the organic fraction of sediment is related with the radioactivity concentration of the upper layer of bulk sediment as $C_{prey,0}=ø_{org}\cdot C_s$. Here $ø_{org}$ is an empirical parameter $ø_{org} =(1-p)\cdot f_{org}C_{org}/C_s$, where p is porosity, f_{org} is the organic matter fraction, C_{org}/C_s is the ratio of concentration C_{org} (Bq kg^{-1}-dry) in the organic matter to the bulk sediment concentration C_s (Bq kg^{-1}-dry). The value of $ø_{org}$ is in the range of 0.1–0.01[5].

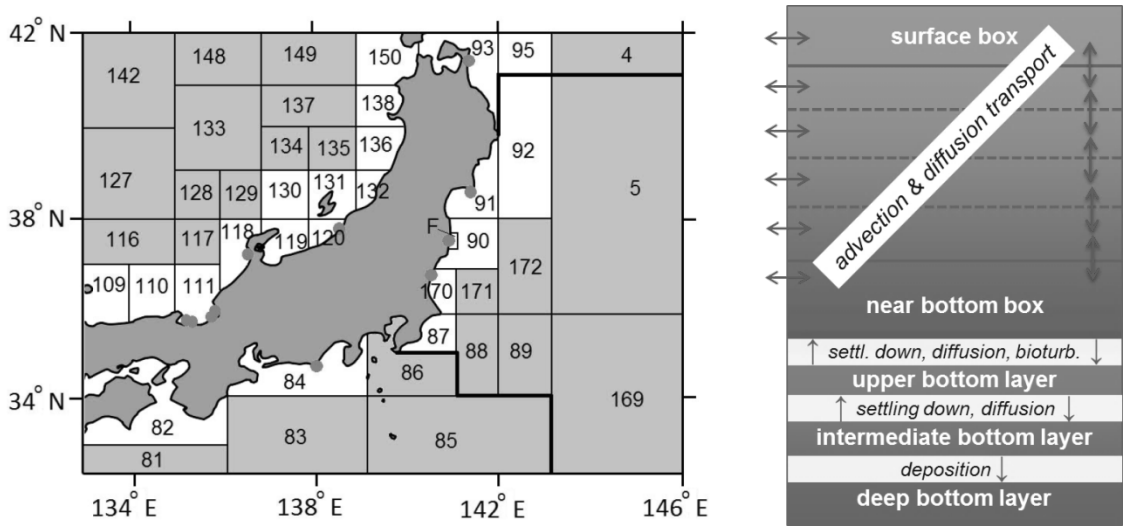
Table 1 Parameters of dynamic food chain model.

i	Organism	drw	$K_{f,i}$ (d^{-1})	a_i	$K_{w,i}$ (m^{3}kg^{-1}d^{-1})	b_i	$T_{0.5,i}$ (d)
1	Phytoplankton	0.1					
2	Zooplankton	0.1	1.0	0.2	1.5	0.001	5
3	Non-piscivorous fish	0.25	0.03	0.5	0.1	0.001	Tissue dep.
4	Piscivorous fish	0.3	0.007	0.7	0.075	0.001	Tissue dep.
5	Macroalgae	0.1			0.6	0.001	60
6	Deposit-feeding invertebrate	0.1	0.02	0.3	0.1	0.001	15
7	Mollusk	0.1	0.06	0.5	0.15	0.001	50
8	Crustacean	0.1	0.015	0.5	0.1	0.001	100
9	Demersal fish	0.25	0.007	0.5	0.05	0.001	Tissue dep.
10	Bottom predator	0.3	0.007	0.7	0.05	0.001	Tissue dep.
11	Coastal predator	0.3	0.007	0.7	0.075	0.001	Tissue dep.

Values of the model parameters are given in Table 1. The parameters for pelagic and benthic food webs were compiled from published data[6]. The biological half-life data for fish flesh, where ^{137}Cs is accumulated, show variability in a large range (35–180 days) due to the differences between species and due to the differences in the experiment methodology.

The dynamic food chain model is integrated in the POSEIDON-R[1,7-8] model where the marine environment is modelled as a system of compartments representing the water column, bottom sediment, and biota. The compartments describing the water column are subdivided into a number of

vertical layers as shown in Figure 2b. The model assumes partition of the radionuclides between the dissolved and particulate fractions in the water column, described by a distribution coefficient. The radionuclide concentration for each compartment is governed by a set of differential equations including the temporal variations of concentration, the exchange with adjacent compartments and with the suspended and bottom sediments, radioactive sources, and decay. The exchange between the water column boxes is described by fluxes of radionuclides due to advection, sediment settling, and turbulent diffusion processes. The activity loss in suspended sediments occurs through settling in underlying compartments and, finally, to the bottom. A three-layer model describes the transfer of radionuclides in the bottom sediments. The transfer of radioactivity from the upper sediment layer to the water column is described by diffusion in the interstitial water and by bioturbation. Radioactivity in the upper sediment layer migrates downwards by diffusion and by burial at a rate assumed to be the same at which particles settle from the overlying water. The upwards transfer of radioactivity from the mid-sediment layer to the top sediment layer occurs only by diffusion. Burial causes an effective loss of radioactivity from the middle to the deep sediment layer, from which no upward transfer occurs. The model equations are given in the Supplement. The model for the pelagic food web component was implemented for the whole area of study, whereas the benthic component was included in the shallow, single water-column layer compartments adjacent to the shore (white boxes on the Figure 2a).



(a) (b)

Figure 2 POSEIDON-R compartment model around the FDNPP (a) and the vertical water column with the three sediment layers (b).

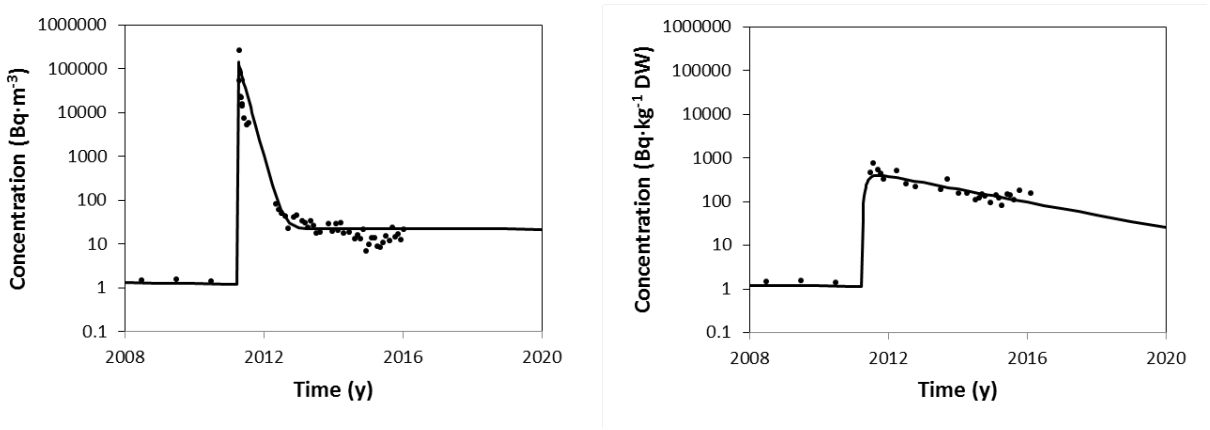
2.2 Model setup

The model was customized for the north-western Pacific Ocean, the East China and Yellow seas, and the Sea of Japan (East Sea). A total of 176 boxes cover this entire region. In the deep-sea regions a three-layer box system was built to describe the vertical structure of the radioactivity transport in the upper layer (0–200 m), intermediate layer (200–1000 m), and deeper layer (> 1000 m). The compartments around the Fukushima Dai-ichi Nuclear Power Plant (FDNPP) are shown in Figure 2a. The "coastal" box (placed at "F" in Figure 2a) covers 15-30 km and is nested into a large "regional" box (box 90) in order to provide more detailed description in the area around the FDNPP. It covers a circular-shaped surface area of a radius 15 km centred at the FDNPP where observation data were collected. The coastal box has one vertical layer for the water column and three bottom sediment layers. It is shallower than the single-layer outer box 90. The water exchanges with the outer box are equal in both directions. The averaged advective and diffusive fluxes between regional compartments were calculated for a 10-year period (2000–2009) using the Regional Ocean Modeling System (ROMS). Details of customization are given by Maderich et al.[7-8]. The value for parameter $ø_{org}$=0.01 was used. The simulation of dispersion and fate of ^{137}Cs was carried for the period 1945–2010 to provide background concentrations of radiocaesium for the radiological assessment of the FDNPP

accident for the period 2011–2020 and to verify the model with available data. The main source of ^{137}Cs in the north-western Pacific in the period 1945–2010 was from fallout due to atmospheric nuclear weapon tests. The fallout includes a global component, caused by the transport of radioactivity due to the general atmospheric circulation and subsequent deposition on the surface of the ocean and a regional component, caused by fallout from weapon tests carried out in the Marshall Islands, resulting in the contamination of the surface layer of the ocean.
The simulation for the period 1945–2010 was continued for the period of 2011–2020 with a source term estimated from the Fukushima accident. It was assumed that the release of activity directly to the ocean took place over the period 1–10 April 2011. Amounts of 5 PBq of ^{134}Cs, and 4 PBq of ^{137}Cs were transferred directly into the coastal box. These quantities are in accordance with widely accepted source terms for the Fukushima accident simulations (see Povinec et al., 2013). The atmospheric deposition data were obtained from simulations with the MATCH model[9] where the dispersion of ^{137}Cs for the period 12 March – 5 April was computed[7]. The area of the atmospheric deposition is bounded by the bold line on the Figure 2a.

3 RESULTS

The results from the modelling of the ^{137}Cs concentration in the water and in the upper layer of sediments of the coastal box are shown in Figure 3. Model results for the water demonstrate good agreement both with yearly averaged observations by MEXT (the Japanese Ministry of Education, Culture, Sports, Science and Technology) for the period 1950–2010[10] and with observation by TEPCO (Tokyo Electric Power Company) for the period of 2011–2016[11]. Figure 3a with confirms the almost constant concentration of ^{137}Cs in the water up to 2011 and the subsequent rise and fall in concentration following the release form the FDNPP. The model also predicts well the concentration of ^{137}Cs in the bottom sediment before the accident and the sudden increase in concentration by more than 3 orders of magnitude as a result of the accident. For an accurate prediction after 2013 exchange terms were added to take account of the vertical ^{137}Cs transfer[6].



(a) (b)

Figure 3 Comparison between calculated and one month averaged observed ^{137}Cs concentration in seawater (a) and in bulk bottom sediment (b) in the coastal box around the FDNPP. Values in (b) are given in becquerels (Bq) per kilogram (kg) of dry weight (DW).

The simulated ^{137}Cs concentrations in deposit-feeding invertebrates, demersal fishes, bottom predators, and coastal predators in the coastal box (placed at "F" in Figure 2a) are shown in Figure 4 along with observed concentrations by the Japan Fisheries Research Agency[12]. The symbols in Figure 4 are observation data for sea urchins (*Strongylocentrotus nudus*) (a), flounders (*Microstomus achne*, *Kareius bicoloratus*, *Pleuronectes yokohamae*) (b), and Japanese rockfish (*Sebastes cheni*) (c). The symbols in Figure 4d are data for seabass (*Lateolabrax japonicus*) and fat greenling (*Hexagrammos otakii*). Just after the accident, the simulated ^{137}Cs concentration in the deposit feeding invertebrates and the observed concentration in the sea urchin increase due to the high concentration of ^{137}Cs in the water (Figure 4a). After that, the concentration trend becomes similar to trends in the sediment (Figure 3b). This is consistent with the model diet that includes macroalgae and deposit

organic matter grossly representing the diet of *S. nudus*[13]. The macroalgae contribution to contamination from feeding first prevails. After 2012, the contamination from ingestion of bottom deposits dominates. The decrease in concentration (depuration constant) is close to the decrease constant for the sediment observations and agrees with observations by Sohtome et al.[14] that concentration in sediment and in deposit feeding benthic invertebrates show almost identical rates of decrease. The results of simulation of the ^{137}Cs concentration in the demersal fishes (Figure 4b) agree well with observations documented for several species of flounders. The gradual decrease of activity in demersal fish caused by the transfer of activity from organic matter deposited in the sediment is similar to observations by Wada et al.[15]. Comparison of simulations with observations for a bottom predator (Japanese rockfish) in Figure 4c shows also good agreement. The comparison of simulated and observed concentrations of ^{137}Cs in coastal predators is given in Figure 4d.

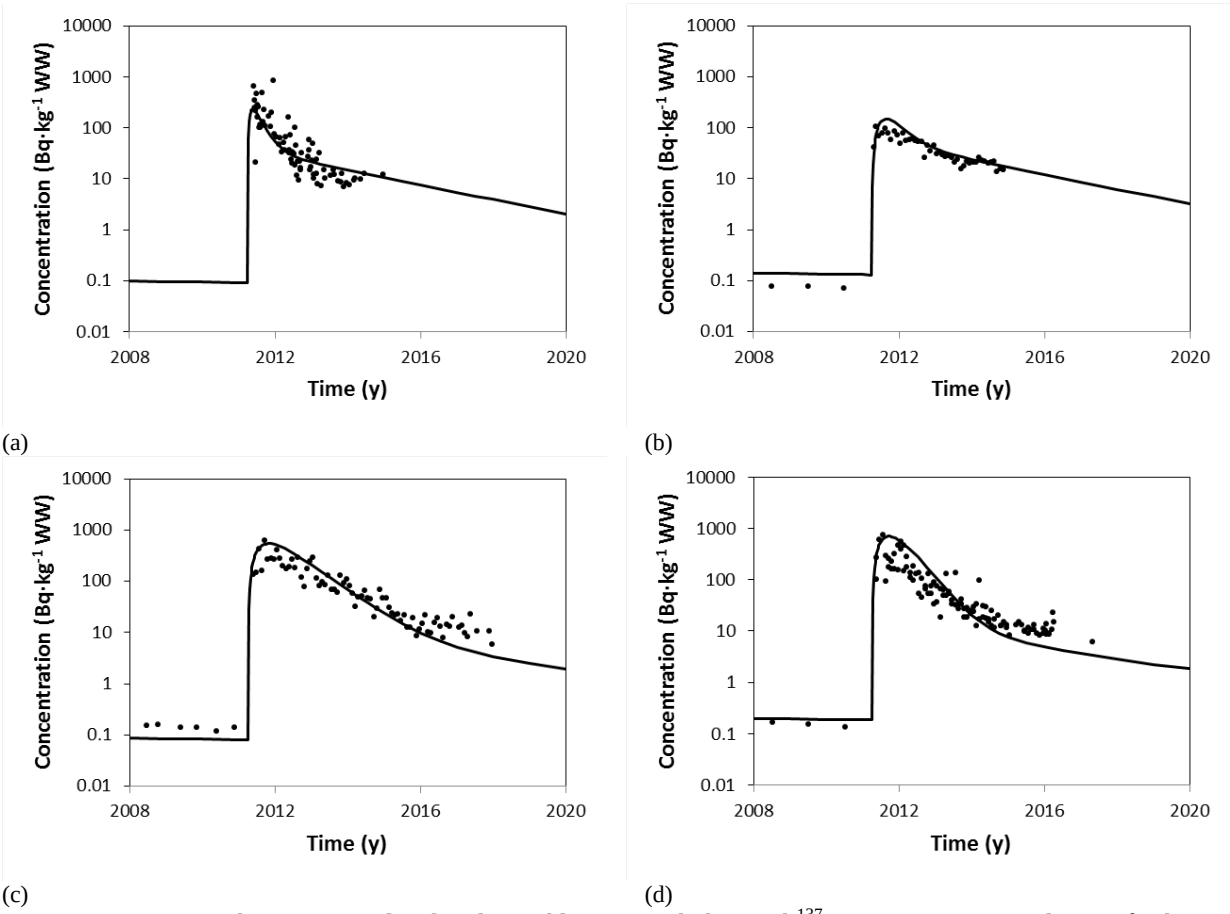


Figure 4 Comparison between simulated and monthly averaged observed ^{137}Cs concentration in deposit-feeding invertebrate (a), demersal fish (b), bottom predator (c), and coastal predator (d) around the FDNPP. Values are given in becquerels (Bq) per kilogram (kg) of wet weight (WW).

The estimated dose rate to the four types of marine organisms presented in Figure 4 is shown in Table 2. The dose estimate includes both internal and external exposure from ^{137}Cs and its progeny ^{137m}Ba, and is based on the maximum concentration levels found in water, sediment and species. The absorbed dose rates are computed using the dose coefficients for non-human biota reported in the ICRP publication 136[16]. For the first three species, it is assumed that the external exposure is determined by 50% submersion in sediment and 50% submersion in water. For the coastal predator full submersion in water is assumed. The maximum absorbed dose rates in the aftermath of the accident are in the order of 2 to 4 $\mu Gy \cdot d^{-1}$. Although the dose rate from other important nuclides such as ^{110m}Ag, ^{134}Cs and ^{90}Sr is not included in this analysis, the dose rates are well away from the derived consideration reference levels (DCRL) for environmental protection that start from 1 $mGy \cdot d^{-1}$ for aquatic reference animals[17].

Table 2 Estimated maximum absorbed dose rates from ^{137}Cs for the various marine organisms.

deposit-feeding invertebrate ($\mu Gy \cdot d^{-1}$)	Demersal fish ($\mu Gy \cdot d^{-1}$)	Bottom predator ($\mu Gy \cdot d^{-1}$)	Coastal predator ($\mu Gy \cdot d^{-1}$)
2.2	2.1	3.7	3.1

4 CONCLUSIONS

The software POSEIDON-R, designed for modelling the uptake of radioactivity in the aquatic water system is extended with a dynamic food web model for the benthic zone and applied to the Fukushima accident. The simulated and observed ^{137}Cs concentrations in water and sediment demonstrate good agreement, both prior to the accident and as well as afterwards. The ^{137}Cs concentration in the various aquatic species confirms experimental data. The sudden increase as well as the subsequent exponential decline as predicted are consistent with their sediment/water interaction and their predation diet. Based on the computed ^{137}Cs concentrations the maximum absorbed dose rate for the different marine organisms is estimated at around 2 to 4 $\mu Gy \cdot d^{-1}$.

5 ACKNOWLEDGEMENT

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Overview of Accidents in Industrial Accelerator Facilities

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Abstract. From about 30 000 accelerators at work worldwide about half of them are used in industry. In the last decade sterilization became an important part of many industries. The dose rates required might be up to thousands of Gy/s. Due to such extremely intense radiation fields, defense in depth shall be implemented in the design of irradiation facilities and high safety culture shall be in place as described in IAEA SSG-8. Sterilisation facilities are based on a use of radioactive materials of Category I, e.g. Co-60, or accelerators. The IAEA published a database of gamma irradiation facilities but no such database for accelerator's facilities for sterilisation exists. In order to understand accidents related to industrial irradiators based on acelerator technology available literature was analyzed, e.g. IAEA, OTHEA-RELIR database and IRPA proceedings. While accidents related to radioactive sources are well described, including detailed analyses of accidents with fatalities, e.g. in Kjeller 1982 and Nasvizg in 1991, less information is available when accelerators are involved. It must be noted that the very first accident related to industrial sterilization using accelerator happened in 1965 in USA. The accident resulted in amputation of worker's leg and arm. The analysis of databases enabled identification of only six accidents. No fatalities were reported. This fact can be linked to safety features installed in accelerator's safety systems enabling easier maintenance of safety in such facilities than in facilities using radioactive sources. Collimated fields used at accelerator's facilities contribute to partial exposure to bodies in case of accidents, e.g. in general amputation of limbs could be required in worst-case scenario. The analysis is presented systematically focusing on understanding physical phenomena such as dark currents as well as safety systems. The analysis might help designers, users and regulatory bodies to better understand risks, associated with such industrial accelerators.

KEYWORDS: risk, irradiators, sterilization, sterilisation, radiation, accident, accelerator, radioactive source, safety, defense in depth, industrial, dose rates

1. INTRODUCTION

Accelerator technology is widely used in last fifty years. As given in [1], today around 30 000 accelerators are at work worldwide in various areas. The very first use of accelerator technology was in research as described in [2] while a list of accelerators used in research today is given for example at the IAEA web IAEA's Accelerator Knowledge Portal [3]. Although a use of accelerators in research and in medical fields are widely known, there are many applications of accelerators in industry such as:

- ion implantation for fabrication of semiconductor devices and materials
- electron beam materials irradiators
- accelerator production of radionuclides
- ion beam analysis
- production and applications of neutrons using particle accelerators
- nondestructive testing and inspection using electron linacs
- industrial use of synchrotron radiation.

A brief presentation of industrial accelerators available today is given elsewhere [2]. In fact as noted in [4] around half of all accelerators mentioned which have been built in the last 60 years are actually used in industry. The authors also stress that users in industry are often interested only in cost-effectiveness of accelerator technology and are less interested in technical details, i.e. the accelerator is very often perceived as a "black box". Such approach to accelerator technology might quickly lead to accidents associated with ionizing radiation.

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Accelerators used in industry have very different characteristics, e.g. electron beam irradiation accelerators used for curing thin film coatings can apply voltage from 100 to 300 keV and beam currents form 10 to 2000 mA while on the other hand accelerators used for sterilisation of medical products or accelerators used as x-ray generators for food irradiation and waste water remediation apply voltage from 5 to 10 MeV and are as a high energy scanned beam systems capable of 25 to 700 kW beam as noted in presentation given in [5]. Such broad ranges of physical parameters also poses broad ranges of radiation risks associated with a use of accelerator technology in industry. In order to assure radiation safety accelerator technology includes numerous safety systems. Its use requires well established management of the risks including procedures addressing responsibilities, initial training of workers, involvement of radiation protection officer and qualified experts. All involved in the accelerator technology, e.g. designers, constructers of facilities, maintenance staff, users and regulatory authority staff could use lessons learned from accidents which already happed when using accelerator technology.

In particular, accidents related to accelerators used in sterilisation should well understood as a use of accelerators is associated with high dose rates in order to make an efficient sterilisation. Namely, radiation field of up to hundred thousand of Gy/s might be present during sterilisation. Due to such extremely intense radiation fields, defence in depth shall be implemented in the design of irradiation facilities and strong safety culture shall be in place.

A use of sterilisation of medical products or food is becoming widely used today. As pointed out in the IAEA document [6]: “Industrial irradiators produce very high dose rates during irradiation, such that a person accidentally present in the radiation room could receive a lethal dose within minutes or even seconds.” The facilities causing such high risk are based either on accelerator technology or on a use of radioactive materials, i.e. Co-60 or Cs-137. Such sealed radioactive sources are categories as source so Category I of the IAEA Categorisation [7] reflecting the risk associated with them. Namely, as given in [7]: »The sources of Category I if not safely managed or securely protected, would be likely to cause permanent injury to a person who handled it or who was otherwise in contact with it for more than a few minutes. It would probably be fatal to be close to this amount of unshielded radioactive material for a period in the range of a few minutes to an hour.”

In the last decade irradiators with radioactive sources became obsolete in order to avoid security issues as well as handling a disused source which requires complex safety measures to be in place even when disused. In particular, due to security reasons a use of Cs-137 for irradiation became obsolete. So today the accelerator based industrial irradiators are either replacing a use of industrial irradiators using a radioactive source of Category I or the accelerator based industrial irradiators are a first choice when introducing sterilisation process in industrial facility.

2. ANALSIS OF ACCIDENTS

2.1 Accident Databases

The very first attempt to the systematic analysis of radiation accidents associated with industrial accelerator facilities where sterilisation is taking place is given in [8] using limited data from open literature. This analysis has been enlarged here by additional data from [9]. A list of literature used in the analysis of accidents with industrial sterilisation facilities using accelerator technology includes:

- IAEA document on lessons learned from accident in industrial irradiation facilities [10]
- Database given in Table 1 in [11]
- UK IRID database [12]
- OTHEA/RELIR database [13]
- presentation Accelerator Health Physics, 2008 HPS [14]

- Radiation accidents and other events causing radiation casualties-tabulated data compiled by Wm. Robert Johnston [9].
- CRC Handbook of Management of Radiation Protection Programme [15].

By purpose the present analysis is focused on a very particular industrial application, e.g. sterilisation. In such facility products, e.g. food to be sterilised, are packed in the standardised cartons which are placed outside the irradiation room on a conveyor belt by workers. Very often the cartons are put in a standardised box just before placing them on the belt in order to avoid any misplacement of cartons in irradiation rooms, e.g. sterilisation facilities are using specially designed metal boxes where usually only few cartons can be placed in. The belt with cartons or boxes travels in the irradiation room. During irradiation cartons or boxes are moved through irradiation fields. The path of the conveyor belt might be quite complex in order to achieve homogenous irradiation of the product. Complexity of this path might be partly replaced by using complex radiation field such as a field with few modules of radioactive sources or a field produced by two accelerators instead of one. After the irradiation, which lasts from about a minute up to an hour, cartons or boxes with products leave the irradiation room. They are picked up from the conveyor belt by workers and marked as irradiated products.

A use of industrial irradiators have some specifics.

- Due to complexity of the building, which must among others provide sufficient shield, facilities are usually placed in industry dedicated area where safety and security measures are going to be put in place without jeopardising practical aspects of a use of irradiator such as a need to have storages for incoming and outcoming products and easy access with trucks.
- Under normal operational conditions the crew during one shift is composed of only few persons, e.g. operator of an accelerator, few workers handling cartons and boxes and radiation protection officer available all the time when sterilisation is taking place.
- A person who is operating a command panel in the command room should be capable to manage highly routine operations for hours as very routine operations are taking place during normal operation. At the same time the person should be highly trained to manage high risks associated with the facility in case of incidents or accidents.
- The facility might have huge workload and might provide nonstop service, i.e. on a 24/7 basis.

2.2 Reported Accidents

Nowadays reporting of accident and lessons learned are an integral part of radiation safety, e.g. in the [16] but this was not the case in the past. In particular so-called near-missed events have not been reported in the past. It must be noted that some databases mentioned above include limited details related to accidents, e.g. in some cases even the year of the accident is not known and technical details are lacking. In addition, authors of the databases are not using the same categorisation of accidents, e.g. some authors do not distinguish between accident in an industrial facility and in a research facility. Moreover, some accelerators might be used partly for research and partly for irradiation of items for the purpose of sterilisation. In general, accidents with radiation sources used in industrial irradiators are well described including detailed analyses of accidents with fatalities, e.g. one fatality was a consequence of the Kjeller accident in Norway in 1982 as well as of the Nasvizg accident in Belarus in 1991 [11]. Much less information is available when industrial accelerator facilities are involved in accidents although the first accident related to industrial sterilisation happened in 1965 in USA where accelerator was involved resulting in amputation of worker's leg and arm [11].

The search for accidents with industrial irradiator reveals that a list of accidents related to industrial irradiators using radioactive sources is, as a rule, longer than the list of accidents where accelerators were involved, e.g. in the document IAEA from 1996 [10] three of eight cases are related to accelerators. In this document only cases related either to fatal consequences or severe radiation

injuries are described. It should be pointed out that no fatal consequences were related to a use of accelerator in sterilisation.

The database given in [9] also reveals that a list of accidents with irradiators using radioactive sources is much longer than a list of accidents with accelerators. Namely, from 26 cases related to so-called “irradiator accident”, i.e. accident related to irradiation, only four of them are related to accelerators. All other cases are related to a use of Co-60. The analysis also reveals few facts.

- Exposure to a member of the public was not involved in none of the reported 26 cases, i.e. only occupational exposures which resulted in causalities occurred.
- Fatalities occurred only in cases where a source was a radioactive source, i.e. a use of accelerators in industry did not cause any reported fatality.
- When accident resulted in a fatality as a rule one or two persons died.

The database also contains around ten additional cases where a use of accelerators for other purposes lead to acute radiation exposure to humans sufficient to cause causalities. None of these cases is related to fatalities. Table 1 gives a comparison of reported data from [9] mentioned above which have been collected till 2014. It must be stressed that number of fatalities were result of 8 cases in some more than one person died.

Table 1: Analysis of database given in [9] giving also details related to other users of accelerator tehnology.

		Number of Reported Accidents	Number of Reported Accidents resulting in Fatalities
Irradiators	Accelerators	4	0
	Radioactive Sources	22	8
Other Uses of Accelerators		13	0

As already pointed out in [8] reporting methodology might be an issue when comparing the databases. Namely, international databases are based on voluntary basis while national databases can be based on different criteria. For example, looking to Figure 4 given in [17] it seems that majority accidents in South East Asia are related to orphan sources, i.e. sources outside a regulatory control, and majority of overexposures in Russia are related to industry. On the other hand, the UK IRID database [12] does not report any accident related to gamma irradiators or use of accelerators in industry. The OTHEA/RELIR database contains two accidents, both happened in France. Some databases are more focused on technical parts of accidents and safety fetaures which were not installed, propery maintained or used while some databases are focused on medical treatment of patient. The literature do not contain any information related to any standards to be used at the time of the accidents. Today the ANSI N43.1. from 2011 on radiation safety for the design and operation of particle accelerators can be used.

As mentioned already in [8] it can be concluded there are much less reports on accidents related to industrial accelerators than to irradiators using radioactive materials as safety features installed enable easier maintenance of safety in such facilities than in facilities using radioactive sources. In addition, collimated fields used at accelerator’s facilities contribute to partial exposure to bodies in case of an accidents, e.g. in general amputation of limbs could be required in worst-case scenario.

A short overview of events reported based on the available open literature is given in structured in a table 2 in a way to enable the analysis of these very few accidents which happened with irradiators based on accelerator technology used in industry. The table is enlarged table already given in [8]. All six accidents happened in the period from 1965 to 1992 taking into account that in one case the date of the accident is not known. The so-called “Hanoi-Accident”, Hanoi, Viet Nam, is also included although it did not happened in industry but in research associated facility where samples have been

routinely irradiation for about one hour. The accident is described in details in [18]. The lessons learned from this accident which happened in 1992 are of so high importance that they are worth mentioned in the table in particular as the lessons tackle high underestimation of a risk associated with an irradiator based on accelerator technology. The accident is also included in [9].

Six identified accidents are presented. The table which is largely taken from [8] is giving initial events, selected contributing factors, responsibilities and lessons learned. Selection of contributing factor is somehow arbitrary as as a rule a list of contributing factor exists. The reader is highly recommended to use IAEA or UNSCEAR documents where additional data are available at least for some of the accidents.

Accident No. 1 Linear accelerator with a 10 MeV beam, Illinois (USA), 1965	
Initial Event	The worker entered to the irradiation room via a gap under the door, i.e. without tripping the interlock, during the irradiation. Up to 2400 Gy were received by various parts of his body.
Contributing Factor	The gap under the door enabled violation of operator' procedures.
Responsibility	The operator did not follow the rules to be used when entering the irradiation room.
Lessons Learned	The gap under the door used to accommodate the convey belt was not closed, i.e. the licensee was responsible for this error.
Accident No. 2 Van de Graaff linear accelerator with a 3 MeV beam used by Gulf Oil Company to irradiate soil samples, Harmarville (USA), 1967	
Initial Event	The workers were irradiated when entering the irradiation room.
Contributing Factor	The maximum whole body dose was approximately 6 Gy.
Responsibility	One interlock was not working and several interlocks were taped.
Lessons Learned	Licensee should assure functioning of interlocks.
Accident No. 3 Linear accelerator with a 3 MV potential drop, Maryland (USA), 1991	
Initial Event	The maintenance worker, i.e. operator in was conducting maintenance when the filament current was off. He was unaware of the risk of existing so-called "dark currents" also called "cold currents" producing the dose rates up to 13 Gy/s. The estimated doses were up to 55 Gy.
Contributing Factor	The interlock systems and other safety features were not designed in line with safety rules, e.g. independence, redundancy and diversity, allowing the entrance to the irradiation room when dose rates in the room were high.
Responsibility	The maintenance worker did not follow the rules.
Lessons Learned	The maintenance staff did not have enough knowledge to provide maintenance of the accelerator and to understand the importance to follow the procedures.
Initial Event	The safety features shall be redesigned, i.e. the licensee was responsible for this error.
Accident No. 4 Linear accelerator with a 2.5 MV, maximum current of 35 mA and maximum dose rate 80 000 Gy/s, Forbach (France) 1991	
Initial Event	The part-time workers entered to the irradiation rooms in order to provide maintenance of auxiliary equipment. "Dark current" caused dose rates up to 0.1 Gy/s. The whole body doses were up to 1 Gy and doses to the skin up to 40 Gy.
Contributing Factor	The second-hand facility had equipment which was not designed for such high exposure fields. The licensee did not have enough knowledge about safety

	requirements for such facility.
Responsibility	The licensee using second-hand facility did not assure safety measures to be in place due to a lack of understanding.
Lessons Learned	The licensee did not assure the competence of the workers.
Initial Event	The workers did not follow safety procedures.
Accident No. 5 Microtron MT-17 operating at the time of the accident with 15 MeV, Hanoi (Viet Nam)	
Initial Event	The physicist entered in the irradiation room when the beam using 15 MeV was turned on as no interlock existed. The doses received were between 10 to 25 Sv rad to the left hand and 20 to 50 Sv to the right hand.
Contributing Factor	Safety systems were not in place as the 20 year old accelerator from Dubna has been installed in Hanoi without appropriate safety features.
Responsibility	The persons involved in the accident although scientists did not have appropriate knowledge regarding a risk associated with irradiator as the risk have been highly underestimated.
Lessons Learned	The licensee did not assure safety features and appropriate training of the personnel.
Initial Event	Safety features of second-hand irradiator shall be installed, checked and maintained as well as upgraded in line with international standards.
Accident No. 6 Linear accelerator with a 800 kV and maximum current of 100 mA (France) date not known	
Initial Event	Three workers entered to the irradiation rooms in order to check ventilation system when the operator started with irradiation. The effective doses received were between 30 to 35 mSv.
Contributing Factor	Safety procedures were not in place.
Responsibility	The persons involved in the accident did not have appropriate retraining.
Lessons Learned	The licensee did not assure safety features and appropriate retraining of the personnel.

3. LESSONS LEARNED

The analysis of accidents which happened enables identification of critical safety issues related to a use of accelerators for irradiation for routine purposes in industry. A list of safety issues is given for example in [6] and is briefly described also in [8] describing two main components:

- layout of the facility enabling appropriate shielding and installation of numerous safety systems installed, i.e few tens of them, which shall be regularly checked and maintained
- safety procedures which shall be strictly followed by all workers who should have appropriate education and training related to safe operation and maintenance of accelerators.

The analysis of the six cases reveals some lessons to be learned.

a. Second-hand accelerators require special attention before their reinstallation as in two cases second-hand accelerators were involved in the accident. It might be even questionable if an old accelerator might be upgraded to a level to be fully in line with standards available today.
b. In two accidents “dark current” caused exposures. These accidents are nearly identical. Accelerator technology requires specific knowledge which shall be present in the industrial facility.
c. In one case initial design of the accelerator did not have safety systems based on independence, redundancy and diversity.

d. Safety procedures shall never be ignored, e.g. taping an interlock should be strictly prohibited and entering to the irradiation room without appropriate measuring equipment and knowing a status of the accelerator should be prohibited. The deliberate taping of safety system was identified in two cases.
e. Underestimation of risk associated with the operation of such accelerator by workers involved is identified in all six cases. Understanding the risks should be a prerequisite to allow the facility to operate. The risk should be understood by all workers including so-called maintenance workers or outside workers.

All accidents resulted in causalities due to relatively short exposure time, i.e. of the order of one minute. As the beams are highly collimated only part of the bodies is highly exposed. As a rule, amputation of one or more limbs or skin grafting of exposed persons was necessary. In one particular case a person was 6 weeks in coma and placed in a sterile chamber, while in few cases a complete bone marrow transplant was necessary. Just in one case no such extensive medical procedures were required.

4. CONCLUSIONS

Accelerators used for industry in order to irradiate products are becoming a widespread all over the world nowadays. They have better safety records as irradiators using radioactive sources of Category I, i.e. around 20% of accidents related to industrial irradiators are related to accelerators and no fatalities related to such accelerators have been reported. In addition, the security issues can be handled somehow easely. At the end of the lifetime no concerns related to a disused source of a mentioned radioactive source exists.

Despite these facts the risk associated with operation of accelerators is high as a worker might be in a very short time exposed to doses requiring amputation of a limb or skin grafting, i.e. exposure of less than a one minute might result is serious causalities. In huge analyses of different accients such as given in [19, 20] these particular lessons learned associated with a use of accelerator technology could be very quickly ovelooked.

A risk associated with industrial accelerators used for sterilisation require that only well trained workers understanding the risk could use numerous safety systems and features which should be installed, maintained and used in such irradiation facility. Lessons from six accidents related to accelerators used for irradiation in industry might encourage designers, constructors, operators, maintenance companies, training providers, qualified experts to critically review safety standards applicable in such facilities as as well as safety culture present in such facilities.

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Organisation of the environmental monitoring: lessons learnt from Fukushima

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Abstract. This paper proposes an analysis of the Japanese situation 7 years after the Fukushima accident, in order to provide feedback experiences on the environmental monitoring implemented to cope with the post-accident situation. This analysis has been achieved by interviewing national and local actors involved in the practical setting up of the environmental monitoring within the Fukushima prefecture, as well as reviewing different documents and public environmental data bases. Results of this study clearly show that the environmental monitoring implemented in Japan after the Fukushima accident gathers multiple actors on both national and local levels. The 'Comprehensive Radiation Monitoring Plan' (CRMP), set up by the Japanese government since August 2011, proposes a national monitoring system concerted, coherent and embracing all environmental compartments. Concerning the local level, the mistrust towards government and the willingness to better understand their own situation lead people living in the affected territory to implement their own environmental monitoring. However, these local data are heterogeneous and often redundant with the CRMP. In this context, the remaining issue consists in knowing how to go towards a better sharing between results produced by institutional and non-institutional actors.

Keywords: Environmental Monitoring, Experts, Authorities, Citizen Vigilance

1. GENERAL INTRODUCTION

In post-accident situations, the implementation of the environmental monitoring is essential for characterising the radiological situation of the affected territories, as well as, allowing people living in such territories to understand what is at stake in their own environment and helping them to become actors of their own radiological protection [1, 2, 3]. In this context, roles playing by institutional and non-institutional actors are determining factors to set up a sustainable monitoring, reach a consensus and so encourage the citizen vigilance. Besides, the recent feedback from the Japanese situation - 7 years after the Fukushima accident –also shows that, thanks to the progress of digital technologies, there is the possibility for people living in contaminated territories to have now the means to measure the radioactivity of their environment, share these results through various networks and so, regain progressively control of their daily life [4].

In this context, this paper focuses on the Japanese situation, in order to provide feedback experiences and analysis of the environmental monitoring implemented after the Fukushima accident. This analysis consists in (i) identifying the environmental schemes implemented following the Fukushima accident, (ii) mapping the different actors who come into play in such situations and (iii) highlighting some local experiences developed by local associations or municipalities within the affected territories.

These overall goals have been achieved by interviewing different Japanese actors involved in the practical setting up of the environmental monitoring within the Fukushima prefecture. In this way, feedback experiences, points of view and comments have been collected from both institutional actors (e.g. Japan Nuclear Regulatory Authority, Health and Labour Ministry, Fukushima prefecture, etc.) and local actors (e.g. local associations, municipalities, citizens, etc.) in November 2016.

Therefore, the following paragraphs present the results obtained from these interviews as well as from documents review and analysis of public environmental data bases available on the internet. These results focus essentially on two parts:

- The official monitoring set up at national level by the Japanese government;
- Highlight from some local initiatives.

2. THE OFFICIAL ENVIRONMENTAL MONITORING SET UP IN JAPAN: THE *Comprehensive Radiation Monitoring Plan* (CRMP)

2.1. Historical and objectives of the national environmental monitoring programme

Following the accident of the Fukushima-Daïchi Nuclear Power Plant (NPP) in March 2011, the government decided to develop and implement a national environmental monitoring programme, with the help of actors and institutions already involved in environmental monitoring before this accident (e.g. ministries, national agencies, *etc*.). Thus, on August 2, 2011, the *Comprehensive Radiation Monitoring Plan* (CRMP) - was created [5]. Based on the environmental monitoring programs implemented before March 2011, the CRMP aims to set up a global monitoring of the environment with a special focus on the radiological situation of the whole Japan. Each year, this national programme is reviewed and adapted according to the radiological evolution of the territory. Note that since the creation of the Nuclear Regulation Authority (NRA) in September 2012, the CRMP is directly coordinated by this national authority.

2.2. The environmental monitoring implemented in the framework of the CRMP

To ensure the best monitoring of the radiological quality of the environment, the CRMP declines different monitoring on each environmental compartment: ambient air, soil, lakes, rivers, drinking water, forests, wildlife and flora, marine environment, *etc.*

As already mentioned, this global monitoring concerns the entire Japanese territory. However, the frequency of the measurements and the sampling grids are adapted as the distance to the NPP decreases. In this way, measurements are concentrated within a radius of 250 km around the NPP, and are more and more intensified within radius going from 80 km to 20 km around the plant.

2.3. Various actors involved in the CRMP

Coordinated by the NRA, the CRMP brings together various actors involved in environmental monitoring. Among these actors, there are in the first instance the different ministries (Ministry of Environment - MOE, Ministry of Agriculture, Forestry and Fisheries - MAFF, Ministry of Education, Culture, Sports, Science and Technology - MEXT, Ministry of Health, Labour and Welfare - MHLW) which were chosen to supervise various environmental studies depending on their own specificities as well as the monitoring that they implemented before 2011.

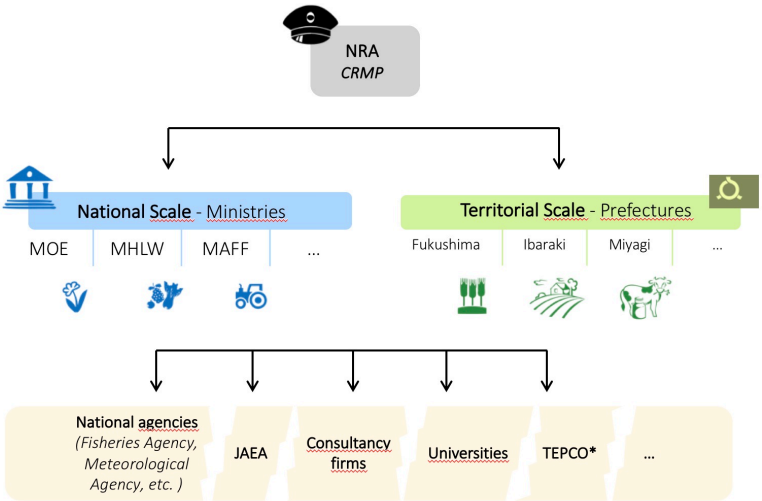
However, it is worth to mention that, in order to obtain adequate results and be able to transmit them to the NRA, the ministries convened additional actors much more qualified to carry out field works and environmental measurements. So, the ministries rely on:

- the support of national agencies and institutes (Japan Atomic Energy Agency - JAEA, National Institute of Environmental Studies - NIES, Forestry and Forest Products Research Institute - FFPRI);
- private providers (consulting firms) or public providers (universities);
- territorial administrations (prefectures) that focus on developing surveillance systems adapted to their territory, relying also on national agencies and institutions, or public and private providers.

Note that, considering the large number of actors convened for this CRMP, and in order to ensure the coherence and the harmonization of the obtained results, ministries and NRA have drown up since 2011 recommendations and guidelines, defining the methodology and prerequisites to respect.

The Figure 1, illustrates the previous paragraphs by showing the general organization of the national environmental monitoring system set up in Japan, since the Fukushima accident.

Figure 1 – Overall structure of the national environmental monitoring programme set up in Japan since the Fukushima accident.



However, from this general organization, a number of special cases has to be mentioned. First, it should be noted that for environmental monitoring in the 20 km radius around the Fukushima Daïchi plant, only TEPCO and the NRA are in charge of the field studies. It should also be noted that, in their own initiatives, national agencies and institutes (JAEA, AIST for example) have launched research programs dedicated to the evolution of the environmental quality of affected territories in parallel with the CRMP. Therefore, they are carrying out additional environmental measurements.

2.4. Publication of various results in the framework of the CRMP

NRA, as coordinator of the CRMP, is responsible for collecting all the results obtained by the different actors implementing environmental studies. Then, these results are published on the NRA website[1], which, for the occasion, has set up mapping tools for better visualizing the radiological quality of the Japanese territory. For the same purpose, the NRA also asked JAEA to publish the obtained results on a dedicated website[2] proposing various interactive maps with the latest measurement results (e.g. ambient dose rate, soil, water) over the whole of Japan.

In addition to this 'official dissemination', ministries also publish the results obtained as part of their own environmental studies on their own websites. These results are often disseminated as tables of data without specific explanation or information on the general trend observed. Territorial administrations (prefectures) also disseminate the results of their monitoring on their own websites. Again, these results can take various forms: from the 'simple' data table to an interactive map of ambient dose rates.

In conclusion, it is interesting to note that multiple actors, from national to territorial scales, are involved in the implementation of the CRMP. This organisation allows to obtain a precise monitoring of the radiological quality of the whole Japanese territory, by notably embracing all the environmental compartments.

However, it should be noted that multiple and abundant data are published by the different actors of the CRMP without seeking homogenization of the results or putting them into perspective. And this abundance of information, with often no explanations, may lead to some confusion for the users. Therefore, a better standardization and integration of these data seems necessary.

[1] http://radioactivity.nsr.go.jp/en/list/309/list-1.html
[2] https://emdb.jaea.go.jp/emdb/en/

3. LOCAL INITIATIVES TO MEASURE THE ENVIRONMENTAL QUALITY

3.1. Precisions concerning the environmental monitoring implemented by the prefectures

As mentioned above, the prefectures are in charge, in the framework of the CRMP, of the environmental monitoring of their own territory. Thus, this monitoring is carried out in all the municipalities belonging to the prefectures and concerns more particularly the aquatic compartment (drinking water, river and lake water and bathing water), the atmospheric compartment (fallout) and the measurements of ambient dose rate.

To ensure such follow-up, public agencies and institutes (JAEA for example), as well as universities support the prefectures. Then, this can lead to the construction of joint measurement laboratories, as is the case of the *Center For Environmental Creation*, inaugurated in Miharu (Fukushima Prefecture) in October 2015.

3.2. Environmental measurements set up by the local municipalities

Although prefectures ensure the environmental monitoring of each of their municipalities, many of these municipalities have launched, on their own, additional environmental studies. Thus, various universities or consultancy firms have been solicited directly by these municipalities to carry out radioactivity measurements on their local territories. Then, the results provided by these initiatives allow the municipalities to obtain a second expertise which completes the ones performed by the prefectures, the final objective being to check the veracity of the official environmental monitoring.

In addition to these studies, some municipalities also decided after the Fukushima accident to provide monitoring devices to their local populations, in order to give them the means to measure their own environment (*e.g.* ambient dose rate, foodstuff, *etc.*). This is how we can find around the Fukushima NPP some local municipalities which, for example, have developed their own contamination map of their local territory. It is worth to mention that, in many case, these local communities are often accompanied by Radiation Protection experts, who give them advises and try to respond to their expectations and concerns related to their daily life in affected territories.

3.3. Local citizen initiatives

In addition to the environmental monitoring implemented in the framework of the CRMP, or at the municipal scale, local initiatives developed by citizen networks or NGOs have been also implemented since March 2011.

These initiatives aim in particular to carry out additional measurements of the radiological quality of the environment in general (*e.g.* ambient dose rate, river, soils, *etc.*) or of the local foodstuff (*e.g.* fruits and vegetables from gardens, mushrooms, fish, *etc.*).

Concerning the NGOs, two main types of actions are proposed:

- Conduct independent measurement campaigns (soils, rivers, ambient dose rate) and disseminate the results on their websites and (sometimes) to the local municipalities;
- Propose measures (which have to be paid) of the radiological contamination of foodstuff brought by local citizens.

In general, these associations work thanks to private subsidies and contributions from their members. Quite often, these funds allow them to buy measuring devices.

It should be noted that, despite the multitude of NGOs committed to the radiological characterisation of the territories affected by the Fukushima accident, there is no significant sharing and networking of the results produced by these various associations. Indeed, each of these NGOs produces and publishes results on its own, with no attempt to compare them with the results obtained by other NGOs on the one hand, or by the official institutions on the other.

However, some exceptions can be highlighted and particularly concern the citizen networks. Indeed, based on the current digital progress, these networks have developed innovative approaches which try to better share environmental results, at least among their users. This is for example the case of the SAFECAST network[3], which offers to users to measure their environment using a mobile and connected device. The obtained results can be shared with the entire users' community through an interactive map, displaying the various results in real time. The approach of the 'Minna No Data Site'[4] also seeks to gather and disseminate results of foodstuff measurements produced by more than 30 independent laboratories.

4. MAIN LESSONS LEARNT

4.1. The multiplication of measurements and actors

The analysis of feedback experiences from the Japanese situation reveals that various actors, from local to national scale, intervene in the implementation of a post-accident environmental monitoring. At the national level, the CRMP -coordinated by the NRA- brings together more than twenty actors (ministries and national institutions) taking care of different types of environmental studies. The obtained results are then posted on the websites of these different actors, without adding comments or explanations that could facilitate their interpretation. This implies a profusion of raw data accessible from all sides, and which can be presented in very different formats depending on the platform (*e.g.* data table, real-time mapping, simple mapping, *etc.*). Thus, this lack of coherence and uniformity can cause confusion and loss of the users.

Besides, since the Fukushima accident, the government is facing a deep loss of confidence from local citizens and communities. Thus, private consultancy firms or universities are solicitated by local municipalities to carry out additional radiological measurements of their environment. The objective is to obtain counter-expertise results, even as these results are most often consistent with those produced within the CRMP. Generally, these data are neither exploited nor exchanged between neighbouring municipalities, nor even communicated to the public.

In addition to that, there are also the environmental measurements produced by the local NGOs, who do not seek to exchange their results with other NGOs, or with local municipalities or even with public institutions.

So, the current situation leads to a post-accident environmental monitoring involving multiple actors without a framework allowing them to share the results of their actions (with the exception of institutional actors involving in the CRMP). This leads to an environmental monitoring which is heterogeneous and sometimes redundant. *'Some places are measured 10 times by 10 different people while other places have never been measured since the accident*' says one of the interviewees, regretting the lack of coordination and information between actors of the environmental surveillance.

However, it is worth to highlight innovative approaches like SAFECAST networks or the Mina No Data Site project which seek to create a common and open database collecting results produced by independent citizens.

4.2. The data produced by local initiatives

In the context of mistrust towards authorities and official institutions since March 2011, citizens and local communities have developed their own radiological characterisation of the territory, resulting today in the production of abundant local data. However, the question of the scientific robustness of these data remains unsolved. In fact, citizens who carry out these measurements are not always trained in radioactivity measurement protocols. Although guidelines have been produced by the NRA and ministries, these recommendations remain generally unsuited to the local communities, which do not have the adequate means. In this way, the quality and veracity of these measures can be questioned. However, these results have all the confidence of the local populations. Therefore, even without

[3] https://blog.safecast.org/
[4] http://en.minnanods.net/

scientific robustness as such, these data represent for citizens the information on which they adapt their behaviour.

Also, it is important to have in mind that these local data could represent for researchers a rich and interesting source of information to understand the evolution of the radiological state of the environment at the local scale and for the radiation protection experts a crucial source for favouring the involvement of local stakeholders in the recovery programme.

4.3. What is the evolution of the Japanese environmental monitoring system?

The multiplication of the measurements carried out on the territory affected by the Fukushima accident is an element admitted and recognized by many actors involved in the environmental monitoring process. The argument that, the radiological quality of the environment follows unsurprisingly the radioactive decay of caesium 134 and 137, could argue in favour of a decrease in the frequency of environmental measurements. NGOs that we have interviewed also acknowledge that visits from local inhabitants are decreasing. As a result, some NGOs have closed down their local offices because of a lack of attendance.

In this context, the question of the medium to long-term evolution of the environmental measurements can be raised, as well as the sustainability of the current system. Some interviewees believe that it is up to local citizens to decide about the future of the environmental monitoring: *'when people will feel safe at home and they will no longer need environmental measurements to reassure themselves, environmental surveillance could be reduced. In the meantime, we must continue'.*

5. GENERAL CONCLUSION AND PERSPECTIVES

Seven years after the accident at the Fukushima plant, the analysis of the post-accident environmental monitoring implemented in Japan reveals the multiplication of actors from local to national scales. If - through the CRMP - the national surveillance system seems coherent and complete, the abundance of results posted online can cause some form of confusion. It might be interesting to accompany each publication of results with comments explaining the observed trend.

At the local level, the mistrust towards authorities and official institutions has induced citizens and local communities to implement their own monitoring, which leads today to the production of abundant local data which can represent a very interesting source of information.

In this context, the remaining issue consists in knowing how to go towards a better sharing between results produced by institutional and non-institutional actors. It appears that scientific experts, often involved in both sides, could play a key role in sharing these results, which represents a strong lesson learnt for the preparedness phase.

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Development and Current Status of a Carborne Gamma-ray Survey System, KURAMA-II

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Abstract. A carborne γ-ray survey system, named KURAMA (Kyoto University RAdiation MApping system), was developed as a response to the nuclear accident at TEPCO Fukushima Daiichi Nuclear Power Plant in 2011. The system has evolved into KURAMA-II, characterized by its compactness, ruggedness, autonomous operation, and acquisition of pulse-height spectrum data. A large number of in-vehicle units of KURAMA-II have served for various types of monitoring activities in Fukushima prefecture and eastern Japan. More than 100 vehicles with KURAMA-II deployed for periodical monitoring activities throughout eastern Japan. A monitoring scheme in residential areas on a long-term basis with KURAMA-II installed on vehicles continuously operated in residential areas, such as local buses has been conducted as a joint project among Kyoto University, JAEA, and the Fukushima prefectural government. Developments for other purposes, such as prompt soil-contamination estimation scheme using KURAMA-II, are on the way. In this paper, the outline and the current status of KURAMA-II along with some results from its applications are introduced.

Keywords : radiometry, mapping, γ-ray, carborne survey, air dose rate, soil contamination, Fukushima Daiichi nuclear power plant

1 INTRODUCTION

The magnitude-9 earthquake in eastern Japan and the following massive tsunami caused a serious nuclear disaster for the Fukushima Daiichi nuclear power plant. Serious contamination by radioactive isotopes was caused in Fukushima and surrounding prefectures, but the existing radiation-monitoring schemes were incompetent for this situation due to damage and chaos caused by the earthquake.

KURAMA [1] was developed to overcome difficulties in radiation surveys and to establish air dose-rate maps during and after the incident. The design of KURAMA was intended to enable a large number of in-vehicle apparatuses to be prepared within a short period of time by using consumer products. The data-sharing scheme based on cloud technology has enabled high flexibility in the configuration of data-processing hubs or monitoring cars. KURAMA has been successfully applied to various activities concerning radiation measurements and the compilation of radiation maps in Fukushima and surrounding areas.

As the situation became stabilized, the main interest in measurements moved to the long-term (several tens of years) monitoring of radiation from radioactive materials remaining in the environment. KURAMA-II [2] was developed for such purpose by introducing the concept of

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continuous monitoring from vehicles moving around residential areas, such as local buses and postal motorcycles. The ruggedness, stability, autonomous operation and compactness were well taken into consideration in its design, and an additional measurement capability of pulse-height information along with location data was also introduced thanks to a USB bus-powered CsI(Tl) detector from Hamamatsu Photonics. KURAMA-II has been successfully introduced to the continuous monitoring by vehicles in residential areas, as expected, and now started being applied to other applications, such as the rapid estimation of Cesium density in the soil of farmland.

In this paper, the outline and the current status of KURAMA-II along with some results from its applications are introduced.

2 KURAMA-II

2.1 In-vehicle Unit

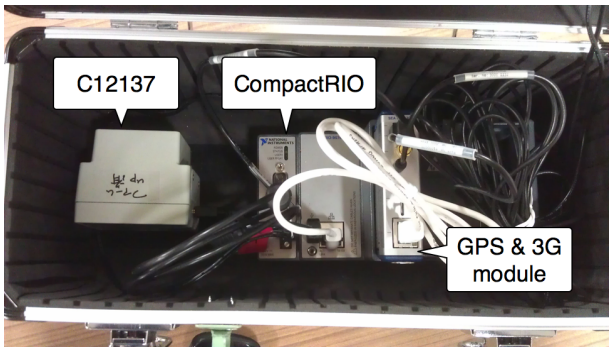
KURAMA succeeded in the simultaneous radiation monitoring extended over a wide area, in contrast to other conventional carborne survey systems failed due to the lack of scalability.

As the situation changed from an emergency to a steady state, The cost of conducting carborne surveys of KURAMA, such as manpower for drivers and operators in monitoring vehicles, became serious problem for contiuing monitoring activities for a period for over tens of years. Then, the concept of KURAMA-II, i.e., autonomous and continuous measurements by KURAMA units on vehicles operated for other purposes in residential area, is proposed to realize a sustainable radiation monitoring.

To obtain sufficient ruggedness, stability, compactness and autonomous operation feature for such long-term and continuous monitoring, the in-vehicle part has been totally redesigned based on CompactRIO series of National Instruments [3] and C12137 series, a CsI(Tl) detector series of Hamamatsu Photonics [4]. All of the components of the in-vehicle part are placed in a small tool box (34.5 cm × 17.5 cm × 19.5 cm) made of wood covered with thin aluminum sheet for the better handling (Fig. 4).

Figure 1: (a) In-vehicle part of KURAMA-II. (b) A typical installation of in-vehicle unit under operation. The in-vehicle unit is always placed on the right side of the rear part in a car so that the in-vehicle unit is in the center part of the road.

(a)



(b)

2.2 Radiation Detection in KURAMA-II

The radiation-detection part of KURAMA-II is the C12137 series by Hamamatsu Photonics, a CsI(Tl) detector series characterized by its compactness, high efficiency, direct ADC output

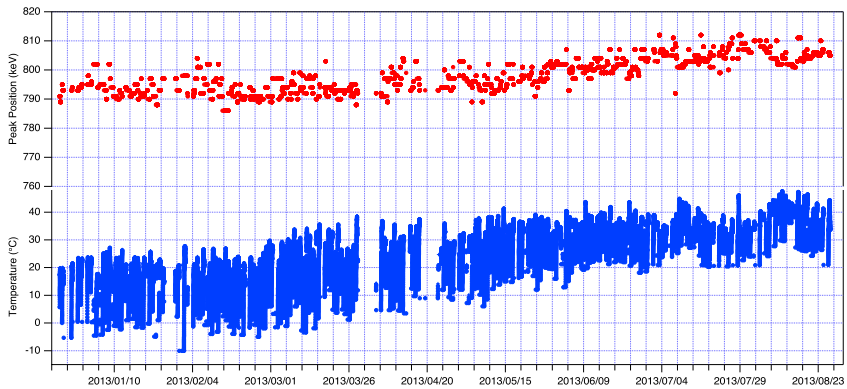
and USB bus power operation. The specifications of the C12137 series are summarized in Table 1. The ambient air dose rate, $H^*(10)$, for each measurement point is calculated from the pulse height spectrum obtained for each measurement point by using G(E) function method [5][6]. Details concerning the G(E) functions and the characteristics of KURAMA-II on radiation detection are available in ref. [7].

Table 1: Specification of the C12137 Series [4] used for KURAMA-II. Three types of C12137 series are usually used, depending on the measurement conditions, such as the expected range of air dose rate.

Type	C12137-00	C12137-01	C12137-10
Scintillator	CsI(Tl)		
Scintillator Size	$13 \times 13 \times 20$ mm	$38 \times 38 \times 25$ mm	$\phi 110 \times 25$ mm
Energy Resolution (662 keV)	8 %	8.5 %	10 %
Efficiency (662 keV, 0.01 μSv/h)	40 cpm	400 cpm	2000 cpm
Detecting Device	MPPC		

The 796 keV peak of ^{134}Cs, typically observed as a well-isolated peak in Fukushima area, is monitored for detecting possible gain shifts during its operation. Fig. 2 is the peak drift of 796 keV γ-ray of ^{134}Cs and the temperature at MPPC for an in-vehicle unit of KURAMA-II placed in a local bus operated at Fukushima city from the beginning of winter to the end of summer in 2013. Regardless of the large variation of the temperature under a tough environment, the peak drift observed was at most 3%, corresponding to 5% of the drift in the air dose rate, i.e., one-third of the tolerance for typical portable survey meters used for the air dose rate measurements in Fukushima.

Figure 2: Observed peak drift of 796 keV γ-ray of ^{134}Cs along with the temperature inside the detector during the feasibility test of KURAMA-II in Fukushima city in 2013. The temperature of the detector drastically and frequently varied due to on/off of the air-conditioning system even within each run.



2.3 Data Handling in KURAMA-II

In-vehicle units of KURAMA-II are based on VxWorks, one of the operating systems that client softwares for typical cloud data sharing systems don't support. Therefore, another file transfer protocol of KURAMA-II has been designed to send data without any loss under the poor coverage of the mobile network expected in emergency situations, as well as to comply to the standard protocols that are widely used in today's networks, such as Web Services.

In this protocol, two timestamped files of the air dose rates and the pulse-height spectrums are separately produced for every three measurement points as data files. A timestamped file for air dose rate is a csv file including of Date/Time, location data, the air dose rate, the temperature of the detector. As for pulse-height spectrum data, a 32-bit binary file, including Date/Time, location data, and the series of pairs consisting of an ADC channel and its count number for the ADC channels of non-zero counts, is generated.

Generated data files are transferred to a remote "gateway server" by the POST method. All communications between in-vehicle units and a remote "gateway server" are based on RESTful API. Unsent files are archived inside an in-vehicle unit as a single zip file and these are sent as soon as the network connection is recovered.

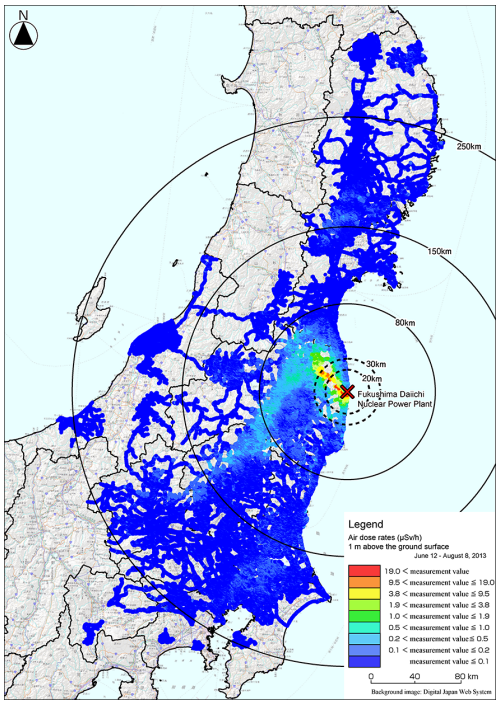
Once data files are received from in-vehicle units by the "gateway server", two files, a data file for the air dose rate and that for the pulse height spectrum, are updated. Those files are shared by remote servers using the same scheme using Dropbox, as was done in the original KURAMA. Now we are under the field test of a new cloud data storage service based on ownCloud especially for KURAMA-II, allowing the full authority of the cloud system to the users for the immediate recovery and easier operation of the system under emergency situations.

3 APPLICATIONS OF KURAMA-II

3.1 Periodical Survey by Japanese Government

KURAMA-II has been introduced by MEXT since March 2012 (Fig. 3) [8], as a successor for periodic carborne surveys in eastern Japan started in June 2011[9]. In each survey, around

Figure 3: Periodic carborne surveys in eastern Japan have been performed by MEXT and NRA. The results are available from the interactive web sites [10] [11].



one hundred KURAMA-II were deployed to the local municipalities in East Japan. Now these periodical surveys are currently continued by Nuclear Regulations Authority, Japan (NRA). Their measurement results are released to the public through a web site [10] [11],

and accumulated knowledge and procedures regarding the radiation monitoring by carborne surveys using KURAMA-II have been summarized and standardized by NRA [12] for the prompt radiation monitoring activities in emeregncy.

3.2 Continuous Monitoring by Local Buses

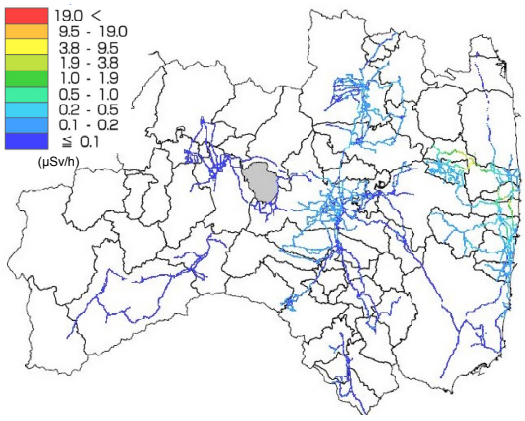
One of the characteristic applications of KURAMA-II is continuous monitoring using local buses and other vehicles continuously operated for other purposes.

The first field test was started in Fukushima city at the end of December 2011, then other major cities in Fukushima, i.e., Koriyama city, Iwaki city, and Aizuwakamatsu city, were also covered in this field test since the end of December 2012.

Now the monitoring scheme by local buses has been performed as an official project by the Fukushima prefectural government under the collaboration of Kyoto University and JAEA. More than fifty in-vehicle units of KURAMA-II deployed to local buses and other vehicles, and both realtime data and the summarized results on a weekly basis are released to the public (Fig. 4(a)) [13] [14]. The results from this measurement are quite useful for practical purposes, such as confirming the effect of decontamination in residential areas (Fig. 4(b)) [15].

Figure 4: (a) Radiation map of continuous radiation monitoring in residential areas by local buses etc. from 14 to 20 Jan. 2018, released from Fukushima prefecture. These kinds of maps as well as kml files are released to the public on a weekly basis via the web page of Fukushima prefecture [14]. (b) A typical example of the decontamination effects observed through monitoring by KURAMA-II on a local bus [15].

(a)



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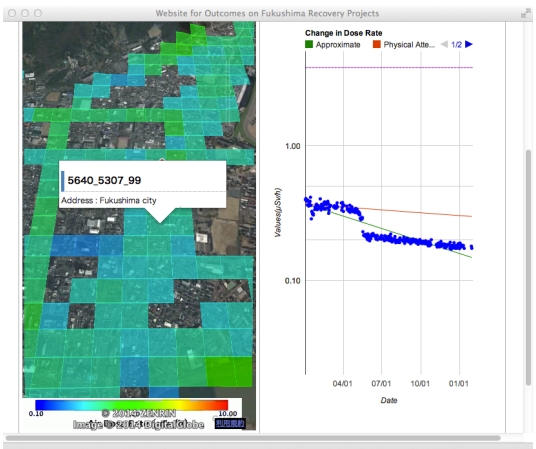
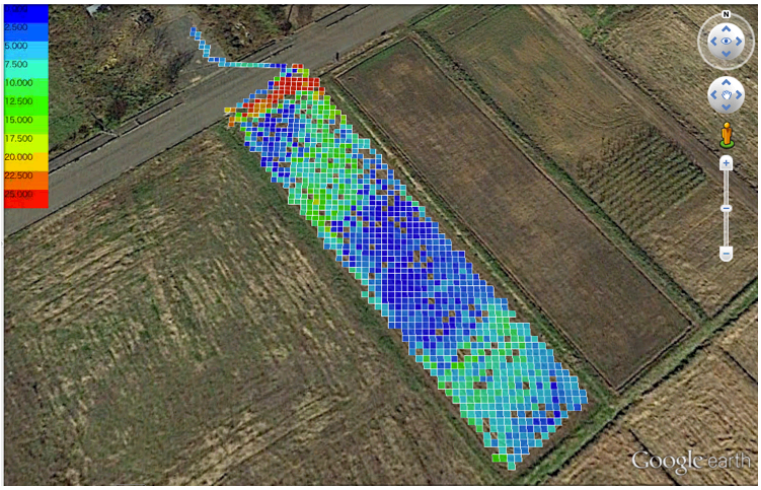


3.3 Other Applications

The characteristics of KURAMA-II enable it to be applied to various activities. One of the important application is the evaluation of soil contamination of farmlands [16]. The differential measurement of two detectors, one is collimated towards the ground surface and the other is nondirectional, is performed to subtract the contribution from the surrounding from the air dose rate detected by the collimated detector. A typical example is shown in Fig.5. This is an experimental rice field decontaminated by removing surface soil at 5 cm thick and by the soil dressing. Even though this field was confirmed to be "decontaminated" by confirming the air dose rate 1 m above the ground to be less than 0.23 μSv/h, a large variation of remaining soil contamination from less than 1 kBq/kg up to 15 kBq/kg was observed by the present method, and then confirmed by the measurement by soil sampling. We have been working

Figure 5: A typical example of soil contamination measurement performed at an experiment rice field near the TEPCO Fukushima Daiichi Nuclear Power Plant. This rice field was already decontaminated by removing the surface soil and soil dressing. A large variation of remaining soil contamination from less than 1 kBq/kg (blue) up to 15 kBq/kg (green) was observed. The red area at the top end is outside the field and no decontamination was performed. This result corresponds to the result from the precise measurement by Ge detector using the soils sampled from this field.



for this method to practically apply the other types of farmlands under the collaboration with Fukushima Agricultural Technology Centre, and plan to propose as a new standard method to confirm the decontamination of farmlands.

Another application is the walk survey at road shoulder of Joban Expressway near TEPCO Fukushima Daiichi Nuclear Power Plant. This measurement has been performed to establish a reasonable estimation of possible exposure during going through this section or under evacuation from possible car accident in this section. We have found that a large differences in reconstructed pulse height spectra in the points whether the expressway was constructed before or after the accident (Fig. 6). This result can be understood whether ^{134,137}Cs stick to the pavement of the road is the dominant source or not. This may affect the reduction rate of air dose rate in these sections. We are now continuing the survey to track the changes of pulse height spectra and air dose rates, that should be important for the prediction of air dose rate in future.

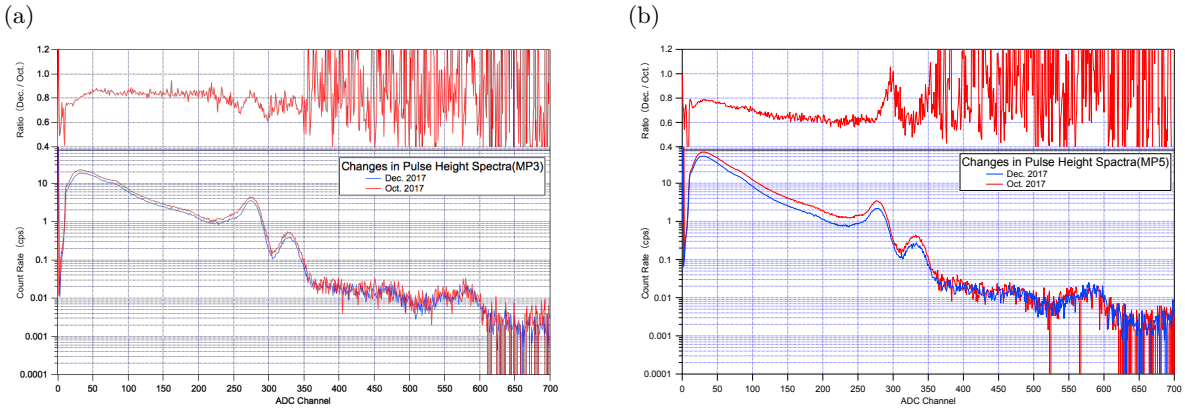
4 CONCLUSION

KURAMA-II, an advanced version of KURAMA, characterized by its ruggedness, flexibility, compactness, autonomous operation, and pulse height spectrum measurement, has been developed to enable long-term monitoring of the air dose rate in residential areas. KURAMA-II serves carborne surveys for periodical air dose rate monitoring in eastern Japan or continuous monitoring of residential areas in Fukushima prefecture. These methods and accumulated knowledge are now standardized by NRA for the prompt monitoring activities in nuclear disaster. Other applications, such as contamination surveys in farmland or the road shoulder survey in expressways, are on the way.

5 ACKNOWLEDGEMENTS

The authors are grateful to Dr. Mizuno, Mr. Abe, Mr. Kimura, Mr. Ito and Mr. Kojima of the Fukushima prefectural government for continuous support to field tests of KURAMA and

Figure 6: Pulse height spectra obtained in the section that was constructed (a)before the accident (MP3) and (b)after the accident (MP5). Pulse height spectrum in MP5 has higher counting rate in lower energy than, and the reduction of counting rate is more rapid than MP3. This can be understood that ^{134,137}Cs sticking to the pavement of the road is dominant source in the case of MP3, while the components from the unpaved area surrounding the road is dominant in MP5.



KURAMA-II and the establishment of a monitoring scheme based on KURAMA-II on city buses. Technical support regarding LabVIEW and CompactRIO was served by National Instruments Japan Corporation under a support program, named as "Monotsukuri Fukkou Shien Project" by National Instruments Japan Corporation, aiming to help in the recovery from the Great East Japan Earthquake. Mr. Muto and Mr. Inomata at Fukushima Transportation Inc., Mr. Suzuki at Shin Joban Kotsu co., ltd., and Mr. Sugihara and Mr. Ishikawa at Aizu Bus Co. Ltd. for their cooperations in the field test of KURAMA-II on city buses. This work was partly supported by the distribution-mapping projects organized by Nuclear Regulation Authority, Japan.
Finally, the authors would like to express their gratitude to Mr. and Mrs. Takahashi and the staff members at "Matsushimaya Ryokan", an inn at Iizaka hot spring in Fukushima city, for their heart-warming hospitality and the offer of a foothold for our activities in Fukushima, regardless of their severe circumstances due to the earthquake and the following nuclear accident.

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Availability and reliability of meteorological data for atmospheric dispersion models

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Abstract. Atmospheric dispersion calculations are widely used in the nuclear industry, not only for environmental impact assessment of normal operation and safety analysis for DBC (design basis conditions) or DEC (design extension conditions), but also for decision support systems for nuclear emergency response. Appropriate information about the source term and adequate meteorological data are essential for these calculations. Different methods and numerical models can be used for the different purposes, and also various input data are applicable for the calculations. Assuming that the source term is known, meteorological data are necessary to specify radioactivity distribution in the environment and to provide adequate dose consequences for areas affected by the emission. The most important meteorological data for activity dispersion calculations are wind speed and wind direction, precipitation, atmospheric stability and height of atmospheric boundary layers. These meteorological data may originate from local measurements or from numerical weather prediction data. These databases have different spatial and time resolution, accuracy and also credibility of the data varies. Data available in different databases, reliability of the data and their consequences on the results of dose calculations will be presented; advantages and disadvantages of these data will be discussed.

KEYWORDS: atmospheric dispersion, meteorological data, radioactivity, environmental and public dose, uncertainty, sensitivity analysis, Gaussian puff model, nuclear safety

6. INTRODUCTION

Atmospheric dispersion calculations are used for a wide variety in the nuclear industry for certain assessments, such as environmental impact assessment of normal operation and safety analysis for DBC (design basis conditions) or DEC (design extension conditions), as well as for decision support systems for analyzing the environmental consequences in the case of nuclear accidents. Atmospheric dispersion model predictions are associated with uncertainties though, which should be identified and quantified in order to interpret the outcomes of the atmospheric dispersion modeling, evaluate the reliability of the activity dispersion calculations, and increase the credibility of the results in decision-making.

Uncertainty in atmospheric dispersion model calculations is associated with several uncertain components. Uncertainties in the given dispersion model, resulting from the inadequate treatment and simplifications of the modeled processes, numerical approximations and misspecifications of the model structure. Further contributor to the overall uncertainty in dispersion modeling is the parameter uncertainty, which is a result of the uncertainties in the input data, the model parameters and initial and boundary conditions. The stochastic uncertainty term in atmospheric dispersion modeling arises from the natural variability and non-predictive capability of the atmosphere [1, 2].

This paper discusses the consequences of the uncertainties related to the meteorological parameters on the results of the activity dispersion calculations. In order to assess the effect of variations in model input values on the model output, a sensitivity analysis was used. Sensitivity analysis enabled the identification of the meteorological parameter with the greatest importance on the output of the atmospheric dispersion modeling, as well as the quantification of the impact of the changes in the evaluated meteorological data between the typical uncertainty ranges on the dispersion calculation's results.

7. METHODS

7.1 Availability and uncertainties of meteorological variables

For activity dispersion calculations, meteorological variables like wind direction, wind speed, atmospheric stability, precipitation, height of the atmospheric boundary layer are required. The accuracy of the atmospheric dispersion and deposition calculations is highly dependent on the

representativeness and availability of the input meteorological data. The data of the meteorological variables may originate from different sources, from meteorological measurement and observations, and numerical weather prediction (NWP) models. The World Meteorological Organization (WMO) made several recommendations and general requirements regarding these meteorological measurements and numerical weather predictions according to [3-5] (see in Table 1 and Table 2).

If both sources of meteorological data are available it is beneficial to compare their uncertainties. In case of emergency response and decision support, measurement data for the meteorology is usually not available (at least in the early phase) so only the data provided by the numerical weather prediction models can be used. Based on the performance requirements, it can be stated that the uncertainties of the wind field's parameters are typically lower for measurements and observations than for the numerical weather prediction models. However the numerical weather prediction modeling may enable lower uncertainty in the determination of precipitation intensity values. Concerning the availability of the meteorological data, the time resolution in the case of measurements and observations is typically better than in the case of numerical weather prediction models. Synoptic meteorological stations are accurate for small-scale measurements and observations, while with the use of numerical weather prediction models the spatial variation of meteorological variables can be also taken into account. Numerical weather prediction models may be considered in the case of lower station densities, since the special resolution of the numerical calculation makes it possible to obtain meteorological values at isolated locations where the number of installed measurement or observation station, thus the measurement and observation frequency is limited.

Table 3: The availability and uncertainty requirements of meteorological parameters from measurements and observations [3].

Measurements and observations				
Meteorological variable	Availability		Uncertainties less than	
	Resolution	Time resolution		
Wind direction	1°	2 and/or 10 min	± 5°	
Wind velocity	0.5 m/s	2 and/or 10 min	≤ 5 m/s > 5 m/s	± 0,5 m/s ± 10%
Precipitation intensity	0.1 mm/h	10 min	0.02 – 0.2 mm/h 0.2 – 2 mm/h > 2 mm/h	n.a. ± 1 mm/h ± 5%

7.2 Atmospheric dispersion modelling

To evaluate the effect the data uncertainties of the used meteorological parameters have on the result of atmospheric dispersion modeling a sensitivity analysis of the Gaussian puff model in the SINAC software [6] was conducted. The SINAC (Simulator Software for Interactive Modelling of Environmental Consequences of Nuclear Accidents) code system was developed in the Centre for Energy Research of the Hungarian Academy of Sciences and is used at the Centre for Emergency Response, Training and Analysis (CERTA) of the Hungarian Atomic Energy Authority (HAEA) as a decision support system for analyzing the environmental consequences in case of an accident at a nuclear power plant. The software models the atmospheric dispersion, the dry and wet deposition of radioactive contaminants and calculates the dose consequences of the public for the exposure pathways of cloudshine, groundshine, inhalation and ingestion. The calculation can take into account different countermeasures like sheltering, evacuation or iodine tablets.

Table 4: The availability and uncertainty requirements of meteorological parameters from regional numerical weather prediction models [4].

Regional numerical weather prediction models				
Meteorological variable	Availability		Uncertainties less than	
	Spatial resolution	Time resolution		
Wind field	10 km (horizontal) 0.1 km (vertical)	1 h	troposphere stratosphere	± 2 m/s ± 3 m/s
Precipitation intensity	10 km (horizontal) 0.1 km (vertical)	1 h	± 0.1 mm/h	

The atmospheric dispersion in SINAC is calculated with a Gaussian puff model [7], in which the release of the contaminants is assumed to occur in separate packages, puffs. The activity concentration of the puffs is considered to have a Gaussian distribution, and the model takes into account the physical processes of advection and turbulent diffusion. The spread of the puffs are described by the atmospheric dispersion parameters of which the downwind and horizontal crosswind dispersion parameters are assumed to be equal ($\sigma_x = \sigma_y = \sigma_r$) in SINAC. In the software, the atmospheric dispersion parameters are dependent on the atmospheric stability and are calculated based on the Pasquill stability classes. The vertical Gaussian distribution of the activity concentration of the puffs are modified by the atmospheric boundary layer and the ground, which is computed in the SINAC software with reflection corrections. The meteorological data used in the code system are from the AROME numerical weather prediction model [8] provided by the Hungarian Meteorological Service (OMSZ).

7.3 Sensitivity analyses

Parametric sensitivity analysis was conducted with the Gaussian puff model of the SINAC code to evaluate the influence of the meteorological input uncertainties on the atmospheric dispersion calculation results. The values of each meteorological parameters were varied separately and the change in the air activity concentration of ^{137}Cs and ^{135}Xe was investigated. The doses can be computed from the environmental activity concentration by multiplication with the dose conversion factor, which do not affect the uncertainties resulting from the varied meteorological parameters. The default meteorological values and their perturbations are summarized in Table 3. The meteorological values were considered to be constant throughout the dispersion calculation. The release quantity was 10^{10} Bq for both nuclides, the effective release height was 120 m and the release duration was 1 hour. The results of the time integrated air activity concentrations were calculated at receptor points along the original 180° wind direction and perpendicularly at the distance of 3 km with 1 m height from the ground. The time resolution of the results was 15 minutes and the time interval of the integration was 6 hours.

Wind velocity. In the sensitivity analysis, the wind velocity was varied between the minimum and maximum values of 1 and 12 m/s, with the default value of 3 m/s. The relationship between the atmospheric stability and the wind direction was not investigated, the Pasquill stability D category was fixed during the assessment of the wind velocity uncertainty.

Wind direction. The sensitivity of the atmospheric dispersion towards the change in wind direction was evaluated with the maximum perturbation of 10° with the default direction of 180°.

Table 5: The summary of the default values and perturbations of the meteorological parameter for the sensitivity analysis.

Meteorological variable	Default value	Perturbation
Wind velocity [m/s]	3	-2, -1, +2, +5, +9
Wind direction [°]	180	+1, 3, 5, 7, 10
Pasquill stability category	D	A, B, C, E, F
Precipitation intensity [mm/h]	0	+1, 3, 5, 8, 12

Atmospheric stability. The atmospheric stability can be derived from different meteorological variables such as the wind velocity, the radiation flux, the cloud cover, the temperature gradient or the horizontal fluctuation of the wind direction, depending on the given atmospheric stability classification scheme. In the sensitivity analysis instead of investigating the change of these meteorological parameters, the Pasquill categories were changed with keeping all the other parameters fixed. This is justified because for the 3 m/s default wind velocity, corresponding radiation flux and temperature gradient values can be found for all 6 types of Pasquill categories.

Precipitation intensity. To evaluate the effects of the precipitation intensity uncertainty, different values of rain intensity were considered to be constant through the atmospheric dispersion. As the rain does not affect the air activity concentration of noble gasses, only the result for ^{137}Cs were examined for this part of the analysis.

8. RESULTS

The results of the sensitivity analysis show that the change of the wind velocity caused about one order of magnitude maximum difference in time integrated air activity concentration for a distance up to 20 km. Further than that, the difference between the time integrated activity concentration for the slower and the faster wind increases rapidly, as with the slower wind the contaminants do not reach these distances by the end of the first 6 hours of the release (Figure 1). When changing the wind direction, we find that the air activity concentration along the original plume centerline decreases (Figure 2) and the maximum ratio of the results for 0° and 10° wind direction decreases with the distance from the release point as well.

This is the result of the shift of the plume center with changing wind direction, thus the air activity concentration decreases based on a Gaussian function. Figure 3 shows the spread of the plume perpendicular to the wind direction for different atmospheric stability categories at 3 km distance from the release point. For more stable atmosphere, the plume is more dense with higher maximum air activity concentration values, and for instable conditions the plume is more spread out and therefore, the maximum air activity concentration values are lower. Concerning the rain intensity, the calculation indicates that the more intensive the rain is, the less the air activity concentration will be shown in Figure 4.

The maximum ratio of the results due to the different meteorological perturbations shown in Table 4, were combined with the uncertainties of the different meteorological dataset to describe the overall uncertainties of the atmospheric dispersion modeling. The error for the ground measurements of wind velocity is ±0.5 m/s below 5 m/s, and 10% above. So using this data source for the dispersion modeling would cause about 10% uncertainty in the activity concentration results, since an inverse relationship exists between the air activity concentration and the wind velocity (in the Gaussian puff model). Using wind velocity values from numerical weather prediction data, which has a bias of 2-5 m/s compared to measurements, can result in more than a 50% uncertainty. The ground measurement error for the wind direction is ±5°, which can produce an uncertainty of 50-80% in the air activity concentration.

Figure 5: The time integrated air activity concentration of ^{135}Xe along the original centerline of the plume for different wind velocities.

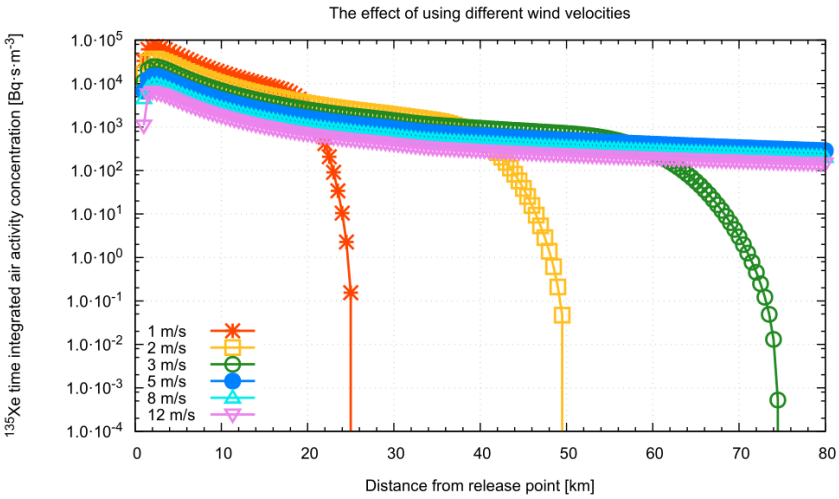
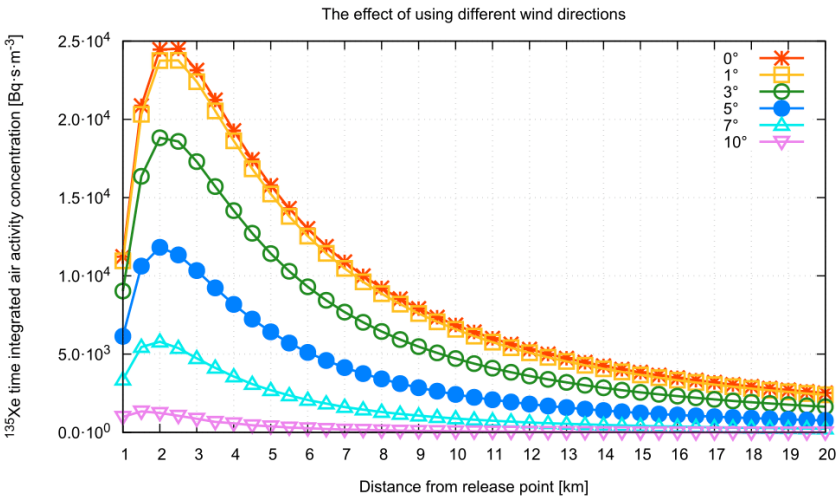


Figure 6: The time integrated air activity concentration of ^{135}Xe along the original plume centerline for different wind directions.



Our analysis yields that close to the release point (1-5 km), the atmospheric stability has biggest effect on the time integrated air activity concentration. One category difference in Pasquill stability category results in two order of magnitude change in the case of stable atmosphere at a given receptor point. To evaluate the effect of rain intensity, the results for 1 and 12 mm/h rain intensity were compared, which resulted in a maximum difference of three orders of magnitude at a distance of 20-30 km from the release point. The error of ground measurements of rain intensity is around ±0.1 mm/h below 2 mm/h, and ±5% above. This leads to a couple percent of the time integrated air activity concentration uncertainty close to the release point. When using numerical weather prediction models as a data source of rain intensity data which has an average bias of 0.1 mm/h, the uncertainty of the time integrated air activity concentration result is similar to uncertainty obtained with using measurement data. We note that with numerical meteorological models, the predicted rain intensity value is averaged over an area and is not representative for specific locations, thus when using the rain intensity results of numerical weather prediction calculations for atmospheric dispersion, this aspect should be considered carefully.

Figure 7: The time integrated air activity concentration of ^{135}Xe at 3 km from the release point perpendicular to the plume centerline for different Pasquill stability categories.

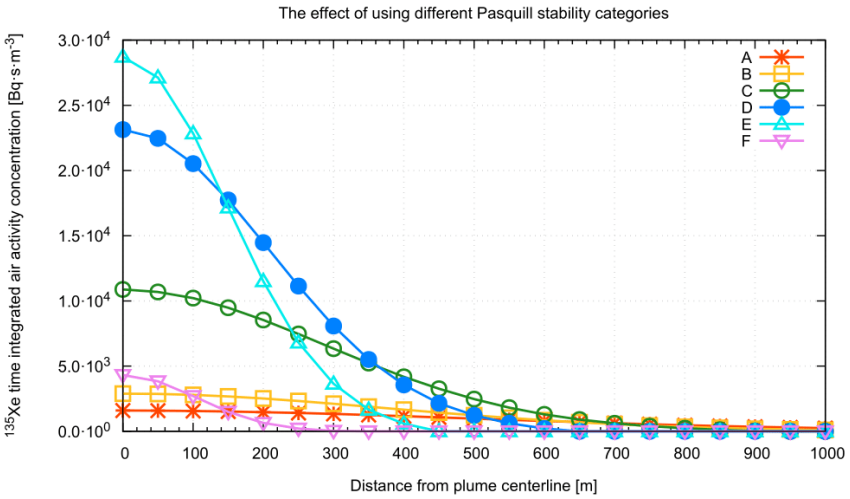


Figure 8: The time integrated air activity concentration of ^{137}Cs along the original plume centerline for different precipitation intensities.

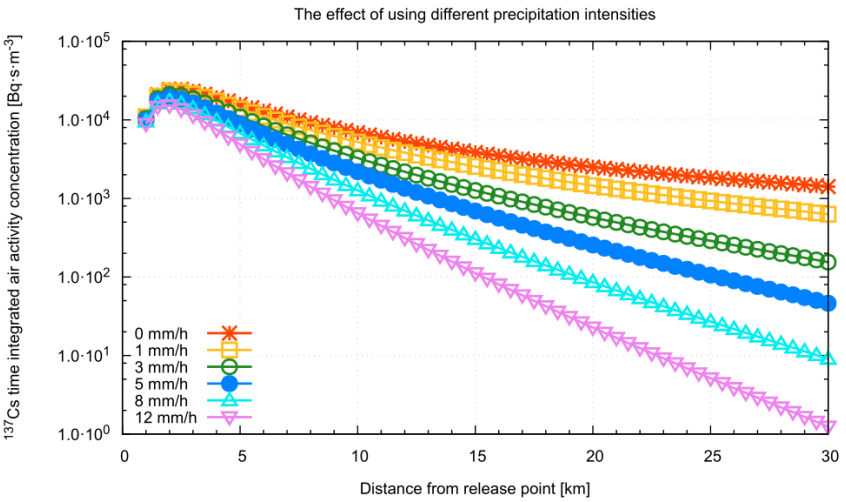


Table 6: The maximum ratio of the calculated air activity concentration at the original plume centerline due to the perturbation of meteorological parameters.

Difference of the evaluated meteorological parameters	The maximum ratio of the air activity concentration in the original plume centerline for different perturbations		
	Distance from the release point [km]		
	1-5	5-20	20-30
Wind velocity: 11 m/s	$9\cdot10^{-2}$	$1\cdot10^{-1}$	$3\cdot10^{3}$
Wind direction: 10°	$6\cdot10^{-2}$	$1\cdot10^{-3}$	n.a.
Pasquill category: 1 category	$1\cdot10^{-2}$	$4\cdot10^{0}$	$3\cdot10^{0}$
Precipitation intensity: 11 mm/h	$7\cdot10^{-1}$	$2\cdot10^{-2}$	$2\cdot10^{-3}$

9. CONCLUSIONS

In this paper, the effect of using different meteorological input data – with different uncertainties – on the results of atmospheric dispersion modeling was investigated. Two main sources of meteorological data, the meteorological measurements including observations and numerical weather prediction models were assessed to summarize the parameter uncertainties and availability of the wind direction, the wind velocity and the precipitation intensity parameters.

To examine the effect of the meteorological parameter uncertainties on the results of atmospheric dispersion modeling, a sensitivity analysis was carried out for the Gaussian puff model of the SINAC decision support system. The time integrated air activity concentration of ^{135}Xe and ^{137}Cs were calculated for different perturbations of wind velocity, wind direction, Pasquill stability category, and precipitation intensity. The maximum change of the activity concentrations for different meteorological input values were computed and combined with the uncertainty of the meteorological data to characterize the overall uncertainties of the atmospheric dispersion calculation.

To conclude, the advantage of using the measurement and observation data is that the uncertainty of wind velocity and the precipitation intensity for values below 2 mm/h is lower than that of the numerical weather prediction models, thus the uncertainty of the resulting air activity concentrations will be low as well. Furthermore, the measurements describe the real conditions of the atmosphere. The disadvantage of using meteorological values from measurements and observations is the spatial sparseness of the measurement stations and possibility of errors due to observations not being representative of the region in which they are located. The advantage of using meteorological data from numerical weather prediction models is the good spatial coverage of the area of the dispersion, and the disadvantage is that the attained data is only a prediction of the actual processes.

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NERIS roadmap for further research development on preparedness for nuclear and radiological emergency response and recovery

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Abstract. Following the Chernobyl accident, European research programmes have been set up to further investigate and improve nuclear emergency and recovery preparedness and management. Created in 2010, NERIS platform aims at fostering the cooperation in this field. Currently, more than 60 organisations are members of this European platform with 26 supporting organisations, members of the legal association. The main objectives of the platform are to: improve the effectiveness and coherency of current approaches to preparedness, identify further development needs, improve know-how and technical expertise, and establish a forum for dialogue and methodological development. The aim of this paper is to present the first roadmap adopted by NERIS taking into account the latest developments and the preliminary lessons learned following the management of the Fukushima accident. Three main challenges have been identified for the NERIS roadmap: 1) Challenges in radiological impact assessment during all phases of nuclear and radiological events; 2) Challenges in countermeasures and countermeasure strategies in emergency & recovery, decision support and disaster informatics; 3) Challenges in setting-up a trans-disciplinary and inclusive framework for preparedness for emergency response and recovery.

1. INTRODUCTION

The NERIS research platform has been created in 2010 in order to further improve preparedness for emergency and recovery management of nuclear accidents. Currently, 63 organisations are members of the NERIS platform with 26 supporting organisations being members of the legal association. Research programmes have been set up after the Chernobyl accident covering emergency and recovery phases. For the emergency phase the main goals are to facilitate regaining control on the situation and to prevent or mitigate consequences for people and the environment by implementing early countermeasure strategies, with a focus on decision-aiding systems. For the recovery phase, the main goals are to develop protection strategies coping with long lasting contamination of the environment and integrating societal and economic considerations for the future of the life in the affected territories. The management of the consequences of the Fukushima accident highlighted the importance of further improving preparedness taking into account the recent evolution of the risk assessment and management approaches, the role of social media, as well as the development in social sciences and humanities.

In November 2017, NERIS has published its first research roadmap addressing the key challenges for improving emergency and recovery preparedness and response [1]. Three main challenges have been identified for the updated NERIS strategic research agenda (adopted in December 2017) [2]. These challenges are developed in the first NERIS roadmap: 1) Challenges in radiological impact assessment during all phases of nuclear and radiological events; 2) Challenges in countermeasures and countermeasure strategies in emergency & recovery, decision support and disaster informatics; 3) Challenges in setting-up a trans-disciplinary and inclusive framework for preparedness for emergency response and recovery. The aim of this paper is to present the main research needs for the three challenges as well as the main goals and vision for the future of emergency and recovery preparedness.

2. CHALLENGES IN RADIOLOGICAL IMPACT ASSESSMENT DURING ALL PHASES OF NUCLEAR AND RADIOLOGICAL EVENTS

Three key topics have to be further addressed for improving and developing impact assessment: improved modelling, improved monitoring and data assimilation.

2.1 Improved modelling

For the models, the objective is to make more reliable and accurate forecasts on dispersion of radioactive materials in different media, human radiation doses and effects on the environment.

For atmospheric transport and dispersion modelling, new developments are expected based on fluid dynamics modelling as well as on better assessment of uncertainties in the models. In the longer term, the management of big data and improved mathematical methods could also help to cope with the complexity of the transfer of radionuclide in the environment (e.g. wet deposition by snow). In the future, it is expected to extend the capability of the models to better address non-conventional emissions (e.g. explosions, aerosol sprays, fire…).

For hydrological modelling, the main challenges are related on one hand to improvement in marine modelling - with a focus on marine food as well as on development of local coastal models (notably implementation of relocatable hydrodynamic 3D models of coastal circulation) - and on the other hand to develop urban run-off models as well as to better address the sensitive issue for local populations of urban water supply models. The first sets of models could be available in the following years, while more flexible approaches could be envisaged for the longer term.

For dose models, efforts should be made to integrate all available environmental monitoring data in prediction and reconstruction of doses in the different environments (urban, forests…) and to better consider the real behaviour of the population and the efficacy of protective actions for the predictions. Due to the recent developments in environmental monitoring and individual monitoring, the models should be adapted.

For environmental modelling, the main challenges consist in improving the database for radioecological models as well as to include regional parameters with a focus on the transfer soil-to-plant and raw-to-product in the various environments. In this context, estimating appropriate uncertainties in the models is crucial and requires specific developments. In addition, further investigations are needed to address the impact of multiple stressors on the environment in case of a nuclear accident.

2.2 Improved monitoring

The main objective is to improve monitoring capabilities and efficiency in emergency and post-emergency/existing exposure situations, with a focus on optimised monitoring and monitoring strategies as well as on improved link between modelling and monitoring.

For monitoring techniques and strategies, the main challenges rely on the development and implementation of new devices and techniques for monitoring people and the environment, taking into account the possible spectrum of radionuclides of concern following an accident. Among the new devices and techniques, it is worth to mention the increasing role of personal monitors with the need to integrate the results into strategies and decision as well as the potential capabilities providing by drones. In addition, a key challenge is to improve approaches combining modelling and monitoring.

For data collection and sharing, it is essential to better identify the key parameters to be collected due to the new capabilities of monitoring techniques and to develop guidance to be applied in case of an accident. Historical data should be organised and database should be developed and shared, notably to be widely used for model validation.

For optimisation of monitoring strategies, the main issue is to organise the interaction between the monitoring and the simulations in order to optimise the monitoring network. Refined methods and tools could be developed in the future to cope with various accidental scenarios allowing to better deploy the monitoring network, more specifically for the early phase.

2.3 Data assimilation

The aims of further developments in data assimilation are to provide a better source term reconstruction based on monitoring and inverse modelling and to decrease uncertainty on impact assessments by combining monitoring and modelling efforts.

For the source term estimation, the main issue is to improve capabilities to estimate source locations and better characterise the source term as well as the plant status and its evolution using dedicated analytical tools and data assimilation methods. Besides the methodological developments for inverse modelling, this requires the combination of tools with mobile and automated equipment to improve the collection of monitoring data.

For improving the impact assessment, the development of operational data assimilation methods in models for doses and concentrations in the environment are needed. A specific focus has to be made on the treatment of uncertainties in the different phases of the accident to cope with limited monitoring for some areas. In the longer term, challenges are associated with the possible combination of bio-dosimetric approaches with the other dose monitoring approaches, notably for assessing doses for large groups of people.

Concerning big data and data fusion, the objective is to develop combined tools for improving decision making using big data capabilities within decision support systems. In this perspective, the main issue is to develop and refine existing computational structures (e.g. platforms, aggregators) allowing the storage, processing and combination of large volumes of heterogeneous and of different origins data.

2. CHALLENGES IN COUNTERMEASURES AND COUNTERMEASURE STRATEGIES IN EMERGENCY & RECOVERY, DECISION SUPPORT AND DISASTER INFORMATICS

Three key topics have to be further addressed for improving and developing response to emergency and recovery phases: countermeasures and countermeasure strategies, formal decision support, disaster informatics.

3.1. Countermeasures and countermeasure strategies

The aim is to investigate the needs for new developments following the large effort performed after the Chernobyl accident and to focus on the elaboration of coherent and harmonised strategies including the evolution with time of these strategies.

The first challenge relies on improving the characterisation and evaluation of efficiency of the various countermeasures to better build and implement countermeasure strategies, addressing preparedness,

emergency response and recovery. For this purpose, lessons have to be drawn from Fukushima management and better analysis of uncertainties, notably regarding spatio-temporal behaviour and effectiveness of countermeasures. Further investigations are also needed to better estimate the factors impacting the efficiency of countermeasures and countermeasure strategies (effectiveness, costs, non-radiological effects…). In the longer term, it is expected to develop intelligent wizards proposing optimised countermeasures and countermeasure strategies based on available information from decision support systems.

To allow an efficient implementation of countermeasures and countermeasure strategies (including considerations on their lifting), it is expected to develop a methodological framework to help decision makers, based on monitoring, modelling and guidance on optimisation approach.

3.2. Formal decision support

For the formal decision support, the main objective is to further develop multi-criteria analysis in particular for the pre-planning and the recovery phases, with a focus on elicitation of stakeholder preferences. In this perspective, methodological developments and elaboration of guidelines are needed to promote "good decision-making practice" and define generic scenarios for preparedness and planning taking into account different driving forces (e.g. technical, societal, economic, environmental…).

Additionally, uncertainty management has to be further considered in multi-criteria analysis. Notably, the elicitation of preferences of stakeholders in the case of high uncertainty has to be addressed. In the longer term, the combination of agent-based simulation systems with multi-criteria analysis could provide useful input for addressing uncertainties.

3.3. Disaster informatics

The recent developments in information technology would provide useful input for further refining disaster informatics. The main objective in this domain is to further develop analytical platform and knowledge database to support decision making when little information is available on the concerned accidental situation. In this perspective, the current analytical platform could be improved based on findings from exercises and applications to better cope with the needs of end-users and progressively to combine the analytical platform with big data approaches (see 2.3).

In particular, tools or mechanisms have to be developed to collect information from the internet and the user interfaces of decision support systems have to be adapted to the new environment of information technologies. Finally, the capabilities offered by virtual and augmented reality could be investigated in order to provide new training facilities for first responders, decision makers and other stakeholders.

4. CHALLENGES IN SETTING-UP A TRANS-DISCIPLINARY AND INCLUSIVE FRAMEWORK FOR PREPAREDNESS FOR EMERGENCY RESPONSE AND RECOVERY

Four key topics have been identified for further developing the trans-disciplinary and inclusive framework: emergency and recovery framework, stakeholder engagement, integrated emergency management, uncertainty and incomplete information handling.

4.1. Emergency response and recovery framework, including reference levels

The main objective of elaborating and further refining emergency and recovery frameworks is to develop radiological decision criteria and guidance for implementation of strategies, taking into account societal

and ethical issues. The aim is to improve the sustainability of emergency response and recovery management. In this perspective, the development of a harmonised framework is needed as well as better addressing the long-term management of contaminated areas, including societal aspects and the management of goods from contaminated areas.

For the harmonised framework, it is essential to develop socially and scientifically robust operational radiological decision criteria, especially to deal with the transition phase and the long-term phase. In addition, success criteria have to be defined to assess the application of the countermeasures as well as the future risks and vulnerabilities for the population and the environment. Decision support systems could be adapted accordingly. In the longer term, it would be useful to consider the integration of the radiation protection approach in a broader environmental protection framework.

4.2. Stakeholder engagement, involvement of the public & communication

The aim for this topic is to ensure a successful and respectful stakeholder engagement process, to promote the integration of citizen science in radiological risk governance for emergency and recovery situations and to favour efficient communication for different contexts and time scales.

For the stakeholder engagement process, the main issue is to further identify the roles, constraints, responsibilities and cooperation among the stakeholders from different levels to improve preparedness frameworks. Existing tools and methodologies for stakeholder participation have to be reviewed and adapted to cope with emergency and recovery situations. In the longer term, preservation of knowledge and experiences of stakeholder engagement have to be considered and analysis of societal needs for the evaluation of legal instruments and governance framework related to stakeholder participation has to be performed. In addition, to better address the integration of citizen science into the radiological risk governance, the key issue is to determine the factors influencing trust between different actors. For improving communication, it is useful to develop methods and procedures to analyse the information flow related to social trust, including traditional information source and social media in order to elaborate guidance on communication for operators, regulators, decision makers and journalists.

4.3. Integrated emergency management – non-radiological aspects

The main objective for the integrated emergency management is to better address non-radiological aspects for developing guidance and framework to improve emergency response and recovery management. This includes considerations on health surveillance, ethical and socio-economic aspects, the development of an integrated surveillance and monitoring framework, and the radiological protection culture.

For ethical and socio-economic aspects, the key features are to investigate their roles in the various decisions to be made during emergency and recovery phases: i.e. evacuation, sheltering, food restrictions, management of the contaminated areas etc. The effect on the daily life of people (notably their perception of the situation and the disturbances on their daily life) and on the environment has to be further analysed to elaborate guidance for decision-makers for the elaboration of countermeasures and countermeasure strategies during the different phases of the accident. This could notably include considerations on the elaboration of a guidance framework for establishing a sustainable radiological protection culture with development of tools, methods and processes to build, maintain and transmit this culture.

For the integrated surveillance and monitoring, the aim is to progressively develop a framework establishing a comprehensive surveillance and monitoring system addressing health surveillance, human dose assessment, environmental monitoring and food monitoring in meaningful way for local populations.

4.4. Uncertainty and incomplete information handling

The aim of this topic is to improve the capabilities to perform sensible and robust decisions under high uncertainty. This includes communication and visualisation of uncertainties in results of the models but also the consideration on how uncertainties are used by decision makers. For this purpose, it is expected to provide a guidance framework and advanced tools to better identify, address and communicate uncertainties.

The first objective will be to investigate overall uncertainties in models and how they can be communicated in the perspective of better characterising the radiological situation for the different stakeholders and decision makers. In addition, it will be useful to investigate media communication about ionizing radiation, more specifically on issues associated with emergency and recovery situations. A special focus will be made on how local actors and non-institutional stakeholders perceive uncertainty in their own decision-making processes and which governance mechanisms can facilitate these processes. All these analyses would allow to develop tools and methods for a two-way communication on uncertainty between experts and non-experts as well as to develop education and training material for decision-makers on uncertainty management.

5. CONCLUSION

With the adoption of its first research roadmap, NERIS has identified the main challenges for the following years and the way forward for developing research activities in order to improve preparedness for emergency response and recovery management. Currently, related research is under development, notably within the European joint research project CONCERT. Within this research project, a first joint roadmap has been elaborated in interaction with the other European research platforms in radiation protection (i.e. MELODI for low-doses, ALLIANCE for radioecology, EURADOS for dosimetry and EURAMED for medical exposure) [3]. Joint research challenges have been identified for further developments, including the following challenges which are directly or indirectly relevant for NERIS issues: understanding radiation related human health effects, improving the concept of effective dose and other quantities, studying the biological effects on biota, optimising emergency and recovery preparedness and response, and enhanced integration of radiation protection science with society. In this context, the reinforcement of the joint research involving the different research platforms aims to promote useful and efficient research in radiation protection.

In addition, the connection with the international organisations involved in the management of emergency and recovery issues is essential for promoting the results of the research, and for sharing experience and drawing lessons on the assessment and management of the Chernobyl and Fukushima accidents, as well as on preparedness experiences. This connection will also contribute to improve and harmonise emergency and recovery approaches in Europe and worldwide.

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Advanced training of first responders at Forschungszentrum Jülich - Cross-border co-operation –

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Abstract. Forschungszentrum Jülich (FZJ) is geographically located in the west of Germany in the border region between Germany, the Netherlands and Belgium. It has a specialized Division for Safety and Radiation Protection (S) and, as part of it, a radiological emergency response team with special experience and knowledge. That's why Forschungszentrum Jülich is a valued partner in the training of fire brigades in radiation protection. Radioactive clouds, international traffic and terrorists don't stop at borders. Therefore, cross-border cooperation is important. Common scenarios for rescue forces like an accident at a nuclear power plant near the border, an accident on a cross-border road or a terrorist attack with a "dirty bomb", must be considered. This paper describes examples of cross-border cooperation in the advanced training of first responders from Brandweer Zuid Limburg (Netherland) and the Region of Aachen (Städteregion Aachen, Germany). The focus of such a practice-oriented training is on different exercise scenarios under realistic conditions. Exercises such as search and best practical experience in finding lost radioactive products or a traffic accident involving radioactive material are examples of training with sealed radioactive sources. Alongside these exercises lectures furthermore support the know-how-transfer in the fields of emergency response, measuring techniques, dosimetry, etc. from Forschungszentrum Jülich to the participants of the training. Together with additional partners an international cross-border-project involving three countries has been proposed, which also includes the education of students and young scientists beside the training of first responders and evaluating possibilities of support of professional non-police response forces, i.e. information course with FZJ experts, e-learning courses for radiation protection experts, teacher training. The planned contributions of Forschungszentrum Jülich in the development of different training modules in radiation protection will be outlined.

KEYWORDS: Training, first responders, rescue forces

1 INTRODUCTION

In December 1956, the state parliament of North Rhine-Westphalia decided to build a "nuclear research facility" – The birth of today's Forschungszentrum Jülich. The main concern of the founders was the peaceful use of the total nuclear research. The foundation stone laying of the MERLIN (**M**edium **E**nergy **R**esearch **L**ight Water Moderated **I**ndustrial **N**uclear Reactor) reactor on June 10 (Fig. 1).

Figure 9: Foundation stone laying of the MERLIN reactor 1956

Photo: Archive FZJ

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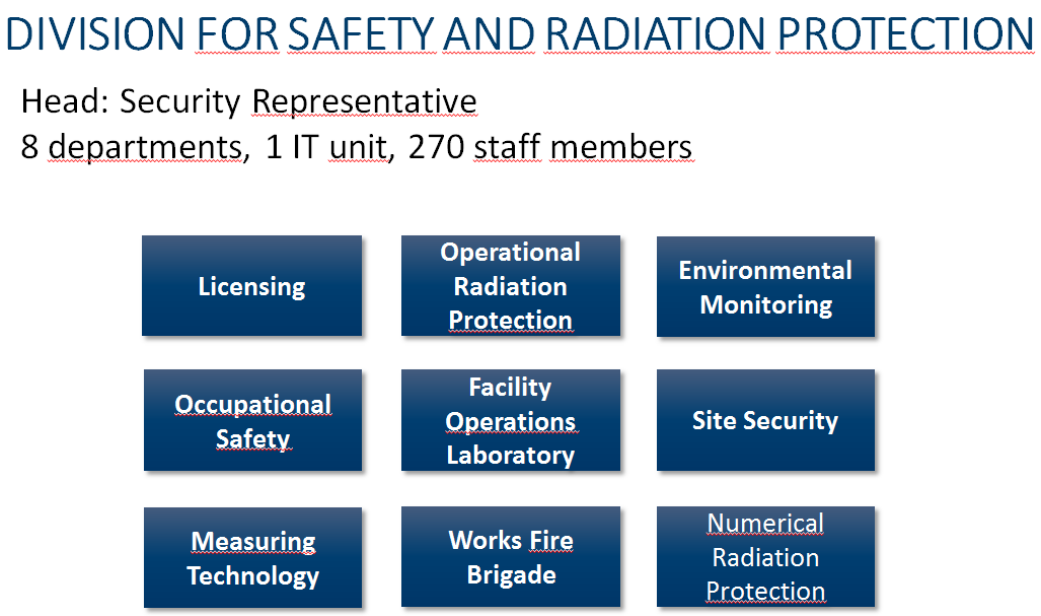
In addition to MERLIN, the Forschungszentrum Jülich has operated another research reactor ("DIDO", the short form DIDO comes from the chemical form of heavy water D2O (the working principle of the reactor), a spelling of this form is DDO or the name DIDO, Shutdown: 2006) for neutron research and was also involved in the development of technologies for the neighboring high-temperature reactor of ***A**rbeitsgemeinschaft **V**ersuchsreaktor* GmbH (AVR). All three reactors have been shut down for many years or decades respectively. The MERLIN research reactor has already been completely dismantled, while the other two are currently in the dismantling process.

Since the 01.09.2015, the task of the dismantling has been taken over by **J**ülicher **E**ntsorgungsgesellschaft für **N**uklearanlagen mbH (JEN), which bundles all the nuclear decommissioning, dismantling and disposal expertise that has been built up in Jülich over five decades. The research center acts with its emergency organization as a service provider for JEN.

From the mid-1980s until today, Forschunszentrum Jülich focused increasingly on "non-nuclear" research. Only minor research is being carried out relating to the safe handling of nuclear waste and international measures for the monitoring of nuclear material ("safeguards"). The emergency protection organization, which was created by Forschungszentrum Jülich (FZJ) for the safe handling of "nuclear research", remained, however, due to the still existing nuclear dismantling activities and the handling of radioactive substances. The shutdown of the last operating reactor DIDO (FRJ-2) in 2006 increased the dismantling activities on the site of the research center. However, the organizations created by the Forschungszentrum Jülich (FZJ) for the safe handling of "nuclear research" remain in operation.

Essentially, this "security organization" consists of todays "Safety and Radiation Protection" Division of the Forschungszentrum Jülich headed by Forschungszentrum's security representative. Part of this business area is e.g. the operation of the Plant Fire Brigade for the FZJ, the radiation protection emergency response and various laboratories and mobile measuring units (Fig. 2).

Figure 2: Organization of the Safety and Radiation Protection Division



In future, the experience, the know-how and the technical possibilities are planned to be used in addition to the original function for the further training of Dutch, Belgian and German public fire brigades.

2 ”HISTORY“AND”KNOWHOW“

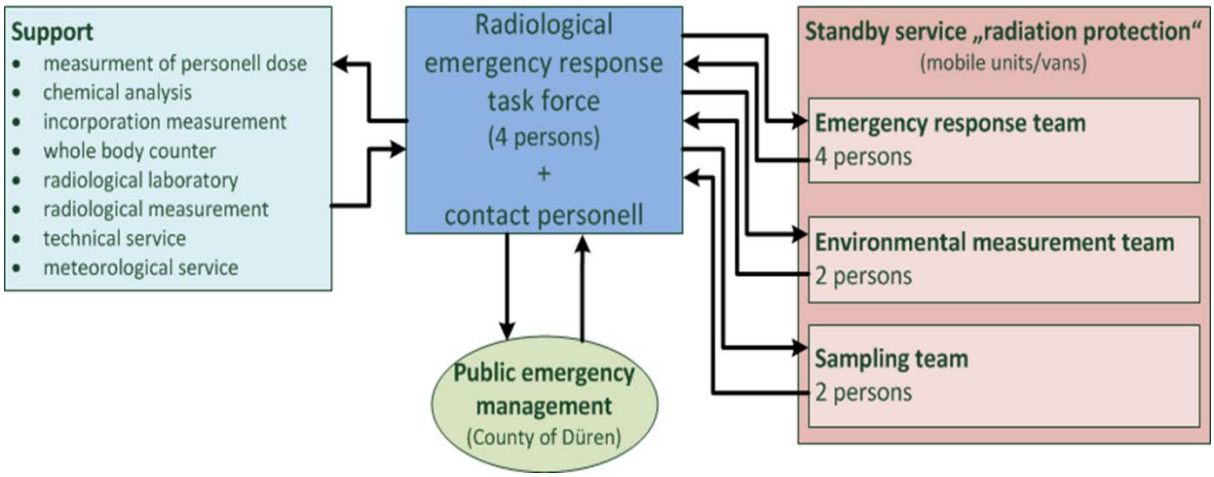
Forschungszentrum Jülich has built up extensive competencies by handling, measuring and analyzing radioactive substances due to its nuclear past. The Safety and Radiation Protection Division (S) has its core competence in the field of radiation protection and emergency response. With the establishment and operation of the research reactors "MERLIN" (Fig. 3), "DIDO" and the "AVR" reactor on the Jülich campus, a radiation protection emergency response organization was necessary. It was established under paragraph 53 of the German Radiation Protection Ordinance [1]. This FZJ organization is headed by the Security Representative (SBV).

Figure 3: Construction of the "MERLIN" reactor"

Photo: Archive FZJ

The emergency organization (Fig. 4), which consists of an emergency response task force, associated emergency response team and various support organizations to which the plant fire brigade must also be counted, is until today unchanged existent. By the further dismantling of nuclear facilities on the site, the organization of a radiation protection emergency response team might become at some time in the future dispensable.

Figure 4: Radiation protection emergency response organization at Forschungszentrum Jülich



The existing know-how of the operational forces, radiation protectors and scientists of the S division at the site can be used for the training of emergency personnel and first responders of public authorities.

3 TRAININGPLAN FOR FIRST RESPONDERS

The fire brigade and emergency protection organization, which belongs to the Safety and Radiation Protection Division, has long been in contact with fire brigade forces of cities nearby. There is a need to deal with radioactive substances in cases of an emergency. This resulted in a first training of operational forces on the site of Forschungszentrum Jülich GmbH.

3.1 Training “City Region” Aachen

The research center possesses a license according to the Radiation Protection Ordinance, which allows training with "real" radioactive sources on the site of the FZJ. This enables the first responders to carry out an exercise under "real conditions". One scenario for the fire brigade from the City Region Aachen (Fig. 5) is, a spill of contaminated wastewater coming from a leaking suction car for radioactive waste water which task is transporting waste water to the conditioning plant where they should be evaporated.

Figure 5: First responders of the City Region of Aachen at Forschungszentrum Jülich

Photo: S. Loup, FZJ

A leaking valve was assumed. Subsequently the car lost radioactive contaminated wastewater on its journey. The resulting training focused on:

- Carrying out measurements with the ABC-brigade for searching the source and the localization of presumed radioactive objects or contaminated areas.
- Working with measuring devices for the detection of sources (stationary and deposed) considering the risk of contamination (Fig. 6).
- Implementation of measures for localization of radioactive sources and secure storage after localization

Figure 6: Working with measuring instruments considering the risk of contamination

Photo: S. Loup, FZJ

After the first responders have fulfilled the tasks under professional observation, a meeting will be held with all participants at the end of the training. The special training goals are evaluated and discussed in detail.

3.2 Training Brandweer Zuid Limburg

Fire brigades from the Netherlands are also interested in the training opportunities in the field of radiation protection by the FZJ. After a first discussion, a pilot training was held in September 2017 for operational forces from Zuid Limburg. For this training a program was compiled covering theoretical aspects of radiation protection and practical exercises. The theoretical part was focused on necessary knowledge for the following practical application.

In various lectures, general radiation protection skills, tactical procedures in use, knowledge of regulation concerning the transport of dangerous goods, measurement technology and emergency measurements were taught. In the practical exercises, an accident scenario was simulated with a dangerous goods transporter who lost parts of its cargo during the accident. Part of the exercise was the assessment of the hazard potential by means of the dangerous goods identification of the transporter and the detection of lost radioactive substances. The second practical part focused on tactical actions (e.g. use of natural conditions as shielding to protect against radioactive radiation) and detection of radioactive sources.

4 MEASURING TECHNOLOGY, ENVIRONMENTAL MONITORING AND DOSIMETRY

Due to a complex and intense monitoring program for environmental radioactivity around FZJ, a special know-how is also available in this field. The environmental measuring stations located in the surroundings of FZJ are still operated today as if the reactors are actively in operation. This includes sampling, measuring and evaluating of environmental samples. The measuring devices, which are used specifically for these purposes, are maintained, calibrated and repaired in own workshops. It does not matter from which manufacturer the measuring instruments come from.

In the field of radiological incorporation monitoring, a certified measuring organization is operated by the Safety and Radiation Protection Division. A body counter, a radio-chemical laboratory and a weather station with a weather tower are also available. These services (see also fig. 4) still cover the entire range of emergency response in case of an accident involving radioactive material.

5 COUNTRY-SPECIFIC DIFFERENCES

In the case of transnational training, it is necessary to consider the specificities that exist for the various countries in the field of nuclear accident prevention. An example of this is the equipment that is used. While the German emergency services can provide individual measuring devices for each fire brigade, the Dutch colleagues use uniform equipment on all vehicles. It is necessary to adjust the training of the first responders.

In addition to the differences in equipment, there may also be differences in the general rules for the radiation protection emergency response. To identify these differences, the research Centre has participated in a transnational proposal.

6 INTERNATIONAL KNOWLEDGE AND INFORMATION CENTRE

To offer training opportunities in the security field to an extended public (including other countries), Forschungszentrum Jülich has joined the project group IKIC (**I**nternational **K**nowledge and **I**nformation **C**entre). This prosed project considers the country-specific differences in the field of security. On this basis, training courses for pupils, teachers, students, professional staff and interested citizens are then developed. The proposed project could start in September 2018 until 2021 if funding is available.

7 CONCLUSIONS

Using radiation protection specialists in the field of training of firefighters, the know-how available at Forschungszentrum Jülich in radiation protection and in nuclear security can be transferred. For this purpose, various training modules are developed and offered by FZJ's Division for Safety and Radiation Protection.

8 REFERENCES

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Presentation of evolutions of the ceres platform used to evaluate the consequences on population of pollutants releases

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Abstract. This paper presents the CERES platform (Code d'Evaluations Rapides Environnementales et Sanitaires) developed in French Energy Atomic Commission (CEA), in order that all evaluations of impact on human relative to CEA installations releases will be done using the same tools and methods.

KEYWORDS: impact assessment, public, radiological impact, chemical impact

1. INTRODUCTION

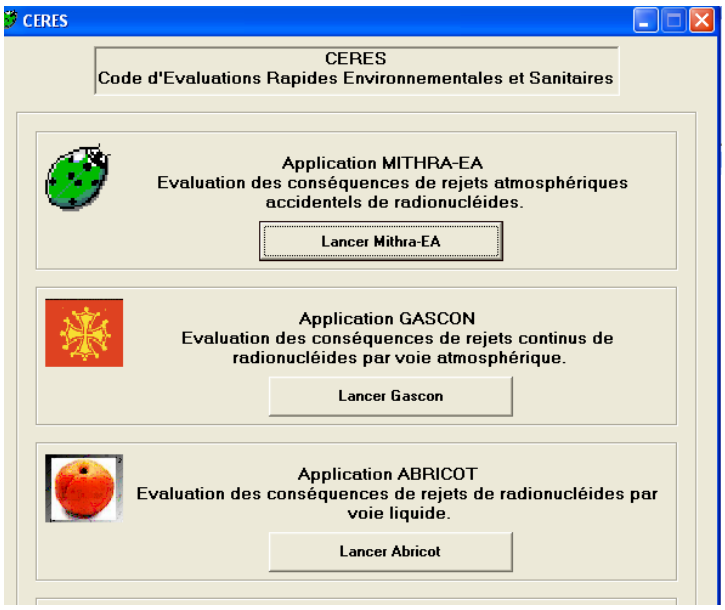
As, according to DPSN* decision, the Radioanalysis, Chemistry, Environment Division (DAM/DIF/DASE/SRCE**) of the French Atomic Energy Commission and Sustainable Development (CEA) is responsible of the "Pôle de Compétences Impact radiologique et Chimique" for all CEA installations, it is in charge of the definition of methods and the development of modelling tools to evaluate the consequences on human health of releases of radionuclides or toxic chemicals in the environment. In this context, the laboratory has developed the CERES® (Code d'Evaluations Rapides Environnementales et Sanitaires), in order to ensure that all impact evaluations relative to CEA installations are performed using same methods and tools. This application is used for emergency situations planning and for the realization of assessment calculations within a regulatory framework, for example in the safety documents relative to nuclear installations. It helps to evaluate either the consequences of accidental situations supposed to occur on installations or the impact of routine releases from single installations or nuclear sites on their near environment. About 50 studies per year are realized using CERES® platform.

Several emission situations can be simulated (Fig 1):

1. accidental releases in the atmosphere (MITHRA),
2. routine releases in the atmosphere (GASCON),
3. routine releases in rivers (ABRICOT)

Two versions of the CERES platform are available. The first one, called "emergency version" is used in order to evaluate the consequences of the emission of pollutants in the atmosphere during emergency situations. In this version (number 1 above), several options are not available in order to simplify the definition of the parameters used for the evaluation. In addition, many parameters are proposed as default values and can not be modified. In the second version, called "expertise version", all the modules can be used and many parameters can be modified, even if most of them are pre defined with default values.

Figure 10. User Interface



2. DISPERSION CALCULATIONS

2.1 Accidental atmospheric releases

For accidental atmospheric releases and in case of emergency situations, atmospheric transport modelling (ATM) is carried out with the Gaussian puff MITHRA model [1]. Different standard deviations equations can be used: Doury's formulas [2], function of travel time, that is the default option, or standard deviations based on Pasquill's theory: Briggs' [3] or Turner's [4]. If necessary, other models can be used to take into account topography.

Following an accidental release in the atmosphere, MITHRA evaluates instantaneous (Bqm^{-3}) and time integrated activities concentrations ($Bqsm^{-3}$) for different points and instants defined by the user. Deposits on the ground result from mechanisms of diffusion, impaction and sedimentation and from washing out of puffs during rainy situations. The dry deposition velocity is considered independent of distance from the emission. Rain may lead to more significant deposits. The calculation of wet deposition velocity with distance from emission is available by taking into account the washing out rate. These parameters combined with the diffusion parameters (release height, wind speed, vertical standard deviation) enable to evaluate the depletion coefficients and depositions. For aerosols or vapours, depletion due to dry and wet deposition is calculated. For aerosols, a dry deposition velocity of 5.10^{-3} ms^{-1} is proposed [5] as default value but can be modified.

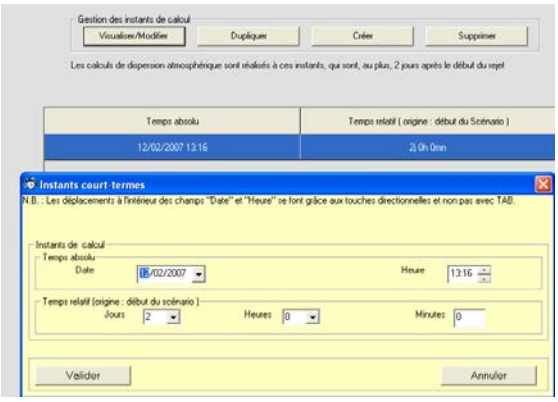
A specific module is used to evaluate the atmospheric dispersion of tritium and transformation from HT gas to tritiated water [6].

The activity emitted from the facility to the environment can be evaluated using the ERASTEM system [7], that is a box model, taking into account the transfers from the different compartments of the studied reactor (fuel, pool, containment, environment) using ventilation rates, retention, filtration and transfer coefficients.

As MITHRA is a Gaussian puff model, it can take into account releases or meteorological conditions varying with time. Radioactive decay during atmospheric transfer is calculated, using Bateman equations [8]. If the release is emitted with high temperature and speed, as it can be the case for significant fires, it is possible to evaluate a plume rise, using Briggs or Holland equations. The effective height is then used as the release emission height.

It is possible to take into account a wind meandering factor (Fig 2). This can be necessary for long duration releases, if meteorological data are not known with great accuracy. Indeed, if the release is quite long, meteorological conditions, especially wind directions, are not constant during release time and observation. The wind meandering factor increases the width of the plume; the concentrations in the wind axis are then lower. If meteorological conditions are well known, it is possible to define meteorological steps, *i.e.* time intervals during which meteorological data (wind speed and direction, atmospheric stability and rain) are supposed to be constant.

Figure 11. Meteorological Data Definition - accidental situations



Depending on the type of calculation, meteorological data are either observations coming from the meteorological station located on the site (if release is ongoing, that means diagnostic calculations) or

forecasts (in case of emergency situations, when the objectives are to implement the counter-measures before the releases begun) coming for example from MEDICIS system outputs described in [9].
Calculations are carried out on a gridded domain and at points defined in the data base relative to the studied site or at points whose coordinates are defined by the user.

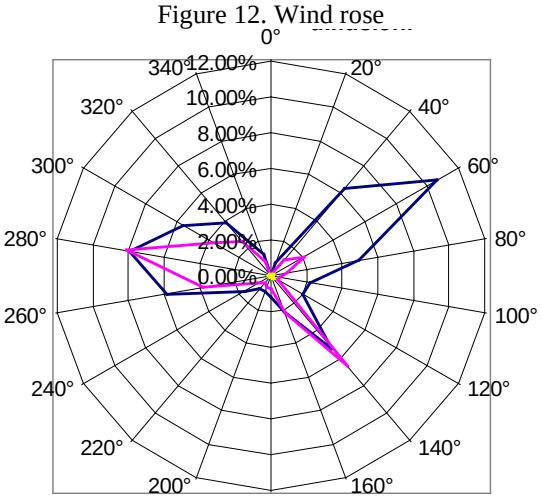
2.2 Routine atmospheric releases

For atmospheric releases in normal situations (routine emissions), dispersion calculations are carried out with GASCON model [10], based on the same Gaussian puff model as in MITHRA code describes previously.

To evaluate the impact of atmospheric releases during normal operation of the plant, the meteorological data needed are the observations carried out for one or more years at the meteorological station of the site or nearby.

The variability of the weather conditions throughout the year is taken into account by associating a probability of occurrence at each triplet (wind speed, wind direction and atmospheric stability) and by summing the contribution of each one of these weather conditions to obtain the average atmospheric concentration. Thus, observations are processed statistically. They are distributed by classes of occurrence in 18 sectors of 20° width into normal diffusion associated to dry weather class (unstable atmosphere, important turbulences), normal diffusion with rain class, weak diffusion (stable atmosphere) (Fig 3).

Figure 12. Wind rose



As rain increases atmospheric turbulences, the situation of stable atmosphere with rain is not taken into account because it appears not to be realistic.

During a single calculation, it is possible to evaluate the global activity concentrations and deposits due to several emissions points.

GASCON provides average activity concentrations (Bqm^{-3}) and dry and wet depositions fluxes ($Bqs^{-1}m^{-2}$) at different points defined by the user and on a gridded domain. Results can then be shown on Geographical Information System.

2.3 Routine liquid releases

For normal release in rivers, dispersion calculations are carried with the ABRICOT model [11]. A homogeneous and immediate dilution is assumed. The water activity depends on the quantity emitted and on the river flow. ABRICOT can take into account the fact that the drinking water, the fishing water or the water used to irrigate crops come from groundwater, pond or is used after purification. Coefficients representing dilution phenomena and sedimentation or adsorption on the suspended matter can be used to take into account of contributions of water and sedimentation. These factors depend on the site and the isotopes studied.
If the water used come from groundwater, it is necessary to take into account the retention of the isotopes in the alluvia that involve a transfer time during which radioactive decay occurs. These phenomena are simulated using a distribution coefficient (Kd) between water and sediments, the distance between the river and the intake point and the water speed in the alluvia.

If releases occur in a pond, the renewal of water is evaluated by considering the relative volume of the pond to its annual throughput exit. The volume activity in the pond is calculated using dilution, due to rain and sources, sedimentation, radioactive decay and renewal of the water of the pond.
In the case of passage through a sewer, deposits on the suspended matter are taken into account using a distribution coefficient function of the isotope and the mass of suspended matter. Impact through mud from sewer on crops can be taken into account.

3. RADIOLOGICAL IMPACT ASSESSMENT

Impact evaluations are performed for groups of population. It is possible to take into account the fact that all the food is not locally produced or that people move during the exposition duration. It is then possible to define the time percentage spent in different places, the percentages of consumption of products of garden, fruits, and cereals as well as livestock products coming from different locations. The characteristics of various groups can be different but for a given group, these characteristics are identical whatever the age, except for diet habits. These latter correspond to the food potentially impacted by the releases, i.e. locally produced food. All the data (diet habits, crops…) are defined in a data base characteristic of the site and they can be modified.

The consequences, expressed in terms of effective dose or dose to the thyroid, including all the exposure pathways described below, are then estimated.

In case of atmospheric emission, the exposure pathways simulated are:

- immersion in the plume, which leads to an internal exposure by inhalation and an external exposure by irradiation,
- presence on the deposits, which leads to an external exposure by irradiation,
- consumption of plants, for which the activity results mainly from the deposits of aerosols and rainfall but also from the transfers starting from the ground (root uptake), and then leads to an internal exposure by ingestion,
- consumption of animal products (meat and milk), from animals which have consumed contaminated food.

In case of tritium emission, the exposure ways are slightly different:

- immersion in the plume leads to an internal exposure by inhalation and skin-passage,
- as the tritium is a pure β transmitter, it does not lead to an external exposure by the deposits; because of the mobility of tritium in the ground, the effect of accumulation can be neglected,
- it leads to contamination by inhalation and by ingestion after transfer through the food chain.

In case of liquid emission, the exposure pathways simulated are:

- inhalation due to activity due to deposits resuspension, following agricultural work as example,
- presence on the deposits, which leads to an external exposure by irradiation,
- ingestion of water,
- ingestion of fishes,
- consumption of plants, for which the activity results mainly from the deposits of aerosols from irrigation but also from the root transfers, and which leads to an internal exposure by ingestion,
- consumption of animal products (meat and milk), from animals which have consumed contaminated food.

Results are available as tables and graphs (Fig 4), giving as an example the contribution of the isotopes or the contribution of the exposure pathways to the dose. The entire hypothesis and the main parameters of the scenario are indicated with the results.

For accidental releases, a colour code indicates whether the intervention levels for radiological emergencies defined by decree [12] (see *Table 1*) are reached for short-term impact (.e. taking into account irradiation by the plume and the deposits, integrated on maximum 48 hours, and dose due to inhalation).

Figure 13. Typical graph of result

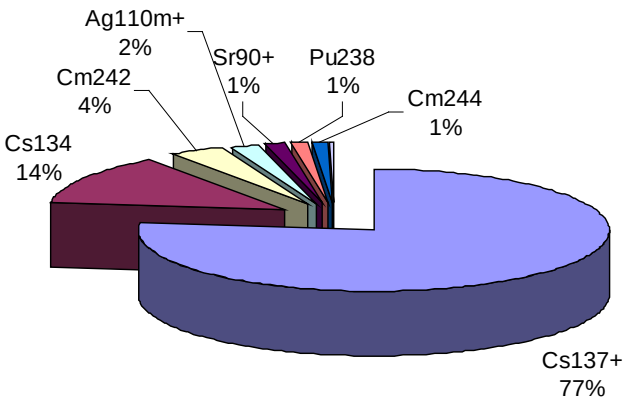


Table 1. Intervention levels for public

Dose	Action of protection
50 mSv – effective dose	Evacuation
10 mSv – effective dose	Sheltering
50 mSv – thyroid dose	Iodine prophylaxis

For the evaluation of the external exposure due to the plume passage, evaluated only for atmospheric releases, dose coefficients come from the Federal Guidance Report n°12 [13]. For accidental releases, the dose rate during the plume passage is accessible. The dose coefficients for the external irradiation by the deposits also come from this document. For atmospheric releases, internal exposure results from the activity inhaled during the plume passage, without protection. Dose coefficients come from the decree of September 1, 2003 [14] or from ICRP publications [15]. The dose to thyroid is presented only for accidental situations when at least one iodine isotope is present in the results.

In accidental situation, the breathing rates correspond to a moderate physical activity. They come from publication 66 of ICRP [16] and are 1.2 m^3h^{-1} for adults, 0.9 m^3h^{-1} for 10 years old children and 0.3 m^3h^{-1} for 1-2 years old children. For atmospheric normal releases, breathing rates, also coming from ICRP66, were calculated using the volume of air inhaled daily, according to different activities, *i.e.* integrating periods of awakening (work, leisure) and sleep. They are thus weaker than in accidental situation and are respectively 0.96 m^3h^{-1} for adults, 0.64 m^3h^{-1} for 10 years children and 0.22 m^3h^{-1} for 1-2 years children. For liquid releases, as inhalation occurs mainly during work, the breathing rates considered correspond to a moderate physical activity.

Transfer mechanisms of isotopes to the plants, then to animals are used to evaluate doses due to ingestion of crops and animal products.

4. CHEMICAL TOXIC IMPACT ASSESSMENT

Dispersion models and transfer in the foodstuff and feedstuff are identical for toxic and radiological emissions. Impact evaluations are difficult. For normal releases, the use of toxicological values helps to evaluate either “Risk Quotient” (for non carcinogenic risk) or Individual Risk Excess - ERI - (for carcinogenic risk) by inhalation and / or ingestion. For accidental releases, the use of emergency levels helps to evaluate accidental impact. For toxic risk, the main difficulty is that toxic levels (mainly for normal situations) are not defined by law. At each study, it is then necessary to check whether the level used in a previous study can still be used. It is so very difficult to build a database up to date that can be distributed to users. In addition, a transfer model has been built in order to evaluate the transfer of organic substances in foodstuff. It must be validated.

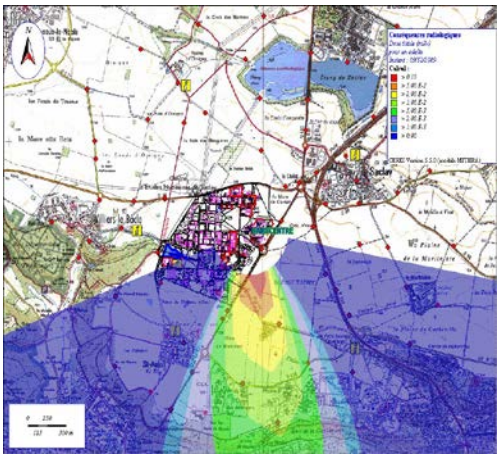
5. DATABASE OF RESULTS

CERES has a database containing the characteristics of about 800 isotopes (dose coefficients, transfer coefficients from soil to plants, from plants to animals…).

Several databases containing environmental characteristics of CEA nuclear sites (installations, height of stacks, inventories, environmental measurements points, meteorological data, diets …) exist.

Atmospheric results can be visualized on a map as shown in Figure 5. All results are stored in a specific database in order to be used again if need.

Figure 14. total effective dose (mSv)



6. VALIDATION OF THE APPLICATION

Results obtained with Doury's parameters have been validated by comparison with experiments of atmospheric tracing. MITHRA atmospheric dispersion results were compared with to those obtained by SIROCCO2 assessment code developed by the French Support to Nuclear Safety [17].

According to the assessment results, due to lack of experimental data available, the validation carried out until now only consists of comparison exercises between the results provided by various models. In addition, studies results carried out previously with a plume model were compared to results obtained with MITHRA. The results obtained with the two codes are similar, in particular with regard to short-term calculations. However, in the event of height release, during stable situation, results provided by MITHRA are higher than those provided by the plume model. This difference is due to the well-known non accuracy of plume model in these situations [18].

7. EVOLUTIONS

This platform is widely used by all CEA sites and some other nuclear operators. Recent evolutions of these platforms follow regulation evolutions and demands expressed by users.

Work is on-going on integration of chemical reactions during atmospheric transfer, development of heavy gases models and use of topography and buildings effects, by example in accidental situations.

A new version of user interface is being developed.

8. CONCLUSION

By developing the CERES® platform, CEA can develop its own models to evaluate impact of releases on population near the sites. As this platform is used either in emergency situation or for safety reports, *i.e.* before releases, date required to evaluate assessment need to be simple and robust. About 50 reports are written per year, covering different stages of installation cycle (creation, accidental, dismantling …).

Results obtained are conservative in order to be sure to overestimate real impact on population.

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A site specific accidental aquatic transport model for radioactive release to the Danube at the Paks NNP

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Abstract. To determine the environmental radiation burden of a possible accidental liquid radioactive release, we need to consider the radionuclide transport in the Danube. Therefore, modeling the migration of radioactive material is crucial. Our objective is to create a dynamic transport decision support program with realistic local parameters which describes the hydrological properties and the habit data of the population.

1 INTRODUCTION

The two main processes during the modeling are the calculation of the activity concentration including the connection with the source and sink terms, and the calculation of the doses. The river is divided into three section considering the status of the vertical and horizontal mixing. The first 1-2 kilometer from the release point is the initial mixing stage, where neither the vertical nor the transversal mixing is complete. After the vertical mixing is completed, the second stage is along a 150-200 kilometer, what lasts until the complete horizontal mixing. Here the activity concentration difference between the two banks is under 10%. The last section is the far field section, where both vertical and transversal mixing are complete.

In this study the primary focus is on the section of the Danube under Paks Nuclear Power Plant (in Hungary). Altogether, it operates four VVER440-213 pressurized water reactors. In the future, the NPP will be expanded with two new VVER-1200 blocks. The receptor point is chosen as the first village under the NPP at the same riverbank called Gerjen. The village is approximately 10 km south from the NPP (Fig.1).

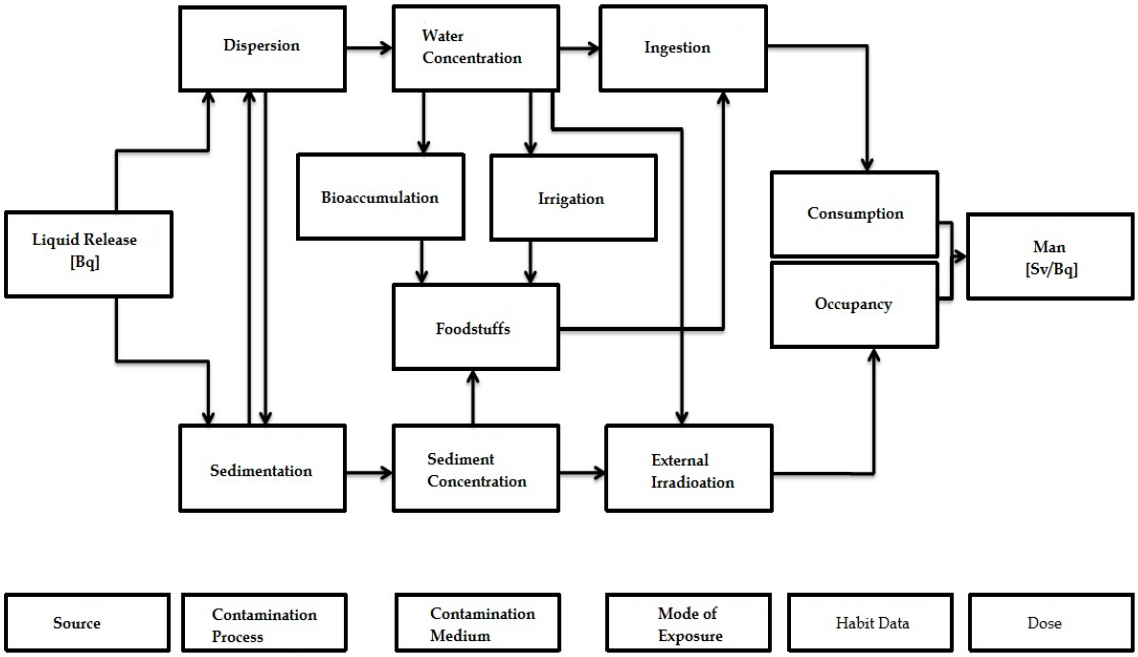
Figure 1: The location of town Paks, the NPP and village Gerjen from north to south

Although the general equations for the calculations of the hydrological part of the transport can be found in the literature, these equations are not detailed enough to be used for an accidental site specific release. Therefore, local geological and hydrological parameters of the Danube are needed, which include the local shape of the river's cross-section, the stream stage dependent flow rate, flow velocity and also the local sediment data. With these, the conversions of the general equations to site specific ones are made and the structure of the transport model is set.

2 METHODOLOGY

The method of aquatic modelling is schematically represented by Fig. 2 [1].

Figure 2: Schematic representation of aquatic pathways [1]



Calculations are made in the following major steps:

1. First the activity concentrations are calculated at the receptor point in water, sediment and biota.
2. From the activity concentrations doses are calculated for different exposure pathways.

Most transport models rely on the advective-diffusion equation or a simplified version of it [2]:

$$\frac{\partial C}{\partial t} = A + D - R + P - \lambda C \tag{1}$$

where:

C: is the activity concentration of the radionuclide of interest in the water,
t: is the time,
A: is the advection term,
D: is the diffusion term,
R: is a term relating to pick-up of radionuclides by suspended sediments,
P: is a source or sink term for sedimentation, e. g. biological uptake, etc.,
λ: is the decay constant.

The two driving forces (advection and diffusion) are being considered together, creating the advective-diffusion equation. The derivation of the equation lays on the rule of superposition: the advection and diffusion terms are additive, since they are linearly independent. Equation 2 describes the two dimension form of the advective-diffusion equation, while Equation 3 the analytical solutions.

$$\frac{\partial C}{\partial t} + u\frac{\partial C}{\partial x} + v\frac{\partial C}{\partial y} = D_x\frac{\partial^2 C}{\partial x^2} + D_y\frac{\partial^2 C}{\partial y^2} - \lambda C \tag{2}$$

$$C(x,y,t) = \frac{M}{h4\pi t\sqrt{D_x D_y}} \exp\left(-\frac{(x-ut)^2}{4D_x t} - \frac{y^2}{4D_y t} - \lambda t\right) \quad (3)$$

where:

x,y: are the directions in downstream and lateral [m],
M: is the activity of the released radionuclide [Bq],
A: is the cross section area [m^2],
t: is the time [s],
$D_{x,y}$: are the diffusion terms in direction x,y [m^2/s],
u: is the average flow velocity [m/s],
λ: is the decay constant [1/s].

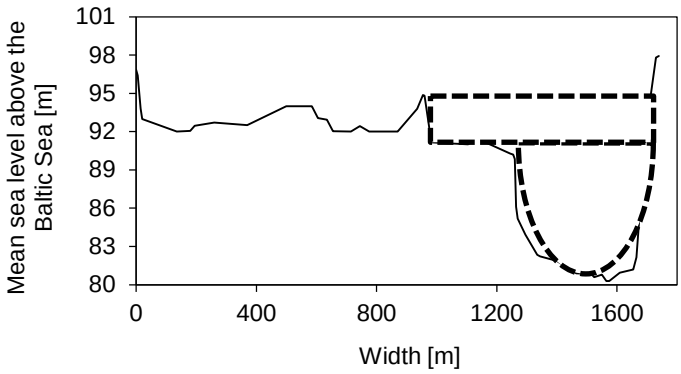
2.1 Hydrological data

The model calculates the actual hydrological variables from the daily measured stream stage (f [m]), which are:

- the depth of the river (h [m])
- the width of the river (B [m])
- the flow rate of the river (Q [m^3/s])

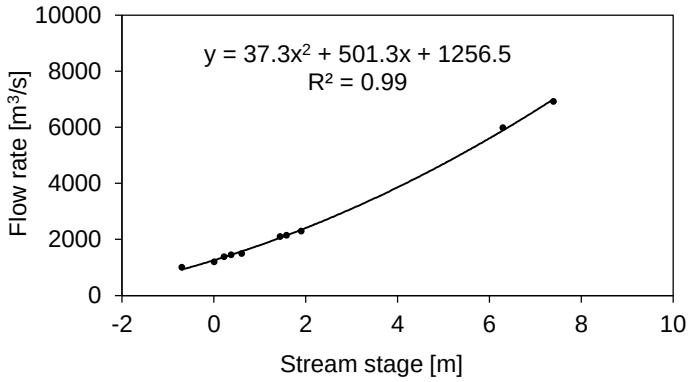
At a stream stage f=0 (85.38 mBf: mean sea level above the Baltic Sea) the depth of the river is 5 meter, derived from the cross sectional shape (Fig. 3). The stream stage can vary into positive or negative direction, therefore the actual depth is described as: h=5+f.

Figure 3: The cross section shape of the Danube under Paks



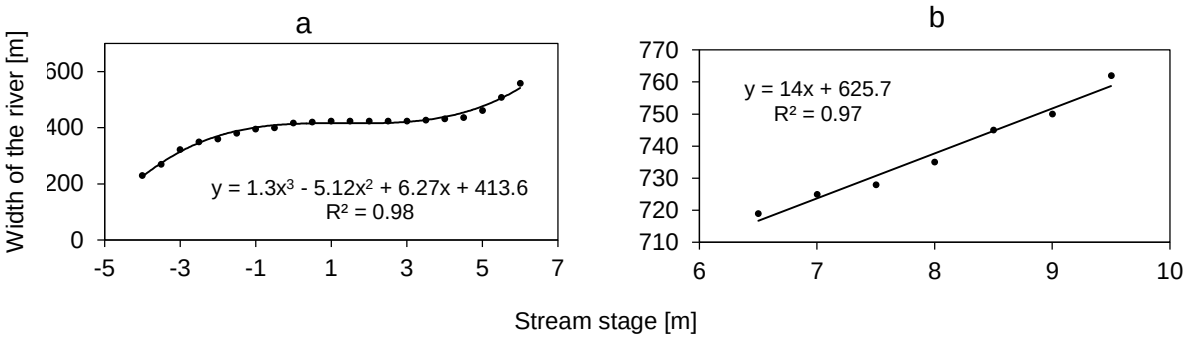
To determine the actual flow rate [Q] in the function of stream stage, a flow rate curve is needed. The measured curve for Paks is shown in Fig. 4.

Figure 4: Measured flow rate curve for Paks [3]



The width of the river is derived from the cross section diagram (Fig. 3). At a stream stage of 6.5 m the river floods the left side terrace, therefore two equations are used to calculate the width. These two sections are shown on Fig. 5/a and 5/b.

Figure 5: The two sections of the river for calculating the width



The average flow velocity of the river (u [m/s]) is calculated by dividing the flow rate with the cross sectional area (Equation 4), where latter is derived from the cross sectional shape (Equation 5).

$$u = Q/A \tag{4}$$

$$A = \frac{2Bh}{3} + ((h - h_{base}) * B_{base}) \tag{5}$$

where:

h: is the depth of the river [m],
h_{base}: is the depth until parabola shape, (constant: 9.62 m),
B_{base}: is the width of the rectangular shape (constant: 720 m),
B: is the actual width of the river [m]

It is considered that between the emission and the receptor point the calculated hydrological variables are constant.

2.2 Connection with the sediment and with the fish

The dissolved nuclides interact with the sediment and biota. The role of the sediment becomes important during a long term release, but in the case of an accidental emission, special emphasis has to be directed towards flooding areas, while contaminated sediments could be left on the shore side [4].

To calculate the activity concentration in the mentioned flooding area, the activity concentration in suspended sediment (C_{sus} [Bq/kg]) need to be considered:

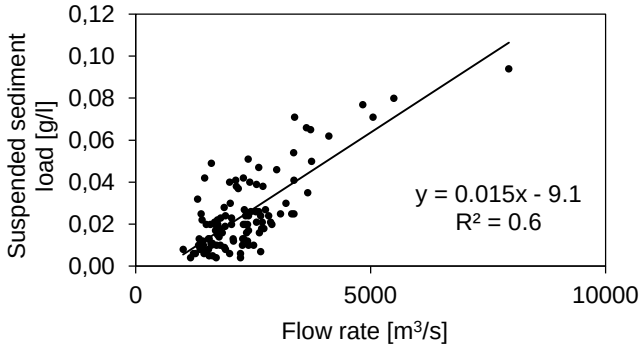
$$C_{sus} = C \frac{0.001\, K_d}{1 + 0.001\, S_s\, K_d} \tag{6}$$

where

C: is the activity concentration is water [Bq/m^3],
K_d: is a distribution coefficient describing the exchange processes of radionuclides between the dissolved and the sediment phase [l/kg],
0,001: is a converter factor for K_d, to turn l/kg into m^3/kg,
S_s: is the suspended sediment load [kg/m^3].

Measurements of suspended sediment load are available for 6 years at village Dombori (~20 km South from Paks). The value of the suspended sediment load between Paks and Dombori does not change significantly. Therefore, these measurements can be used for the calculations (Fig. 6).

Figure 6: Correlation between the suspended sediment load and the flow rate from 2007 until 2012



Activity concentration in the bottom is assumed to be 10% of the concentration in the suspended sediment in the whole riverbed area between the release point and the receptor point.

The discharged radionuclides are also interacting with the living organisms. The radionuclides can reach humans from fish consumption, water ingestion, and via terrestrial food chain after irrigation. In this paper, only the activity concentration in fish is considered, as the contribution from the estimated maximum daily intake rate of fish (k_f)*:*

$$k_f = \frac{D_{max}\, T_e}{86400} \tag{7}$$

where

k_f: is the intake amount [kg],
T_e: is the accumulation time [s],
D_{max}: is the maximum daily intake of the fish [%] what depends on the water temperature. A fish with a weight of 10 kg at water temperature 10-15 °C intakes the 5% of it's whole weight. At a temperature of 20-25 °C this amount is 10%.

Activity concentration in fish (C_{fish} [Bq/kg]) is given by Equation 8:

$$C_{fish} = \frac{\sum_{Te} C \cdot k_f/1000}{w} \tag{8}$$

where

C_{fish}: is the activity concentration in the fish [Bq/kg],
$\sum_{Te} C$: is the accumulation time-integrated activity concentration in water,
k_f: is the uptake rate [kg],
w: is the weight of the fish [kg].

2.3 Calculation of the doses

Exposure pathways are divided into two types: internal and external. Internal contamination occurs via aquatic food chain, and via ingestion of water. External exposures have more different pathways. Exposure due to the water mass itself occurs via swimming and boating, and from contaminated sediment via sunbathing. The calculation of the dose can be described with the following general equation:

$$D = \int K\, C\, dU_p \tag{9}$$

where:

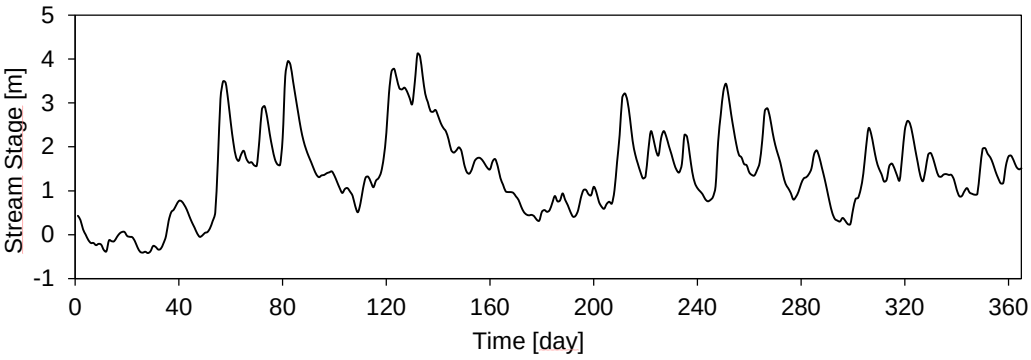
C: is the activity concentration of the media (water, sediment, fish),
U_p: is the habit data,
K: is the dose conversion factor.

Dose conversion factors for ingestion can be found in literature like IAEA BSS [5]. For external doses, calculations were made with the program Microshield [8].
The other main variable is the habit data. Parameters can be found in literature for different age categories [1; 6; 7]. These data are averaged values, describing the whole population.

3 RESULTS

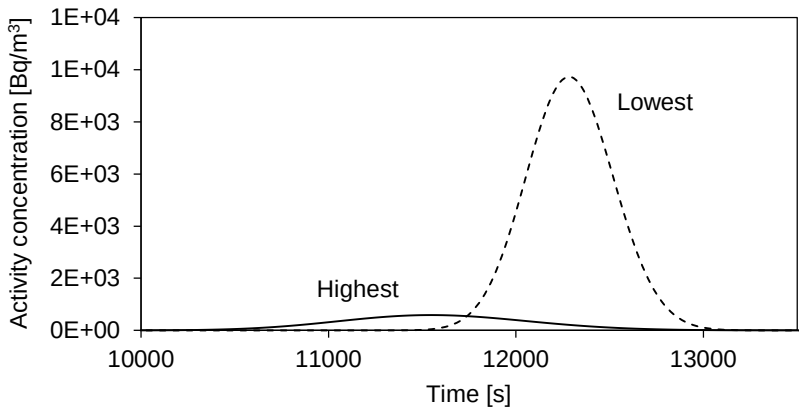
One of the most important parameter is the river stream stage as it changes daily (Fig. 7). Sensitivity analyses were made to see the change of the activity concentration in different media in the function of the stream stage. Analyses were carried out for isotope I-131 based on a hypothetical aquatic release of 1 GBq activity. The release was taken as an instantaneous emission from a shore side point source.

Figure 7: The daily change of the stream stage at Paks during 2017



The ever measured lowest and highest f values for Paks are: -0.58 m, 8.91 m. Calculations were made with these two values, and it can be seen in Fig. 8, there is a magnitude difference between the peak values.

Figure 8: Activity concentration at the shore side for the lowest and highest measured stream stage



Analyzing the average monthly data of the stream stage for the last 10 years, it can be seen that the highest river stages occur in the late spring and early summer. Studies were conducted to see the connection between the stream stage and the activity concentration in water, suspended sediment and in the fish. Figures 9-11 show these connections. In all the three cases, the increase of the water level decreases the activity concentration in the media can be noticed.

Figure 9: Connection between the stream stage and the activity concentration in water

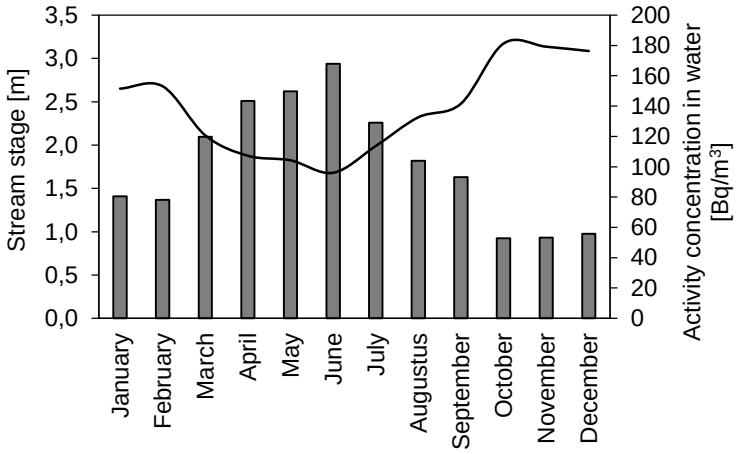


Figure 10: Connection between the stream stage and the activity concentration in suspended sediment

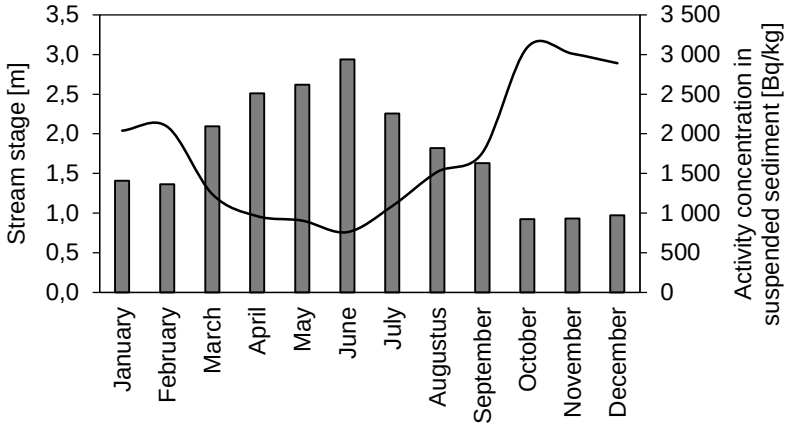
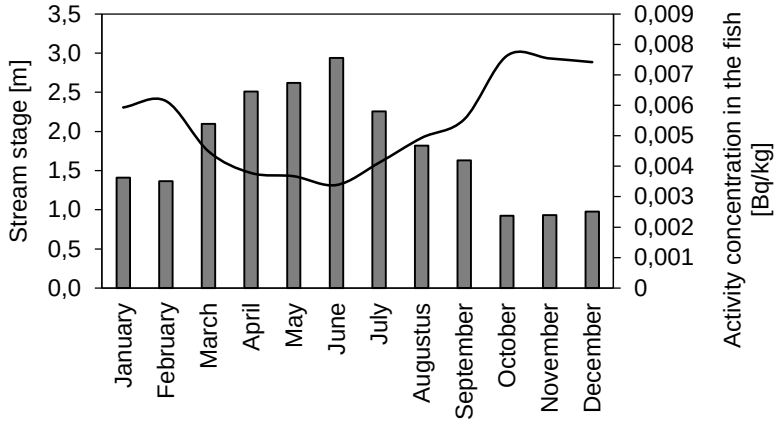


Figure 11: Connection between the stream stage and the activity concentration in the fish



4 CONCLUSION

After an accidental radionuclide release, it is important to use the best-estimated site specific hydrological variables during a decision support modelling. Most of these parameters are derived from the daily measured stream stage. Studies were conducted to find connection between the stream stage, and the activity concentration in different media. The results shows, that increase of the water level cause a better lateral mixing, therefore the activity concentration decreases in water, suspended sediment and in the fish. These results indicate that using a maximum, minimum or yearly averaged flow rate value for calculating can lead to incorrect estimations, and thus decisions.

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Occupational

Monte Carlo simulations for individual dosimetry of workers in disposal facilities for spent nuclear fuel

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Abstract. This paper provides information about potential radiation exposure of workers involved in emplacement of casks with spent nuclear fuel in generic deep geological disposal facilities. Monte Carlo simulations were employed for a close to reality estimation of the personal dose during individual working scenarios. To have an indication of quality of the employed approach, simulations were compared with experiments performed with 2.5 MeV neutrons of a neutron generator. Based on simulation results, comparative studies were performed that analysed emplacement scenarios of an approximately equal amount of spent nuclear fuel into claystone and rock salt.

KEYWORDS: Deep geological disposal facility, spent nuclear fuel, radiation exposure for workers, MCNP6

1 INTRODUCTION

A proper and safe management of spent nuclear fuel (SNF) discharged from commercial nuclear power plants is a subject of societal concern that cannot be solved by engineering alone. This work was hence realized within the interdisciplinary research platform "ENTRIA" [1]. This platform was established with the purpose of comparing different disposal options from various technical, ethical-moral, legal, and social aspects. Three disposal concepts were proposed in the platform, i.e., deep geological disposal with or without provisions of retrieval and long-term aboveground storage.
Certain working procedures in the storage / disposal facilities might lead to an enhanced level of radiation exposure for workers. Hence, a realistic estimation of the personal dose during individual working scenarios is desired, in order to compare the different disposal options from the aspect of radiological risk on the workers.
The Monte Carlo code MCNP6 [2] was employed to simulate the radiation fields and exposure in a final disposal in a rock salt and a claystone repository. In both cases, POLLUX® casks [3] containing a mixture of spent UOX and MOX fuel assemblies were investigated.
As a result of the investigations it was shown that neutron radiation dominates the radiation field and in turn determines the dosimetry. The contribution of backscattered neutrons also plays an important role [4].
To assess the modelling and simulation approaches applied in MCNP6, experiments with cask-type components were carried out at the TU Dresden neutron generator (neutron energy 2.5 MeV) located at the Helmholtz Zentrum Dresden Rossendorf (HZDR).
Finally comparative studies were performed that analysed emplacement scenarios of an equal amount of SNF into claystone and rock salt.

2 MATERIAL AND METHODS

2.1 Basis

The investigation of emplacement scenarios is based on the existing concept of the POLLUX®-10 cask [3]. This cask is designed as a technical barrier to prevent the release of the radioactive nuclides contained in the nuclear waste as well as to facilitate a proper protection of the workers from radiation. Working scenarios for emplacement of a POLLUX®-10 cask in rock salt are based on the proposal of

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DBE TECHNOLOGY GmbH [5]. The geometric parameters for a horizontal emplacement drift located in a generic rock salt and claystone repository were taken from reference [6].

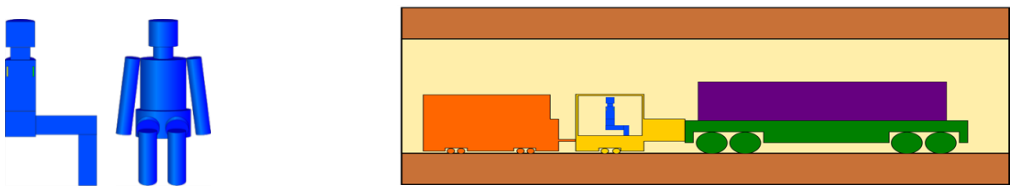
2.2 Waste Inventory

The representative waste inventory for the casks is a mixture of ~90% UOX and ~10% MOX SNF assemblies, which complies with the average composition of SNF from German reactors to be disposed of [7]. Hence, for rock salt nine UOX and one MOX fuel rods were considered as content of a POLLUX®-10. Moreover, an average burnup of 55 GWd/tHM and a cooling time of 50 years after unloading from reactor was taken into account. Regarding disposal in an argillaceous host rock, a lower waste load per cask, compared to POLLUX®-10 loading for a rock salt repository, was considered to keep the thermal impact below the level of heat-induced irreversible alteration of clay minerals. Thus, eight UOX and one MOX fuel rods were distributed to three casks with the outer dimensions of a POLLUX®-10 cask.

2.3 Monte Carlo

The geometrical layout of the POLLUX®-10 cask was chosen according to the report of Janberg and Spilker [3]. However, some simplifications were made to model the cask for MCNP6. The area with the nuclear waste inside the internal cask was assumed as a homogeneous mixture. For this mixture an effective density of 3.23 g/cm3 was assumed, including fission and activation products, structure materials (e.g. Zircaloy), and void volume filled with air.
The MCNP6 model for a horizontal emplacement drift located in a generic rock salt and claystone repository was adapted from reference [6].
The distance between a POLLUX®-10 cask and the end site of the drift amounts to 2.63 m and the distance between the casks was chosen as half the length of the cask.
The wall thickness of the emplacement gallery model was limited to 1 m, which accounts for most of the radiation interactions with the walls. In case of claystone, concrete as material of the supporting liner for a drift in claystone was used.

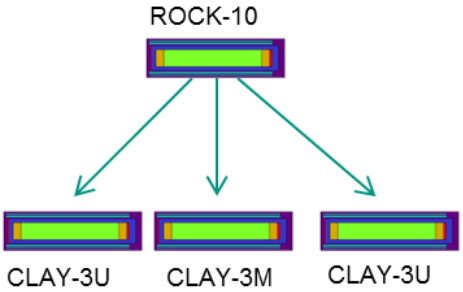
Figure 1: Employed phantom (left) and cross section of a snapshot of an emplacement scenario (right). Blue: phantom, yellow: drivers cab of the electric locomotive, orange: locomotive, green: carriage, brown: walls of emplacement drift, beige: air.



To determine personal dose equivalent $H_P(10)$ in working scenarios, a model of a worker was given by a simplified anthropomorphic phantom. Details are described in publication [8]. The employed phantom is a virtual representation of the BOMAB (BOttle MAnnikin ABsorber) phantom [9]. Fig. 1 shows a presentation of the sitting phantom for MCNP6. A representative "model snapshot" of the DBE TECHNOLOGY GmbH based working scenarios for emplacement of a POLLUX®-10 cask, the first of the four investigated steps, is also shown in Fig. 1: the POLLUX®-10 cask is transported on a carriage through the drift by means of an electric locomotive with the phantom sitting as driver inside the drivers cab.
For comparison of radiation exposure from a POLLUX®-10 cask emplaced in rock salt to radiation exposure from POLLUX®-10 type casks emplaced in claystone, the following acronyms are introduced: "ROCK-10" stands for a POLLUX®-10 cask containing 9 UOX and 1 MOX fuel rods in rock salt, "CLAY-3M" for a POLLUX®-10 cask with 2 UOX and 1 MOX fuel rods in claystone, and "CLAY-3U" for a POLLUX®-10 cask containing 3 UOX fuel rods in claystone.
With the aim that a similar amount of SNF should be emplaced, a ROCK-10 corresponds approximately

to one CLAY-3M plus two CLAY-3U. This is illustrated in Fig. 2.

Figure 2: Schematic sketch for the disposal of a similar amount of SNF in rock salt and claystone drifts. For details, see text.



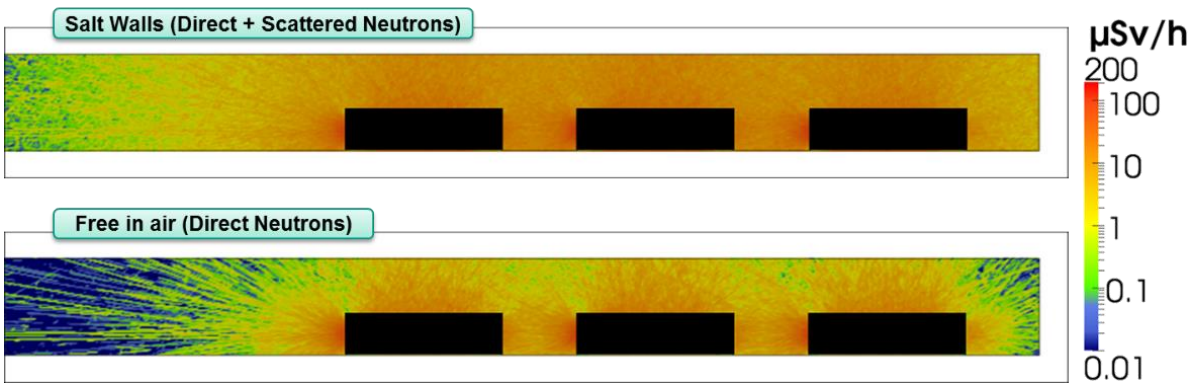
2.4 Neutron Generator

In addition to the numerical simulations of the emplacement scenarios, the radiation field calculations were experimentally tested. In view of the dominating neutron radiation, the simulations were compared to experiments performed at a neutron generator.
The TU Dresden neutron generator, located at the Helmholtz Zentrum Dresden Rossendorf (HZDR), was utilized producing 2.5 MeV neutrons from D-D fusion reaction at a neutron generation rate up to $5*10^9$ n/s. Different material combinations based on POLLUX® components were positioned between the target of the neutron generator and a NE213 detector. Neutron flux spectra were measured and corresponding simulations with the code MCNP6 were performed [10, 11].

3 RESULTS

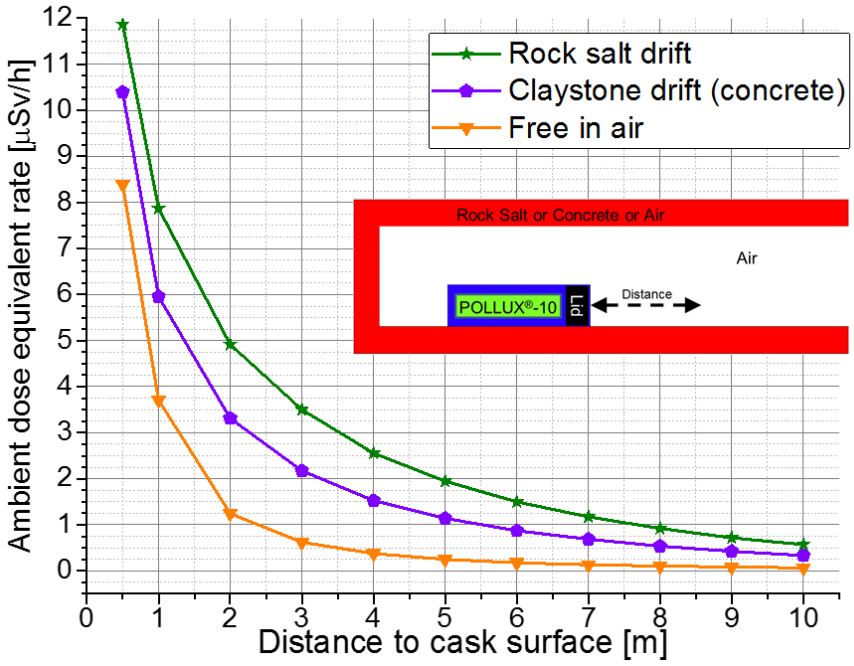
According to previous studies [4], for the considered waste inventory, neutrons dominate the radiation field inside the emplacement drift. The effect of the walls of the rock salt drift on the neutron dose rate is illustrated in Fig. 3. As shown in reference [4], due to the neutrons backscattered by the rock salt layers, the neutron dose equivalent rate in the emplacement drift is increased about an order of magnitude at 10 m distance to the POLLUX®-10 casks.

Figure 3: Comparison of ambient neutron dose-rate distribution $\dot{H}^*(10)$ around an ensemble of three POLLUX®-10 casks in a rock salt drift (top) and free in air (bottom).



As demonstrated in reference [8], the distance and material dependent effect of the gallery on the ambient dose rate equivalent is shown in Fig. 4. The higher amount of backscattering in rock salt and the higher moderation in the concrete walls in claystone yield higher dose rates in the rock salt drift.

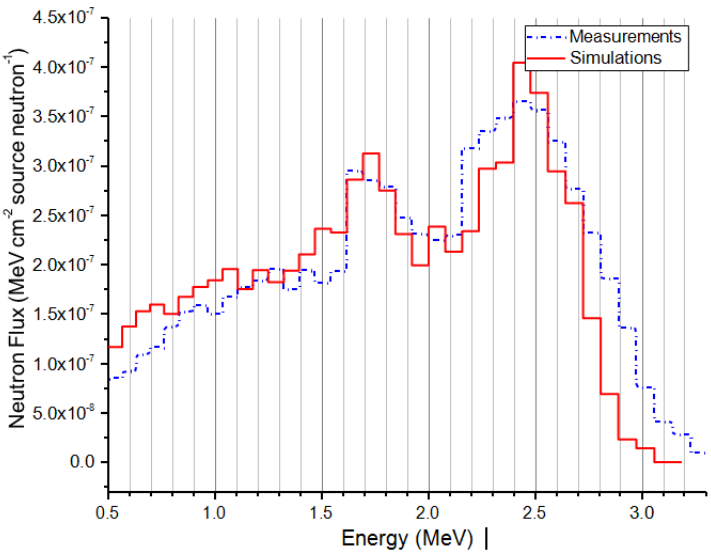
Figure 4: Ambient dose equivalent rate $\dot{H}^*(10)$ at different distances to the lid side of a POLLUX®-10 cask in rock salt, claystone, and free in air (based on data derived from [8, 11]).



At this stage, claystone seems to be favorable from the radiation exposure point of view.

The numerical simulations of the emplacement scenarios were experimentally tested. The MCNP6 simulations were compared to experiments performed at the neutron generator mentioned above. Different material combinations based on POLLUX® components were employed and a reasonable agreement between experiments and simulations was demonstrated. A representative result is shown in Fig. 5, obtained by a configuration of steel and polyethylene plates, interposed between the neutron generator target and the NE213 detector (more details are given in references [10, 11]).

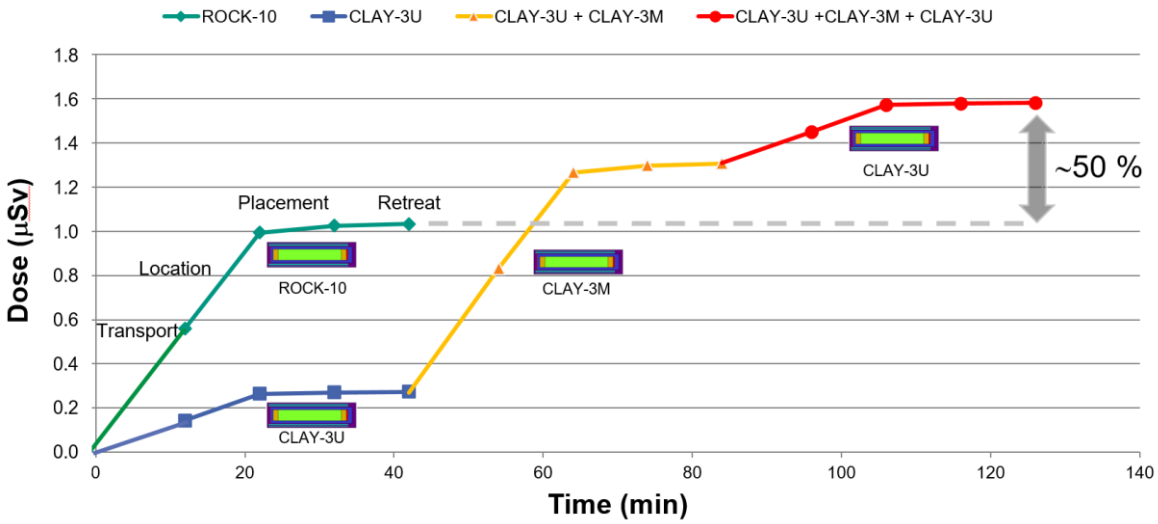
Figure 5: Comparison of the simulated und measured neutron flux spectra for the material configuration "4 cm steel + 10 cm polyethylene + 4 cm steel" between the neutron generator target and the NE213 detector (data derived from references [10, 11]).



Assured that our methodology is promising to assess simulation scenarios with shielding casks containing SNF, we turn towards the investigation of emplacement scenarios.

The first of the four investigated steps (Transport) is described above and shown in Fig. 1. In the next step (Location), when the transport ensemble reaches the emplacement position, the cask is positioned under a storage equipment, which elevates the cask from the carriage to allow the entity locomotive and carriage to drive back. In a third step (Placement), the entity locomotive and carriage is driven back and with the help of the storage equipment, the cask is placed on the ground. Finally (Retreat), the entity locomotive and carriage connects to the rail mounted storage equipment and moves it to the next emplacement position.

Figure 6: Emplacement of same amount of waste in rock salt and claystone. Working time for the different steps vs. the dose $H_P(10)$.

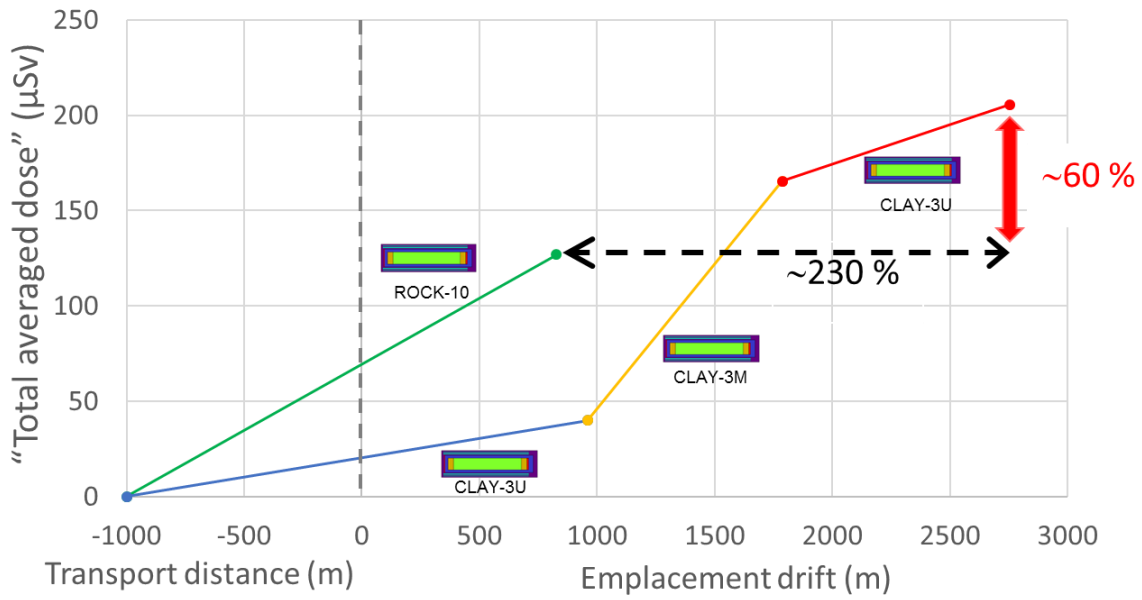


These four steps were considered for the emplacement of a similar amount of SNF in rock salt and claystone. The thermal impact on the host rock determines the load of the casks so that, as above-mentioned, a ROCK-10 disposal is compared to one CLAY-3M plus two CLAY-3U. Results are displayed in Fig. 6. The steps for the hypothetical example of 1 km Transport with 5 km/h, and Location, Placement and Retreat, each of the latter with a duration of 10 min, are displayed. For the claystone scenario, first CLAY-3U is disposed, followed by CLAY-3M and terminated with CLAY-3U. The steps Transport and Location are connected to the highest dose. In these cases, the driver is all the time located closer to the cask than for the other steps. For ROCK-10 a total dose of more than 1 µSv is estimated. The same amount can be attributed to CLAY-3M. As two CLAY-3U have to be added, the total dose is about 50% higher for the claystone scenario reaching up to 1.6 µSv. More details can be found in the publications [8, 11].

This hypothetical scenario was extended to a larger amount of casks, which possibly can be handled per annum. In a second example the emplacement of one hundred ROCK-10 are compared to one hundred CLAY-3M plus 233 CLAY-3U. Fig. 7 shows the result. Moreover, in this example, the number of UOX and MOX fuel rods disposed of in the different host rock types is equal.

The “Total average dose” and the needed emplacement drift space are compared, when disposing one hundred ROCK-10 vs. 116 CLAY-3U plus one hundred CLAY-3M plus 117 CLAY-3U casks. As a result, the claystone scenario yield about 60 % more dose and about 1.8 km more emplacement drift space.

Figure 7: Emplacement of one hundred ROCK-10 vs. one hundred CLAY-3M and 233 CLAY-3U casks. Transport and emplacement drift distance vs. the "Total average dose" $H_P(10)$. The lines connecting the points are added to guide the eye.



4 SUMMARY and CONCLUSION

The investigations were performed with Monte Carlo simulations for individual dosimetry in disposal facilities. For the considered SNF, the neutron radiation was found to be the dominant component of the radiation field and in turn rules the respective dosimetric considerations. The contribution of backscattered radiation plays an important role with an increased contribution in rock salt.
For the test of the simulations, experiments were performed with 2.5 MeV neutrons produced by the neutron generator of TU-Dresden located at HZDR. From the considered neutron flux point of view, the approach was confirmed to be promising to assess simulation scenarios with shielding casks containing SNF.
Comparison studies of emplacement scenarios in rock salt and claystone considered the same working steps and waste amount. However, a lower waste load per cask was considered to keep the thermal impact below the level of heat-induced irreversible alteration of clay minerals.
As a result the emplacement of the same amount of SNF in claystone yields a higher dose and needs a longer emplacement drift compared to the rock salt scenario.
It should be emphasized that the hypothetical scenarios are rather fast working approaches and the worker/phantom is shielded by the drives cabin at a certain distance to the cask. In this case, the estimated annual dose for one hundred ROCK-10 vs. one hundred CLAY-3M and 233 CLAY-3U casks is in the range of 120 to 210 µSv. Other scenarios, when workers are staying closer to the cask would be associated with higher doses.
To conclude, the described methodology is ready to be applied for new concepts. After what has been said so far, detailed description of the working steps and the dimensions of disposal facilities are required to evaluate dose expositions more accurately.

5 ACKNOWLEDGEMENTS

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EURADOS Action on Harmonisation of Individual Monitoring for External Radiation: Intercomparisons, Surveys, and Training and Networking Activities

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Abstract

EURADOS is a network of more than seventy institutions involved in research and other radiation dosimetry activities, from across Europe. By means of training courses, intercomparisons and networking activities, EURADOS promotes the maintenance of expertise and hence of sustainability in radiation protection. This paper gives and overview of relevant EURADOS activities, focussing specifically on its work on harmonisation for measurements of occupational doses from external radiation.

1 INTRODUCTION

In the field of radiation protection, activities are often carried out by small numbers of highly-specialised staff. One such activity is monitoring of the doses received by individual workers from ionising radiation, which helps employers to fulfil their health and safety responsibilities for their employees. Across Europe there is a significant variation between Individual Monitoring Services (IMS) in terms of their methods, their required staffing levels, their access to technical support and their contexts within parent organisations. In larger IMSs, staff often work full-time on operation of the service, and the higher throughputs mean that experience builds relatively quickly; whilst staff of smaller IMSs are likely to have other responsibilities that occupy their time and restrict the building of relevant IMS experience. There are also differences in the amount of dialogue that takes place

between IMSs within the same country. Thus, the opportunities for personal development can be limited. In order to sustain expertise for the future, these risks need to be addressed.

2 EURADOS

EURADOS (www.eurados.org) is a network of more than seventy institutions involved in research and other radiation dosimetry activities, from across Europe. It is registered as EURADOS e. V. in the German Register of Societies as a non-profit association and carries out a range of work, promoting scientific research though its members and by collaboration with other research platforms. It helps to raise the profile of radiation protection and supports young scientists by means of awards, and it also carries out a number of harmonisation and training activities. As well as the institutions (voting members), EURADOS can call upon the expertise of over five hundred individual scientists (associate members).

EURADOS operates by:

- Organization of scientific meetings and training activities.
- Organization of intercomparisons and benchmark studies.
- Coordination of working groups which promote both technical development and its implementation in routine work, which contribute to compatibility within Europe and to conformance with international practices.

EURADOS covers a wide range of topics in radiation dosimetry [1], as illustrated by the themes for the current working groups in Table 1.

Table 1. EURADOS Working Groups

Working Group	Topic
WG2	Harmonisation of individual monitoring
WG3	Environmental dosimetry
WG6	Computational dosimetry
WG7	Internal dosimetry
WG9	Radiation dosimetry in radiotherapy
WG10	Retrospective dosimetry
WG11	High energy radiation fields
WG12	Dosimetry in medical imaging

Dissemination of research findings, recommendations and so on is an important way of promoting sustainability, and all working groups contribute to this. As well as papers in peer-reviewed journals and presentations at conferences and workshops, EURADOS produces its own series of reports that are available from the web site. EURADOS also arranges well-attended annual meetings and individual monitoring conferences, and with other platforms contributes to events such as the European Radiation Protection Week. Amongst the events at the EURADOS annual meeting are Winter Schools: these are lecture-based and intended for students and those new to the subject area. Recent topics have included:

- Application of physical and computational phantoms in dose assessment.
- Internal dosimetry for radiation protection and medicine.
- Dosimetry for epidemiological cohorts.

3 HARMONISATION – EXTERNAL RADIATION

In 1996, the European Commission published a Basic Safety Standards Directive [2] for radiation protection. Further, at the time more states were joining the European Union. There was, therefore, a need to help ensure consistency of standards across Europe. With these factors in mind EURADOS set up a task group [3] with the aim of promoting Harmonisation of Individual Monitoring in Europe. ("Harmonisation" did not require that all IMSs should work in identical ways, but rather that the various methods and approaches would achieve similar standards of performance and protection.) Past work on harmonisation has included:

- Comprehensive surveys of legislation, standards and practice [4],[5],[6].
- Organisation of Individual Monitoring Conferences.
- Update of EC Technical Recommendations (see below).
- QA Surveys, comparing methods used, quality management systems, and common sources of uncertainty and error [7],[8].
- Organisation of benchmarking intercomparisons.

Since then EURADOS has pursued a number of activities that have successfully met that aim, but that have incidentally provided opportunities for training, development and networking. These opportunities have helped to ensure that best practice is rapidly shared amongst staff, both new and experienced, of IMSs. Harmonisation activities continue to include: regular self-sustaining IMS intercomparisons and the associated participants' meetings; training courses based on the technical recommendations produced by EURADOS for the European Commission; surveys, to establish perceptions and needs; "learning network" events at the EURADOS annual meetings; and the operation of an online discussion group.

4 HARMONISATION - GENERAL

The work on harmonisation in measurement of occupational doses from external radiation, described in detail below, is only part of the contribution of EURADOS to harmonisation and hence to sustainability. For example, benchmarking intercomparisons have been carried out for a variety of methods, including eye lens dosimetry (Working Group 12), intakes of radionuclides (WG7), environmental dosimetry calibration and measurements (WG3), codes for aircrew dosimetry (WG11), retrospective dosimetry (WG10) and computational methods (WG6). In addition to the Winter Schools mentioned above, specialised training courses have been held, for example those covering dosimetric techniques such as the use of voxel phantoms, uncertainty estimation in retrospective dosimetry, and the application of Monte Carlo methods.

Underlying all of this is substantial collaborative research work, between EURADOS members and with other institutions, leading not only to the advancement of radiation dosimetry knowledge but also to the sharing of expertise and dissemination of the results. Further details of the range of EURADOS' work can be found on the web site (see above), where information on the strategic research agenda is also available [9],[10].

5 TECHNICAL RECOMMENDATIONS

Important among the outputs of the EURADOS harmonisation program have been the publication of revised Technical Recommendations for Monitoring Individuals Occupationally Exposed to External

Radiation [11], [12] under contract for the European Commission, covering a range of topics from calibration and traceability to quality assurance and record keeping. The recommendations have been published as commission report no. RP160. Meanwhile, companion recommendations for the measurement of doses arising from intakes of radionuclides have also been developed [13].

6 HARMONISATION IN MEASUREMENT OF OCCUPATIONAL DOSES FROM EXTERNAL RADIATION – CURRENT ACTIVITIES

6.1 Self-Sustaining Intercomparisons

Following its work on harmonisation in the early 2000s, EURADOS identified [6] a number of areas for work, one of which was the revision of the technical recommendations mentioned above [11]; another was the "organisation of intercomparison exercises and performance characteristics testing on a regular basis". This was developed as a series of self-sustaining intercomparisons for systems assessing occupational doses from external radiation, beginning in 2008 with an intercomparison for whole-body photon dosemeters and going on to include exercises for extremity and neutron dosemeters. In the intercomparisons, exposures are selected to include a variety of radiation energies and angles as well as common calibration conditions and, to provide a thorough test of the systems, mixed exposures are also included. The intercomparisons are carried out with strict anonymity, and results are reported to participants within a few months of the end of the exercise. This is followed by a participants' meeting, normally held during the EURADOS annual meeting, at which results are presented, questions are answered and discussions take place. Costs of the irradiations are met by means of charges on the participants, who can administer their entries via an online portal.

Detailed reports of the individual intercomparisons are published on the EURADOS website, and companion papers are frequently presented at conferences or in the open literature: see, for example, [14] and [15]. The intercomparisons have proved popular, with around 100 systems now entering for the whole-body photon exercises.

The intercomparisons thus support harmonisation by promoting sharing of best practice via reports and discussions.

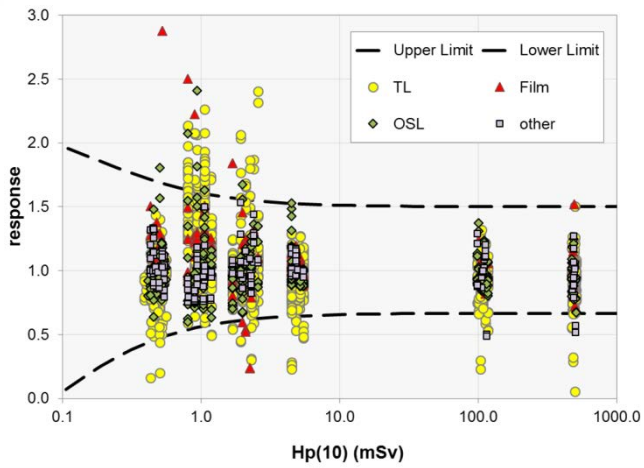

Figure 1: Example analysis chart from an intercomparison report, showing the performance of different systems at different dose levels.

Figure 2: Dosemeter types submitted for the 2016 intercomparison.

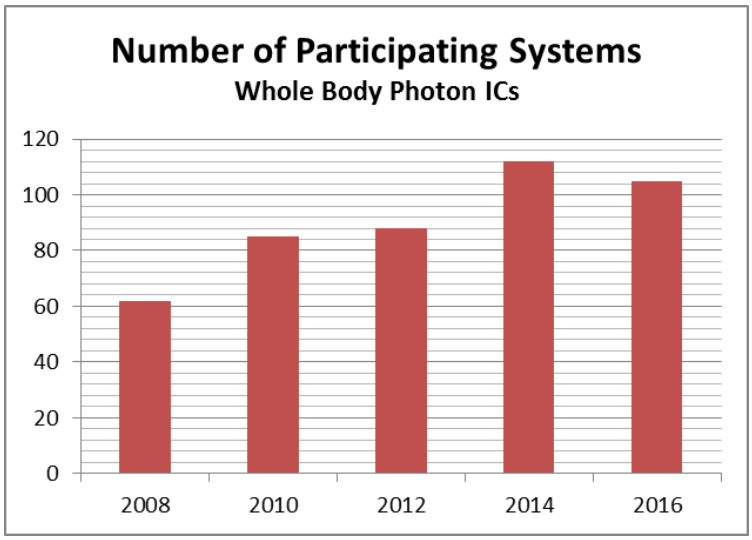

Figure 3: Number of participants in the whole-body photon intercomparisons.

6.2 Training Courses

The production of a set of technical recommendations is a significant achievement, but there is no guarantee that they will be used unless they are disseminated. Following the publication of the "RP160" recommendations, presentations were given at conferences but, most importantly, EURADOS also embarked on a series of courses aimed at training the staff of IMSs – particularly junior and trainee managers. The courses involve lectures, exercises and laboratory or workplace visits and, like the intercomparisons, aim to cover their costs via the fees charged. The content mainly follows that of the recommendations [11], but has evolved to take account of new developments, for example in the increased importance of eye lens dose [16]. Topics include:

- Framework for Individual Monitoring.
- Dosimetry Concepts and Quantities.
- Individual Monitoring Procedures.

- Uncertainties and Accuracy Requirements.
- Calibration and Type Testing.
- Criteria for Approval of Dosimetry Services.
- Dose Reporting, Record Keeping and Information Systems.
- Reliability, Quality Assurance and Quality Control.

The courses have continued to prove popular despite the time that has lapsed since the publication of "RP160", showing that there is a continuing need for specialised training in this area of radiation protection. The first course was held in 2012 in Krakow, and the most recent in 2017 in Florence.

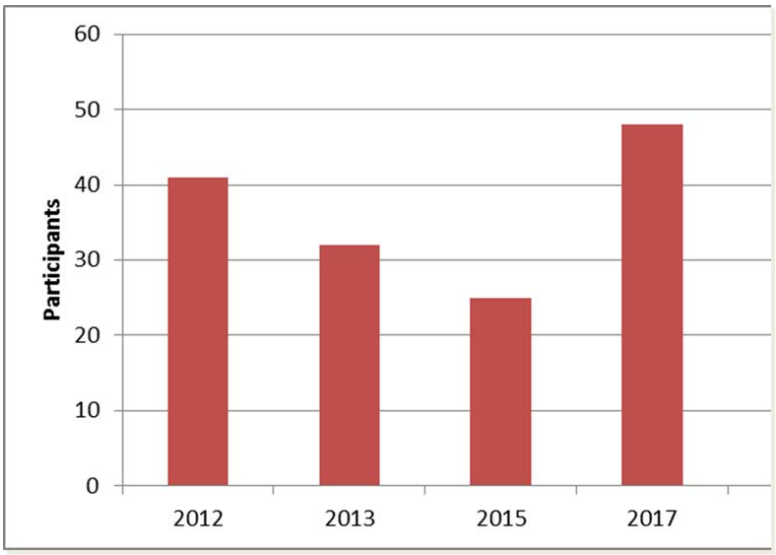

Figure 4: Number of participants in the training courses on the recommendations for monitoring of doses from external radiation.

Planning is already under way for a further course in 2019. Meanwhile, EURADOS Working Group 7 is planning to hold a similar course for the new recommendations on assessment of occupational doses from intakes [13] – this will be announced on the EURADOS web site.

6.3 Networking Activities

Experience in running the intercomparison participants' meetings and the "RP160" training courses has been very positive. Discussions are often lively, and participants have often expressed enjoyment of the informal discussions that also take place, e.g. coffee-break conversations. Accordingly Working Group 2 has developed two new Networking Activities.

The first is a "Learning Network" event which is designed to give participants just such opportunities for discussion. Two Learning Networks have taken place so far. They have been held during EURADOS annual meetings, so that no extra journeys are necessary for the participants, and they have included focussed discussions on a range of relevant topics, for example on calibration, or the treatment of damaged dosemeters. The events have proved popular, providing as they do a way for people to become more involved in the work of EURADOS, as well as a way of sharing and disseminating good practice.

Of course, not everyone in the IMS community has time to attend the EURADOS annual meeting. The second development, therefore, is an online discussion forum known as "IMS-DG" (Individual Monitoring Services Discussion Group, which can be found at https://ims-dg.org/). The group allows for different discussion topics and threads, with e-mail alerts, and is moderated by EURADOS WG2 members.

6.4 Guidance on Interpretation of ISO 17025

One recommendation contained in "RP160" was that IMSs should follow a formal quality management system, amongst which the most appropriate was ISO 17025, *General Requirements for the Competence of Testing and Calibration Laboratories* [17]; and the subsequent survey of quality assurance conducted by WG2 in 2012 [8] showed the increasing interest in this standard amongst IMSs. However, discussions at Training Course and Learning Network events showed that, because IMSs represent a relatively novel application for ISO 17025, there could be differences in how the various requirements were applied. The most recent work stream for WG2, therefore, is the development of guidance on how to interpret ISO 17025 in the context of Individual Monitoring Services. The guidance is intended for the use of both IMSs and auditors, although it is stressed that the authoritative judgements remain with the auditors, and that following the guidance will not necessarily lead to accreditation. The work stream is intended to conclude during 2019, with a report that refers to the new version of ISO 17025 [18].

7 CONCLUSIONS

By means of its various activities in promoting research, in promoting collaboration and dialogue and in delivering training, EURADOS supports the dissemination of knowledge and skills and hence the sustainability of expertise in radiation protection. Amongst the work of EURADOS, Working Group 2 focuses on harmonisation in the particular field of individual monitoring for external radiation, by means of self-sustaining intercomparisons, training courses, networking activities and the production of recommendations.

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Regulatory oversight of safety culture in the nuclear industry

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Abstract: Safety culture is a well-known concept in the nuclear industry and is recognized as an important factor in achieving high levels of safety performance. The term was first used to explain how the lack of knowledge about risk and safety and failure to act appropriately contributed to the Chernobyl incident.

In the Netherlands, both regulators and licensees in the nuclear industry are required to strengthen and improve their safety culture. According to Schein, any culture should be studied at three levels, from the very visible (conscious) to the tacit and invisible (unconscious) ones. The IAEA developed a framework for a strong safety culture, consisting of five overarching safety culture characteristics, which can be used by organizations as a reference when assessing and improving safety culture.

It should be clear that in the case of non-compliance or identification of unforeseen risks, the ANVS requires the licensee to take appropriate corrective actions. Regulatory oversight of safety culture complements compliance-based control with proactive control activities. In this paper, I explain five methods that support regulatory oversight of safety culture, and thereafter share lessons learned.

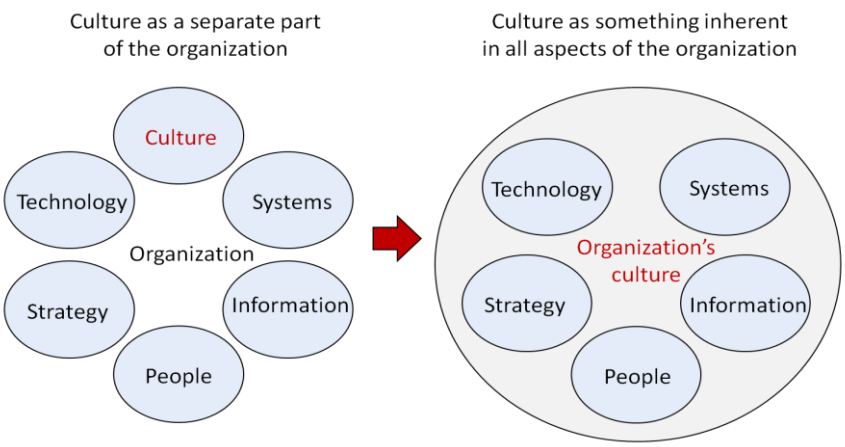
1 Introduction

An organization's culture is formed, shared and supported by people. The way of thinking about an organizational culture has evolved over time and has been the topic of interest for managers, organization specialists and scientists. Organization culture, originally, was considered to be an elusive social process, but it has become increasingly clear that understanding culture is critical to understanding organization performance.

1.1 An organization is its culture

Nowadays, culture is considered to be inherent in all aspects of the organization and not as one variable amongst others [1], see Fig. 1. An organization's culture is not an isolated phenomenon, nor can it be 'implemented' or 'removed'. Therefore, it can be deduced that an organization is its culture.

Figure 1: Culture is inherent in all aspects of an organization.



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The impact of an organization's culture on its performance is due to the nature of interactions between individuals, departments and hierarchies, as well as relationships with external organizations. The quality of these interactions determines how people collectively engage, share information and integrate efforts. An organization's culture influences every aspect of how the organization's members behave, from the development of the key performance indicators to the manifestation of defence-in-depth principles. A culture shapes what the organization considers to be 'right' and 'wrong'. A strong organizational culture is crucial in avoiding and managing incidents and in addressing safety as well as other competing priorities like security, integrity and continuous improvement.

1.2 Edgar Schein's iceberg metaphor

The ANVS embraces the IAEA's approach to organizational culture, which is based on Edgar Schein's iceberg metaphor. According to Schein, organizational culture comprises three levels: behaviour/artefacts, espoused values and basic assumptions [2], see Fig. 2.

Figure 2: Schein's iceberg metaphor.

Behaviours and artefacts (which comprise the top level of organizational culture) represent the external physical manifestation of a culture. Examples are safety performance data, policies, standards of housekeeping, material conditions, organization charts, communication styles, and how findings are addressed and incidents investigated.

The next level of organizational culture is comprised of espoused values. This level reflects how members represent the organization. This is often expressed in official philosophies and public statements of identity and frequently includes a projection for the future of what members hope to become. Occasionally, inconsistencies between the observable behaviour and the espoused values are evident. An example of this is when management argues that safety is the highest priority (espoused value), but in practice employees take risks to achieve production objectives (behaviour). Such an inconsistency clarifies that behaviour is driven by an even deeper cultural layer of beliefs, perception and feelings, for instance: beliefs like "we are safe" because "the design is robust, highly automated and there are ample safety margins".

This deepest level of culture represents sets of underlying assumptions, i.e. unconscious beliefs, perceptions and feelings. Schein indicates that assumptions are often taken for granted to the extent that there will be little variation within social units. This deepest cultural layer strongly determines behaviour .

2 Definition of and framework for a strong safety culture

The safety culture concept was first applied formally to the nuclear power industry by the International Atomic Energy Agency's (IAEA) International Nuclear Safety Advisory Group [3]. The term was used to explain how the lack of knowledge about risk and safety and failure to act appropriately contributed to the Chernobyl incident. The human factor was determined to be a major element contributing to the incident. It was found that personnel had an insufficient understanding of technical procedures involved with the nuclear reactor, and knowingly ignored regulations to speed up test completion.

2.1 IAEA framework for a strong safety culture

The IAEA defines a strong safety culture as "*the assembly of characteristics and attitudes in organizations and individuals which establishes that, as an overriding priority, protection and safety issues receive the attention warranted by their significance*". The IAEA has also developed an international framework for strong safety culture, consisting of the following five overarching safety culture characteristics [4]:

- Safety is a clearly recognized value;
- Leadership for safety is clear;
- Accountability for safety is clear;
- Safety is integrated into all activities;
- Safety is learning-driven.

Each of these safety culture characteristics encompasses attributes that have been identified as essential for achieving a strong safety culture. These are described in what follows.

Safety is a clearly recognized value:
- The priority given to safety is reflected in documentation, communications and decision-making.
- Safety is a primary consideration in the allocation of resources.
- The strategic business importance of safety is reflected in the business plan.
- Individuals are convinced that safety and production go hand-in-hand.
- A proactive and long-term approach to safety issues is reflected in decision-making.
- Safety-conscious behaviour is socially accepted and supported (both formally and informally).

Leadership for safety is clear:
- Senior management is clearly committed to safety.
- Commitment to safety is evident at all levels of management.
- There is visible leadership which reflects the involvement of management in safety-related activities.
- Leadership skills are systematically developed.
- Management ensures that there are sufficient competent individuals.
- Management seeks the active involvement of individuals in improving safety.
- Safety implications are considered in change management processes.
- Management shows a continuous effort to strive for openness and good communication throughout the organization.
- Management can resolve conflicts as necessary.
- Relationships between managers and individuals are built on trust.

Accountability for safety is clear:

- An appropriate relationship with the regulatory body that ensures that the accountability for safety remains with the licensee exists.
- Roles and responsibilities are clearly defined and understood.
- There is compliance with regulations and procedures.
- Management delegates responsibility with appropriate authority to enable clear accountability to be established.
- 'Ownership' for safety is evident at all organizational levels and for all individuals.

Safety is integrated into all activities:

- Trust permeates the organization.
- Consideration of all types of safety, including industrial safety and environmental safety, and of security, is evident.
- The quality of documentation and procedures is good.
- The quality of processes, from planning to implementation and review, is good.
- Individuals have the necessary knowledge and understanding of the work processes.
- Factors affecting work motivation and job satisfaction are considered.
- Good working conditions regarding time pressure, workload and stress exist.
- There is cross-functional and interdisciplinary cooperation and teamwork.
- Housekeeping and material conditions reflect commitment to excellence.

Safety is learning-driven:

- A questioning attitude prevails at all organizational levels.
- Open reporting of deviations and errors is encouraged.
- Internal and external assessments, including self-assessments, are used.
- Organizational experience and operating experience (both internal and external to the facility) are used.
- Learning is facilitated through the ability to recognize and diagnose deviations, formulate and implement solutions and monitor the effects of corrective actions.
- Safety performance indicators are tracked, trended, evaluated and acted upon.
- There is systematic development of individual competences.

These attributes and characteristics can be used by organizations as a reference when assessing and improving safety culture and they also provide the framework for developing a common understanding of the safety culture concept.

3 Regulation of safety culture

Worldwide, the safety of nuclear installations is governed by national legislation and international conventions. Within the EU, this is supplemented by EU Directives. The objective of the amended Nuclear Safety Directive (2014/87/Euratom) is to maintain and promote the continuous improvement of nuclear safety. This Directive was enacted in August 2014 and was transposed into Netherlands' legislation in 2017.

3.1 Importance of safety culture is highlighted in EU Nuclear Safety Directive

The importance of safety culture is highlighted in this amended Nuclear Safety Directive, due mainly to a review of the EU framework on nuclear safety following the Fukushima incident in 2011. The Fukushima nuclear incident in Japan renewed worldwide attention to the measures needed to minimize risk and ensure the most robust levels of nuclear safety. The lack of a strong nuclear safety culture contributed significantly to the Fukushima incident and, as a result, safety culture was once again

highlighted as a major concern. Despite efforts to strengthen its safety culture, TEPCO[2] had not cultivated a questioning attitude, practiced safety-first decision-making, promoted organizational learning and challenged assumptions. A shared basic assumption that plants were safe was developed over time by the stakeholders of the Japanese nuclear industry [5].

3.2 Regulators and licensees must strengthen and improve their safety culture

In the Netherlands the regulatory framework consists of the Nuclear Energy Act, ministerial arrangements, directives and licensing conditions. Due to the transposition of the amended Nuclear Safety Directive into the Netherlands' regulatory framework, both regulator and licensee must strengthen and improve their safety culture. The regulatory framework also includes requirements for staffing, management systems, continuous improvement, learning organization, safety culture and quality assurance programmes for safe operation.

4. Regulatory oversight of safety culture

The prime responsibility for the safety of a nuclear installation lies with the license holder and is supervised by the regulator. The ANVS is responsible for establishing a regulatory framework and ensuring the licensee company fulfils its responsibilities.

The ANVS expects licensees to foster a strong safety culture in their organizations. The goal of regulatory safety culture oversight is to prevent potential future performance degradation and to initiate safety improvements. History has shown that organizations rarely recognize the early signs of deterioration. Incidents often occur because organizations gradually accept declining conditions and prioritize other concerns over safety.

Initiating safety improvements requires a deeper understanding of an organization's safety culture. To understand safety culture, behaviour and artefacts (the visible aspects) must be observed. By interpreting what people say, the behaviour of leaders and staff, and other visible aspects, strengths and weaknesses within a culture can be identified.

4.1 Regulatory oversight of safety culture complements compliance-based control

In the case of a violation during the safety culture oversight process, like non-compliance or the identification of unforeseen risks, the ANVS requires that the licensee take appropriate corrective actions or implement enforcements measures. However, it is possible to identify a latent condition that could lead to potential safety degradation which does not include a violation. In those situations, the regulator can complement compliance-based control with regulatory oversight of safety culture and perform proactive control activities. Compliance-based control only allows access to the upper levels of culture (behaviour and artefacts). Regulatory oversight of safety culture includes also the lower levels of culture like espoused values and basic assumptions.

The ANVS uses various methods for the regulatory oversight of safety culture. These include:
- independent safety culture assessment;
- self-assessments;
- inspections of management systems;
- gathering information during on-site inspections;
- special investigations.

The data collection methods are explained below.

[2] Tokyo Electric Power Company.

4.2 Independent assessment and self-assessment

An independent safety culture assessments as well as a self-assessments can be proactively conducted by the licensee or enforced by the ANVS. A combination of interviews, questionnaires, focus groups, document reviews and observations are to assess behaviour, artefacts, espoused values and basic assumptions. For instance: The aim of conducting observations is to reveal actual performance and behaviour in real time and they are used to address behaviour and artefacts. They provide information about people in their actual work context, e.g. how people interact, their work practices and what people pay attention to in everyday work situations. Interviews, on the other hand, can obtain in-depth information and points of view from individuals, thus leading to a better understanding of how interviewees think and feel, e.g. what is s/he passionate about, what is s/he concerned about and what tends to be ignored and avoided? Interviews can therefore help to access perceptions, values, beliefs and attitudes. A combination of different tools is used to systematically collect a wide variety of organizational data [6].

Contrary to self-assessments, independent assessments are conducted by an independent, external expert organization. The IAEA has developed a method to analyse and consolidate data into an 'image' of the organization's culture, which is then compared to the IAEA safety culture framework.

4.3 Inspections of management systems of licensee

Some of the ANVS's inspections focus on the licensee's management system due to the close relation between the safety culture of an organization and its safety management system. Licensees are obliged[3] to use the management system to promote and support a strong safety culture by [3]:

- ensuring a common understanding of the key aspects of safety culture within the organization;
- providing how the organization supports individuals and teams in carrying out their tasks safely and successfully, considering the interaction between individuals, technology and the organization;
- reinforcing a learning and questioning attitude at all levels of the organization;
- providing how the organization continuously seeks to develop and improve its safety culture.

However, reality tends to be more complex: an organization's safety culture does not always comply with the formal safety management system, nor can the safety culture of an organization be created or changed overnight [7].

4.4 Gathering information during on-site inspections

The ANVS makes use of a tool that supports the use of on-site inspections as a source of information in the personnel-organizational area, like quality of written documents, knowledge and competence, workload and housekeeping. The tool can be used during every supervision visit and helps inspectors with a technical background to be more aware of non-technical issues. An effective application of this tool requires suitable training and coaching. This tool mainly supports gathering information about artefacts. Its use in the continuous collection of information reflects its purpose as a long-term monitoring device, and it can also be used to help understand and/or detect negative developments in the area early.

4.5 Special investigations

A special investigation may be triggered by specific circumstances like an occurrence of an event. The design of a special investigation, and the reason for it, can vary greatly. In exceptional cases, the ANVS conducts this type of investigation but no standard approach is available. The tools, methods

[3] Sometimes a license condition requires the license holder to fulfil all requirements stated in a specific IAEA Safety Standard.

and topics need to be considered for each specific situation. Relevant organizational functions and responsibilities should always be included.

5. Lessons learned

The enhancement of a safety culture is a dynamic process: a culture cannot change rapidly and requires systematic long-term work, consistency and perseverance. After obtaining a better understanding of an organization's safety culture, we should be aware that a culture is seen as something we can influence, rather than something we can control.

The main lessons learned regarding regulatory oversight of safety culture are to:
- use a safety culture framework;
- invest in a common understanding of safety culture;
- combine different approaches;
- enable dialogue;
- question existing shared norms, values and basic assumptions.

5.1 Use a framework and invest in a common understanding of safety culture

In the oversight process, it is important that supervisors, license holders and other parties involved use the same language. The ANVS therefore invests in developing shared insight into the understanding of the key aspects of safety culture. Using a framework for safety culture supports this process. A regulator must avoid vague conclusions about safety culture as a whole, e.g. the culture is 'good or bad'[8]. These vague conclusions are not constructive. By comparing an organizational culture with the IAEA safety culture framework, specific weaknesses within a particular culture[4] can be identified.

5.2 A combination of approaches is most beneficial

The ANVS uses a combination of approaches for regulatory safety culture oversight. A combination is most beneficial because all the methods have their own pros and cons. The methods of 'inspecting a management system' and 'gathering information during on-site inspections' support long-time monitoring at a relatively superficial level. The safety culture assessments and special investigations provide greater understanding but also require more effort in a relatively short timeframe. The collection and analysis of data during the oversight process is an area the ANVS seeks to improve.

Over time, the ANVS has come to understand the difficulties involved in creating an all-encompassing image of an organization's culture based on the information/data gathered during the oversight process. The data collected by the combination of methods mostly represents the 'tip of the iceberg[5]', and analysing this data requires multidisciplinary expertise. The analysis must certainly consider the existence of various subcultures. The relationship between subcultures may have both positive and negative implications for the organization.

We have seen that the best results are achieved with an Independent Safety Culture Assessment (ISCA) conducted by the IAEA. An ISCA investigation lasts, on average, two to three weeks, comprises a minimum of two safety culture experts, depending on the size of the organization[9, 10]. Assessments of this kind have resulted in a positive dialogue between regulator and licensee, and

[4] This paper discusses regulatory safety culture oversight, but Schein's metaphor and the basics oversights approach are quite similar in terms of security culture, despite the differences in frameworks for assessing the related artefacts, behaviours, values and assumptions. An assumption regarding security culture could be 'credible treats exits' and related artefacts could be viewed as a visible security policy as well as safety performance data and communication styles.

[5] Artefacts are the most accessible and therefore constitute the largest part of the collected data.

increased the understanding of strengths and weaknesses within the licensee organization. Opportunities for improvements were identified and the licensees were committed to follow ups.

5.3 Enable dialogue

After analysing the data, the results can be communicated with the licensee. Regulatory oversight of safety culture will not generally lead to clear-cut and implementable results, but it will lead to an increased understanding of why different safety-related issues appear, and provide insight into what may be done to enhance safety. A dialogue is the preferred communication method because it is not about winning acceptance of a viewpoint but exploring every option and agreeing to do what is right.

5.3 Existing shared norms, values and basic assumptions should be questioned

Basic assumptions like “a severe incident can happen here” or “the technical design is inherently safe” can change behaviour. Unfortunately, a group is seldom aware of its shared understandings as they are seldom voiced or visible. Organizations with a strong safety culture are capable of questioning existing shared norms, values and basic assumptions like “are we safe?” and “what do we not pay attention to?” According to Schein, any process that challenges underlying assumptions will cause distress and anxiety among group members [11]. Therefore, in order to make changes and to ease anxiety, understanding and giving up of deep-rooted assumptions is necessary. A high level of trust and respect between individuals, both horizontally and vertically in an organization, facilitates a constructive process whereby underlying assumptions are challenged.

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Uncertainty estimation of thyroid activity measurements and its consequences in dose estimation

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Abstract. For the great majority of incorporated radionuclides the internal dose is estimated in two steps. In the first step the actual activity present in the body is determined by direct or indirect monitoring methods. By direct measurements activity in the whole or part of the human body can be determined (*in-vivo*), while by the indirect methods the radioactivity of physical and biological (*in-vitro*) samples are assessed. In the second step the intake value and the associated committed dose can be estimated on the basis of measured data considering necessary assumptions on exposure conditions (time and route of intake, chemical form etc.).
In the *in-vivo* measurement the low activity of the human body, the non-standard geometry and the limited measurement times considerably increase the measurement uncertainty in comparison to that of environmental or biological *in-vitro* samples. The adequate thyroid measurement results are influenced by various parameters e.g. the shape of thyroid, position of the thyroid inside the human body, the detector distance from the body surface, the distribution of the activity within the organ and the position of the detector during calibration with a thyroid phantom. All these factors affect the accuracy of dose estimation as well. The results of uncertainty assessments were included in dose estimation calculations with MONDAL-3 codes. The influence of the uncertainties of assumptions such as the date and route of intake, the physical and chemical form were also investigated in terms of the accuracy of the final internal dose assessment.

KEYWORDS: thyroid, iodine-131, dose estimation, uncertainty

1 INTRODUCTION

In the case of severe reactor accidents, complex combination and large quantities of radionuclides can be released into the air (Fig.1). It may be necessary for the population living in the vicinity of the reactor to use individual observations, as these observations enable more accurate estimation of the exposure of members of the public due to radioactive discharges into the environment. In these accidents, primarily the volatile iodine isotopes contribute to the initial internal exposure of the population [1, 2]. Possible pathways of intake of radioiodine in an accident situation is inhalation of the contaminated air and ingestion via food chain. Accumulation of radioiodine in thyroid depends on the availability of stable iodine and the metabolism of ^{131}I in the human body. Intake of stable iodine can be achieved with daily nutrition (e.g. iodized salt, water, sea food) and with iodine thyroid blocking (ITB) administration before or within few hours after the intake of radioiodine as well. Iodine thyroid blocking is an urgent protective action that is required by emergency response and safety regulations [3]. One of the operational intervention levels is the dose rate from the thyroid. For the whole body and individual organs like the thyroid, the relevant internal exposure cannot be directly measured.

Figure 1: Radioactive material emission to the atmosphere

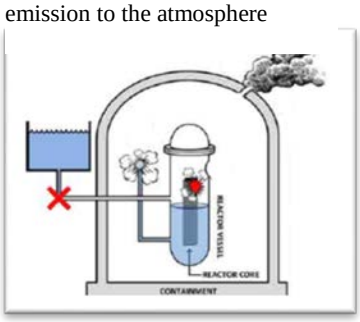
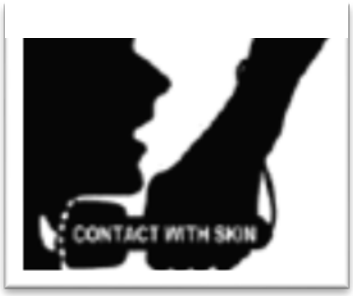


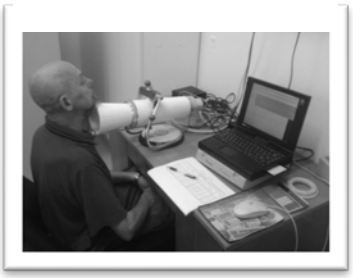
Figure 2: Dose rate in contact with the skin in front of the thyroid



2 MEASUREMENT METHOD AND EQUIPMENT

Thyroid activity was measured by lead shielded NaI(Tl) scintillation detector standing on a vertically adjustable stand for lead shielding, which allows the optimal installation of the equipment. The detector connects to the computer via USB, where the spectrum recording is performed by MULTIACT software. The person to be measured is situated in front of the detector and the detector's collimator is directed at the neck towards the position of the thyroid (Fig 3).

Figure 3: Measurement setting



3 CALIBRATION WITH DIFFERENT PHANTOMS

To perform the efficiency calibration, a plastic phantom was used, which imitates a human neck, within the radioactive material is placed in a cuvette. The available ANSI thyroid phantom [4] is made of plexi-glass cylinder with a diameter of 150 mm and a height of 146 mm. It has only one cavity (20 ml), modelling the thyroid of an adult. The efficiency calibration measurements were carried out in 3 different measuring positions (Table 1) using 3 calibration sources with different ^{131}I activities (Table 2).

Figure 4.: Measurement setting for the measurement geometry in III. position.

Table 1: Different measurement geometries

Measurement geometry No.	Distance [cm]	Collimator
I.	12.5	Yes
II.	4	Yes
III.	4.5	No

Vial No.	Activity [Bq]
1	10.3
2	29.4
3	48.5

Table 2: Different activities of the used ^{131}I sources

A cylinder phantom with a diameter of 130 mm and a height of 120 mm was also used in this study. The cylinder phantom has three pair of holes with different sizes to imitate the size of the thyroid of 5 and 10 years old children and adults. The activities to be measured were filled in two vials, each vial contained ^{131}I in liquid form (Table 3).

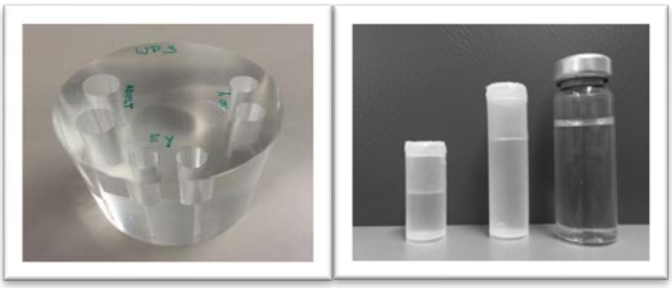


Figure 5: Cylinder phantom geometries and cuvettes

Table 3: Parameters of cylinder phantom

Vial Name	Height [mm]	Diameter [mm]	Capacity [ml]	Activity [Bq]
5 years	3.2	1.35	3.2	9.09
10 years	5	1.4	7.5	8.02
Adult	5.8	2	19	7.64

Based on the measurement results, about 20% difference was determined between the two different phantom efficiency values (see in Table 4).

Table 4: Efficiencies with different phantoms

Vial No.	ANSI	Cylinder	Vial Name
1.	2.38	1.98	Adult
2.	2.51		
3.	2.22		

In the case of the phantom which formation takes into account the different thyroid size of the certain ages, about 20% difference was also shown between the efficiencies. In Table 5. efficiencies for the different phantoms for certain age groups are summarized, which efficiency values were normalized to the efficiency for the adult phantom. These results indicate that thyroid measurements of children of different ages can be performed with higher efficiency than the adult thyroid measurements.

Table 5: Normalized efficiencies for the phantoms for different ages

Vial Name	Normalized efficiency (to adult)
5 years	1.20
10 years	1.16
Adult	1

4 DOSE ESTIMATION

For the dose estimation analyses MONDAL-3 software was used [5], in compliance with the pertinent ICRP recommendations [6]. Measurement of ^{131}I was chosen from the intake modes, according to the initial scenario, indicating the early period of severe reactor accidents. The people surveyed were classified in the examined population subgroups. For ^{131}I dose calculations, the AMAD (Activity Median Aerodynamic Diameter) characteristic of the inhaled aerosols for the different age groups could be adjusted in the program, as the size and function of the thyroid gland affects the dose from

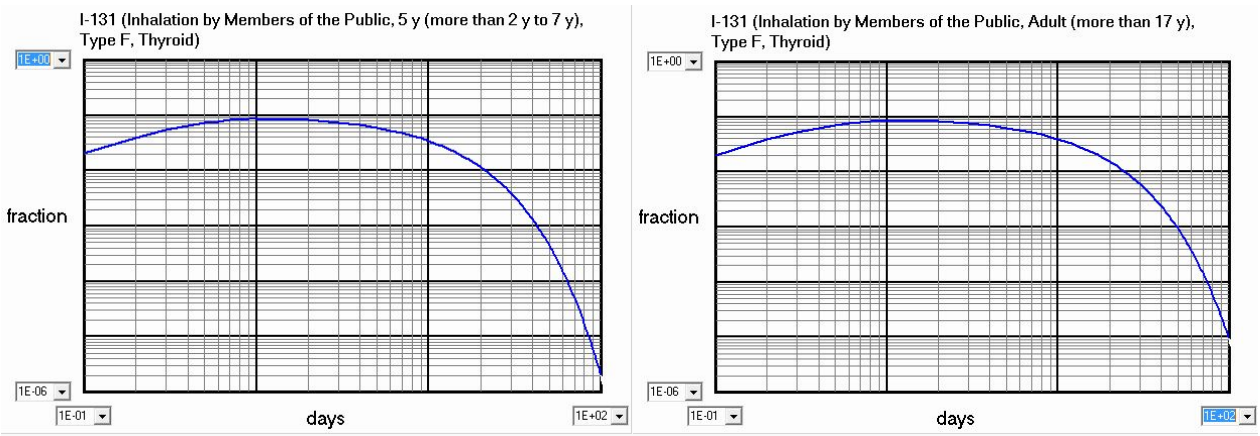
radioiodine. The calculations were performed on the first day following the record of the accidental discharge of radioactive contamination, as these values are expected to be measured shortly after recording [7]. We have described the deviation and applications of the age model of ^{131}I dosimetry for the thyroid.

In this study the measured activity was the fixed parameter (A_m=1000 Bq), with the variation of the other variables (intake route, elapse time between the intake and measurement, age) it was found that the uncertainty of the dose estimation could reach about 50% level.

Table 6: Dose estimation with different parameters

Intake route	Absorption type	Time between intake and measurement [days]	Age	Activity of intake [kBq]	Effective dose [µSv]
Inhalation	Type F	1	5 y	11	420
			10 y	11	210
			Adult	12	86
		7	5 y	21	770
			10 y	19	350
			Adult	19	140
Ingestion	f1=1.0	1	5 y	4	400
			10 y	4	210
			Adult	4	87
		7	5 y	7.30	730
			10 y	6.6	340
			Adult	6.5	140

Figure 6: The predicted inhalation fraction of the received dose from ^{131}I to the thyroid in 100 days following the accident. .



The results of the dose calculations from the intake activities show that the estimated effective doses for members of the public, thus the obtained doses resulting from the accident are less than 1 mSv. In the thyroid, the organ equivalent dose is 25 times higher than the effective dose. For radiolabel, it is recommended to calculate the equivalent dose for thyroid, since 50 mSv is the recommended baseline

criterion. Since radioiodine is largely concentrated in the thyroid, it causes a significant dose in the initial period following the accident due to the short half-life of radioiodine isotopes. In short-term exposures the children and infants are generally at more risk, because their thyroid gland is more sensitive to the carcinogenic effect of radiation. In addition, it is important to observe the high-hormone-level population groups, such as teenagers, pregnant women, nursing mothers, and women with variable sex habits because cell division is faster in their body due to increased hormone production, so the process of iodine incorporation will be faster.

5 CONCLUSION

The results of the analysis showed that the adequate thyroid measurement results are influenced by various parameters e.g. the shape of thyroid, position of the thyroid inside the human body, the distance between the detector and the body surface, as well as the distribution of the radioactivity within the organ. The experiences also demonstrated that the determination of the uncertainty of dose estimation is a complex process. The influence of the uncertainties of assumptions such as the date and route of intake, the physical and chemical form were also investigated in terms of the accuracy of the final internal dose assessment. The results of the uncertainty assessments were involved in dose estimation calculations with MONDAL-3 code. By varying the parameters, it was found that the uncertainty of the dose estimation is about 50% level. By studying the metabolism of the thyroid gland, the need for continuous and adequate iodine consumption has become clear for the safety of the population. These tests showed large absolute variability in the dose for children but relative measures of dispersion, such as the coefficient of variation differ little among age groups.

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Induced activation studies for the LHC upgrade to High Luminosity LHC

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Abstract. The Large Hadron Collider (LHC) will be upgraded in 2019/2020 to increase its luminosity (rate of collisions) by a factor of five beyond its design value and the integrated luminosity by a factor ten, in order to maintain scientific progress and exploit its full capacity. The novel machine configuration, called High Luminosity LHC (HL-LHC), will increase consequently the level of activation of its components. The evaluation of the radiological impact of the HL-LHC operation in the Long Straight Sections of the Insertion Region 1 (ATLAS) and Insertion Region 5 (CMS) is presented. Using the Monte Carlo code FLUKA, ambient dose equivalent rate estimations have been performed on the basis of two announced operating scenarios and using the latest available machine layout. The HL-LHC project requires new technical infrastructure with caverns and 300 m long tunnels along the Insertion Regions 1 and 5. The new underground service galleries will be accessible during the operation of the accelerator machine. The radiological risk assessment for the Civil Engineering work foreseen to start excavating the new galleries in the next LHC Long Shutdown and the radiological impact of the machine operation will be discussed.

1 INTRODUCTION

1.1 Introduction to CERN

CERN, the European Organization for Nuclear Research, is an intergovernmental organization with over 20 Member States. Its seat is in Geneva but its premises are located on both sides of the French-Swiss border (http://cern.ch/fplinks/map.html).
CERN's mission is to enable international collaboration in the field of high-energy particle physics research and to this end it designs, builds and operates particle accelerators and the associated experimental areas. At present more than 11 000 scientific users from research institutes all over the world are using CERN's installations for their experiments.
The accelerator complex at CERN is a succession of machines with increasingly higher energies. Each machine injects the beam into the next one, which takes over to bring the beam to an even higher energy, and so on. The flagship of this complex is the Large Hadron Collider (LHC).

1.2 Introduction to LHC and HL-LHC

The Large Hadron Collider (LHC) is the biggest accelerator constructed on the CERN site. The LHC machine accelerates and collides proton beams but also heavier ions up to lead. It is installed in a 27 km circumference tunnel, about 100 m underground.
High Luminosity LHC (HL-LHC) is a project aiming to upgrade the LHC collider during the Long Shutdowns (LS) 2 and 3, in order to maintain scientific progress and exploit its full capacity. By increasing its peak luminosity by a factor five over nominal value, it will be able to reach a higher level of integrated luminosity, nearly ten times the initial LHC design target.

2 RADIATION PROTECTION STUDIES USING FLUKA MONTE CARLO CODE

The radiological impact of the LHC proton-proton operation in the Long Straight Section (LSS) areas of the Insertion Region (IR) in Point 1 and 5 was evaluated in order to prepare the work for the modification of the LHC machine for its upgrade to HL-LHC. Using the Monte Carlo particle transport code FLUKA [1], [2], ambient dose equivalent rate estimations for the IRs have been performed on the basis of available operating scenario and using the latest available machine layout.

2.1 The p-p accelerator machine operating scenario

Proton-proton collisions in the LHC experiments in Point 1 (ATLAS) and in Point 5 (CMS) produce a secondary high energy radiation field that penetrates into the adjacent accelerator tunnels and causes severe activation of beam-line elements.
The simulations are performed considering the luminosity achieved up to 2017 and the target values announced by the LHC Operation team (LHC machine) and by the HL-LHC project (HL machine).

Table 1. Operating scenario of the LHC and HL-LHC machine.

	Year of (HL-)LHC Operation	Peak / levelled luminosity [$cm^{-2}s^{-1}$]		Integrated luminosity [fb^{-1}]	
	≤2012	0.8E+34		30	
LS1					
past	2015	6.30E+33		4	
	2016	1.40E+34		40	
	2017	1.60E+34		50	
	2018	1.60E+34		50	
LS2	(2 years)				
	2021	2.0E+34		60	
	2022	2.0E+34		60	
	2023	2.0E+34		60	~TOT 350 fb^{-1}
LS3	(3 years)	Nominal	Ultimate	Nominal	Ultimate
HL-LHC	2027	5.0E+34	7.5E+34	250	300
	2028	5.0E+34	7.5E+34	250	300
	2029	5.0E+34	7.5E+34	250	300
LS4	(1 year)				
	2031	5.0E+34	7.5E+34	250	300
	2032	5.0E+34	7.5E+34	250	300
	2033	5.0E+34	7.5E+34	250	300
LS5	(1 year)				
	2035	5.0E+34	7.5E+34	250	300
	2036	5.0E+34	7.5E+34	250	300
	2037	5.0E+34	7.5E+34	250	300
LS6				TOT ~2600 fb^{-1}	TOT ~3050 fb^{-1}

The operating scenarios as well as the integrated luminosity targets are summarized in Table 1.

2.2 Simulation geometry

LHC and HL-LHC geometry of the Insertion Region in Point 1 and in Point 5 has been reproduced in FLUKA [3], [4], all the machine elements as well as the tunnel are implemented in the simulation geometry.
The machine element layout and optics in the two points are the same, and from a radiological point of view the two Insertion Regions are equivalent.

2.3 LHC machine simulation

Ambient dose equivalent rate maps have been calculated from around 18 m distance from the Interaction Point (IP) around the two main LHC experiments in Point 1 and 5 (ATLAS and CMS respectively) up to around 260 m distance, and for different cooling times in the LS2 and LS3 (as in Table 1). The simulations were done for 7+7 TeV pp-collisions and using DPMJET-III [5] as the event generator; for the inelastic proton-proton cross section the value of 85 mb is used, on the basis of the extrapolation from [6].
In order to benchmark the simulation results, during LHC machine operation, ambient dose equivalent rate maps have been calculated on the basis of the actual machine operating scenario and for typical cooling times during the Technical Stop of the machine.
For example, Figure 1 shows the comparison between the simulation results and the measurement taken in the LHC tunnel during the first Technical Stop in 2016. In the upper part of the plots the sequence of the machine elements is depicted. The simulation results are in a very good agreement with the measurements.
Figure 2 shows the ambient dose equivalent rate map after one month of cooling time during LS3 in IR1.

Figure 1. Calculated ambient dose equivalent rate profile (yellow) and residual dose measurements taken during LHC-TS1 (in blue) along the LHC machine elements in IR5 (left) and IR1 (right).

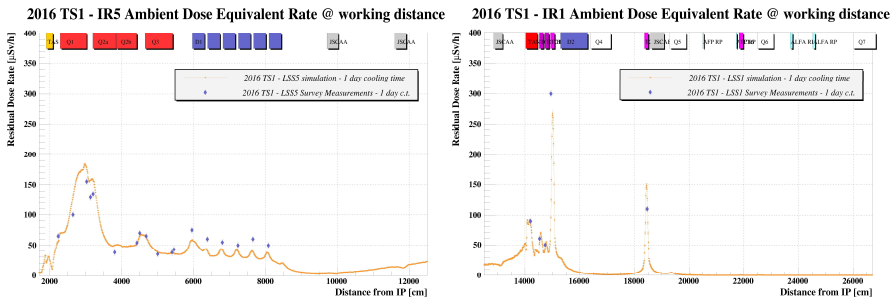


Figure 2. Ambient dose equivalent rate map in the Insertion Region 1 after one month cooling time during LHC machine Long Shutdown 3. The residual dose in the vertical dimension is averaged on 30 cm around the beam pipe.

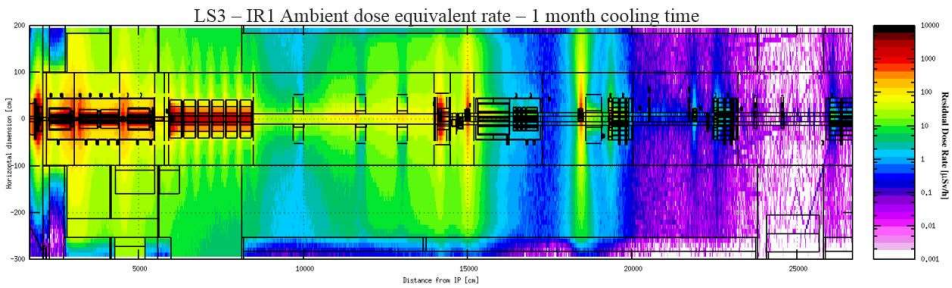


Figure 3. Ambient dose equivalent rate profile in the Insertion Region 1 after 1week, 4 weeks, 4 months and 1 year cooling time during LHC machine Long Shutdown 3. The residual dose in the vertical dimension is averaged on 30 cm.

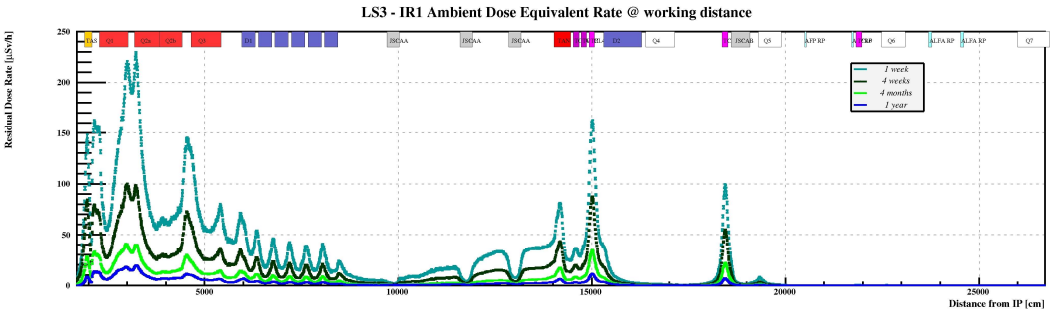


Figure 3 shows the residual dose rate profiles taken at a working distance (about 40 cm) from the machine elements in the tunnel aisle, at different cooling times (1 week, 4 weeks, 4 months and 1 year) during LS3. The higher residual dose rates are around the elements closest to the IP (element on the left in plots), with the highest levels in the aisle corresponding to the connections between elements, where the shielding effect due to self-absorption of the element is less effective, and around the collimators (element in purple in the upper part of the profile plots, where the sequence of the machine elements is depicted).

2.4 HL-LHC machine simulation

For the HL-LHC proton operation, the ultimate scenario is considered (7+7 TeV pp-collisions at the average luminosity value of 7.5×10^{34} $cm^{-2}s^{-1}$ with a 295 μrad half-angle crossing in the interaction point using DPMJET-III [5] as the event generator; for the inelastic proton-proton cross section the value of 85 mb is used, on the basis of the extrapolation from [6]).
Figure 5 shows the ambient dose equivalent rate maps after one month of cooling time during the first Long Shutdown during the HL-LHC era, so called LS4, in IR1 (upper plot) and in IR5 (bottom plot). The small difference in between the two ambient dose equivalent rate maps is due to the different crossing angle plane, which is vertical in IP1 and horizontal in IP5.

Figure 5. Ambient dose equivalent rate maps in the Insertion Region 1 after one month cooling time during HL-LHC machine Long Shutdown 4. The residual dose in the vertical dimension is averaged on 30 cm around the beam pipe.

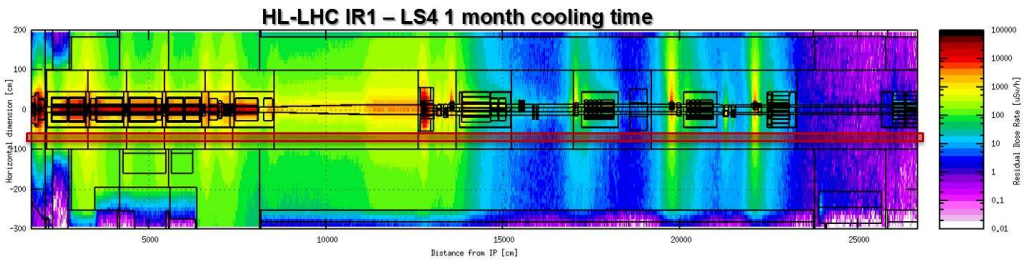
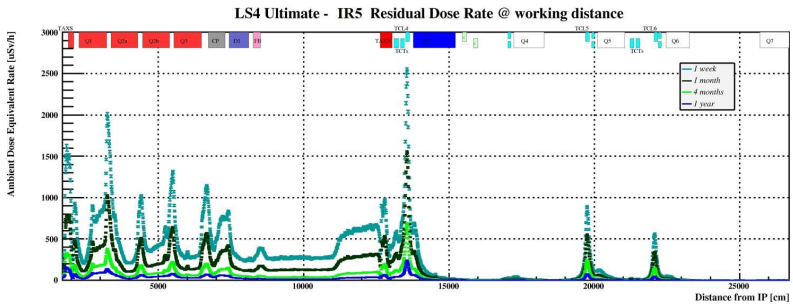


Figure 6 shows the ambient dose equivalent rate profile in the aisle at a working distance from the machine elements at four different cooling times (1 week, 1 month, 4 months and 1 year), which are typical cooling times for maintenance intervention in the machine tunnel during the scheduled technical stops, end of the year technical stops and long shutdowns. As for the LHC machine, the higher residual dose rates are around the elements closest to the IP (element on the left in plots) where the highest levels in the aisle correspond to the connections between elements, where the shielding effect due to self-absorption of the element is less effective, and around the collimators (element in purple in the upper part of the profile plots, where the sequence of the machine elements is depicted).

Figure 6. Ambient dose equivalent rate profile in the Insertion Region 5 after 1 week, 4 weeks, 4 months and 1 year cooling time during HL-LHC machine Long Shutdown 4. The residual dose in the horizontal and vertical dimensions are averaged on 30 cm.



As reported in Table 1, for the time being, three long shutdown are schedule for the HL-LHC. The operating scenario is foreseen as three blocks of three operational years with a long shutdown in between. For the operational years, constant operating parameters are assumed.

Figure 7 compares the ambient dose equivalent rate after one month of cooling time during the first HL-LHC long shutdown (LS4) and the last (LS6). The ratios for all the considered cooling times are reported in Table 2; the ratio is almost 1 for the shortest cooling time and it increase with the longer cooling times due to the accumulation in the activated equipment of the induced radionuclides with longer half-lives.
Figure 8 compares the ambient dose equivalent rate after one month of cooling time during the first HL-LHC long shutdown (LS4) between the nominal and the ultimate case (as reported in Table 1). The ratios for all the considered cooling times are reported in Table 3; ratio is higher for the shortest cooling time due to the higher production in the activated equipment of the induced radionuclides with shorter half-lives, and it decreases with the longer cooling times.

Figure 7. Ambient dose equivalent rate profiles in the Insertion Region 1 after one month cooling time during HL-LHC machine LS4 and LS6. The residual dose in the horizontal and vertical dimensions are averaged on 30 cm.

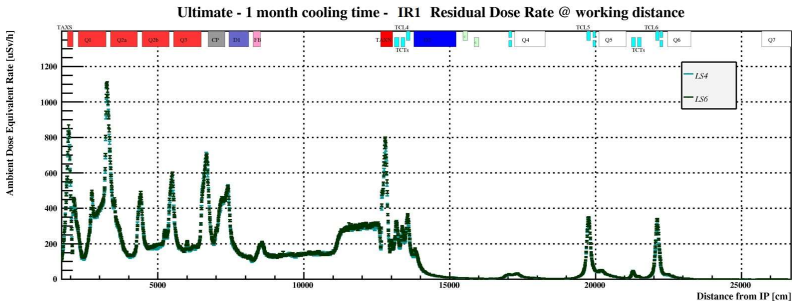
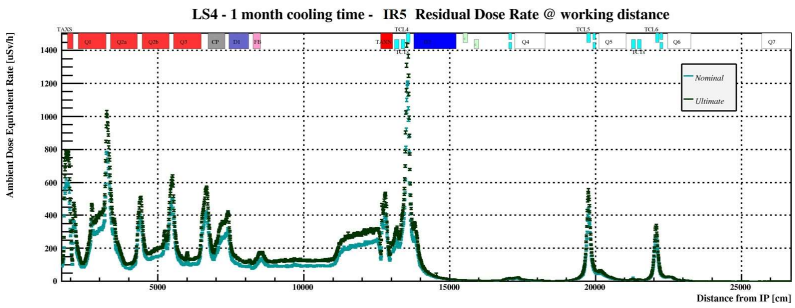


Table 2 and 3. (On the left) Comparison residual dose rate ratios between LS4 and LS6 results for different cooling times. (On the right) Comparison residual dose rate ratios between Nominal and Ultimate results for different cooling times.

Cooling time	LS6/LS4
1 hour	1.02
1 day	1.03
1 week	1.06
1 month	1.09
4 months	1.18
1 year	1.29

Cooling time	Ultimate/ Nominal
1 hour	1.50
1 day	1.46
1 week	1.35
1 month	1.30
4 months	1.24
1 year	1.23

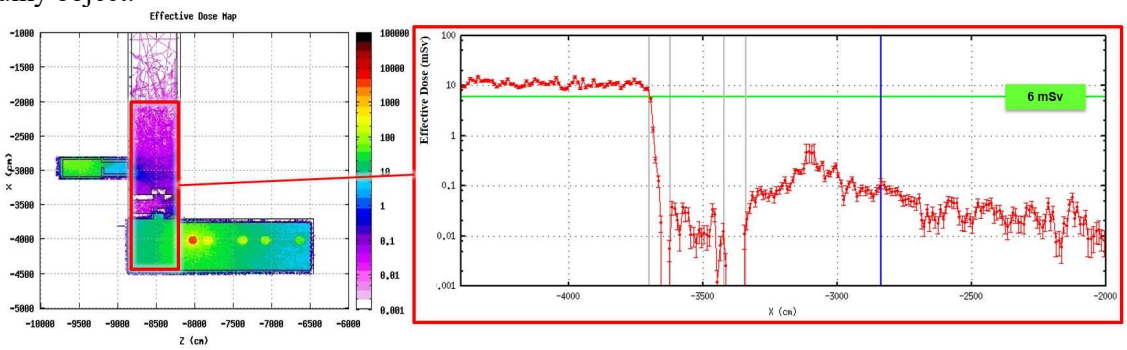
Figure 8. Ambient dose equivalent rate profiles in the Insertion Region 5 after one month cooling time during HL-LHC machine LS4 for the Nominal and Ultimate operating scenarios. The residual dose in the horizontal and vertical dimensions are averaged on 30 cm.



2.5 HL-LHC new underground galleries

In order to accommodate the new technical infrastructure needed for the upgrade of the LHC machine to HL-LHC, new caverns and 300 m long galleries along the Insertion Region of IP1 (ATLAS) and IP5 (CMS) are required.
The new service galleries will be accessible during the operation of the accelerator machine. For this reason, the radiological impact of the machine operation has to be assessed.
Two different scenarios are kept into account in the study: an accidental scenario, represented by one full proton beam lost into a bulky object in front of the most exposing connection to the galleries, and the stray radiation from the normal operation of the machine. For both cases the ultimate beam intensity (2808 bunches and 2.2×10^{11} ppb) and luminosity (7.5×10^{34} $cm^{-2}s^{-1}$) are considered. The requirements are set by the two considered scenarios. In the accidental case the effective dose delivered to personnel has not to exceed the legal annual limit for the class B radiation worker, i.e. 6 mSv. During normal operation of the accelerator machine, in the new service galleries the ambient equivalent dose has to be the lowest possible, and in any case, it has to be lower than 15 µSv/h, in order to classify the areas at the lowest Radiation Area level, i.e. Supervised Radiation Areas.
Figure 11 shows on the left the effective dose map and on the right effective dose profile in the area indicated by the red square, due to the full loss of one proton beam on a bulky object in front a connection to the new galleries. The implemented shielding provide an attenuation of a factor of about 10^{-8} from the LHC tunnel to the closest accessible area in the new galleries (indicated by the blue line in profile plot, on the right of Figure), where the effective dose level drops to a delivered dose lower than 0.1 mSv.

Figure 11. Effective dose delivered in the new underground galleries due to a full proton beam lost on a bulky object.



During machine operation, the maximum dose rate next to the tunnel wall is 50 Sv/h, which applying the attenuation factor of 10^{-8}, lead to a value of 0.5 µSv/h ambient dose equivalent rate due to stray radiation in the new underground gallery closest accessible point.

3 ACTIVATION STUDIES USING ActiWiz3 CREATOR CODE

In order to assess the level of activation of the spoil when the new underground galleries will be excavated (LS2), the results obtained with the FLUKA Monte Carlo code (particle fluence spectra) were used as input for the ActiWiz3 Creator © code [7].
An activated material is defined as radioactive if the specific or total activity of any radionuclides of artificial origin exceed the corresponding clearance limit, i.e.:

$$\sum_{i=1}^{n} \frac{a_i}{LL_i} < 1$$

where a_i is the specific activity (Bq/g) or the total activity (Bq) of the ith radionuclide of artificial origin in the material, LL_i is the respective CERN clearance limit for the radionuclide i in the material and n is the number of radionuclides present.
During last end of the year technical stop of the LHC machine, some concrete and rock samples were taken out from the LHC tunnel wall, in the place where the excavation will occur. γ-spectrometry measurements were performed on the rock and concrete samples, after a cooling time of about 3 months. The activation of the samples was also evaluated using ActiWiz3 Creator code. The results are reported in Table 4, in the form of the sum of specific activity over clearance limit.

Table 4. Activation level of concrete and soil samples as measured by the γ-spectrometry and calculated using the ActiWiz3 Creator© code.

$\sum_i \frac{a_i}{LE_i}$	***3 months cooling time***		
UPR	γ spectrometry measurements	ActiWiz3 Creator©	
		only γ	*full*
concrete	1.15E-01	9.70E-02	*4.96E-01*
soil	8.17E-03	5.30E-03	*1.65E-02*

In the last column of Table 4, the results of all the radionuclides is reported, including the ones that cannot be measured with the γ-spectrometry technique. The γ results, second and third column in Table 4, are in good agreement. The induced radionuclide list calculated is reported in Table 5 (concrete sample) and in Table 6 (soil sample) with the contributing percentages to the sum; the most contributing radionuclide in both cases is Ca-45, which is a pure β-emitter, thus not measurable with the γ-spectrometry technique.

Table 5

Concrete sample	
radionuclide	Contribution to $\Sigma_i \frac{a_i}{LL_i}$
Ca-45	80%
Na-22	7%
Zn-65	3%
Fe-55	3%
S-35	1%
Mn-54	1%

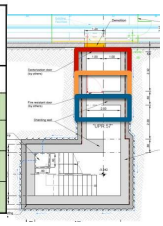
Table 6

Soil sample	
radionuclide	Contribution to $\Sigma_i \frac{a_i}{LL_i}$
Ca-45	68%
Na-22	28%
S-35	2%

The activity over limit sum was evaluated for the spoils coming from the excavation of the new underground galleries where they are connected to the existing LHC tunnel. The results of the calculation, for the first three meters of excavation spoils is reported in Table 7, for one and four months cooling times; the first meter of excavated spoils will be radioactive, the second and the third meters are below the legal limits in both the cases.

Table 7. Activation level of excavated spoils.

$\sum_i \frac{a_i}{LL_i}$	***LS2***	
UPR	ActiWiz 3 Creator©	
	1 month	4 months
1st meter	5.19E+00	4.15E+00
2nd meter	3.19E-01	2.90E-01
3rd meter	1.87E-02	1.70E-02



4 SUMMARY

FLUKA Monte Carlo code is extensively used to perform induced activation studies for the LHC accelerator machine and its High Luminosity upgrade. The simulations are performed using a very long and detailed geometry which reproduce the accelerator elements and the tunnel environment in great detail. The simulation results are continuously benchmarked with the radiation measurements performed in the LHC machine.
Actiwiz3 Creator© were used in order to predict the level of activation of excavation spoils and the radionuclide inventory in view of the excavation work in order to build the new underground galleries for the HL-LHC additional infrastructure and services.

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Physics, Chemistry & Biology

Biota dose rate calculations in PC-CREAM 08®

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Abstract PC-CREAM 08® was developed by Public Health England as an update to the original PC-CREAM, which was developed under contract to the European Commission and implements the European Union methodology for routine releases. The tool is a suite of models and data intended to be used to assess the radiological impact of continuous and constant releases of radioactive effluents to atmosphere, rivers and marine waters arising as a result of normal operations. Until now, PC-CREAM only assessed radiological impact to humans; work is currently under way to implement the calculation of dose rates to the Reference Animals and Plants defined by the International Commission on Radiological Protection. The user will identify the area of interest (the target area) and PC-CREAM will calculate mean activity concentrations in the media in which the biota live, e.g. air, on the soil surface, underground. Activity concentration ratios, taken from *ICRP Publication 114*, will be used to estimate the organisms' internal activity concentration, and dose conversion factors, taken from *ICRP Publication 108*, will be used to calculate the internal and external dose rates to the organisms. Dose rates to a selection of biota were calculated using the proposed methodology and the results were compared with output from the ERICA tool for assessing the radiological risks to biota. There was strong agreement between external dose rates calculated by the two systems. Internal dose rates agreed less well, but all differences can be explained by a difference in methodology and a newer source of data for dose conversion factors in the latest version of the ERICA tool.

KEYWORDS: PC-CREAM; environmental transfer; wildlife; biota

1 INTRODUCTION

In *Publication 103* the International Commission on Radiological Protection (ICRP) introduced the requirement of protecting the environment, specifically "preventing or reducing the frequency of deleterious radiation to a level where they would have a negligible impact on the maintenance of biological diversity, the conservation of species or the health and status of natural habitats, communities and ecosystems." [1]. Since then, ICRP have defined a set of Reference Animals and Plants (RAPs), intended to be hypothetical entities with biological similarities to particular types of animal or plant at a taxonomic level of family and typical of different environments [2]. ICRP also specified dose conversion factors for each RAP to translate internal activity concentration and activity concentration in external media into internal and external dose rates respectively [2]. Finally, ICRP recommended using concentration ratios as a means of estimating activity concentration in the organism from activity concentration in the surrounding media [3].

Public Health England (PHE) are in the process of incorporating the ICRP methodology into PC-CREAM 08® [4], a suite of models and data for assessing the radiological impact of continuous and constant discharges of radioactivity to the environment. The aim is to provide the user with a method of estimating doses to non-human biota that is consistent with that for assessing doses to humans. To this end, the new feature will use the existing dispersion modelling in PC-CREAM, with an enhancement to calculate an average activity concentration in and on soil in the area of interest as a result of discharges to atmosphere, average activity concentrations in a 10 km length of river from discharges to freshwater and average activity concentrations in seawater and seabed sediment from discharges to the marine environment. From these media activity concentrations, PC-CREAM will then estimate dose rates by applying the ICRP recommended concentration ratios and dose conversion factors. In this paper we apply the PC-CREAM dispersion and averaging modelling to discharges from a hypothetical nuclear power station to derive activity concentrations in media at the site of interest. We then compare biota dose rates arising from those media activity concentrations calculated by the proposed PC-CREAM methodology and the ERICA tool [5] which was developed to implement the

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ERICA Integrated Approach [6], a system of assessing the radiological risk to biota. Since its release in 2007, the ERICA tool has been widely used in varied situations (for example, assessing the impact of deep geological and near-surface disposal facilities, uranium mining impact assessments, assessing releases from medical facilities) [7]. The ERICA tool has also been used in model intercomparison [8, 9] and therefore provides a benchmark against which to compare the PC-CREAM proposed methodology.

2 METHODOLOGY

2.1.1 Calculating activity concentration in environmental media

A hypothetical nuclear power station with a pressurised water reactor (PWR) discharges radioactive effluent to atmosphere and to the river on whose banks it sits. The surrounding area is typical of a temperate environment and includes species similar to the RAPs. The discharged radionuclides and the quantity discharged are characteristic of a PWR. Average activity concentrations in air, on soil and in soil at a nearby site of interest from discharges to atmosphere were derived for use in this comparison exercise; similarly, activity concentrations in river water, sea water and seabed sediment from liquid discharges were derived. Table 1 shows the activity concentrations in the various media.

Table 1: Activity concentrations in environmental media

Radionuclide	In air (Bq m^{-3})	On soil (Bq kg^{-1})	In soil (Bq kg^{-1})	In river water (Bq L^{-1})	In sea water (Bq L^{-1})	In seabed sediment (Bq kg^{-1})
^{3}H	1.30 10^{-3}	—	—	7.01 10^{0}	7.01 10^{0}	1.87 10^{1}
^{14}C	3.73 10^{-4}	—	—	5.86 10^{-3}	5.86 10^{-3}	1.56 10^{1}
^{54}Mn	—	—	—	9.49 10^{-7}	9.49 10^{-7}	5.06 10^{0}
^{58}Co	—	2.99 10^{-8}	1.50 10^{-8}	6.41 10^{-6}	6.41 10^{-6}	5.12 10^{0}
^{60}Co	—	4.86 10^{-7}	2.54 10^{-7}	8.09 10^{-6}	8.09 10^{-6}	6.47 10^{0}
^{63}Ni	—	—	—	2.41 10^{-6}	2.41 10^{-6}	1.28 10^{-1}
^{110m}Ag	—	—	—	2.04 10^{-5}	2.04 10^{-5}	5.43 10^{-1}
^{124}Sb	—	—	—	1.29 10^{-6}	1.29 10^{-6}	6.87 10^{-3}
^{125}Sb	—	—	—	3.60 10^{-6}	3.60 10^{-6}	1.92 10^{-2}
^{131}I	—	4.91 10^{-7}	2.46 10^{-7}	1.42 10^{-6}	1.42 10^{-6}	2.65 10^{-4}
^{133}I	—	3.18 10^{-8}	1.59 10^{-8}	—	—	—
^{134}Cs	—	1.71 10^{-7}	8.63 10^{-8}	1.21 10^{-6}	1.21 10^{-6}	1.29 10^{-2}
^{135}Cs	—	5.83 10^{-12}	3.85 10^{-12}	—	—	—
^{137}Cs	—	1.62 10^{-6}	9.92 10^{-7}	1.45 10^{-6}	1.45 10^{-6}	1.55 10^{-2}

2.1.2 Selecting biota

Of the reference biota, several spend time in multiple environments, for example reference Rat spends some time above ground and some time underground, whereas reference Duck spends time on soil and on water. For simplicity, this study selected RAPs from each habitat (terrestrial, freshwater, marine) that spend 100% of their time in a single environment: reference Pine Tree (100 % on soil); reference Trout (100% in fresh water); and reference Flatfish (100% at the sea water-seabed sediment interface). The ERICA tool refers to these organisms as Tree, Pelagic Fish and Benthic Fish respectively.

2.1.3 Calculating and comparing biota dose rates

Assessments in the ERICA tool are organised into three tiers. Tier 1 is a simple, conservative approach that compares media activity concentrations to pre-calculated limits for the most limiting organism for each radionuclide; it does not provide dose rates and requires no specialist knowledge to set up. Tier 2 assessments allow the user to select from a set of pre-defined biota and to tailor the exposure conditions and transfer parameters for a more site-specific assessment. Tier 3 assessments are more

realistic in nature, are complex to set up and require considerable specialist knowledge [5]. In ERICA (version 1.2.1) [7], a Tier 2 assessment was created for each selected RAP. Media activity concentrations were entered as in Table 1 and ERICA default values used for distribution coefficients, concentration ratios and dose conversion coefficients (called dose conversion factors in ICRP recommendations and the PC-CREAM methodology).

A prototype of PC-CREAM was developed to implement the proposed methodology that uses the ICRP recommended values for concentration ratios and dose conversion factors [2, 3]. The prototype was used to calculate dose rates for each of the selected RAPs.

3 RESULTS

Table 2 lists the external dose rates calculated by PC-CREAM and ERICA for each of the selected RAPs from identical activity concentrations in the relevant media and Table 3 below lists the internal dose rates.

Table 2: External dose rates (µGy h^{-1})

	Terrestrial: pine tree		Freshwater: trout		Marine: flatfish	
Radionuclide	ERICA	PC-CREAM	ERICA	PC-CREAM	ERICA	PC-CREAM
^{3}H	—	—	2.52 10^{-12}	2.48 10^{-12}	1.02 10^{-11}	1.02 10^{-11}
^{14}C	—	—	1.05 10^{-10}	1.05 10^{-10}	1.41 10^{-7}	1.40 10^{-7}
^{54}Mn	—	—	3.99 10^{-10}	3.95 10^{-10}	1.11 10^{-3}	1.16 10^{-3}
^{58}Co	4.78 10^{-12}	4.61 10^{-12}	3.14 10^{-9}	3.21 10^{-9}	1.31 10^{-3}	1.28 10^{-3}
^{60}Co	1.90 10^{-10}	1.88 10^{-10}	1.05 10^{-8}	1.04 10^{-8}	4.20 10^{-3}	4.31 10^{-3}
^{63}Ni	—	—	2.65 10^{-15}	2.71 10^{-15}	7.06 10^{-11}	6.93 10^{-11}
^{110m}Ag	—	—	2.86 10^{-8}	2.89 10^{-8}	4.08 10^{-4}	3.96 10^{-4}
^{124}Sb	—	—	1.21 10^{-9}	1.18 10^{-9}	3.37 10^{-6}	3.44 10^{-6}
^{125}Sb	—	—	7.56 10^{-10}	7.65 10^{-10}	2.11 10^{-6}	2.16 10^{-6}
^{131}I	3.00 10^{-11}	3.07 10^{-11}	2.70 10^{-10}	2.72 10^{-10}	2.66 10^{-8}	2.66 10^{-8}
^{133}I	3.12 10^{-12}	3.05 10^{-12}	—	—	—	—
^{134}Cs	4.28 10^{-11}	4.28 10^{-11}	9.56 10^{-10}	9.58 10^{-10}	5.29 10^{-6}	5.38 10^{-6}
^{137}Cs	1.46 10^{-10}	1.49 10^{-10}	4.21 10^{-10}	4.11 10^{-10}	2.32 10^{-6}	2.33 10^{-6}

Table 3: Internal dose rates (µGy h^{-1})

	Terrestrial: pine tree		Freshwater: trout		Marine: flatfish	
Radionuclide	ERICA	PC-CREAM	ERICA	PC-CREAM	ERICA	PC-CREAM
^{3}H	1.61 10^{-6}	6.42 10^{-7}	5.78 10^{-5}	2.31 10^{-5}	5.78 10^{-5}	2.31 10^{-5}
^{14}C	1.43 10^{-5}	1.37 10^{-5}	3.12 10^{-2}	7.14 10^{-3}	2.95 10^{-4}	1.99 10^{-3}
^{54}Mn	—	—	1.81 10^{-7}	1.78 10^{-7}	1.19 10^{-7}	9.89 10^{-9}
^{58}Co	5.30 10^{-14}	1.34 10^{-14}	1.39 10^{-7}	5.58 10^{-8}	2.48 10^{-6}	1.41 10^{-7}
^{60}Co	1.93 10^{-12}	5.10 10^{-13}	3.93 10^{-7}	1.63 10^{-7}	7.29 10^{-6}	4.45 10^{-7}
^{63}Ni	—	—	5.92 10^{-9}	3.13 10^{-10}	7.40 10^{-9}	6.51 10^{-9}
^{110m}Ag	—	—	2.01 10^{-6}	4.85 10^{-7}	4.04 10^{-5}	2.96 10^{-5}
^{124}Sb	—	—	1.46 10^{-8}	4.21 10^{-9}	2.24 10^{-7}	2.26 10^{-7}
^{125}Sb	—	—	1.22 10^{-8}	3.23 10^{-9}	1.94 10^{-7}	1.80 10^{-7}
^{131}I	1.72 10^{-11}	6.40 10^{-12}	6.33 10^{-8}	1.21 10^{-8}	1.66 10^{-9}	1.65 10^{-9}
^{133}I	1.91 10^{-12}	7.02 10^{-13}	—	—	—	—
^{134}Cs	1.34 10^{-11}	7.48 10^{-12}	8.57 10^{-7}	6.67 10^{-7}	1.73 10^{-8}	7.44 10^{-9}
^{135}Cs	3.14 10^{-17}	1.69 10^{-17}	—	—	—	—
^{137}Cs	7.03 10^{-11}	3.95 10^{-11}	8.80 10^{-7}	7.18 10^{-7}	2.07 10^{-8}	8.92 10^{-9}

4 DISCUSSION

4.1.1 External dose rates

The external dose rates calculated by the proposed PC-CREAM methodology agree well with those calculated by the ERICA tool for the same media activity concentrations. This result was anticipated since the ERICA dose conversion coefficients closely match the ICRP recommended dose conversion factors, varying by a few per cent at the most.

4.1.2 Internal dose rates

The internal dose rates calculated by PC-CREAM agree less well with those calculated by the ERICA tool than the external dose rates. In general the PC-CREAM dose rates are lower than those calculated by ERICA but are within an order of magnitude; exceptionally, the difference is greater than an order of magnitude (e.g. trout, ^{63}Ni; flatfish, ^{54}Mn). This can be explained by differences in the concentration ratios used by the two systems, and by the incorporation of radiation weighting factors into the calculation of dose conversion coefficients in ERICA.

The values for internal dose conversion factors recommended by ICRP [2] came from the Wildlife Transfer Database (WTD) [10] which was initially populated using the database from the original release in 2007 of the ERICA tool, one of the most comprehensive resources of wildlife transfer values at the time [11, 12]. By the time the ERICA tool database was updated in 2013, the WTD had amassed an additional 17,000 concentration ratio values [13]. After quality checking, revised summary values were generated from the WTD to fill in gaps where no empirical data existed before and to improve on the quality of existing data; these revised values were incorporated into the 2013 update (version 1.2) of the ERICA tool.

ICRP's current recommended values for dose conversion factors [2] do not take account of the relative biological effectiveness of different forms of ionising radiation [14] and simply sum the contributions to obtain an overall figure for each radionuclide. By contrast, the ERICA tool applies weighting factors of 10 to alpha radiation, 3 to low energy beta radiation and 1 to (high energy) beta and gamma radiation [5] as a default; if these weighting factors are all set to 1, the value of the ERICA dose conversion factors are the same as, or very close to, those recommended by ICRP.

5 CONCLUSION

This study compared the proposed PC-CREAM methodology for calculating dose rates to non-human biota with that implemented in the ERICA tool. Activity concentrations in environmental media were inferred from discharges from a hypothetical PWR, and those activity concentrations were used in both the PC-CREAM and the ERICA methodology to calculate dose rates to three of ICRP's Reference Animals and Plants: pine tree, trout and flatfish. External dose rates calculated by the two methodologies agreed well. Internal dose rates were generally within an order of magnitude, and differences can be explained by (i) the application of radiation weighting factors in the ERICA tool, and (ii) concentration ratios in the current version of the ERICA tools were generated from the Wildlife Transfer Database more recently than the ICRP recommended values.

PHE intend to continue this work, with the goal of producing a future version of PC-CREAM that will allow the user to estimate doses to non-human biota. It will use the same dispersion modelling for humans and non-human biota, providing consistency in the calculation of activity concentrations in the environment, and will incorporate the ICRP methodology for estimating doses to non-human biota from activity concentrations in the surrounding media.

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Optimization of dosimetry in alphatherapy by a multi-scale approach: application to the treatment of bone metastases with ^{223}Ra.

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Abstract. In nuclear medicine, beyond providing a starting administered activity for clinical studies, dosimetry has an important role in guiding clinical trial design to help maximize the likelihood of a successful, minimally toxic implementation. For alpha-emitter radiopharmaceuticals, a personalized dosimetry is challenging because of the short range of alpha-particles. The aim of this work was to develop tools to optimize patient-specific dosimetry in alphatherapy by focusing on ^{223}Ra (Xofigo®) which is already used in clinical routine for the treatment of patients with castration-resistant prostate cancer metastasized to bone. In the MIRD pamphlet 22, several recommendations have been made in order to carry out a dosimetric study in alphatherapy. First, the repartition of the radiopharmaceutical in the patient body has to be determined. Therefore, an imaging protocol has been developed and optimized. The quantification accuracy and its limitations have been studied thanks to many experiments conducted on a gamma camera. Then, as bone marrow is the organ at risk, absorbed dose to this region has to be calculated. So, the most realistic dosimetric model was used and the dose was compared to the one from the previous dosimetric model. Finally, because of the short range of alpha-particles, a heterogeneous distribution of the radiopharmaceutical can lead to a highly non-uniform irradiation of the target volume. The microdistribution of ^{223}Ra has then to be investigated. So, animal models were developed. The results help gain more insight into the various aspects of the ^{223}Ra dosimetry from a macroscopic scale to a microscopic scale. These studies will allow later to develop simpler tools for more practical use in the clinic. This work is crucial to optimize treatments and to better understand the therapeutic response and dose limits for organs at risk. It also offers methods to go further in personalizing the dosimetry of new alpha-emitter radiopharmaceuticals.

KEYWORDS: Dosimetry; Nuclear Medecine; Monte Carlo; Bone Marrow Dosimetry; Quantitative imaging; Microdistribution.

1 INTRODUCTION

Targeted therapy is based on the administration of a radiopharmaceutical (radionuclide often tied to a vector), which will bind specifically in tumor regions in order to destroy them. Until recently, routine treatments in therapeutic nuclear medicine were performed with β^- emitters. But, the development of new radiopharmaceuticals, especially alpha-emitters, in particular due to progress on more and more specific vectors, makes the discipline particularly promising. Indeed, advances in the targeted delivery of radionuclides, in radionuclide conjugation chemistry, and in the increased availability of alpha-

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emitters appropriate for clinical use have recently led to patient trials of alpha-particle-emitter labeled radiopharmaceuticals. Although alpha-emitters have been studied for many decades, their current use in humans for targeted therapy is an important milestone. The physical and radiobiological characteristics of these particles confer a greater cytotoxicity to tumor cells while minimizing the unwanted radiation to healthy tissues in comparison with β^- emitters. This makes alphatherapy very interesting for the treatment of disseminated cells or metastases, tumor cell clusters or radiation-resistant cells like potentially tumor stem cells whereas current approaches are largely ineffective once the tumor has metastasized and tumor cells are disseminated throughout the body.

More and more studies show the very promising results of alphatherapy. Xofigo® (^{223}Ra) is the first alpha-emitter that has received marketing authorization (MA) from the European Commission in November 2013 for the treatment of patients with castration-resistant prostate cancer metastasized to bones. According to a Thomson Reuters Pharma Matters report [1], Xofigo® is expected to become the first blockbuster radiotherapeutic. In England, the total number of patients with bone metastases treated with radiopharmaceuticals increased by nearly 400% from 2007 to 2015 due to ^{223}Ra [2]. In addition, this MA has boosted other clinical and preclinical trials. For example, in 2016 and 2017, new publications on ^{225}Ac-PSMA and ^{213}Bi-PSMA (prostate-specific membrane antigen, a target in prostate cancer) showed outstanding results on the first patients. The first clinical trials have shown complete responses on patients in therapeutic failure [3,4]. These new radiopharmaceuticals should arrive quickly on the market. Many preclinical trials also show promising results, including the use of nanoparticles that can potentially contain several alpha-emitters and prevent the migration of daughters to healthy tissue [5]. Thanks to these new developments, a huge field of applications now opens.

Nevertheless, despite the preferential targeting of the targeted pathological lesions, a fraction of the non-targeted tissue can be irradiated. The evaluation of the absorbed doses (energy absorbed per unit of mass in each organ) delivered to these tissues is then crucial in order to ensure that the ratio between the risks related to the irradiation of the non-targeted tissues and the benefits related to the treatment remains in favor of the latter. To do that, this evaluation should ideally be done in the most personalized way possible. In addition, it must optimize the radiopharmaceutical administration protocol in order to maximize the effectiveness of the treatment while limiting the irradiation of organs at risk. For alpha-emitter radiopharmaceuticals, the dosimetric evaluation is therefore a real challenge given the short path of these particles.

Many challenges are to be met before a clinical application of this dosimetry. Indeed, this increased importance of dosimetric analysis is coupled with a greater difficulty in obtaining the human data necessary to perform this evaluation. As highlighted in the MIRD pamphlet 22 [6], three challenges related to dosimetry have to be solved.

Clinical cameras are not able to detect alpha-particles because of their short path. So, the first challenge is to perform quantitative imaging in order to characterize the macroscopic biodistribution of the radiopharmaceutical in the patient body. Then, bone marrow is often the dose-limiting organ in alphatherapy. However, the dosimetry of bone marrow is difficult due to its complex geometry as well as the presence of tissue inhomogeneities. The second challenge is therefore to determine the dose to this region. Finally, the short range of alpha-particles relative to the typical scale of human organ dimensions can lead to a highly non-uniform irradiation of the target volume. So, the final challenge is to investigate the microdistribution of ^{223}Ra.

To tackle these challenges, the proposed study will focus on ^{223}Ra (Xofigo®). Several hospitals in France used ^{223}Ra in clinical routine. Moreover, an ongoing multicenter phase I/II clinical trial studies the effectiveness of ^{223}Ra for renal cell carcinoma patients with bone metastases [7]. In these types of cancer, patient will mainly suffer from osteolytic bone metastases wheareas patient with prostate cancer will mainly suffer from osteoblastic bone metastases. An underlying objective is also to develop techniques that will be used for other alpha-emitter radiopharmaceuticals.

2 ACQUISITION OF QUANTITATIVE ^{223}RA IMAGES

The first challenge was to perform ^{223}Ra imaging in order to determine the activity distribution in patient body. This information is crucial to evaluate the successfulness of the therapy. Thanks to this image, the radiopharmaceutical fixation on the lesions will be ascertained and any toxicity in organs at risk could be prevented.

A few publications have reported ^{223}Ra planar images [8,9]. ^{223}Ra and its daughters have several gamma emissions in addition to the alpha emissions. Nevertheless, planar images do not give 3D information on activity distribution as opposed to tomographic images. The improved quantitative accuracy of 3D tomographic imaging modalities over 2-dimensional planar imaging is well established [10–12]. So, the goal of this first challenge was to establish a quantitative tomographic ^{223}Ra imaging protocol with clinically achievable conditions and to investigate its limitations.

First, several experiments were performed, with the *Infinia Hawkeye 4* gamma camera (General Electric, USA) at the European George Pompidou Hospital in Paris, France, using physical phantoms (the vial of Xofigo®, a syringe or a Triple Line phantom) in order to determine the best acquisition and reconstruction parameters such as the windows setting, the iteration number and filter for the reconstruction algorithm, compensation for attenuation, scatter correction.

Then, based on the MIRD pamphlet 23 [13], more complex physical phantoms were used to calibrate the gamma camera and investigate the accuracy of the quantitative ^{223}Ra SPECT imaging : a NEMA phantom (Data Spectrum™, USA) and an anthropomorphic TORSO® phantom (Orion, France) (Fig. 1).

The NEMA phantom contains six fillable spheres of different diameters: 10, 13, 17, 22, 28 and 37 mm. Each of these spheres was filled with a solution of ^{223}Ra. The rest of the phantom was filled with water (Fig. 1, a). Several SPECT/CT images were acquired and reconstructed with the optimized parameters over a clinically relevant ^{223}Ra concentration range from 1.8 kBq/mL to 22.8 kBq/mL. This study showed that the calibration factor on reconstructed images in the three biggest spheres (26.5, 11.5, and 5.6 mL) was constant for ^{223}Ra concentrations from 22.8 to 6.5 kBq/mL. For the three smallest spheres (2.6, 1.1, 0.5 mL), the contrast and the sensibility were low (Fig. 1, b) mainly because of the spatial volume effect. Below 6.5 kBq/mL and up to 1.8 kBq/mL, the calibration factor, with the exception of the biggest sphere, exhibits greater variability. This shows the limitations of SPECT/CT imaging. So, thanks to this phantom, a calibration factor was established for each sphere dimension.

In Pacilio *et al.* [14] and Murray *et al.* [15], the authors report a mean volume for osteoblastic metastases of 87 mL in patients with prostate cancer. However, osteolytic metastases in patients with renal cell carcinoma are much smaller: approximately 0.6 mL [16]. So the optimized protocol enables the activity quantification in osteoblastic metastases whereas a partial volume correction will be necessary for better activity quantification in osteolytic metastases.

Finally, the accuracy of the quantitative ^{223}Ra SPECT imaging was investigated using an anthropomorphic TORSO® phantom (Orion, France) to be as close as possible to clinic conditions [17]. This phantom contains a liver insert, lung inserts and cylindrical insert of 156 mL nominal volume. 0.5 mL and 5.6 mL spheres were placed in this last insert and another 5.6 mL sphere was fixed on it (Fig. 1, c). To mimic ^{223}Ra uptake in the healthy bone and the lesions, several tumor to normal tissue (TNT) ratios between the spheres and the cylindrical insert were performed: 6, 10, and 30. The other insert and the phantom background were filled with water. Several SPECT/CT acquisitions were performed with several clinically relevant concentration ranges: from 2.3 kBq/mL to 8.1 kBq/mL with a TNT = 30, from 8.7 kBq/mL to 21.5 kBq/mL with a TNT = 10, and from 22.8 kBq/mL to 64.0 kBq/mL with a TNT = 6. SPECT/CT images were acquired and reconstructed with the optimized parameters (Fig. 1, d).

The sphere of 0.5 mL was not analyzed because it could not be distinguished on reconstructed images (Fig. 1, d). Thanks to the calibration established with the NEMA phantom, the results showed that activity could be quantified to within an error < 20% for each TNT ratio in a 5.6 mL lesion for ^{223}Ra concentrations higher than 8 kBq/mL.

These results have enabled this protocol to be accepted in the new multicenter phase I/II clinical trial for the treatment of renal cell carcinoma with bone metastases in which a set of planar and SPECT/CT images will be acquired for each patient. So this protocol has been implemented in each hospital involved in this clinical trial and the gamma-cameras have been calibrated using à NEMA phantom.

The images acquired thanks to this protocol will give the macroscopic distribution of Xofigo®, and any other radiopharmaceutical with ^{223}Ra, in the patient body. However, bone is often an organ at risk in alphatherapy and especially its two radiosensitive regions: endosteum and red bone marrow. These regions are described at a microscopic scale and thus cannot be segmented in patients images. So voxelized models describing the bone at a microscopic scale will be used to perform a bone marrow dosimetry.

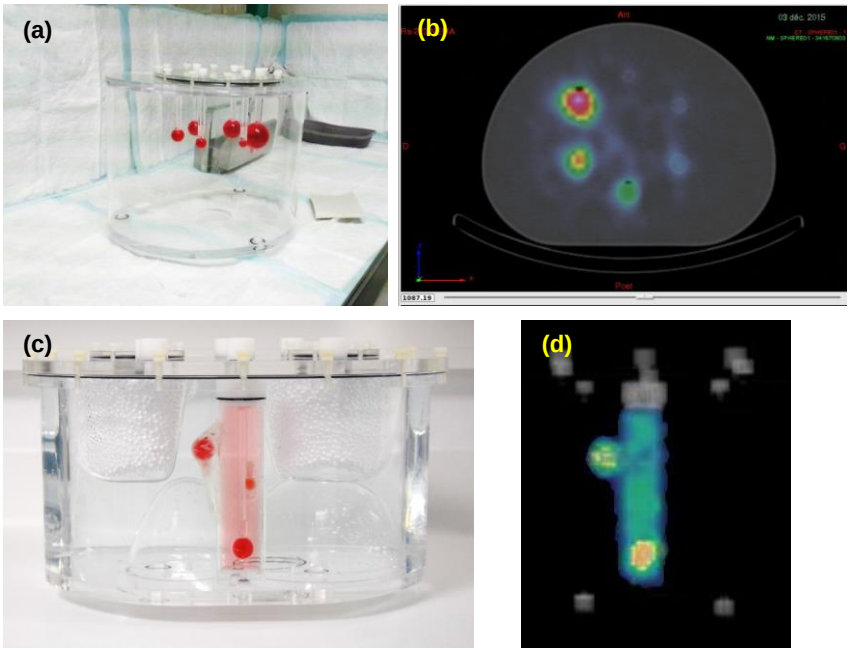

Figure 1: NEMA phantom (a): each of these spheres was filled with a solution of 22.8 kBq/mL of ^{223}Ra and the background was filled with water. Transaxial section of the reconstructed SPECT images merged with the CT (b). TORSO phantom (c): each sphere contained in the phantom was filled with a solution of 64 kBq/mL of ^{223}Ra. The cylindrical insert was filled with a solution of ^{223}Ra in order to have a TNT ratio equal to 6. 3D view of reconstructed SPECT/CT images of the TORSO phantom with TNT = 6 (d).

3 BONE MARROW DOSIMETRY

The dosimetry of bone marrow is difficult because of its complex geometry and the presence of tissue inhomogeneities. Nevertheless, the dosimetry of bone marrow is indispensable to avoid any toxicity such as myelotoxicity. As said before, patient images cannot be used to perform bone marrow dosimetry because the two radiosensitive regions in the bone are described at a microscopic scale: the active bone marrow and the endosteum. The active bone marrow is responsible for hematopoiesis (formation of blood cellular components). The endosteum is a thin layer lining the surfaces of trabecular bones and contains osteoprogenitor cells (cells responsible of the growth and remodeling of bone tissue). Moreover, the bone architecture depends of the type of the bone: for example, long and short bones. This architecture also evolves with the age of the patient as the proportion of red bone marrow decreases with age.

Thus, approximations models used in the previous studies, must be replaced by more realistic geometries. Indeed, these models did not consider the alpha-particle energy, the bone site or the bone marrow proportion [18,19]. So, in order to tackle the second challenge, dose calculations were optimized using the most recent and realistic bone models.

In collaboration with the University of Florida, several voxelized phantoms for adult male were developed from µCt images (Fig. 2). Alpha-particles transportations in 34 bone sites were simulated with the MCNP6 Monte Carlo code for an energy range from 2 to 12 MeV. The results from the simulations enabled us to calculate the energy absorbed in the red bone marrow and in the endosteum in each bone site and in the skeleton. Then, the doses to the red bone marrow and to the endosteum were calculated for the ^{223}Ra using its biokinetic model form the ICRP 67 [20]. These results were compared with the ones published by Lassmann *et al.*. The comparisons of absorbed doses to the active marrow and to the endosteum with that obtained by considering the ICRP 30 data show an overestimation of the doses by the previous dosimetric model. This overestimation may explain the lack of toxicity observed in clinical studies [21].

Moreover, the evolution of the dosimetry of bone marrow with the marrow cellularity was investigated. These results allow the adaptation of the dosimetric model to the age of the patient, thus a more personalized dosimetry.

In these calculations, the repartition of the alpha emitters was considered homogeneous in each bone region. However, a non-uniform distribution of the radiopharmaceutical can lead to a highly heterogeneous dose distribution because of the short path of alpha-particles. So, to better understand the biological effects of alphatherapy, especially ^{223}Ra therapy, it is necessary to determine the exact distribution of the radiopharmaceutical at a microscopic scale.

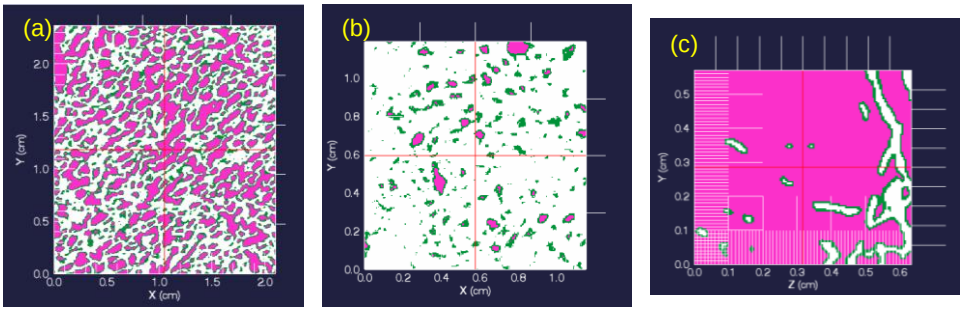

Figure 2: Transversal sections of voxelized phantoms for different bone sites: cranium parietal (a), cranium occipital (b), and clavicle (c). Trabecular bone is showed in white, endosteum is showed in green and active bone marrow is showed in pink.

4 MICROLOCALISATION OF ^{223}RA

Lastly, the third challenge was to characterize the distribution of ^{223}Ra at the microscopic level in order to better assess the relationship between dose and biological effects. However, the microscopic repartition of the radiopharmaceutical cannot be properly characterized on human. Studies were then performed on mice.

Three animal models were developed in collaboration with the CIPA in Orléans, France: a control model with healthy mice, a diseased model with osteoblastic/osteolytic metastasis (IGRCaP 1 cells kindly provided by Dr Chauchereau) and a diseased model with osteolytic metastasis (786-O cells). The metastasis cells were selected to modelize the osteoblastic lesions generated by the prostate cancer which are treated in clinical routine and the osteolytic lesions generated by the renal cell carcinoma which are the subject of the new clinical trial. This way, differences of uptake between both types of metastases were studied. The metastasis cells were inoculated intratibially in the left tibia so that each mouse in the diseased groups has a healthy tibia and one with metastasis (Fig. 3).

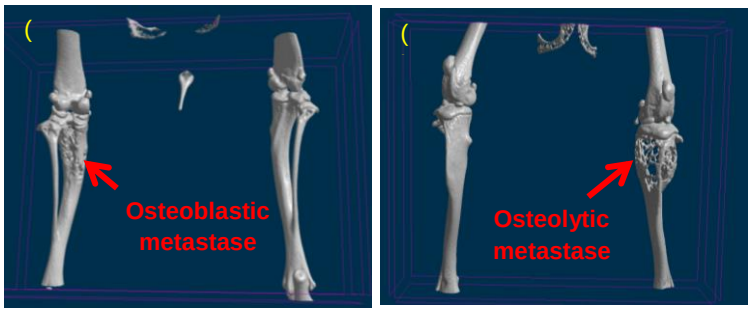

Figure 3: Structural differences between osteoblastic lesions (a) and osteolytic lesions (b).

Development of tumor was checked in mice (by bioluminescence imaging and) high resolution X ray computed tomography (Bruker 1278 CT). Mice were dosed with 30 kBq of ^{223}Ra at the CRCINA in Nantes, France. Mice of each model were divided in 5 groups (5 mice per experimentation group) and killed at 15 min, 4, 24, 48 and 96 hours for prompt dissection. Tissue activity was assayed by gamma counting for several organs in order to determine the macroscopic biodistribution of ^{223}Ra. These results showed a rapid renal clearance and an important uptake in the bones from 15 min for each model. No significant difference was observed at a macroscopic scale between the healthy tibia and the diseased tibia in each mouse of the metastasis models.

Both tibias of diseased mice were then used to achieve fresh frozen, undecalcified tissue sections. Microdistribution analysis was performed using a new digital autoradiographic system called the Beaver™ [22,23]. Autoradiographies of both tibias for each euthanasia time were acquired (Fig. 4, b and d). The same sections used for autoradiography acquisitions were then used for haematoxylin and eosin (H&E) stains in order to correlate ^{223}Ra fixations with anatomical regions (Fig. 4, a and c). The first acquired images (Fig. 4) showed different localizations of ^{223}Ra uptakes between the healthy tibia and the diseased one. In both tibias, ^{223}Ra is homogeneously distributed in the cortical and trabecular bone. Moreover, there is an important uptake of ^{223}Ra in the active bone modeling site, that is the growth plate, in both tibias. However, this uptake seems higher in the healthy tibias than in the diseased ones. ^{223}Ra does not localize directly to the tumor (Fig. 4, d), regardless of type. Instead, activity accumulates at the apposite bone surface surrounding the lesion.

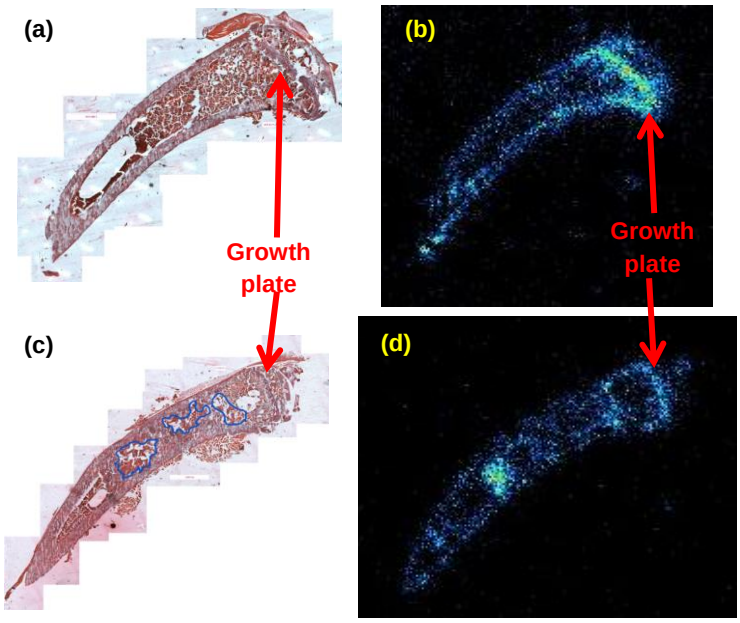

Figure 4: Mouse inoculated with osteoblastic metastasis due to prostate cancer and euthanized at 48 hours: (a) H&E stain of the healthy tibia, (b) autoradiography of the healthy tibia, (c) H&E stain of the diseased tibia with tumors circled in blue, and (d) autoradiography of the diseased tibia.

These data will make possible the study and quantification of the differences of ^{223}Ra repartition

between the healthy tibia and the diseased tibia and between metastasis due to prostate cancer and metastasis due to renal cell carcinoma. Furthermore, a biokinetic model will be deduced for each metastasis model thanks to the images at different times in order to perform a dosimetric study at a microscopic scale.

5 CONCLUSION

Therapeutic nuclear medicine is already a highly multidisciplinary field. Indeed, this study has been conducted at the interface of several fields: physics, medicine, and biology. It has then provided more insight into the various aspects of the ^{223}Ra dosimetry and now offers tools to go further in dosimetry personalization from the macroscopic to the microscopic scale.

First, a SPECT/CT imaging protocol for ^{223}Ra has been established and optimized and its limitations have been studied. Thanks to this protocol, ^{223}Ra uptakes on osteoblastic metastasis can be quantified. However, as osteolytic metastasis are smaller, additional optimizations, such as partial volume correction, will be necessary to correctly quantify the ^{223}Ra uptakes. This protocol has been implemented in several hospitals and will help determine the radiopharmaceutical repartition in patient at a macroscopic scale.

Then, the absorbed energy in the radiosensitive regions in bone was established at a microscopic scale. The dose calculations were optimized using the most realistic voxelized model. This model explicitly accounts for the volume distribution of cortical and trabecular bones and the proportion of bone marrow by skeletal site. The results showed the importance of personalizing the dosimetric phantoms in order to better understand the biological effects, in particular the lack of toxicity observed in clinical trials.

Finally, the microscopic distribution of ^{223}Ra in bones was determined thanks to animal models. Three models were developed: a group of healthy mice, a group of mice inoculated with osteoblastic metastasis coming from prostate cancer and a group inoculated with osteolytic metastasis coming from renal cell carcinoma. The first images give first insight on the repartition of the radiopharmaceutical, thus the dose distribution at a microscopic scale. This will have important implications for the design and interpretation of clinical studies evaluating treatment with ^{223}Ra, to guide clinical application with adapted dosing, and ultimately for more effective application in human.

In conclusion, this work conducted prior to the clinical trial is crucial and will allow us to develop a methodology for clinical routine. Moreover, alphatherapy is an extremely promising field and will most likely be one of the most innovative therapeutic strategies of the next decade. This study will thus give a strong foundation for the other alpha emitting radiopharmaceuticals.

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Tell me what populations eat, I will tell you how much they have been exposed

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Abstract. Realism of exposure scenario used for the evaluation of health impact due to radiological exposition and more precisely the ingestion dose has become an emerging thematic. A way to improve realism of exposure scenario is to enhance the knowledge of local population's dietary habits.
This assessment of ingestion doses is currently performed by mathematic models which need to collate food stuffs into fixed category as for example leafy vegetables, root vegetable and so on. The amount that is consumed in each category is associated by autarky data. These reflect the proportion of food locally produced that is consumed. Furthermore, dose assessment is performed for several age categories. Periodically, national surveys are mostly carried out for health-related purpose (nutrition, chemistry …) and data provided do not necessarily fit our needs. For this reason, they must be treated before being implemented into mathematic models. Furthermore, data provided by surveys may not be directly comparable due to the different methods used for their achievement. A part of study was dedicated to the development of a method for choosing the fit-for-purpose data.
To conclude, the analyse of information collected gathered from our review and the appreciation of the sensitivity of the dose led us to put more realism into our evaluation and improve a better mastering of those one.

KEYWORDS: ingestion effective dose, consumption dietary habits, data processing

1. INTRODUCTION

Realism of exposure scenario used to assess radiological exposure and more precisely ingestion dose has become an emerging thematic. Estimating the ingestion of potentially contaminated foodstuffs around nuclear industrial sites requires data about the food practices and eating habits of the residents, especially the consumption of locally-produced and home produced food [1]. In France, data about food practices and eating habits are investigated trough national diet and nutrition surveys [2-6] or more specific surveys conducted by the French Institute for Radioprotection and Nuclear Study (IRSN) [7-11] in the vicinity of nuclear sites. A portrait of these data is presented on the second paragraph.

The assessment of ingestion doses is currently performed by mathematic models which need to collate foodstuffs into fixed category as for example leafy vegetables, roots vegetables and so on. Furthermore, this assessment is also performed for several age categories. Data concerning eating habits do not necessarily fit our needs. For this reason, they must be treated before being implemented into mathematic models. A data processing is proposed in the third paragraph with the objective to be reproducible.

A wide range of data is available for assessing the ingestion dose. A method for choosing the fit-for-purpose data is proposed in the fourth paragraph.

2. PORTRAIT OF THE AVAILABLE DATA

Diet or nutrition surveys provide photography of food practices and eating habits of a specific population. In France, data are provided by national surveys [2-6] or by surveys conducted by the IRSN [7-11] in the vicinity if nuclear site. These data sets and their limits are presented in this paragraph.

2.1. Data provided by national surveys

National Institute of Statistics and Economic Studies (INSEE) conducted a survey of eating habits of the French household in 1991[2]. Data provided by this survey are currently used as a reference by IRSN.

Currently, national dietary surveys are conducted by French governmental agency concerned by public health, such as the French Agency for food, environmental and occupational Health and Safety (ANSES) or the French Institute for Public Health Surveillance (InVS). ANSES conducts periodically national surveys of the eating habits in Metropolitan France called INCA surveys. Currently, there are three available surveys INCA 1 (1998-1999) [3], INCA 2 (2006-2007) [4] and INCA 3 (2014-2015) [5]. INVS conducts surveys on particular health

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hazards. A notable example is the survey published in 2009 concerning eating habits associated with the consumption of home-produced food in the vicinity of household waste incineration plan [6].

Limits of these data sets are mainly due to the fact that each survey is subjected to a specific methodology. That makes hard to compare the different data sets. Furthermore, data concerning the consumption of home-produced food are not accessible from INCA survey report.

2.2. Surveys in the vicinity of nuclear sites

Since 2004, in partnership with French Operators (ORANOS, CEA, EDF) and BEGEAT, IRSN studied the eating habits of the residents in the vicinity of several nuclear sites:

- Residents of Bollene, near the Tricastin plant along the Rhone valley in 2004 and 2005 [7] ;
- Residents in the vicinity of the site (5 km) of Chinon Avoine NPP along the Loire in 2008 [8];
- Residents in the vicinity of the Marcoule plant (5 to 10 km) along the Rhone valley in 2010 [9];
- Residents in the vicinity of the Graveline plan NPP (5 to 10 km) along the North see in 2012 [10];
- Residents in the vicinity of the Blayais NPP (5 to 10 km) along the Atlantic ocean in 2013 [11];

One of the specific feature of this survey is to target people leaving downwind, usually in rural areas, and with a high autarky (consumption of locally or home produce food).

2.3. Discussion

From this portrait, richness and diversity of available data sets may be highlighted. However, limits associated to these data have to be kept in mind when results of the evaluation are discussed. Limits most commonly concerns:

- *Representativeness of the data for the entire population which was studied* : a strong hypothesis considering that selected persons have the same food consumption habits than the rest of the population is taken;
- *Reliability of the selected persons*: selected persons may slightly change their habits during the survey period. It is also due to the accuracy of the estimation of intake quantities;
- *Reliability of data*: whole collected data are put into an information system leading to mistakes.

As if these limits are identified, it remains hard to associate them with uncertainties.

Furthermore, data concerning the consumption of locally or home produced food are not always available. That limits the use of concerned data sets for assessing the ingestion dose.

3. DATA PROCESSING FOR THE IMPLEMENTATION INTO MATHEMATICAL MODELS

Ingestion dose assessment is performed with mathematical models (e.g. CONDOR [12], SYMBIOSE [13]) with fixed category of foodstuffs. Thus, a data processing is needed to fit these fixed categories. Moreover, data processing has to be reproducible and independent from the context of the evaluation.

The present work proposes a standardized method in order to fit both objectives. This method is presented into the following paragraph through two aspects which are the food categorization and the food intakes distribution by age categories.

3.1. Classification of foodstuffs into calculation tools category

Process followed to classify foodstuffs consists in:

1. Review of available data and foodstuff category definition from available sources;
2. Affiliation of foodstuff category identified on the first step to category asked in the model and identification of association problem;
3. Identification of criteria depending on the foodstuff itself and radio-ecological models;
4. Formalizing this procedure with a decision tree.

From the step 1 and 2, it was noted that association problems split into two typologies which are aggregation difficulties and disaggregation difficulties.

Concerning the aggregation difficulties, surveys data sets are too detailed compared with available models categories. The figure below shows an example of aggregation for root vegetables.

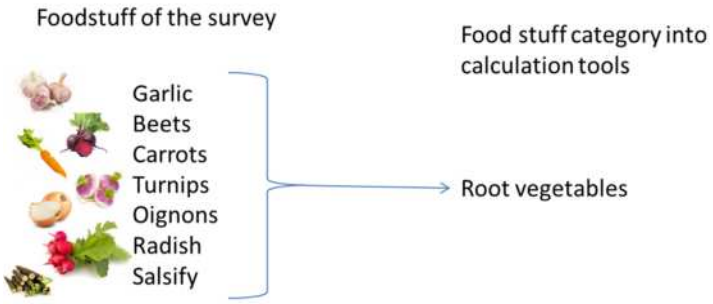

Figure 1: Aggregation of foodstuff groups

For some vegetables, the aggregation is not trivial [1,2].

Disaggregation difficulties occur when categories in a survey data set have a larger scope than those in models. Figure 2 present an example of disaggregation for milk.

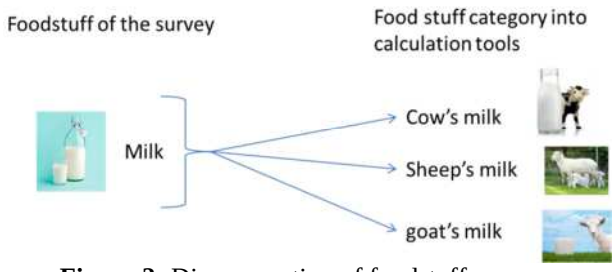

Figure 2: Disaggregation of foodstuff groups

Classification criteria define in the step 3 are listed below:

- *Possibility of foodstuff contamination:* only foodstuff which shall be subject to autarky are selected
- *Need to disaggregate foodstuff:* only concerns food mixing or processed food. In that case, proportion coming from other source of information may be used.
- *Origin of the foodstuff:* a foodstuff has animal or vegetal origins. Classification of food is based on the Technical report publication n°472 of the IAEA [14].

Finally, all these elements were formalized into classification process. Decision tree format was selected because it leads to a reproducible treatment of the data and it is easy to evolve in time.
These process lead to introduce bias due to manipulation of data. Up to now, the estimation of the impact of these biases is difficult to estimate.

3.2. Intakes distribution by age group

Data obtained by food surveys are express in terms of age categories whose limits are variable. Furthermore, age groups of models are constrained by the use of dose coefficients expressed for fixed age groups [15] as for example: under 1 y (year), 1-2 y and so on. A review of available data leads to the identification of two main difficulties for data treatment which are:

- lack of data for several categories;
- mismatch between surveys and model.

In reply to the first point, mixing of surveys data sets may be considered. Even if it introduces more bias, mixing data sets may be sufficient generic or screening calculations.

Regarding to the second point, a review of the way to deal with this difficulty shows two possibilities which are:

- modification of ages groups with linear correlation;
- simple association between ages groups.

The second solution was chosen because it limits complexity add to data treatments even if a loose of realism is introduced for concerned age group.
To illustrate this solution an example is shown in the figure below.

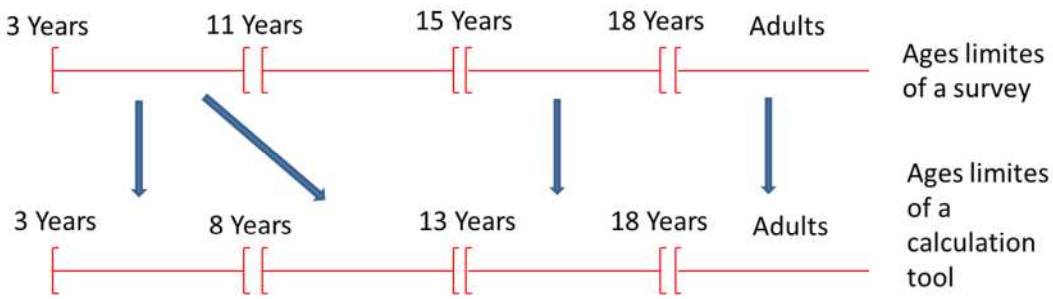

Figure 4: Implementation of simple association on data extracted from the survey INCA 2 [3].

In this example, it can be seen adult group correspond in both configurations. Indeed, it is considered that children have the same eating habits between 3 years old and 13 years old.

3.3. Discussion

As see in this paragraph, data provided by foodstuff surveys may not be directly implemented in models. A procedure of classification has been implemented in response to the need of reproducibility of the classification and independence in relation to the context of the evaluation.

Data processing leads to introduce some bias in the evaluation. Those bias mainly deal with intakes quantification. Currently, the impact of bias on the ingestion effective dose has not been investigated.

4. SELECTION CRITERIA FOR CHOSING THE FIT FOR PURPOSE DATA

A wide range of data sets is available. A weighting process to choose data set to make them opposable.

Many factors may be taken into account for selection of the fit for purpose data set. Some are summed up in the figure below.

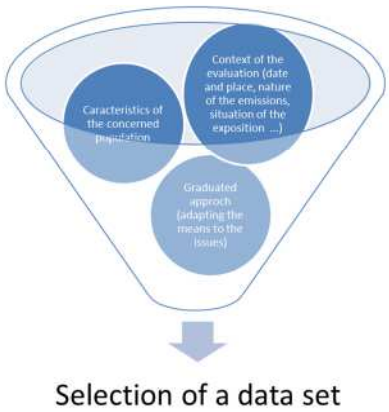

Figure 5: factors for selecting datas sets.

4.1. Criteria to compare food surveys

As seen above, the direct comparison of surveys data sets is not recommended. Comparison criteria are proposed. They can be split into two categories:

- those which lead to improve the realism of the evaluation: they quantify the balance between data sets and the context of the assessment;
- Those which lead to limit approximations: they assess the mismatch between data sets and the constraints imposed by the model.

The chart below sums up criteria used to compare data set.

Table 1: Criteria used to compare data set listed by category

Category	Criteria
Improve the realism of the evaluation	Year of implementation, survey's localization, representativeness and sampling rates, survey's duration, survey's period, results localization, statistics
Limit approximations	Amount of intake, autarky, ages group, foodstuffs categories

Once criterions are defined, it seems necessary to weight the influence of each criterion on the ingestion and on the effective dose. A way to do this is to implement sensitivity studies on each criterion. Results of study performed for results localization is shown below.

4.2. Sensitivity studies of result's localization

Within the same region, eating habits may be significantly different [16] The sensitivity of this parameter on the effective dose was performed with CONDOR [12], a dose assessment model, based on Gaussian plume atmospheric dispersion, and equilibrium transfer factors to the human food chain. The scenario is a realease of 1 TBq of a mixture of plutonium isotopes (^{238}Pu : 5,69.10^{-2} TBq, ^{239}Pu : 2,17.10^{-3} TBq, ^{240}Pu : 4,03.10^{-3} TBq, ^{241}Pu : 9,37.10^{-1} TBq and ^{242}Pu : 2,62.10^{-5} TBq), from a fictive nuclear supply localized in the vicinity of Marcoule. Thereafter, the rMain parameters used in this case-study are shown Table 2.

Table 2: Synthesis of the parameters used for the sensitivity evaluation

Parameters	Values
Release height	On the ground
Wind speed	3 m.s^{-1}
Atmospheric diffusion	low
Distance from emissary	1000 m
Wind share	1
Diffusion height	200 m
Duration of the release	30 s
Age group	adults
Duration of exposure	1 y

This study was performed with three different data sets :

- Data from the reference national survey, INSEE 91 [1], representative of French average population];
- Data from the same survey [1], representative of Mediterranean average population ;
- Data from [9]., representative ofl residents in the vicinity of Marcoule plant, with a high autarky

Results (Figure 6) illustrate the major impact on the ingestion dose assessment, and to a lesser extent on the effective dose assessment, of considering an IRSN local survey, focused on people leaving downwind, in rural areas, and with a high autarky.

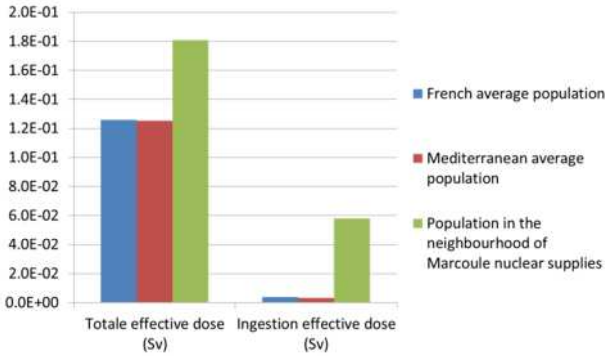

Figure 6: sensitivity of the localization of survey results on the effective dose

Regarding to the results, it can be noted that bias of considering people leaving downwind, usually in rural areas, and with a high autarky has a significant impact on the ingestion dose which can be found on the effective dose.

5. CONCLUSION

To sum up, a lot of data concerning dietary habits are available but they may not fit constrains related to the use of mathematical models. Data processing is proposed but it introduces bias and loss of realism. The investigation of the impact of those biases on the effective dose needs to be achieved. Finally, criteria are defined for choosing the fit for purpose data sets. These criteria are implemented into a weighting process. Furthermore, eating habits are only one of the human behavior features used for dose assessments. Similar approaches could be developed for example for occupancy factors (time budget).

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Radon / Thoron

"Lock up your daughters": Exploring an alternative method of radon hazard remediation.

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Abstract. The hazard associated with radon is primarily from the inhalation of its progeny. Traditional remediation focuses on the removal or preclusion of radon gas from a volume of air and can be unfavourable due to cost and practical considerations. In this study, ion generation, low humidity and a combination of both were assessed as alternative means of remediation which focus on the direct removal of radon progeny from a volume of indoor air. Data gathering was conducted in a "real-life" workplace setting to confirm the techniques are efficacious away from model ideal testing scenarios. Ion generation reduced potential alpha energy concentration (PAEC), which has been shown to consequently decrease dose in other studies, by an average of 78%. Relative humidity (RH) was found to positively correlate with PAEC, providing evidence that decreasing RH decreases dose. The combination of low RH and ion generation did not result in an additive decrease in PAEC, suggesting a mutually shared mechanism of action. Work presented here and in previous literature provides evidence that when aerosol conditions are atypical e.g. when using remedial techniques that focus on progeny removal, radon gas concentration measurement is unsuitable to assess the hazard associated with radon.

KEYWORDS: radon progeny, remediation, ion generation, humidity, unattached progeny

1 INTRODUCTION

1.1 The Hazard Associated with Radon – Radon Progeny

Radon-222 is a radioactive gas produced by the decay of radium-226, in the uranium-238 decay chain. Radon-222 decays by the emission of an alpha particle to its radioactive short-lived daughter products, or progeny: polonium-218, lead-214, bismuth-214 and polonium-214. It is well known that the principal hazard associated with radon is not from the gas itself, but radon progeny [1], hereafter referred to as progeny.

The Potential Alpha Energy (PAE) of an atom in the radon decay chain is the total alpha energy that will be emitted by that atom during its decay to stable lead-210. The Potential Alpha Energy Concentration (PAEC) of a mixture of any short-lived progeny is the sum of the PAE of these atoms in a volume of air. The SI unit for PAEC is J m^{-3}.

After radon decay, progeny grow by cluster formation to form particles approximately 1 nm in diameter (unattached progeny). Unattached progeny may attach to aerosols and form larger particles (attached progeny), which can have a trimodal activity size distribution (nucleation, accumulation and course modes) [2]. The activity of the progeny is less than that of radon gas, primarily because progeny "plate-out" onto surfaces of an area and the prevention of "grow-in" of progeny by factors such as ventilation [1]. The measure of disequilibrium between radon and progeny concentration is known as the equilibrium factor, F, and is the quotient of the PAEC of an area divided by the maximum possible PAEC for that area (i.e. if radon was in secular equilibrium with its progeny).

The unattached fraction of progeny, f_p, is the quotient of the PAEC of unattached progeny divided by the PAEC of total progeny [3]. It should be remembered that as this is a quotient, an increase in f_p does not necessarily mean the absolute PAEC of unattached progeny in a volume of air has increased.

In normal circumstances, the unattached progeny have a greater plate-out, or deposition rate than attached progeny [4]. This is because unattached progeny have a greater electrical mobility than attached progeny. Electrical mobility depends on particle size and electrical charge: the smaller the

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particle and the greater its electrical charge, the greater its mobility [5]. This explains why f_p is usually relatively low – it is given as 0.08 in the dosimetric models for all standard situations in ICRP 137 [1].

Radon progeny emit alpha particles as they undergo decay and therefore irradiate the surfaces of the respiratory system when taken into the body by inhalation. Both attached and unattached progeny may be inhaled, but due to their small size and high diffusion coefficient, the unattached progeny more effectively deposit in the bronchial region of the respiratory system and consequently contribute a greater dose per unit of airborne activity than attached progeny [6].

1.2 Index of Dose

The two main factors that affect the radon hazard source term are: radon gas concentration (as this directly affects total PAEC); and aerosol conditions (as this affects the magnitude and proportion of attached and unattached PAEC by influencing progeny attachment and deposition rates). Due to these effects of aerosol conditions, a negative correlation is observed between F and f_p. Currently, in stating an index of dose, the ICRP assumes typical aerosol conditions which can be described by $F = 0.4$ and $f_p = 0.08$. The ICRP considers this approach to be applicable to the majority of circumstances with no adjustment for aerosol characteristics.

This, with practical considerations, is the reason that radon gas concentration is used as the index of dose for the radon hazard. In this system, as one of the two parameters that affects the source term is assumed to be constant, measurement of the other (radon concentration) is deemed sufficient to gauge any change in the hazard. This approach is applicable for areas where the aerosol characteristics are those which are normally encountered and do not fluctuate significantly.

1.3 Remediation Considerations

The United Kingdom (UK) Ministry of Defence (MOD) operates over hundreds of sites and in compliance with Public Health England (PHE) guidelines, carries out an extensive radon monitoring programme. In the UK, the workplace application level of the Ionising Radiations Regulations 2017 (IRR17) is 300 Bq m^{-3} averaged over an annual period. If radon levels are found to exceed this concentration, measures must be introduced to reduce exposure to personnel (IRR17, Regulation 3 and 9). Additionally, the MOD is a landlord; if the radon concentration exceeds 200 Bq m^{-3} averaged over an annual period in dwellings, remediation must also be introduced.

Currently, most readily available remedial techniques act to reduce the radon concentration of an air space. This may be the case, in part, because it is known as the quantity of interest regarding the hazard associated with radon by ICRP and the general radiation protection community. The previous Ionising Radiations Regulations (1999) included a PAEC threshold level for application of the Regulations, regardless of the radon gas concentration; however, IRR17 includes no equivalent level. This may be because of the aforementioned negative correlation between F and f_p, which is thought by some to render dose constant for a given radon concentration regardless of PAEC fluctuation.

Installation of remedial measures which are currently commonly available is costly; reduces efficiency and use of buildings during installation and in some cases, is not practical. A recent study [7] detailed that in 199 cases where radon remediation was recommended in Switzerland (radon gas concentration > 1000 Bq m^{-3}); only 46% went on to perform remediation. Concerns regarding the financial and/or invasive nature of remediation work were found to be the most important in those who chose not to install it.

For this project, a scoping study was performed to gather information on alternative means to reduce radiation exposure in areas where the radon gas concentration exceeds the relevant UK action level. This literature review yielded two possible methods to reduce exposure: ion generation and decreasing relative humidity (RH) of an air space.

1.4 Ion Generation and Relative Humidity

The use of unipolar ion generators (IGs) to remove progeny in air spaces has been documented in the literature over the past four decades [8-10]. Mutual repulsion of unipolar air ions near the IG creates

an electric field gradient directed radially from the IG. Simultaneously, progeny as well as aerosols to which they may attach become charged to the polarity of the IG. The electrostatic force causes their migration towards the boundaries of the air space, and results in their deposition onto surfaces and removal from the air [9].

There is also evidence in the literature that the use of dehumidifiers and low RH environments decreases the PAEC of an air space [4, 11, 12].

1.5 Research Aims

The main purpose of this study was to evaluate the use of ion generation and low RH to reduce PAEC in an occupied workplace setting. Further evidence was sought that they are effective means to reduce PAEC and ascertain the practicalities of using the techniques in a "real-life" environment. It was also an objective of the project to assess whether the simultaneous use of both ion generation and low RH would result in an additive decrease in PAEC. Furthermore, by assessing the reduction of PAEC from ion generation in both a partially occupied space and unoccupied settings, a comparison could be made of the efficacy of the technique in the two scenarios.

Evaluate whether the negative correlation between F and f_p is sufficient to justify the use of radon concentration as the most appropriate index of dose from the radon hazard when artificial means are introduced to reduce PAEC.

The study also examined whether increased deposition of progeny from the techniques studied leads to a detectable increase in alpha particle contamination on room surfaces, and whether ozone was generated via IG operation.

2 METHODOLOGY

2.1 Locations Used

Data collection was primarily undertaken in an instrument calibration facility control room denoted "Calibration Room B" at the Institute of Naval Medicine (INM), Gosport, United Kingdom. During data collection periods, occupancy of the room ranged from approximately 10% to 50%. The radon concentration in this room was measured as 119 Bq m^{-3} and the volume is approximately 46 m^3.

Other locations used included two unused office rooms, MBG55 (22 Bq m^{-3}; 72 m^3) and B34 (21 Bq m^{-3}; 18 m^3), and a minor source store room (MSSR) (90 Bq m^{-3}; 29 m^3) all also at INM. A cellar store room at RAF Waddington was also used as a relatively high radon concentration area (970 Bq m^{-3}; 40 m^3) and an unused office denoted G1 (34 Bq m^{-3}; 57 m^3) was used to compare radon gas and progeny concentration. Occupation of these areas was 0 % during the data gathering period.

2.2 Ion Generator

A negative ion generator, the Wein VI-2500 High Density Negative Ioniser, was used which produces 450 trillion ions sec^{-1} cm^2 at 2.5 cm. The ion coverage is approximately 75 m^2 when used with a power of less than 6 watts.

2.3 Radon Monitoring

Radon concentration was measured before data collection to provide information on the range of effectiveness of the remedial technique. Measurement in Calibration Room B, MBG55, B34 and the MSSR was conducted using an Alphaguard monitor. The Alphaguard uses an ion chamber and alpha spectroscopy. The radon concentration at RAF Waddington was measured using poly allyl diglycol carbonate (PADC) etch-track dosimetry.

2.4 Progeny Monitoring

A Thomson and Nielsen Working Level (WL) meter was used to determine PAEC. It is an active progeny meter which utilises a silicon semi-conductor detection system. The intake pump has a flow

rate of 60 litres per hour. The air is filtered through Wattman filter paper to remove the progeny from the air and allow for detection.

Progeny monitoring was conducted in cycles, first with then immediately without the IG for direct comparison and to limit seasonal variability in PAEC. The time periods used for monitoring varied depending on availability of the location. The range of time periods was 18 – 100 hours. Varying time periods were normalised within the methodology for calculating PAEC from alpha counts of the Thomson Nielsen WL meter. PAEC diurnal variation was minimised by prioritising the consistency of data collection start and end times (+/-2 hours).

Additionally, to assess the suitability of using radon as the dose index in situations where progeny deposition is intentionally increased, radon concentration was measured in G1 simultaneously with PAEC measurement, both with and without ion generation. These measurements were taken with the Alphaguard (radon measurements) and Thomson Nielsen Working Level meter (PAEC measurements). Monitoring was conducted over five consecutive days with and without ion generation.

2.5 RH Monitoring

RH was measured using a Hygrometer Testo 610 humidistat. Humidity data was collected in Calibration Room B. The mean RH was used during each data collection period.

2.6 Ozone Monitoring

A Model 205 Dual Beam Ozone Monitor was used to measure ozone. The instrument uses UV light to measure and compare intensities of ozone scrubbed air and non-scrubbed air.

2.7 Surface Contamination Monitoring

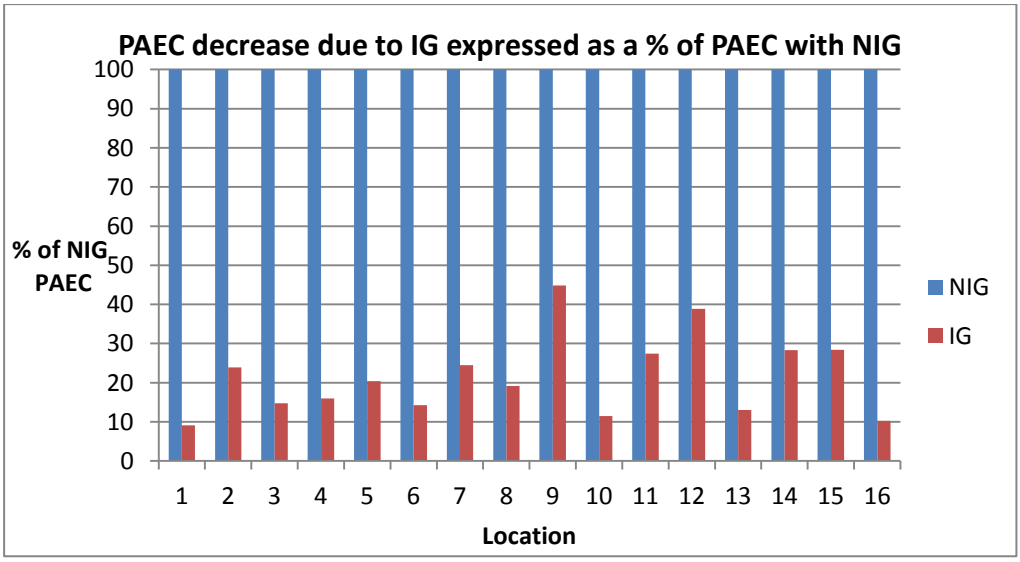
Smears of a 30 cm^2 area were conducted with Wattman filter papers when the IG was activated and in control scenarios. The filter paper was then analysed using a Canberra SC 105 Scaler-Counter & alpha drawer.

3 RESULTS AND DISCUSSION

3.1 PAEC Decreases due to Ion Generation in Occupied and Unoccupied Areas

Figure 1: Bar chart showing the difference in PAEC with (IG) and without (NIG) Ion Generation in five locations.

Location legend: 1: MSSR, 2: B34, 3 - 7: MBG55, 8 - 15: Calibration Room B, 16: RAF Waddington.



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The data in Figure 1 show that PAEC was always lower when the IG was present. The range of PAEC decrease was 55 – 91% with a mean of 78%. The highest original PAEC (RAF Waddington) decreased by 90% with ion generation. A paired *t*-test was run on the two data sets (PAEC with and PAEC without ion generation). There is a statistically significant difference between the data sets, P value = 0.02.

The mean decrease in PAEC in periods where people were present during data collection (approximately 10-50% occupancy) was 74%. The corresponding mean decrease in unoccupied areas was 81%. A *t*-test: two-sample assuming unequal variance was performed on these data and it was found that this difference is not statistically significant (P value = 0.15). This is further evidence that ion generation is an effective tool to reduce PAEC in model scenarios and novel evidence that IGs increase progeny deposition when introduced to the "real-life" occupied workplace scenario.

In this study, it was not possible to model dose due to financial and equipment constraints. Therefore, it is not possible to state categorically that the action of the IG decreased potential dose here. However, studies have shown that PAEC decrease due to ion generation [9,10], confers a decrease in the overall dose (using acceptable dose models of the time), despite any consequent increase in f_p. This is because the resultant decrease in total PAEC (and therefore F) is large enough that the corresponding increase in f_p is insufficient to keep dose constant. This point, which is discussed further below, is consistent with ICRP 137 Annex A which states that in uncharacteristic aerosol conditions (i.e. force ventilated mines – F given as 0.2) radon hazard control is based on PAE exposure [1].

3.2 Mechanism of Dose Reduction due to IGs

IGs increase the deposition rate of all aerosols in an area, including radon progeny. This has two conflicting effects on unattached progeny PAEC. Increased deposition of the unattached progeny themselves reduces the unattached PAEC. Conversely, newly formed progeny have fewer aerosols to attach to, which increases the unattached PAEC. The overall change in dose is significantly affected by the relative magnitudes of these effects. Both effects are themselves dependent on the extent to which ion generation increases aerosol deposition rate. It is possible then that these effects will be of approximately the same magnitude and will effectively cancel, resulting in the unattached PAEC for the area remaining approximately constant. Maher et al. found that negative IGs increased unattached PAEC but still decreased bronchial dose at normal room ventilation rates, while positive IGs decreased both unattached PAEC and bronchial dose (more significantly). This is likely because polonium-214 and lead-218 initially possess a positive charge when formed and thus are affected by a positive IG more rapidly than a negative IG, which must first impart a negative charge on progeny by diffusion charging [9].

IGs decrease attached progeny PAEC, as they increase the rate of deposition of the attached progeny and also reduce the rate of progeny attachment by the mechanism described above.

Thus IGs have the following effects on relevant quantities, they: may increase or decrease unattached PAEC; decrease attached PAEC significantly; increase the quotient unattached PAEC / total PAEC, commonly referred to as unattached fraction, f_p; and decrease the hazard associated with radon (dose).

3.3 Comparison of Radon Gas Concentration and PAEC change in Response to Ion Generation: Implications for the Suitable Index of Dose in Atypical Aerosol Conditions

In addition to the above experiments, the radon concentration and PAEC were measured simultaneously both with and without ion generation in room G1 and recorded at the end of five consecutive 24 hour periods. The mean decrease of PAEC was 82%, whereas there was no significant difference in radon concentration. This suggests that radon concentration measurement does not provide a valid reflection of the hazard in cases where aerosol conditions are artificially altered to reduce dose. This is in agreement with ICRP 137 which only states that radon concentration is the better index of dose when aerosol conditions are those which are normally encountered [1]. The large

differences in *F* observed using IGs clearly demonstrates that the use of an IG confers atypical aerosol conditions.

Given the results discussed above and elsewhere [9,10], it is likely that PAEC correlates better with dose than radon concentration in some scenarios and is therefore a more suitable dose index in these scenarios.

Recently developed direct radon progeny sensors (DRPSs) are able to passively measure radon progeny concentration in an air space and are able to differentiate between attached and unattached progeny [13]. This is evidence that practical considerations in passive monitoring for quantities more directly relevant to dose are becoming less onerous to achieve as technology develops.

3.4 Relationship between RH on PAEC without Ion Generation

Figure 2: Graph demonstrating the relationship between progeny (PAEC) and mean RH (%) in Calibration Room B without ion generation.

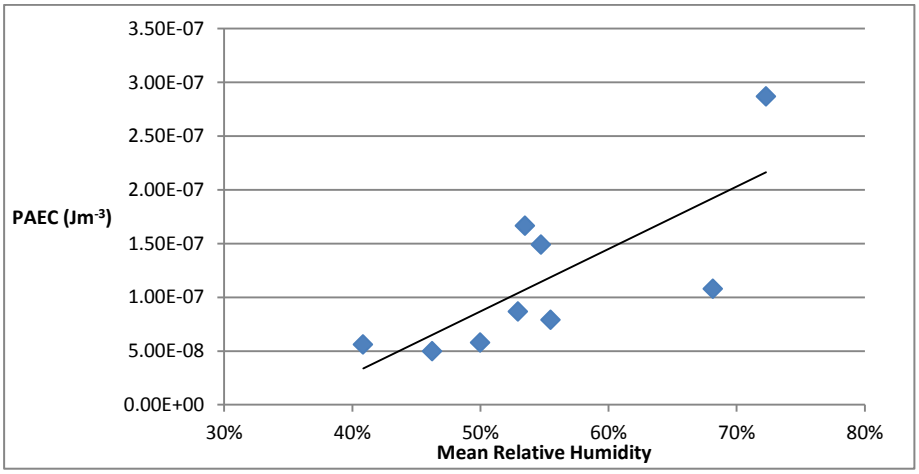

Figure 2 shows measurements of PAEC versus RH in the absence of an IG. There is a strong positive correlation between PAEC and mean RH (*R* value: 0.74; *P* value: 0.009). This finding supports previous work [11] where it was suggested that an increase in RH contributes to an increase in aerosols, which raises *F* and decreases f_p. One possible mechanism is that increased water vapour results in increased condensation events with existing aerosols, which in turn increases aerosol number in an air space as more liquid water becomes present in the air.

Chen et al. [4] and Grządziel et al. [12] demonstrated a statistically significant correlation between effective dose and RH in indoor air. This suggests that while it is likely that f_p increases in response to low RH, the overall dose decreases because *F* is the more important factor for dose. As low RH decreases PAEC through the same mechanism of action as IGs (i.e. decreasing aerosol concentration [4, 9] the explanation given in paragraph 3.2 regarding factors which increase aerosol deposition is also applicable to the low RH scenario.

3.5 Relationship between RH and PAEC with Ion Generation

Figure 3: Graph demonstrating the relationship between progeny (PAEC) and RH (%) in Calibration Room B with ion generation.

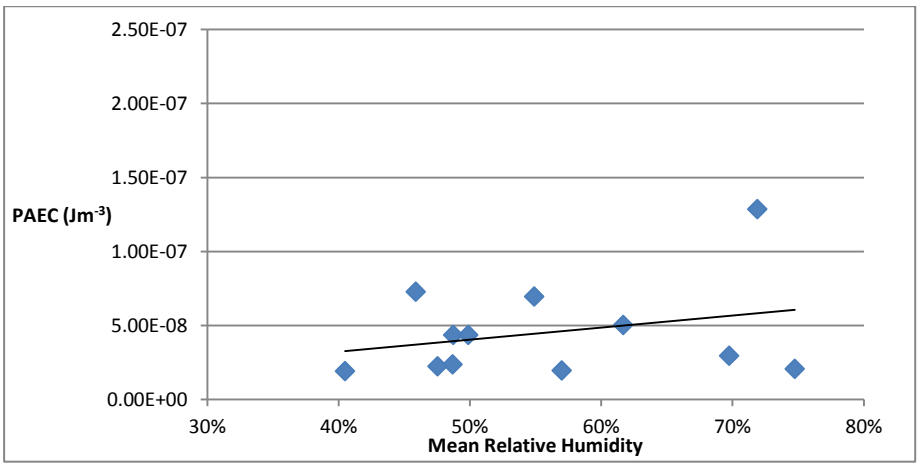

Figure 3 shows measurements of PAEC versus RH with ion generation. There was no statistically significant correlation between PAEC and mean RH when IG was applied (R value: 0.27: P value: 0.58). There is the possibility of a trend, but the sample size was insufficient to elucidate this further.

Figure 3 provides evidence that there are no obvious additive effects of utilising low RH conditions and IGs to decrease PAEC in a volume of air. This provides further evidence that low RH and IGs decrease PAEC by the same mechanism of action i.e. by increasing deposition of aerosol. As the two methods share the same functionality there is a "bottleneck" effect whereby there is no increased effectiveness in utilising both. It is likely that low RH confers this increased deposition by modulating the electrostatic properties of the air space which thus becomes more conducive to aerosol deposition.

3.6 Surface Contamination and Ozone

There was no statistically significant increase in either surface alpha contamination when the IG or low RH was utilised. There was no statistically significant increase in ozone concentration when the IG was used.

4 CONCLUSION

This study has built upon previous literature to provide evidence that remediating the hazard associated with radon by focussing on reduction of radon progeny is an efficacious alternative method in "real-life" situations. No practical issues or drawbacks have been raised. Therefore this method should be considered if traditional means targeting radon gas are not possible or favourable.

The effect of using both ion generation and low RH simultaneously to decrease PAEC was assessed. It was found that the combination of the two conferred no additive effects. It is thought that this is due to the methods working by the same mechanism of action and thus a bottleneck effect was observed.

This study and others have shown that in scenarios where aerosol conditions are not typical e.g. where radon progeny is the focus of remedial action, radon gas concentration is not a suitable index of dose. In this case the increase in f_p is not sufficient to offset the decrease in F in terms of dose. PAEC measurement is more accurate in providing an index of dose in situations where F is significantly decreased, although it is recognised that due to the more significant effect on dose of unattached progeny, it too is an imperfect measure of radiological risk.

Further work to quantify dose reduction from positive IG, negative IG and low RH, therefore clarifying the most effective radon progeny remediation strategy is suggested. Work should also be undertaken to elucidate the most suitable long-term assessment of dose in the "low F" scenario.

Due to economic and other considerations, employers, landlords and private persons may look to alternative methods of radon hazard remediation such as ion generation increasingly in the future. The

development of passive dosimetry, able to differentiate attached and unattached PAEC, is very promising for this type of remediation strategy.

5 ACKNOWLEDGEMENTS

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Robust measurement of the thoron exhalation rate from building materials

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Abstract. Thoron (^{220}Rn) exhalation from building materials has become increasingly recognized as a potential source for radiation exposure in dwellings. However, contrary to radon (^{222}Rn), limited information on thoron exposure is available. As a result no harmonized test procedures for determining thoron exhalation from building materials are available at present. This study is a first interlaboratory comparison of different test methods to determine the thoron exhalation and a pre-step to a harmonized standard. The purpose of this study is to compare the experimental findings from a set of three building materials that are tested, and identify future challenges in the development of a harmonised standard.

KEYWORDS: natural radioactivity, ^{220}Rn, exhalation rate, building materials, interlaboratory comparison

1 INTRODUCTION

The presence of natural radioactivity in mineral based building materials is generally well known, and stems from small quantities of the ^{238}U (uranium), ^{232}Th (thorium) and ^{40}K (potassium) in the mineral grains. These so-called primordial radionuclides together with its progeny are in part responsible for the radiation exposure in dwellings. This includes external exposure through the photon's emitted by the primordial radionuclides and its progeny, as well as internal exposure through the release from ^{222}Rn (radon) and ^{220}Rn (thoron) that is formed in the uranium and thorium decay series. While external radiation exposure from building materials is increasingly regulated[1] and authorities are obliged to monitor radon exposure in dwellings, relatively little is known about the exposure to thoron.

A limited number of survey's on thoron(progeny) in dwellings have been performed in Europe and Asia. In 2015 a nation-wide survey including around 2500 dwellings was completed in the Netherlands[2]. The findings from the survey demonstrate an average thoron progeny concentration (Equilibrium Equivalent Thoron Concentration, EETC) of around 0.65 Bq·m^{-3} with a median value of 0.53 Bq·m^{-3} and a maximum of 13 Bq·m^{-3}. Based on the dose coefficients reported in UNSCEAR[3] and 80% time spent indoors this result in a mean dose of around 0.18 mSv per year with an estimated maximum of more than 2 mSv. Other studies performed in thoron prone areas e.g. in China[4] and India[5] have demonstrated that thoron exposure can be well in excess of 4 mSv per year based on 80% indoor time, using more refined dosimetric models[6].

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Determination methods to measure the thoron exhalation rate from building materials have been developed since the early 80´s and include work by Keller et al.[7] and Folkerts et al.[8]. The determination methods reported in literature are by and large using either one of the following three thoron detection techniques. These are: i. total alpha counting using solid state nuclear track detectors (SSNTD) to determine the total alpha activity from thoron (progeny), ii. alpha particle spectroscopy using silicon semiconductor detectors to perform dedicated alpha-spectrometry and iii. alpha particle scintillation counting using a scintillation cell with zinc sulfide coating also known as Lucas cell.

Sharma and Virk[9] as well as Hafez et al.[10] and Ujic et al.[11] describe a determination method for thoron exhalation based on SSNTD technology. These methods use a closed accumulation chamber placed on the sample. The chamber is equipped with multiple track detectors located at variable distances from the sample and in certain cases one of the detectors is equipped with a seal to determine radon only. By doing so radon and thoron concentrations can be distinguished and subsequently the thoron concentration near the sample surface is used to compute the thoron exhalation rate of the sample.

Studies by Folkerts et al.[8], Keller and Schütz[12] and Howard et al.[13] used a silicon semiconductor based detector combined with a metallic hemisphere applied with high voltage. The electric field is used to actively deposit thoron's first progeny ^{216}Po on the detector surface, which is then detected, amplified and processed in a spectrum analyser. In recent years also commercial semiconductor detectors dedicated to radon and thoron measurement have become available and are used to determine thoron exhalation from building materials[14,15]. Methods used on scintillation technology are for example described by Zahorowski and Whittlestone[16], Saegusa et al.[17] and Tokonami et al.[18]. The methods are based on a Lucas cell with ZnS(Ag) scintillation coating, and in some case an electric field is applied to improve detection efficiency. The Lucas cell is connected with a photomultiplier and a data analyser.

In recent years also other researchers have developed dedicated facilities to determine the thoron exhalation rate from building materials. However, as of today these facilities have not been compared through an international round robin testing. The purpose of this study is to report the findings from such interlaboratory comparison, which is dedicated to the measurement of the thoron exhalation rate from building materials. A total of five laboratories have participated in the study each of the laboratories using their own measurement procedure and testing conditions.

This paper provides a description of the tested materials and the test procedures used by the individual laboratories followed by a summary and discussion of the test results. The paper is completed by a short list of recommendations to improve robustness and consistency of the determination methods.

2 MATERIALS AND METHODS

2.1 Samples

Separate material samples of the same origin and same production time were distributed among the laboratories. These included two types of concrete, and a phosphor based gypsum. Details of the sample geometry together with the activity concentrations ^{228}Ra and ^{228}Th are shown in Table 2. Where needed some laboratories have reduced the size of the gypsum sample to allow measurement on their test facility. For both types of concrete the ^{228}Ra and ^{228}Th concentration are identical implying that the ^{232}Th decay series is in equilibrium. In contrast the gypsum sample indicates a variation in ^{228}Ra and ^{228}Th concentration of around 12%, suggesting the ^{232}Th decay series is not in full equilibrium. Considering the uncertainty in the concentrations ^{228}Ra and ^{228}Th the results agree within a 99% probability level and are therefore, not significantly different. Nevertheless, if one still considers the difference in concentration statistically significant there might be time dependent variations in the ^{228}Th concentration resulting in a variation in the thoron production. However, when computing the ^{228}Th concentration for both ingrowth and decay the variation in thoron source during a

half-year period is limited to 1.7% only. As all measurements are carried out well within this time period, no correction to the measured exhalation rate is applied.

2.2 Methods

A total of five test methods have been involved in this study, operated by: Nuclear Research and consultancy Group (NRG), Helmholtz Centre, University of Pannonia, Durridge UK and Meisenberg. Details on each of the methods are described below and a listing of the key features is provided in Table 1.

Table 7 Test methods and conditions.

	Chamber ventilation	Sample exposure	Chamber* geometry	Ventilation rate	Test conditions	Detector
Laboratory A	Ventilated	Total surface	0.3×0.3×0.15 m^3	300 $ml \cdot min^{-1}$	20°C, 50%	RAD7
Laboratory B	Closed	Total surface	0.11×0.11×0.08 m^3	-	20°C, 55%	RAD7
Laboratory C	Closed	Single surface	0.3×0.3×0.2 m^3	-	21°C, 50%	CR-39
Laboratory D	Ventilated	Total surface	0.01 m^3	1000 $ml \cdot min^{-1}$	20°C, 50%	RAD7
Laboratory E	Closed	Total surface	$\frac{\pi}{4}0.25^2 \times 0.05$ m^3	-	20°C, 55%	ERS-2-S

* This is the space available for a test sample.

2.2.1 Laboratory A

The test setup is based on the test arrangement used for measuring the radon exhalation rate from building materials. This arrangement and the required test procedures are described in the Dutch standard NEN 5699[19,20]. For the determination of the thoron exhalation rate, the test setup is equipped with an active silicon based semiconductor detector (RAD7, Durridge, USA). The silicon detector is added to the test setup using a by-pass. In addition the exhalation chamber is equipped with additional fans to ensure uniform mixing of the thoron. Based on the measured thoron concentration the exhalation rate E_{Tn} is calculated as follows:

$$E_{Tn} = \frac{C_{Tn} \cdot V \cdot \lambda_{eff}}{S}, \qquad (5)$$

where $\lambda_{eff} = \lambda_{Tn} + \lambda_v$. E_{Tn} is the thoron exhalation rate ($Bq \cdot m^{-2} \cdot s^{-1}$); S is the surface area (m^2); C_{Tn} is the thoron activity concentration in the chamber; λ_{eff} is the effective decay constant (s^{-1}); V is the free volume (m^3); λ_{Tn} is thoron's decay constant (s^{-1}) and λ_v is the ventilation exchange rate (s^{-1}). The ventilation exchange rate is defined as $\lambda_v = \phi / V$, where ϕ is the nitrogen flow in $m^3 \cdot s^{-1}$. For accurate climate control the exhalation chamber with a volume of around 30 ltr is equipped with temperature control and purged with a nitrogen flow of 300 $ml \cdot min^{-1}$ and 50% relative humidity. The temperature in the exhalation chamber is set to 20°C and the measurement time is by default set to 24h to enable good statistics.

The test samples were conditioned at 20°C, 50% relative humidity (RH) prior to the exhalation measurements. The conditioning was continued until the decrease in moisture content of the material was less than 0.07% measured over a period of 7 days[21]. As thoron is primarily a surface phenomenon and only the thoron formed in the outer surface area is responsible for the thoron exhalation to the environment, accurate temperature and humidity conditioning during the measurement is very important. The geometry of the material samples is to be chosen freely; however,

where possible it should reflect de materials intended use. Based on a measurement of multiple regularly used building materials, including concrete, limestone, gypsum, brick and mortar the thoron exhalation was found to range from around 0.5 $Bq \cdot m^{-2} \cdot s^{-1}$ to 0.01 $Bq \cdot m^{-2} \cdot s^{-1}$[15].

2.2.2 Laboratory B

Measurements of thoron exhalation rates from samples are conducted using the radon and thoron measurement device RAD7. It uses the electrostatic deposition of thoron decay products. Air is conveyed with a volume flow rate of about 1 $ltr \cdot min^{-1}$ from the end of an inlet tube. For the exhalation rate measurements, it was connected in closed circuit to an air-tight chamber containing the sample, a so-called accumulation chamber[14]. The thoron exhalation rate is calculated according to Eq. (1) where the ventilation rate (λ_v) is assumed zero as small leakages of thoron from the chamber and back-diffusion into the sample are neglected because of the short half-life of the nuclide. Samples of building material were tested with and without thoron seal. The thoron-proof seal is based on an acrylic lacquer, which contains a synthetic resin as binding agent and is applied in two layers on all sides but one to provide exhalation to only one side as it is the case with a wall plastering or a floor. Unless otherwise noted, the humidity inside the chamber was set to about 50% as a supersaturated solution of NH_4NO_3 was added to the chamber. Other humidity's were achieved with different salts. To condition the moisture content of the samples to the respective moisture content all samples were placed in the chamber for at least one day prior to the measurement.

2.2.3 Laboratory C

The thoron exhalation rate is measured using an accumulation chamber. The dimensions of the chamber are 30×30×20 cm^3. The thoron (and radon) concentration are determined with a solid-state nuclear track detector (CR-39 type). Thoron as well as radon can escape from the sample and migrate in the chamber. Thoron's diffusion length in air is approximately 2.9 cm, so the distance between the nuclear track detector and the sample is 2.5 cm. During the measurement period the conditions in the chamber were controlled at 21°C and 50% RH. The calibration of the CR-39 was carried out in the radon-thoron calibration chamber. The calibration source was prepared at the Institute from a thorium-oxide solution mixing kaolinite[21].

2.2.4 Laboratory D

The laboratory uses a 10 ltr cylindrical emission chamber into which samples are placed, in a closed loop with a DURRIDGE RAD7 and a DURRIDGE ADS-144-3-P2 Active DRYSTIK. The purpose of the DRYSTIK is to exchange water vapour from the RAD7's input air stream back to the emission chamber, such that the RH inside the RAD7 is maintained at < 10% whilst the RH in the chamber remains between 45 - 55%. The chamber has air-tight electrical feedthroughs to power a small mixing fan. It is assumed that the fan is capable of instantly mixing emitted thoron evenly throughout the 10 ltr volume. A constant air flow rate of 1 $ltr \cdot min^{-1}$ is maintained by the DRYSTIK's internal pump. Combining the measured tubing internal volume and flow rate with the half-life of thoron reveals that < 0.5 % of the thoron emitted into the chamber survives a full loop of the system to be counted again by the RAD7. No correction was made for this small extra background signal. The thoron exhalation rate E_{Tn} is determined according to Eq. (1). The emission chamber is purged using the RAD7 internal pump using lab air (~ 5 Bqm^{-3} radon).

Calibration of the device is performed using a home-made ashed mantle thoron source that provides a calibration with a 25% uncertainty (1 standard deviation). The ashed mantle thoron source has been calibrated against a similar in-house source. This source, in turn, has had its thoron emission determined twice by gamma spectroscopy.

2.2.5 Laboratory E

The test setup is using a 3 ltr accumulation chamber combined with a TracerLab ERS-2-S device for measuring both radon and thoron. The device is a semiconductor type detector that is attached to the chamber in a closed loop. Where needed the detector is purged with clean air prior to any measurement.
The cylindrical shaped chamber has a 25 cm diameter and allows for a maximum sample height of 5 cm. The maximum length and width of samples is 13×13 cm^2. The chamber operates at a temperature of 20°C and 55% RH. Prior to the measurement the sample is kept at those conditions for several days. The humidity inside the accumulation chamber is controlled by using a saturated solution of potassium carbonate. The measurement device is calibrated for the ambient thoron concentration, and the exhalation rate E_{Tn} is determined according to Eq. (1). Each sample is measured three times each using a measurement time of 2 h for each determination. An additional correction factor is applied for an inhomogeneous concentration inside the closed system[22].

3 RESULTS

The results from the round robin are presented in Table 2. The results include exhalation rates from three sample materials: concrete – CEM 1, concrete – CEM 2 and gypsum.

Table 8 Sample size, activity concentrations $C_{Ra\text{-}228}$ and $C_{Th\text{-}228}$, and the ^{220}Rn exhalation rate E_{Tn} as reported by the participating laboratories.

	Concrete – CEM I	**Concrete – CEM II**	**Gypsum**
	(cm)	(cm)	(cm)
Sample size	10×10×7	10×10×7	30×30×2
	($Bq \cdot kg^{-1}$)	($Bq \cdot kg^{-1}$)	($Bq \cdot kg^{-1}$)
$C_{Ra\text{-}228}$	15±1	8±1	54±2
$C_{Th\text{-}228}$	15±1	8±1	47±2
	($mBq \cdot m^{-2} \cdot s^{-1}$)	($mBq \cdot m^{-2} \cdot s^{-1}$)	($mBq \cdot m^{-2} \cdot s^{-1}$)
Laboratory A	90±12	60±7	339±41
Laboratory B	93±30	61±25	493±118*
Laboratory C	82±13	101±19	1200±200
Laboratory D	56±20	46±20	277±22
Laboratory E	220±40	180±40	640±70

* The presented value is the mean value obtained from two individual measurements.

The results for both types of concrete are broadly consistent with the exception of laboratory E where the thoron exhalation is exceeded by approximately a factor of 2. In addition all laboratories, with the exception of laboratory C, have reported a higher exhalation rate for concrete 1. Although it is important to state that for nearly all laboratories the differences between the two types of concrete are not statistically significant. It is possible that the exhalation rate from each of the surfaces varies as the surface treatment is not identical for all of the six surfaces. Thoron exhalation particularly from concrete is predominantly a surface phenomenon due to its short diffusion length of around 0.1 cm[23]. As a result local variations in concrete curing and material composition may affect surface porosity, grain distribution and pore size distribution. These mechanical features could well contribute to the variation in exhalation rate as found in this work.

The variations in the thoron exhalation rate found with the gypsum were more profound. The results from the laboratories A and D are with approximately 300 $mBq \cdot m^{-2} \cdot s^{-1}$ broadly comparable. The thoron exhalation rate found by laboratory B is considerably higher, but considering its uncertainty the result is statistically not significantly different from the exhalation rate found by laboratories A and D. The result from laboratory C with around 1200 $mBq \cdot m^{-2} \cdot s^{-1}$ is higher and its difference with those found by others is statistically significant. The exhalation rate from laboratory E is around 640 $mBq \cdot m^{-2} \cdot s^{-1}$ and exceeds the values from laboratories A and C by a factor of around 2. Interestingly, such variation was also found with both types of concrete. Finally, an important remark on the findings from the gypsum sample. The sample size that laboratories can accommodate for varies significantly; as a result the gypsum samples have been cut or only some fraction of the sample surface has been measured. More detailed analysis, not presented in this paper, has demonstrated spatial variations in the exhalation rate of gypsum, this phenomenon could have contributed to the variations found with this type of sample material used in this study. Therefore, further work is scheduled to address the issue.

4 DISCUSSION

Consistent measurement of the thoron exhalation rate remains challenging as the findings from this study have demonstrated. Three areas have been identified that require good consideration when accomplishing consistent results, these are: i. adequate calibration of the detector and/or test setup, ii. stable climate conditions during conditioning and testing of the samples, and iii. consideration on spatial variations and aging of the sample.

Traceable thoron standards with low uncertainty needed for good calibration are not widely available and generally costly when obtained commercially. For this reason most laboratories have constructed their own thoron standard by obtained a material high on thorium and measuring the concentration of the thoron parent (^{228}Th and ^{224}Ra) and the thoron progeny (^{212}Bi and ^{212}Pb) using gamma-spectrometry. The difference between both concentrations is used as a measure for the thorium exhalation from the source. Although this approach is generally well suitable many of the thoron sources have demonstrated considerable uncertainty, which in many cases reaches as much as 15%. It is envisaged that these, at least, in part are a result of changing climatic conditions. Other approaches such as developed by PTB[24] measured ^{224}Ra and ^{212}Bi continuously and showed variations in the exhalation properties of their thoron source during calibration testing even when climatic conditions remained unchanged. In summary, good thoron sources that are traceable to a primary standard are still much needed.

Climatic conditions such as temperature and humidity are key parameters when determining the thoron exhalation from a building material. As the relevant building material are of a mineral nature they have hygroscopic properties and the uptake of moisture will influence the pore structure and subsequently the emanation and diffusion properties of the material. Results presented by De With[25] have demonstrated that the thoron exhalation from phosphogypsum can vary by as much as 15% when the humidity ranges between 40 and 60%. Other building materials have not been studied but may demonstrate comparable behaviour.

Robust measurement of the thoron exhalation rate remains challenging, particularly as the exhalation is a result of the thorium activity and the mechanical properties of the sample. This is consistent with measuring radon exhalation; however, in addition thoron exhalation is a surface phenomenon and may vary also locally. The thorium activity can be distributed unevenly, in addition surface roughness can raise effective exhalation surface, and porosity and pore size distribution may vary due to complex binding chemistry of the material structure. Recent findings by De With[25] have shown that the thoron exhalation from a single concrete cubic block may hold different exhalation rates from each of the six surfaces with variations in the order of 10 to 20%. It is demonstrated that these results emanate from variations in material composition, surface roughness and pore structure within the sample.

5 CONCLUSIONS

This work has reported first findings from an interlaboratory comparison on the thoron exhalation rate from building materials. The results have demonstrated that there are still considerable variations in the data reported by the laboratories. To improve consistent measurement of the thoron exhalation the development of an harmonised standard would be much welcomed. A future standard for the determination of the thoron exhalation rate from building materials should address:

- Suitable climate conditions during sample preparation and testing.
- Guidance on appropriate calibration of the test setup.
- Development of a generic description of the test setup and test conditions.
- Procedures for determination of the exhalation rate and its uncertainty characteristics.
- Guidance on material aging and spatial variations.

Finally, a further round robin testing by the above five laboratories using two new material samples is scheduled and aimed to incorporate the lessons learned.

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The Relationship between Social Deprivation and Domestic Radon Levels : a Study in the East Midlands, UK

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Abstract. The natural radioactive gas, radon, is present in the built environment, and at high levels is associated with an increased risk of lung cancer. This risk is significantly further enhanced when occupants also smoke. The UK government now publishes the Index of Multiple Deprivation (IMD), a measure of average overall deprivation, as well as indices of subsets such as health, education status, income and living environment. Studies investigating a number of environmental pollutants suggest that the more-deprived populations are exposed to higher levels of many pollutants, with direct adverse impact on their health. In contrast, however, two recent studies have suggested that increased social deprivation in the UK is associated with lower environmental radon exposure. This paper considers this suggestion by reviewing social deprivation and radon levels in 231 postcode sectors - small areas of around 7500 people - in both urban and rural settings, in the East Midlands of England, using the IMD for 2015. The area includes a number of major towns, among them Leicester, Bedford, Northampton, Wellingborough, Kettering and Rugby, together with many villages in rural areas. There is an apparent trend to greater overall deprivation in low radon areas, in contrast to many other environmental pollutants. The trend only has a weak correlation. This study identifies two factors that show some correlation with this trend. Firstly, urban areas have lower average deprivation than rural areas. Secondly, since urban areas tend to contain higher proportions of multi-storey apartment blocks than rural areas, the lower radon exposure experienced by residents of the higher floors reduces the overall average residential radon exposure in the locality. In addition, a significant contributor to urban radon exposure is the location of most major urban centres in the UK on intrinsically low-radon alluvial geologies.

KEYWORDS : Radon; Environmental Pollution; Social Deprivation; Index of Multiple Deprivation

1 INTRODUCTION

Tobacco smoking is the primary cause of a range of diseases responsible for preventable morbidity and premature mortality, accounting for 81,400 deaths in England in 2009 [1]. Although smoking is the most significant risk factor for lung-cancer, being implicated in 86% of all lung-cancer deaths, Radon is now identified as the second most significant risk factor for lung-cancer. Case-control studies have demonstrated an increase in lung-cancer in people with raised radon levels in their homes [2] and the risks from radon and smoking are considered to be multiplicative [3].

Radon is a naturally occurring radioactive gas, produced by the decay of the long-lived naturally occurring Uranium and its daughter products, notably Radium, and its varying distribution in underlying rocks and soils is a key, but not exclusive, factor determining radon concentration levels in the built environment. Studies have shown the influence of numerous factors, including house type, building materials, foundations, ventilation and double glazing, on domestic radon levels [4, 5], and have led to the development of a model [6] suggesting that 25% of the total variation in indoor radon in England and Wales could be explained by bedrock and superficial geology.

Since the early 1990s there has been increasing concern that the location of hazardous industries and the spatial distribution of environmental pollutants have resulted in higher exposures to more deprived populations. This led Jerrett et al. [7] to postulate the 'triple jeopardy' of environmental inequality, poor

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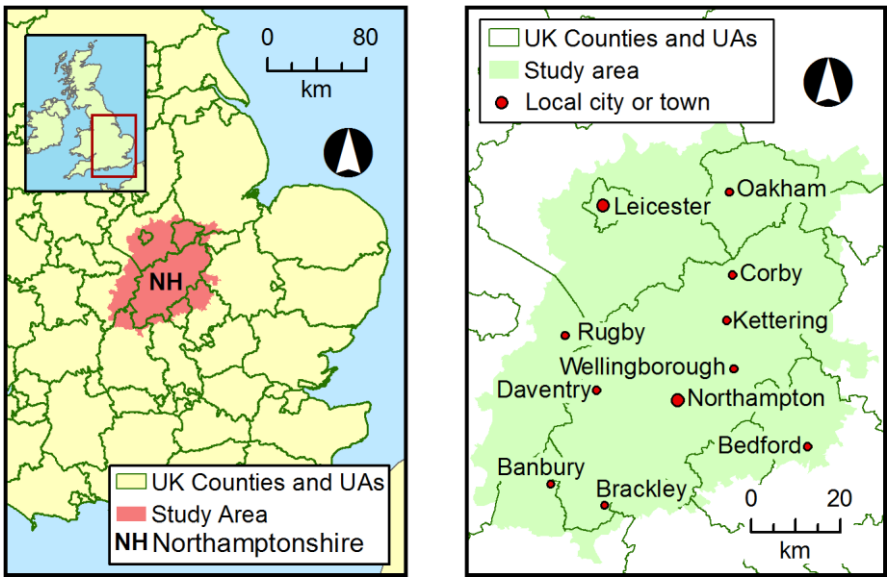
socio-economic status, poor living environment leading to poor health, and many research groups [8, 9] are now studying different pollutants to evaluate this potential relation. Briggs et al. [8] investigated a wide range of pollutants in England, including radon, and showed strong associations with Socio-Economic Status for air pollution from volatile organic compounds (VOC) and NO_2, weaker associations for some other pollutants such as SO_2, but a weak negative correlation for ozone and radon. More recently, Kendall et al. [10], using data from the UK Childhood Cancer Study of the 1990s [11, 12], have suggested that greater social deprivation is associated with lower radon areas in the UK, lending support to the findings of Briggs et al. [8].Finally, Riaz et al. [13] have shown for the first time that urbanisation is an additional factor to consider when investigating deprivation and lung-cancer incidence. This has not been considered before in the context of radon.

This paper analyses the population of an area in the East Midlands of England in a cross-sectional case study, looking at deprivation, house style, urbanisation and radon levels in small geographical areas, to investigate the relation between social deprivation and radon in more detail.

2 METHOD

The study area, in the East Midlands of England, shown in Fig. 1, includes the counties of Northamptonshire and Rutland and parts of adjoining counties. The area is largely rural, with villages and towns with populations ranging from a few hundred to a few thousand, but also includes the major conurbations of Leicester, Northampton, Wellingborough, Kettering, Corby, Bedford and Rugby. It has good rail links to London and the North of England, and the M1 motorway runs through the area.

Figure 1: Location of the Study Area in Central England



Postcode sectors (e.g. NN12 3) are the smallest areas in the UK for which average domestic radon levels are published. They are areas with mean populations of around 7,500, but with a significant variation. There are 231 postcode sectors in the study area, with average population ranging from 15 to 17,365 (mean 7,820, standard deviation 3,926). A rural postcode sector may include one or more villages together with surrounding countryside areas, and inevitably covers a much larger area than its urban counterpart. The percentage of houses found to be over the domestic radon Action Level for each postcode sector in the study area was taken from the Indicative Radon Atlas for England and Wales - 2010 Data Review [14], published by the former Health Protection Agency (HPA) [now Public Health England (PHE)].

The UK Government Department for Communities and Local Government (DCLG) publishes data for England on deprivation in seven domains – Income, Employment, Education, Health, Crime, Barriers to Housing & Services, and the Living Environment. The smallest units for which data are available are

Lower-layer Super Output Areas (LSOAs), based on the 2011 UK census. There are 32,844 LSOAs in England, with an average population of 1500. Each LSOA is assigned a Deprivation Score under each of the seven headings, and these are then amalgamated to provide a single Multiple Deprivation score. The LSOA Scores are ranked in ascending order to generate the Index of Multiple Deprivation (IMD), which currently ranges from 1 (most deprived) to 32,844 (least deprived), the published tabulations also allocating LSOAs into 10 equal-sized deciles. The most recent IMD tabulation was published in 2015 [15]. Middle Super-Output Areas (MSOAs) are larger areas, combining around 4 LSOAs and matching local authority boundaries where appropriate. Technical guidance [16] provides an algorithm for calculating the average IMD score and rank for an individual postcode sector, using the 2011 census data for the population of each full postcode (e.g. NN12 3XY), and this was done for each postcode sector in the study area. The calculated IMD Scores in this dataset ranged from 5.05 (least deprived) to 85.36 (most deprived), with an average of 18.11.

Under the UK Government Rural-Urban Classification (RUC) scheme [17], introduced in 2001 and updated at the 2011 Census, LSOAs are classified as one of four Urban or six Rural categories. Postcode sectors in the study area were classified as either Rural or Urban depending on the classification of their LSOAs, with a few sectors containing both Urban and Rural LSOAs being classified as 'Mixed'. However, it was not possible to allocate some postcode sectors to one of the specific categories as many of their constituent LSOAs were in different categories. The results were tabulated and plotted on maps.

3 RESULTS

The deciles for the 2015 Index of Multiple Deprivation (IMD) Ranking for each postcode sector in the study area are shown in Fig. 2(a), where 1 is the most deprived decile, and 10 is least deprived. The study area contains postcode sectors covering the whole range of deciles, with the most deprived areas (Decile 1) being found in the centres of Leicester, Bedford, Northampton and Corby, the least deprived postcodes (Decile 10) being rural areas around Market Deeping in southern Lincolnshire, Olney in Buckinghamshire and Broughton Abbey in Leicestershire. The mean Decile for the study area is 6.23, suggesting an average deprivation slightly less than the average for the whole of England.

3.1 Radon

Fig. 2(b) shows the percentage of existing houses in each postcode sector with radon levels over the UK domestic Action Level, taken from the HPA Radon Atlas [14].

The relationship between IMD Scores and Radon levels is shown in Fig. 3, which plots the percentage of houses with radon above the Action Level versus the calculated IMD score for each postcode sector. Although the correlation is modest, it is significant (p=4.3%), areas with more deprivation tend to be in low radon areas, with a logarithmic trend-line having a relation $Rn = -0.014\ln(IMD)+0.0729$, with a low coefficient of determination $R^2=0.0177$.

3.2 Population

As noted in the Methods section, postcode sector populations vary considerably, so it is appropriate to consider postcode population density when considering any impact of population. The population density at the 2011 census in each postcode sector in the study area is shown in Fig. 2(c).

A plot of population density against IMD Score is shown in Fig. 4. Again there is considerable scatter, with a moderate but significant trend ($p \approx 0$) to increased deprivation in areas with higher population density, $R^2 = 0.1407$. Higher radon areas appear to have lower population density.

3.3 Urban – Rural Classification

As noted above, each postcode sector was coded as Urban, Rural or Mixed, giving a total of 143 Urban, 75 Rural, and 13 Mixed sectors. These postcode sectors are identified in Fig. 5, where it can be clearly seen that urban postcodes are predominantly in low radon areas, while high radon areas are generally rural. There is no significant trend relating radon and IMD in any of the three classifications.

Figure 2: Study Area - (a) Social Deprivation Deciles; (b) Percentage of Homes with radon levels over the Action Level; (c) Population Density

(a) (b) (c)

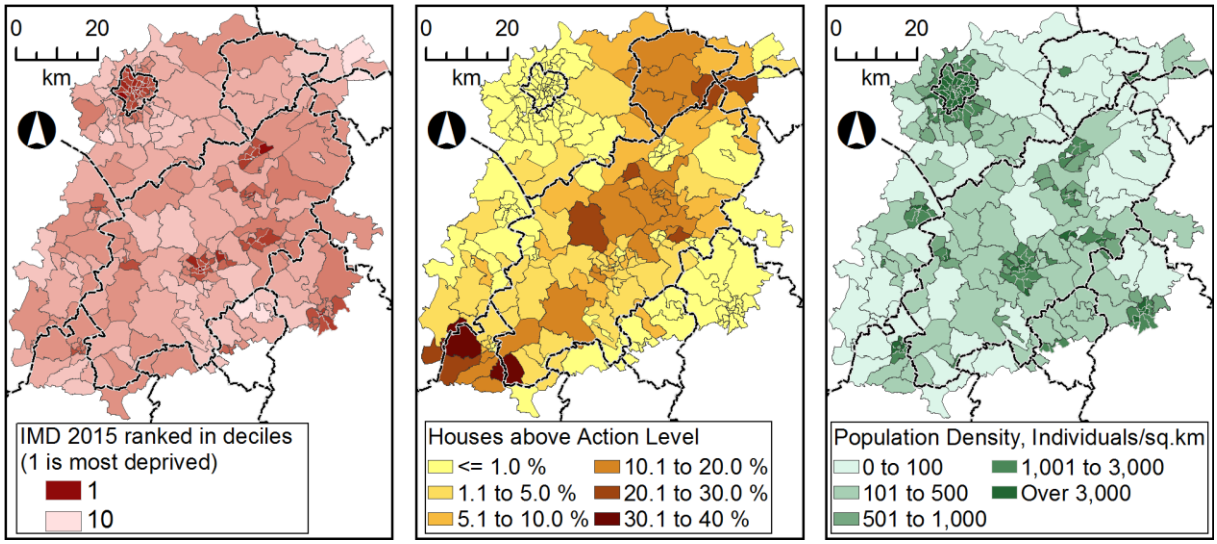


Figure 3: Relationship between Radon Potential (% of houses above the AL) with IMD Score

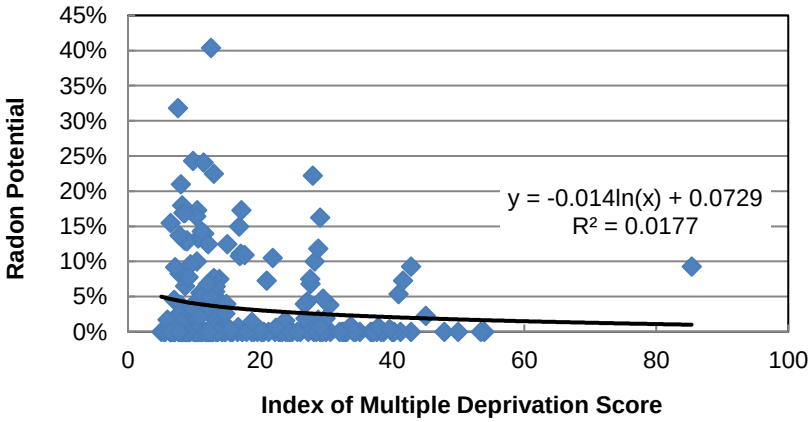


Figure 4: Population Density plotted against IMD Score

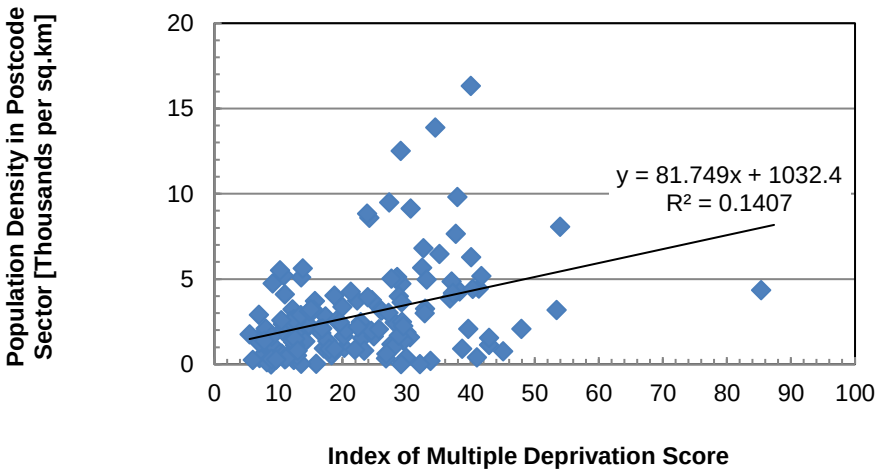
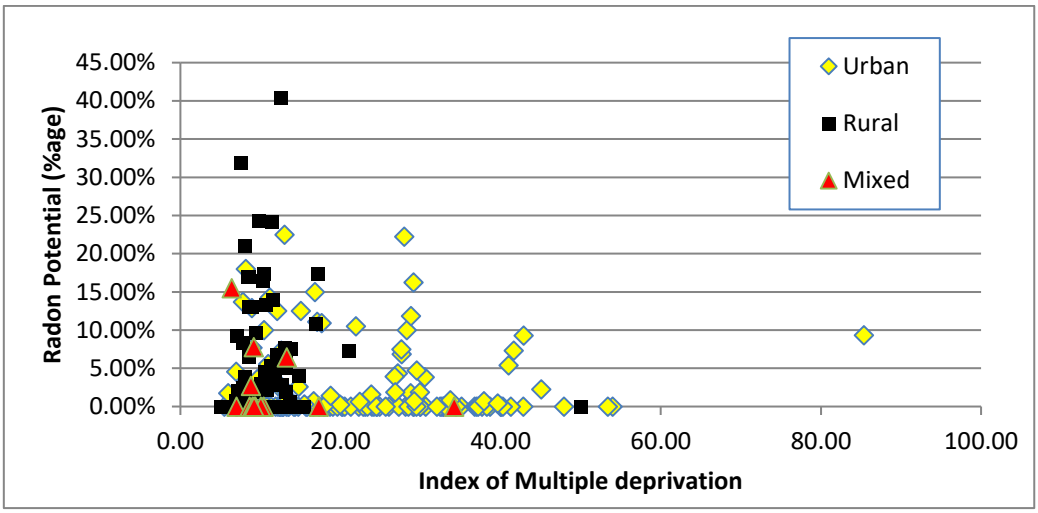


Figure 5: Radon levels and Social deprivation in Urban and Rural settings

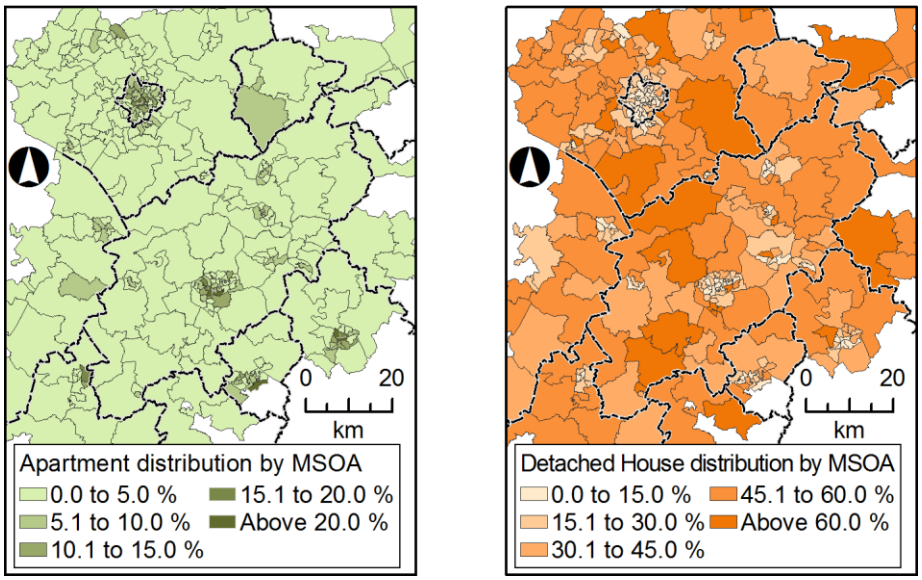


3.4 Building Types

The 2011 UK census classifies residential accommodation into 6 types – 3 for houses (detached, semi-detached and terraced) and 3 for apartments (purpose-built, converted or shared, and commercial). Although this data is available for LSOAs, and therefore potentially convertible to postcode sectors, this study considers Medium Super Output Area (MSOA) level, as both IMD and the Rural-Urban Classification are available at this level. The study area contains 308 MSOAs, of which 216 were urban, 50 were rural town and fringe, and 42 were rural villages. The distribution of apartments and detached houses in the study area is shown in Fig. 6.

Apartments form an average of 5.8% (range 0.1% to 82.9%) of the housing stock in urban MSOAs in the study area, but only 2.0% (0.3% to 6.3%) in rural towns and 1.4% (0.4% to 3.7%) in rural villages and dispersed areas. For detached houses, this situation is reversed, with 54.0% (35.9% to 70.2%) rural villages and dispersed areas, 47.2% (25.1% to 63.6%) in rural towns and 26.1% (1.6% to 71.0%) in urban MSOAs.

Figure 6: Distribution of House Types in 2011 across the Study Area
(a) all apartments (b) Detached Houses



4. DISCUSSION

The trend-line in Fig. 4 suggests that areas of higher deprivation have, on average, lower radon levels, replicating the findings of Kendall et al. [10] and Briggs et al. [8]. Kendall et al. [10] used deprivation data from a case-control study of 6000 participants dating from 2000, and deprivation data using SES methodology for electoral wards from 1988, while Briggs et al. [8] used radon data from 2004 and IMD data from the 2001 UK Census. This trend is therefore consistent over a number of years and independent of methodology. The data in Fig. 4 has a large scatter, suggesting a weak correlation and that other factors may be more significant.

Other authors have noted that indoor radon levels are influenced by the underlying geology. In England, the major urban conurbations of London, Leeds, Manchester, Birmingham and Newcastle are all in low radon areas. This is not surprising, as major towns in England were established at strategic points with access to the sea or inland at the crossing points of rivers and consequently, as Briggs et al. [8] suggest, at sites where radon levels are low. Such settlements would have developed where alluvial silts and muds would have been deposited. Such deposits, if clay-rich, as most will be, will be less permeable and will tend to act as a barrier to radon.

An additional factor reducing radon exposure in urban areas is the greater predominance of high-rise buildings and apartments. On average, radon levels decrease to 70% in each higher storey [5]. Assuming an average of 4 storeys and 4 apartments per storey, Denman et al. [18] estimated that average radon exposure to apartment block occupants was around 45% that of occupants of a two-storey house. Denman et al. [18] also noted that, in 2009, apartments comprised 38% of all dwellings in London, but only 9% in the East Midlands Region, and, at similar radon levels, the population in London would be exposed, on average, to 83% of the radon exposure in the East Midlands. Thus, the higher density of apartments in urban areas could explain at least some of the variation of radon exposure with deprivation.

IMD ranking, by its nature, does not permit longitudinal study of changes in deprivation and there have also been a number of changes in the geographical definition of MSOAs between the 2001 and 2011 Censuses. Together, these factors mean that it is difficult to study changes in deprivation over time, a particular issue when considering changes in rural deprivation. Villages in Northamptonshire, for example, saw a population decline from 1880 to the 1930s by 26%, but since then have seen a net influx of immigration of high-status ex-urban households and extensive new house building [20]. The increase is as a result of the advent of the motor car, permitting rural residents to drive into the nearby town for work, or even commute to London by train. Sherwood [19], in his study of South Northamptonshire, reported that by 1981, at least 50% the working population of the majority of wards worked outside the district, with a quarter of wards having over 70% working away. Commenting on the social status of such villages, he noted "*Superimposed upon a predominantly elderly demographic structure with a strong orientation to agriculture, these parishes are gaining a veneer of new, younger, high-status households living in substantial dwellings built in small numbers and at low densities*". The trend in house-building and net migration to such villages continues. For example, Brixworth, a large village in Northamptonshire, had a population of 1173 in 1931 [21], increasing to 5,162 at the 2001 census, and 5,228 at the 2011 census [22], with current building of new estates expanding the village further. In this scenario, it would be expected that the average IMD Score would increase as the population grows. It is also true that pockets of rural deprivation would be small, consisting of a few families in a village, and this is unlikely to be detected even in the small LSOA areas. Such changes over time and, of course, also changes in the degree of deprivation in the urban environment, could be expected to have little direct impact on the relationship between IMD Score and radon, being most likely to affect the degree of scatter.

One arena where it is important to take into account variations in the levels of deprivation is in the epidemiological assessment of the health risks of radon. The studies showing environmental inequalities and ‘triple jeopardy’, and those showing that those living in areas of higher deprivation smoke more, all

demonstrate reduced life-expectancy of those who are more deprived. In addition, smoking and radon together increase the risk of lung-cancer. Thus lung-cancer incidence will be higher and life expectancy lower in urban areas, even though radon exposure will be lower. These factors need to be taken into account when studying the risks of radon to the population.

The current UK policy for reducing the risk of radon is to encourage householders who live in RAAs to test radon levels and, if these exceed the Action Level, to remediate their homes. Previous studies have shown that householders are not always willing to pay the cost of this work, that only around 15% do so and that those with lower incomes are less likely to pay [23]. As these are more likely to be smokers, the outcome is that current initiatives to reduce radon risk are not reaching those most at risk. However, other studies have shown that smokers are more likely to live in urban areas [1, 24] where radon is lower, and so the issue of "willingness to pay" may not be as significant on a nationwide scale as might be thought.

5 CONCLUSIONS

This study shows a small decrease in deprivation score with radon exposure, with weak correlation, consistent with the previous studies of Kendall et al. and Briggs et al., both of which used older datasets and different methodologies and study areas in the UK. This is, in part, due to the predominance of multi-storey accommodation in urban relative to rural areas. In addition, since the major centres of urbanisation in England and Wales are in lower radon areas, we suggest that it is not appropriate to regard the weak correlation between deprivation and radon exposure as a causative link. However, it is important to consider the correlation in epidemiological studies of radon exposure, as deprivation is linked with a shorter life span, and other confounding factors.

6 ACKNOWLEDGEMENTS

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A radioecological study of radon and thoron in soil gas and water

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Abstract. Inhalation of radon is the second largest cause of lung cancer in Europe after smoking. The new EU council directive 2013/59/EURATOM, the implementation of which into national law is currently ongoing, stipulates national action plans for the protection of the population from excessive radon exposure. The aim of this project is a radioecological study of radon and thoron activity concentrations in soil gas and water in selected locations with increased radon potential in Austria. Soil gas measurements were carried out at eight different locations by inserting steel probes into the ground and pumping the soil gas into measuring instruments based on ionization chambers. Additionally, the radon content in water samples was determined using an ionization chamber and the activity concentrations of radionuclides of the uranium-radium and the thorium decay chains in soil samples were measured using low-level gamma spectrometry. Activity concentrations of radon and thoron of over 100 kBq/m³ were detected in soil gas. The radon content in the water samples was found to be between 42 Bq/L to 118 Bq/L.

KEYWORDS: radon; thoron; activity concentration; radioecology; low-level gamma spectrometry.

1 INTRODUCTION

Radon (Rn-222) is a radioactive noble gas that occurs naturally in soil and water as a product of the uranium-238 decay chain. In 1998, it was classified as a human lung carcinogen based on data from epidemiological studies of underground miners exposed to high radon activity concentrations [1]. Evidence has shown that even long-term average radon activity concentration levels below 200 Bq/m^3 are associated with a risk of lung cancer [2]. Over one third of the natural radiation dose received by the general population in Austria is due to the inhalation of radon and radon progeny [3]. The average annual effective dose due to thoron (Rn-220) is much smaller than that due to radon [4]. Therefore, its influence has often been neglected in studies of radon exposure. The aim of this project is a radioecological study of radon and thoron activity concentrations in soil gas and water in selected locations with increased radon potential. The Waldviertel in Lower Austria and the Mühlviertel in Upper Austria are part of a geological zone called the "Granite and Gneiss Plateau" situated in the north of Austria. Due to the abundance of granite, the ground in this area has a high uranium and thorium content compared to other geological zones. Measurements of the activity concentrations of radon and thoron in soil gas were taken at eight different locations within this region (Figure 1).

Figure 1: Map of the measurement locations.

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2 METHODS

2.1 Radon and thoron activity concentration in soil gas

To measure the activity concentration of radon and thoron in soil gas, a steel probe of 1.2 m or 1.6 m length and 12 mm diameter is used. A metal tip is placed at the bottom of the probe, which is then hammered into the ground to the desired depth. A thin metal rod inserted into the probe is then used to hammer the tip 5 cm further into the ground. This allows the soil air to be drawn in from the top of the probe using either a syringe or a pump.

The soil gas measurements in July 2017 (locations 1-3, see Figure 7) were conducted by taking a volume of 100 mL of soil gas from the probe using a syringe and inserting it into an AlphaGUARD (model PQ2000 PRO RnTn by Saphymo - a radon detector with ionization chamber). For this measurement technique, the AlphaGUARD is set to 10 min flow mode and its openings are closed after inserting the soil gas into the chamber. Since the AlphaGUARD has an active detector volume 0.56 L, but only 100 mL of soil gas are inserted into it, the displayed value is lower than the actual radon activity concentration in the soil gas sample. It therefore has to be multiplied by a previously determined calibration factor. The calibration factor for the AG BEV is 8.07 ± 0.01 [5]. For the other AlphaGUARDs it is assumed to be 8 ± 10 %. At least four measurements are taken the first one is discarded, and the following are used to calculate a mean value. This eliminates the effect of thoron on the radon measurement, as it will have decayed almost completely after the first 10 min measurement due to its short half-life of 55.6 s. To simultaneously measure the activity concentrations of radon and thoron, the soil gas is pumped into the AlphaGUARD (in radon/thoron mode) using an AlphaPump. The setup is shown in Figure 2. This measurement technique was used for the soil gas measurements in September 2017 (locations 4-8, see Figure 7). Measurements were taken for at least 40 min and one measurement was taken over night.

Figure 2: Setup to measure the activity concentration of radon and thoron in soil gas.

2.2 Soil Permeability

Permeability refers to the ability of the soil to allow fluids and gases to pass through it. Figure 3 shows the setup used to measure the soil permeability. An AlphaPump set to 1 L/min draws in soil gas through the probe. A manometer and a flowmeter are connected to measure the pressure and flow rate of the soil gas. The permeability of the soil can then be calculated using the following formula [6].

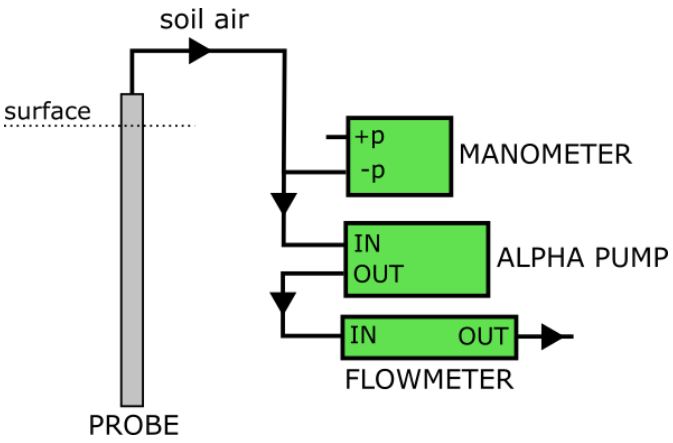
$$k = \frac{Q\mu}{Fp} \tag{1}$$

Q ... Flow rate (m^3/s), k ... Permeability (m^2), p ... Pressure (Pa), μ ... Viscosity of air (1.75×10^{-5} Pa s), F ... Form factor (m).

$$F = \frac{2\pi l}{\ln\left(\frac{2l}{d}\sqrt{\frac{4D-l}{4D+l}}\right)} \quad (2)$$

l ... Length of the effective probe volume (m), d ... Diameter of the effective probe volume (m), D ... Lower depth of the effective probe volume underneath the surface (m).

Figure 3: Setup to measure the soil permeability.



The soil permeability is used in combination with the radon activity concentration in soil gas to determine the radon potential [7,8].

2.3 Gamma spectrometry of soil samples

At each measurement location soil samples were taken to determine the activity concentrations U-238, Ra-226, Th-228 and K-40 via low-level gamma spectrometry. At least two soil samples were taken at each location, one close to the surface (0-10 cm below the surface) and one deeper underground (30-50 cm below the surface). The samples were weighed before and after drying at 80 °C for a few days. They were then filled into suitable air-tight containers of 11 mm height and 64 mm diameter. The gamma spectrometry measurements were carried out using a p-type high purity germanium semiconductor detector. It is cooled with liquid nitrogen and makes use of a fully automated sample changer. The measurement time for each sample was between one and one and a half days.

2.4 Gamma dose rate

The measuring instrument was placed on the workbench (at a height of approximately 80 cm) close to the probes and the measurement was carried out over a time period of about an hour. Two different measuring instruments were used for the measurements[2] in July 2017 and September 2017, but in both cases the measured quantity was the photon dose equivalent rate H_x.

2.5 Radon in water

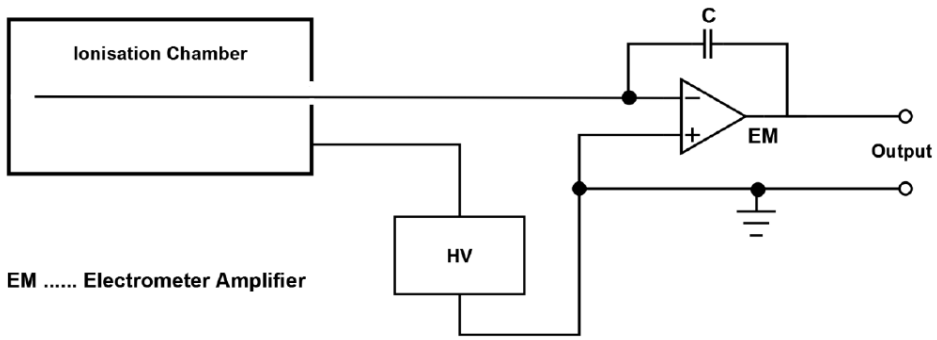
Water samples were collected from a house in Königswiesen, Upper Austria (07/09/2017), the waterworks in Bärnkopf, Lower Austria (06/09/2017) and from a spa in Bad Zell, Upper Austria (08/09/2017). All samples were collected using commercially available 1 L glass bottles.

The radon content in the water samples was determined using an ionization chamber with a volume of 9.3 L and an applied voltage of 260 V. The experimental setup is shown in Figures 4 and 5.

[2] July: UMo Universeller Monitor für Strahlenschutz, EG&G Berthold. Type LB1236, SNr. 1513.
September: NBR-Detector FHZ672E, ESM Eberline. Z.-Nr. 42540/61, F.-Nr. 0137.

Figure 4: Photograph of the setup to measure the radon activity concentration in a water sample.

Figure 5: Schematic diagram of the setup to measure the radon activity concentration in water. Diagram by Konrad Lotter [9].



First, air is pumped out of the ionization chamber to reduce the pressure. Then, the cap of the bottle containing the water sample is removed and replaced by a plug with two tubes connected to it. When connecting one of those tubes to the ionization chamber, the reduced pressure causes air to flow into the chamber through the water bottle. The resulting bubbles cause the emanation of radon from the water. After an amount of air three times the volume of the water has passed through the bottle, virtually all the radon will have emanated from the water [10].

An electrometer was used to measure the charge collected over a time interval. By dividing the collected charge by the time interval, the current of the ions produced in the chamber can be calculated. The radon activity A in the ionization chamber is directly proportional to the produced current I and is given by

$$A(\mathrm{Bq}) = 19.24 \cdot I(\mathrm{pA}) \tag{3}$$

At least five measurements were taken from each water sample, about half an hour after the radon had finished transferring from the bottle to the ionization chamber. However, Equation 3 is only valid when the radon in the ionization chamber is in equilibrium with its daughter products, which happens after about three hours. Because the measurements were taken before equilibrium was reached, they were multiplied by a factor of 1.3 ± 0.1, which was obtained by dividing the calculated total maximum activity in the ionization chamber by the calculated activity after half an hour. Furthermore, because the measurements took place a few days after the collection of the samples, the results were corrected to account for the radioactive decay of radon between the sampling time and the measurement time.

3 RESULTS

3.1 Soil gas measurements

The radon activity concentrations measured from individual soil gas probes lie between 22 kBq/m^3 and 246 kBq/m^3. Thoron activity concentrations lie between 44 kBq/m^3 and 310 kBq/m^3. The radon activity concentration in soil gas depends on many factors, including soil moisture, rainfall, temperature, air pressure and wind [5]. These conditions can change from one day to the next, and therefore the measured radon activity concentrations are only momentary values. This is illustrated in Figure 6, which shows the relationship between the radon activity concentration and the ambient air temperature during an overnight measurement (Hörzenschlag, 07/09/2017-08/09/2017). Nevertheless, the momentary values can give a rough indication of the radon and thoron activity concentrations in the area.

Figure 6: Relationship between radon activity concentration and ambient temperature (Hörzenschlag, 07/09/2017-08/09/2017).

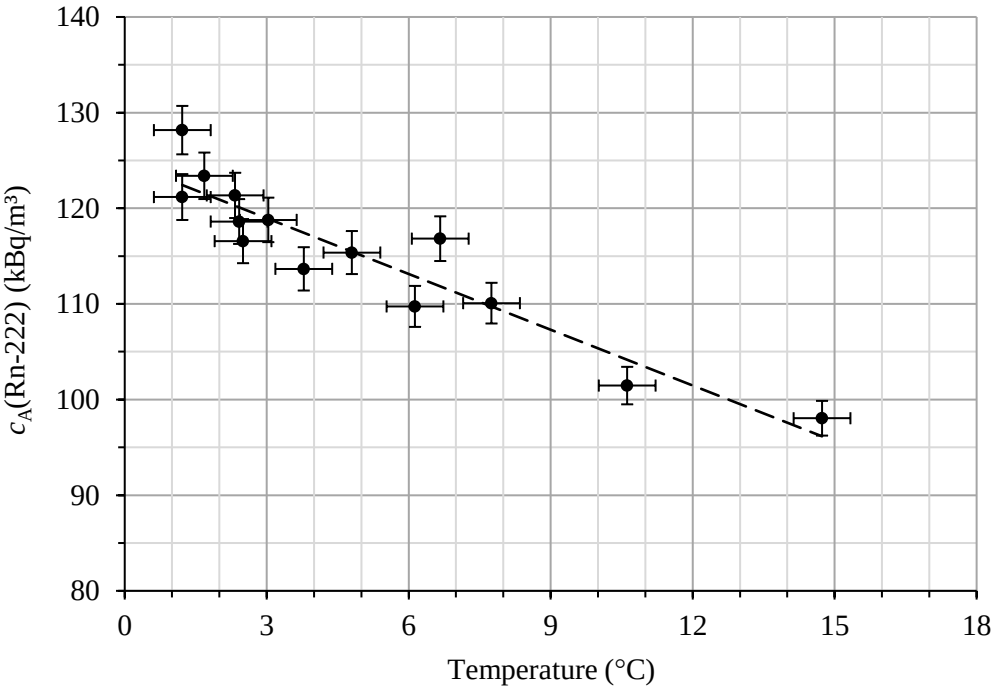


Figure 7 shows the average radon and thoron activity concentrations at each of the measurement locations, as well as the average soil gas permeabilities. The radon activity concentrations have been corrected to a depth of 1 m.

The average soil gas permeabilities can be used with the maximum radon activity concentration to determine the geogenic radon potential of an area. This can be done by using an empirical ranking classification that divides the radon activity concentration and gas permeability into three classes each, to obtain a radon potential class between 1 and 6 [7]. The high soil gas permeabilities and medium to high radon activity concentrations shown in Figure 6 lead to a radon potential class of 4 to 5.

3.2 Radionuclides in the soil

Figure 8 shows the activity concentrations of Ra-226, Th-228, U-238 and K-40 in the soil samples measured by low-level gamma spectrometry.

Figure 7: Average radon and thoron activity concentrations and soil permeabilities (25/07/2017-07/09/2017).

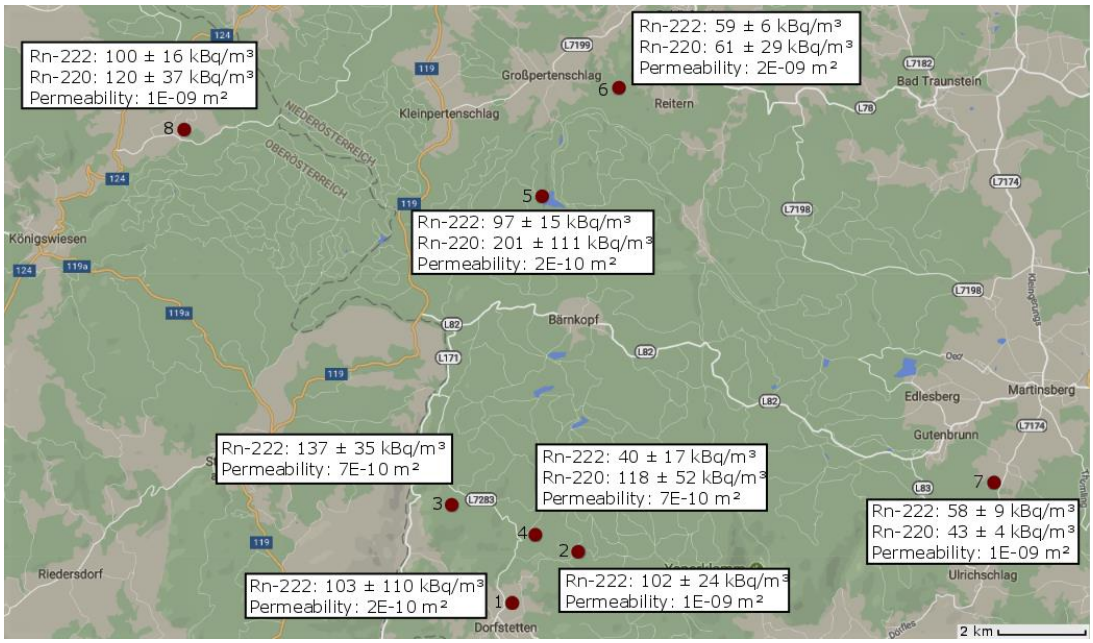
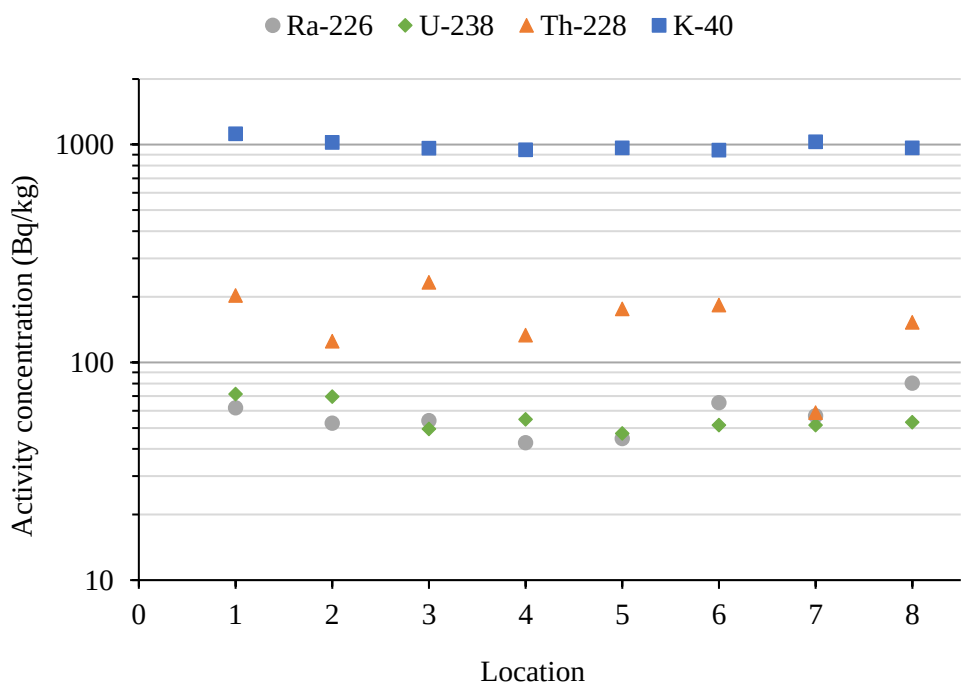


Figure 8: Activity concentrations of Ra-226, Th-228, U-238 and K-40 in the soil samples (25/07/2017-07/09/2017).

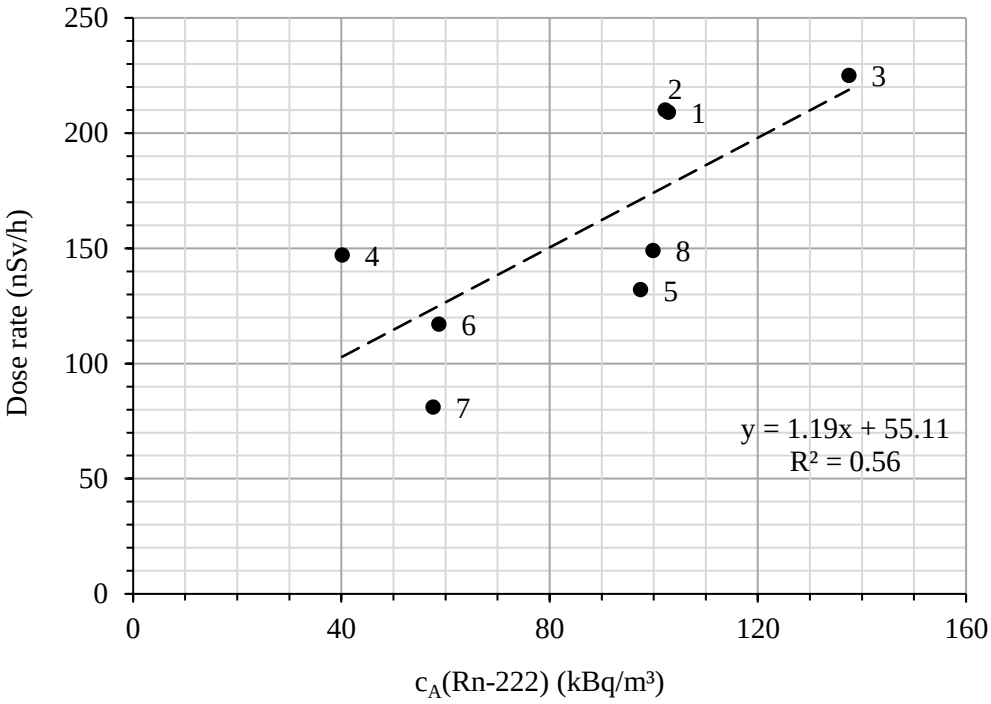


3.3 Dose rate measurements

The detected gamma dose rates lie between 81 nSv/h and 225 nSv/h. Figure 9 shows the relationship between the radon activity concentration in soil gas and the gamma dose rate. The gamma activity of radon progeny in the ground contributes to the dose rate above ground, which is why the dose rate increases with increased radon activity concentration. Information about the correlation between these two quantities is very useful, as gamma dose rate measurements are faster and easier to carry out than

soil gas measurements. Dose rate measurements could therefore be used as an alternative way to examine the radon characteristics of an area [11].

Figure 9: Relationship between Rn-222 activity concentration and gamma dose rate (25/07/2017-07/09/2017). The Rn-222 activity concentrations have been corrected to a depth of 1 m. The data point labels indicate the measurement locations (see Figure 7).



3.4 Radon in water

The results of measurements of the radon content in water samples are summarised in Table 1.

Table 1: Radon activity concentrations of water samples.

Sample name	Sampling date	c_A(Rn-222) (Bq/L)
Alter Brunnen 1	07.09.2017 18:15	60 ± 9
Alter Brunnen 2	07.09.2017 18:18	59 ± 9
Unterer (neuer) Brunnen 1	07.09.2017 18:25	87 ± 10
Unterer (neuer) Brunnen 2	07.09.2017 18:27	98 ± 24
Bärnkopf Wasserwerk Dreiblöchelberg Quelle	06.09.2017 14:35	118 ± 11
Bärnkopf Wasserwerk Q2	06.09.2017 14:40	45 ± 19
Bärnkopf Wasserwerk Q3 2	06.09.2017 14:30	42 ± 8
Bad Zell Radonarium	08.09.2017 10:50	102 ± 17

Alter Brunnen (old well) and Unterer (neuer) Brunnen (lower (new) well) refer to the water supply of the house in Königswiesen in which the indoor radon and thoron measurements were carried out. Dreiblöchelberg Quelle, Q2 and Q3 are the names of the sources from the waterworks in Bärnkopf. Bad Zell Radonarium refers to the spa in Bad Zell, where the water sample was taken from a water fountain in the lobby.

4 CONCLUSION

Soil gas measurements were carried out in 8 different locations with increased radon potential (radon priority areas, RPAs) in Austria. In addition to measuring the activity concentrations of radon and thoron, the soil gas permeability, the gamma dose rate, and the activity concentrations of radionuclides in the soil (the uranium-radium and thorium decay chains) were determined. Furthermore, measurements of the radon activity concentration in water samples were carried out. Activity concentrations of radon and thoron of more than 100 kBq/m³ were detected in soil gas. The radon content in the water samples was found to be between 42 Bq/L to 118 Bq/L. An overnight soil gas measurement showed an inverse relationship between the radon activity concentration and ambient temperature. A correlation was also found between the radon activity concentration in soil gas and the detected gamma dose rates, indicating that dose rate measurements could be used to estimate the radon potential of an area.
The results of the project help to plan and carry out future radioecological research in radon priority areas for the implementation of radon and thoron mitigation and precaution programmes.

5 ACKNOWLEDGEMENTS

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The gamma spectrometry measurements were carried out in cooperation with Dr. Michael Stietka at the University of Natural Resources and Life Sciences (BOKU).

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Regulation

IAEA and EU Basic Safety Standards – Analysis of Differences

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Abstract. The harmonization of basic safety standards related to radiation protection enabling equal health standards has been in the focus of international organizations for decades. A list of reasons for that can be very long, e.g. harmonization is needed to enable among other movement of people and goods such as food, feedings stuff and radiation sources and related equipment. The EU as well as the IAEA have been publishing basic safety standards following ICRP recommendations for decades. In 2013 the EU published the Council Directive 2013/59/Euratom. On the other hand, in 2014 the IAEA published International Basic Safety Standards, i.e. IAEA SS GSR Part 3. Even through both texts have been drafted in the same period and are based on the ICRP 103, i.e. introducing three exposure situations, the analysis of differences identifies main areas where full harmonisation between both texts was not achieved. In particular, differences lie in introducing a control of exposures related to natural radiation sources, e.g. exposures related to radon and thoron, building materials and exposures in NORM industries. In addition, some conceptual differences are identified, e.g. optimisation principle is somehow enlarged in the EU document while security is incorporated in the IAEA document and not in the EU document. The analysis of about ten basic differences between both documents focusing in particular on consequences of the differences shall facilitate communication among EU MSs and other countries. Such communication is particularly important considering world market of equipment producing ionising radiation, world market of materials and control of exposure of itinerant workers in all exposure situations.

KEYWORDS: basic safety standards, directive, radiation protection, European Union, IAEA, ICRP, Council Directive 2013/59/Euratom, GSR Part 3, radiation, sources, planned exposure situation, emergency exposure situation, existing exposure situation

1. INTRODUCTION

New scientific achievements and experiences regarding implementation of radiation protection guide countries and international organisations to update safety standards for protection against the danger arising from exposure to ionising radiation. Updating of such standards is a huge project requiring years of study of scientific data and mentioned experiences. Namely, international organisations are publishing new documents with a period of about 15 years. Moreover, many different organisations are involved in providing data to upgrade the standards, e.g. UNSCEAR, ICRP, IRPA, WHO, ERA, HERCA and EURADOS, just to mention few of them.

In 2013 the EU published the new basic safety standards “for protection against the dangerous arising from exposure to ionising radiation” as given in the text of the Council Directive 2013/59/Euratom (EU BSS) [1]. The EU Member States (MSs) are required to implement the document which has about 100 Articles, 19 Annexes and more than 1000 technical details. The document repeals five directives and is one of only few directives related to radiation and nuclear safety. Details are given elsewhere, e.g. in [2].

On the other hand, in 2014 the IAEA published International Basic Safety Standards regarding radiation protection and safety of radiation sources, i.e. IAEA General Safety Requirements Part 3 (IAEA BSS) [3] to be used in the IAEA MSs. Around 160 countries contributed in drafting the document. The document contains around 400 requirements, 52 among them are so-called overarching requirements. It also contains numerous technical specifications applicable for calculations, majority of them given in Schedules I- IV.

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Both documents, i.e. EU BSS and IAEA BSS, are largely harmonised as they are based on the same scientific data. This harmonisation was achieved by strong communication among authors of both texts in the period of preparation of both documents. However, it must be noted that some differences still exist in standards mentioned which can, if not taking into account, pose a challenge in everyday communication between an IAEA MS, which is not a member of the EU, and an EU MS. Such challenges span from different technical standards regarding facilities, equipment or calculation models to differences in emergency preparedness and manging itinerant workers.

Some of the differences originating in the fact that EU BSS is based on EURATOM Treaty [4] which defines a scope of every document based on this Treaty. In addition, in the EU some other EU legal acts, i.e. not the EU BSS, are providing requirements regarding specific area, such as standards related to drinking water. On the other hand, the IAEA BSS is only one of the seven IAEA General Safety Requirements (IAEA GSR), e.g. IAEA GSR Part 4 tackles specifically safety assessment. Details regarding legal background of both documents are given elsewhere, e.g. in [5], where the very first analysis of differences between both documents is published.

2. DIFFERENCES BETWEEN EU BSS AND IAEA BSS

The very outstanding difference between the EU BSS and the IAEA BSS can be noted looking only to the organisation of the documents. Namely, both documents introduced three exposure situations, i.e. *planned*, *emergency* and *existing exposure situations*. While IAEA BSS requirements are based on these situations and within each situation requirements related to exposure to workers, member of the public and patients as appropriate are given, this is not the case in the EU BSS. The structure of the EU BSS is quite different. Namely after first five somehow general chapters three chapters follows, i.e.:

- *Chapter VI: Occupational Exposures,*
- *Chapter VII: Medical Exposures,*
- *Chapter VIII: Public Exposures,*

and the requirements related to three exposure situations are given within each chapter. Different organisation of both texts somehow inhibits straightforward analysis of differences.

A list of differences between requirements given in both texts is actually very long when considering level of details given in both huge texts. A list might be started by pointing out already one of the basic principles of radiation protection which is not equal in both documents. Namely optimisation as a general principle of radiation protection is not the same. In the EU BSS the Art. 5 (b) gives: "*Optimisation: Radiation protection of individuals subject to public or occupational exposure shall be optimised with the when aim of keeping the magnitude of individual doses, the likelihood of exposure and the number of individuals exposed as low as reasonably achievable considering the current state of technical knowledge and economic and societal factors…*" The IAEA BSS does not mention "*current state of technical knowledge*" as one of the component when judging if optimisation has been implemented in line with requirements. In everyday regulatory control or implementation of optimisation at a facility such basic conceptual differences should be noted when appropriate.

Another very important difference is related to the scope of both texts. Namely, the scope of the IAEA BSS includes provisions for security. Although security related publications are issued in the *IAEA Nuclear Security Series,* holistic approach to safety in security is taken into account in the IAEA BSS. Namely, security and safety measures should be integrated. The IAEA BSS do not deal with security measures but require: »*arrangements are in place for the interfaces between safety and the security of radioactive sources*" as given in para. 2.27. Security is not tackled in the EU BSS as Euratom Treaty as a legal background for the EU BSS mentioned already above does not provide the background.

In addition, so-called "free market of goods and services" within the EU requires that the level of technical details in some requirements is very high in the EU BSS while such details are not tackled in the IAEA BSS. Three examples are given:

- Art. 89 and Annex XIV of the EU BSS give a detailed *Standard Record Sheet for High-Activity Sealed Sources (HASS)*, i.e. a specific form, which should be prepared by the undertaking and provided to the competent authority to assure safety of such sealed sources and timely exchange of detailed information regarding HASS.
- Arts. 43, 44 and 51 of the EU BSS are giving details related to data systems for individual radiological monitoring related to workers which can be realised either as a network or as a national register. Issuing an individual radiological monitoring document is optional. Such networks or national registers should facilitate tracking of worker's doses within the EU with a specific emphasis to control doses of so-called "outside workers".
- In line with the EU legislation the EU BSS address communication among MSs with the EC as well as communication between MSs. Such provisions are given for example in Art. 79 regarding recognition requirements for services and experts which should be communicated to the EC. Another example is communication related to so-called consumer products as MSs shall inform the point of contact for the competent authorities of other MSs when receiving a request regarding production or import of new type or class of a consumer product. Details are given in the EU BSS.

Some noted differences between both texts are based or somehow an arbitrary decision of authors. While EU BSS tackles lessons learn from Fukushima and requires in Art. 65 that "*radioactive contamination liable to extend to the ground beneath the facility*" should be analysed before acceptance into service a facility, such technical detail is not given in the IAEA BSS. On the other hand, the IAEA BSS give in para. 2.37 requirement related to "*ensuring protection and safety in the handling of deceased persons or human remains that are known to contain sealed or unsealed radioactive sources*" while such specific requirement is not given in the EU BSS. Details are given in [5].

The present analysis is focused on differences in areas which have been fully introduced in both texts in details for the first time and might have large impact on the radiation protection framework in the next decade in EU MSs, i.e. areas regarding natural radiation sources. Three areas are analysed:

- radon and thoron exposures
- exposures related to building materials
- exposures related to industry involving naturally-occurring radioactive materials (NORM).

The analysis also considers requirements regarding:

- *planned exposure situation* and dose limits
- *emergency exposure situation* where lessons from the Fukushima accident were taken into account in both texts
- *existing exposure situation*
- use of ICRP documents.

2.1 NATURAL RADIATION SOURCES

2.1.1 Radon and Thoron Exposures

As pointed out in EU BSS preamble 22: "Recent epidemiological findings from residential studies demonstrate a statistically significant increase of lung cancer risk from prolonged exposure to indoor radon at levels of the order of 100 Bq m^{-3} . Radon exposure is responsible for about 3-14% of all lung cancers [6], i.e. the total number of deaths in EU MSs is about 20 000 per year. These facts

require further attention of the UNSCEAR and other organisations studying effect of radon on health. Details are given in [7]. The EU BSS and the IAEA BSS are addressing this issue by introducing reference level for the indoor annual average activity concentration in air. But while the reference level for the annual average activity concentration in air at workplaces shall not be higher than 300 Bqm^{-3}, unless it is warranted by national prevailing circumstances as required in the EU BSS, such reference level is 1000 Bqm^{-3} in the IAEA BSS. The maximum reference level to be applied in dwellings is 300 Bqm^{-3} in both documents.

While the EU BSS is tackling also thoron, no requirements related to thoron are given in the IAEA BSS. Namely, the indoor exposure to radon and thoron in workplaces, dwellings and in other buildings is put in the indicative list of the *existing exposure situation* in Annex XVII of the EU BSS.

2.1.2 Exposures related to Building Materials

Once building materials enter EU market so-called free movement of goods in the EU MSs applies. Some types of building materials can cause exposures which should be under regulatory control. Namely, indoor gamma radiation emitted by such building materials might cause exposures which cannot be disregarded. The EU BSS addresses these issues in Art. 75 by:

- setting the reference level for external radiation due to building materials of 1 mSv per year in addition to outdoor external exposure
- providing an indicative list of materials to be used for identification of building materials which might pose annual dose above a reference level, such as alum-shale, lava, fly ash and residues from steel production
- requiring determination of activity concentration of Ra-226, Th-232 and K-40 in such building materials before putting them on the EU market
- requiring the implementation of appropriate measures in building codes or restrictions on the use.

These requirements might largely influence development of specific protocols regarding sampling of building materials, development of measurement protocols and development of specific laboratories in EU MSs. They might also influence reporting protocols as competent authorities should base their measures and restrictions on measurements.

The IAEA BSS does not address this issue in such details. Namely, the exposure due to construction materials is considered as *existing exposure situation* implementing the maximum reference level of annual exposure to the representative person of 1 mSv as given in para. 5.22 [3]. Different approach to building materials in both texts shall be carefully analysed by EU MSs and by undertakings in EU MSs importing building materials which might be a subject of regulatory control in EU MSs.

2.1.3 Exposures related to Industry involving NORM

The discussion how to incorporate exposures related to industry involving NORM in radiation protection system took years. The question was not related if radiation protection measures are necessary but to what extent they should be introduced, i.e. managing such industry within a framework of a *planned exposure situation* or of an *existing exposure situation*.

The EU BSS is trying to fully integrate protection against natural radiation sources within the overall requirements. As a result, industrial sectors involving NORM shall be controlled within the scope of requirements applicable for *planned exposure situations*, i.e. the sectors are controlled as a practice in line with the Art. 23 of the EU BSS. A list of industrial sectors is given in the Annex VI of the EU BSS. The list includes: production of thorium compounds and manufacture of thorium-containing products, oil and gas production, ground water filtration facilities and mining of ores other than uranium ore. When implementing EU BSS MSs are identifying the industry involving NORM and

managing such industries as practices. This might be challenging due to large diversity of industrial facilities from the list mentioned.

According to the para. 3.4. of the IAEA BSS, the exposure due to natural sources is, in general, considered an *existing exposure situation*. However, requirements for *planned exposure situations* apply if exposures are a result of involvement of materials with enhanced activity concentration, e.g. in industrial facility. Namely, if activity concentration of any radionuclide from the uranium decay chain or the thorium decay chain is greater than 1 Bq/g or the activity concentration of K-40 is greater than 10 Bq/g. Such requirements apply also for public exposure due to discharges or due to the management of radioactive waste from such facilities or activities. But no further detail standards are given in the IAEA BSS.

2.2 PLANNED EXPOSURE SITUATION AND DOSE LIMITS

Dose limits are always in the focus of radiation protection as they can be easily understood and introduced in the legislation. Two differences in application of dose limits in both BSS are noted, one related to apprentices and students and other to the public.

- While the limit of an equivalent dose to the lens of the eye for apprentices and students aged between 16 to 18 years is 15 mSv in a year in the EU BSS, this limit is 20 mSv in the IAEA BSS.
- The annual limit of the effective dose for the public exposure is 1 mSv in both texts, however the IAEA BSS has an additional provision in para. III.3. (b) allowing higher value of effective dose in a single year provided that the average effective dose over five consecutive years does not exceed 1 mSv per year. Such higher value is related for example for authorized, justified and planned operational conditions that lead to transitory increases in exposures.

The EU BSS introduces two concepts related to occupational exposures which are not considered in the IAEA BSS:

- For the purpose of monitoring and surveillance of exposures the workers are classified in two categories, i.e. category A and B, as required in Art. 40.
- The concept of the “specially authorised exposure” is introduced in the Art. 52 EU BSS, to be used in exceptional circumstances evaluated case by case, excluding emergencies. Specifically, the competent authority may, where a specific operation requires so, authorise individual occupational exposures of identified workers exceeding the dose limits for occupational exposures, provided that such exposures are limited in time, confined to certain working areas and within the maximum exposure levels defined for the particular case by the competent authority. Only workers of the category A can be a subject of “specially authorised exposure”.

According to the EU BSS the exposure of air crew to cosmic radiation should be managed as a *planned exposure situation*. However, not all requirements might be applicable as given in Art. 35 of the EU BSS. The arrangements of the workplaces include several requirements. One of them is organisation of working schedules in order to reduce the dosed of the highly exposed crew. MSs shall also ensure that the exposure of spacecraft crew above the dose limits is managed as a “specially authorised exposure” as given in the Art. 52. On the other hand, considering the IAEA BSS para. 5.30.-5.33 exposure of aircrew and space crew is considered to be *existing exposure situation* in the IAEA BSS, i.e. no dose limits apply but reference levels should be used.

2.3 EMERGENCY EXPOSURE SITUATION

Authors of both texts introduce *emergency exposure situation* in a largely harmonized way. However, some differences between EU BSS and IAEA BSS regarding management of *emergency exposure situation* including emergency preparedness are noted. The EU BSS is addressing exposures in *emergency exposure situation* by introducing so-called "accidental exposure" and "emergency exposure". An "accidental exposure" is any exposure related to an accident except exposure of emergency workers. No such concept exists in the IAEA BSS. According to the EU BSS Art. 42 the undertaking is responsible for the assessment of the relevant doses and their distribution in the body related to accidental exposure. For example, accidental exposure due to an accident in industrial radiography shall be assessed by the undertaking. In the IAEA BSS so-called accidental exposure is related only to medical exposures. In addition to this conceptual difference three others are identified.

- In the EU BSS the reference level related to external radiation for emergency occupational exposure shall be set, in general below an effective dose of 100 mSv as given in Art. 53. In exceptional situations, in order to save life, prevent severe radiation-induced health effects, or prevent the development of catastrophic conditions, a reference level for an effective dose from external radiation of emergency workers may be set above 100 mSv, but not exceeding 500 mSv. Such workers shall be volunteers as noted in Art. 53. On the other hand, in the IAEA BSS requires that the workers are volunteers only in cases when guidance level of 500 mSv is exceeded as stated in Table IV.2 of the IAEA BSS.
- The EU BSS is addressing two different types of informing of members of the public likely or actually affected in the event of an emergency. The EU BSS actually repeats the requirements from one of the legal acts prepared few years after the Chernobyl accident. Specifically, Council Directive 89/618/Euratom from 1989 given in [8] was repealed by the EU BSS but its requirements were fully incorporated in the EU BSS. In line with the Art. 70 and 71 there MSs shall assure not only information to the members of the public actually affected in the event of an emergency but also so-called "prior information" shall be given to members of the public likely to be affected. Moreover, such information shall not be linked to requests and shall be updated and permanently available. A list of data required is given in the Annex XII. No such requirement is given in the IAEA BSS.
- The EU BSS does not provide generic criteria to be used for protection actions in case of an *emergency exposure situation* in EU MS, e.g. a MS can use its own criteria for urgent decontamination, immediate decorporation and registration for long term medical follow-up. A lack of harmonisation might be a huge challenge in a real *emergency exposure situation.* On a contrary, the IAEA BSS gives a table, i.e. Table IV.1, to be used to avoid or to minimize severe deterministic effect in case of an emergency.

2.4 EXISTING EXPOSURE SITUATION

The concept of *existing exposure situation* is relatively new and there are practically no regulatory experiences with it. The situations to be treated as *existing exposure situations* in both texts are not always the same as already mentioned few times above. The EU BSS is introducing in Annex XVII so-called indicative list of *existing exposure situations* as referred to in Art. 100 which is not full in line with Req. 51 and para. 5.22 of the IAEA BSS. Namely, food, feed and drinking water are excluded from the list in line with the mentioned Annex. Details are given there.

2.5 USE OF ICRP DOCUMENTS

The authors of the EU BSS also address updating of ICRP documents to be used in calculation of doses other than those caused by external exposures. The authors of the EU BSS were aware of the fact that the preparation of new BSS document requires years. The EU MSs shall be in line with the development of the ICRP methodology when using values and relationship given in ICRP 116 [9] and

ICRP 119 [10]. The EU BSS require either EU BSS methodology or a MS approval of values to be used in methodology as given in the Article 4 (96). In the ICRP 103 [11], a new methodology was introduced by the ICRP to calculate doses based on the latest knowledge on radiation risks, and this should, where possible, be considered by MSs. No such provision is given in the IAEA BSS.

3. CONCLUSION

The EU BSS and the IAEA BSS incorporated three exposure situations from the ICRP 103 in largely harmonised way. However, the detailed analysis reveals numerous differences which are based among others on legal background of both documents. The difference can be noted in applications of safety measures in all exposures situation, namely in *planned*, *emergency* and *existing exposure situations*. These differences shall be taken into account in cooperation between those IAEA MSs which also implement the EURATOM legislation and another IAEA MSs which did not implement the EU BSS.

Identification of differences in regulatory approaches can facilitate efficient cooperation among designers, providers and suppliers of radiation sources and associated equipment as well as others responsible for radiation safety and security, e.g. operators and waste management facilities, qualified experts and technical support organisations. Timely identification of differences can be a fruitful starting point when sharing experiences among MSs of both organisations.

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Integration of Risks from Multiple Hazards into a Holistic ALARA / ALARP demonstration Radiation Protection Instrumentation: IEC International Standards for Performance Requirements

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Abstract. The principle of As Low As Reasonable Achievable (ALARA) stems from the field of Radiological Protection. In the UK, this principle has been incorporated into the Health and Safety at Work Act 1974 and rather than applying solely to radiological hazards, applies to all hazards in totality. Given the current methods for assessing hazards are somewhat isolated, in that one hazard is assessed independently of another, it can be challenging to ensure a truly holistic view of the risks, and demonstrate they have been reduced to ALARA or As Low As Is Reasonably Practicable (ALARP) as required in the UK Regulatory Regime. The following paper presents a summary of a presentation and paper presented at a Workshop organised on behalf of the International Radiation Protection Association (IRPA) by the French Society for Radiation Protection (SFRP) in Paris, 23-24 February 2017. The paper / presentation proposed a framework for the integrated assessment of risks from multiple hazards. In addition, it presented an overview of some of the key challenges that may be encountered when producing a holistic ALARA demonstration.

KEYWORDS: Risk, ALARA, ALARP, Integrated Risk Assessment

1. INTRODUCTION

The "optimization" principle is at the core of radiation protection. It is often referred to as ALARA, short for keeping all doses "As Low As Reasonably Achievable". In the UK context, the word "Achievable" is replaced by "Practicable", to give ALARP; to be consistent with UK case law. Whilst there are some subtle differences, they are in practice interchangeable terms.

In some cases, particularly in the nuclear sector, radiological risks are perceived to be the dominant hazard. However, the vast majority of situations involve a range of hazards. There is a natural tendency to assess radiological and other hazards, such as chemical and manual handling, independently. This can result in a lack of proportionality in the treatment of each hazard or even the introduction of new hazards as a result of mitigating another hazard.

In practice, taking a balanced view of a range of hazards and risks is something that operational safety staff do every day. What is often missed though is recording the basis of decision making, so that one can learn from non-optimal situations. As the complexity of situations increases it becomes more and more important to have clear, systematic and integrated processes for making assessments and judgements; with demonstrable underpinning.

A framework has been developed built on the work that has been undertaken within the UK Nuclear Industry, where increasing emphasis has been placed on ensuring a balanced view of all hazards. This was presented at a Workshop organised on behalf of the International Radiation Protection Association (IRPA) by the French Society for Radiation Protection (SFRP) in Paris, 23-24 February 2017. Further details of the framework can be found in [1] and a summary presented in this paper.

2. BACKGROUND

In the UK, there is legal requirement to reduce the risks from all hazards to ALARP. Despite this each hazard is assessed somewhat in isolation and independently of one another, making it challenging to ensure a truly holistic view is taken of all the risks.

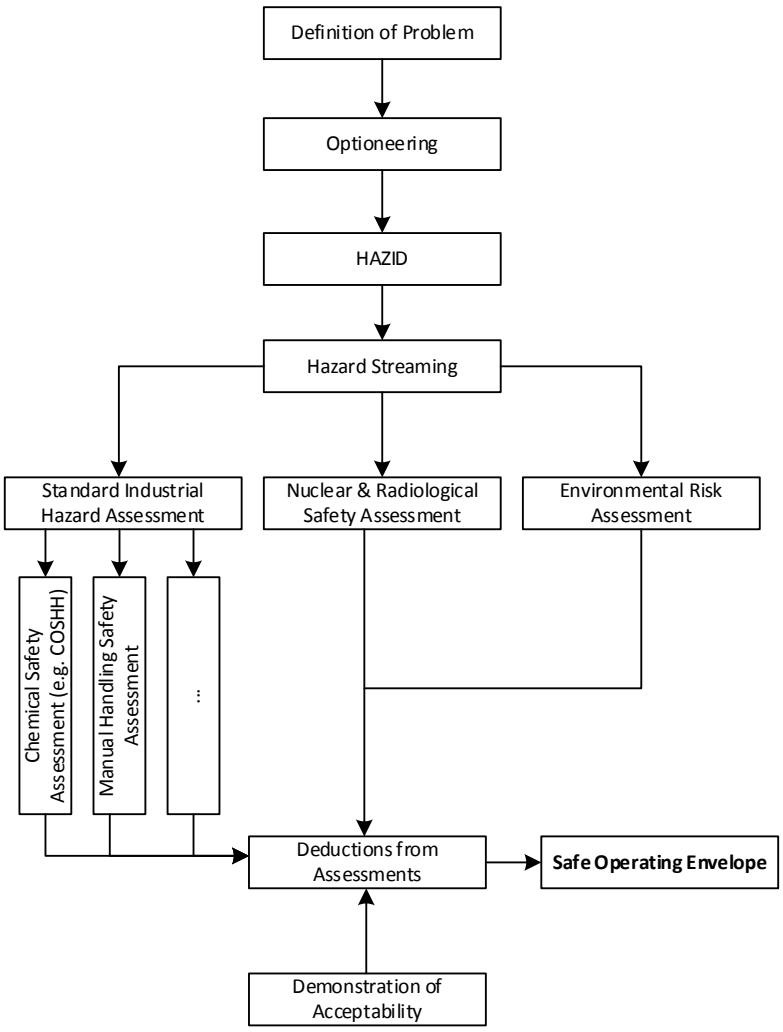
In the UK, high hazard industries, such as the Nuclear Industry, have invested significant effort in developing approaches for the integration of risks from multiple hazards. This is due to the high level of regulation and associated regulatory regime. Whereas in the case of the lower hazard industries such as the medical sector or small users of radioactive materials, such as university laboratories, are less strictly monitored by the regulators, there has been less of a driver to ensure an “all risk” approach.

Taking on board the lessons from several Nuclear Site Licensees, in the production of a Nuclear and Non-Nuclear Safety Cases (suite of documentation that demonstrates how the operator has reduced the risks from all hazards to ALARP), along with guidance from the UK Regulators, a framework is proposed for the integration of risks from multiple hazards.

3. FRAMEWORK

Figure 1 shows an overview of the proposed framework. The framework is presented at a relatively high level, where the level of assessment can be tailored to the level of risk posed by the sector, it is hoped it provides a learning opportunity for a variety of industries not just high hazard sectors. A summary of the key points of the framework is defined in subsequent sections.

Figure 1: Framework for the Integration of Risks from Multiple Hazards into a Holistic ALARA / ALARP demonstration



Problem Definition and Optioneering

The first few steps in the process lay the foundation to the ALARP argument. First off there must be a clear definition of the problem. After defining the problem, the next step is to carry out an Optioneering Assessment to identify possible solutions to the problem and select a preferred option or options. The Optioneering Assessment should be accurately recorded and documented as this provides an auditable trail that underpins the justification and supports the ALARP argument.

Hazard Identification (HAZID)

Once the preferred option(s) has been determined the next step requires a systematic method of hazard identification. This provides a means of testing the preferred option(s). The identified hazards are then subsequently allocated to an appropriate assessment stream.

Hazard Assessment

Regardless of the Hazard Type whether Nuclear / Radiological, Industrial or Environmental in nature a proportionate approach should be undertaken to assessing the hazard. It should be noted that in certain cases controls put in place to limit or reduce the risk associated with one hazard type, may also be used to limit or reduce the risk associated with another hazard type (e.g. chemical and radiological contamination hazards). In such cases, it is important to make sure any claims placed on the control are recorded in both hazard assessments.

Deductions from Assessment

The output of the various assessments should be reviewed in combination, to ensure that there are no conflicts, for instance any controls or mitigations put in place for one hazard type have not created any new hazards, or impacted on any of the other hazard assessments. These deductions are then used to define the Safe Operating Envelope (SOE).

Demonstration of Acceptability

The output of the Hazard Assessments will result in claims being placed on Engineered Controls and Managerial Controls, for instance, the effectiveness of a radiological shield. These claims need to be substantiated to demonstrate they can be met.

4. CONCLUSION

A proposed framework has been described in this paper for the integration of different risks as part of a holistic ALARA / ALARP demonstration. The framework is not without its challenges when trying to ensure a balanced approach to the assessment of the risks from multiple hazards, and noting the variation in the approaches to assessing risks in different industries. Further guidance on the implementation of the framework is presented in [1] this includes details of some of the key challenges that may be encountered when producing a holistic ALARA demonstration and how to overcome them.

5. ACKNOWLEDGEMENTS

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Radiation Protection Instrumentation: IEC International Standards for Performance Requirements

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Abstract. This paper provides information on the international standards developed by IEC Subcommittee (SC) 45B "Radiation Protection Instrumentation". It also provides information about IEC, Technical Committee 45 "Nuclear Instrumentation", SC45A and SC45B, membership, participation and collaboration with partners like ISO, ICRP, ICRU, IAEA. The paper lists all SC45B working groups, their scope and the latest published and under development international standards. The IEC standards serve as basis for national standardization, as references when drafting international tenders and contracts, and for conformity evaluation of instrumentation. IEC standards are considered by CENELEC for adoption as European standards. All active IEC SC 45B standards are listed in the reference.

KEYWORDS: *IEC; standards; radiation protection; instrumentation.*

1. INTRODUCTION

The International Electrotechnical Commission (IEC) is the world's leading organization that prepares and publishes globally relevant International Standards for all electric and electronic devices and systems. It brings together 170 countries (84 Member and 86 Affiliates countries), representing 98% of the world population and 96% of world energy generation. Close to 20 000 experts cooperate on the global IEC platform. The IEC also supports all forms of conformity assessment and administers four Conformity Assessment Systems.

2. INTERNATIONAL ELECTROTECHNICAL COMMISSION (IEC)

The IEC was founded in 1906 in London, UK, and ever since has been giving the very best global standards to the world's electrotechnical industries. IEC members are National Committees (NCs) and there can only be one per country. Individuals participate in the IEC's work through the National Committees. The National Committees (NCs) represent the interests in their country, notably companies and businesses, industry associations, educational bodies, governmental and regulatory bodies and consumers. By participating in the creation of a standard, a NC can be sure that the interests of its country have been taken into account.

The IEC standards serve as basis for national standardization, as references when drafting international tenders and contracts, and for conformity evaluation of instrumentation. IEC standards are considered by CENELEC for adoption as European standards.

2.1 Technical Committees and SubCommittees

IEC Technical Committees (TC) and SubCommittees (SC) prepare technical documents on specific subjects within their respective scopes. Experts carry out the standardization work in the IEC in TCs and SCs, in working groups (WG), project (PT) and maintenance teams (MT). There are currently 207 TC and SC and 567 WG. The working groups are composed of representatives from all over the world who are experts in their own field and who are members of research and testing laboratories, regulatory agencies, academia, manufacturers, and user organizations. IEC also establishes project teams (PT) to prepare

individual standards that do not fall within the scope of an existing TC or SC. Project teams are disbanded once a standard has been published. Currently there are 270 Project Teams.

2.2 IEC membership

There are two levels of IEC membership:
- Full members - NC has access to all technical and managerial activities and functions, at all levels of the IEC, including voting rights in Council.
- Associate members - NC has full access to all working documents but limited voting rights in the technical work and no eligibility to managerial functions within the IEC.

While individuals participate in the work of the IEC, they are not members. The IEC NCs (National Committees) are the members. Individuals can participate in the standardization work of the IEC as experts or delegates.

2.3 Experts, delegates and participation

Experts are individuals with specialist knowledge in a particular technical field. Each NC (National Committee) participating in a technical committee's work can appoint experts to take part in specific technical work through working groups, project teams or maintenance teams. Experts participate in IEC technical work in a personal capacity and do not represent their company / organization or NC. Category A liaison organizations may also appoint experts to working groups and project teams.

Delegates are representatives of their NC at a TC (Technical Committee) or SC (Subcommittee) meeting and should be fully briefed by their NC before attending a meeting.

If you are in a country that already participates in the work of the IEC contact your NC directly.
If you are in a country that does not yet have an NC, contact IEC Central Office to investigate how you can get involved.

The IEC also runs a programme for newly-industrializing countries around the world through the Affiliate Country Programme. The Programme offers such countries a form of participation in the IEC without the financial burden of actual membership, making full use of IEC 100% electronic environment.

2.4 IEC publications

IEC TCs/SCs (Technical Committees and Subcommittees) develop International Standards and other types of publications for a specific area of electrotechnology. These publications fall into two broad categories:
- Normative publications reflect agreements on the technical description of the characteristics to be fulfilled by the product, system, service or object in question.
- Informative publications provide background information such as implementation procedures or guidelines

International Standards and other publications are the result of full or limited international consensus among the IEC's members (National Committees). Any member of the IEC may participate in the preparatory work of an International Standard, and any liaison organization i.e. international, broad regional organizations, consortia and fora may also participate in this preparation.

IEC publications are bilingual in English and French, while the Russian Federation National Committee prepares Russian-language editions. Certain publications have also been translated into Spanish.

2.5 Cooperation and partners

The IEC also cooperates with several international, regional and national partners to produce joint publications, help promote the importance of standardization and encourage implementation of its international standards around the world and to coordinate and reduce any potential overlaps in work.

The IEC has Cooperation Agreements or Memorandum of Understanding with 32 organizations like ISO, WTO, CENELEC, ITU, IEEE and another 199 Partners like ICRP, ICRU, IAEA, IEA, WHO, EC-JRC. One of the IEC's principal partners is the WTO (World Trade Organization). The IEC works with its sister international standardization organizations, ISO (International Organization for Standardization) and ITU (International Telecommunication Union), on a bilateral basis in specific technical areas.

3. TECHICAL COMMITTEE 45 "NUCLEAR INSTRUMENTATION"

IEC Technical Committee 45 addresses international standards development for instrumentation specific to nuclear applications. TC 45 has 18 Participating members and 16 Observing members. TC45 has two subcommittees (SC): SC 45A and SC 45B and 3 WGs and PT "Industrial non-destructive testing equipment – Electron linear accelerator". TC 45 working groups are:

- WG 1 "Terminology"
- WG 9 "Detectors and systems"
- WG 18 "Mobile unmanned automated systems for nuclear and radiological applications"

Working groups numbering in TC 45, SC 45A and SC 45B is not consecutive because some groups in the past have merged or ceased to exist due to inactivity.

4. SUBCOMMITTEE 45A "INSTRUMENTATION, CONTROL and ELECTRICAL POWER SYSTEMS of NUCLEAR FACILITIES"

The scope of IEC SC 45A is to prepare standards applicable to the electronic and electrical functions and associated systems and equipment used in nuclear energy generation facilities (nuclear power plants, fuel handling and processing plants, interim and final repositories for spent fuel and nuclear waste) to improve the efficiency, safety and security of nuclear energy generation. The core domain is the instrumentation, control and electrical power systems important to safety in nuclear energy generation facilities. The core domain includes the radiation monitoring instrumentation used for monitoring, control and safety actuation functions and instrumentation, control and electrical power systems used in nuclear energy generation facilities to manage and control nuclear materials in the frame of international agreements and to safeguard nuclear material.

SC 45A has 8 working groups and 21 Participating members and 12 Observing members. The working groups are:

- WG A2 "Sensors and measurement techniques"
- WG A3 "Instrumentation and control systems: architecture and system specific aspects"
- WG A5 "Special process measurement and radiation monitoring"
- WG A7 "Functional and safety fundamentals of instrumentation, control and electrical power systems"
- WG A8 "Control rooms"
- WG A9 "System performance and robustness toward external stress"
- WG A10 "Ageing management of instrumentation, control and electrical power systems in NPP"
- WG A11 "Electrical power systems: architecture and system specific aspects"

5. SUBCOMMITTEE 45B "RADIATION PROTECTION INSTRUMENTATION"

IEC SubCommittee 45B "Radiation protection instrumentation" covers all the fields of radiation protection instrumentation for measurements under normal and accident conditions of external and internal individual exposure, to workers, the public and in the workplace and environment. SC 45B scope is to prepare standards that address instrumentation used for:

- the measurement of ionizing radiation in the workplace, to the public, and in the environment for radiation protection purposes;
- illicit trafficking detection and identification of radionuclides;
- radiation-based security screening.

SC 45B has 21 Participating countries, 14 Observer countries, 7 working groups (WG) and close to 150 experts. SC 45B working groups are:

WG 5 "Measurements of Environmental Radiation"
The WG 5 scope is to prepare standards on instrumentation to measure radiation and contamination to which members of the public may be exposed to, either from natural radiation or as a result of normal operation or emergency conditions from any nuclear facility. The standards cover measurements of dose rates, contamination levels of both water and ground, and airborne spectrometers. There are 14 active standards issued and maintained by WG 5.

WG 8 "Active pocket and portable dose (rate) meters and monitors and passive dosimetry systems"
WG 8 scope is to prepare standards on active pocket and portable dose (rate) meters and monitors and on passive dosimetry systems for photon, beta, and neutron radiation that are used for the direct or indirect supervision of dose limits (e.g. for persons or controlled areas), for the characterization of workplaces, etc. The standards will specify requirements for the dosemeter, monitor, and, if supplied, for its associated equipment and software, e.g. its readout system. There are 6 active standards issued and maintained by WG 8.

WG 9 "Installed equipment for radiation and activity monitoring in nuclear facilities"
The scope of WG 9 is to prepare standards (a) for installed dose rate meters, warning assemblies and monitors, for X, gamma and neutron radiations, (b) for centralized monitoring systems including data collection and processing for continuous surveillance of radiological parameters (dose rate, activity monitoring, etc.) in nuclear facilities, (c) to liaise with SC 45A in the preparation of standards covering the requirements for installed monitoring systems in nuclear power plants. There are 7 active standards issued and maintained by WG 9.

WG 10 "Radon and radon daughter measuring instruments"
WG 10 scope is to prepare standards on radon and radon daughter measuring instruments for personnel and ambient atmosphere monitoring. There are 4 active standards issued and maintained by WG 10.

WG 15 "Illicit trafficking control instrumentation using spectrometry, personal electronic dosimeter and portable dose rate instrumentation"
The scope of WG 15 is to develop international standards for Illicit trafficking control instrumentation using spectrometry, personal electronic dosimeter and portable dose rate instrumentation. There are 11 active standards issued and maintained by WG 15.

WG 16 "Contamination meters and monitors"
The scope of WG 16 is to prepare international standards for survey meters and monitors which measure surface radioactive contamination and activity and/or activity concentration. The scope does not include instruments specifically designed for environmental monitoring and for monitoring activity concentration

of work place, gas effluent and liquid effluent. There are 8 active standards issued and maintained by WG 16.

WG 17 "Security inspection systems using active interrogation with radiation"
WG 17 scope is to prepare standards for inspection systems that actively employ radiation for security-screening purposes to detect explosives, special nuclear material, and other contraband. The scope includes both technical performance (image quality) standards, as well as radiation safety standards for systems that use ionizing radiation. There are 3 active standards issued and maintained by WG 17 and 3 more standards in the development stage.

Additionally 3 SC 45B project teams (PT) have developed and published standard documents that do not fall into the scope of SC 45B working groups.

6. IEC SC 45B PUBLICATIONS

IEC SC 45B standards specify the consensus based minimum performance requirements for each type of radiation protection instrumentation and the tests and the criteria for compliance with those requirements. The general composition of IEC SC45B standards is:
- scope and applicability of the particular standard in terms of instrument types and usage
- normative references to other standards to be used in conjunction
- terms, definitions and units used in the particular standard
- general requirements for the radiation detection instrument, warm up time, battery, etc.
- marking and communication interface
- reference test sources
- statistical fluctuations
- overload characteristics
- personal protective alarms
- response time
- radiation alarms (gamma, neutron)
- radiation data format for transferring data to external device (computer, network, internet)
- gamma exposure and ambient dose rate equivalent, effective range, dose rate linearity
- energy dependence of the detector response
- variation of the detector response due to the angle of incidence
- alpha, beta and neutron detection
- radionuclide identification of single and multiple radionuclides
- algorithm to evaluate indicated value
- environmental requirements: temperature, humidity, moist, dust, etc.
- mechanical requirements: vibration, microphonics, mechanical shock
- electromagnetic requirements: radiofrequency and magnetic field susceptibility, electrostatic discharge, electromagnetic emissions
- documentation

6.1 Latest IEC SC 45B publications

- IEC 62401 Ed. 2 (2017) Alarming Personal Radiation Devices (PRD) for detection of illicit trafficking of radioactive material
- EC 62327 Ed. 2 (2017) Hand-held instruments for the detection and identification of radionuclides and for the estimation of ambient dose equivalent rate from photon radiation
- IEC 62957-1 (2017) Semi-empirical method for performance evaluation of detection and radionuclide identification

- IEC 61017 (2016) Transportable, mobile or installed equipment to measure photon radiation for environmental monitoring

6.2 Current IEC SC 45B projects in dose (rate) measurements and contamination monitors

- IEC 62387 Ed. 2 Dosimetry systems with integrating passive detectors for individual, workplace
- and environmental monitoring of photon and beta radiation (publication in 2019)
- IEC/TS 63050 Dosemeters for pulsed fields of ionizing radiation (publication in 2018)
- IEC 61322 Ed. 2 Installed dose equivalent rate meters, warning assemblies and monitors for neutrons
- of energy from thermal to 20 MeV (publication in 2019)
- IEC 61563 Ed. 2 Equipment for measuring specific activity of gamma-emitting radionuclides in foodstuffs (publication in 2018)
- IEC 61098 Ed. 3 Installed personnel surface contamination monitoring assemblies (publication in 2020)

6.3 Current IEC SC 45B projects in detection of illicit trafficking of radioactive material

- IEC 62244 Ed. 2 Installed radiation portal monitors (RPMs) for the detection of illicit trafficking of radioactive and nuclear materials (publication in 2019)
- IEC 63121 Vehicle-mounted mobile systems for the detection of illicit trafficking of radioactive materials (publication in 2020)
- IEC 62484 Ed. 2 Spectroscopy-based portal monitors used for the detection and identification of illicit trafficking of radioactive material (publication in 2021)

6.4 Current IEC 45B projects in security inspection systems

- IEC 62945 Evaluating the Image Quality of X-ray Computed Tomography (CT) Security-Screening Systems (publication in 2018)
- IEC 62963 Bottle/can liquid X-ray computed tomography (CT) inspection systems (publication in 2020)
- IEC 63085 System of spectral identification of liquids in transparent and semitransparent containers (publication in 2021)

7. CONCLUSION

IEC international standard for radiation protection instruments establish performance requirements, give examples of acceptable test methods and specify the general characteristics, general testing procedures, radiological, mechanical, electrical and environmental requirements for the instrument.

IEC/SC 45B with its 54 active international standards greatly contributes for the high quality of existing instrumentation for radiation protection purposes. Compliance with such standard requirements provides the manufacturers with internationally acceptable specifications and the instruments users with assurance of the rigorous quality and accuracy of the measurements.

Experts from all countries are welcome to contribute to the IEC work (registration through the national committees). The next plenary SC 45B meeting will be held in April 2019 in Paris (France).

8. ACKNOWLEDGEMENTS

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IEC 60761-3 Ed. 2, Equipment for continuous monitoring of radioactivity in gaseous effluents - Part 3: Specific requirements for radioactive noble gas monitors, 01/2002
IEC 60761-4 Ed. 2, Equipment for continuous monitoring of radioactivity in gaseous effluents - Part 4: Specific requirements for radioactive iodine monitors, 01/2002
IEC 60761-5 Ed. 2, Equipment for continuous monitoring of radioactivity in gaseous effluents - Part 5: Specific requirements for tritium monitors, 01/2002
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Chemical and radiological risk-assessment methodology for soil contamination in Belgium: a comparison

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Abstract. In Belgium, environmental authorities have published detailed guidance on the chemical risk-assessment methodology for contaminated sites. These methodologies address both the risk-assessment to human health as to ecosystems and to groundwater and allow deriving generic as well as site-specific clean-up levels. For assessing the impact on human health of carcinogenic contaminants, a reference value of excess lifetime cancer risk of 10^{-5} is used; if the risk induced by the exposure to the contaminants exceeds this value, the soil contamination is considered to form a substantial threat. Moreover, the measured or predicted concentration of the contaminant in the environment has to be compared with relevant regulatory values, such as drinking water standards.

The clean-up levels are derived on basis of standard exposure scenarios defined for the five following ground-use: natural, agricultural, residential, recreational and industrial. The evaluation of human health risk from soil contaminants is made using the S-Risk model developed by the Flemish environmental institute VITO and used as reference model in all regions of Belgium. A former version of the S-Risk model was used a few years ago by soil contamination experts to assess the chemical risk of a Belgian site contaminated with uranium.

This methodology for assessing chemical risk to human-health is very similar to the methodologies used for assessing radiological risk for contaminated sites and could be used to derive clean-up levels for radioactive contaminants. A comparison between the methodologies for chemical and radiological human risk-assessment is presented. The present study confirms the conclusion of a recent US EPA paper where the consistency of US EPA and UK Environmental Agency methodologies for chemical and radiological risk-assessment of contaminated sites was demonstrated.

KEYWORDS: risk-assessment, contaminated site, mixed contamination, slope-factors.

1. INTRODUCTION

Most, if not all, sites contaminated with radionuclides are also contaminated with non-radioactive pollutants, such as heavy metals. This is in particular true for NORM contaminated sites where in many cases the non-radioactive part of the contamination constitutes the main health risk and impacting factor on the environment. Some substances, such as uranium, present both a radiological and a non-radiological hazard. This entanglement between radiological and non-radiological hazards appeals for a consistent approach both in the risk-assessment as in the decision making process regarding a contaminated site.

Such a consistent approach has already been derived and applied in different countries: in particular the US EPA has made the policy decision that risks from radionuclides exposures at remedial site should be estimated in the same manner as chemical contaminants; in the context of its Superfund program, EPA has defined slope-factors for radionuclides allowing to sum the excess cancer risk of both radioactive and non-radioactive contaminants so as to provide an estimate of the combined risks [1][2]. In the UK as well, the Radioactively Contaminated Land Exposure Assessment Methodology (RCLEA) [4] is based on the original Contaminated Land Exposure Assessment (CLEA) [5] approach that was developed for the assessment of non-radioactive contamination. Other studies (e.g. [6] in Germany) have used the assumptions of non-radiological soil protection regulations to derive reference values for radionuclides in soil.

In Belgium, a standard model has been developed to assess the risk to human health of land contamination. Assumptions and parameters used in this model are comparable to the ones commonly used for radiological dose-assessment, confirming the consistency demonstrated in e.g. UK or US EPA approaches [20].

2. RISK-ASSESSMENT VERSUS DOSE-ASSESSMENT: SLOPE FACTORS AND DOSE CONVERSION FACTORS

In the derivation of clean-up levels for non-radioactive contaminants, a distinction is generally made between threshold effect and non-threshold effect. The latter essentially corresponds to carcinogenic, mutagenic or teratogenic effects, which obviously also applies to radioactive substances. In Belgium, the environmental authorities use for carcinogenic substances a criteria of 10^{-5} for individual excess life-time cancer risk[4]. The life-time cancer risk is calculated by multiplying the dose (intake of the contaminant expressed as a concentration per body weight) or the concentration with respectively a slope factor or a unit risk. Next to the criteria based on excess cancer risk, a second criteria based on compliance with legal limits for concentration of contaminant in the environment is also used in the assessment [9][10]. This is similar to the approach of the US EPA which considers that the remediation of a contaminated site (including radioactive contaminated sites) should achieve a level of risk within the 10^{-4} to 10^{-6} carcinogenic risk range based on the reasonable maximum exposure for an individual [2]. EPA also requests that compliance with so-called ARARs (*Applicable or Relevant and Appropriate Requirements*), such as drinking water standards or radon indoor level is taken into account in the risk-assessment.

In radiation protection, effective dose is generally used to characterize the risk and ICRP has attributed a nominal risk coefficient for cancer of 5.5 10^{-5} per mSv for the whole population [11]. In most cases, assessment of the impact of a site contaminated with radioactive substances will be done on basis of dose-conversion factors for the various nuclides and comparison with a reference level expressed as an effective dose. US EPA on the other hand has derived slope-factors for estimating incremental cancer risks resulting from exposure to radionuclides though inhalation, ingestion and external exposure pathways [3]. Slope factors for radionuclide represent the probability of cancer incidence as a result of a unit exposure to a given radionuclide averaged over a life-time using the linear non-threshold model [2]. This approach allows to assess in a consistent manner the impact of both radioactive and non-radioactive contaminants and to sum the risks of both radionuclide and non-radioactive contaminants with non-threshold effects. However, EPA studies have shown that there is not a simple one-to-one relationship between risks calculated with the effective dose and risks calculated using slope factors for each nuclide. The risks estimated directly from dose tend to be greater than those estimated with slope factors [1].

3. COMPARISON BETWEEN EXPOSURE SCENARIOS AND DEFAULT PARAMETERS

Various international and national guidance and models have been developed to assess the impact of radioactively contaminated land: e.g. in UK [4], Germany [12] or France [13]. As mentioned above, in Belgium, the model S-Risk is used as the standard model for assessing exposure and human health risks from (non-radioactive) contaminants present in soil [8]. This model has been developed by the Flemish environmental institute VITO and incorporates state-of-the-art values for assessment parameters taking into account specific circumstances for Belgium or its different regions (regarding e.g. diet or standard soil profile). Belgian soil protection regulations defines 5 standard soil use and S-Risk has implemented different standard scenarios corresponding to these standard uses. They are summarized in Table 1.

[4] One risk of additional cancer for 100 000 exposed persons on their life-time.

Table 1: Standard soil use and corresponding exposure scenario

Type of soil use		S-Risk default scenario
Type I	Nature area	REC-dayout (day recreation – incl. sport)
Type II	Agriculture	AGR (residence with vegetable garden in agricultural area)
Type III	Residential	RES-veg (residential with vegetable garden) RES (residential with garden) RES-ng (residential without garden)
Type IV	recreational	REC-dayout (day recreation – incl. sport)
		REC-dayin (day recreation indoor sport scenario)
Type V	industrial	IND-l (light industry)
		IND-h (heavy industry with outside activity)

S-Risk incorporates following exposure pathways:

a) *Oral*
- **Ingestion of soil and indoor settled dust;**
- **Intake of vegetables from local production (home-grown);**
- **Intake of meat and milk from local production;**
- **Intake of water (drinking-water or groundwater);**

b) *Dermal*
- Absorption from soil and indoor settled dust;
- Absorption from water during showering and bathing;

c) *Inhalation*
- **Inhalation of outdoor air (gas-phase + particles);**
- **Inhalation of indoor air (gas-phase + particles);**
- Inhalation during showering (gas-phase).

In bold, we have marked the pathways which are also relevant for radiological impact. External exposure pathway is obviously a missing exposure pathway for non-radioactive contaminants but needs to be taken properly into account in radiological impact assessment.

S-Risk incorporates also the Volasoil model [14] which allows calculating indoor air concentration of volatile compound due to vapour intrusion from soil or groundwater into the building. The model considers three different building types: with basement, with crawl space and with slab-on-grade; it takes into account both diffusive and convective transport. It also allows two options for the floor of the building: intact or with gaps and holes. Further work will focus on the applicability of this model to estimate indoor radon concentration from radon concentration in soil-gas or radium activity concentration in soil.

S-Risk, like the models for radiological dose-assessment mentioned above, proposes a set of default values for the parameters used in the assessment. These default values differ from one model to another depending on national circumstances or different expert judgment. A comparison between default values used in [4], [13] , S-Risk and in the RESRAD-OFFSITE model had been performed in [16] : this report showed some discrepancies between parameters such as soil ingestion values, dust concentration or exposure time in different scenario. Table 2 gives an example of this comparison for inadvertent soil ingestion by an adult in a residential scenario with vegetable garden.

Table 2: soil ingestion values for an adult in a residential scenario with vegetable garden

	[13]	[4]	RESRAD	S-Risk [8]
Quantity of ingested soil (g/a)	0.8	22	36.5	28

A similar conclusion has been reached in [7] where default parameters and assumptions of, e.g. EPA PRG calculator, RCLEA and RESRAD-ONSITE have been compared. This underlines the need for carefulness in the choice of any default parameters: it needs to take into account site-specific circumstances (e.g. the diet must correspond to the one of the affected population) and an appropriate

degree of conservativeness. Default values of a specific model should only be used in the assessment context for which this model had been developed. Site-specific values, when available, should be preferred to generic assumptions.

4. THE CASE OF URANIUM

It is well known that the chemical toxicity of uranium is higher than its radiotoxicity. This is why the World Health Organisation recommendation of 30 µg/liter for uranium concentration in drinking water [17] is more restrictive than the derived concentration for U-238 and U-234 as defined in the European directive 2013/51/euratom (respectively 3 and 2.8 Bq/l) [18].

In 2009, a plume of uranium contamination in the groundwater of a Brussels neighbourhood had been fortuitously discovered during a routine control of groundwater parameters. The contamination reached a maximum level of 660 µg/liter. Consequently, a characterization study allowed defining the vertical and horizontal contour of the groundwater contamination plume; limited uranium contamination was also found in some backfilling material. A risk-assessment study [19] was performed according to the rules of regional soil protection regulations; the latter rules do not take into account radiological aspects but the chemical toxicity of uranium had to be addressed in the assessment.

That study used a former version of the S-Risk model (Vlier-Humaan) and calculated the daily intake of uranium due to e.g. ingestion of vegetables. As decision criteria, the study followed the WHO recommendations of that time and used a Tolerable Daily Intake (TDI) for uranium of 0.6 µg/kg of body weight per day[5]. Chemical toxicity of uranium is thus considered as a threshold effect. The way chemical and radiological risks of uranium are assessed is therefore quite different: threshold effect and comparison with a TDI for the chemical toxicity, non-threshold effect and comparison with an excess cancer risk or an effective dose for the radiological toxicity.

The model calculated for the oral exposure pathway (ingestion of vegetable and of soil) a value of 1.27 µg/kg per day for adult and 2.9 µg/kg per day for a child, essentially due to ingestion of vegetables. As the TDI for uranium was exceeded, it concluded to the existence of a significant risk. The risk can however simply be mitigated by preventing consumption of vegetables grown in the contaminated area.

5. CONCLUSIONS

Models used to assess chemical and radiological risks show essential similarities. With the exception of external exposure, the exposure pathways will be the same and some regulators, such as US EPA, already made the policy decision to assess chemical and radiological risks in an integrated scheme and to sum both risks. Models for chemical risk-assessment, such as S-Risk, could be used to perform radiological assessment, either by applying the dose-conversion factors to the calculated values in terms of intake of Becquerel, either – following the EPA approach – in applying nuclide-specific slope factor to the calculation of excess cancer risk.

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Benchmarking Radiation Protection Departments

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Abstract. The staffing level of a radiation protection department can be a matter of debate between management, licensing government and the radiation protection expert. Benchmarking can be an effective tool to assist in this debate. In a recent study, benchmark parameters have been determined for US medical facilities. In this work similar benchmark parameters for Dutch medical and non-medical facilities are determined and compared to the US. The response to the questionnaire used to determine the Dutch benchmark parameters was high, indicating that the result can be considered to be representative for the Dutch facilities.

In the US, the benchmark parameters used to assess the staffing levels are mostly comparable to, or higher than, those in the Netherlands. Valuable information on benchmarking parameters can be deduced from the variation in the answers. The topics that showed low variation overlap to a large extent in the US and Dutch medical facilities. The variation in benchmarking parameters for the Dutch non-medical facilities appears to be somewhat higher than for medical facilities.

The most useful benchmark parameters for both US and Dutch facilities are those related to the number of I-131 therapy patients (internal and external), gamma camera's, dosimetry wearers, CT scanners, linacs and PET/CT scanners and CT scanners. Specific for the Netherlands additional useful benchmark parameters are the number of High Active Shielded Sources and the involvement of the radiation protection department with quality assurance of radiation measurement equipment and the leak-check of radioactive sources.

1 INTRODUCTION

The staffing level of a Radiation Protection Department (RPD) is an important factor in determining how well equipped a RPD is to fulfill its obligations. The staffing level, expressed in Full Time Equivalent (FTE), refers to the number of working hours attributed to the RPD. The required staffing level can be a subject of debate between management, the RPD and licensing authorities. Management has to be convinced to supply the required resources, and authorities must be shown that the staffing level is adequate. As discussed in [1], benchmarking can be a valuable tool in these discussions. Data for a benchmark study is obtained from organizations, similar to the organization that has to be assessed. From this data, benchmark parameters (i.e. specific items that show good correlation with the staffing level) are derived and they can subsequently help to assess the performance of the organization under investigation. In this study, the benchmark focuses on staffing levels, so the resulting benchmark parameters can be used to help assessing the staffing level of a specific organization.

A concise benchmark on staffing of the RPD in medical facilities in the United States of America (US) is described in [2]. Without additional research, these results cannot be used to help assessing the staffing of Dutch RPD's. In this article results of a benchmark of Dutch RPD's is described, and a comparison with the US results is made.

Requirements on the staffing of the RPD are derived from Dutch law: facilities with a "Complexvergunning", the equivalent of the US "Broad License", are obliged to have an independent RPD. This obligation follows from the Basic Safety Standards for Radiation Protection Regulation (Rbs) [3, Article 5.28] that is based on the Nuclear Energy Act [4] and the Basic Safety Standards for Radiation Protection Decree [5]. Rbs and the guidance document for the application of a license [6], describe the RPD's tasks, but only state that it has to be "adequately staffed", i.e. be able to perform and fulfill the tasks and responsibilities required by law and regulations.

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2 METHOD

The first step in the benchmark design is the selection of facilities that will asked to participate in the study. The US benchmark [2] strictly focuses on academic and medical facilities having a RPD. In order to compare the situation in the Netherlands with that in the USA, Dutch facilities with a RPD, implying Broad License, were selected. Although Dutch nuclear installations are also obliged to have a Broad License, they are not considered here: their practices differ too much from the practices of non-nuclear facilities. That implies that their requirements on the RPD will be that much different that a comparison is not viable.

The scope of the Dutch benchmark has been extended as compared to [2]: not only medical facilities have been considered, but universities, research institutes and other (industrial) facilities as well. Currently there are only 23 licensees in the Netherlands with a Broad License that fall within this scope. Of these, 9 licensees are in the category "medical" (the facility can be either a hospital or a university and an academic hospital sharing a single license). The other category, "non-medical", consists of 14 facilities (research institutes, universities and industry).

2.1 Design of the questionnaire

Data for the benchmark has been collected using a questionnaire. For a valid comparison, the US benchmark questionnaire was used as a basis for the Dutch benchmark. A detailed description of questions in the US benchmark and their its relevance can be found in [2]. A question on the increased controls regulation, a security item, was dropped since it has no Dutch equivalent. The questionnaire was translated to Dutch, and extended to include the non-medical category. In addition, questions, specific for the Dutch situation, were added. These were questions on:

- the number of internal permits (ITs);
- the number of High Active Sealed Sources (HASS);
- the level of education of the RPD staff;
- the different types of applications of radiation;
- the number and type of laboratories where radiation can be used (radionuclide laboratories);
- who performs the quality assurance of radiation measurement equipment (QA/QC) and
- who performs the leak-check of radioactive sources.

For the additional questions the assumption is that they also will be correlated with the staffing level of the RPD: having for instance many radionuclide laboratories could mean a larger workload on the RPD, and thus more FTE. For the university/hospital combinations sharing a single license, questions were added in order to be able to attribute the staffing to respectively the university and the hospital. In the translation some assumptions were made, especially regarding terminology:

- The "Authorized Users (AU)" in the US are considered to be equivalent to those staff members that are mentioned in person in internal permits, and are allowed to use specified applications ionizing radiation without further supervision of the RPD. A distinction is made between those AU allowed to administer radioactive material to humans (human use AU) and those that are not allowed to do so (non-human use AU). This was described in some detail in the questionnaire for clarification.
- "Dental medicine/dentistry" was translated into "tandheelkunde en kaakchirurgie" (dental care and surgery of the jaw), since the latter is a specialization of dentistry in the US.
- The "academic tubes" are research instruments for non-clinical use.
- The question on the number of lasers refers to class 3 and 4 lasers only: they can be part of the RPD's responsibility.

The questionnaire was emailed to the Radiation Protection Experts of the different organizations, with a detailed explanation of the purpose of the study, and the offer to keep all respondents informed about the results. Because of the small group of facilities with a Broad License, a high response rate was required in order to perform a meaningful comparison. This was the reason for adding the detailed explanation of the purpose of the study. The confidentiality of the answers was emphasized by

explaining that all data were to be presented anonymously and in a format that makes identifying the facility very difficult.

3 DATA ANALYSIS

The benchmark is number-based. Since some answers were non-numerical, these answers are transposed to a number. These questions, and the method to assigned a number to them, are:

- The number and type of radionuclide laboratories. The numerical value is obtained by adding the number of laboratories.
- The number of applications of ionizing radiation is obtained by summing the applications as returned by the respondents.
- Who performs the QA/QC and the leak-check of radioactive sources? If the QA/QC is performed only by the SBE, a 2 is assigned to this answer, if it is jointly performed by the RPD and others a 1 is assigned, when there is no participation by the RPD a 0 is assigned. This assignment is also used for the question on leak testing and the values obtained for both questions are added.

As one of the goals of the US/Dutch comparison is to determine appropriate benchmark parameters, the results are normalized. An additional advantage of normalizing the data is that it helps maintaining the respondent's anonymity. The normalization is done using the number of FTE as scaling parameter. Obviously the number of people working in the RPD is not a useful normalization factor, as it does not take part-time work into account. The normalization procedure is adopted from [2]. To clarify the procedure, the number of HASS is taken as an example. Assuming that a RPD of 3 FTE is part of a facility that uses 60 HASS, the normalized answer on the number of HASS therefore amounts to 3 FTE/60 sources = 0.05. In this normalization procedure only the number of FTE of the radiation protection experts in the RPD is used. The number of FTE of the support personnel is not used: it will be less well defined and can depend on other topics not covered in the questionnaire, for instance whether the RPD is part of a Worker Safety department. Note that the effect of ignoring support staff in the normalization procedure is limited: except for one organization the relative number of support staff is less than 10%.

3.1 Statistical parameters

The small dataset size, in combination with a large variation in the data, makes an extensive statistical analysis infeasible. As the main goal of this study is to determine suitable benchmark parameters, the focus in the analysis will be on the variation of the parameters: a parameter that shows small variation is more suited for benchmarking than one showing large variation. The statistical parameters, based on the normalized data that will be used, in addition to the number of responses (*Nr*), are:

- the median value (*Median*)
- The minimum and maximum value (*Min, Max*)
- The relative range, defined as *rRange* = (*Max*-*Min*)/*Median*
- The average (*Avg*)
- The relative standard deviation of the average σ_{N-1}.

In the US benchmark only *Median, Min* and *Max* are used, so the other parameters have been derived from the US dataset.

4 RESULTS AND DISCUSSION

To increase readability all tables are included at the end of this paper. A first analysis of the data showed that the answers on three questions could not be analyzed because a lack of usable answers. In the subsequent presentation of the results these answers will not be reproduced.

4.1 Response rate

After 6 weeks, the positive response rate was 83% (19 respondents), a very satisfying result, especially considering that filling in the questionnaire takes quite some time. As a comparison: in the US

benchmark the response rate, after 6 months, was 30%. Sending out a reminder email, with a more pressing request, presumably helped in obtaining this high response rate.

4.2 Benchmark parameters

In this section the normalized results are presented. The statistical data, derived from the normalized data, is discussed and a comparison between the different types of facilities is made. Normalized data for the Dutch facilities is shown in table 1. The derived statistical data is shown in table 2 for both the Dutch and US facilities. Inspection of the results shows that *rRange* shows a large variation in all datasets. This is not surprising since it, by definition, is strongly dependent on the extreme results. The standard deviation, taking all data into account, shows a lower variation and will be used in the subsequent discussion.

For most parameters, the variation in the Dutch medical facilities dataset is equal to or lower than in the US dataset. A possible explanation is that the group of Dutch medical facilities is more alike than the US facilities, even though they are all Broad License facilities. An indication for this is that the size of the US medical facilities seems to vary more than in the Dutch group, as inferred from e.g. the number of instruments (X-ray tubes, CT scanners).
The parameters showing the lowest variation largely overlap for both US and Dutch medical facilities. The most notable exception is "#Authorized Users", presumably due to somewhat differing responsibilities for Authorized Users in both countries.
A comparison of *Avg* and *Median* shows that they are usually larger in the US. This might be caused by possible additional tasks (as compared to the Netherlands) for the RPD in the US, that are not reflected in the questionnaire. Another possible explanation is that Authorized Users in the Netherlands have tasks and responsibilities that in the US belong to the RPD.

The parameters in table 2 strongly focus on medical applications, so a comparison with results for the non-medical facilities will be limited. The variation for most questions that do overlap appears to be somewhat higher for the non-medical facilities than for the medical facilities. This is probably caused by the larger variation in the application of radioactivity in the non-medical facilities.

The derived statistical data for the additional questions is shown in table 3. Apparently the number of IT's has limited effect on staffing of the RPD. The work related to issuing IT's is apparently relatively low, probably because they are not issued very often. Differences in variation of the results between the medical and non-medical facilities are, as before, attributed to the larger variation the use of radioactivity in the non-medical facilities. For the medical facilities, the number of HASS shows little variation, making it especially useful in benchmarking.

5 CONCLUSION

The response rate of the selected group of users with a Broad License was high: apparently it is a topic with some significance for the Dutch radiation protection community.
The US not only has the largest RPD's, but also has facilities with much larger numbers for, for instance authorized users and dosimetry wearers than in the Netherlands. This presumably is related the larger overall size of many US facilities. Contrary to the situation in the US, the range of FTE's in Dutch RPD's is relative small: although the variation in the facility size is large, the variation in the staffing of RPD's varies from less than 0.5 to about 6 FTE.
Noticeable is the large variation in the number of dosimetry wearers in non-medical facilities as compared with the medical facilities. It would be interesting to investigate whether this is explained by differences of the total number of employees or by other reasons.
In the US, the benchmark parameters used to assess the staffing levels are mostly comparable to or higher than those in the Netherlands. Valuable information on benchmarking parameters can be deduced from the variation in the answers. The topics that showed low variation overlap to a large extent in the US and Dutch medical facilities.

A part of the variation in the answers can be attributed to the fact that the questionnaire only covers part of the work of the RPD: for instance, training and education is a significant part of the work-package of some RPD's, but not of all.
Based on the variation in the answers (both US and the Netherlands) and an arbitrary maximum variation of 33% we conclude that the following parameters can be used in a benchmark of the RPD (sorted on increasing variation for Dutch medical facilities). Note that not all parameters are applicable for each facility:

- #I-131 therapy patients
- #gamma camera's
- #I-131 therapy internal patients
- #dosimetry wearers
- #CT scanners
- #linacs
- #PET/PET-CT scanners
- #Academic tubes (NL only)
- #lasers
- #Authorized users for human use (NL only)
- #dental X-ray tubes
- Time spend for committee meeting (US only)
- Time spend for committee work

For the questions specific for the Netherlands:

- HASS
- involvement of the RPD with QA/QC and the leak-check of radioactive sources
- different applications

The parameters showing the lowest variation are useful for a quantitative comparison between facilities, respecting the inherent variation due to the differences in the facilities. It is important to note that this type of benchmark can be considered as a tool to compare RPD staffing of similar facilities, but not as a means to determine the required RPD staffing level.

6 REFERENCES

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Table 1: Normalized data for the benchmark parameters for Dutch facilities.

	Authorized users(AU)	Non-human use AUs	Human use AUs	Dosimetry wearers	#Gamma cams	#PET/PET-CT	I-131 therapies	Internal I-131 therapies	#X-ray tubes	#CT scanners	#linacs	#Dental X-ray tubes	#Academic tubes	#Committee meeting hours	#Committee total hours	#lasers
Medical facilities																
20	0.004	0.004	0.045	0.004	0.675	2.70	0.0386	0.054	0.054	0.142	0.193	0.225	0.675	0.061	0.038	0.090
21	0.117	0.023	0.070	0.008	1.167	3.50	0.0292	0.058	0.438	0.292	0.583	1.167	0.350	0.044	0.022	0.175
22	0.074	0.035	0.095	0.008	1.033	3.10	0.0235	0.025	0.310	0.689	0.477	0.238	1.033	0.258	0.129	0.238
23	0.157	0.157	0.157	0.010	1.573	1.05			0.262	0.450	0.629	1.049		0.039	0.030	
24	0.013			0.004		2.00	0.0139	0.024		0.400	0.400	0.400	1.000	0.063	0.063	
25				0.004	0.595	0.79	0.0238	0.048		0.298	0.476	0.595				
26	0.708			0.009	1.063	4.25	0.0280	0.053	0.033	0.425	1.063	0.170		0.156	0.156	
Non-medical facilities																
1	0.157	0.157	0.157	0.016					0.063					0.039	0.030	
2	0.104	0.016		0.017					0.036					0.010	0.006	0.015
3	0.019	0.005		0.030					0.150					0.038	0.008	0.075
4	0.066	0.066		0.058												
5				0.010					0.018					0.029	0.022	
6	0.500			0.400												
7	0.154	0.007												0.025	0.013	
8	0.067	0.500		0.250										0.014	0.007	
9	0.007			0.125					0.020							0.030
10	0.183	0.016		0.004					0.020					0.069	0.046	
11				0.090												
12	0.120	1.000		0.013					0.300					0.013	0.006	
13	0.023												4.000	0.063	0.063	
14				0.013									5.882			
15	0.375			0.025							0.375			0.156	0.156	

Table 2: Summary of the statistical parameters. The Dutch medical facilities are shown in the upper part of table, US medical facilities in the middle and Dutch non-medical facilities are shown in the lower part. For comparison all σ_{N-1} values up to 33% are marked in yellow.

	Authorized users(AU)	Non-human use AUs	Human use AUs	Dosimetry wearers	#Gamma cams	#PET/ PET-CT	I-131 therapies	Internal I-131 therapies	#X-ray tubes	#CT scanners	#linacs	#Dental X-ray tubes	#Aca-demic tubes	#Committee meeting hours	#Committee total hours	#lasers
Dutch medical facilities																
Median	0.095	0.029	0.083	0.0078	1.048	2.700	0.026	0.050	0.262	0.400	0.477	0.400	0.838	0.062	0.050	0.175
Nr	6	4	4	7	6	7	6	6	5	7	7	7	4	6	6	3
Min	0.004	0.004	0.045	0.0036	0.595	0.793	0.014	0.024	0.033	0.142	0.193	0.170	0.350	0.039	0.022	0.090
Max	0.708	0.157	0.157	0.0105	1.573	4.250	0.039	0.058	0.438	0.689	1.063	1.167	1.033	0.258	0.156	0.238
rRange	7.40	5.22	1.36	0.89	0.93	1.28	0.95	0.69	1.54	1.37	1.82	2.49	0.82	3.54	2.69	0.85
Avg	0.18	0.06	0.09	0.0066	1.02	2.5	0.026	0.044	0.22	0.39	0.55	0.55	0.77	0.10	0.07	0.17
σ_{N-1}	61%	63%	26%	16%	14%	19%	13%	14%	35%	17%	19%	28%	21%	34%	32%	26%
US medical facilities																
Median	0.065	0.084	0.25	0.0095	0.646	2.000	0.071	1.000	0.0745	0.500	1.429	0.333	0.3465	0.102	0.0365	0.28
Nr	27	21	21	26	20	17	19	13	26	23	15	16	18	24	22	15
Min	0.011	0.013	0.015	0.001	0.067	0.333	0.008	0.057	0.005	0.053	0.25	0.057	0.1	0.014	0.007	0.037
Max	0.250	4.000	4.000	0.067	5.000	17.000	0.607	3.600	3.000	7.000	7.000	2.400	12.000	1.500	0.600	1.000
rRange	3.68	47.46	15.94	6.95	7.64	8.33	8.44	3.54	40.20	13.89	4.72	7.04	34.34	14.57	16.25	3.44
Avg	0.085	0.32	0.59	0.014	1.2	3.1	0.14	1.2	0.45	1.3	1.8	0.54	1.3	0.26	0.09	0.32
σ_{N-1}	14%	58%	35%	22%	28%	29%	29%	27%	40%	30%	27%	30%	52%	29%	33%	22%
Dutch non-medical facilities																
Median	0.112	0.041		0.0250					0.036				4.941	0.033	0.017	0.030
Nr	12	8		13					7				2	10	10	3
Min	0.007	0.005		0.0044					0.018				4.000	0.010	0.006	0.015
Max	0.500	1.000		0.4000					0.300				5.882	0.156	0.156	0.075
rRange	4.40	24.38		15.82					7.88				0.38	4.38	8.73	1.99
Avg	0.15	0.22		0.08					0.09				4.9	0.046	0.036	0.040
σ_{N-1}	29%	57%		40%					46%				19%	30%	41%	45%

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Table 3: Summary of the statistical parameters for the questions specific for the Netherlands. For comparison all σ_{N-1} values up to 33% are marked in yellow.

Dutch medical facilities	#HASS	# IT's	RPD: QC, equipment check and leak testing	#different applications	#radionuclide labs (B,C,D-labs)
Median	0.886	0.052	1.583	0.457	0.238
Nr	7	7	6	4	7
Min	0.629	0.014	0.500	0.386	0.118
Max	1.190	0.304	3.147	0.886	3.147
rRange	0.63	5.61	1.67	1.09	12.70
Avg	0.88	0.11	1.7	0.55	0.7
σ_{N-1}	9%	44%	21%	22%	57%
Dutch non-medical facilities					
Median	0.725	0.056	0.833	0.333	0.725
Nr	9	13	12	12	13
Min	0.023	0.010	0.167	0.060	0.060
Max	1.573	0.100	1.833	4.500	5.500
rRange	2.14	1.61	2.00	13.34	7.50
Avg	0.67	0.046	0.84	0.8	1.5
σ_{N-1}	23%	20%	19%	45%	35%

Knowledge Management for the Radiological Protection and Criticality Specialism in the Office for Nuclear Regulation

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Abstract. Within the next 5 years the Office for Nuclear Regulation has the potential to lose valuable knowledge due to the demographic profile of the organisation. It is important as an organisation we react to this, and develop sustainable processes for capturing and disseminating knowledge and information that can used to train, develop and sustain the next generation of inspectors. Our corporate goal is to ensure the Office for Nuclear Regulation develops and maintains the necessary regulatory culture and appropriate understanding of standards, competency and regulatory history needed to regulate effectively and efficiently.

KEYWORDS: *Knowledge Management, Radiological Protection, Sustainability, Resilience*

1. INTRODUCTION

Within the next 5 years the Office for Nuclear Regulation (ONR) has the potential to lose valuable knowledge due to the demographic profile of the organisation.

It is important as an organisation that we react to this and develop sustainable processes for capturing and disseminating knowledge and information that can used to train, develop and sustain the next generation of inspectors.

Our corporate goal is to ensure ONR develops and maintains the necessary regulatory culture and appropriate understanding of standards, competency and regulatory history needed to regulate effectively and efficiently.

Within the Radiological Protection and Criticality specialism development of these processes are particularly important due to the demographic profile of our current inspectors, and in recognition of the experience of new inspectors who may not have experience of the nuclear industry.

Within the specialism we aim to appropriately and effectively acquire, retain and maintain relevant knowledge to support the ONR mission to provide efficient and effective regulation of the nuclear industry, holding it to account on behalf of the public, and make this knowledge accessible to all of ONR by:

- Appropriately capturing, cataloguing and making accessible knowledge from new starters, movers and leavers of the specialism.
- Encouraging and facilitating collective input from all Radiological Protection and Criticality Inspectors.
- Providing on-going learning and development opportunities for specialism members to maintain and develop expertise in their field.
- Encouraging specialism members to obtain and maintain membership of relevant professional societies and institutions.
- Defining core technical knowledge requirements
- Assessing staff competencies and developing Specialism Resilience Matrices and using these to fill gaps.
- Putting knowledge management delivery plans and Information Maps in place
- Maintaining two way communications between the specialism knowledge management group and other ONR knowledge management specialism sub-groups, and the ONR corporate knowledge management Lead.

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2 CHALLENGES FOR THE RADIOLOGICAL PROTECTION AND CRITICALITY SPECIALISM

2.1 Experience

The specialism is made up of a diverse range of disciplines and experience. There are 33 members who are considered to be the core of the specialism, who cover the following disciplines:

- Radiological protection
- Criticality
- Radiation consequences
- Emergency preparedness and response
- Transport
- Shielding

The specialism is comprised of the following:

- Nuclear Graduates – ONR sponsors the '*Nucleargraduate'* training scheme – graduates undertake three placements over a 2 year period, with the offer of a permanent position as a nuclear associate on completion. Normally the nuclear graduate would have finished either a degree or a master's degree, and as such will have limited work or nuclear experience.
- Nuclear associates – this role is primarily to provide technical support to ONR inspectors. This role provides a development route to Inspector. Although normally this is the role the nuclear graduate would be recruited into, nuclear associates are also recruited directly with either limited work experience of the nuclear industry or with radiological protection experience from non-nuclear sectors.
- Inspectors – in general these are recruited directly from the nuclear industry with extensive experience in their specialist discipline.

Although each member will have specific discipline experience, there is considerable variance in regulatory experience, ranging from less than a year to over 30 years.

The specialism is moving to recruiting more at the nuclear associate level and providing extensive training to develop the recruit to the inspector position. With this in mind, it is important that there needs to be the mechanisms available to capture the extensive regulatory knowledge that exists within the specialism to enable the new recruits to learn and develop.

2.2 Demographic Profile

The demographic profile of the specialism is that there are more inspectors closer to self-selected retirement, than new those just starting their career.

In the United Kingdom (UK) the default retirement age, which was previously 65 for the general population and 60 for those in public service, was fully abolished in October 2011. This allows employees to stay in employment longer with retirement at their discretion (unless they are retired by the employer on very specific grounds).

Experience has shown that inspectors have retired from ONR in their early 60s, with many undertaking amended working arrangements, such as reduced hours or days two to three years before actually retiring.

The main impact of this is around succession planning, although this should not wait until the individual employee has indicated a desire to retire.
Although knowledge management activities should be an on-going process, based on the retirement profile information available, it is important that knowledge capture activities for those towards the end of their career are underway before they exceed 60.

2.3 Personal Aspirations and Career Development

It is important that there are opportunities for inspectors to develop their career. For inspectors in the specialism this has generally involved a move to site inspection activities, which will take them away from the radiological protection and criticality specialism. It is important to ensure before the move takes place appropriate knowledge capture activities have been instigated, or if this has not been possible to ensure time and access is made available to undertake these activities after starting their new role.

2.4 Personal Information Libraries

We hold information, either in electronic or printed form, especially if the information is regularly used. It is important that this is shared and made available to others. A large amount of information is now in electronic form, which allows for easy access and for links to be created to the source material. One challenge is old established publications that may only exist in printed form, and which are not easy to scan. There is always a concern these could go missing.

2.5 Corporate Initiates and IT Platform

To develop knowledge management awareness across the organisation a number of initiatives have been instigated and include:

- Changes to the training and assimilation of new inspectors.
- Training of a group of Knowledge Champions in knowledge elicitation techniques to address the concern of knowledge loss due to the demographic profile of the organisation.
- Use of resilience maps to identify vulnerabilities in knowledge areas and to target training and recruitment of inspectors.
- Development of knowledge management training courses

Since 31st March 2014, the ONR has been a public corporation and a fully independent nuclear regulator, having previously been an agency of the Health and Safety Executive. Since this time, ONR has been developing its own IT strategy and systems that suit the size of the organisation. The current IT platform does not provide the flexibility required to enable easy development of the specialism community pages which provide a portal for the specialism to access information.

Corporately the importance of knowledge management has been recognised with initiatives being implemented, such as the Well Informed Regulatory Decisions project which will deliver modern systems and tools and is due for delivery in 2020.

It is important that knowledge management activities do not stand still waiting for the corporate initiatives to be delivered, and that engagement with the corporate knowledge management lead and other specialisms continues so we can develop our plans to ensure a seamless integration.

3 ACQUIRE, RETAIN, MAINTAIN AND SHARE RELEVANT KNOWLEDGE

3.1 Acquiring Knowledge from Members of the Specialism

Members of the specialism have been encouraged to provide information that can be shared on the specialism knowledge management intranet community pages.

There are two areas of current focus, knowledge capture from potential retirees, and obtaining examples of enabling regulation related to radiation protection, or criticality.

The current method for capturing knowledge from retirees has been to undertake audio recordings of discussions based around their career with the ONR. Transcripts and the audio recordings are then made available to the specialism. To enable the discussion to be structured a set of questions were developed and shared with the individual ahead of the recording session. This approach was very successful in focussing the session.

Aside from the typical type of information, such as links to relevant web-sites or publications, we are seeking to include examples of where inspectors have made key decisions relating to radiation protection or criticality which can be used to inform future assessments and decisions. This enables a consistent approach to be taken.

3.2 Retaining Knowledge and Maintaining Knowledge

As an organisation we hold an extensive amount of information, much of which is retained in the electronic document management system. Due to the age and design of this system, information can be difficult to search for.

The specialism has established a dedicated intranet community page that provides links to information help in the document management system. Members of the specialism are encouraged to highlight any work which will be of interest so a dedicated link can be created.

Some key information is only available in printed form, and if this is the case, efforts are made to scan this so an electronic copy is available for easy reference. This is not possible for some publications, so a small technical library is maintained by the specialism.

One of the key factors in maintaining knowledge management is the people who develop and then share the knowledge they have gained as they progress in their career. It is important that the organisation recognises the need for people to be trained and to maintain competency in their specialist area.

Within the specialism a competency matrix has been developed in line with that required by professional bodies. The matric identifies four levels of competency:

- **Level 1: Aware -** Performs activity with significant supervision and guidance. Performs basic routine tasks, little / no responsibility.
- **Level 2: Familiar** - Performs actively in a range of contexts, supervision required in more complex circumstances. Some individual responsibility or autonomy.
- **Level 3: Skilled -** Performs activity in some complex or non-routine contexts. Significant responsibility and autonomy - can oversee the work of others.
- **Level 4: Expert -** Performs activity in a wide range of complex or non-routine contexts. Substantial personal autonomy - can develop others within the activity.

The competency areas currently assessed are:

- Radiological protection
- Criticality
- Emergency preparedness and response

Competency areas are under development for radiation consequences, transport and shielding.

Within the Radiation Protection competency there are such topics as:

- Understanding of Radiological Protection Principles
- Legislation and Standards
- Application of Shielding Knowledge & Understanding
- Application of Atmospheric Dispersion Knowledge & Understanding
- Application of Respiratory Protective Equipment in relation to Radiological Protection

Also with the specialism subject matter experts have been assigned for the following subjects:

Personal Protective Equipment	Shielding
Respiratory Protective Equipment	Radiation Metrology & Instrumentation
Radiation Protection Training	Emergency Planning & Response National Arrangements
Level 3 Probabilistic Safety Assessment	Source Terms

Legislation
Basic Safety Standards Directive
Implementation
Occupational Exposure / ALARA[6]
Public Exposure / ALARA
Criticality
International Forums
Land Use Planning
Research
De-licensing
Internal Dosimetry
Dosimetry Approvals
Transport

3.3 Resilience Plans

Resilience plans have been developed to highlight a range of factors to ensure the specialism demographics is correct to support the organisation. Reviews are undertaken annually and cover such topics as:

- Potential retirees
- Staff aspirations – staff may want to move to site or project inspection, or corporate roles, and secondments outside of ONR
- Competencies
- Specialism training needs analysis
- Time taken to recruit new inspectors

3.4 Sharing Knowledge

It important that the information held is easily accessible to members of the specialism and others in ONR. To enable a single point of information, a community page has been developed. Due to current IT limitations, this currently provides links to relevant information held in the ONR electronic document management system. It is hoped that with future IT developments, the community page will operate more like a web page.

The key focus is expanding the links available on the community pages in the following areas:

- Regulatory decisions relevant to the specialism
- Examples of enabling regulation[7] [1]
- Operational learning from national and international events
- New technologies
- Feedback from international meetings

The aim is that the community page are the 'go to' point of resource for members of the specialism, and to assist in this knowledge management is a standing agenda item at the regular specialism meetings. This is also supplemented with targeted knowledge management awareness sessions.

4 CONCLUSION

Knowledge management is a continual process which should continue to develop and grow. For the specialism the key areas are:

- Knowledge Management should be part of everyday work with each inspector asking '*how can I share the work I'm doing now to help inform others'*.
- Improvements to the ONR career end to end knowledge capture process
- Encourage members of the specialism to use the community pages on a regular basis, and extend this to the whole of ONR.

[6] As low as reasonably achievable (ALARA) equates to as low as reasonably practicable (ALARP) in the UK

[7] Enabling Regulation is defined as "A constructive approach with dutyholders and other relevant stakeholders to enable effective delivery against clear and prioritised safety and security outcomes."

At the ONR corporate level, knowledge management has been recognised as part of the top 10 priorities for 2018/19 and to support this ONR are developing more appropriate IT infrastructure and platforms suitable for the organisation.

In addition, ONR has instigated the Well Informed Regulatory Decisions project with the aim of '*Modernising our regulatory management systems and procedures to provide efficient processes that enable our teams to deliver effective regulation more flexibly*' which is due for delivery in 2020.

To support this project, knowledge management champions from across the organisation have provided input in to the initial scoping project and will be part of a pilot study to review the infrastructure changes before roll out across the organisation.

All disciplines within the ONR, and in particular radiological protection and criticality are developing in the field of knowledge management to ensure that ONR inspectors are at the forefront of technological and regulatory developments ensuring the continued capability of the organisation as a world leading regulator for years to come.

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Practical implications/consequences of new Dutch legislation due to Council Directive 2013/59/Euratom of 5 December 2013

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Abstract:

Due to new European legislation, a large part of the Dutch policy on radiation protection has changed resulting in major consequences for the day-to-day practice in radiation protection.

Main implications/consequences:

- Change of exposure situations (planned exposures, existing situations and incidents)
- Lower dose limit to the lens of the eye
- Lower clearance and exemption levels for a large number of isotopes
- Change of system of licensing, registration and reporting
- Incorporation of dose as a result of exposure to radon
- Registration of dosimetric information of imaging systems
- Use of diagnostic reference levels
- Decommissioning plan for accelerators

Some issues will increase the workload for the radiation protection management for example writing a decommissioning plan for accelerators. Other issues will increase the costs, for example the substantial lowering of the clearance level, resulting in an increase in radioactive waste that has to be sent to the COVRA.
Most of the approximately 30 “Complex Permit holders” in the Netherlands are united in an association called the “Large Permit Holders”. This group meets on regular basis to discuss issues regarding radiation protection in large institutes and companies. One of the main issue in the past years is the implementation of the new Council Directive 2013/59/Euratom of 5 December 2013 and the practical implications and consequences for the new Dutch legislation.

Because the changes directly influence the local policy on radiation protection in institutes and companies, these issues are addressed during “Large Permit Holders” meetings. Working groups initiated by the Large Permit Holders, the NVS board and/or the Authorities on several topics discuss consequences and propose alterations in the legislation.

Formed working groups in association with other parties and authority:

- NORM working group, mainly on exemption and clearance
- Working group on exemption and clearance of artificial isotopes
- Working group on writing a decommissioning plan of accelerators
- Working group on writing an emergency plan for the institute of company

Within the Large Permit Holders group an extensive cooperation between the approximately 30 institutes and companies is present. The main goal is working safely with ionizing radiation but where legislation is to strict or not practical, to discuss this with the government and, in discussion, adapt regulation to a more practical regulation.

By joining the expertise of radiation protection experts in an association that seeks the connection with the authorities, a save but still workable radiation protection environment can be established. This poster will address the issues and discuss the possible solutions.

Introduction:

Due to new European legislation, a large part of the Dutch policy on radiation protection has changed, resulting in major consequences for the day-to-day practice in radiation protection. These consequences concern issues like an increase work load of increase in costs of waste disposal.

Main implications/consequences:

- Change of exposure situations (planned exposures, existing situations and incidents)
- Lower dose limit to the lens of the eye
- Lower clearance and exemption levels for a large number of isotopes
- Change of system of licensing, registration and reporting
- Change of system of Education & Training of RPOs
- Incorporation of exposure to radon
- Registration of dosimetric information of imaging systems
- Use of diagnostic reference levels
- Decommissioning plan for accelerators

Some issues will increase the workload for the radiation protection management for example writing a decommissioning plan for accelerators. Other issues will increase the costs, for example the substantial lowering of the clearance level, resulting in an increase in radioactive waste that has to be sent to the COVRA (national waste storage facility).

Large Permit Holders:

Most of the approximately 30 “Complex Permit holders” in the Netherlands are united in an association called the “Large Permit Holders”. This group is established in 2007 and consists of 47 members divided over 31 institutes. The so called “Complex Permit” in the Netherlands is a permit which gives the General Radiation Protection Expert the authority to issue internal permits in the organization within the limits of the “Complex Permit”.

The group meets on regular basis to discuss issues regarding radiation protection in large institutes and companies. The authorities and radiation physicians are invited on regular basis to join a meetings and discuss topical issues. One of the main issue in the past years is the implementation of the new Council Directive 2013/59/Euratom of 5 December 2013 and the practical implications and consequences for the new Dutch legislation.

Because the changes directly influence the local policy on radiation protection in institutes and companies, these issues are addressed during “Large Permit Holders” meetings. Working groups initiated by the Large Permit Holders, the NVS board and/or the Authorities on several topics discuss consequences and propose alterations in the legislation.

Working groups:

Formed working groups in association with other parties and authority:

- NORM working group, mainly on exemption and clearance
- Working group on exemption and clearance of artificial isotopes
- Working group on writing a decommissioning plan of accelerators
- Working group on writing an emergency plan for the institute of company
- Working group on eye lens dosimetry

Topics and solutions

NORM clearance levels

A workgroup has been assembled to address the following issues: higher specific clearance level for sludges, specific clearance levels for potassium salts, specific arrangements for clearance of construction materials and designing a calculation tool based on RP-122.

The result of this working group is that a specific paragraph has been included in legislation in which sludges can be cleared in higher concentrations.

Clearance levels artificial nuclides.
By lowering the clearance levels, more material will get the label “radioactive waste” and have to be discarded via the COVRA. This will increase the costs for waste disposable substantially. A typical example is a biochemical lab that produces about 10 waste bins annually with H-3 and C-14 in low concentration. In the situation before the 6th of February this material could be discarded as chemical waste. In the new situation these waste bins have to be discarded via the COVA indicating an increase in waste cost of 25.000 euro annually. This is a substantial part of the research budged available for this group and directly results in less budged for research.

By forming a working group that analyses the amount of produced waste and, based on a German study on clearance of radioactive substances[1], it can be calculated that when this amount is incinerated, the dose near the incineration facility will not exceed the 10µSv annually.
The workgroup will suggest to the authorities to broaden the limits for H-3 and C-14. The work of this group is still in progress.

Decommissioning plan
With the coming in act of the new legislation, a decommissioning plan for accelerators above 8 MeV has to be present. This plan is a detailed description on the procedure of decommissioning an accelerator above 8 MeV and the costs involved. To prevent that multiple institutes are doing similar work, a working group has been established to divided the work load and to benefit from institutes that have already been working on a decommissioning plan. In addition, a discussion with authority regarding unclear or unnecessary requested data will be organized. The work of this group is still in progress.

Emergency plan
Under the old legislation, an institute was already obliged to have an emergency plan. Under the new legislation this plan has to be more extensive and detailed. This will increase the work load of institutes. By joining knowledge of different institutes, the work load per institute can be decreased.

Eye lens dosimetry
The consequences of the lowering of the limit of the dose to the eye is discussed on regular basis in meeting, during several meeting also a radiation physician was invited and joined the discussion. A working group has produced a report on consequences of lower dose to the eye and practical guidelines.
A special committee has published a report on practical solutions in dosimetry and protection of the eye lens mainly during intervention radiology.

Conclusions:

By joining the expertise of radiation protection experts in an association that seeks the connection with the authorities, in harmonization with the NVS Board, a safe but still workable radiation protection environment can be established.
Within the Large Permit Holders group an extensive cooperation between the approximately 30 institutes and companies is present. The main goal is working safely with ionizing radiation but where legislation is to strict or not practical, to discuss this with the government and, in discussion, adapt regulation to a more practical regulation.

[1] Strahlenschutskommission (SSK), Freigabe von Stoffen zur Beseitigung, Empfehlung der Strahlenschutskommission, December 2006.

Security

Nordic exercise for unmanned systems – NEXUS 2017

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Abstract. The NEXUS exercise offered challenges in urban environments, contaminated fields and scenarios for fixed wing systems.
The use of fixed wing platforms was tested and demonstrated briefly.
Unmanned measurements in urban environments was tested and demonstrated in two scenarios, the street market and the 2-storey building. In particular the scenario around the 2-storey building demonstrated the 3D survey advantages with rotary wing systems. Survey of contaminated areas was tested and demonstrated in a scenario with dispersed activity in a pattern.
The exercise demonstrated that the capacities in the Nordic countries are in ongoing development. Exercise in an area where the teams can see each other's approach and solutions is most inspiring for the exchange and growth of knowledge.

KEYWORDS: Preparedness, unmanned, mobile gamma spectrometry, exercise.

1 INTRODUCTION

There are several measurement and sampling scenarios that may constitute very high risks for humans to carry out, e.g. reactor accidents, such as Chernobyl and Fukushima, RDDs (radiological dispersal devices) before and after explosion, search of MORC (material out of regulatory control), or search inside buildings that are under the threat of collapsing. For these scenarios, remotely controlled radiation measurements and sampling using unmanned aircraft systems (UAS) are developed.

The NKS-B activity SemUnaRS – Seminar on Unmanned Radiometric Systems, was held in 2014 in Linköping, Sweden. The seminar was the start-up and an inventory of the capacities for unmanned measurements in the Nordic countries [1].

The NKS-B activity NORDUM - Intercomparison of Nordic unmanned aerial monitoring platforms, was held in 2016 in Norway. The exercise gave five teams the opportunity to test their rotary wing UAS in three scenarios [2].

The NORDUM exercise was the first joint Nordic exercise for unmanned systems. The NEXUS exercise further expanded the challenges to urban environments, contaminated fields and scenarios for fixed wing systems. The exercise was held at an open, joint exercise area where the teams could observe each other's systems and techniques directly. The scenarios included small areas for rotary wing UASs searching for point sources, larger areas for assessment of contaminated areas, and surveys in urban environment.

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2 THE NEXUS EXERCISE

A more detailed description of the exercise area and each team's implementation and results are to be found in the project report [3]. Examples are given below.

The objective of the exercise was to test unmanned aerial platforms with respect to locating, identifying and estimating the activity of radioactive sources under field conditions. Thereby acquire competence within the Nordic countries.

The NEXUS exercise covered scenarios in these aspects:

- The use of fixed wing platforms that are beneficial in covering larger areas. They are intended to solve survey missions with assessments of ground activity concentrations or search for sources over larger areas. Fixed wing vehicles have generally longer flight times than rotary wing, but greater cruise speed which challenges the detector systems and data processing.
- The use of unmanned measurements in urban environments. Localization, identification, assessment of activities and recommendations of rescue routes.
- The team's reports to a figurative reach back were evaluated according to value as decision support in the scenarios.

The exercise area was Björka exercise field in the south of Sweden. The enclosed area houses a runway, an artificial urban street, forest and grass fields.

2.1 Scenario Street market

In the middle of the runway there was a built urban environment of several houses and sheds. A fenced area about 50x25 meters offered the setting of a street market with groupings of market stands and sheds. The teams were to assess the area from outside the market.

Four sources were placed on different heights; one ^{133}Ba, two ^{60}Co, and one ^{137}Cs source.

The sources were identified, and the activities assessed as displayed by the Norwegian team in Figure 1.

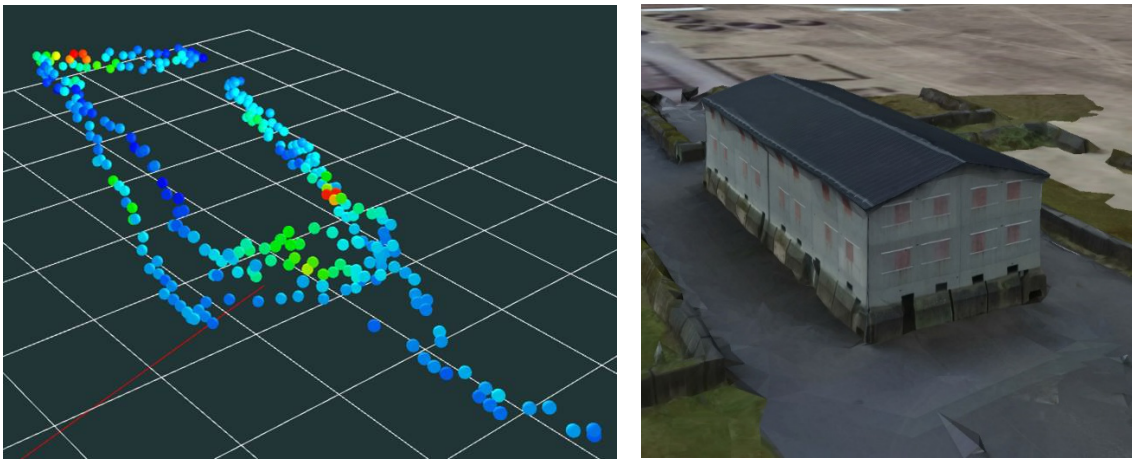
Figure 1: Map overview, detected sources at the Street market, Norwegian team.

2.2 The 2-storey building

The urban environment is also populated by a 2-storey building. 5 sources (one ^{133}Ba and four ^{137}Cs sources) were placed on different floors and collimated by their container and by the building's window frames.

Measurements at different altitudes and above the roof gave indications of the sources and a 3D rendering was made by the Norwegian team as seen in Figure 2.

Figure 2: NRPA 3D viewer (In-House software for manufacturing flight data in 3D with color scaling adjusted according to dosage value.). 3D reconstruction to the right from the Finnish team.



2.3 Single strong source

At the end of the runway, a 1 GBq ^{137}Cs source was placed on the ground for MDA (Minimum Detectable Activity) testing. The source was collimated upwards.

The empirical testing of the MDA showed that at altitude above 20 m, the source was hard to detect by a 2" NaI(Tl) scintillator as indicated in Table 1.

Table 1: Dose rates (SDI) recalculated to 1 m AGL(height Above Ground Level). Swedish team.

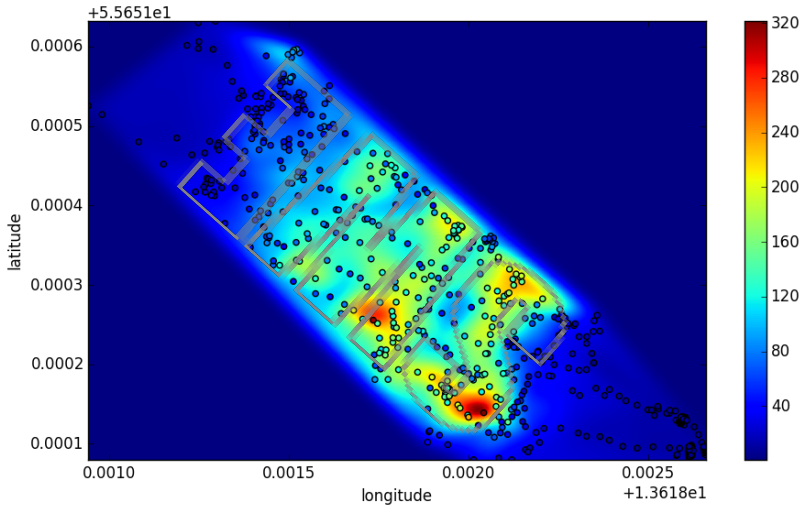
Altitude [m]	SDI	SDI-bkg	1 m AGL
7	850	750	36750
3,5	3500	3400	41650
10	550	450	45000
18	250	150	48600
30	100		
35	100		

2.4 The contaminated area

A part of the runway was contaminated with the total activity of 1150 MBq ^{99m}Tc.
The contamination was sprayed in the pattern of the letters S, W and E, with different activity concentrations, the letter ”S” had 2 times the activity than ”W” and ”E” was weak.

The contaminated pattern was hard to reproduce by measurements, presumably because of some diluting rain showers. The measurements still indicated the activity gradients, as seen in the Finnish heatmap in Figure 3, where the letters S, W, E were expected from lower right to upper left.

Figure 3: Heat map of the contaminated runway. Letters for illustration. Finnish team.



2.5 Fixed wing

The Swedish team had an ambitious, successful, first flight, but unfortunately, during the next flight, the prop broke with a following hard landing. The detector system survived, and they could have just replaced the motor and propeller with minor damage to the fuselage and carried on. However, we didn’t try our luck anymore as the sun set.

Figure 4: The Swedish team after the first, successful landing. Photo: Marie Carlsson

3 PARTICIPANTS

The exercise was coordinated by Linköping University, Sweden, who participated with one team, as did Norwegian Radiation Protection Authority and University of Oulu, Finland.
The exercise was facilitated by the Radiation Physics department at Lund University.

4 CONCLUSION

The use of fixed wing platforms was tested and demonstrated briefly. The Swedish team had their fixed wing designed for carrying a detector system in the air. The use of fixed wing UAS platforms could fill a gap between rotary wing UAS and full scale fixed wing systems in surveying larger areas.

The use of unmanned measurements in urban environments was tested and demonstrated in two scenarios. In particular the scenario around the 2-storey building demonstrated the 3D survey advantages with rotary wing systems.

Survey of contaminated areas in contrast to separate point sources was tested and demonstrated in the scenario with the contaminated area with dispersed activity in a pattern.

The aim to have team's report to reach back failed, presumably due to lack of time and preparations. This is something to develop since the end result should be decision support.

The capacities in the Nordic countries are still in development.

Exercise in an area where the teams can see each other's approaches and solutions is most inspiring for the exchange and growth of knowledge.

5 ACKNOWLEDGEMENTS

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Securing the High-Activity Sealed Source Life Cycle

From cradle to grave.

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Abstract. Securing the life cycle of a High-Activity Sealed Source is a road with quite some hurdles. Safety and security are abstract words and it depends on the perception of the radiation security expert which measures must be taken. Of course radiation security experts have consensus to reduce the likelihood of malicious acts involving these sources, but what if legislation lags behind the recommendations? Even with the International Atomic Energy Agency's recommendations about nuclear security, different insights between governments are noticeable and sometimes form obstacles in securing high activity sealed sources.

KEYWORDS: Radiation Safety, Radiation Security, Brachytherapy, Leksell Gamma Knife®, Nuclear Security Series.

1. INTRODUCTION

Sealed sources that are used in Radiotherapy are usually sources with a relative longer half-value-time compared with open sources used in the department of Nuclear Medicine. Sealed sources have hardly any contamination issues however security of High-Activity Sealed Sources (HASS[2]) does pose security challenges. For health protection of patients, public and healthcare providers administering therapy as well as environmental protection, security measures must be taken to counteract theft, abuse or sabotage of HASS.

1.1 Brachytherapy afterloader

In contrast to External Beam Radiation Therapy (EBRT) where high-energy x-rays from outside the body are focused on a lesion, Brachytherapy afterloaders are able to position a HASS directly with pinpoint precision at the area requiring treatment using catheters and special designed applicators. By moving the HASS with minute steps through the catheters with different time-intervals an optimal dose effect relation to the tumour can be achieved. An important characteristic of Brachytherapy afterloaders is that the radiation only affects a very local area around the sealed source. Exposure to radiation of healthy tissue that is located further from the source is thus reduced. The self-shielding afterloader is both a medical treatment device and simultaneously serves as a storage unit for the HASS. Most commonly used isotopes in Brachytherapy afterloaders are Iridium-192 and Cobalt-60.

1.2 Stereotactic radiosurgery (SRS)

Another usage of HASS in Radiotherapy is in Leksell Gamma Knife® surgery. This treatment is a non-invasive method specifically developed to facilitate pinpoint irradiation of brain lesions. It is the delivery of a single, high dose of irradiation to a small and critically located intracranial volume through the intact skull. The Leksell Gamma Knife® uses up to 201 precisely focused rays of gamma radiation. On its own, each gamma-ray results in a relatively low dose, but when they converge the gamma-rays combine to provide a precise, intense dose of radiation. The patient is positioned in such a way that the isocenter and the lesion coincide as precise as possible. The self-shielding Leksell Gamma Knife® is both a medical treatment device and at the same time functions as storage unit for the HASS. The isotope used is Cobalt-60.

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[2] Please note that in order to keep the text readable, the acronym HASS is used for both single as multiple sources.

2 CATEGORIES

The risk of criminal or intentional unauthorized use of radioactive material creates a threat to international security. IAEA gives a clear indication that HASS are a high risk target to perpetrators [1]. In recent history we have seen several malicious acts of smuggled [2] and stolen consignments [3]. Radioactive sources are expected to be very attractive to adversaries depending on radioactivity levels, as well as depending on the possibility to obtain them.

The source categorization used in the International Atomic Energy Agency (IAEA) Code of Conduct [4] is based on the activity (A) and the "danger-value" (D) [5] of the sources. The D value is the quantity of radioactive material which is considered a dangerous source. A dangerous source is a source that, if uncontrolled, could result in deterministic health effects. In order to specify the appropriate security level, we have to calculate the A/D ratio. Depending on the A/D ratio the source is assigned to a category that has a certain level of security measures associated to it. Category 1 sources are required to have security measures which meet the security objectives of Security level A. Security measures for category 2 sources must comply with the objectives of security level B. Where the goal of security level B is to *minimize* the likelihood of unauthorized removal of radioactive sources the goal of security level A is to *prevent* any unauthorized removal of radioactive sources. A detailed description of the requirements of the security levels can be found in the IAEA implementation guide for security of radioactive sources [1].

2.1 Brachytherapy afterloaders

In terms of A/D ratios, brachytherapy sources used in afterloaders are categorized as Category 3 sources. It is important to realize these sources are abundantly available and therefore easier to acquire for malevolent purposes, than SRS sources. Although the radioactivity of individual brachytherapy sources is much lower than individual SRS sources, they reside under Category 2, Security level B, due to their 'level of attractiveness' (e.g. ease of handling, storage, self-shielded portable devices).

2.2 Stereotactic radiosurgery (SRS)

According to the IAEA's categorization scheme [6] Leksell Gamma Knife® sources belong to Category 1. Not only due to the activity of the individual sources, but also the fact that multiple sources are located in the same equipment which might be of interest to perpetrators. Immediate detection and intervention during tampering with the Leksell Gamma Knife® is therefore required. These sources reside under Category 1, Security level A

3 REGISTRATION

The security of the life cycle of Elekta HASS starts even before a source is ordered. Part of the recommended registration process of these type of medical devices is to investigate if there are any circumstances which could lead to a breach of national or international legislation. Preferably before the sales department acquires leads and ventures into a new market territory, but certainly before any sales quotes, it is recommended that sales teams and supporting staff verify with the radiation protection expert if legal export of radioactive materials into the new market is cleared. The new market needs to be assessed for any economic sanctions related to exporting radioactive materials and embargos to that specific country. Additionally a recent list of countries supporting the IAEA's Code of Conduct must be checked. In case the targeted county is not mentioned in the IAEA list, further assessment is required before moving forward in the sales process. The first step is to check if there is a national regulatory body for the control of radiation sources. The second step is to look-up if the country does have a contact person for import and export of radioactive sources as mentioned in the contact list of the IAEA [7].

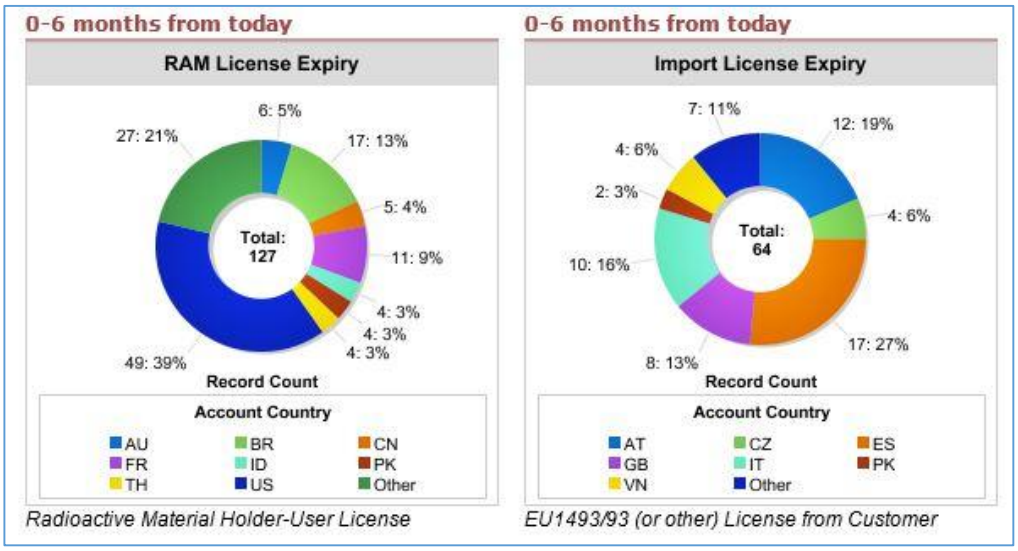
Countries with a regulatory body in place are much easier to asses for radiation safety import & export advice. For countries without a regulatory body it is a more cumbersome process as in these rare scenarios one has to take a deeper look into the local legislation to see if the country complies with the most basic safety and security principles. In most of these cases, the responsibility for radioactive sources is fragmented over multiple governmental departments.

War-torn countries as well as politically unstable countries were the safety of staff cannot be guaranteed anymore and that are still mentioned in the list of countries supporting the IAEA's Code of Conduct are considered a special group. In these cases the question arises should we refrain from shipping sources to these countries and in doing so deny the population in these regions, access to state of the art, lifesaving treatments? Obviously there is no unambiguous answer for this question. Every case must be considered separately. In case of serious doubt the process should be put on hold. Matter of fact; in some countries it took several years before the first radioactive source was allowed to be delivered.

4 ORDERING

After the registration process has been completed the next mandatory security phase is to check whether the hospital has a valid relevant nuclear energy license (also referred to as radioactive material -RAM-license). Requesting a copy of this license is the most effective solution, challenges such as native languages can be countered be requesting a certified English translation. The license must be checked on validity and contents by a radiation protection expert. An example of licenses expiry check is shown in Fig.1. Extra attention should be paid to the maximum admissible activity that may be present on the

Figure 1: Example of Import and RAM licenses expiry.



premises of the hospital. Source exchanges usually occur after one half-life-time, the total activity during a source exchange will be about 1.5 times higher than the maximum activity of a single source; the new source and the partially depleted old source. It is the responsibility of the hospital that their license covers the total present activity during source exchange. The legal manufacturer is legally required to insure that license of the recipient hospital covers the total amount of activity even before a source is put on transport.

5 SECURITY

After the publication of IAEA's Nuclear Security Series, measures have been taken in several countries to counteract theft, abuse or sabotage of HASS installed in clinical treatment units. Detection with motion sensors or tampering devices are the most commonly used devices to detect unauthorized removal or sabotage. Most security measures are based on the principle of delaying a perpetrator. The resilience time of the location where the sources are stored must be longer than the response time. The response time starts as soon as one of the sensors detects an unauthorized attempt to access. The response time includes the time it takes to verify if the alarm is genuine added with the time it takes security staff to arrive at the scene. The necessary resilience, delay time, must be achieved with a combination of organizational, constructional and electronic measures.

A security system with biometric access control like iris scanners and fingerprint scanners is one of the highest possible levels of security. It is the least fraud-sensitive security application on the market and

at first glance the solution for access control to clinical treatment units using HASS. However, consideration of suitability in daily use of such system is necessary before implementation. When determining the access security systems the safety of patients should be taken into account as well. If it takes medical staff too long to reach patients in case of an emergency, such as a cardiac arrest, the access security system will soon be bypassed or switched off completely. There should be an equilibrium between securing the HASS and using the sources safely by authorized staff. Cooperation with the medical staff and yearly security awareness training is essential to maintain the desired level of security. In some countries the use of biometrical access control can lead to privacy issues.

The new General Data Protection Regulation (GDPR) [8] has been taken into effect in the European Union (EU) on May 25th, 2018. This means that from that date onwards, the same privacy legislation applies throughout the EU. The GDPR provides specific rules for the processing of 'biometric data'. This means that the use of biometrics in security will only be permitted if the national legislature makes an arrangement for this. Although this new Regulation was adopted by all EU member states on April 26th, 2016 there are countries where the national legislature has not yet made an exception act, hence the use of biometrical access is in those countries is indicated illegal. Nevertheless the security of HASS in the hospitals within the EU is arranged sufficiently.

6 TRANSPORT

Transportation of Category 1 & 2 sources remains challenging. Transport of radioactive material can be seen as an interim mobile state between two stationary areas. The potential consequences are identical and therefore the level of security needs to be at least comparable to the security level of the hospital. These shipments are mainly achieved by air or by road transport and therefore constructional security is limited. In order to maintain the same security regime as in the stationary areas operators[1] have to rely on organizational and electronic measures.

6.1 Organisational measures

Part of the organisational measures will be to obtain a Certificate of Trustworthiness and Reliability for all staff involved. Due to privacy legislation such certificates are sometimes hard to obtain. In addition during a source exchange the double quantity of sources is present at the site of the hospital, which makes it a more desirable target of a perpetrator. Prior knowledge of vital transport information like the day and arrival time of the consignment and the security measures applied to the transport should be restricted to the minimum number of staff required.

The most optimal situation is that the hospital staff is unaware when exactly the HASS will arrive. The hospital and security staff receive a time-slot of 48 hours in which the transport will arrive. Just ten minutes in advance the carrier calls to ask the security staff to be ready to escort the consignment. In several hospitals this procedure has been implemented successfully. Unfortunately, this is not always possible to realize, certainly not in case of Leksell Gamma Knife® sources, due to the size of transport.

6.2 Transport security

IAEA's recommendations [9] that deserve extra attention, are security requirements for radioactive material in transport that are meant to be implemented by individual governments to minimize the likelihood of loss of control or malicious acts. Regrettably a solid implementation is not the norm and this applies in particular to politically unstable regions. Another imperfection is that national legislations are not "matched" at the borders to enable seamless security measures for international transports. This is particularly visible with cross-border road transport of HASS. In one state the transport is provided

[1] The term 'operator' is used by the IAEA to describe an entity authorized to use, store or transport radioactive material. Such an entity would normally hold a license from a competent authority.

with an armed motorcade, but as soon as the transport is crossing the border the motorcade turns around and no additional security is escorting for the remainder of the transport.

6.3 Source logistics

Since the hospital purchases the HASS, ownership of the source(s) and responsibility for clinical use and storage, reside with the hospital from the moment of delivery. When taking a closer look at the source exchange procedure of the Brachytherapy afterloaders a few shortcomings become visible. During the source exchange the task of the Field Service Engineer (FSE) is to responsibly carry out the exchange and to prepare the old source for return transport to the manufacturer. This part of the process is regulated by the license of the legal manufacturer. The manufacturer is obliged to have the depleted source returned and disposed of in a correct manner. However, sending the consignment from the hospital back to the manufacturer (or approved disposal site) appertain to the duties of the radiation expert of the hospital. After all, the hospital has become the consignor. Therefore the hospital needs to arrange a legitimate transport company to pick up the consignment. The IAEA's recommended 'basic security level' should be maintained and in case of Leksell Gamma Knife® source exchange, the security level should even be elevated to the 'enhanced security level'. Experience has taught that the expertise of such specific transport regarding these specific requirements is not available in every hospital, and therefore must be supported by staff of the legal manufacturer.

6.4 Orphan sources

Due to unclear division of tasks source return shipments remain on the hospital premises longer than strictly necessary. In some hospitals obsolete sources are gathered in unequipped or inappropriately equipped storage rooms. Combined with a lack of awareness of potential health and security risks, a stagnating source-return process or a downright refusal of hospital management to carry the transportation costs, these sources pose a serious hazard of becoming "loss of control" sources.

As mentioned in §4, the legal manufacturer carries out vital checks prior to shipment (e.g. RAM license checks of recipient hospitals). Despite controlling inappropriate storage of depleted sources is in fact a governmental task, it is in the interest of manufacturers to ensure the maximum activity limits of recipient hospitals is not exceeded. By monitoring hospital licenses the legal manufacturer should refrain from sending new sources if the combined activity of the stationary sources, exceeds the amount of activity stated in the permit. To keep track of radioactive sources which are no longer in possession nor property of the legal manufacturer, the legal manufacturer is forced to make extra time and resources investments. The benefit of decrease of IAEA [10] interventions in security and removal of obsolete radioactive sources is to be kept in mind when weighting these extra tasks.

In the same light the occurrence of orphan sources due to the bankruptcy of -mainly private- hospitals must be considered. Once a hospital is closed down by court of law, it becomes very difficult to repatriate and secure the radioactive source. Not all countries have regulated a financial guarantee for HASS sources in the hospital permit. Putting in place a financial guarantee in the beginning of the sales process can safeguard repatriation of sealed sources in the event of hospital closure. Since not all countries have regulated financial guarantees for HASS by government it is important to check early in the sales process if this is the case. In countries where no financial guarantees exist a financial guarantee for repatriating HASS must be part of the financial sales contract. Instead of keeping track of financial guarantees as well as changing situations in countries a better and more feasible solution could lead to a more sustainable way of securing HASS from cradle to grave. If the hospitals would rent or lease radioactive sources from the legal manufacturer, the legal manufacturer could keep a better track of the sources. After a source exchange the FSE could send the old sources back to the manufacturer and would act as consignor. Although it would be a completely different business plan, after all transport, safety and security are not core business process for a radioactive source manufacturer, but it might work. From a security point of view it is recommended to investigate this option.

6.5 Electronic measures

6.5.1 Data security

In addition to personal data and employee certificates, transport information and customer data have to be stored in databases of the manufacturer. Privacy and security sensitive information that must be secure and at the same time accessible to daily work. A balance should be found between workability and security. Of course, a hardware encrypted SSD is the safest solution to save the data, but how do you share this data with the carrier and other operators? And in particular how to work with it in the field, close to where the actual operations occur, where many "heterogeneous" entities need to be informed about the aspects of the transport. At this moment the industry is lacking a clear standard, a clear directive how to work with this kind of sensitive information. Safety (external servers, regular backups) and security (keep sensitive information contained to those that need information to execute their profession) seems to be each other's opposites.

One of the digital challenges is the safe forwarding of the transport information. The basic idea is that as few people as possible are aware of the transports. In order to keep a brachytherapy HASS transport running smoothly, about thirty people spread over various operators are needed. In these situations it is vital that the information remains fragmented and that the number of people who have the total overview of the project is kept to a minimum. All operators must work meticulously and diligently in their implementation of secure storage of data and the methods used to communicate confidential information. Staff should only be authorized to information needed to perform their tasks.

Especially when communicating information with other operators it is important that such projects have multiple authorization levels. In addition to standard password protections, it is advisable to work with multi authentication such as a two-factor authentication (2FA). In contrast to two-step verification where a random number is sent by SMS, the user also needs a physical token to log in. This could be a telephone, an authenticator or a USB stick. Biometric information can also be used to gain access to more vital information, but again this could cause a privacy violation.

6.5.2 Ransomware

In May 2017, a worldwide cyberattack by the WannaCry ransomware crypto worm, targeted computers running the Microsoft Windows operating system by encrypting data and demanding ransom payments in Bitcoin cryptocurrency. Large seaports like Rotterdam [11] thought they were immune to cyber-attacks that are increasingly sweeping through the business world [12]. However, Rotterdam was shut down almost completely by WannaCry. It took Port of Rotterdam more than a week to recover from the attack. The ransomware campaign was unprecedented in scale according to Europol [13] which estimates that around 200,000 computers were infected across 150 countries. Transport companies, airports and healthcare organizations were among those affected.

6.5.3 Hacking

Hacking computers of transport companies is not a new phenomenon. Hackers are known to help Somali pirates to choose their targets by viewing navigational data online [14] and hackers infiltrated computers connected to the Belgian port of Antwerp locating specific containers with smuggled drugs and deleted the records.[15]

To keep track of consignments carrying radioactive materials an 'enhanced security level' with tracking devices is recommended in the IAEA Nuclear Security Series no.9. This could be a simple track and trace system with bar or QR-codes. Unfortunately such a system only tracks in which section of transport the consignment may be found, not the actually whereabouts of the HASS. More preferable would be an on-line tracking system in conjunction with a communication system in the conveyance. These on-line tracking devices are using Global Positioning System (GPS) navigation to transmit their position. As soon as the transport diverts from the pre-set routes an alarm will be sent out and immediate action can be taken.

6.5.4 Spoofing

In the aftermath of the WannaCry ransomware it became clear how vulnerable GPS based navigation is by another type of cyberattack. In June 2017, a number of ships in the Black Sea reported anomalies with their GPS-derived position [16]. Their navigation system showed their ships where positioned on an airfield some 30 kilometres away. This occurrence may have been created possibly with a GPS-spoofing device which overrules the signals coming from the satellites feeding incorrect position data into the navigation system. In 2013, this was already tested by deliberately breaking into the computer system of a super yacht [17]. In this way ships can be diverted using a laptop, a small antenna and a GPS-spoofing device. It might be staggering to realize that the necessary equipment to spoof GPS signals is just a $15 device [18]. In theory, any transport device can be diverted or sabotaged by hacking computer systems.

7 CONCLUSION

IAEA gives a clear indication that HASS are a high risk target. In order to prevent any unauthorized access the occurrence of orphan source should be prevented. Pre-sales assessment for any economic sanctions or embargoes must be executed. The legal manufacturer should refrain from sending new sources if the combined activity of the stationary sources exceeds the amount of activity stated in the permit of the hospital. Regular checks whether the hospital has a valid relevant RAM-license is mandatory and a financial guarantee for repatriation of HASS must be in place.

The main goal is to reach an equilibrium between securing the HASS and using the sources safely by authorized staff. Cooperation with the medical staff and annual security awareness training is essential.

Vital transport information of consignments containing radioactive material and the security measures applied to the transport should be restricted to the minimum number of staff required. Tracking consignments carrying HASS using Global Positioning System (GPS) navigation to transmit their position is recommended. Although enhancing organisational security measures are preferable, as GPS navigation and computers may be exposed to respectively spoofing and hacking.

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Waste management

Radioecology of tritium in sewage water discharges of a landfill in northern Germany

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Abstract. In purified sewage water ("permeate") of a landfill in northern Germany, a tritium concentration of 576 Bq/l was measured in 2011. From the data gathered up to 2016, it can be estimated that a total activity inventory of about 330 GBq of tritium occurred in 2011. For assessing the radiological significance of the measured tritium concentrations, radiation exposures of persons of the public were modeled based on the German calculation guide "AVV zu § 47 StrlSchV" [1]. In some parameters, the specific local conditions differ from the assumptions used to derive the clearance values of the German Radiation Protection Ordinance (StrlSchV). The highest annual doses were obtained for the age group of infants and are below 1 µSv for 2015. The main contribution results from the consumption of breast milk. A comparison of the site-related model parameters with parameters used for the derivation of clearance values for landfills [2] shows that the scenario "use of surface water" in [2] is not conservative for the landfill considered here.

KEYWORDS: *Landfill, clearance, tritium, radioecology, dose, persons of the public*

1. INTRODUCTION

The purified leachate (permeate) of a landfill in northern Germany has been inspected for radioactive contamination since 2011 by the responsible state agency. The first result obtained from sampling at the discharge point of permeate into a ditch was 567 Bq/l. Although this concentration by no means represents an extreme value of the tritium concentrations in German landfill leachates, it exceeded the drinking water standard of 100 Bq/l. This finding has raised the concern of citizens regarding the origin of the tritium. For being able to elucidate these concerns on a professional basis, it was decided to launch further studies.

In Figure 1 the tritium concentrations in the permeate obtained from samplings up to 2016 are shown. Due to the treatment of the leachate, which removes organic matter from the water nearly complete, the chemical form of tritium can be assumed to be exclusively HTO.

Figure 1: Temporal change of tritium concentration in leachate (permeate) of the landfill site

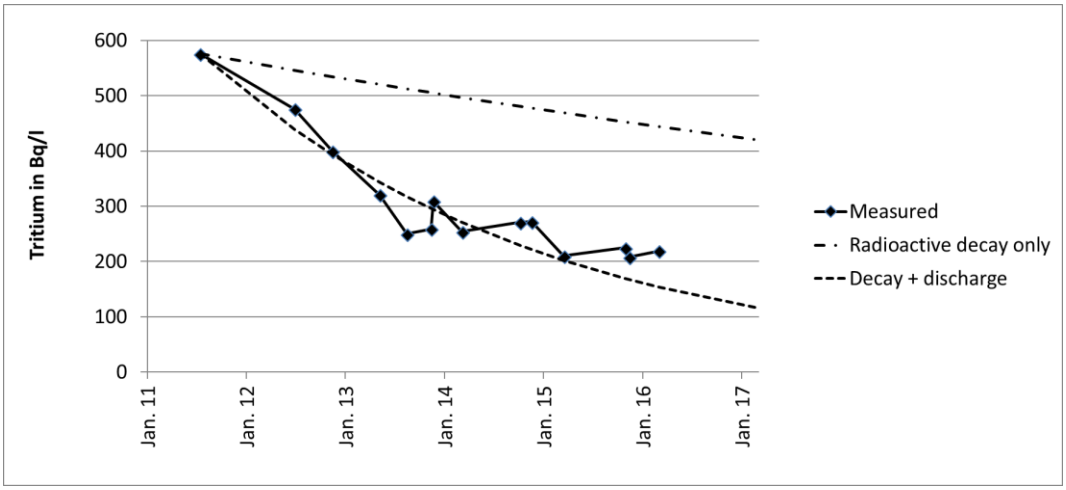
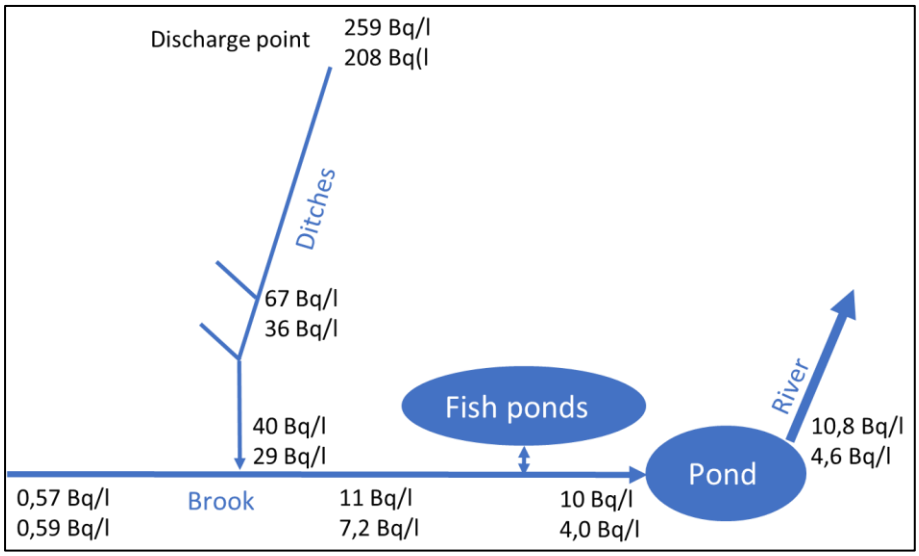


Figure 2 shows tritium concentrations in the watercourses downstream the discharge point of the landfill site. The displayed values demonstrate a significant dilution of tritium during flowing through the ditches system. However, the tritium concentrations exceed the natural level of less than 0.6 Bq/l that occurs upstream to the inflow of the ditches in the brook, and the influence of the landfill discharges is evident.

Figure 2: Schematized watercourse system with measured tritium concentrations in 2013 (upper value) and 2015 (lower value)



2. ESTIMATION OF THE TRITIUM INVENTORY

From measured concentrations and data of the landfill leachate quantities, the temporal change of total tritium discharge was analyzed using various models, and a tritium balance of the landfill body was estimated.

Total tritium discharge of the landfill amounted to approximately 80-90 GBq in 2011. This discharge decreased up to 2016 to approximately 20 GBq. Fitting the temporal change with an exponential model demonstrated: leaching contributes 4.1 times more to the decrease in tritium concentration than the radioactive decay. The total inventory contributing to the observed discharges was estimated by using the exponential model to approximately 330 GBq for the reference date 01.01.2011, and to less than 100 GBq in spring 2016.

The origin of tritium inventory could not be unambiguously clarified. Waste from an NPP that was deposited on the landfill after clearance was checked as a possible source of tritium but could be excluded with a high level of confidence. More likely seems that waste containing consumer goods caused the tritium inventory.

3. DOSE ESTIMATIONS FOR PERSONS OF THE PUBLIC

For assessing the radiological significance of the measured tritium concentrations, dose estimates were carried out. A rough estimate showed the inhalation doses of evaporated HTO to be so low, even for workers, that the contribution of inhalation can be disregarded for the exposure of persons of the public. Moreover, due to site conditions, consumption of contaminated drinking water by persons can be excluded.

The exposure pathway with the highest radiological significance for persons of the public results from the discharge of the purified leachate (permeate). Even though neither regular irrigation of garden areas nor

livestock drinking from the influenced watercourses was known to be indeed occurring the following exposure pathways and most unfavorable receiving points were considered [1]:

- Consumption of locally produced vegetable products and irrigation of plants from the brook after the inflow of the drainage ditch (see Figure 2).
- Consumption of milk, dairy products, and meat from livestock that drink water from the ditch before discharging into the brook (see Figure 2).
- Consumption of fish that lived in the brook downstream the inflow of the drainage ditch.

Since the intake of tritium via drinking of cattle leads to higher tritium concentrations in the milk and meat than the irrigation of the pasture areas, the pathways irrigation → fodder plant → cow →- milk/meat are covered by the model assumptions. According to the provisions of the AVV [1] for tritium in the form of tritiated water, the activity concentration in vegetable products and pasture plants should be set equal to that of irrigation water. Furthermore, the pasture-period was set to be 0.5 years, and the tritium concentration in surface water limits the activity in meat and milk during this time.

The tritium concentration of the surface water at different sites of the watercourses system was modeled from the concentration in the permeate using dilution factors.

With these model assumptions and a mean tritium concentration in the permeate of 217 Bq/l (average of measurements from 2015) and taking into account a background activity of 0.6 Bq/l, the results listed in Table 1 were obtained. As expected, the highest exposed reference person is an infant, and the most crucial exposure pathway is the breast milk of a mother who exclusively eats locally grown food.

Table 1: Annual effective doses of reference persons calculated by the model parameters from the discharge of tritium-containing permeate of the landfill leachate

pn=1 (100 % locally produced food)	<1 a µSv	1-2 a µSv	2-7 a µSv	7-12 a µSv	12-17 a µSv	>17 a µSv
Fresh fruit (Ob)	0.031	0.042	0.039	0.029	0.039	0.039
Root vegetables (Wg)	0.037	0.037	0.027	0.024	0.027	0.027
Leafy vegetables (Bl)	0.004	0.006	0.004	0.004	0.004	0.004
Vegetables (Gm)	0.006	0.016	0.018	0.016	0.018	0.018
Milk (Mi)	0.231	0.616	0.398	0.314	0.246	0.188
Meat (Fl)	0.017	0.033	0.083	0.080	0.077	0.087
Fish (Fi)	0.0013	0.0058	0.0037	0.0042	0.0036	0.0054
Breast milk (MM)	0.556					
Sum	**0.88**	**0.76**	**0.57**	**0.47**	**0.41**	**0.37**

4. COMPARISON WITH CLEARANCE MODELS

The German clearance values for the disposal of waste on landfills (Annex III Table 1 Column 9a and 9c StrlSchV) are based on the radioecological models described in [2]. In Table 2 the parameter values applied in [2] for modeling the scenario "Use of surface water" are listed and compared to values of the landfill. The essential result of the comparison is: The generic landfill model described in [2] is not conservative in the specific case analyzed here. In particular, the dilution of the discharged permeate of the landfill in the receiving water is about 1.5 orders of magnitude lower (factor 39) than assumed in [2].

Table 2: Comparison of parameter values according to [2] for the scenario „utilization of surface water“ with real values of the considered landfill site

Parameter	Unit	Value according to [2]	Value of the landfill considered here
Discharge landfill	m³/a	27.000	100.000
Throughput wastewater treatment plant	m³/a	1.000.000	100.000
Runoff receiving watercourse	m³/a	31.536.000	Ca. 3.000.000
Amount sewage sludge (TM)	Mg/a	500	(k.A.)
Distribution sewage sludge	Mg/(a ha)	1	0

Reasons for this significant difference are the following:

- The considered landfill is larger than the reference landfill. Therefore more seepage water accumulates.
- The considered landfill site has an own treatment plant for sewage water. The sewage sludge is not spread on arable land but disposed of on the landfill itself.
- Runoff of the receiving water (brook) is significantly lower than assumed in the model according to [2].

As a consequence, the tritium inventory of 6000 GBq (100 Mg x 6E+04 Bq/g), which can be annually released under the StrlSchV for disposal on a landfill, will result in doses significantly exceeding 10 µSv if applying the rule of three in case of the landfill site considered here.

5. CONCLUSIONS

In the context of the still very high levels of public and media attention on radioactivity as well as a skeptical stakeholder stance on the clearance of NPP waste (in Germany), analyses of discharge waters of landfills should be done accompanying to the disposal of radioactively contaminated waste on landfill sites after release from regulatory surveillance. Landfill operators that intend to dispose of radioactively contaminated waste should determine the initial levels of radionuclides in the leachate before accepting such waste. The data obtained from such measurements should be used to check the parameters of radioecological models. This feedback could help in further developing the model basis and in adapting models to the actual conditions.

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Radioactive waste management strategy in the Netherlands

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ABSTRACT
The Netherlands literally means low lands; large parts of the country are indeed below the sea level. In their century-long struggle with water, the Dutch have relied on an approach of long-term, inclusive and pragmatic solutions. This approach is also reflected in the radioactive waste management strategy.
In the Netherlands the policy is based on the philosophy that hazardous materials must be isolated, controlled and monitored. This is done by securing containment of radioactive waste in terms of organisational aspects as well as in terms of physical treatment. Really long-term, i.e. at least 100 years, storage in above ground engineered structures of low-, medium- and high-level waste, including spent fuel and (TE)NORM is the first element in the Dutch policy.
Secondly, and included in the policy are all necessary steps to be taken for the longer term. The burden of the waste will not solely be transferred to the next generations because a clear system of liabilities is created and the availability of finances is guaranteed for the future. For a country with a small nuclear power programme and an important amount of waste from other applications of radioactive materials, it is a practical solution that works.

1. INTRODUCTION

In the Netherlands there are some 200 producers of radioactive waste, varying from nuclear power plants, research establishments, all sorts of industries and hospitals. Most of them generate only small volumes of low- and medium-level waste. These small volumes however cover a wide range of waste forms: solids, liquids of all natures, slurries, animal carcasses, machines, equipment, sealed sources etc. Some processing industries generate larger volumes of solid low-level radioactive material. The concentration of the radioactivity is low but the nuclides present are alpha emitting nuclides, which are highly radiotoxic. These materials are commonly called NORM or TENORM waste: (Technically Enhanced) Naturally Occurring Radioactive Material. Most of the NORM waste can be disposed of at one of the landfills authorized to accept the waste. A smaller volume of NORM cannot be accepted at these landfills and is brought to COVRA. This includes also the depleted uranium resulting from the uranium enrichment plant in the Netherlands.

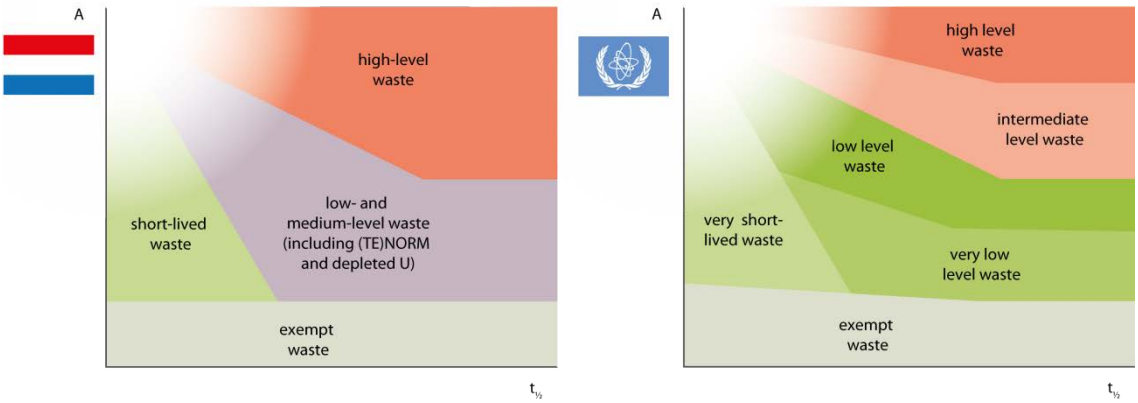

Figure 1: Waste classification in the Netherlands compared to IAEA classification.

Even smaller volumes of high-level waste are produced by the nuclear power plants and by the research reactors. Research reactors at Petten and Delft were constructed in the late fifties. The research reactor at Petten is also used for radioisotope production: important quantities of molybdenum for medical use are produced here. In the late sixties and early seventies two nuclear power plants in Dodewaard and Borssele started producing electricity. The Dodewaard reactor was shut down in 1997. The Borssele plant will continue to operate as long as safety and economics permit: foreseen is a lifetime of sixty years till 2033.

2. LEGAL FRAMEWORK

The legal foundation of the policy governing nuclear activities, including all activities with radioactive materials, is contained in the Nuclear Energy Act and the legislation based on it, together with the licences and relevant licence conditions. All these documents can be found via www.overheid.nl. The act lays down rules in the nuclear field, makes provisions for radiation protection and designates the different competent authorities.

This institutional arrangement for radioactive waste management follows the IAEA 'classical triangle' model. The model separates the three roles of the regulator, the waste producer and the waste manager/disposer. Each has separate responsibilities and must exhibit independence from the other. The nuclear sector in the Netherlands is regulated by the Authority for Nuclear Safety and Radiation Protection (ANVS), which began operating on 1 January 2015. Before, the nuclear expertise and the regulatory functions had been fragmented, and spread out over several ministries and executing organisation. The ANVS drafts legislation, develops safety requirements, issues permits and carries out inspections. The ANVS will also be jointly responsible for emergency preparedness in the event of incidents that could result in the release of radiation.

The application of radioactive materials is permitted only if licensed under the Nuclear Energy Act and this act stipulates that a licensee can dispose of its radioactive waste only by handing it over to the authorised waste management organisation. As such the Central Organisation for Radioactive Waste (COVRA) is the organisation authorised by the government of The Netherlands.

All activities with radioactive substances, excluding shipments, have to comply with the rules laid down in the Radiation Protection Decree, which is part of the Nuclear Energy Act. In the decree a definition is given for radioactive waste:

> *Radioactive waste: a substance can be considered to be waste if for this substance no use, reuse or recycling is foreseen and the substance is not to be discharged.*

This definition is in compliance with the IAEA Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management. In the decree also the rules for exemption and clearance of radioactive materials are described. The limits themselves are laid down in specific regulation on the execution of radiation protection and are based on the Basic Safety Standards Directive (2013/59Euratom) .

3. ISOLATE, CONTROL AND MONITOR

In general, waste prevention and reuse of materials is an important environmental goal. Waste producers are required to prevent the generation of radioactive waste as much as reasonably achievable. Recycling options must also be considered to this end. If the activity has decreased below the clearance levels, radioactive material can be released from supervisory control and reused or managed as conventional waste. Also the justification of the application of radioactive materials – a condition for a license - contributes to the goal of minimization. But when the production of hazardous waste is unavoidable, then these materials must be isolated, controlled and monitored. Of course this philosophy is also applied to radioactive waste management. Radioactive materials for which no use, reuse or recycling is foreseen, are isolated from our living environment. The radioactive materials are contained in such way that we control the risk of the ionizing radiation. Finally, the waste is monitored until a passively safe situation has been achieved.

Over thirty years already, the government of the Netherlands follows a straightforward policy based on the above-mentioned philosophy. Radioactive waste must be safely managed as long as it poses a risk to humans and the environment. All kinds and categories of radioactive waste are stored for at least 100 years above ground, in engineered structures, which allow retrieval at all times. This long-term storage, together with a central treatment facility is considered as a normal industrial activity and is located on one single industrial site. COVRA has been established to manage all Dutch radioactive waste. In principle, all the costs for radioactive waste management are borne by the waste producers.

This includes all costs incurred by COVRA for collection, conditioning, storage and disposal. These costs are charged to the waste generators through COVRA's fees.

Geological disposal is currently regarded as the only safe option for the management of radioactive waste over the very long term. Any kind of storage (including long-term storage) of radioactive waste will always be a temporary solution and is not considered as an alternative to disposal. Long-term storage is essential because direct disposal is not yet feasible in the Netherlands. A disposal site for this type of waste is not available, the public acceptability for geologic disposal is low and the small volumes of waste available for direct disposal do not require an immediate final solution. Also the financial burden of a direct disposal facility is prohibitive for the small quantities concerned. Therefore, research is performed on the geological disposal possibilities within the Netherlands or within an international framework. In addition, a disposal fund has been established. The money in the fund needs to grow over a substantial time period to reach the required amounts.

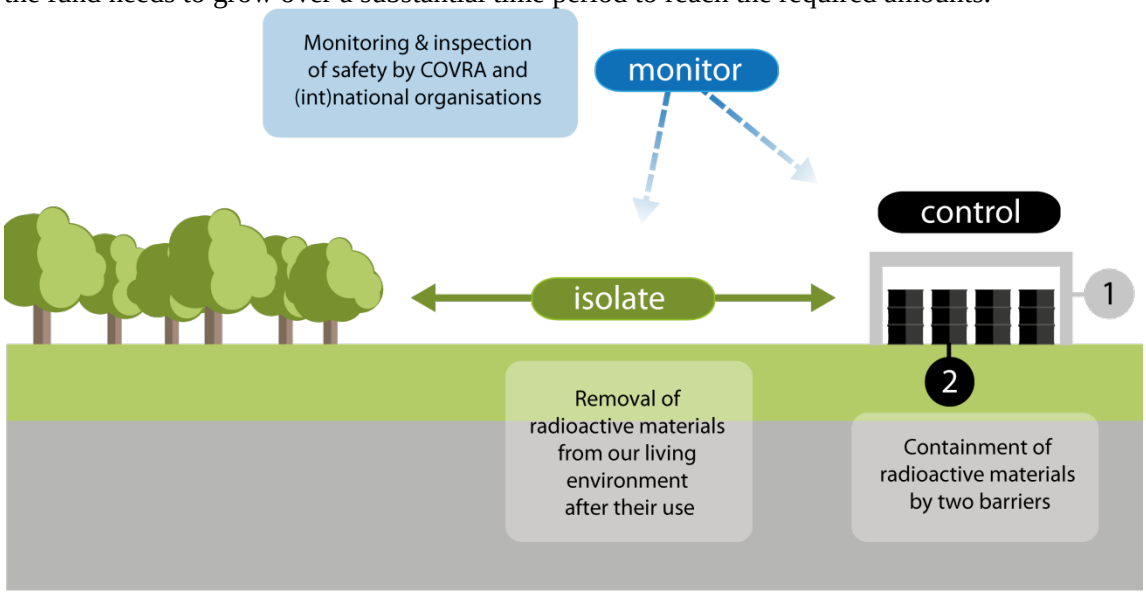

Figure 2: Isolate, control and monitor

The choice to store for a long time was well considered and was not taken as a 'wait and see' option. This is clearly demonstrated by the fact that integral parts of the policy are: the establishment of the capital growth fund and a clear choice for the ownership of the waste within COVRA. This policy does not leave the burden of waste generated today to future generations. Only the execution of the disposal action is left as a task for the future. A disposal solution is at principle available and the money will become available in the capital growth fund.

4. CENTRAL ORGANISATION FOR RADIOACTRIVE WASTE (COVRA)

COVRA is the only organisation in the Netherlands that is legally authorized as a radioactive waste retrieval company. COVRA is a joint stock company (JSC). Since 2002 all shares have been held by the Dutch government. As COVRA takes over ownership of the waste, its fees must at a minimum cover costs, now and in the future, hence also including future disposal costs. The Dutch government does not provide structural financial support.

COVRA was founded in 1982 and has grown to become the company which provides the sustainable management of Dutch radioactive waste. During its first ten years the organisation and logistics for the collection of Dutch radioactive waste was established; a location was found for the company activities and all activities at the temporary location in Petten were transferred to the long-term location in Vlissingen-Oost. In the years that followed, the facilities for the processing and storage of all types of radioactive waste were constructed and operationalised, and a stable and professional organisation was formed.

COVRA has a small but well-equipped organisation, with personnel and material for the collection, processing and storage of all types of radioactive waste. The size of COVRA is derived from the organisation's tasks. COVRA:

- accepts ownership and full liability for the radioactive waste transferred,
- collects and transports the waste to be treated,
- treats low- and medium-level radioactive waste, creating an encapsulated stable product,
- stores low-, medium- and high-level radioactive waste in purpose-designed buildings,
- monitors and controls all stored waste and accurately registers all waste,
- ensures that the radiation emitted during processing and storage is kept within the legal limits,
- coordinates research on geological disposal,
- Informs actively the population about these tasks and about the work done.

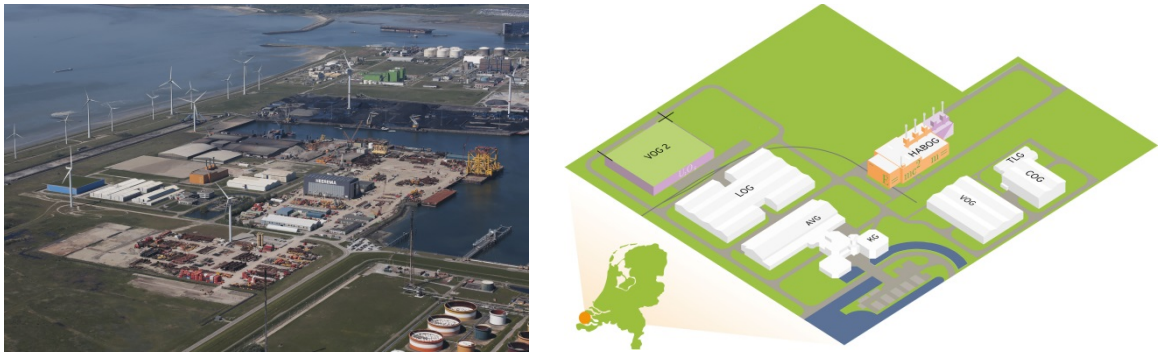

Figure 3: air photo of the COVRA site.
KG= Office building, AVG= treatment building for LMLW, LOG = storage building for LMLW, HABOG = treatment and storage building for HLW (purple part is planned extension), VOG = depleted uranium storage building, COG = container storage building (calcinated waste from phosphor production), TLG = Transport and logistics building.

COVRA has a site available of about 25 ha at the harbour and industrial area Vlissingen-Oost. This site offers enough space for the storage of existing waste and the waste expected to be produced in the next hundred years. At this site COVRA operates a facility including the following:

- an office building including an exhibition centre;
- a building for the treatment of low- and medium-level wastes;
- various storage buildings for conditioned low- and medium-level waste, for NORM waste, (including depleted uranium) and for high-level waste.

5. TRANSPARENCY AND COMMUNICATION

Transparency and communication are an integrated part of the operations of COVRA. Because of the long-term activities, COVRA can only function effectively when it has a good, open and transparent relationship with the public and particularly with the local population.

When COVRA in 1992 constructed its facilities at the present site, it took it as a challenge to build a good relationship with the local population. From the beginning attention was paid to psychological and emotional factors in the design of the technical facilities. All the installations have been designed so that visitors can have a look at the work as it is done. Creating a good working atmosphere open to visitors was aimed at. The idea was not to create just a visitors centre at the site, but to make the site and all of its facilities the visitors centre.

HABOG features a bright orange exterior and the prominent display of Albert Einstein's equation $E=mc^2$ and Max Planck's $E=h\nu$. The high-level waste inside HABOG is planned to remain there for at least 100 years, during which time its radioactivity will decrease through decay and hence the heat

production of the waste will decline to insignificant levels. This process is symbolised by the colour of the building's exterior, which is to be repainted every 20 years in lighter and lighter shades of orange until reaching white. The orange colour was chosen because it is halfway between red and green, colours that usually symbolise respectively danger and safety.

HABOG is more than an interim storage, it is a work of art created by the artist William Verstraeten. As a work of art it is a communication tool. It helps to explain the concept of radioactivity in simple not technical way. It is an 'attraction' that draws people to the COVRA facilities, people from the region, but also from all over the country and abroad. It provokes questions and stimulates discussion about radioactive waste and its management. People remember the story of the building, the changing colour which helps them to understand the process of decay and the safety of radioactive waste storage.

The new storage buildings planned for storage of depleted uranium as well as the extension of the HLW-building offer further opportunities to communicate. To start a dialogue. And to tell a positive story about the management of radioactive waste. Now that the storage capacity both for depleted uranium as well as for high-level waste has to be extended, art will be included again. Currently, the construction of a new building for the storage of depleted uranium has started. This new building is necessary to accommodate the generation of the depleted uranium. The artist William Verstraeten created again a spectacular new work. The building will become the largest sun dial of Europe, indicating the phenomenon of time as important factor to render radioactivity harmless.

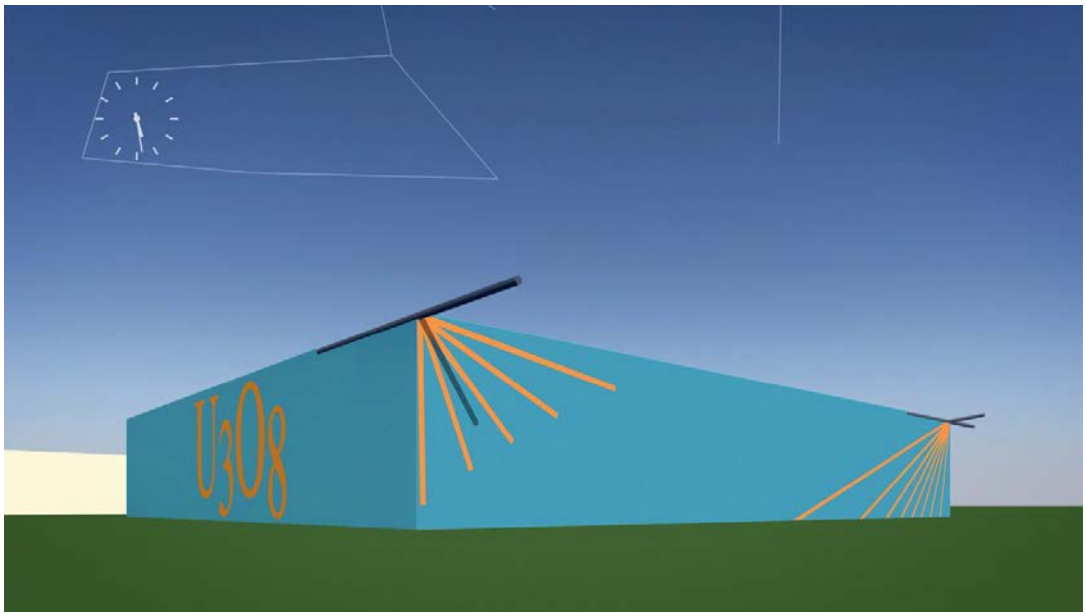

Figure 4: The design of the new building for depleted uranium (VOG2).

6. QUANTITIES, TREATMENT & STORAGE

Figure 5 gives an overview of annual waste production in the Netherlands and the treatment of the waste. Most of the NORM is landfilled and almost half of the LMLW is reused. This is in line with the policy of prevention and reuse of radioactive waste. Radioactive waste, that has been reused or landfilled at a conventional site, is no longer considered as radioactive waste. Only the radioactive waste that cannot be avoided or reused is transferred to COVRA. All of the HLW and parts of the LMLW and NORM is transferred to COVRA. Important to note here that SF from the nuclear power plant is not regarded as waste. It is recycled (reprocessed) in France and resulting HLW is shipped to COVRA.

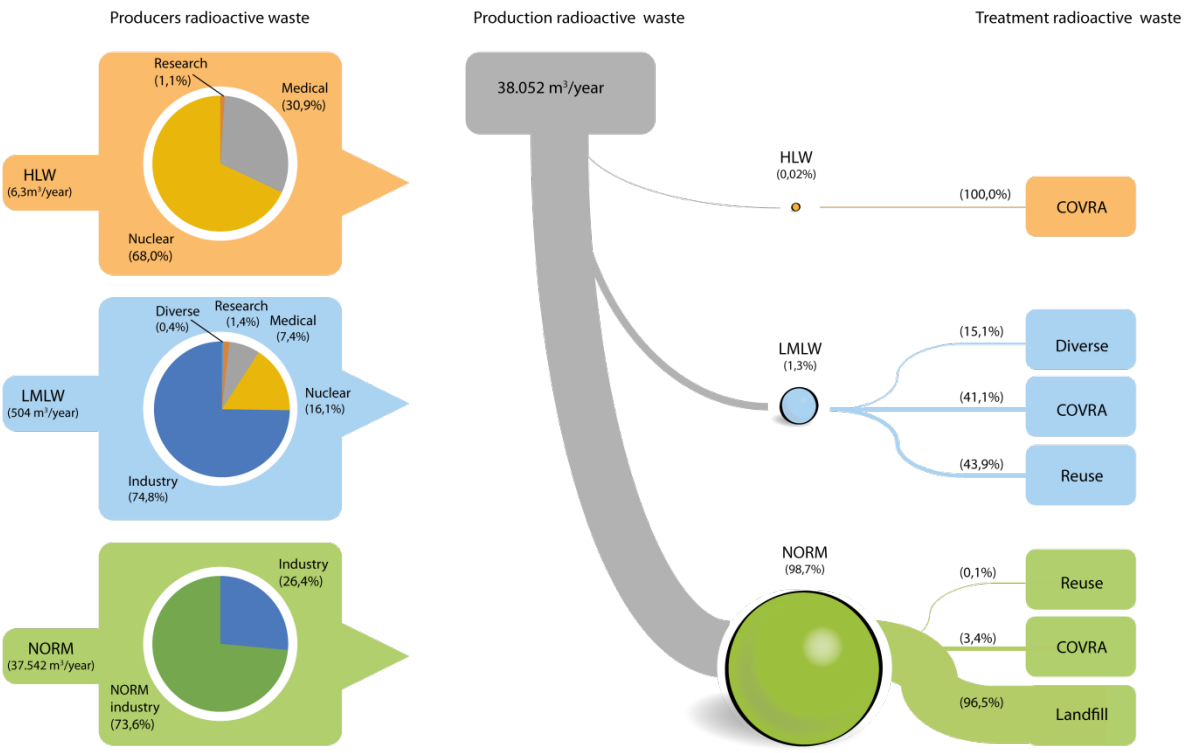

Figure 5: Radioactive waste generation and treatment in the Netherlands

Low- and medium-level waste

Annually some 500 m^3 of low- and medium-level waste is produced. About half of it is conditioned at the COVRA facilities. Only resins and evaporator sludges are directly conditioned with cement at the nuclear power plant Borssele. Other wastes are reused, contaminated metals for example.

At COVRA, various installations are available in the treatment building, such as a super compactor, an incinerator for biological waste, an incinerator for organic liquids, shearing and cutting installations, a cementing station and wastewater treatment systems. The final conditioned waste form is a cemented package of 200 or 1000 litre. These packages are stored in concrete storage buildings.

At the moment COVRA operates four storage units, each unit has a capacity of approximately 5,000 m^3 conditioned waste. A reception bay connects the storage units. Blocks of waste packages are placed in rows, which leave open corridors for inspection. Lower dose rate packages are stored along the outer walls of the modules, and on the top layers in order to provide additional shielding for higher dose rate packages at the interior. Humidity in the storage building is kept below 60% in order to prevent condensation of air moisture on the packages.

(TE)NORM and depleted uranium

Most of the (TE)NORM waste is landfilled. Some 3 percent of the (TE)NORM generated is shipped to COVRA. Depleted uranium from uranium enrichment is (TE)NORM but generally not regarded as waste. The depleted material still contains uranium that -depending on economic factors- can be used as feed material for an enrichment process. However, when larger quantities are in stock and real use is not foreseen within some tens of years, the product comes close to the definition of waste. In the Netherlands a maximum of 5,500 tonnes of depleted uranium is produced per year. In this material ^{238}U and all daughters are present. The activity is around 14,000 Bq/gram. The depleted uranium is stored in DV-70 containers, that can hold some 11 tonnes.

The depleted uranium is produced as UF_6. This is not a chemically stable product suitable for long-term storage and therefore this depleted UF_6 is converted in the stable U_3O_8. This is done in

installations in France. Currently, one building with six storage modules, each with a storage capacity of 650 containers is full and a new building (commissioned in 2017) is in operation (figure 4). The new storage building has 3 modules, each with a capacity of almost 2200 containers per module (i.e. more than 15 years of generation in total).

Another (TE)NORM waste stored at COVRA is a calcined product resulting from the production of phosphor in a dry/high-temperature process. As the depleted uranium, it is a stable product that does not need further conditioning to assure safe storage. Economics played an important role in the implementation of this storage solution. As activity levels are relatively low (alpha-emitting radionuclides) radiation levels are low. Little shielding is required. The waste is stored in a building consisting of a galvanised steel construction frame with steel insulation panels, designed for 150 years lifetime with minimum maintenance. This building also, can be modularly expanded. After decay of the radionuclides the material will be cleared and brought outside the nuclear domain. Therefore, this waste is not conditioned either. The calcinate produced at the phosphor plant is dried at the plant and collected and stored in a 20-ft container. The phosphor plant ceased operation in 2013 and no more calcinate waste from its operations is expected.

High-level waste

All the spent fuel of the Borssele plant is reprocessed by Areva (Orano) in France. The resulting reprocessing waste will be sent back to the Netherlands. The vitrified residues (CSD-v) and compacted hulls and endcaps (CSD-c) will be stored in a storage vault that was commissioned in 2003. High- and low-enriched spent fuel from the research reactors, uranium collection (UCW) filters from the medical isotope production and some other high-level (legacy) waste from research activities will be stored in this building also. A distinction is made between heat- and non-heat-generating waste, since the former category requires cooling. The non-heat-generating waste is, remotely controlled, stacked in well-shielded storage areas. The heat generating waste such as the vitrified residues, is put into vertical storage wells cooled by natural ventilation. This method is proven technology in the storage facilities of Areva at La Hague.

The spent fuel elements of the research reactors are delivered to COVRA in a cask containing a basket with 33 elements. The basket with elements is removed from the cask and placed in a steel canister, which is welded tight and filled with helium. These sealed canisters are placed in wells, in the same way as the vitrified residues. The wells are filled with argon to prevent corrosion. The UCW filters follow the same route.

Figure 6: Loading room for heat-generating high-level waste.

7. DISMANTLING & DISPOSAL

The necessary levels of isolation are achieved in a first phase by containing the radioactive materials in safe and secure storage facilities. Storage of radioactive waste in surface facilities for periods up to many decades is a proven safe technology and is applied globally. Nonetheless, this storage method is not a long-term or final solution for wastes that remain radioactive for very long times and for which the necessary continued active monitoring and inspection, security and maintenance cannot be assured. For materials that remain hazardous for thousands to hundreds of thousands of years, the acknowledged approach to long-term isolation and confinement is disposal in a stable geological environment, deep enough beneath the surface of Earth to exclude disruptions due to near-surface processes and events. This is referred to as emplacement in a Geological Disposal Facility (GDF). Geological processes in the deep underground occur at slow and predictable rates over very long periods of time. At the current state of science and technology, geological disposal is the only solution that can ensure no radioactivity will ever return to the human environment at concentrations that can be harmful. For this above reasons, the Netherlands, along with other countries with significant quantities of long-lived radioactive wastes, has chosen geological disposal as the official national policy. The decision-in-principle to dispose of Dutch radioactive waste in a GDF was taken by the government in 1984. The Dutch policy is for more than thirty years based the above ground storage of the radioactive waste for a period of at least 100 years, after which disposal deep below ground is foreseen around 2130 and dismantling of COVRA afterwards.

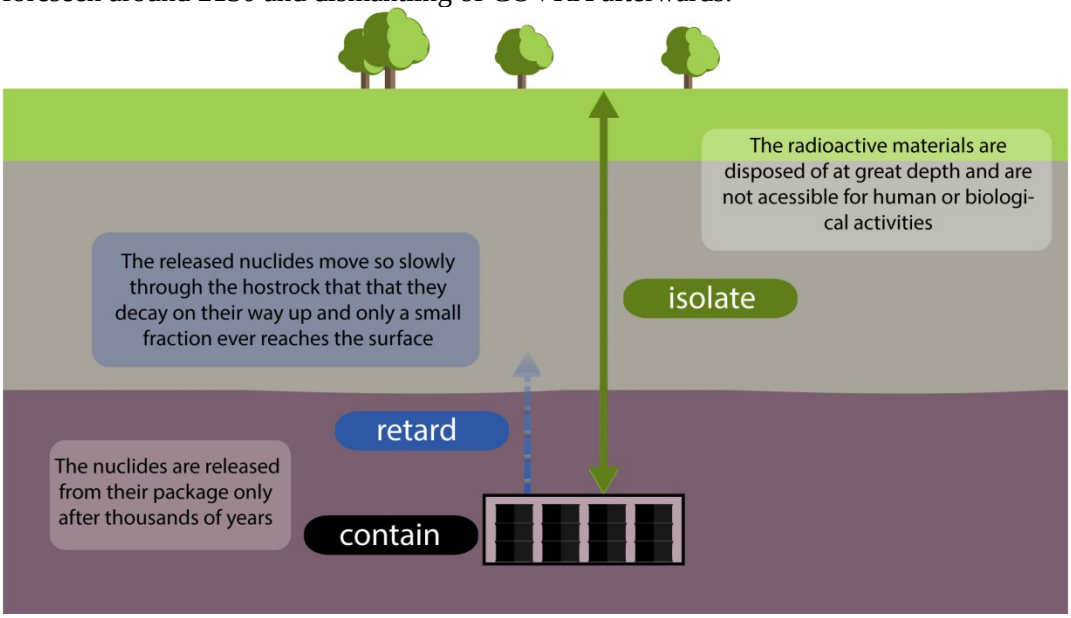

Figure 7: isolate, retard and contain

8. CONCLUSION

The Netherlands followed a straightforward and clear line to implement the governmental policy to manage the small volume of radioactive waste. COVRA executes this policy: facilities are in operation to store low-, medium- and high-level waste for a period of at least 100 years and disposal in 2130. COVRA takes over full title of the waste and prepares financially, technically and socially the steps to be taken after this period of storage. This is a dedicated management solution for the Netherlands that works!

International Network for Nuclear Waste Characterization IAEA-LABONET: Facts and Future

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ABSTRACT

LABONET was launched early 2011 by the International Atomic Energy Agency (IAEA) as an international network on nuclear waste characterization. Its objective is to improve the quality of, and harmonize, activities related to the radioactive waste characterization in IAEA Member States. It is widely accepted that proper characterization of radioactive waste through its entire flow-sheet is a key activity and will improve the sustainability of waste (e.g. immobilized waste packages). In particular the needs and problems in radioactive waste characterization in the participating countries are a main concern within LABONET, as the programs are not elaborated to the same level in the various countries. LABONET wants to facilitate the exchange of international experience in the application of proven, quality assured practices for the characterization of waste and waste packages. This will be beneficial for the less developed countries. The objectives of LABONET are achieved via different actions: a) exchanging information and expertise via presentations and discussions at technical meetings; b) expert meetings on particular topics of interest, for instance e-learning and e-platforms; c) training sessions in less developed countries, for instance, on non-destructive assay of waste packages; d) facilitate the exchange of information amongst its members ; and e) assist Member States in establishing characterization plans that are cost effective and of good quality. LABONET is continuously encouraging institutions or professionals who need some assistance or can help in solving questions about characterization to be a member. Currently, LABONET is preparing for its future, including projects addressing common problems of its members. Aiming at the dissemination of this important network for the professionals involved in radioactive waste characterization, this paper will discuss the following issues: (1) the objectives of LABONET, (2) its structure, membership and organization of meetings, (3) some achievements, and (4) plans for the future.

KEYWORDS: IAEA-LABONET, radioactive waste, characterization, exchanging information, training

INTRODUCTION

The safe management and a sustainable disposal of low and intermediate level radioactive waste requires accurate and quality assured characterization by non-destructive and destructive methods or other means, to determine the radionuclide inventory, chemical, physical properties in the different stages of waste management. Relevant procedures, standards and practices have been developed and continue to be refined in waste characterization facilities in Member States [1, 2, 3].

[1] INTERNATIONAL ATOMIC ENERGY AGENCY. Characterization of Radioactive Waste Forms and Packages. Technical Report Series No 383, Vienna. (1997).
[2] INTERNATIONAL ATOMIC ENERGY AGENCY. Strategy and Methodology for Radioactive Waste Characterization, TECDOC-1537, Vienna. (2007).
[3] INTERNATIONAL ATOMIC ENERGY AGENCY . Determination and Use of Scaling Factors for Waste Characterization in Nuclear Power Plants. Nuclear Energy Series, Technical Reports No. NW-T-1.18. Vienna (2009).

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Sharing of information between organizations and practitioners underpin the on-going development of such procedures, standards and practices [5, 5] .

A number of Member States with less developed programs however may not have such laboratories or organizations. Consequently, for these countries, achieving satisfactory characterization programs is a complex technical challenge requiring both scientific and financial resources. The IAEA wishes to support organizations, either currently engaged in or seeking to develop, such characterization programs, through their inclusion in a network to cooperate in and coordinate relevant actions, training and technical progress. LABONET is the network created for this.

LABONET, the International Network of laboratories for Nuclear Waste Characterization was launched in 2011 and focuses on proven practices and their successful implementation. It builds on world class research and characterization activities both nationally and internationally by sharing information between Member States. Since then LABONET ran a number of activities including training sessions and technical meetings. In 2017 the Terms of Reference of the network were revised to gain closer alignment with the needs of Member States and to reflect the priorities of the IAEA - laid down in a 5-year working plan.

In particular LABONET is aiming to:

- Facilitate the exchange of knowledge and experience among organizations with characterization facilities.
- Support organizations or Member States with less advanced capabilities for characterization of radioactive waste, by facilitating access to the relevant skills, knowledge, management practices and approaches and expertise from Member States with mature characterization capabilities.
- Develop and implement training and demonstration activities with a global, regional or thematic focus, provide hands-on, user-oriented training and demonstration of proven procedures and technology.
- Plan and implement projects through dedicated working groups to address recognized needs of Member States in waste characterization.
- Propose Coordinated Research Projects (CRP) to the IAEA for relevant technical needs of Member States.
- Contribute waste characterization expertise to the IAEA and be a forum in which experts' advice and technical guidance may be provided to IAEA's relevant programs.

LABONET produces outcomes directly relevant to the needs of the Member States and in alignment with the objectives of the IAEA. The Steering Committee proposes to the IAEA at a regular basis topics to take into account as these topics are addressed by representatives of Member States at annual plenary and technical meetings of LABONET. As example the following projects proposals have been defined in the new 5-year plan, covering the period 2017-2021.

- Assist Member States to find and apply safe, cost-effective solutions to characterize radioactive waste
- Facilitate exchange of information and expertise between Member States on waste characterization
- Provide training and education in waste characterization activities

[5] VANISEGHEM, P., BRUNEL, G., LIERSE, CH., MORALES, A., ODOJ, R., TROIANI, F., HUGON, M., The European Network for Quality Checking of Waste Packages: objectives and status. Management and Disposal of Radioactive Waste, Editor T. McMenamin, EUR 17543, 286-295 (1997).

[5] TIETZE-JAENSCH, H., VANISEGHEM, P., DODARO, A., ANTHONI, S., NECKEL, W., PINA, G., VAN VELZEN, L., GUISSET, J.P., KEKKI, T., STEYER, S., DIONISI, M., VICO DEL CERRO, E., LIERSE, CH., FUKS, L., ENTRAP and its potential interaction with the IGD-TP". Mineralogical Magazine, Vol 79(6), 1515-1520 (2015).

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- Assist Member States to improve the effectiveness and efficiency of waste characterization activities.

In the 5-Year plan of 2017-2021 it is also stated how to produce valuable outcomes of these projects. LABONET proposes to organize thematic technical meetings. These technical meetings have to be held on a regular basis and will provide a platform for LABONET members and other participants of these projects to interact with each other, sharing information and experience on the mentioned topics. The foreseen outcome will be published as a "Wiki"-document at the IAEA-LABONET website.

LABONET structure

Membership of LABONET is open to individuals as well as organizations. The network welcomes the participation of commercial entities where their activities within the network are not commercial in nature.

LABONET comprises IAEA and its Scientific Secretary, Members, Working Groups, Steering Committee, LABONET Chairperson and Vice-chairperson, and Sponsors. Individuals and organizations may apply to become a member by contacting the Scientific Secretary. Once appointed, members will remain in LABONET until resignation.

Members of LABONET might be individuals involved in characterization of low and intermediate level waste (scientific, technical or management responsibilities), but also organizations with waste characterization capabilities. These capabilities might be suitable for development, demonstration and/or training and a willingness to share their resources with other Member States or organizations engaged in planning, implementing, improving or operating capabilities for characterization of low and intermediate level waste, and who are willing to participate and actively support in the activities of LABONET.

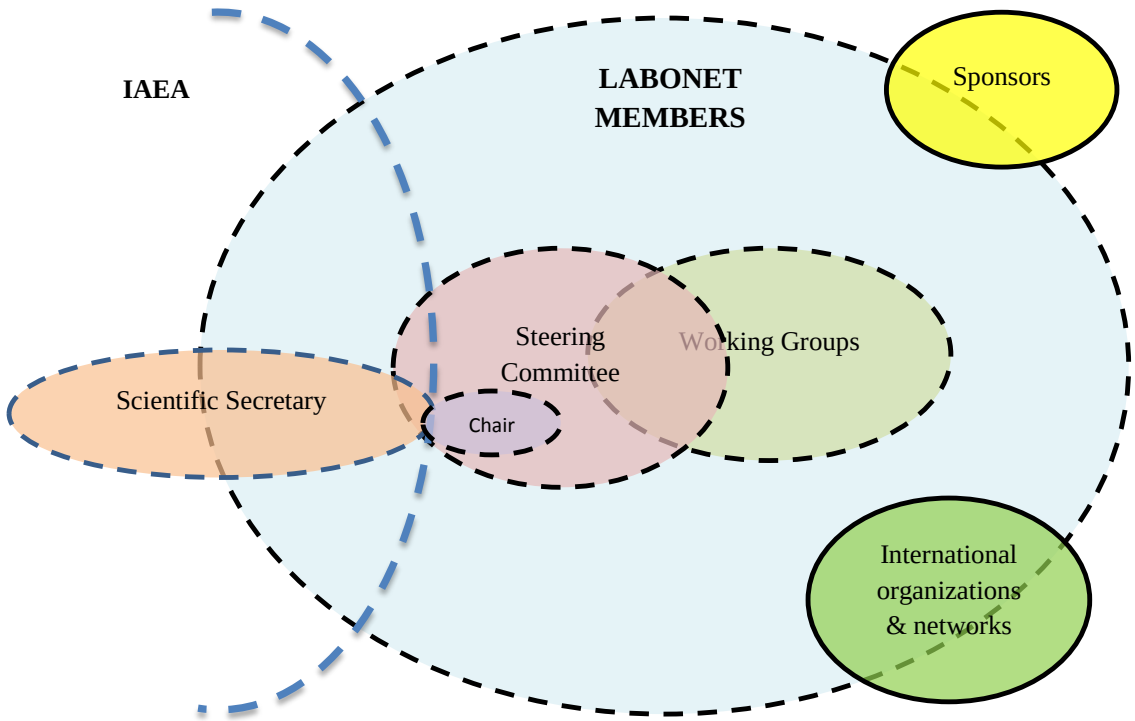

Fig. 1 Structure of LABONET and the interrelations of participants

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Working Groups are non-permanent groups that conduct the target projects and comprise members specifically appointed by the Steering Committee. Among their responsibilities, they deliver measurable and targeted outputs that are relevant to Member States and supported by the IAEA, organize technical visits and meetings to progress their projects as required and may interface with other working groups, other networks or external organizations.

The Steering Committee is composed of individual members and the Scientific Secretary of LABONET provided by the IAEA. This Committee reviews LABONET activities and provides advice to the IAEA. The collective responsibilities of the Steering Committee are to liaise with Member States to understand the needs of Member States in waste characterization and influence and persuade LABONET members to ensure their active participation in the network activities. The Chair and Vice-chairperson of the LABONET are members of the Steering Committee appointed by the Steering Committee for a 3-year term. The Chair of the LABONET should mainly support the Scientific Secretary to coordinate LABONET activities and organize and chair all network and steering committee meetings. The Vice-Chair should support the chairperson as required.

Sponsors are organizations that will work with the IAEA to provide either financial or technical support to LABONET activities. These may include commercial entities.

Figure 1 shows the structure of LABONET and the interrelations of participants.

LABONET achievements

Since LABONET was launched in 2011, some 45 Member States joined LABONET. Six technical meetings have been held in Slovakia, Italy, Austria, France, Belgium, the Netherlands and Czech Republic, as well as training sessions and dedicated expert meetings. Up to 25 participants joined each of these technical meetings.

The presentations and meeting reports were made available on a dedicated website [6] () accessible by its members. Topics covered in the presentations during the technical meetings can be grouped into 5 categories: (1) overviews of the national programs on radioactive waste management; (2) determination of the radionuclide inventory through analysis (e.g. non-destructive, destructive), modelling calculations, scaling factors; (3) other characterization techniques; (4) immobilized waste; (5) other topics like the segregation of ILW in high-ILW, LLW and free release (if possible). LABONET is primarily focusing on very low, low and intermediate level radioactive waste.

Plans for the future

A 5-year plan has been discussed and set up at the end of August. It will guide the activities of LABONET and ensure specific, measurable and realistic outcomes, directly relevant to the needs of the Member States, aligned with the objectives of the IAEA and make sure that this is achieved in a timely manner.

These activities will be delivered as ‘projects’. Table 1 gives an impression of the proposed activities in this 5-Year plan to the IAEA. The projects will be scheduled in the next 5 year period subject to resources and the availability of LABONET members to join the respective Working Groups.

The project “Guideline for Sampling Radioactive Waste” aims to elaborate guidance for LABONET members to establish sampling procedures for their waste characterization program. This guidance will provide a tool with recommendations for the elaboration of country-based sampling procedures adapted to the different realities of Member States.

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[6] https://nucleus.iaea.org/sites/connect/LABONETpublic (2017)

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Table 1: Project proposals for next 5 years

Project / Activity	**Primary Output**	**Expected duration**
Develop a Guideline for Sampling Radioactive Waste	LABONET Wiki article on guidelines for sampling radioactive waste	3 years
Develop a Guideline for Analysis of toxic, non-radioactive constituents in radioactive waste	LABONET Wiki article on non-radioactive toxic substances in radioactive waste and their management options.	4 years
Develop a catalogue of methodology for demonstrating compliance with WAC (Waste Acceptance Criteria)	LABONET Wiki article with a catalogue of methods for demonstrating compliance with WAC	5 years
Develop a Guideline for Data quality management of the declared characteristics of waste	A LABONET Wiki article on data quality management of the declared characteristics of waste	2 years
Inter-comparisons; Destructive Analysis (DA) proficiency tests on samples and a Non-Destructive Analysis (NDA) proficiency tests on full size re-usable 220 litre drums including certified re-usable calibration sources	A report on NDA and DA inter-comparison on full size 220 litre drums	5 years
Developing e-Learning modules for Scaling Factor method	Waste Characterisation e-Learning Modules published on CONNECT.	3 years

The project “Guideline for Analysis of toxic, non-radioactive constituents in radioactive waste” aims to publish a report, consisting of (1) an overview of the toxic, non-radioactive constituents in radioactive waste of relevance for the Member States; (2) possible techniques to determine these inventories; and (3) recommendations for further research if needed, and subsequent recommendations for cooperation between members of LABONET.

The project “Catalogue of methodologies for demonstrating compliance with Waste Acceptance Criteria” aims to create a catalogue with recommended methodologies for a number of selected waste fluxes (immobilized or not) of interest for Member States, and list the according waste acceptance criteria; in addition, collect the known information on the radionuclide inventory; select a few waste fluxes, and perform estimations using different combinations e.g. (1) no DA, only NDA and Scaling Factor (SF), (2) NDA, DA, SF, (3) NDA, modeling calculations, SF.

The project “Guideline for Data quality management of the declared characteristics of waste” aims to develop a guideline that should be applicable for waste producers and waste management organizations of Member States to collect the data of the origin as well of the non- as of the radioactively contaminated materials present in wastes, so that the radioactivity present in this waste can be obtained by routine ND characterization with the best accuracy.

The project "Inter-comparisons; DA proficiency tests on samples and a NDA proficiency tests on full size 220 litre drums" aims to design and perform, due to the repeated requests of participants of Members States at LABONET meetings, a full-size NDA inter-comparison on 220 liter drums. It is worth to mention that the designed full-size 220 litre drums will be re-usable (e.g. simulating homogeneous and non-homogeneous raw waste and compacted waste) including the incorporated calibration sources.

The project "Developing e-Learning modules for Scaling Factor method" aims to develop an "e-Learning module" for studying and developing lesson materials/procedures to apply the Scaling Factor method correctly.

Conclusions

The IAEA created LABONET in 2011 to facilitate the international exchange of expertise on radioactive waste characterization. Another major objective was to assist countries with less advanced programs on radioactive waste characterization in order to improve their capabilities. Presently more than 45 Member States have already been participating in LABONET events. LABONET is very successful through its annual technical meetings and training sessions. Currently, the SC of LABONET and the IAEA prepare to strengthen the operation of LABONET, by adding to the present approach a number of technical projects that will meet the needs of Member States and to strengthen the interaction with other IAEA networks, like IDN and ENVIRONET, respectively IAEA's International Decommissioning Network and IAEA's Environmental Remediation Network.

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