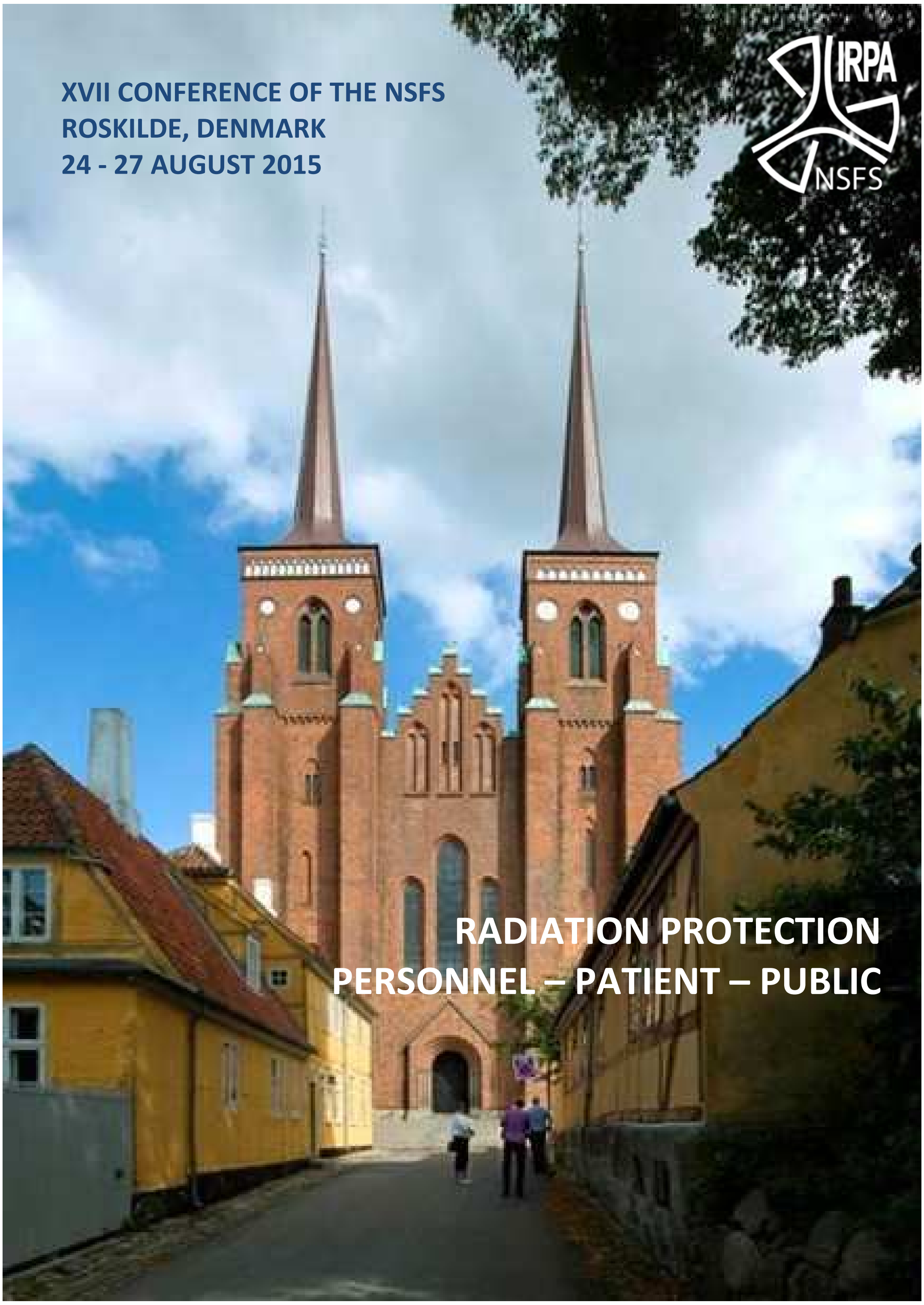

NORDIC SOCIETY FOR RADIATION PROTECTION

Proceedings of the NSFS XVII Conference

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NSFS 2015

PROGRAM AND ORGANIZING COMMITTEE:
Mette Øhlenschlæger
Hanne N. Waltenburg
Jens Søgaard-Hansen
Bent Lauritzen
Carsten Israelson
Søren Holm

Dear colleagues

It was a great pleasure to welcome more than 120 participants from around the world to the XVII conference of the Nordic Society for Radiation Protection – held in Roskilde 24-27 August 2015.

The theme for the conference was “Radiation Protection – Personnel, Patient and Public” which reflected the very broad spectrum of our professional field. You will find that this theme was very well represented in the conference program. We received many high quality papers, making it easy for the organizing committee to put together a very interesting program with 40 oral presentations and 20 posters.

The 2015 receiver of the Bo Lindell award was Sigurður M. Magnússon, who gave a presentation on ’Nordic co-operation in an international context’.

In addition, three invited speakers introduced very hot radiation protection issues: the latest international developments and cooperation in the field of emergency preparedness and response, the European Spallation Source in Lund in Sweden, and developments and justification of applications using ionizing radiation in the medical field.

The organizing committee would like to thank all oral and poster presenters, and other participants for taking active part in the discussions.

We hope that you in addition to all the professional activities found to time to relax and mingle among colleagues and to enjoy Roskilde, a famous Danish town founded more than 1000 years ago at the bay of Roskilde Fjord.

Mette Øhlenschlæger

President of NSFS 2011-2015

The opinions and conclusions presented in the articles of the proceedings are those of the authors and do not necessarily represent the official position of the Nordic Society for Radiation Protection.

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S1-O1

Bo Lindell Lecture:
Nordic co-operation in an international context

Sigurður M. Magnússon

Icelandic Radiation Safety Authority

The Nordic countries have a long history of co-operation in nuclear and radiation issues. What began with sporadic contacts in the time after World War 2 evolved over the next 10 – 15 years into an extensive and formalized co-operation that is still important in spite of radical societal and technological changes taking place since then.

The Nordic co-operation has adapted well, to both external events and changing political climate in the Nordic countries. It has stood the test of time and severe nuclear accidents, is robust and fit for purpose. Through the wide ranging co-operation leading to important Nordic documents, such as the Flag books, a common Nordic view on nuclear and radiation safety issues has been established.

The Nordic nuclear and radiation co-operation is dual in its nature. On one hand there is the traditional co-operation between the authorities and on the other hand there is the research co-operation within the framework of NKS.

The lecture will give an overview of the Nordic co-operation in an international context, where we are and what the future may hold.

S1-O2

The Bo Lindell book translation project

Jack Valentin

Former ICRP secretary

Almost all NSFS members and most other IRPA members have heard of the admirable Bo Lindell (former ICRP Chair, former UNSCEAR Chair, former head of SSI, the Radiation Protection Authority in Sweden, etc etc). Many IRPA members around the world have met him over the years, and many are aware of his significant book series, 'Strålningens, radioaktivitetens och strålskyddets historia' (The history of radiation, radioactivity, and radiological protection), comprising 2637 pages of accurate, important, and exciting information with a Nordic perspective on developments world-wide. Unfortunately, fewer have been able to read the complete set of four books, since they are written in Swedish. Bo Lindell has met most of the significant contributors to the development of radiation science and radiological protection in the 20th century, he contributed personally to many of the crucial events, and he is well known world-wide both for his razor-sharp intellect and encyclopaedic knowledge and for his ever-lasting politeness, kindness, and patience. Because of this, and given that there are not that many other documents summarising the history of radiation and protection, there is a considerable demand for a translation into English, both within the Nordic countries and internationally. However, in spite of that demand, commercial translation and publishing of the books was not a viable proposition. Therefore, NSFS has launched a Translation Project, organising professional translation through a proper agency, Quality Assurance through an editorial committee of senior Nordic and Anglo-Saxon experts, and engaging additional sponsors to co-fund the costs. In addition to NSFS, the co-sponsors are NKS, IRPA, and the five Nordic licensing authorities. The project is now well under way, with the first volume (Pandora's Box) completed. The second volume (The Sword of Damocles) is in the final stages of preparation, the third volume (The Labours of Hercules) is under Quality Assurance editing, and the translator is half-way through the raw translation of the final volume (The Pains of Sisyphus).The resulting English versions will be published in two ways: by making PDF files available for free downloading from the NKS and/or NSFS web sites, and by providing print-on-demand hard copies available at cost price.

S2-O1

New developments and growing international cooperation in the field of emergency preparedness and response

Patrick Majerus

Ministry of Health, Department of Radiation Protection, Luxembourg

Emergency preparedness and response (EP&R) has mostly been developed and implemented at national level without giving great importance to cross-border issues. In case of the nuclear accident in Fukushima Daiichi, this resulted in different advises given to the own citizens by European and other states. In the event of a nuclear emergency in Europe, immediate protective actions may have to be taken in more than one country. Exercises, such as the one of the Greater Region in 2012, involving France, Germany, Belgium and Luxembourg, clearly demonstrate the difficulties to coordinate the response between several countries.

Internationally, many efforts have been taken in recent years to improve the situation. HERCA has worked on recommendations for a better coordination during the response on accidents taking place in Europe or elsewhere. Together with WENRA, recommendations concerning fast kinetic severe accidents were also put forward. The European Commission has significantly improved its ECURIE information exchange platform and performed a review on EP&R in Europe. The IAEA, based on its action plan on nuclear safety, gave additional competences to its Incident and Emergency Centre (IEC), particularly with regard to assessment and prognosis. Other international initiatives, such as NERIS, MELODI and PREPARE, develop enhanced scientific and technical tools to support EP&R.

In the legal area, Directive 2013/71/EURATOM contains several new approaches. Member States may implement these provisions based on purely national considerations, with the risk that differences between the States will again increase. The following years will therefore be decisive in terms of promoting a coherent implementation of the various provisions and recommendations at all levels. In addition, a testing and verification mechanism will be needed to ensure the effectiveness of the new EP&R arrangements.

S3-O1

ESS status - focusing on the perspectives for international research, and the challenges related to radiation protection for the staff, the public and the environment

Peter Jacobsson

European Spallation Source ESS AB, Environment, Safety & Health (ESH) Division

The European Spallation Source (ESS) is a multi-disciplinary research center based on the world's most powerful neutron source. This new facility will be around 30 times brighter than today's leading neutron-scattering facilities, enabling new opportunities for researchers in the fields of life sciences, energy, environmental technology, cultural heritage and fundamental physics. ESS will be designed to the highest level of safety in order to meet both the expectations of the users, the personnel and the regulatory requirements. This applies to all aspects of safety but in particular radiation safety and radiation protection.

The presentation will describe the fundamental safety functions for the neutron source, the confinement barriers and the aspects of radiation shielding. It will discuss different aspects of possible radiation doses to the staff, users and the third person during normal operation and accidents. The presentation will also point out challenging areas, e.g. material behavior of the tungsten (the ESS target material) during spallation, where research is presently being done.

Current and emerging challenges for Nordic nuclear safety: cooperation through the NKS-R programme

Karin Andgren[1], Kasper G. Andersson[2] Finn Physant[3], Sigurdur M. Magnusson[3]

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[2] NKS, DTU
[3]NKS

Building on the foundation of a common cultural and historical heritage and a long tradition of collaboration, NKS aims to facilitate a common Nordic view on nuclear and radiation safety. A common understanding of rules, practice and measures, and national differences in this context, is an essential requirement. Through collaborative efforts problems may be tackled quicker, more efficiently, more consistently, and at a lower cost. NKS activities are divided into two programme areas, NKS-B and NKS-R. Recent results and on-going activities of the NKS-R (reactor safety) programme will be presented at the seminar. On-going activities include projects within the areas of severe accidents, risk analysis and organisational issues. For example, the research on core melt debris coolability and steam explosion represents the front edge research worldwide for Nordic boiling water reactors. Other activities financed by NKS in 2015 include guidelines on level 3 probabilistic safety analysis (consequences off-site) and a project aiming to enhance organisational learning from successful actions and decisions. A seminar is one of the best ways of disseminating information about the work NKS does and the results it achieves. We therefore invite all interested persons and organisations to participate in a seminar 12-13th of January 2016 in Stockholm. Focus will be on lessons learned from Fukushima and the way to proceed in both Nordic emergency management and nuclear risk assessment, which are strongly interrelated disciplines.

S4-O1

Current and emerging challenges for Nordic nuclear/radiological emergency preparedness: cooperation through the NKS-B programme

Kasper Andersson[1], Karin Andgren[2], Finn Physant[3], Sigurdur Magnusson[4]

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The Nordic countries share a sociocultural heritage, which has historically facilitated cooperation on a wide range of societal themes. Specifically on nuclear and radiological preparedness topics, cooperation over many decades in what has evolved into the core of the NKS-B programme has provided a common understanding of important issues while maintaining close links between organisations with an interest in the field. Following up on a number of preparedness related learning points after the Fukushima accident in 2011 a suite of new dedicated NKS-B activities were launched already in January 2012. The scope and requirements in cooperative Nordic preparedness were illustrated and discussed at the NKS seminar in Stockholm in January 2013 on the Fukushima Accident and Perspectives for Nordic Reactor Safety and Emergency Preparedness, which had 140 participants. For instance the needs for well exercised monitoring strategies and the important role of detailed and operational information pathways and strategies were highlighted. The results of tens of activities have now been published on the NKS website, which are directly aimed at extending and upgrading Nordic capabilities to respond to an emergency in the light of the Fukushima experience. On this background NKS invites all interested persons and organisations to participate in a follow-up workshop on 12-13th of January 2016 on lessons learned and the way to proceed in both Nordic emergency management and nuclear risk assessment, which are strongly interrelated disciplines. Over the years, the NKS-B programme has also produced numerous valuable results in areas as diverse as improvement of routine measurement technologies for the nuclear industry and waste management relating to decommissioning and NORM generating processes.

S4-O2

Societal dimensions in post-accident recovery – return of experience from Fukushima and Chernobyl experience

Inger Margrethe Eikelmann

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Emergency and post-accident situations are of a complex nature insofar as they involve a large number of actors and a large number of intrinsically interlinked dimensions and issues (heath, environmental, economic, social, cultural, ethical dimensions). People and organisations have to face this complexity in a context in which the usual social and institutional routines are particularly challenged and destabilised. Existing return of experience from post-Chernobyl situation in Norway and from post-Fukushima situation in Japan show that the societal dimension plays is of key importance in the recovery process following a nuclear accident. In effect, the capacity of local actors to build their response to the crisis, at a personal and community levels also depends on the capacity of local actors to build new forms of cooperation among themselves and with other actors (e.g. experts, actors providing resource). The societal dimension in post-accident recovery and the understanding of the interactions between the different recovery paths includes a reflection on how some central values (truth, justice, solidarity, dignity, democratic culture), are taken into account in the different paths. In effect this has an influence on the resilience capacities of the actors). The objectives for the study is an analysis of the societal dimension of local post-accident recovery processes, on the basis of the post-Chernobyl experience in Norway and on the post-Fukushima experience in Japan. The work is carried out by case studies in the Norwegian and Japanese context. The case studies cover the following fields: Food and farming, healthcare and local community management. The analysis of the cases identifies the different recovery paths (individuals and families, local authorities, regional and national authorities, experts and TSOs and their interactions. It also shows the role of values in the interaction between the different recovery paths and in the sustainability of recovery.

S4-O3

Uncertainties of Atmospheric Dispersion Calculations for Emergency Preparedness

J. H. Sørensen[1], B. Amstrup[1], J. Bartnicki[2], H. Feddersen[1], S. C. Hoe[3], C. Israelson[3], H. Klein[2], U. S. Korsholm[1], B. Lauritzen[4], J. Lindgren[5]

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Abstract

Atmospheric dispersion model calculations of radionuclide releases from a nuclear accident provide information on possible contamination levels and radiation hazards, thereby facilitating decisions on protective actions. This is implemented in emergency management through Decision Support Systems. Recent developments in numerical weather prediction models include probabilistic forecasting techniques to address the inherent uncertainties in numerical forecasting. This approach may readily be taken over by atmospheric dispersion modelling, and hence the meteorological uncertainties of atmospheric dispersion predictions can be estimated quantitatively.

Introduction

Most current Decision Support Systems (DSSs) in operational use for nuclear emergency preparedness and management do not accommodate uncertainties, but merely allow for presentation of the 'most likely' plume dispersion or deposition pattern. However, recent research and development has enabled incorporation of such new facilities.

In the recently completed NKS-B project ‘Meteorological Uncertainty of atmospheric Dispersion model results’ (MUD), the uncertainties of atmospheric dispersion model calculations are investigated, as well as means for incorporating these into DSSs, allowing for the presentation of uncertainties to decision makers in a comprehensible manner (Sørensen *et al.*, 2013 and 2014). The MUD methodology has been implemented operationally in the Danish setup providing long-range atmospheric dispersion model results for ARGOS by using the DERMA model (Sørensen *et al.*, 2007; Sørensen, 1998).

In the ongoing NKS-B project ‘Fukushima Accident: UNcertainty of Atmospheric dispersion modelling’ (FAUNA), the MUD methodology is applied to the Fukushima Daiichi NPP accident, where the influence of meteorological uncertainties on real-time assessments of atmospheric dispersion and deposition is being investigated, imitating real-time emergency management (Sørensen *et al.*, 2015). The objective of the project is to examine how uncertainty estimates can be presented to experts as well as decisions makers in a manner that meets both the requirements of the experts running the DSS and the decision makers relying on practical decision support.

In this paper, the methodology developed in MUD will be described, and results of MUD and FAUNA presented.

The DMI meteorological Ensemble Prediction System

The DMI meteorological Ensemble Prediction System (EPS), which is currently based on the HIRLAM numerical weather prediction (NWP) model (Sass *et al.*, 2002), involves 25 ensemble members. The horizontal resolution is 0.05°, corresponding to approximately 5.5 km, and vertically the model has 40 layers from the surface and up to 10 hPa (approximately 30 km above the sea surface). The ensemble HIRLAM model is nested into ECMWF's global model. The geographic domain is displayed in Fig. 1.

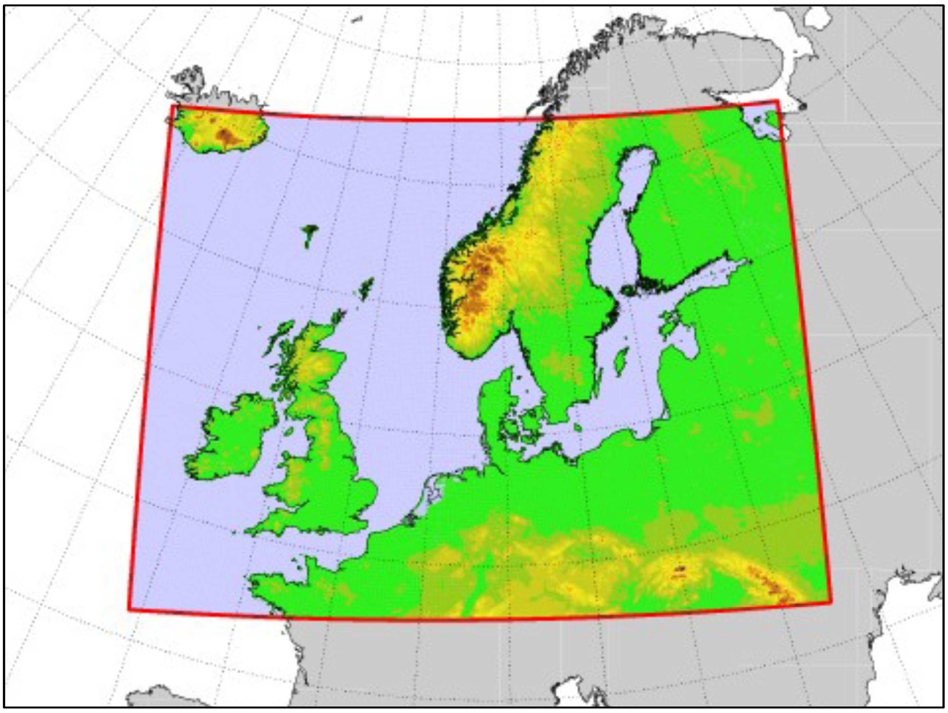

Figure 1. Geographic domain covered by the DMI operational meteorological EPS.

Meteorological forecast uncertainties arise from uncertainties in the initial and lateral boundary conditions and from model short-comings, particularly short-comings associated with parameterization of physical processes that take place on spatial scales that cannot be represented explicitly in the model.

For short-range forecasts, i.e. up to two days in advance, the main uncertainties are those associated with clouds and convection, and so the main application of the DMI EPS has been to provide forecasters at DMI with a tool to predict the risk of severe precipitation events (rain or snow) 12 to 36 hours in advance. Occasionally, there is also substantial uncertainty associated with the wind field, typically in relation with passing weather fronts.

In the MUD project, the DMI meteorological EPS was run for four typical meteorological situations for 54 hour forecast periods. These situations involve windy cyclonic and low-wind anti-cyclonic conditions, as well as convective summer precipitation. For each of four nuclear power plants (NPPs) in vicinity of the Nordic countries, atmospheric dispersion model calculations were carried out for the meteorological ensembles using the same six-hour release scenario.

In the FAUNA project, the MUD methodology is applied to a realistic setting of the Fukushima accident. Thereby, the implications of the uncertainty estimates for the emergency management are investigated. A meteorological ensemble forecasting system has been set up and run for the period of concern and for a geographical domain covering Japan and surroundings, cf. Fig. 2. For the full period, two-day meteorological forecasts have been generated four times a day, as would be the case for an operational system in real time.

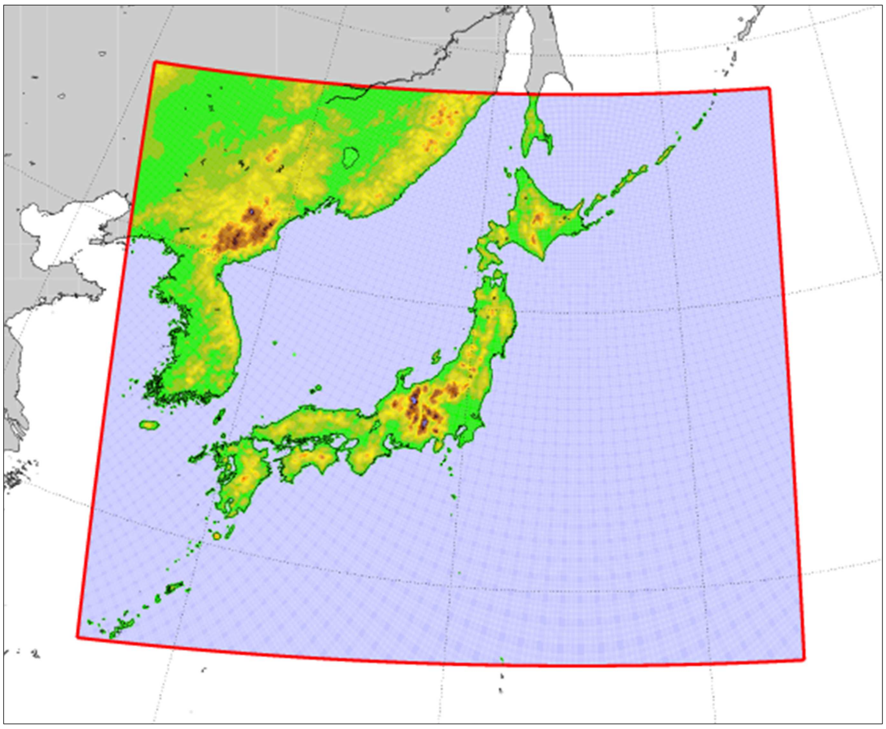

Figure 2. Geographic domain covered by the meteorological EPS for Japan.

For selected dates and times in the release period, the Danish and the Norwegian long-range atmospheric dispersion models, DERMA (Sørensen *et al.*, 2007; Sørensen, 1998) and EEMEP (Bartnicki *et al.*, 2011), respectively, are run based on data of the meteorological ensemble assuming that a realistic source term is available in near real time. Here, the source description by Katata *et al.* (2014) is employed. Corresponding ensemble-statistical parameters are be calculated, e.g. percentiles of the concentration and deposition fields. Further, the predictions are available to the ARGOS decision-support system for display and dose modelling. Thereby, the project imitates real-time emergency management taking into account estimates of the uncertainty of the dispersion model results.

Atmospheric dispersion model runs using EPS NWP model data

In Fig. 3 results are shown for DERMA applied to a scenario with a release from the Ringhals NPP beginning on 2011-05-20 at 18 UTC. The results shown are time-integrated concentration of I-131 at 54 hours after the start of the release. In the upper row, a low percentile, the median and a large percentile are displayed. In the lower row, probabilities for exceeding values of 10^4, 10^3 and 10^2 Bq h/m^3, respectively, are presented. As can be seen, the variability is considerable.

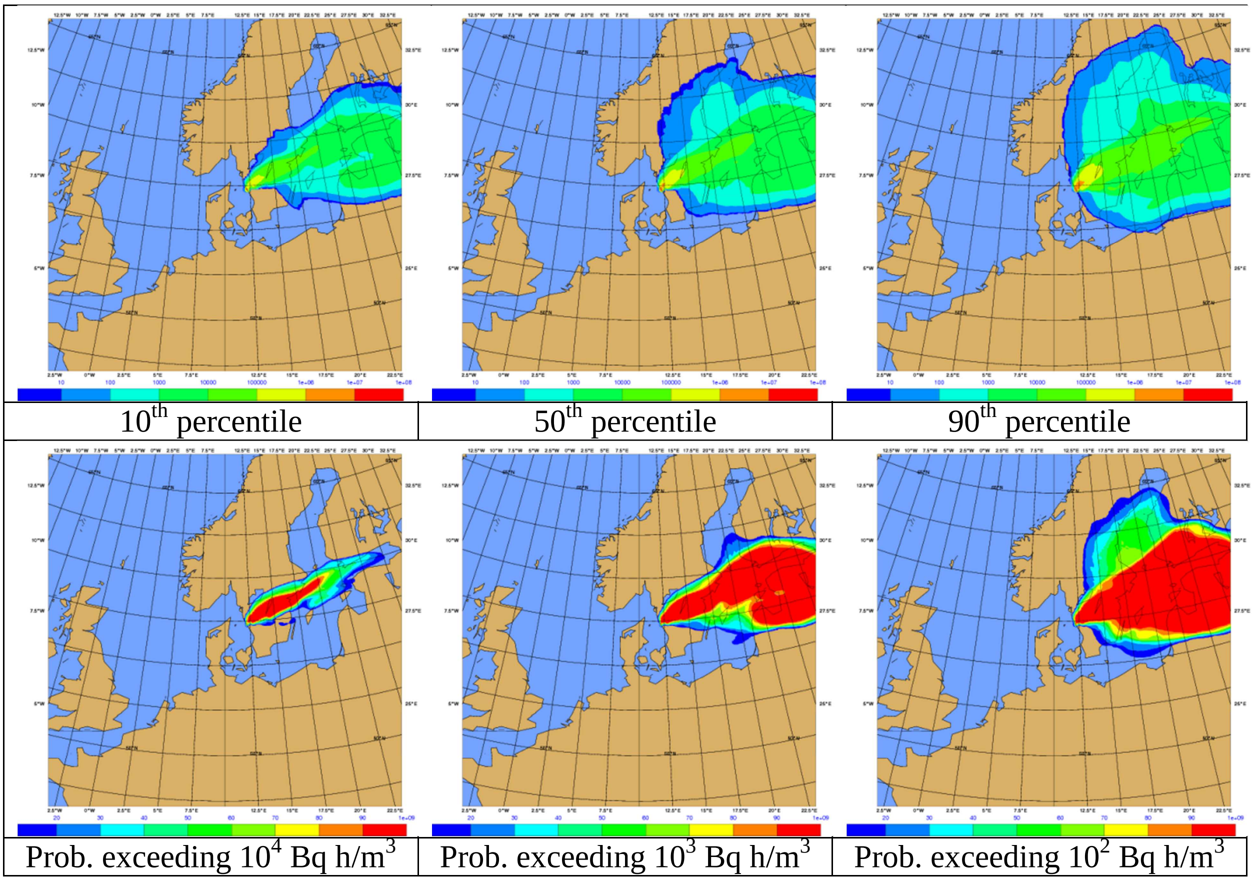

Figure 3 Scenario: 2011-05-23. NPP: Ringhals. Field: Time-integrated concentration 54 hours after start of release. Nuclide: I-131.

In Fig. 4, percentiles of the thyroid dose are presented using two isocurves. The large percentile indicates the maximum area which can possibly be influenced by the plume. The real dose pattern will most likely be confined inside this domain. The median indicates the domain which will most likely become influenced. And the low percentile indicates the domain which will be influenced with large certainty. These quantities may are considered useful for the emergency management.

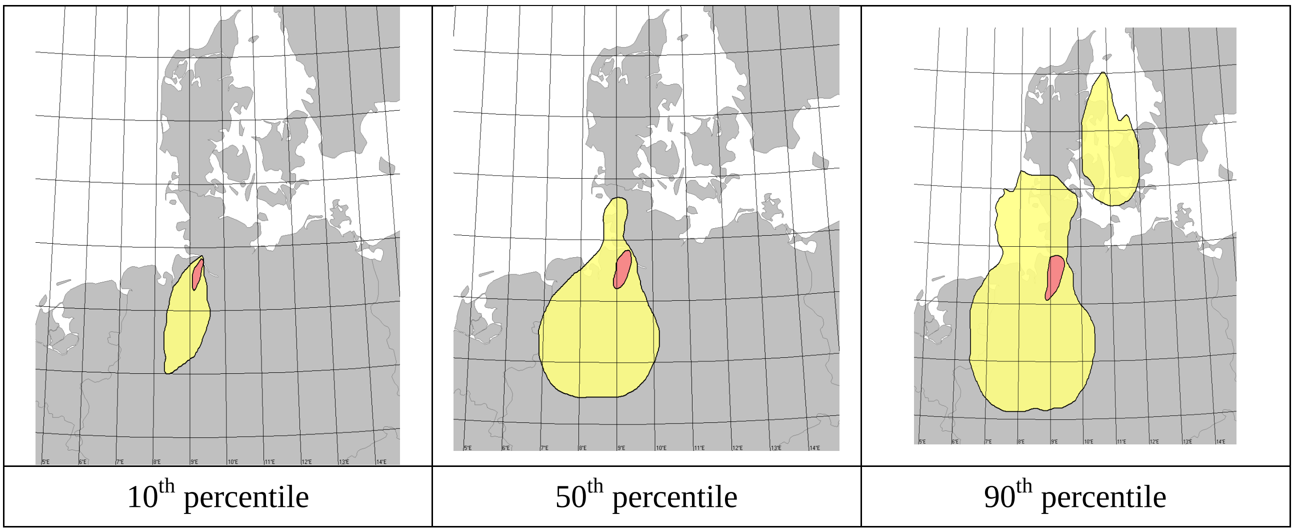

Figure 4 Scenario: 2011-05-23. Field: Thyroid dose 54 hours after start of release. Isocurves at 1 and 100 mGy.

Conclusions

In brief, the conclusions are the following:

- Depending on the meteorological situation, the uncertainties can be large, up to a factor of ten, especially when convective precipitation is involved.
- There is a risk of information overflow when considering presentation to decision makers, and therefore statistical parameters must be selected carefully:
  - The use of percentiles is encouraged, involving a large percentage, the median and a low percentage.
  - Probabilities for exceeding given threshold values are also relevant. However, they rely on radionuclide dependent threshold values, which are not available in all cases.
- The methodology developed in course of MUD is implemented operationally at DMI to be used in ARGOS by the Danish Emergency Management Agency (DEMA).

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S4-O4

Uncertainty in predictions of the ambient dose equivalent rates for 30 years following the Fukushima Daiichi nuclear power plant accident

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Abstract

It is essential to develop long-term prediction models allowing follow-up of the radiological situation and the sustainability of the rehabilitation programme after the Fukushima Daiichi nuclear power plant accident. At the Japan Atomic Energy Agency, long-term prediction models have been developed to assess how ambient dose equivalent rates might change in the future and to analyze radiological situations within the 80 km radius of the Fukushima Daiichi nuclear power plant. The long-term prediction models described using a double exponential form with ecological half-lives for land use, enable affected population to receive information on the level of ambient dose equivalent rates and its space and time distribution for the next 30 years after the accident. In the present study, uncertain predictions of future ambient dose equivalent rates arising from variability in model parameters were assessed using Monte Carlo simulations. The model parameters were the ecological half-life for the fast/slow elimination components and the fractional distribution of fast elimination component. It was found that ambient dose equivalent rates for the next 5-30 years after the accident would be predicted within a factor of approximately 2. The long-term prediction models would be useful for a better understanding of the radiological situation since they provide information on the time variation of the ambient dose equivalent rates in inhabited areas.

Introduction

The Fukushima Daiichi nuclear power plant accident that occurred in March of 2011 resulted in widespread radioactive caesium contamination in the eastern region of Japan. People in inhabited areas could be exposed to radiation from the radioactive caesium deposited on grounds. Models are needed to predict how ambient dose equivalent rates might change in the future and to understand the local radiological situations in the long-term contamination [1, 2, 3, 4]. The Japan Atomic Energy Agency has developed long-term prediction models described using a double exponential form with ecological half-lives -the time for half the ambient dose equivalent rate due to natural removal phenomena and human activites- for land use. To describe the time variations of ambient dose equivalent rates from the radioactive caesium deposited on grounds, the model parameters such as the ecological half-lives for the fast elimination components and the fractional distributions of fast elimination component were derived from observations within the 80 km radius of the Fukushima Daiichi nuclear power plant [5, 6, 7, 8]. In the present study, uncertainties concerning future ambient dose equivalent rates were assessed using Monte Carlo simulations of model parameter variability in conjunction with the long-term prediction models.

Materials and methods

Long-term prediction models and model parameters

Based on the previous studies on the long-term prediction models [1, 2, 3, 4], the following expression can be deduced for the change with time in the ambient dose equivalent rate contribution from radioactive caesium on grounds:

$$Y(t) = Y(0)\left\{\alpha_{\mathrm{fast}}\exp\left(\frac{-\ln 2}{T_{\mathrm{fast}}}\cdot t\right) + (1-\alpha_{\mathrm{fast}})\exp\left(\frac{-\ln 2}{T_{\mathrm{slow}}}\cdot t\right)\right\} \times \frac{k\,\exp(-\lambda_{134}\cdot t)+\exp(-\lambda_{137}\cdot t)}{k+1} + Y_{BG} \quad (1)$$

where *Y(t)* is the ambient dose equivalent rate at time *t*; *Y(0)* is the ambient dose equivalent rate at time zero; α_{fast} is the fractional distribution of fast elimination component; T_{fast} is the ecological half-life for the fast elimination component; T_{slow} is the ecological half-life for the slow elimination component; λ_{134} and λ_{137} are the physical decay constants of ^{134}Cs and ^{137}Cs, respectively; *k* is the ambient dose equivalent rate ratio of ^{134}Cs to ^{137}Cs at time zero, 2.7 (as of March 15, 2011) ; Y_{BG} is the background radiation, approximately 0.05μSv/h.

The model parameters such as the ecological half-lives for the fast elimination component and the fractional distributions of the fast elimination component are summarized in Tables 1 and 2, respectively [8]. They were evaluated using the changes with time in ambient dose equivalent rates through vehicle-borne surveys within the 80 km radius of the Fukushima Daiichi nuclear power plant. They were obtained from the best curve fit when plotting the ambient dose equivalent rates over time. The model parameters were mainly categorized into the evacuation order area and other than the area. The parameters for other than the evacuation order area were categorized according to the precise land-use and land-cover map by the advanced land observing satellite "Daichi" (ALOS) [9]. Eight categories were classified: water, urban, paddy, crop, grass, deciduous forest, evergreen forest and bare surface in ALOS. The parameters for the evacuation area were categorized based on whether or not the land use area falls under deciduous and evergreen forest areas. In accordance with the previous studies [1, 3], the ecological half-life for the slow elimination component is given: 45 y for the 5th percentile, 90 y for the median and 135 y for the 95th percentile. The distribution is assumed to be uniform.

Uncertainty in ambient dose equivalent rate predictions

Uncertainties in model predictions arise from model parameters. In the present study, the uncertainties in predictions of ambient dose equivalent rates for the next 30 years after the accident were assessed using Monte Carlo simulations of model parameter variability. The Monte Carlo simulations with Latin hypercube sampling (LHS) [10] were performed. Use of LHS method ensured that sampling occurred from the complete distribution of each model parameter. Each model parameter distribution was divided into equal probability sections, and random sampling from each equal probability section was carried out to produce a probability distribution function. Three model parameters were taken into consideration: the ecological half-life for the fast/slow elimination components and the fractional distribution of fast elimination component shown in Tables 1 and 2.

Table 1 Ecological half-lives for the fast elimination component [8]

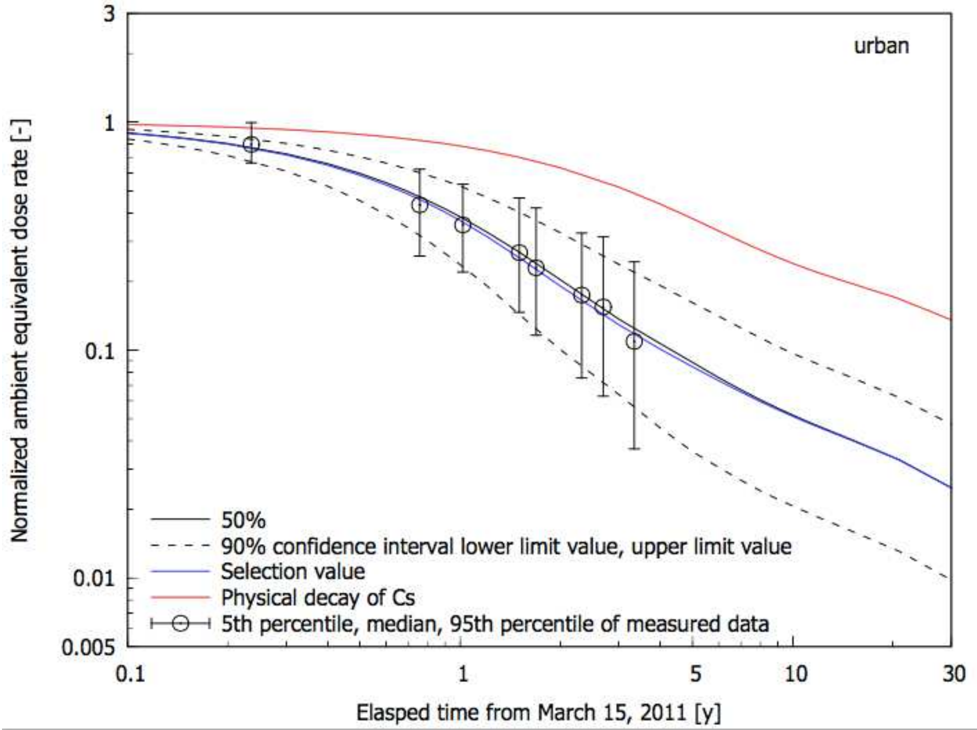
Evacuation order area	Land-use in ALOS	Ecological half-lives for the fast elimination component [y]			
		5th percentile	Median	95th percentile	Distribution
outside	water	0.25	0.56	1.2	log normal
	urban	0.35	0.60	1.7	
	paddy	0.32	0.55	1.5	
	crop	0.32	0.63	1.9	
	grass	0.29	0.58	2.2	
	deciduous forest	0.29	0.66	2.7	
	evergreen forest	0.28	0.94	5.7	
	bare surface	0.31	0.62	1.6	
inside	forest	0.29	0.68	3.1	log normal
	other than forest	0.32	0.60	1.7	

In addition, to validate the long-term prediction models, the prediction values of ambient dose equivalent rates based on the seventh vehicle-borne survey data (from November 5 to December 12, 2013) were compared with the eighth vehicle-borne survey data (from June 23 to August 8, 2014).

Table 2 Fractional distributions of the fast elimination component [8]

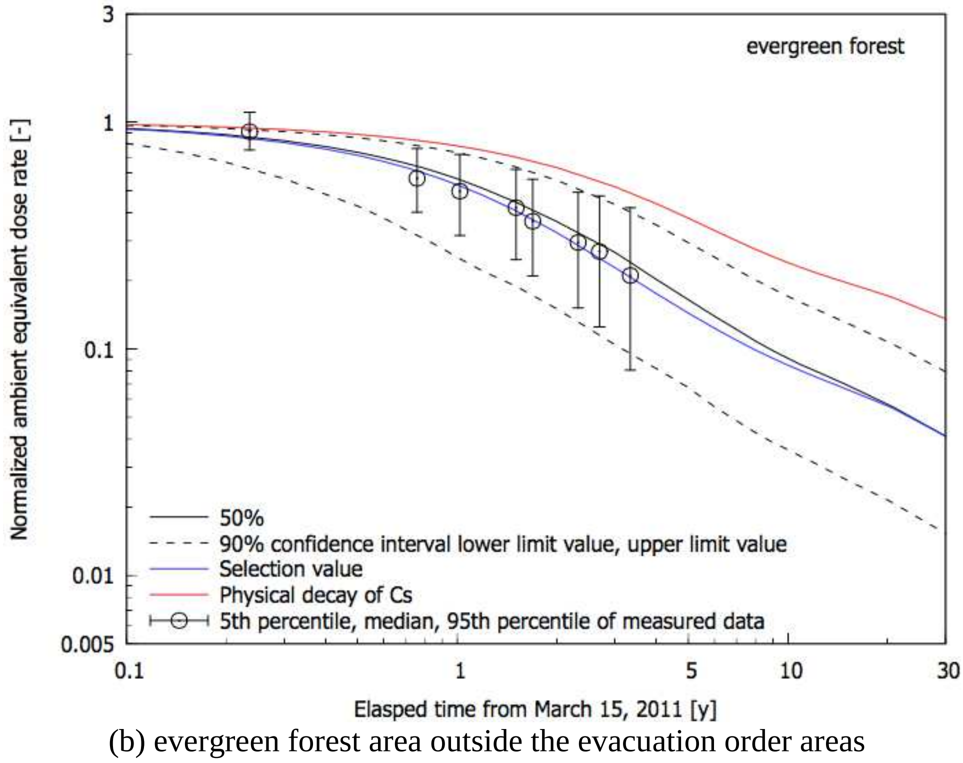
Evacuation order area	Land-use in ALOS	Fractional distributions of the fast elimination component [-]			
		5th percentile	Median	95th percentile	Distribution
outside	water	0.53	0.76	0.89	normal
	urban	0.52	0.77	0.93	
	paddy	0.53	0.75	0.93	
	crop	0.48	0.71	0.89	
	grass	0.47	0.72	0.92	
	deciduous forest	0.41	0.68	0.88	
	evergreen forest	0.26	0.62	0.86	
	bare surface	0.51	0.73	0.90	
inside	difficult to return, forest*	0.32	0.45	0.55	normal
	difficult to return, other than forest	0.33	0.47	0.57	
	restricted residence, forest*	0.37	0.48	0.61	
	restricted residence, other than forest	0.39	0.49	0.65	
	cancel preparation, forest*	0.37	0.51	0.67	
	cancel preparation, other than forest	0.29	0.51	0.69	

* deciduous and evergreen forest areas

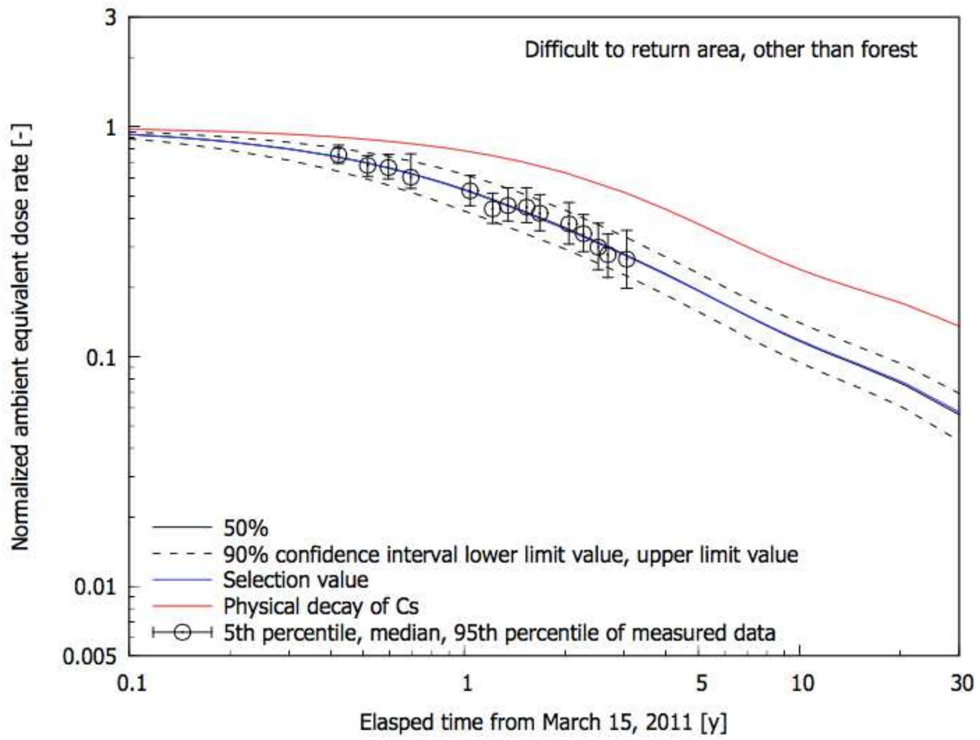


(a) urban area outside the evacuation order areas

Figs. 1 Relative ambient dose equivalent rates outside/inside the evacuation order areas (1/3).



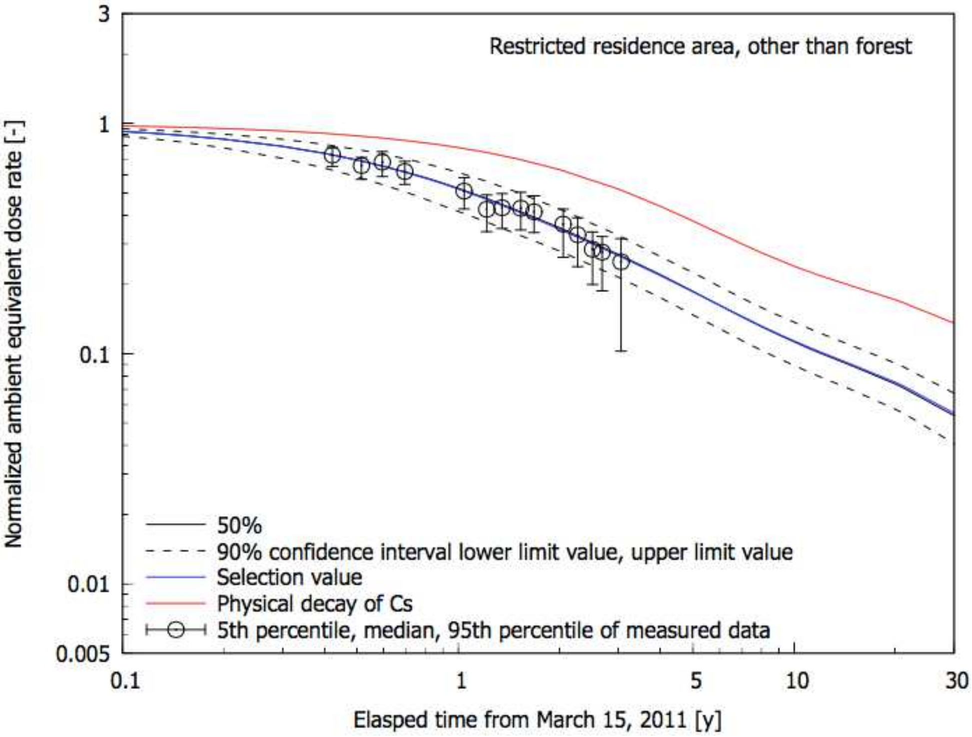
(b) evergreen forest area outside the evacuation order areas



(c) difficult to return area, other than forest area inside the evacuation order areas
Figs. 1 Relative ambient dose equivalent rates outside/inside the evacuation order areas (2/3).

Results and discussion

Four examples of the time variations of the relative ambient dose equivalent rates outside/inside the evacuation order areas are shown in Figs. 1. They are predictions for land-use. In all figures, the 5th to 95th percentile range –the 90% confidence interval- of relative ambient dose equivalent rates through the vehicle-borne surveys and the uncertainties in predictions of relative ambient dose equivalent rates for the next 30 years after the accident are indicated. From the figures, it can be stated that the medians (the 50th percentiles) of predictions of relative ambient dose equivalent rates are in good agreement with data through the vehicle-borne surveys. In addition, it is confirmed that almost all of data through the vehicle-borne surveys are within the prediction intervals.



(d) restricted residence area, other than forest area inside the evacuation order area

Figs. 1 Relative ambient dose equivalent rates outside/inside the evacuation order areas (3/3).

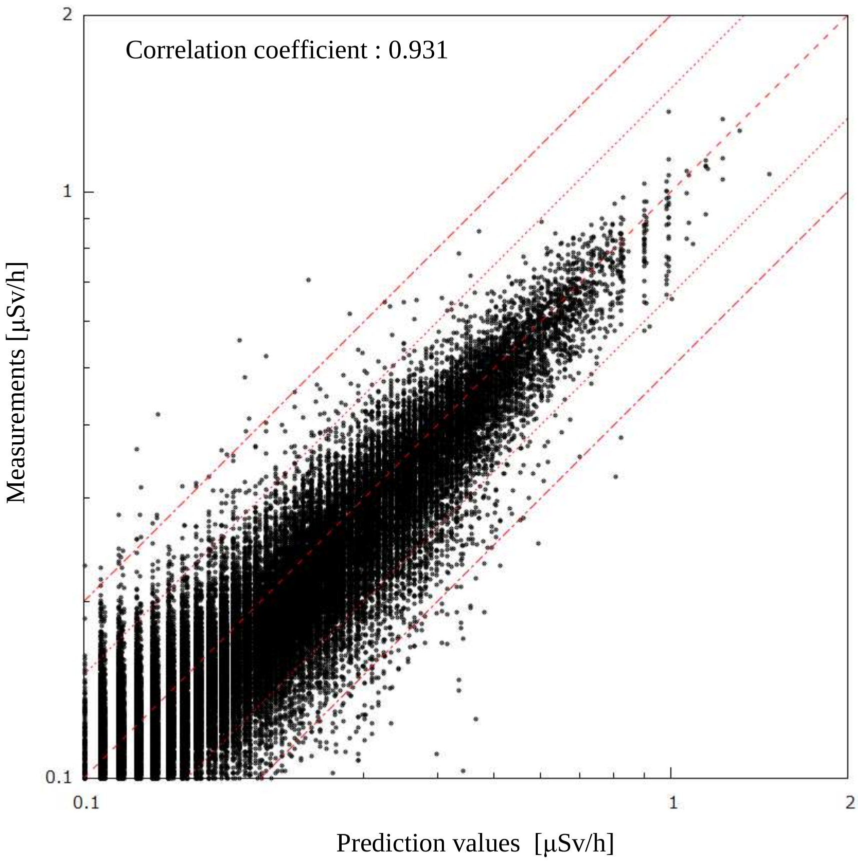


Fig. 2 Comparisons of ambient dose equivalent rates between predictions and the measurements through the eighth vehicle-borne survey.

Figure 2 shows comparisons of ambient dose equivalent rates between those predicted using the long-term prediction models with the seventh vehicle-borne survey data and those through the eighth vehicle-borne survey. They are the ambient dose equivalent rates over 0.1μSv/h for the outside the evacuation order areas. It was found that the predictions agree within a factor of approximately 2 with the actual data. Hence, the long-term prediction models would be useful for a better understanding of the radiological situation.

Conclusions

The uncertainties in predictions of ambient dose equivalent rates for the next 30 years after the accident were assessed using Monte Carlo LHS simulations. It was found that the ambient dose equivalent rates for the next 30 years after the accident would be predicted within a factor of approximately 2. The long-term prediction models would be useful for follow-up of the radiological situation. We plan in the near future to update the long-term prediction models using the latest survey data.

Acknowledgement

This study was conducted in a government-commissioned research project from the Nuclear Regulation Authority of Japan.

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S4-O5

Dispersion model based dose-rate measurement simulation for exercises

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Abstract

In Finland there is a dose rate monitoring station network with over 250 stations. Measurements are available for authority use in real time in the Internet. For exercise purposes STUK has a dedicated exercise Web page where simulated dose rate results are shown. The simulated dose rates are based on sophisticated dispersion models. During exercises the simulated results are shown in the exercise Web page together with measured results. This creates very real feeling for the exercise participants. The exercise system is nowadays in regular use in national exercises in Finland.

Introduction

The external dose rate monitoring network ULJAS covers over 250 monitoring stations geographically evenly distributed over Finland. The dose rate results are collected and presented by the USVA system. It is used to produce the real-time overview of radiological situation in the country. The USVA system also alerts STUK's expert on duty when dose rate exceeds station specific alarm limit.

The Radiation and Nuclear Safety Authority STUK has developed tools to simulate dose rate measurements for these ULJAS stations. The USVA system has a separate Web site for exercise usage. The system is called "Exercise USVA". The Web interface has a similar layout compared to normal dose rate monitoring network view except for the large "exercise" text appearing on the site. In this site dose rate measurements are simulated and values are based on atmospheric dispersion models.

Dose rate measurement simulation process

<u>Usage</u>

The simulation is created and carried out using Web-based emergency management system TIUKU (formerly known as KETALE). TIUKU is also used for running and displaying dispersion model results, dose calculations, analyses and reporting etc. [1] System is very easy to use and even non-experts can be trained to run simulations. By integrating the simulation to TIUKU system it is verified that all the development made for normal dispersion model usage is also available for simulation purposes.

Before starting the simulation, the simulator user asks from the meteorologist of the FMI to make an assumption of the best NWP model for current weather conditions. Then the simulator starts the dispersion calculation using predefined source term and the newest numerical weather prediction (NWP). The model output doses and dose rates which are interpolated to the coordinate points of the dose-rate monitoring stations.

<u>Dispersion Models</u>

Currently the simulator supports long-range dispersion model SILAM developed and provided by Finnish Meteorological Institute (FMI) and short/medium range dispersion models included in JRODOS Decision Support System. The simulations are not geographically limited to Finland and can be performed world-wide. Because of the modular structure of the system additional dispersion models can be easily expanded into the system.

SILAM is a long-range model but even 2.5 km spatial resolution can be achieved when HARMONIE model data is used. SILAM runs on FMI's supercomputers and there is a dedicated Web Service for STUK to run calculations. For SILAM multiple NWP models (ECMWF, HIRLAM, and HARMONIE) are constantly available. For special use cases also historical NWP data can be used. The simulation has two separate SILAM calculations. The first calculation is done for longer distance (approx. 200-500 km) with sparse grid (~10 km) and the second calculation is for shorter distances in the vicinity of nuclear power plant for instance. This is done because the density of the monitoring stations is typically higher near the nuclear power plants. STUK calculates dose rates from SILAM results and produces NetCDF files which can be further used to produce simulated measurements.

JRODOS Decision Support System contains four different short/medium range dispersion models: ATSTEP, RIMPUFF, DIPCOT, and LASAT. User can select if the dispersion is based on NWP data or weather input by hand. In JRODOS there is a nested calculation grid and thus multiple calculations are not necessary. JRODOS can export the calculation results to GIS database from where the simulator reads the values.

Simulator software

The simulator software can simulate dose rate results to all known monitoring stations in Europe. However any location can be used for simulation in addition to existing dose rate monitoring

Before the simulation begins, the exercise database is cleared and three weeks of real measurement data is copied into exercise database. If the simulation time is in the past the real measurement results are overwritten by simulated results.

The dose rate results calculated by the dispersion models do not take background radiation into account. The simulator software uses real measurements as background and the reported value is calculated by adding the value read from dispersion model output to background. If the measured result does not exist for given point, dose rate value 0.1 μSv/h is used as background.

As the exercise progresses the simulated measurements incremented by background values are inserted into database with 10 minutes interval. The simulator user can follow the status of simulation in the Web page. The web page constantly shows the list of simulated stations in every time cycle and the remaining time of the whole simulation process.

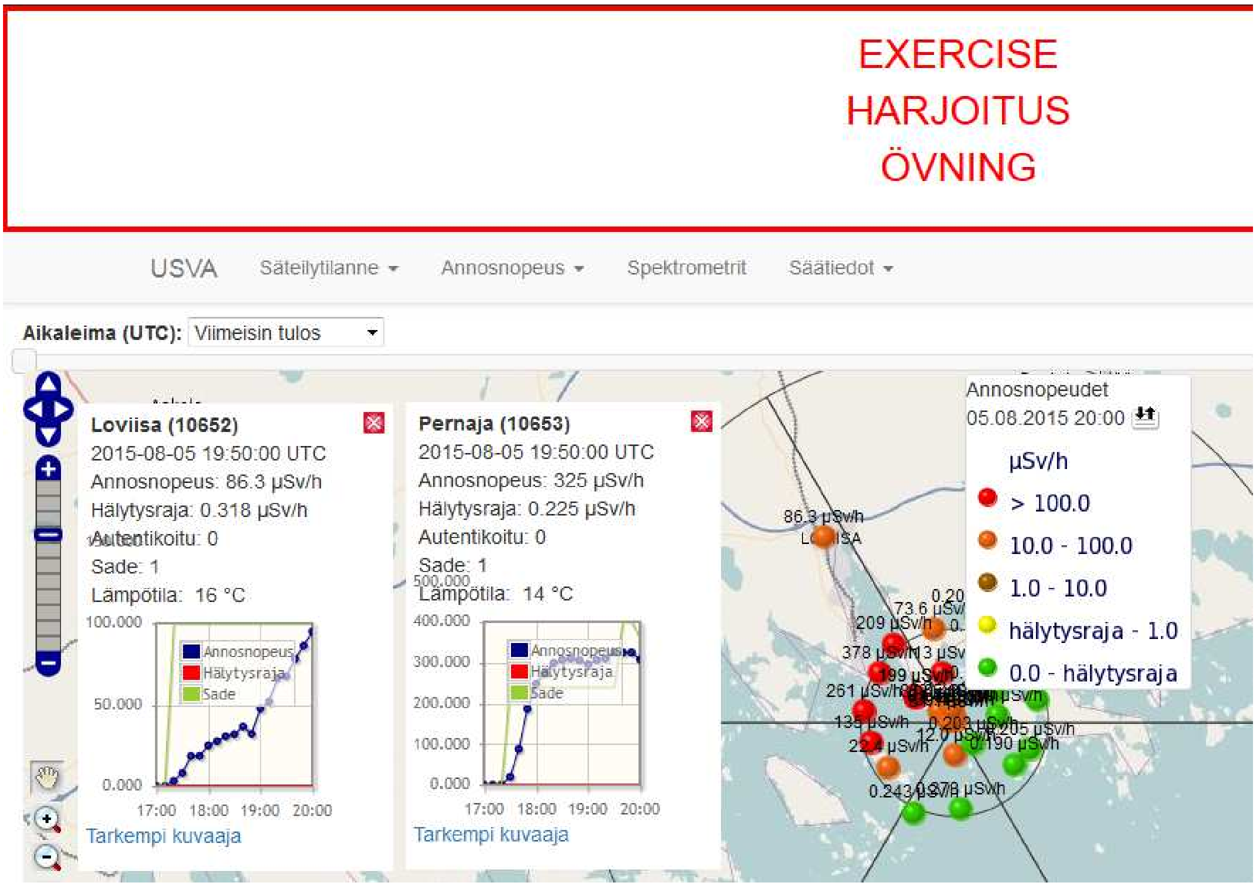
Exercise USVA

The end users can follow the simulation via Exercise USVA Web Page. Like the operational USVA Web page it is available on the Internet but it is password protected. The results are accessible for all the players in the exercise including the rescue officers. In national exercises the URL of the page is written in exercise instructions and participants are encouraged to follow this site during the exercise.

Figure 1: A snapshot of USVA web view with simulated results. The current situation can be represented on the map. Also time dependence in selected stations can be shown. Green is normal dose rate and yellow all the rest values are above mutable alert limit ("hälytysraja" in Finnish).

The results can be viewed in map. Several background maps are available including Google maps. The map view provides a good overview of the situation. In addition number of other methods to view results is available. These include flexible plotting tools for example.

The USVA Web page is not only for viewing dose rate monitoring network results. It also has a capability to display trajectories and weather information. With trajectory information the user can check if the dispersion direction matches with "measured results". Weather information cover radar and weather station network observation overlays. Also each monitoring station have rain detector and time dependence of rain information together with dose rate result can be viewed (see Figure 1).



The results are also available in the TIUKU system. This makes is possible to display model results together with simulated results on the map. If the model parameters match with the simulation dispersion model the results should look the same. Also WMS/WFS service for the results is available and can be used with various GIS applications.

Conclusions

Compared to traditional simulation methods for exercises STUK's system has many advanced features. It can use the most sophisticated dispersion models available and it is not limited to one dispersion model or one weather prediction model. NWP based real weather conditions including historical data are supported but with JRODOS the weather input by hand can also be used. The real-time availability of results online for large number of players is unique feature and following of results via Web makes it possible to have more realistic feeling of an exercise.

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S4-O6

DEMAs experiences with unmanned aerial vehicles for radiological measurements

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Unmanned radiological measurement can be useful for a wide range of scenarios including mapping of plume passage and ground contamination during and after an accident with a nuclear power plant.

The Nuclear Division at DEMA has completed tests with an unmanned X8 helicopter from Danish Aviation Systems ApS carrying an off-the-shelf Canberra Colibri dose rate monitor with a CsI detector. The X8 has a pay load of 1.5 kg and can fly programmed patterns with a fly time of approximately 20 minutes. The Colibri dose rate monitor has a build-in GPS and a logging function that can store dose rates and coordinates every 1 s.

Preliminary tests show that the system readily locates radioactive sources on the ground and the obtained data combined with GIS software can be used to create maps with dose rates and terrain.

In the future, DEMA will among other things work on reducing the weight of the dose rate monitor and possible installing it in a fix wing unmanned aerial vehicle. This will increase the range and the operational functionality of unmanned radiological measurements considerably.

S5-O1

Measurement requirements to maximise recovery phase dose reduction in large contaminated land areas

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Numerous European and Nordic guidance handbooks have been produced over the years for optimisation of recovery countermeasure strategies following a widespread terrestrial radioactive contamination. Here the experience from particularly clean-up efforts made in areas contaminated by the Chernobyl accident has been summarised to provide decision makers in future contamination scenarios with an overview of how to construct optimised strategies for intervention in contamination scenarios with different characteristics. However, all along the way, the focus has been on enabling selection of the most suitable countermeasure options, and not on how to best ensure that they are implemented in the best possible way in practice. Operational optimisation of countermeasure implementation requires dedicated measurement strategies to assess the local contaminant distribution as well as parameters that will enable proper estimation of residual dose. It has previously been demonstrated experimentally in Chernobyl-contaminated areas that where military standard operations failed in reducing contamination levels significantly, essentially the same countermeasure strategy, when carried out in accordance with simple but specific contamination measurements in the area, could reduce dose greatly and highly cost-effectively. Yet there is still a profound gap in the literature on how to carry out measurement-based countermeasure implementation. For example ICRU and IAEA guides for sampling and measurement entirely fail to take into account the purpose of sampling. In connection with the Fukushima accident huge numbers of soil samples were taken with the stated purpose of ‘evaluating Fukushima derived radionuclides’, but the lack in specificity is reflected in generally rather low dose reductive effects in the contaminated areas. The presentation pinpoints the current problems in this context and proposes a way to improve operational effectiveness through measurement recommendations.

An accidental exposure to I-131

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In a company manufacturing radiopharmaceuticals, a laboratory technician was contaminated with I-131 while preparing I-131 capsules for treatment of thyroid carcinoma. She was wearing two pairs of protective gloves and, when changing gloves, noticed a rupture in the right inner glove but did not notice any rupture in the outer glove. Only 3-4 h later, routine monitoring revealed heavy contamination of the back of her right hand. Immediate actions to decontaminate the hand were taken on-site. On the next day, besides persisting heavy contamination of the hand, I-131 was also detected in the thyroid gland.

Based on the measurements on-site and later follow-up at STUK, including surface contamination measurements and whole body counting, the original I-131 activity on the hand was estimated at 12 MBq and the superficial skin dose at 33 Gy, affecting a skin area of about 10 cm^2. The thyroid dose was estimated at 430 mGy. Stable iodine would have reduced the thyroid dose substantially but was not administered.

Eleven days after the incident the skin was dry and slightly desquamated but no severe skin damage developed. Four days later the skin was intact with no desquamation left. No further signs of skin damage had occurred when the person was seen after three months, and none have come to our knowledge later on. Cytogenetic analysis of circulating lymphocytes was performed and a slight elevation of chromosomal aberrations was observed, pointing to a small, partial exposure, which is in line with the incident. Health Phys. 107(4):351–355; 2014

S5-O3

Scenario Based Nuclear and Radiological Emergency Preparedness in a Non-Nuclear Country (Norway)

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Severe nuclear or radiological events may have significant consequences, with a large and immediate need for information, considerable challenges related to implementation of protective measures, advice to the general public and affected stakeholders, and decisions on other mitigating actions. A number of authorities will be involved, and the management of the event will most likely involve many Professional and governmental environments and fields. Recovery and long-term rehabilitation in the aftermath of the event may be demanding. Management of severe events will also require international coordination. In emergency planning, the aim is to form a preparedness that provides the best protection based on the resources available. In May 2010, the Norwegian government established six scenarios in order to priority needs and to plan an as good as possible nuclear and radiological emergency preparedness in Norway. The scenarios are based on experiences from previous events and evaluations of existing and future activities. They form a comprehensive list and qualitatively describe possible events with different consequences, and each represents distinct aspects towards crisis management. The six scenarios are:

1. Large airborne release from a facility in another country
2. Airborne release from a facility in Norway
3. Local event from a mobile source
4. Local event evolving over time
5. Marine release, and/or fear of marine or terrestrial contamination
6. Severe event abroad without direct consequences for Norwegian territory.

The primary users of the scenarios are all authorities with responsibilities in nuclear and radiological emergency preparedness and response. The scenarios id in reviewing crisis management plans, developing specific scenario based communication plans, raise general consciousness on different needs in emergency response, and provide a comprehensive approach towards emergency preparedness planning.

S5-O4

Online courses in radiation protection

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The new master programme in radiation safety at the University of Gothenburg has attracted the interest of many students. To give students a good start, we prepare online courses including recorded lectures, quizzes, calculation exercises, and recommendations to further reading. The aim of the courses is to effectively teach the basics of radiation physics, detectors of ionizing radiation and gamma spectrometry.

Massive Open Online Courses (MOOC) is offered by several Universities. The courses often consists of short recorded lessons (3-15 min) directly followed by a quiz or a calculation exercise. For many students, short bursts of information followed by a quiz are an effective method for learning. With support from the Swedish Radiation Safety Authorities (SSM), some of the courses given at the master programme are also available to medical physicists, PhD-students and professionals working in the nuclear power industry as continuous professional development courses (CPD). The online material will be available to CPD participants as well. In the future, online courses may be an interesting alternative to the intense CPD courses given today. The first online course: Radiation detectors and measurement in radiation protection and preparedness and will be launched in the fall of 2015. The first versions will be given in Swedish; however, material is continuously translated into English.

S5-P1

Elemental Composition and Structure of Commercial Available Personal Radiation Shielding Protective Clothing

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Recently, the demand for personal protective equipment for radiation protection is increasing. The laboratory of Nuclear Protection Department at National Institute for NBC Protection (SUJCHBO v.v.i.), Czech Republic deals with a study of selected personal protection equipment properties. This Research is supported by the project called “Prevention, Preparedness and Mitigation of Severe Accidents at Czech Nuclear Power Plants in Connection with New Findings of Stress Tests after the Fukushima Accident” (project No. VG 20132015105, founded by the Ministry of the Interior of the Czech Republic). As a part of this research a collection of commercially available personal radiation shielding protective clothing (RSPC) was gathered at National Institute for NBC Protection. This protective clothing is a kind of protective garments used not only against radioactive contamination, but due to a special shielding layer RSPC protect against penetrating ionizing radiation as well. RSPC are possible to be used by special emergency response technical workers during accidents in nuclear facilities or by first responders and firemen in emergency situations such as accidents during radioactive material transportation, terrorist incidents involving radioactive dispersal devices or nuclear weapons. As a first part of the study there was determined X and gamma radiation attenuation in the materials of individual RSPC. During further research, it is intended to determine the effective dose to man using this protective garments and staying in a radiation field. This determination will be realized by Monte Carlo simulations using MCNPX code. For this purpose it is necessary to know the elemental composition of RSPC, especially the composition of the shielding layer. The elemental composition was determined by X-ray fluorescence analysis. The results were then refined using chemical analysis. In addition the material structure of RSPC was studied using electron microscopy. During the structure study, the elemental composition of shielding layers of selected shielding materials was determined using Energy-dispersive X-ray spectroscopy as well. In the contribution, the collection of commercially available RSPC will be presented with regard to their shielding material. There will also be presented methods for determining the elemental composition of RSPC and their results. Finally, the shielding materials of RSPC will be compared with respect to their structure.

Probabilistic Off-site Consequences Analysis – development of a guiding document

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Probabilistic Off-site Consequences Analysis, commonly referred to as Level 3 Probabilistic Safety Assessment (Level 3 PSA), is infrequently performed and generally regarded as a less developed analysis when compared to Level 1 and Level 2 PSA. Due to new nuclear construction plans there is a renewed interest in objective and risk-based siting analysis in order to better understand off-site consequences, especially in the wake of the multi-unit disaster at the Fukushima Daiichi site. Based on an inquiry from the Nordic PSA Group and the Nordic Nuclear Safety Research group (NKS), a consortium of Swedish nuclear risk consultancies (Lloyd's Register Consulting, ÅF Industry and Risk Pilot) and the Finnish research institute VTT has begun a multi-year study of Probabilistic Off-site Consequences Analysis (Level 3 PSA). This last year of the project, Vattenfall joined the consortium to provide radiation protection expertise in the pilot studies performed and in the development of a Nordic guiding document on Level 3 PSA. The objective of the project is to further develop understanding within the Nordic countries in the field of Level 3 PSA; the scope of its application, its limitations, the appropriate risk metrics, and the overall need and requirements for performing a Level 3 PSA. The project activities include:

- Performing an industrial survey about expectations on a Level 3 PSA, which included several workshops and meetings with Nordic utilities, regulators, and safety experts
- Investigate the use and benefit of L3 PSA risk metrics
  - Health Effects: Individual and Collective Dose, Early/Late Effects
  - Environmental Effects: Ground contamination, “loss” of land
  - Economic: Food bans, relocation, land decontamination, business impacts, etc.
- Investigate the use and benefit of countermeasures (evacuation, sheltering etc.)
- Investigate the interfacing to L2 PSA: Additional requirements to existing L2 PSA Two pilot studies with the aim to
  - illuminate possibilities and identify key uncertainties and limitations in Level 3 PSA analysis
  - Establish resource requirements for production of a Level 3 PSA

The final activity will be the development of a guiding document for Level 3 PSA. This presentation will focus on radiation protection aspects in performing a Level 3 PSA, especially in the development of this guide.

S5-P3

Characterization of HPGe detectors using Computed Tomography

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Computed Tomography (CT) high-resolution imaging have been used to investigate if there is a significant change in the crystal-to-window distance, i.e. the air gap thickness, in a small n-type detector cooled to 77 K, and in a medium sized p-type HPGe detector when cooled to 100 K. The findings were compared to detector dimension data made available by the manufacturer. The air gap thickness increased by (0.38±0.07) mm for the n-type detector and by (0.40±0.15) mm for the p-type detector when the detectors were cooled to 77 resp. 100 K compared to at room temperature. Monte Carlo calculations indicate that these differences have a significant impact on the efficiency in close geometries (< 5 cm). In the energy range of 40-700 keV with a source placed directly on endcap, the change in detector efficiency with temperature is 1.9-2.9% for the n-type detector and 0.3-2.1% for the p-type detector. The measured air gap thickness when cooling the detector was 1.1 mm thicker than manufacturer data for the n-type detector and 0.2 mm thicker for the p-type detector. In the energy range of 40-700 keV and with a source on endcap, this result in a change in detector efficiency of 5.2-7.1 % for the n-type detector and 0.2-1.0 % for the p-type detector, i.e. the detector efficiency is overestimated using data available by the manufacturer.

Impact of atmosphere on the transport of Ruthenium in the primary circuit of nuclear power plant

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Ruthenium is a semi-volatile element originating as a fission product in nuclear reactors that can be released in case of a severe nuclear accident. This release is promoted by air ingress, high humidity, high temperature and oxidative conditions in the reactor containment when the consistency of primary circuit is lost. In this work impact of atmosphere composition on transport of ruthenium through the primary circuit was examined. In experiments silver nanoparticles were used as a representative aerosol in the primary circuit. Additionally impact of NO2 gas as product of air radiolysis was examined. Quantification of ruthenium transported both as a gas (RuO4) and aerosol was performed, to determine amounts of transported ruthenium. Chemical composition of ruthenium species was evaluated by XPS, XRD and Raman spectroscopy techniques. Significantly increased transport of gaseous ruthenium through the facility was detected while NO2 gas was in the atmosphere. In experiments conducted with both silver aerosols and NO2 in atmosphere transport of ruthenium in both gaseous and aerosol form was promoted. Conclusion was made that modification of the atmosphere composition in the primary circuit will have unneglectable effect on the amount of ruthenium transported to the containment during a severe accident.

S6-O1

Radiochemical analysis of important radionuclides in Nordic nuclear industry

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In the Nordic countries, there are 5 nuclear power plants in operation: Forsmark, OKG and Ringhals NPPs in Sweden and Loviisa and Olkiluoto in Finland. In addition, there are 3 research reactors JEEP II and HBWR in Norway and FiR-1 in Finland. Environmental safety of the nuclear installations is a sensitive and critical issue for the public and authorities. In the past years, an increased and more restrict environmental assessment program has been required by the authorities and initiated in the nuclear power plants and host institutions. For this purpose, some radionuclides difficult to measure, such as 14C, 63Ni, and 55Fe, have been added to the list of routine monitoring programme for discharges and circulation water. In the past years, some radiochemical analytical methods have been developed and applied in Nordic industry and research institutions for waste analysis, environmental monitoring and decommissioning.

However, because no suitable reference material is available, it is not easy to validate these methods. There is a high demand for a good approach for quality control and a great need for standard analytical methods for routine analysis of some important radionuclides. A joint NKS project on standardization of radio analytical methods for important radionuclides in Nordic nuclear industry (STANDMETHOD) has been launched by collaboration of 8 Nordic labs, the activities and achievement of this collaboration project are presented.

S6-O2

Multivariate analysis of release data and environmental monitoring data from Swedish nuclear facilities

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Abstract

On behalf of the Swedish Radiation Safety Authority a multivariate analysis of release and environmental data from some of the Swedish nuclear facilities has been performed. For the facilities in Oskarshamn (three BWR:s and an interim storage for nuclear fuel), Forsmark (three BWR:s) and Studsvik (a waste treatment facility and a closed down research reactor) an analysis for both water and air variables was performed. Earlier a similar study for the facilities in Ringhals (one BWR and three PWR:s) for water variables was performed on behalf of the Swedish University of Agricultural Sciences. For the nuclear power plants in Oskarshamn, Forsmark and Ringhals the radionuclides Co-58, Co-60, Cs-137, Zn-65, Mn-54 and Fe-59 were investigated and for the waste treatment facility in Studsvik Co-60, Cs-134, Cs-137, Eu-152, Eu-154 and Mn-54 were considered.

The aim of this project was to investigate if there are correlations between the measured released activity and the measured concentrations of radionuclides in the environment.

Multivariate analysis is a statistical tool which can analyse many variables at a time. It is here used to find correlations between samples in the environment and the releases of radionuclides to air and water.

Quantifiable correlations between measured releases and measured radioactivity in the environment were identified. Several of the investigated environmental samples showed significant correlations with the releases and mathematical models describing the relationship between releases and concentrations in the environment were identified.

The results from the multivariate analysis enables the identification of the most reliable sample types in the monitoring programs in order to reflect releases from the nuclear facilities. A proper selection of sample types allows the estimation of expected environmental concentrations from the release, in addition to estimation of historic releases based on environmental concentrations.

Introduction

On behalf of the Swedish Radiation Safety Authority a multivariate analysis of release and environmental data from some of the Swedish nuclear facilities has been performed. For the facilities in Oskarshamn [1], Forsmark [2] and Studsvik [3] an analysis, for both water and air variables, was performed. Earlier a similar study for the facilities in Ringhals for water variables was performed on behalf of the Swedish University of Agricultural Sciences [4]. In this report the analysis for Ringhals is not further described, but the analyses for Oskarshamn [1], Forsmark [2] and Studsvik [3] are summarised.

The nuclear facilities in Sweden measure their releases to air and water in accordance with the regulations. They are also measuring environmental concentrations of radionuclides according to a program for environmental control that the Swedish Radiation Safety Authority has decided.

Aim

The aim of this project was to investigate if there are significant correlations between the measured released activity and the measured concentrations of radionuclides in the environment. A goal was also to see how different sample types (species) can work as indicators for measured releases.

Method

Multivariate analysis is a statistical tool which can analyse many variables at a time. It was here used to find correlations between concentrations of radionuclides in samples in the environment and the measures releases of radionuclides to air and water.

Data

For the Oskarshamn facilities (three BWR:s and an interim storage for nuclear fuel) and the Forsmark facilities (three BWR:s) the nuclides Co-58, Co-60, Cs-137, Zn-65, Mn-54 and Fe-59 were considered. For the Studsvik facilities (a waste treatment facility and a closed down research reactor) the radionuclides Co-60, Cs-134, Cs-137, Eu-152, Eu-154 and Mn-54 were considered.

The data consists of annual averages of measured releases to air and water and annual averages of measured concentrations in the environment.

In the data matrix each year is an observation and each variable is defined as nuclide, sampling station and sample type (species). For some variables there are only a few observations and these are excluded. When the missing data is more than 80-85 % the variables are excluded.

Data analysis

As a first step Principal Component Analysis (PCA) is used to discover if a certain amount of data contains correlations. The PCA method can identify similarities or differences in the observations and also see different clusters or extreme values, so called outliers. The first principal component (PC1) shows the direction in the data which shows the largest variation. The next principal component (PC2) shows the second largest variation and so on. Thus PC1 shows a projection of the most important variables in the original data matrix. A score plot shows how similar the different observations are. A loading plot shows a map over how the different variables are correlated to each other.

Partial Least Square or Projection to Latent Structures (PLS) is used to quantify correlations between the multivariate matrix with release data (X) and the matrix with environmental concentrations (Y). The software starts by doing a PCA of Y and then of X and then an optimal alignment of X and Y, but not necessarily in the line of lowest common denominator. Score plots and loading plots for PLS also show, as in the case of PCA, a map of observations and variables respectively.

R2 shows how well the calculation models fit actual data. Q2 shows the predictability of the model for new X and Y values. Q2 is calculated by cross validation which means that one or more observations are excluded and a sub-model is calculated by using the rest of the observations. The sub-model is then used as a basis for prediction of the excluded observation. When all observations has been excluded once, all the uncertainties in the predictions are added and a mean uncertainty can be calculated which leads to Q2.

A Gaussian plot shows if the residuals (the difference between observed and predicted values) are normally distributed. The graph shows the probability as a function of standard deviation (stdv) for every observation. Observations with more than 4 stdv are probably outliers.

Analysis of variance testing of Cross-Validated predictive residuals (CV-ANOVA) is a diagnostic tool for assessing the reliability of PLS models. CV-ANOVA is a crucial test to assess if the calculated model is significant or just stochastic.

More details on the multivariate tool used in the study can be found at www.umetrics.com.

Example of the method applied on Forsmark data

Figure 1 shows a score plot of the releases to air and water from Forsmark for the period 1980-2012 for one model. If the observations are correlated the data points are close to each other in the plot. For example if the releases to air and water were correlated in some way this would show in the plot. In this case the correlation between releases to air and water is in general weak but the releases to water, Fe-59 excepted, of the different nuclides seem to be correlated between the different nuclides. In the case of Fe-59 there are a lot of missing data for both air and water and this can be the reason why the nuclide deviates from the others.

In Figure 2 the loading plot of all the variables, both releases and environmental concentrations for the period 1980-2012 for one model, are seen. The pattern for the releases is similar to the one in Figure 1. To identify correlations between the different environmental variables one must really scrutinise the plot. The computational algorithms seek the best possible fit for X (releases) to Y (environmental concentrations) and when this is done some variables will end up with negative correlations. Although it is not reasonable that increased releases lead to reduced environmental concentrations.

Figure 3 shows how well the particular model represents the actual data (R2) and how well the model can predict new values for releases and environmental concentrations (Q2). Q2 cannot be negative and therefor this model is not useful for Fe-59 in air and water and Zn-65 in air.

The Gaussian plot in Figure 4 shows that the residuals can be regarded as normally distributed except for the 1986 observation (marked with red) that deviates by almost three stdv.

The model M36 with 3 variables is significant and the regression coefficients are showed in Figure 5. M36 can be used to predict releases of Co-58 to water from measurements of environmental concentrations with a prediction ability of 96,7 %. A model with many coefficients will be resource intense. To get more easily handled models the coefficients can be reduced but often also the prediction ability is reduced. If for example the largest coefficient is used, station S111 and bladder wrack (S111_BlåsT) a model with the prediction ability 78,6 % is obtained. But if the next coefficient, station S101 and ongrowth (S101_Påväx), a model with the prediction ability 89,7 % is obtained. Observed and predicted value with this model, M176, can be seen in Figure 6.

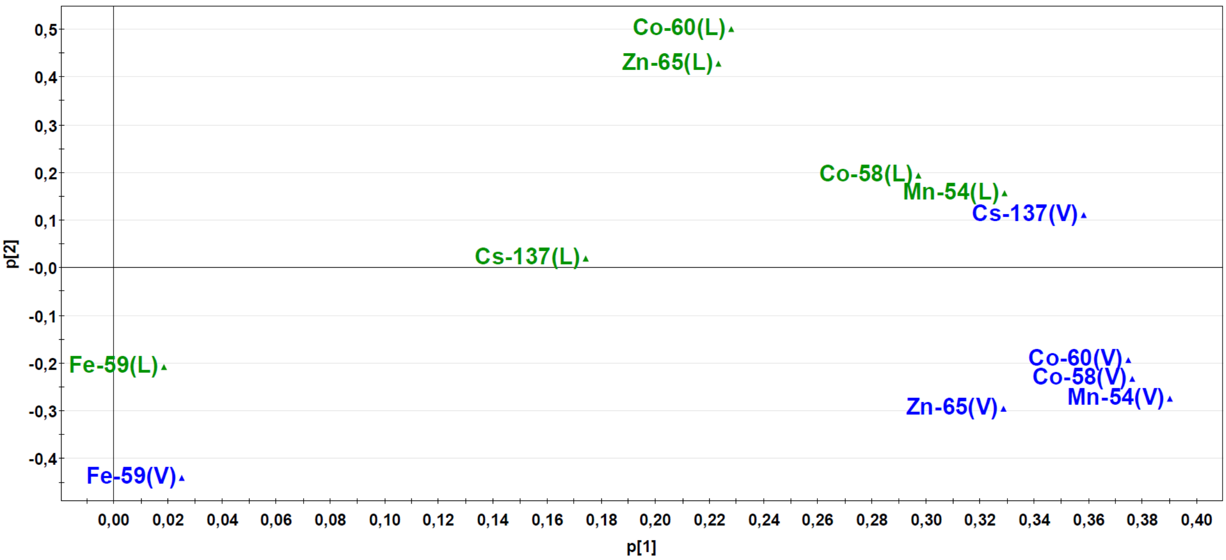

Figure 1. PCA score plot for the releases to air (green) and water (blue) from Forsmark for the period 1980-2012 for one model (M3).

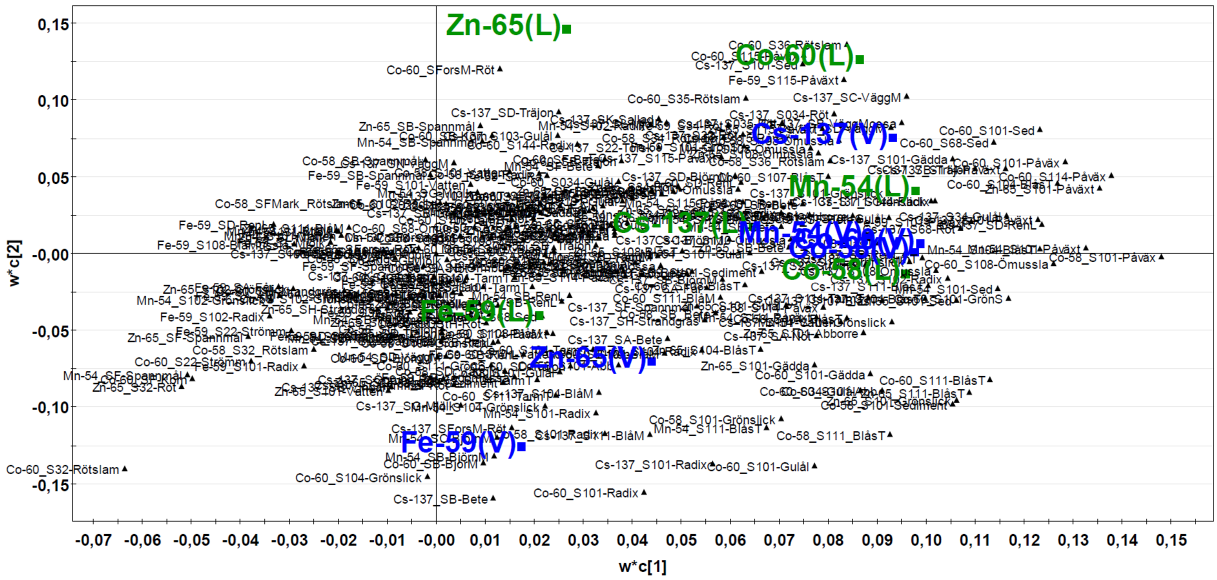

Figure 2. PLS loading plot for all variables, both releases to air (green) and water (blue) from Forsmark and environmental concentrations (black) in the vicinity of Forsmark for the period 1980-2012 for one model (M4).

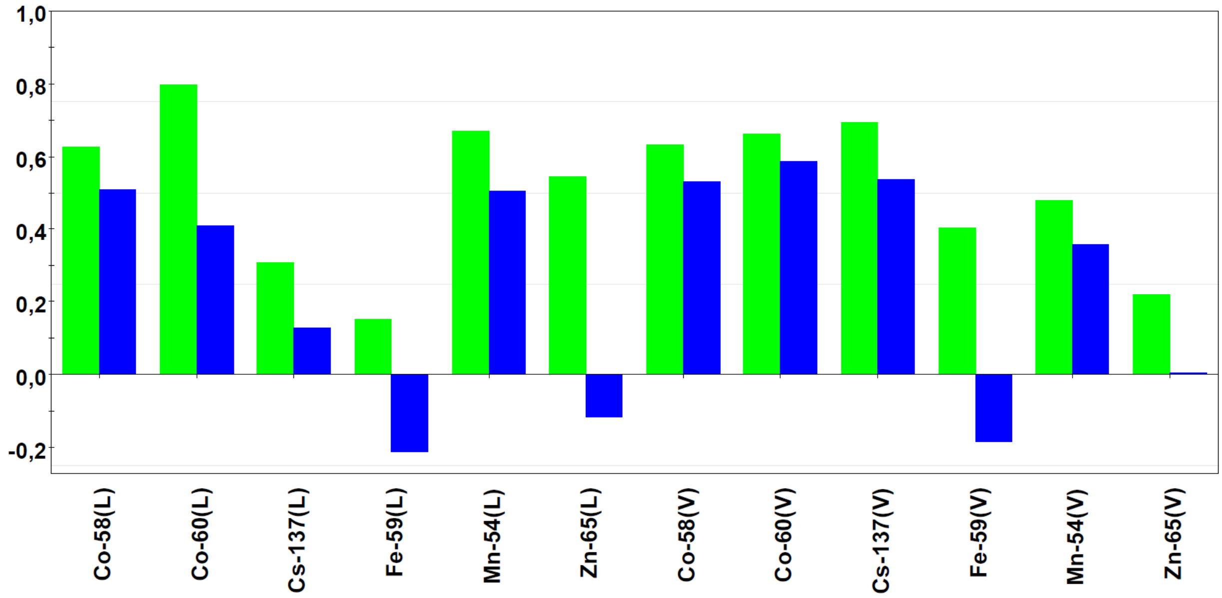

Figure 3. R2 (green) and Q2 (blue) for different nuclides relating to releases to air (L) and water (V) for one model (M4). R2 shows how well the particular model represents the actual data and Q2 shows how well the model can predict new values for releases and environmental concentrations.

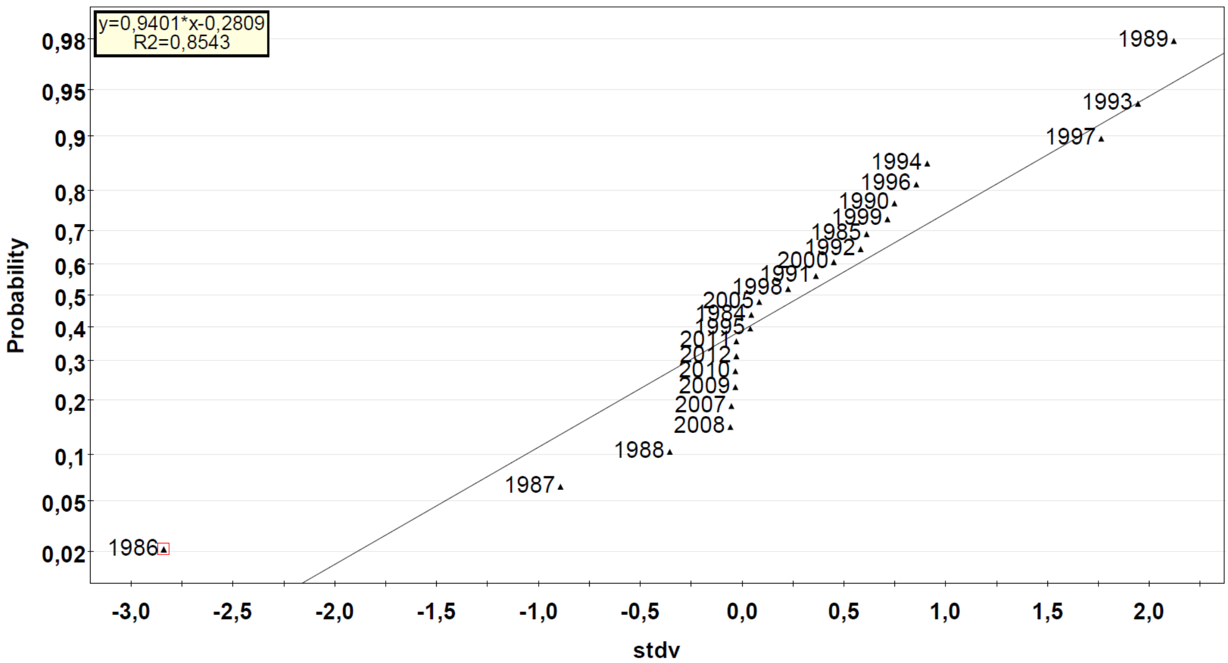

Figure 4. Gaussian plot for Cs-137 to water from Forsmark for the period 1980-2012 for one model (M136). In this case the residuals can be regarded as normally distributed except for the 1986 observation (marked with red).

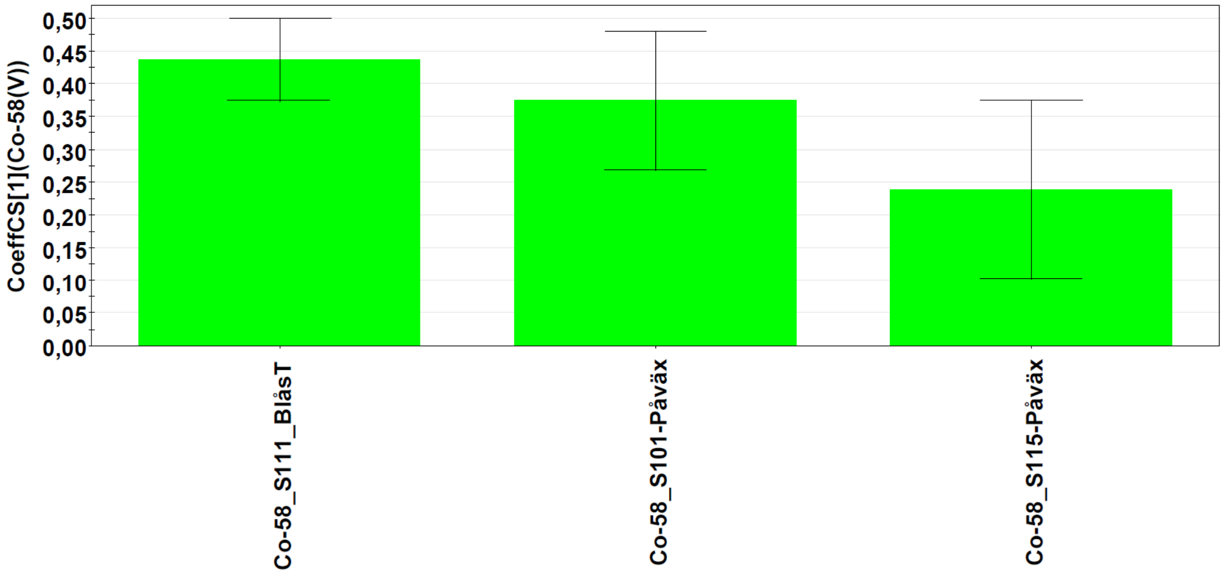

Figure 5. Regression coefficients for one model (M36) for releases of Co-58 to water for different sampling stations and species.

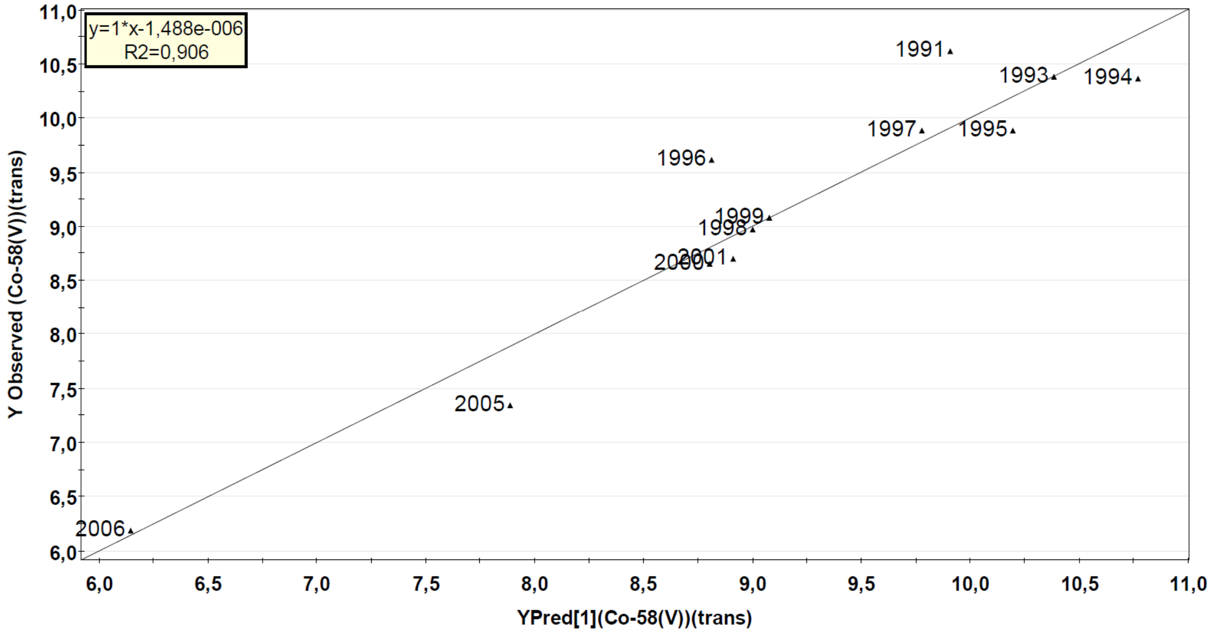

Figure 6. Observed and predicted value for the release of Co-58 to water for one model (M176). Log transformed data.

Results

Quantifiable correlations between measured releases and measured environment concentrations were identified. Several of the investigated environmental samples showed significant correlations with the releases and mathematical models describing the relationship between releases and concentrations in the environment were identified. The correlations are valid for activity values within the interval that has been used to design these models. The models should also be able to point out deviations and indicate when a non-monitored release has occurred.

The summarised result is that it is possible to calculate the releases from the studied facilities from the results of the measurements of a few different species at one or a few of the sampling stations in the surroundings. The results also show that it is possible to calculate the environmental concentration of the studied radionuclides for a number of different species and sampling stations from the releases from the studied facilities.

Results for Oskarshamn

Significant models were identified for five of the six studied nuclides to water, Fe-59 excepted, and for Zn-65 to air. The analysis also shows that for five nuclides to water and Zn-65 to air it is sufficient to measure only one species at one sampling station to get a good idea of the controlled release. That no certain relationships could be established for Fe-59 is probably due to the small number of measurements in the environment for this nuclide. See Table 1.

Table 1. Prediction ability for different models for Oskarshamn.

Nuclide	Number of variables (n)	Prediction ability (%) for model with n variables	One variable (station_species)	Prediction ability (%) for model with one variable
Co-58 water	3	91,7	S12_bladder wrack[1]	76,7
Co-60 water	4	90,0	S12_bladder wrack	89,5
Cs-137 water	18	79,4	S17_bladder wrack	72,0
Mn-54 water	3	70,0	S12_bladder wrack	64,2
Zn-65 water	10	75,8	S1_yellow eel[2]	59,6
Zn-65 air	4	76,3	SB_dryopteris filix-mas	71,4

Results for Forsmark

Significant models were identified for five of the six studied nuclides to water, Fe-59 excepted, and for Co-60 to air. The analysis also shows that for four nuclides to water it is sufficient to measure only one species at one sampling station to get a good idea of the controlled release. See Table 2.

Table 2. Prediction ability for different models for Forsmark.

Nuclide	Number of variables (n)	Prediction ability (%) for model with n variables	One variable (station_species)	Prediction ability (%) for model with one variable
Co-58 water	3	96,7	S101_ongrowth	89,7
Co-60 water	11	85,9	S101_ongrowth	93,6
Cs-137 water	11	37,9	S101_ongrowth	72,9
Mn-54 water	2	64,9	-	-
Zn-65 water	4	90,3	S104_bladder wrack	93,7
Co-60 air	2	86,2	-	-

Results for Studsvik

Significant models were identified for all of the six studied nuclides to water, and for Cs-134 and Cs-137 to air. The analysis also shows that for five nuclides to water and Cs-137 to air it is sufficient to measure only one species at one sampling station to get a good idea of the controlled release. See Table 3.

Table 3. Prediction ability for different models for Studsvik.

Nuclide	Number of variables (n)	Prediction ability (%) for model with n variables	One variable (station_species)	Prediction ability (%) for model with one variable
Co-60 water	3	50,9	S33_bladder wrack	61,7
Cs-134 water	3	33,6	-	-
Cs-137 water	3	85,2	S3B_bladder wrack	84,5
Eu-152 water	3	77,8	S3B_bladder wrack	70,8
Eu-154 water	4	71,8	S7_sediment	73,6
Mn-54 water	4	46,8	S34_bladder wrack	40,2
Cs-134 air	3	64,4	-	-
Cs-137 air	2	48,4	SC_hair moss[3]	90,4

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[1] Fucus
[2] Anguilla
[3] Polytrichum commune

Conclusions

The results from the multivariate analysis enables the identification of the most reliable sample types in the monitoring programs in order to reflect releases from the nuclear facilities. A proper selection of sample types allows the estimation of expected environmental concentrations from the release, in addition to estimation of historic releases based on environmental concentrations.

In general, the more variables used in the model the better the prediction ability. This is the case for all the variables in Oskarshamn, but as can be seen in the Forsmark and Studsvik case, this is not true especially for Cs-137. This deviation can possibly be explained by the fallout from Chernobyl when Forsmark, specifically, and the Swedish east coast, in general, were highly subjected.

The studies also show that specific variables can contribute to the prediction of releases significantly. For the facilities described here it seems that bladder wrack is an important species to include in the environmental control program to be able to predict environmental concentrations form measured releases or vice versa. In Forsmark also the ongrowth samples are good indicators for releases. For Ringhals similar results were obtained; coefficients for samples of bladder wrack showed the least uncertainty [4].

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Application of Rapid and Automated Techniques in Radiochemical Analysis – Inspirations from NKS-B Rapid-Tech Project

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Rapid sample processing techniques are desirable in radionuclide determination for emergency preparedness, environmental monitoring, nuclear decommissioning and waste management to achieve fast analysis, and high sample throughput with low labor intensity and cost. Within the Nordic countries, a few laboratories working with radiochemistry have initiated R&D in developing rapid radiochemical methods using different rapid and effective sample treatment techniques. However, the exploration of rapid techniques is still a fresh area, and very little has been done to share experiences and knowledge on this topic among the Nordic countries. In 2014, within NKS-B Rapid-Tech project [AFT/B(14)7], Nordic scientists gathered together and screened the current analytical methods for common radionuclides (e.g., Sr, actinides). Problems and needs in developing rapid radiochemical methods were identified and applications of distinct rapid sample processing techniques to improve the simplicity and analytical efficiency in radio assays for determination of the most common radioisotopes were assessed. Based on the screening, several consensuses through the screening have been reached:

1) Current application of novel automated techniques in Nordic countries is very limited, many of them have only been exploited for research purpose while most routine analysis are still operated in batch-wised manual fashion.
2) Analytical techniques used for Sr determination vary significantly from lab to lab. Especially for low-level environmental samples, several Nordic labs are still using very traditional methods developed in 1960-70s. These methods are not only problematically slow and labor intensive but also based on the use of harmful chemicals (e.g., fuming nitric acid) wherein laboratory safety issues are worth of concern.
3) There is a need for end users to become more aware of the advantages of improved techniques for radiochemical assays, so that they can become more active in driving the long-term development. Identification of concrete analytical benefits and experience sharing are necessary for selecting purpose-fit novel techniques.

To better improve the application of rapid and automated techniques in radiochemical analysis thus to prompt the development of effective radio-analytical methods, insights and perspectives obtained from the NKS-B Rapid-Tech 2014 project are discussed in this paper in combination of specific analytical protocols (e.g., for 90Sr and Pu isotopes).

S6-O4

Canopy interception and accumulation of Fukushima Dai-ichi derived radiocaesium by forest trees

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Radioactive particles, such as Cs-134 and Cs-137, where released to the atmosphere from the nuclear power plant (NPP) accident in Fukushima Dai-ichi in March 2011. These released radioactive particles were deposited both as dry and wet deposition onto forest trees. Some of the intercepted and post-deposited retained radioactive particles were be taken up directly by the forest vegetation through e.g. leaves and/or bark. The aim of this study was to investigate the canopy interception of radiocaesium and the effects of species characteristics on the canopy interception. Moreover, to analyse the accumulation within different plant parts of forest trees. The investigation was based on 4 different monitoring surveys obtained by the efforts of Fukushima Prefecture. The surveys has been carried out from year 2011 in the radioactivity contaminated forest areas of Fukushima Prefecture in Japan, after the release of radiocaesium (Cs-134 and Cs-137) from the Fukushima Dai-ichi NPP accident in March 2011. Totally approximately 250 measurement points in 7 different forest stands was studied. The different tree species were divided into 2 groups, evergreen species (coniferous) and deciduous species (broad leaves).The average interception of radiocaesium (both Cs-134 and Cs-137) was largest for Oak tree mixtures (282 ± 467 Bq m^{-2}) and lowest for Japanese cedar tree (58 ± 104 Bq m^{-2}). The average f-values of radiocaesium (both Cs-134 and Cs-137) were highest for Beech trees (0.58±0.17). The bark part had in generally the highest activity concentration of radiocaesium whereas the lowest activity concentration was in the wood part (both softwood and heartwood). Moreover, there was found that pre-leaves had a higher activity concentration of radiocaesium then in post leaves. The deciduous trees had in general higher activity concentration in the bark part and in the litter layer then for coniferous trees. This can be explained by that the deposition of radiocaesium occurred before the deciduous trees had started leafing.

S6-O5

Concentrations and inventories of Cs-137 in dated sediments sampled in the Swedish Marine Environmental Monitoring Program

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The Swedish Radiation Safety Authority (SSM) collects, jointly with other Swedish authorities (e.g. Swedish Environmental Protection Agency and Geological Survey of Sweden), samples from the marine environment surrounding Sweden. This activity is a part of the Swedish Marine Environmental Monitoring Program. In a subprogram focusing on metals and organic pollutants in sediment, SSM takes part to monitor radioactivity. Sampling stations in this subproject have been monitored since the 1970s, but monitoring in its present form, comprising 16 stations in open sea which are revisited every 5-6 years, started in 2003. The idea is to study trends in pollutions (including 68 elements and 66 organic compounds) at these stations which are carefully selected to be in deep bottom areas with undisturbed sediments that form excellent archives that can be dated. SSMs role has been to study radioactivity in the sediments and to produce sediment dating results for the other collaborators monitoring metals and organic pollutants. We will present the Cs-137 results from 16 sediment cores collected during the 2008 sampling campaign and compare the results with the 2003 data that are already published in the HELCOM database. The sediments collected during the 2014 sampling campaign have just arrived to the SSM, and hopefully some preliminary results on selected cores will be given at the conference.

S6-O6

Effects of dynamic behaviour of Nordic marine environment to radio ecological assessments (the EFMARE project)

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Within the scope of the EFMARE project, the consequences of severe radioactive releases to the Nordic marine environment were analysed with special attention to the effects of dynamic behaviour, namely kinetic bioaccumulation processes and their modelling and the temporal variability of pollutant dispersal. The hypothetical NPP and a submarine reactor accident in the coastal Nordic marine environment (The Baltic Sea regions; Icelandic and Faroese coastal areas) were under consideration. Our results for estimating the concentration of radionuclides in marine organisms and for dose assessment regarding the implementation of the kinetic bioaccumulation model into the NRPA box model and the DETRA computer code clearly demonstrate that there is a significant quantitative difference between the kinetic modelling approach and the approach based on the assumption of constant concentration ratios. Model results were compared with experimental data on the basis of an improved version of the NRPA box model for the Baltic Sea. This clearly demonstrates that dynamic modelling of bioaccumulation processes can provide a more precise description of the concentration of radionuclides in biota and can be very useful for radio ecological assessment. With a numerical case study the temporal variability of pollutant dispersal in Icelandic waters is demonstrated. The simulations, which contain a pollution source in Denmark Strait northwest of Iceland, are based on flow and turbulence fields provided by the CODE operational ocean model. The results show a spreading directed mainly eastwards over the north Icelandic shelf with the North Icelandic Irminger Current. Another path leads into the southward directed East Greenland Current. The dispersal into both branches shows a high inter-seasonal variability whereas the role of the seasonal signal is, at least in certain areas, only of minor importance. These results emphasize that operational hydrodynamic ocean modelling can provide important additional information for radio ecological assessment.

S6-O7

Really long term radiological assessment of ecosystems

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A major challenge for the assessment of nuclear waste repositories is to make estimates for the long time scales of 1000 to 100 000 of years. The Swedish Nuclear Fuel and Waste Management Co. (SKB) has recently handed in two applications for repositories. One high level repository for spent fuel and one low level waste repository for operational nuclear waste. In both assessments the knowledge of the past history of the sites, the last glaciation period to now, and the deduction from the ongoing processes are combined to produce an illustration of future landscapes. This includes variations in the natural ecosystem as well as potential land use of the surface ecosystems and the potential exposure to humans and the environment. The results from these applications give that humans can be exposed to a variation of several orders of magnitude depending at when and how long releases might occur to the surface. A brief overview of the methods applied will presented, and some examples of results. Finally the balance for a complex “realistic” biosphere assessment versus a “simplistic” assessment will be discussed.

S6-P1

Radioactivity in fertilizers

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Fertilizers are produced from organic materials or minerals deposits. Varying concentrations of radionuclides are found in these raw materials. During industrial processes, radionuclides may concentrate in the product, side product or waste. Therefore, fertilizers in the Finnish market were investigated as a part of the environmental radioactivity surveillance. Radioactivity in fertilizers may transfer from soil to agricultural produce either directly to edible crops or via feed cultivation to livestock and hence cause exposure internally to consumers. Fertilizers may also cause exposure externally to those who handle them professionally. The previous survey into radioactivity in fertilizers had been carried out in 1982–1983. The sample consisted of 45 fertilizers: 32 inorganic, 7 organic, 4 soil amendments and 2 substrates. The samples were the same that Finnish Food Safety Authority – Evira selected for their fertilizer's control analyses. The fertilizers originated from Finland, Russia, Germany, Estonia, Israel, Sweden and Belgium, and were collected in 2012. In all investigated fertilizers, radioactivity content was so small that they do not significantly increase the natural radioactivity content of agricultural soils. The radioactivity content of soil amendments and substrates were in the same range as or smaller than agricultural soils in general. Among those who handle these fertilizers at work, excess effective dose from external exposure does not exceed 1 mSv/year.

S7-O1

Uranium Aerosol Characteristics at a Nuclear Fuel Manufacturing Site – The regulators perspective

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This presentation is based on the research project “Uranium Aerosol Characteristics at a Nuclear Fuel Manufacturing Site - Particle Size, Morphology and Chemical Composition”. This study has been conducted by Linköping University for the Swedish Radiation Safety Authority, SSM. The conclusions and viewpoints presented in the report (2015:18) are those of the author/authors and do not necessarily coincide with those of the SSM. From SSM’s point of view it is interesting to present how this research project will support regulatory supervision at nuclear facilities with airborne radioactive particles present. The authority has a challenge at nuclear facilities to understand and follow up how the uncertainties in the different parameters are handled concerning data collection for calculating the committed effective dose. In operating of nuclear facilities, in the process of controlling the radiation working places, many parameters in the collecting data are crucial for following calculations of committed effective dose. Three such parameters concerning uranium aerosols are the particle size, the morphology and the chemical form. It is in our regulations (SSMFS 2008:51) pointed out basic provisions for the protection of workers and the general public. The operator has to control the amount of intake of radioactivity via respiration, ingestion or through the skin (including wound). For this control the air and surface contamination are monitored in “controlled areas”. Results of calculated committed effective dose are related to the regulating dose limits including equivalent dose to the foetus not getting more than 1 mSv. Results of monitoring people shall also be recorded.

Radiation protection of individuals exposed to ionizing radiation at nuclear facilities is based on regulations (SSMFS 2008:26) where the operator is pointed out to have continuous calibrations and checks of instruments and equipment. Personal contamination checks before leaving controlled area has to be installed. Whole body counting including urine analysis and lung measurements if needed is required. If committed effective dose of 0.25 mSv or more is found, the whole working team shall be measured. Single event of intake which is calculated to give a committed effective dose of 5 mSv or more shall be reported in writing to the Swedish Radiation Safety Authority. Such a report has to declare type of intake and circumstances, committed effective dose and the basis for these calculations.

S7-O2

Uranium Aerosol Characteristics at a Nuclear Fuel Manufacturing Site – Particle Size, Morphology and Chemical Composition

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Characteristics are crucial in order to carry out accurate internal dosimetry calculations following the International Commission on Radiological Protection methodology. Examples of such parameters are Activity Median Aerodynamic Diameter and solubility. Understanding of such parameters requires knowledge of aerosol characteristics such as size distribution, morphology and chemical form. In this pre-study, these parameters have been studied at two process steps (fluidizing bed furnace and burnable absorber grinder) at the Westinghouse Electric Sweden AB nuclear fuel factory in Västerås, Sweden. Aerosols were collected using a cascade impactor and 62nalysed with Scanning Electron Microscopy coupled with Energy Dispersive X-ray analysis. The results show a significant variation in uranium aerosol shape (spherical, near-cylindrical, irregular with sharp edges, conglomerates of small particles, etc.), with particle size distributions to some extent deviating from the expected lognormal distribution, possibly indicating two 'families' of particles. The vast majority of the radioactive aerosols, unsurprisingly, consisted of uranium and oxygen, but at the bed furnace, 1-6% of the uranium aerosols contained fluorine. Other uranium aerosols were shown to be attached to elements such as nitrogen, aluminium, gadolinium and silicon. This is believed to affect the aerosols' physicochemical properties.

Performance of a new NIRP TL-dosimeter – Uncertainty and detection limit estimation

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In July 2012 the Danish Health and Medicines Authority/National Institute of Radiation Protection introduced a new dosimetry system based on thermoluminescent dosimeters (TLD). The system consist of a HarshawTM TL-card with LiF:Mg,Cu,P material and automated ‘8800’ TL-readers from Thermo Fisher Scientific. The TL-card is worn in a designated plastic holder to measure the operational quantities for individual monitoring. The performance of the dosimeters is being tested according to the international standard IEC 62387-1 and irradiations with photon and beta-particle fields are performed according to the standards ISO 4037 and ISO 6980. Results from this test will be presented and uncertainties and detection limit will be presented and discussed in accordance with internal standards.

S7-O4

The start of the decommissioning of the inner parts of the DR3 reactor

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Abstract

The decommissioning of the former DR 3 research reactor at the Risø site has now reached a state, where the first inner parts have been removed. These parts are the top shield plug (TSP) and the top shield ring (TSR), which constituted the top shielding of the reactor tank with the fuel elements. Both the TSP and the TSR were heavy components with very active bottom parts. Therefore careful planning and monitoring from the moment the parts were taken out until they were placed in shielding casks was needed. A characterization of the activity contents in the reactor parts was used to estimate the distribution of activity in these parts. ^{60}Co is the dominant gamma-emitting radionuclide. From the estimated activity contents radiation levels were calculated around the parts both unshielded and when placed in their shielding casks. The parts were handled remotely during the removal until they had been partly shielded. Also the dose rate was monitored remotely. Good agreement was obtained between measured and calculated dose rates. Dose constraints were set for the individual effective dose for the workers participating in the operations. Effective doses to the workers were all well below the dose constraints. Both Danish Decommissioning personal and external workers did participate in the work.

Introduction

The DR3 research reactor was an English Pluto type reactor which was in operation almost continuously from 1960 until 2000 where it was permanently shut down. The reactor had a thermal effect of 10 MW. It was cooled by heavy water and had a graphite reflector. The reactor was primarily used for research in physics (neutron scattering), radionuclide production and silicon doping. 18 experimental tubes gave access to positions close to the core position and neutron beams (thermal and cold neutrons) from two horizontal tubes.

In 2003 it was decided that the reactor was to be decommissioned after a cooling period of 10 years to allow for decay of radioactivity (^{60}Co). The 10 year period has been used to characterize the reactor for the content of radioactivity, different decommissioning options have been evaluated and peripheral equipment has been dismantled. Three years ago the primary cooling system (except for the reactor tank itself and the tubing in the reactor block) was dismantled and in 2014 the decommissioning of the inner parts of the reactor started. For a more general discussion on the health physics issues related to the decommissioning of the DR3 reactor see ref. 1.

Decommissioning approach

After the peripheral systems and the primary cooling system had been dismantled the reactor block (biological concrete shield and the reactor parts on the inner side of the concrete shield), the concrete walls of the room below the reactor block and the reactor containment remained.

Figure 1 shows a schematic cross section of the reactor block. The reactor block can be delineated into the inner parts (within the red lines) and the outer parts (mainly concrete). The concrete room with the primary cooling pumps is not shown on the figure 1.

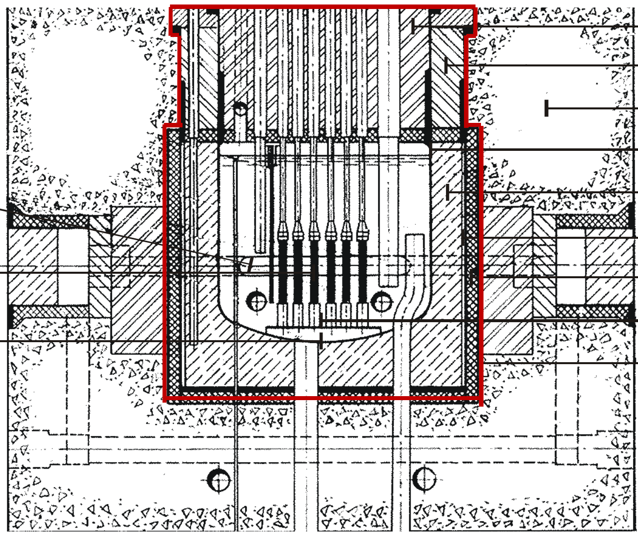

Figure 1. *Schematic cross section of the DR 3 reactor construction. Parts within the red lines are defined as inner parts and the parts outside as outer parts. The reactor aluminium tank contained the fuel and the heavy water moderator. Fuel and moderator were removed just after final shot down. Outside the reactor tank a graphite reflector is placed in a steel tank (inner tank). Outside this steel tank a lead and boral shielding are situated in an outer steel tank. Outermost is the concrete biological shielding. The reactor aluminium tank is closed at the top by a top shield plug (TSP) and likewise the steel tank with the graphite reflector is closed by a top shield ring (TSR).*

The decommissioning is planned to be done according to the following sequence (from top down and from inside out):

Top shield plug (TSP), top part of reactor tank, top shield ring (TSR), lower part of reactor tank including horizontal tubes, graphite moderator, inner steel tank, lead and boral shielding, outer steel tank and the concrete shielding. The outer steel tank containing lead and boral shielding might be removed together with the concrete.

TSP and TSR units and the method of their removal

TSP and TSR can be considered as slabs of different materials (cadmium, iron, lead, shot iron ball concrete) put on top of each other and encased in steel. A priori the parts of the steel casing closest to the reactor core were considered to contain most of the activity.

Table 1 gives some of the characteristics of the TSP and TSR.

Table 1: *Some characteristics of the TSP and TSR.*

	TSP	**TSR**
Geometric shape	Approx. cylinder	Approx. cylindrical shell
Inner diameter	−	206 cm
Outer diameter	197 cm	270 cm
Height	132 cm	118 cm
Mass	~ 22 tons	~ 17 tons
Total estimated activity content	~ 4.5 TBq (primo 2012)	~ 0.4 TBq (primo 2012)

The two units were removed in almost the same way and as follows:

A: vertical wire-lift by hydraulics (strain jack) of approx. 220 cm

B: horizontal displacement of approx. 500 cm on skid tracks by hydraulic pistons

C: vertical lowering onto a bottom shielding

D: assembly of the shielding that had been built specifically for the unit

The vertical and horizontal movements of the units were done remotely and were performed by the external company Mammoet.

When the TSP was removed the now open reactor cave was covered by a moveable top shield (MTS) which ran on tracks.

Preparations before removal

Limitation on doses and radiation levels

A maximum acceptable individual effective dose for the removal of one unit was set to 1 mSv. This is a fairly large fraction of the 5 mSv/y dose constraint on individual dose, which is used for the DR3-internal-parts-project. This 1 mSv level was chosen because considerable shielding of the sources had been made, further shielding was difficult and expensive to apply, the tasks were all considered important and non-frequent. The 1 mSv level was set for both Danish Decommissioning personnel and the external workers and all workers were informed of this requirement in a health physics introduction. The nuclear regulatory authorities did agree with this maximum effective dose beforehand but requested an immediate notification in case the dose level was exceeded.

The two shieldings for the TSP and TSR were designed to reduce the dose rate on the surface of the shieldings to 2 mSv/h.

Determination of activity content

The activity distribution in the reactor inner parts was estimated based on the results of a project in which the reactor was characterized (ref. 2). As the characterization was made by measurements on samples the estimated activity contents in some parts did have large uncertainties.

It was considered necessary to make dose rate measurements to obtain better knowledge of the activity contents on the side of the TSP. Therefore a preliminary lift in which the TSP was lifted approx. 95 cm was made a month before the removal.

<u>Expected radiation levels</u>

Based on the estimated activity content in the inner parts and surrounding concrete shielding, the expected radiation levels were calculated in order to design shieldings and to calculate radiation doses to the workers. The Monte Carlo code MCNP was used to calculate the radiation levels. To assist with the dismantling of the reactor inner parts, a catalogue of radiation levels has been made (ref. 3).

Figure 2 and figure 3 show some examples the calculated dose rate levels.

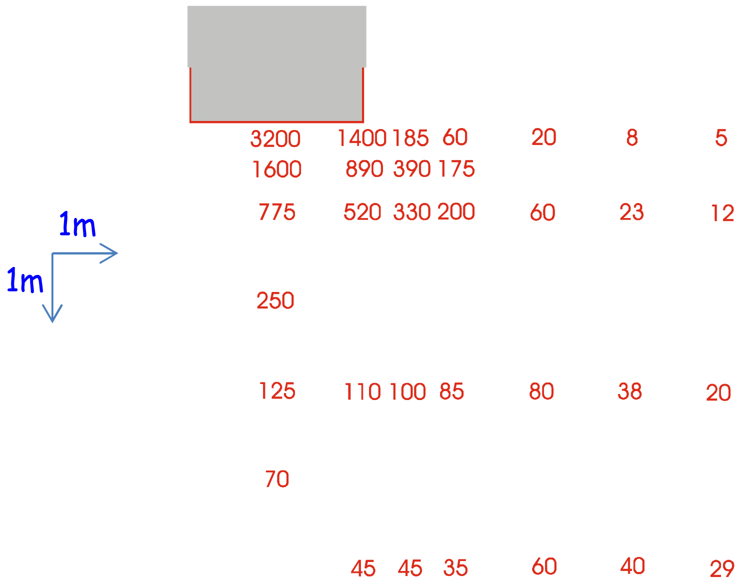

Figure 2. *Modelled dose rates in mSv/h below the TSP unit (TSP situated in mid-air).*

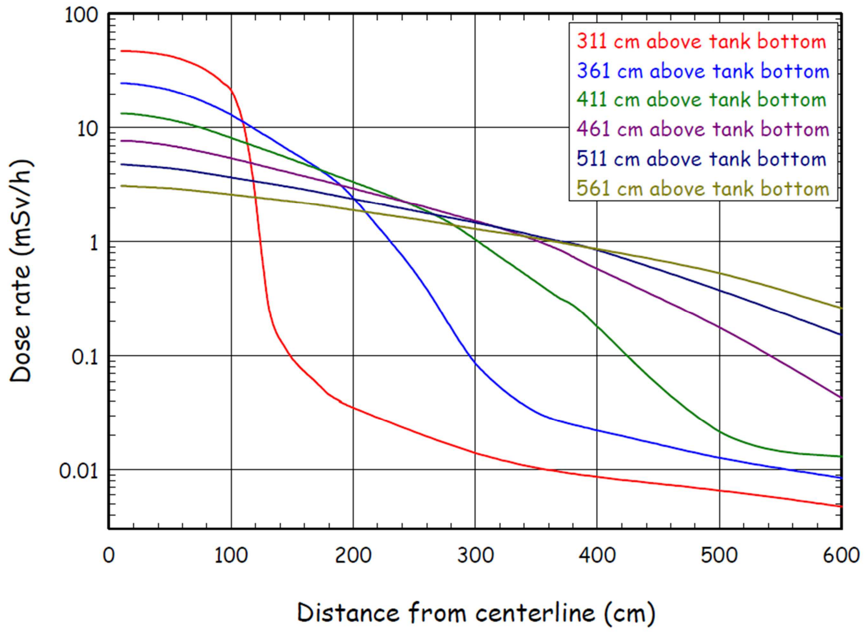

Figure 3. *Modelled dose rate from the inner parts of the reactor as a function of horizontal distance from the reactor centreline for different vertical distances above the bottom of the reactor tank.*

Relation between the dose rate at un-collimated detectors placed at different positions relative to the TSP and TSR and the radioactivity content in the bottom of the units were calculated using the point kernel method and the MCNP tool. Also the relations between activity content in the side of TSP and dose rates at collimated detectors at different positions relative to the TSP were calculated in the same way. These relations were used in comparison of measurements made during the removal of the units.

Shielding design

The shielding packages for TSP and TSR were designed on the basis of the activity contents and the requested dose rate limitation. A shielding wall of concrete blocks surrounding the reactor top area was designed.

Work planning

The work was divided into sub tasks. For each sub task the radiation protection measures were optimized. The calculated radiation levels were used to do this.

Expected effective doses

Based on the predefined work procedures, on positions relative to the radiation sources, on work time consumption in the radiation fields and on the radiation levels, the effective doses were calculated.

Setting no-go criteria

A no-go criterion was set up for the removal of the TSP and TSR.

The estimated maximum individual effective dose in the sub tasks for the TSP removal was in the range 0.20-0.25 mSv i.e. ~1/4 of the maximum acceptable dose. The estimated activity in the TSP side casing was the main contributor to the effective dose hence up to 4 times larger activity than estimated in the side casing could be accepted. If a similar argument was to be used for the TSP bottom a somewhat higher activity than the estimated activity multiplied by 4 could be accepted. However, the bottom of the TSP was the least shielded part during the lift and therefore a factor of 4 was also applied for this activity. The limits for acceptable activity were translated into acceptable dose rates in detectors which could measure in planned stops during the removal. If one or more of the measurements at the side were above the no-go limit the TSP would still be lifted to the highest position before it would be lowered into its original position in the reactor. This was to be made to, should the activity in the TSP side be too large, so that an indication of the activity of the bottom of the TSP could be achieved.

In a similar way a criterion was made for measurements the bottom of the TSR during removal.

Results

External effective doses

The resulting effective doses to the involved workers were registered by digital dose meters and TL dose meters and extremity doses (hands) were registered by TL dose meters. In this paper only the digital readings are reported. Very low doses were received while performing preparatory work prior to the actual lift. Table 2 and table 3 show the effective doses received during the removal of the TSP and TSR respectively.

Table 2: *Effective doses in μSv received by Mammoet (MM) and Danish Decommissioning (DD) personnel when the TSP was removed.*

8 May (removal)	**DD**	**MM**	**DD and MM**
Mean effective dose (for those >0)	15	14	14
Maximum effective dose	31	36	36
Collective effective dose	177	111	288
9 May			
Mean effective dose (for those >0)	8	9	8
Maximum effective dose	26	19	26
Collective effective dose	109	37	146
Total collective effective dose	286	148	434

Table 3: *Effective doses in μSv received by Mammoet (MM) and Danish Decommissioning (DD) personnel when the TSR was removed.*

13 October	**DD**	**MM**	**DD and MM**
Mean effective dose (for those >0)	6	5	5
Maximum effective dose	12	8	12
Collective effective dose	13	19	32
14 October (removal)			
Mean effective dose (for those >0)	12	16	13
Maximum effective dose	54	36	54
Collective effective dose	202	98	300
Total collective effective dose	215	117	332

Dose rates in areas surrounding the DR3 containment were measured during the removal of the TSP and TSR. At the maximum lift height, where the dose rates in the surrounding was largest, the dose rates did not exceed 1 μSv/h (TSP) and 0.3 μSv/h (TSR).

Activity content in the TSP steel bottom plate

Measurements with two un-collimated GM-detectors were made when the TSP was lifted to the highest position, i.e. 217.7 cm. The detectors were positioned 80.7 cm vertically under the bottom and 94.5 cm (radial distance) from the side of the TSP. The detector had an approximately 45 degrees angular separation. The detectors had full view of the TSP bottom. The measured dose rates were 121 mSv/h and 135 mSv/h with a relative 20% uncertainty. If the relation between dose rate and activity which was calculated for this geometry is assumed to be without uncertainty the activity of the bottom of the TSP was 2.89 TBq ± 0.41 TBq (8 May 2014). This is in full agreement with the estimated primo 2012 activity, which corrected for decay results in 3 TBq medio 2014.

Activity content in the TSP steel side casing

Measurements with two collimated detectors GM-detectors pointing towards the lower side of the TSR were made, when the TSR was lifted. The detectors were placed at a radial distance of 130 cm from lower part of the TSP casing. The lead and steel collimation limited both the detectors view to a part of the side casing which was 10 cm in height and 10 degrees wide, and which was just above the reactor pit. The detectors were separated by approximately 40 degrees. Measurements were made at the lifting heights of 110 cm and 125 cm. These heights were chosen as a compromise between having the bottom of the TSP in the pit and measure the side casing near the bottom of the TSP. The lifting heights correspond to parts of the TSP sides which have their lower side 22.1 cm and 7.1 cm, respectively, above the bottom of the TSP. The activity concentrations calculated from the measured dose rates were in all four places a factor of 10 lower than the estimated (and decay corrected) activity concentrations.

Activity content in the TSR steel bottom plate and inner side steel casing

The activity content in the TSR bottom was determined by measuring the dose rate with two un-collimated GM-detectors situated 54 cm below the bottom on the main axis. In this position the detectors had a good view of the bottom and inner side casing of TSR (the two parts which contributed the most). The dose rate to activity ratio used is estimated to have a relative uncertainty of 10% due to the inherited uncertainty in the activity distribution.

Prior to the lift, the dose rate, as a function of position below the bottom had been calculated with the MCNP for a total activity content of 0.41 TBq which was the expected activity content primo 2012. For the purpose of the calculations most of the activity was considered to be in the bottom (58%) and in the inner side casing (40%). For the actual geometry the relation between dose rate and activity was 78 mSv/h per 0.41 TBq.

In the actual lift the two detectors both measured 53 mSv/h. The relative uncertainty on the measured dose rates was 20%. The dose rates measured correspond to an TSR bottom and inner side casing activity of 0.28 TBq ± 0.15 TBq (14 October 2014). Decay correction to October 2014 from the primo 2012 estimated activity (0.41 TBq) results in 0.29 TBq, thus a full agreement was between estimated and measured activity.

Contamination

During the removal air was continuously monitored for tritium and particulate activity by air monitors. No reading above background was seen. Surface smear samples were taken along the transport route for the TSP and TSR but no contamination was found.

Internal doses

As contamination of surfaces and air did not occur, internal doses were not received.

Effectivity of TSR and TSP shieldings

Measurement on the surface of the shielding shows that the maximum surface dose rate (~1.1 mSv/h (TSP) and ~2 mSv/h (TSR)) does not exceed 2 mSv/h.

Photos

Below are some photos (figure 4-9) from the removal of the two units.

Figure 4. *Ready to remove TSP. Side part of TSP shielding is placed over the top of TSP in the center of the picture .The hoist system is ready to be attached. To the right are the bottom part and the inner lower side shielding for the TSP ready to be lowered down. To the left under the support for the hoist system is the moveable top shield (MTS) ready to be moved over the hole which was left open when the TSP was removed. A shielding wall of concrete stones surrounds the reactor top.*

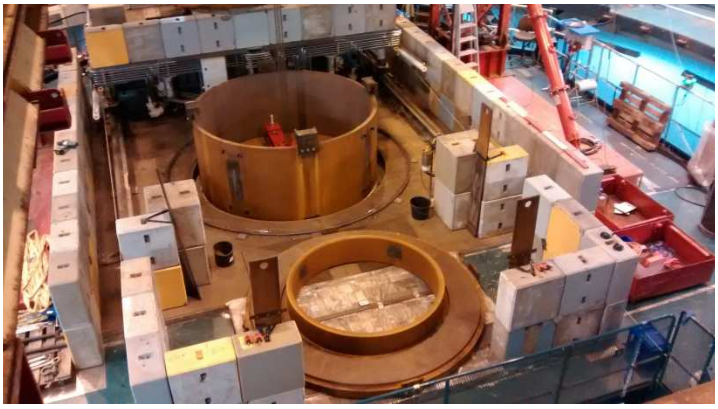

Figure 5. *Ready to remove TSR. Side part of TSR shielding is placed over the top of TSR. The hoist system is attached via a shielding slab to the top of the TSR. In front are the bottom part and the inner lower side shielding for the TSR ready to be lowered down. In the back the MTS is ready to be moved over the hole which was left open when the TSR was removed. A shielding wall of concrete stones surrounds the reactor top.*

Figure 6*. Remote removal of the TSR is made from the control room at the entrance level of the reactor hall.*

Figure 7*. TSP inside its shielding at the reactor top.*

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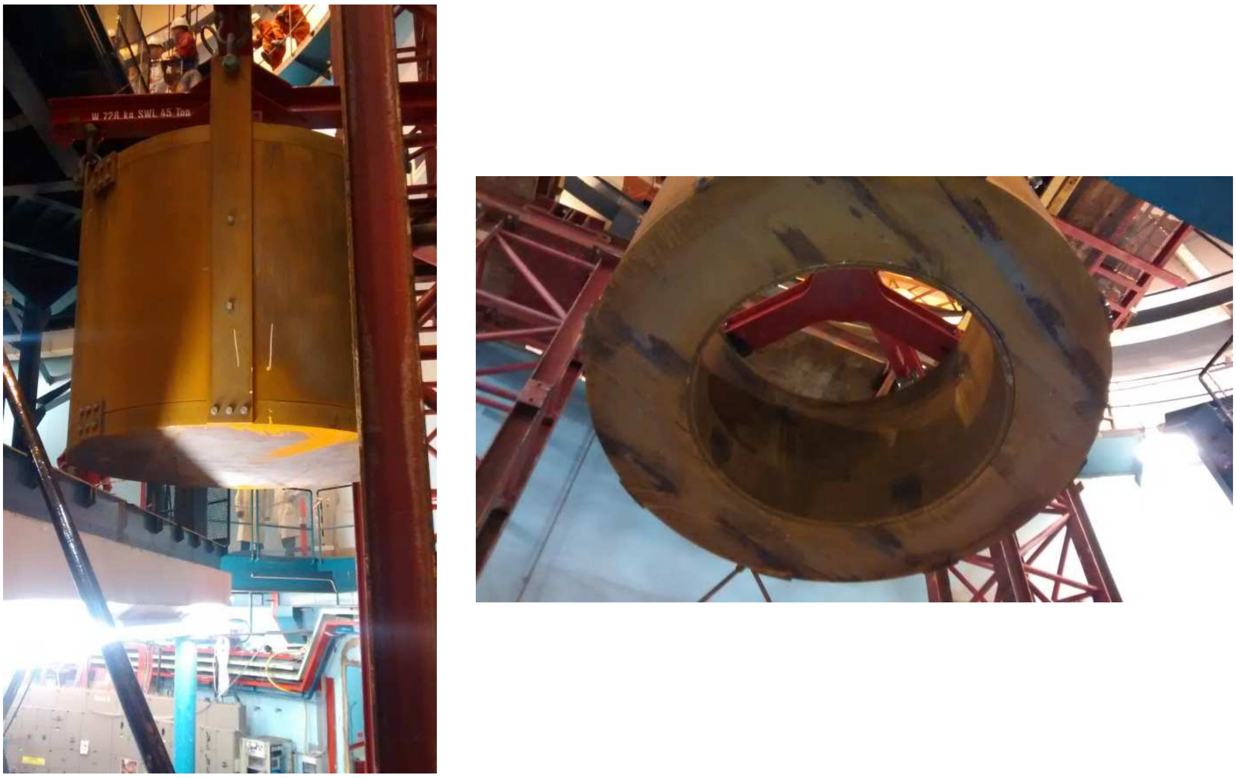

Figure 8. *TSP (left) and TSR (right) inside their shieldings are being lowered down from the reactor top.*

Figure 9. *TSP inside its shielding inside another shielding is transported to the local waste storage.*

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S7-O5

Radioactive Waste Management in Denmark

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Danish Decommissioning (DD) receives and manages all the radioactive waste produced in Denmark. The waste originates from a variety of users and activities including the industry, the health sector, research institutions, schools, and the decommissioning of the nuclear facilities at the Risø area. The waste management is subject to a number of requirements concerning the protection of humans and environment. These requirements concern all stages of the waste management and –treatment from production of the waste over short- or long term storage till the final disposal. During the early stages of waste production and –storage humans and environment must be protected from ingestion of and exposure to direct radiation from radionuclides. At the final stage of the waste management, the disposal, the effort will concentrate on isolating the waste from humans and environment, and to inhibit, reduce and delay the migration of radionuclides at any times from the waste to the biosphere. In order to consider and prepare for all stages of the waste life cycle a number of requirements have been set up for the waste management. The requirements include:

- Volume reduction of the waste;
- Sorting of the waste according to material and origin
- Waste characterization including weight, size, material composition, and identification of radionuclides involved
- Maintaining data about the waste for future generations.

The detailed description of the waste is important as waste properties can be deciding factors when selecting a final design for a long term solution for the waste. We present the Danish Waste management which is based on the requirements above. Furthermore we present the possibilities for a long term solution for the Danish radioactive waste that have been decided politically, and we outline the considerations and models that will be a part of a safety case for a long term solution for the Danish Radioactive waste.

S7-P1

Establishing a method for a more accessible and reliable verification of medical radiation shielding

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Introduction

In the planning of any medical X-ray facility one of the main priorities is to ensure that persons in the vicinity of the x-ray facility are not exposed to levels of radiations that exceed regulatory exposure limits[1]. Verification in the form of measurements and calculations of the radiation shielding are mandatory when it is not entirely unlikely that the exposure limits could be exceed, due to increased workload, new equipment or renovation [2].

The aim of this project is to evaluate current methods in use for assessment of medical radiation shielding and to establish the most practical and reliable method for verification of radiation shielding.

Materials and Methods

The methods included in this project involved; Transmission measurements using x-ray, and determination of lead thicknesses and equivalent thicknesses using two radionuclides. All measurements were performed on 2 lead sheets of know thicknesses and on 3 areas of interest in an x-ray lab**.**

Transmission based assessment

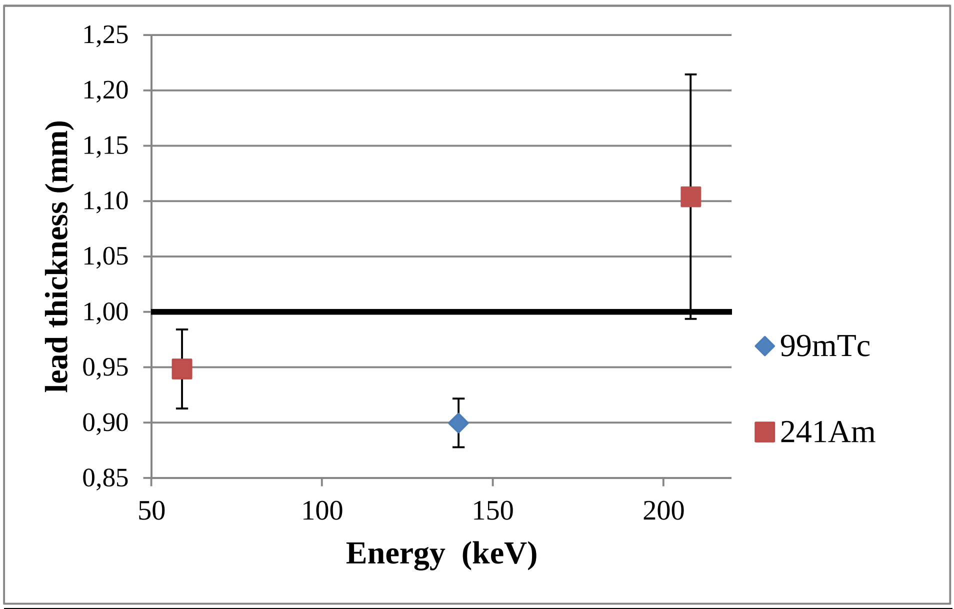
Transmission measurements were performed using mobile x-ray equipment (Siemens Mobilett II) and a dose meter (RTI Piranha).

Methods determining lead equivalencies

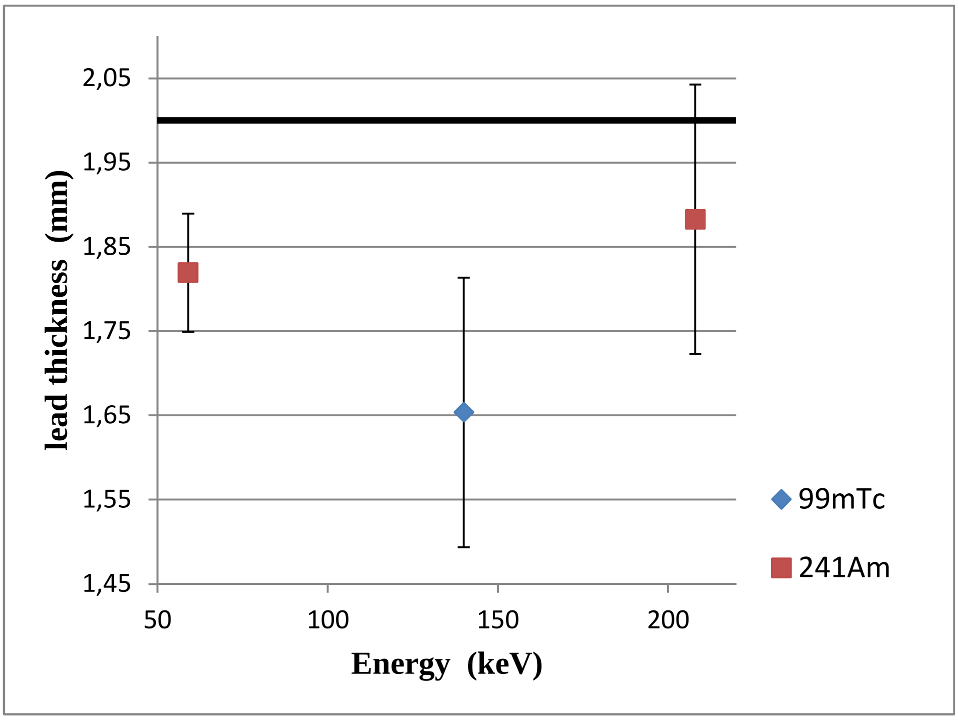
Here the area of interest was exposed to a radioactive point source and the transmitted beam was measured on the other side with a portable HPGe gamma-ray spectrometer (Ortec Detective). Measurements were performed using two different radionuclides, 50 Mbq ^{99m}Tc (141 keV) and 370 MBq ^{241}Am with several gamma energy peaks e.g. 59 keV (36%) and 208 keV (0.0008%).

Results

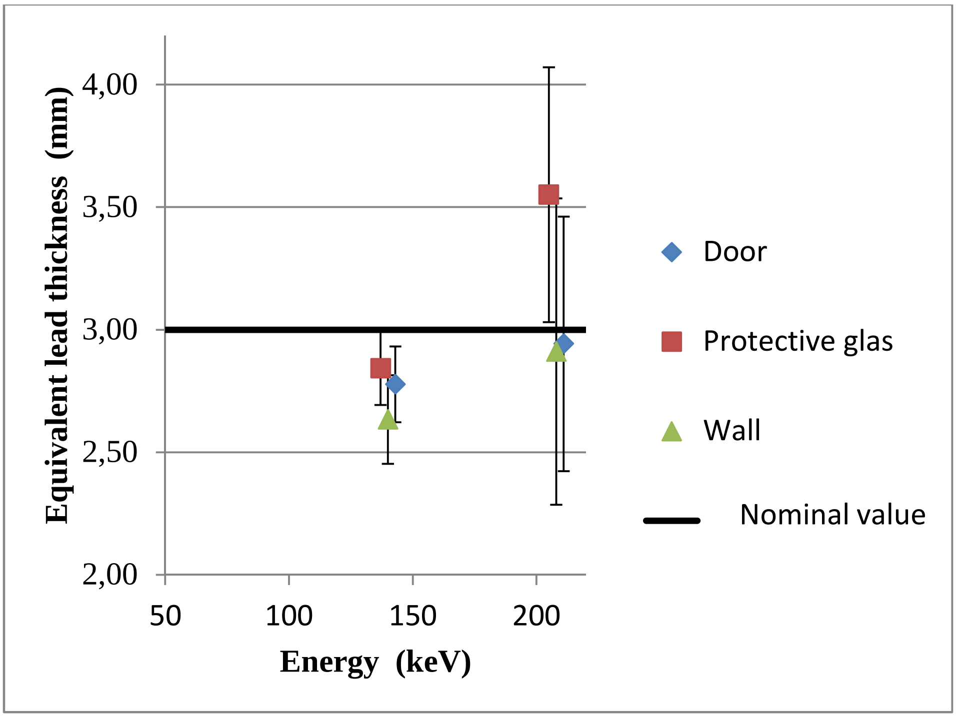
<u>Determination of lead thickness and equivalency</u>



Figur 1: Lead thickness of 1 mm sheet measured using γ-energies 59, 141, 208 keV



Figur 2: Lead thickness of 2 mm sheet measured using γ-energies 59, 141, 208 keV



Figur 3: Equivalent lead thickness of shielding areas (3mm Pb in 110 kV) in an x-ray lab measured using γ-energies 141, 208 keV

<u>Transmission results using x-rays beams</u>

Table 1: % Transmission of three radiation qualities for two lead sheet thicknesses and shielding areas in x-ray lab.

	59 kV (HVL: 2,42 mm Al)	**78 kV (HVL: 3,34 mm Al)**	**131 kV (HVL: 5,43 mm Al)**
1,04 mm Pb	0,014±0,001	0,126±0,005	0,674±0,008
2,00 mm Pb	0,000±0,000	0,023±0,001	0,058±0,002
Wall (3mm Pb)	-	-	0,000±0,000
Shielding glas (3mm Pb)	-	-	0,000±0,000
Door (3mm Pb)	-	-	0,000±0,000

Conclusion

The lead equivalencies measured using energies of 141 keV and 208 keV are in the range of the stated 3 mm Pb. However due to the low yield of the 208 keV rays the uncertainties are high and the 59 keV rays are insufficient for shieldings exceeding 2 mm, making the Am-method not entirely reliable as an assessment method. To be able to use ^{241}Am energies, high potent sources and longer acquisition times are required, which is not feasible or practical.

The use of ^{99m}Tc provides a fast, inexpensive, precise and accessible evaluation method. The downfalls being; a energy higher than diagnostic x-ray beam energies and a short half-life of

6,04 h requiring decay correction. For quantitative assessments the Tc-method is to recommend, especially if the properties of the shielding material is considered.

Transmission measurements using mobile x-ray equipment are a practical solution to verify stated lead equivalent thickness, due to the accessibility of mobile x-ray equipment. However at shielding areas were the lead equivalency is close to or higher than 2 mm lead, the weaker dose rates give no quantitative signal. Making this method mostly suitable for confirmations and not for more accurate characterization of the shielding.

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S8-O1

INVITED: Developments and justification of applications using ionizing radiation in the medical field

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In recent years, those professionals of the radiation protection community who work within the field of medical exposures have embraced and promoted the importance of appropriate justification and optimisation. The focus however has been on individual exposures and the applications of these principles to classes or types of exposure has been largely forgotten. Such exposures can be considered as being at Level II of the ICRP justification hierarchy.

The publication of the Basic Safety Standards Directive 2013/59/Euratom is a timely reminder that Member States need to ensure that new classes or types of exposure must be justified before being adopted. In addition, justification of existing classes or types of exposure must be reviewed in the light of new and important evidence regarding their use and efficacy.

A new requirement of the Directive requires occupational and public exposure must be taken into account when considering medical exposures.

The rate of development in diagnostic and therapeutic medicine is rapid and this applies equally to those areas where ionising radiation is used. This presents specific challenges to those organisations tasked with undertaking the required justification processes.

This presentation considers different approaches to Level II justification, including two examples in the clinical field to illustrate the problems faced by Member States seeking to comply with Directive requirements.

S8-O2

Radiation safety aspects of the Danish Center for Proton Therapy

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The Danish Center for Proton Therapy will be located at Aarhus University Hospital in Skejby 5 km outside Aarhus. The facility will have 3 treatment rooms with gantries and a research room with a fixed beam line. The present status is that Varian Medical Systems has been selected as the supplier of the proton therapy equipment and that the building construction soon will begin. Patient treatment will start in 2018.Secondary neutrons produced by the loss of protons is the main prompt radiation hazard of the facility during operation of the proton therapy equipment, whereas the contribution from other types of radiation normally can be neglected owing to shorter attenuation lengths in matter. Besides the prompt radiation from the therapy equipment, proton and neutron interactions also produce radionuclides in the proton therapy equipment and in surrounding building structures. The facility will be shielded by concrete because concrete has good neutron attenuation properties, is relatively cheap, and has excellent structural properties. The method for determination of the thickness of the concrete shielding will be presented. Long-lived radionuclides in the concrete shielding could be a significant expense for the decommissioning of the building at its end-of-life. Accordingly, the amount of radionuclides in concrete has been estimated and the use of other materials with more favorable activation properties has been investigated. The secondary neutrons also produce radionuclides in air, in the cooling water, and in the groundwater. The radiation safety aspects of these radionuclides will also be presented.

S8-O3

New Danish research laboratory for medical dosimetry

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Abstract

In 2014, the Center for Nuclear Technologies at the Technical University of Denmark (DTU Nutech) inaugurated the “Laboratory for Fundamental Medical Dosimetry” (LFMD) at Risø Campus near Roskilde. The laboratory has been made possible through a large donation from the John and Birthe Meyer Foundation. The objective of this presentation is to outline the design of the new laboratory, and to highlight specific work related to the characterization of radiation protection instruments for measurements in pulsed beams.

Introduction

Cancer patients are increasingly treated with highly conformal, individualized radiotherapy delivered by megavoltage x-ray beams from medical linear accelerators. These advances have first and foremost been made possible by improved imaging procedures and other advances in treatment delivery technology. However, all types of radiotherapy rest on the availability of accurate dosimetry that can establish traceability to the gray in the International System of units (SI), and although radiotherapy dosimetry may be a well-established field for conventional megavoltage radiotherapy, new treatments emerge at a steady rate and these techniques require evaluation and refinement of existing calibration and measurement procedures or development of new ones. Prominent examples of current challenges are the development of accurate dosimetry for therapy in small x-ray beams, in scanned proton beams, and in strong magnetic fields of linear accelerators with magnetic-resonance (MR) imaging capabilities. The overall aim of the new DTU medical dosimetry laboratory is to contribute to this development such that hospitals can make the best and newest types of radiotherapy available to patients.

Research at the laboratory focuses on metrology for the measurement of the absorbed dose to water in MV accelerator beams, however, the laboratory also participates in other projects including, for example, dosimetry for proton therapy and brachytherapy. The work comprises both (i) clinically oriented projects carried out in close collaboration with Danish hospitals, (ii) detector oriented projects based on in-house developments and specific DTU technology (mainly solid-state dosimetry based on luminescence or electron-spin resonance signals), and (iii) metrology projects on new standards and calibration procedures. In addition, radiobiological research is carried out by the Hevesy laboratory, which is also part of DTU Nutech.

Currently much work at the laboratory is still devoted to commissioning of radiation sources and to development of instrumentation and measurement procedures that can be accredited by DANAK, the national accreditation body in Denmark. The aim is to include the medical dosimetry calibration and measurement capabilities (CMC’s) in the existing quality management system of DTU Nutech’s High Dose Reference Laboratory (HDRL). This could lead to an expansion of DTU’s status as Designated Institute within the field of ionizing radiation to also include CMC’s for medical dosimetry that can be declared in the BIPM key comparison database.

The potential outcomes from the laboratory can be classified into four groups: (i) basic research and developments which will be published in the scientific literature, (ii) accredited measurement services that will be made available for hospitals, (iii) training and university teaching including production of MSc and PhD graduates that have specialized in dosimetry during their thesis work conducted at the laboratory, and (iv) innovative instruments and methods that can be commercialized by external companies.

An outline of the laboratory is given in Fig. 1. It contains one large irradiation room with a cobalt-60 irradiator and a medical linear accelerator, a control room, and three rooms dedicated to chemical, optical and electronic developments. The irradiation and control rooms are air-conditioned as to maintain a stable thermal environment and to minimize influence of changes in humidity. The maximum air flow is 21 000 m^3/h. The laboratory is equipped with an advanced instrumentation for dosimetry, and measurement results can be recorded remotely from the control room using in-house developed data acquisition software [1]. The software has a scripting language that facilitates, for example, automated calibration procedures where results from several instruments can be recorded in a standardized way.

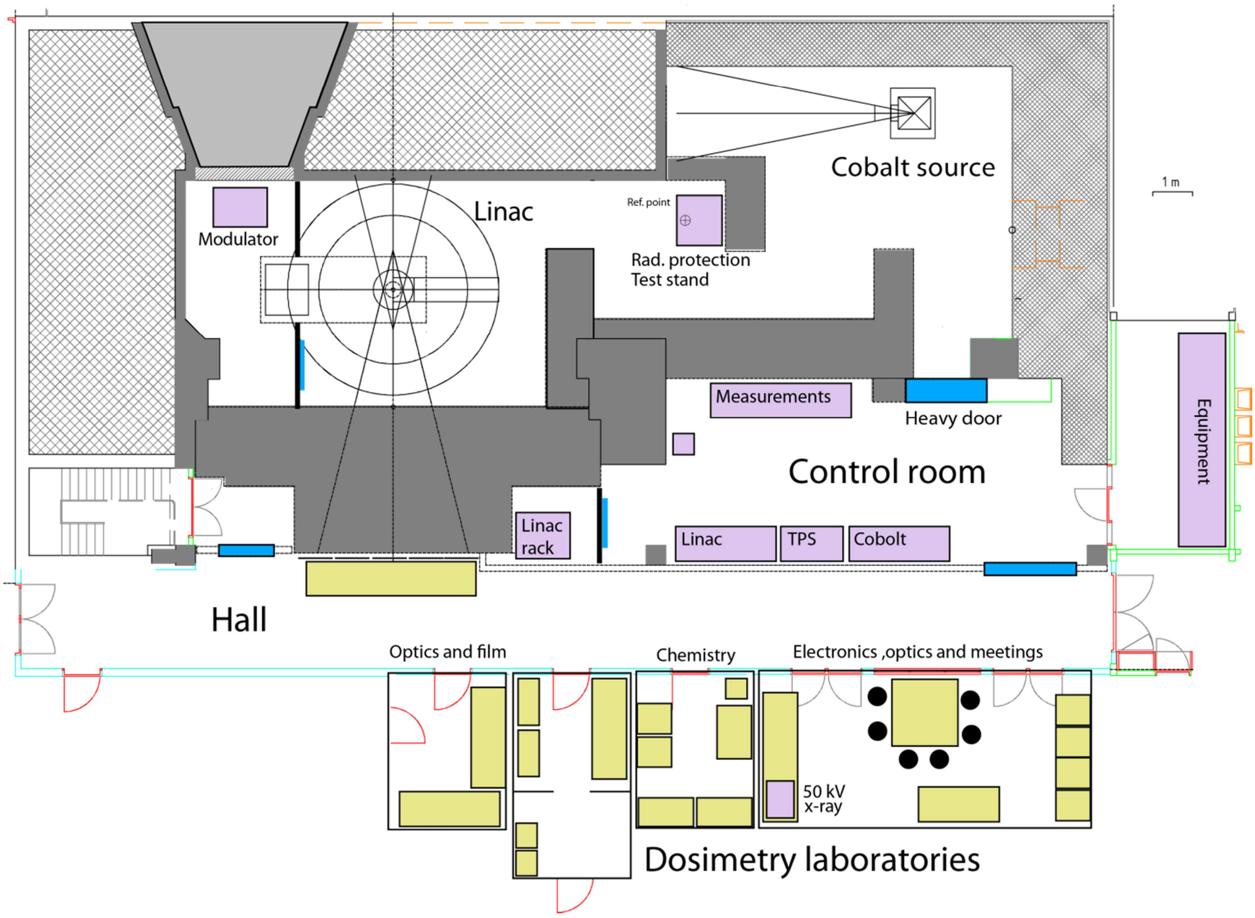
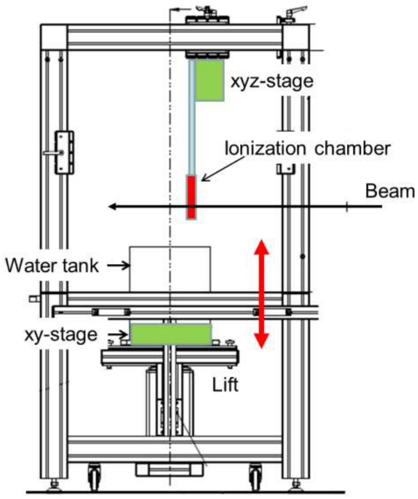


Figure 1 Outline of the Laboratory for Fundamental Medical Dosimetry (LFMD).

Cobalt irradiator

Fig. 2 and 3 show details of the Terabalt-100 cobalt irradiator system from UJP Praha, Czech Republic. The irradiator contains a sealed GK60T03 cobalt-60 source with an activity of 455 GBq / 364 GBq eff. (2012). The dose rate at 100 cm distance is about 1.4 Gy/min. The beam is horizontal, and the collimator can shape the field from 5 x 5 cm^2 to 40 x 40 cm^2. The irradiator is fully computer controlled, and it accepts commands from DTU's data acquisition software.

Calibrations of ionization chambers can be performed in several ways using different water tanks and devices for distance measurements and alignment. In the set-up in Fig. 2 and 3, the



ionization chamber is fixed to the rig using a motorized xyz-stage which allows for micro-meter precision manipulation over a distance of a few cm along all three principal axes. The water tank can be moved vertically using a heavy-duty motorized lift (0.02 mm precision).

Figure 2 Details of the water tank rig

Figure 3 Cobalt-60 irradiator from UJP Praha, Czech Republic.

Accelerator

The medical linear accelerator is a Truebeam (v. 2.0) from Varian Medical Systems. Key elements in the design are a gridded electron gun, a klystron power source, a standing-wave accelerating wave-guide incorporating an energy switch, and a 270° bending magnet. At full power, the accelerator consumes about 50 kW, and cooling is required as not to disturb the thermal environment of the irradiation room. The accelerator can deliver seven different bremsstrahlung radiotherapy beams in the range from 4 to 18 MV, and seven different electron beams in the range from 6 to 20 MeV. The maximum field size is 40 x 40 cm^2, and the maximum dose rate is 6 Gy/min for most beam qualities, however, the flattening-filter free (FFF) 10 MV beam can deliver up to 24 Gy/min. The accelerator is equipped kV and MV imaging systems. The accelerator is dedicated to non-clinical research, and no patients can be treated.

Radiation protection dosimetry in pulsed beams

Electronic Personal Dosimeters (EPD's) and survey meters can be troublesome to use in pulsed beams [2]. The problem is that the instantaneous dose rate during the short duration of a pulse may be a 1000 times or more higher than the dose rate averaged over a second. Instruments that cannot handle such high dose rates typically give a substantial under response. We have, for example, seen a general purpose survey meter that displayed a 25 μSv dose in response to a dose of 2 Gy delivered by a pulsed beam with a 5 μs pulse width. The instrument did not give any indication that something was wrong.

To investigate the suitability of instruments for measurements in pulsed beams, we therefore designed a simple test stand where radiation protection dosimetry equipment could be tested against reference ionization chambers in the scattered radiation from the linear accelerator and the cobalt source. Fig. 4 shows the set-up used for the measurements discussed below.

We generated the pulsed scattered radiation by irradiating a thorax phantom (002LFC from CIRS) with the linear accelerator gantry angle at 0° and using a 10 x 10 cm^2 field. A thimble chamber (Model 30013 from PTW) was placed in the center of the phantom which in turn was at the iso-center of the linear accelerator. The thimble chamber was used for monitoring of changes in output from beam to beam with the same energy. The results are referred to as phantom dose in what follows.

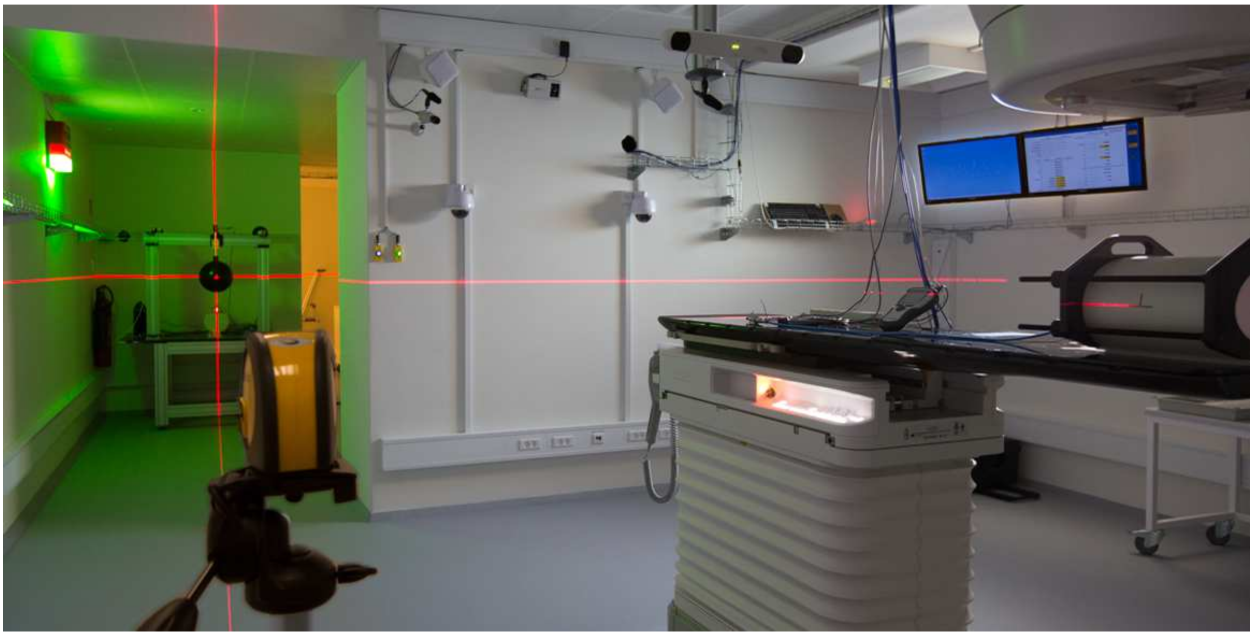


Figure 4 The 10 L spherical ionization chamber in the left part of the photo was at the reference point and the thorax phantom was on the couch (right part of photo). The yellow alignment laser in the foreground is used to verify that instruments at the reference point were at the same height as the accelerator iso-center.

To characterize the time-structure of the accelerator beams, we used an in-house developed instrument (ME40 [3]) for recording the number of gun pulses for each beam and the average time between pulses. The accelerator pulse width was measured using an oscilloscope.

The cobalt source was also used to generate a scattered field of continuous radiation. A 20 x 20 cm^2 beam was used to irradiate the water tank shown in Fig. 1 and 2.

Two spherical ionization chambers of 1 L and 10 L volumes (PTW model 32002 and 32003, respectively) were used to establish the dose rate at the reference position indicated in Fig.1 (about 7.7 m from the accelerator iso-center). The chambers were calibrated for air-kerma rate (free in air), K_a, in cobalt-60 beams, and the measured air-kerma rates were converted to ambient dose equivalent H*(10) using a coefficient of 1.2 Sv/Gy in accordance with the computations by Frota et al. [4] for 6-24 MV beams transmitted through 1 m to 2.5 m of concrete. For simplicity, the H*(10)/ K_a- conversion factor for the scattered radiation from cobalt-60 was also set to 1.2 Sv/Gy [5].

The standard uncertainty of the air-kerma calibrations for the spherical chambers was1.3 %, the energy response of the chambers had a relative uncertainty of about 2 % (k=1), and the conversion factor was estimated to have an uncertainty of 10% (k=1). The combined relative standard uncertainty for the spherical chamber measurements therefore was about 10.3%. The influence of variations in set-up from chamber to chamber or from session to session was found to be negligible.

Table 1 shows the results for tests of a Ludlum 9DP* survey meter designed for ambient dose equivalent measurements and calibrated by the factory. This instrument contains a 230 cm^3 pressurized ionization chamber and it has build-in data logging capabilities. During the measurements reported here, the instrument was set to record the dose rate every 5 s. The Ludlum specifications states that the 9PD*has an accuracy of ±10%, and we take this to mean that the results have a relative standard uncertainty of 10%.

Table 1 Measurement results and standard uncertainties for the two spherical ionization chambers (1 L and 10 L) and the Ludlum 9DP survey meter in scattered beams from the medical accelerator and the cobalt source at the reference position shown in Fig. 1 and 4. The phantom dose was measured with a thimble chamber.*

Beam	Phantom dose Gy	Irradiation time s	Pulse width μs	Pulse-to-pulse ms	No. of pulses	H*(10) 1 L μSv/h	H*(10) 10 L μSv/h	H*(10) 9DP* μSv/h
4 MV	4.0	120	3.7	2.5	48.1×10^3	260 ± 30	260 ± 30	240 ± 30
6 MV	8.5	120	3.7	3.3	36.0×10^3	480 ± 50	500 ± 60	440 ± 40
10 MV	9.4	120	3.5	3.3	36.0×10^3	380 ± 40	400 ± 40	330 ± 30
15 MV	9.8	120	3.2	6.7	18.0×10^3	910± 100	930 ± 100	770 ± 80
18 MV	10.2	120	2.9	6.7	18.0×10^3	1600 ± 180	1700 ± 180	1500 ± 150
6 MV, FFF	8.0	150	4.4	4.9	30.9×10^3	240 ± 30	250 ± 30	220 ± 20
10 MV, FFF	9.0	150	4.1	8.3	18.0×10^3	180 ± 20	190 ± 20	170 ± 20
Cobalt-60	-	-	-	-	-	67 ± 7	67 ± 7	62 ± 6

FFF = Flattening Filter Free

The measurements with the spherical chambers were associated with a high degree of reproducibility (the relative standard deviation of repeated measurements was typically 0.2%). The results for the 1 L chamber tended to be 5% lower than the results for the 10 L chamber, which may reflect true differences in scattered radiation averaged over the respective volumes. Likewise, the Ludlum 9DP* consistently responded about 5% lower than the 1 L chamber. These tests suggest that the results could be significantly influenced by a non-uniform field (which therefore should be included in the uncertainty budget), and it would be of interest to consider alternative positions for such tests where the ambient dose would be more uniform or to test the uniformity for other gantry angles of the linear accelerator. To decrease the uncertainty in the reference doses, it would be of interest to compute specific $H^*(10)/K_a$-conversion coefficient using Monte-Carlo modelling of the photon energy fluence spectra at the measurement position.

With respect to the tested Ludlum 9DP*, it was found that the unit was generally in reasonable agreement with the reference values from the spherical chambers, and the unit worked well in the tested range of pulse widths and dose rates.

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S8-O4

The National System for the Introduction of New Health Technologies within the Specialist Health Service

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The National System for the Introduction of New Health Technologies within the Specialist Health Service was introduces in 2013. The main purpose with the system is to ensure that patients get access to new methods showing a safe and evidence based clinical effect and avoid unsafe and non-effective methods. The Norwegian Radiation Protection Authority (NRPA) became a full-part member in August 2014. The motivation for this implementation is the national radiation protection regulations requirement for a generic justification evaluation of new methods and practices in medical exposure before they are introduced in general clinical practice. This national system coordinates all steps and authorities involved in the process of evaluating new methods before they are put into general practice. The system covers alerts of new methods, evaluation of new methods, prioritizing, decisions and finally implementation. Evaluation of methods follows international principles of health technology assessment (HTA) and is performed at three levels: mini-HTA, fast-HTA and complete-HTA. Mini-HTA is performed locally at the Hospital Trust, while fast and complete HTA are performed at a national level. The Norwegian Medicines Agency (NMA) is responsible for the fast-HTA of pharmaceuticals and the Norwegian Knowledge Centre for the Health Services (NKCHS) is responsible for fast-HTA of medical equipment. A complete-HTA is always performed by NKCHS. NRPA is assisting both NMA and NKCHS in fast and complete-HTA of new radiopharmaceuticals and new equipment and procedures for use in medical exposure. NRPAs role in the national system is to ensure that radiation protection issues are evaluated and taken into account in the total risk-benefit evaluation of the method in all three levels of evaluation. As a part of the system, NRPA get properly involved in all processes related to the introduction of new methods and a national overview of local mini-HTA performed on the different hospital trusts for equipment and procedures within medical exposure.

S9-O1

Computed pediatric tomography exposure and radiation-induced cancers: Results from a national cohort study in France

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Context: The increasing use of computed tomography (CT) scans for the pediatric population raises the question of the possible impact of such ionising radiation (IR) exposure on the occurrence of radio-induced cancers. Recent epidemiological studies have suggested an increased risk of cancer among children receiving CT scans. In France, a nationwide study has been launched to assess cancer risks, especially leukemia and cerebral tumors, associated with the use of CT scans in pediatrics. This study is part of the Epi-CT collaborative European project.

Material and methods: The cohort includes children less than 10 years old, subjected to at least one CT scan between 2000 and 2011 in 23 French University hospitals. Cumulative organ doses were estimated according to the protocols retrieved from the radiology departments, using specifically designed simulation software. Clinical information recorded during hospitalization was used to determine whether the children had medical predisposing factors (PFs) likely to increase their risk of cancer. Cancer incidence and mortality data were retrieved through national registries.

Results: At all, 67 274 children were included, 30% of whom were exposed to a first CT scan before the age of 1 year. Examinations of the head represent 57% of the CT scans. PFs to cancer were observed in 2.3% of the children. During the follow-up from 2000 to 2011, 27 children were diagnosed with cerebral tumors, 25 with leukemia. Hazard ratios (HRs) of 1.06 (95% CI 1.02–1.10) for cerebral tumors and of 1.08 (95% CI 0.80–1.44) for leukemia (17 cases) were estimated for each increment of 10 mGy in CT X-rays organ dose. In children without PFs, hazard ratios were similar to those which were estimated for the whole cohort while HRs decreased in children with PFs. This decrease of risk was possibly due to a concomitant increase in non-cancer mortality risk among children with PFs.

Conclusion: These first results indicate that patients with PFs should have a very different risk of radiation-induced cancer than patients without PFs. Then, in terms of public health, the most relevant risk estimates should be analyzed separately for each group. Confounding bias by indication could nevertheless not be excluded and should be investigated by extending the follow-up of the cohort and by other ongoing studies.

S9-O2

Pediatric protocols and dose reduction devices in CT scanners where few examinations are performed

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Abstract

Increased awareness of patient doses related to CT examinations has inspired the invention of various dose reduction features, as well as stressed the importance of pediatric protocols.

The aim of our study was to evaluate factors related to scanner design and setup that might result in considerable differences in radiation dose between geographic areas.

We examined the preset protocols and the technical features regarding dose reduction; automatic exposure control (AEC), iterative reconstruction (IR), automated kV selection and organ based dose modulation.

As expected the availability of dose reduction devices is highly, but not entirely, dependent on the manufacturing year of the scanner. In general, protocol optimization is more lacking than technical features. The majority of the scanners did not have age and/or weight categorized pediatric protocols and the dose reduction made possible with iterative reconstruction methods is not yet available everywhere.

Introduction

Increasing number of computed tomography (CT) examinations in the last decades has raised concerns especially regarding the pediatric population [1]. There are though indications of improvements, for example in specialized pediatric hospitals where the use of CT imaging has begun to decrease [2] and radiation doses are lower [3].

The increased radiation dose awareness has inspired important technical improvements in the area of dose reduction. Automatic exposure control (AEC) with tube current modulation (TCM) is now a well-established technique which can reduce patient doses considerably [4]. Among the emerging techniques in the last decade that can result in relevant dose reductions are; iterative reconstruction (IR) [5,6], automated kV selection [7], and organ dose modulation [8]. These new technologies are generally not all available in all scanners, especially not in the older ones.

It is known that for the same procedure, patients' radiation dose can vary considerably and more than 10-fold variation in estimated median effective dose for a baby has been reported within trauma center facilities [9]. One of the most important things to do is to "child size" the amount of radiation used [10], by using pediatric protocols and implement all the available dose reduction technologies in each scanner.

There was a 20% increase in the number of CT examinations in Iceland from 2008-2013. A substantial amount of all CT examinations is done in CT scanners in which less than 5000 examinations are done per year and those scanners might not be equipped with optional technical features.

In scanners where pediatric examinations are few and far between it is difficult to establish a reliable evaluation of patient doses for each procedure. One way to evaluate the situation is to examine the preset protocols and the dose reduction techniques available in the respective scanners.

The aim of our study was to evaluate factors related to scanner design and setup that might result in considerable differences in radiation dose between geographic areas.

Materials and method

All scanners outside the Reykjavik-capital area were included, in each of them relatively few CT examinations are performed (less than 5000/year) in contrast to CT scanners in the more densely populated capital area. Existing data about number of examinations in 2013 was mined to collect information about examination frequency and proportion of pediatric examinations.

We visited each scanner and examined the technical features regarding dose reduction; AEC, IR, automated kV selection and organ based dose modulation. Note that the last two are not available from all CT vendors. In the same visit we examined the preset protocols in every scanner to confirm if there were pediatric protocols, how many, how they were labelled and if dose reduction devices were used. At last we asked the radiographers if and when pediatric protocols were used.

Results

Outside the Reykjavík-capital area there are seven scanners, and they were all included in the study. The oldest scanner in the study was from 2004 (11 years), the mean age was 5 years, and two of the scanners were new (2015). There were three General Electric scanners, two Toshiba and two Siemens scanners.

In the year 2013 21% of all CT examinations in Iceland were performed with the seven scanners included in this study, and the remaining 79% were performed in the five clinical CT scanners in the Reykjavík-capital area.

Pediatric examinations are on average 3% of all examinations, and the highest proportion of pediatric examinations is at the University hospital (5%). In all the scanners included in this study (scanners were few examinations are performed, outside the capital area) the proportion of pediatric examinations was 2-4%.

There were pediatric protocols for head CT in all seven scanners, but in the majority of scanners there was only one pediatric protocol as shown in Figure 1.

Two of the seven scanners (29%) had no pediatric abdomen protocols. In four scanners (57%) there was one pediatric abdomen protocol and one scanner had abdomen protocols based on weight groups. The lowest kV was 100kV.

All of the seven scanners had three dimensional AEC but none of them had organ based dose modulation.

One scanner (14%) had the availability of automated kV selection, but at the time of the study the automated kV selection was not yet implemented in the pediatric protocols.

IR is available in the three newer scanners (2013 and newer), 1st generation IR in one scanner and 2nd generation in two, but not in the four older ones (2004-2010). In all three scanners with IR it was implemented in the pediatric abdomen protocols, but only in one scanner in the head protocols.

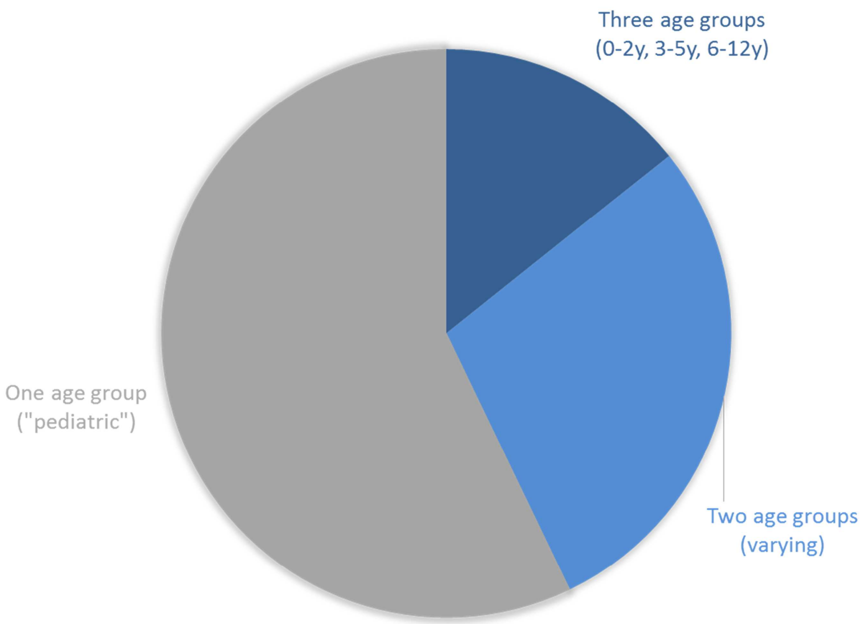

Figure 1. Pediatric head protocols in seven CT scanners outside the capital area, where few examinations are performed. The label "One age group" means that only one pediatric head protocol was present.

Discussion

As expected the availability of dose reduction devices is highly dependent on the manufacturing year of the scanner, but not entirely. An interesting example is that the only scanner with the automated kV selection is the second oldest of all the scanners.

All scanners had some preset protocols for pediatric examinations but some only for head CT. Radiographers confirmed that this reflects the requests for pediatric CT, where head CT is more often requested than abdomen CT. Radiographers in hospitals near Reykjavík claim that they purposefully direct pediatric abdominal CT to the University hospital and our data supports that.

We did though assume that pediatric protocols were necessary in all scanners and that kV should be lower for the smaller children, because one method to lower the radiation dose is to use lower kV when the subject is small (11). Without automated kV selection, automatic tube current selection is generally not enough to optimize dose and image quality in pediatric patient because a lower limit on the tube current may restrain the dose reduction. Pediatric protocols also ensure that an appropriate beam shaping filter is used.

There might be a different need for age or weight based pediatric protocols depending on scanner manufacturer, based on difference in scanner design. An example of this is that while some types of AEC modulate the tube current directly according to image noise, other types take into account that in smaller patient the inherent contrast is less and thus less noise is needed for similar perceived image quality (12).

AEC is not recommended in head protocols in all scanner types and thus these protocols often have fixed mA. We found that this was often the case and unfortunately there was only one pediatric protocol in majority of the scanners, a fact from which we can assume that many of the larger children are scanned with adult protocols. It was confirmed at one site that adult protocol was used for all patients aged 8 years and older.

IR was available in three scanners but interestingly it was only used in head protocols in one of them although it was always used in abdomen protocols when available. This might reflect that dose reduction is considered more important in abdomen CT than head CT, but radiographers also mentioned that radiologists do not accept the altered appearance (texture) of the images that result from the IR.

In general, protocol optimization is more lacking than technical features. The majority of the scanners did not have age and/or weight categorized pediatric protocols (Figure 2) and the dose reduction made possible with IR is not yet available everywhere.

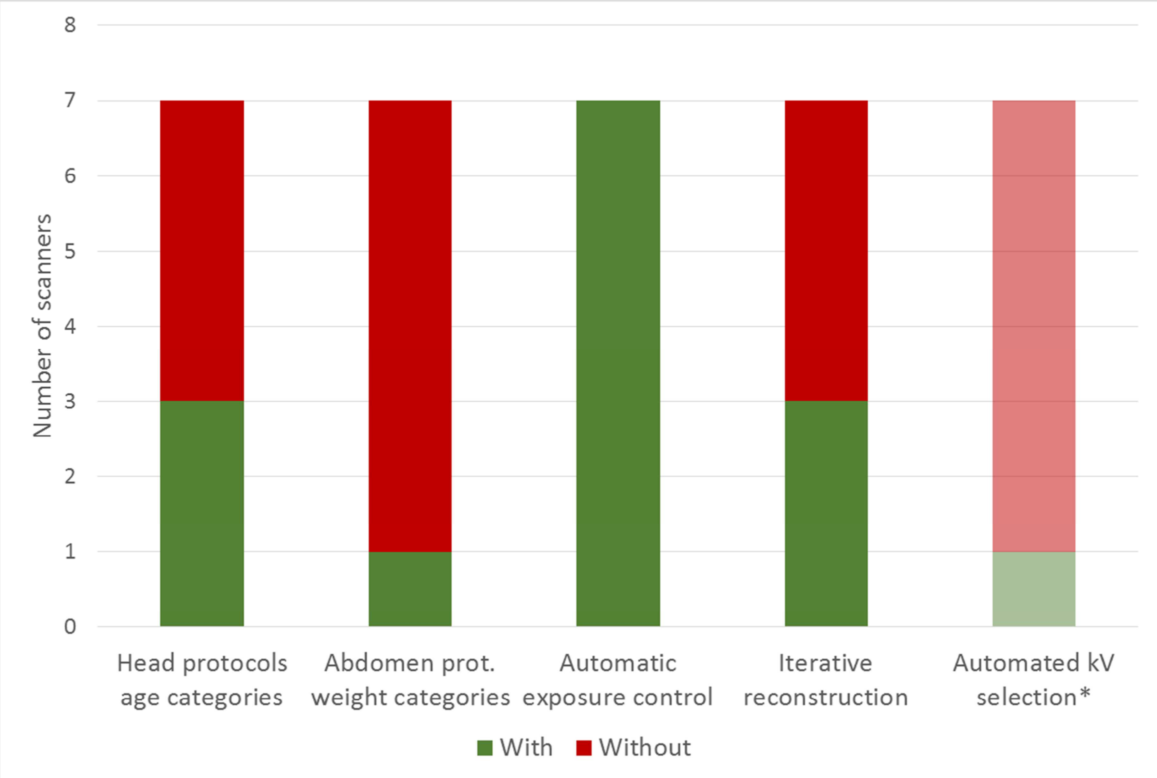

*Figure 2 The number of scanners with (green) and without (red) age categorized pediatric head protocols, weight categorized pediatric abdomen protocols, automatic exposure control, iterative reconstruction and automated kV selection. *Not available from all vendors.*

Although data on examination frequency showed that pediatric examinations were performed in all scanners included in the study we found more than one example indicating that pediatric protocols were neglected. Two examples of this are that; a) where automated kV selection was recently installed the technique was not implemented in the pediatric protocols although it was in the adult protocols, and b) where radiographers were not aware that there were pediatric protocols in the scanner and thus did not use them.

In conclusion, the dose reduction techniques that can be expected according to scanner age are available but the pediatric protocols are few in most scanners and need improvement.

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S9-O3

Population doses from x-ray and nuclear medicine procedures in Nordic countries

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Population doses from radiodiagnostic (x-ray and nuclear medicine) procedures in Nordic countries were estimated based on data collected from 36 European countries in Dose Datamed 2 (DDM2) project. In Nordic countries the mean effective dose per caput from all diagnostic x-ray and nuclear medicine procedures was 0,87 mSv which is lower than the corresponding dose 1,12 mSv in the EU and EFTA countries (except Lichtenstein).

For x-ray procedures in Nordic countries a mean effective dose was 0,83 mSv per caput, being lowest in Finland (0,45 mSv) and highest in Iceland (1,70 mSv). Compared to the average of EU and EFTA (except Lichtenstein) countries included in the survey in which a mean effective dose was 1,05 mSv per caput, only Iceland and Norway had higher per caput doses and also their frequencies of CT procedures were higher than in other Nordic countries.

For nuclear medicine procedures in Nordic countries a mean effective dose was 0,04 mSv per caput, being lowest in Finland (0,02 mSv) and highest in Denmark (0,07 mSv). Compared to the EU and EFTA (except Lichtenstein) countries in which a mean effective dose was 0,06 mSv per caput, only Denmark had higher per caput dose, and the frequency of positron emission tomography (PET) studies was higher than in other Nordic countries.

Recent increases in medical imaging, particularly with respect to computed tomography (CT) and other high dose procedures, have led to significant increase of individual patient doses and of the collective dose to the population as a whole. The overall per caput effective doses from radiodiagnostic procedures in Nordic countries are about half the recent value of per caput effective doses estimated in Australia (Wallace 2012) and about one-third of the corresponding value in the USA (NCRP 2009). Comparing the results with an earlier estimation of population dose in Europe, in the DDM1 countries, there seems to be a trend upwards, in Denmark 92 % and both in Norway and in Sweden 14 %. In Finland the overall per caput effective dose has not changed.

A relatively low value of population dose can be a good sign for the successful implementation of the justification and optimization principles in radiation protection. The Nordic countries are considered to be very similar to each other; however there seems to be variation in use of different imaging modalities.

S9-O4

Sunbeds and sunburns in Iceland

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Abstract

According to two recent lifestyle surveys in years 2007 and 2012, Icelanders received a relatively high proportion of their sunburns abroad and in sunbeds. In the 2007 survey, Icelanders reported more sunburns abroad than domestically and 7% reported burning in sunbeds. In the 1980s with many more sunbeds and much less travel abroad, the proportion of sunburns owing to sunbeds may have been even larger. It may be deduced that skin cancer prevention programs in Iceland should place extra emphases on warnings for travellers and sunbed users.

Introduction

Since UV-exposure is the main risk factor for skin cancers, a low incidence rate of melanoma can be expected in Iceland due to its northern latitude. Between years 1990 and 2000 the incidence rate of melanoma did however rise rapidly in Iceland and among young women it became the highest in the Nordic countries [1]. The incidence rate started to decline after year 2000 and it is currently below the Nordic average (Figure 1). It has been suggested that the increase was due to the introduction of the modern type of sunbeds in the 1980s. Melanoma is increasingly found in body locations normally covered by cloth [2], which makes a connection to intentional UV-exposure likely. A large number of sunbeds in Iceland around year 1990 may have facilitated this exposure [3].

From the on-line Nordic cancer registries [4], the following observations can be made that support the association between incidence of melanoma and use of sunbeds:

1) The incidence increase is most marked among young women, the typical customers of sunbed saloons.
2) The increase was confined to the capital area (Figure 1). It seems likely that sunbed use is part of an urban lifestyle.
3) In addition to the incidence increase of melanoma there was an increase in the mortality rate for women in the capital in the years 1991- 2009 as compared to 1973 – 1990 (according to Figure 1). The mortality rate for women on the countryside did not increase.

These observations may be due to other causes than the use of sunbeds. The increased incidence rate of melanoma might partly be due to an increased effort in finding cancers at their earlier stages even if this does not suffice to explain the apparent increase in mortality. Increase in travel to sunnier places may also have been a contributing factor.

Data for number and usage of sundbeds on the countryside are not available until 2004 and later. It is likely that Reykjavik had more sunbeds per capita than the countryside in the beginning (1980–1990) but it must be noted that recent surveys have not reported large differences between the countryside and the capital in sunbed numbers and use.

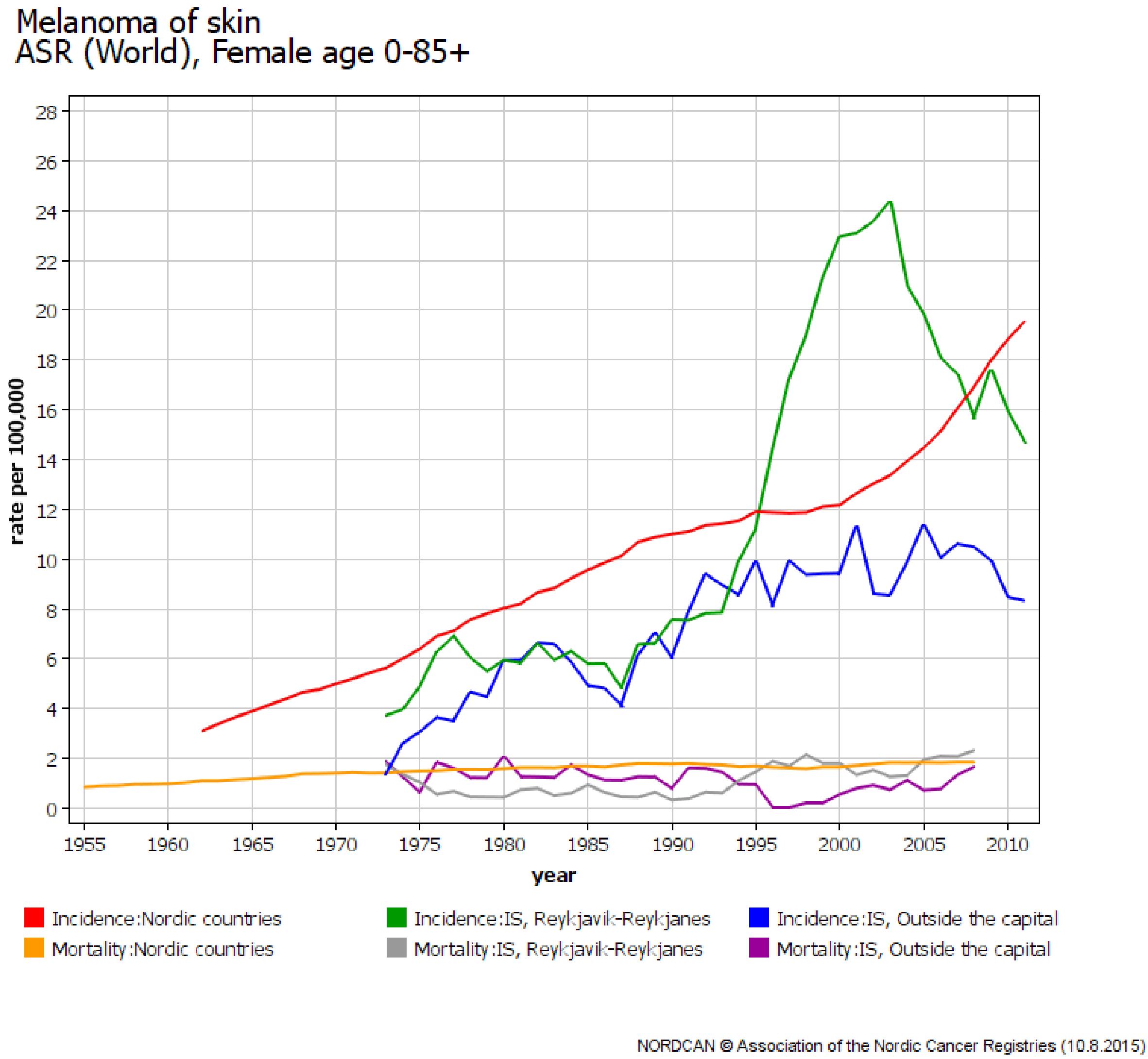

Figure 1. Incidence and mortality rates of melanoma for women in Iceland (capital and countryside) and other Nordic countries. Five years moving average. Age standardized according to WHO- world population. The figure and its data is from on-line Nordic cancer registries, reference [4].

This study provides information on sunburns that Icelanders receive in sunbeds and abroad. The data used in the study originates from the health and lifestyle survey “Heilsa og líðan Íslendinga” carried out by the Public Health Institute of Iceland in year 2007 and by the Directorate of Health 2012 [5, 6]. The sample was a stratified random sample of 10.000 individuals between 18-79 years. The response rate in 2007 was 60.8% and 54.5% in 2012. Questions were asked for present outdoor tanning habits, location (Iceland/abroad) sunbed use and self-reported sunburn, location (Iceland/abroad) and burn from sunbed use during the past 12 months.

Sunburns before and after onset of economic crisis

The economic crisis Iceland in 2008 gives a unique opportunity to study the effect of travel on sun-exposure. With increased financial well-being, Icelanders travelled abroad in ever-greater number. In 2007, at the height of an economic upheaval, Icelanders were subsequently more likely to spend their summer vacation abroad than ever before. In 2007, a higher proportion of Icelanders reported that they had sunburned abroad than domestically. In 2012 after the devaluation of the currency and introduction of capital controls, this was reversed (Table 1).

Table 1 Percentage reporting sunburn during outdoor sunbathing in 2007 and 2012.

Year	Age	Sunburns Domestic	Sunburns Abroad
2007	18 and older	17%	19%
2012	18 and older	25%	11%

The frequency of sunburns abroad was probably not this high in the 20th century when Icelanders travelled far less (Figure 2).

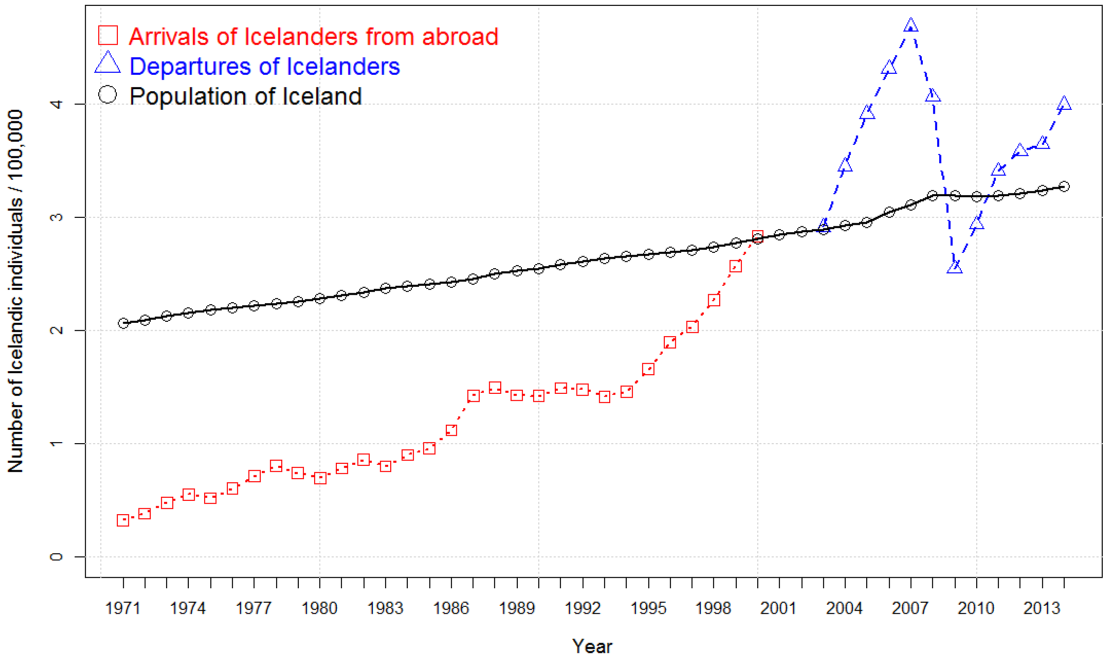

Figure 2 Business travelling as well as holiday trips via airlines (Keflavik) [7].

Sunburns and sunbeds

Another interesting outcome of the lifestyle surveys in 2007 and 2012 is a relatively high percentage of Icelanders reporting burns in sunbeds (Table 2).

Table 2 Percent burning in sunbeds

Year	Age	Burning in sunbeds
2007	18 and older	7.5%
2012	18 and older	4.3%

The number of sunbeds in Reykjavík decreased rapidly between 2007 and 2012, which is probably reflected in these percentages. There had however already been a decrease since 1988 (Table 3). During that time, burns in sunbeds are likely to have been even more common.

Table 3 Number of sunbeds in Reykjavik (capital area)

	1988	2005	2008	2011	2014
Number of sunbeds	207	144	98	76	61
Number per 1000 inhabitants	1.5	0.8	0.5	0.4	0.3

Conclusions

Even with a greatly reduced number of sunbeds, some percent of Icelanders are still reporting burns in sunbeds. This supports the suggestion that sunbeds may have played a role in the increased incidence rate of melanoma between 1990 and 2000.

In the last decades, a large proportion of sunburns occurred during sunbathing abroad. The proportion diminished after the economic crisis in Iceland took hold with reduced travelling but still remained large. Therefore, future skin-cancer-prevention-programs in Iceland might want to emphasize warnings for sunbed users and messages that enhance sun protection among those who travel to sunny resorts in lower latitudes.

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S10-O1

The importance of implementing radiation protection in the national eHealth-strategy

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The technological development in medical exposure is fast and new equipment and methods are continuously introduced and implemented in clinical use. As a consequence, medical exposure is now the largest man-made source of radiation to the population. To ensure for proper medical exposure it is crucial that justification, optimization and dose limitation is properly implemented at health care facilities. International and European organizations have lately increased their focus on the necessity of implementing these radiation protection principles in the health care facilities IT-systems. Norway has now initiated a national eHealth strategy. One of the objectives is to make data available for quality assurance, health surveillance, management and research. The Norwegian Radiation Protection Authority (NRPA) has now been included in the national eHealth strategy. Two major projects are under development at NRPA to strengthen the implementation of justification and optimization in medical exposure. Unjustified radiological examinations are an increased challenge both from a health economical and radiation protection perspective. Evidence based referral criteria and their implementation in clinical decision support systems (CDS) is identified as an important tool to reduce unjustified examinations. NRPA is part of a national working group with the mandate to look into the possibilities for adapting and adopting already available referral criteria, to implement them in CDS systems and connect it to general practitioners electronic referral systems. Local and national overview of patient doses is essential for optimization of radiological examinations and to be able to communicate radiation risks to patients. NRPA strongly recommend that dose parameters from radiological examinations automatically are transmitted from the modality to a local database that support statistics and reporting to national records. NRPA have initiated cooperation with the National Patient Register (NPR) to automatically collect activity and dose data from the health care facilities. These data will further give input to a national surveillance system for medical exposure. This system will provide NRPA with important information of national dose distributions and frequencies of typical radiological examinations and allow for establishing national diagnostic reference levels (DRLs) and frequent estimates of the population dose from medical exposure.

S10-O2

Clinical audits for breast cancer radiotherapy in Norway

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Background: During 2009-2011, external peer review clinical audits for breast cancer radiotherapy were carried out at nine radiotherapy centres in Norway. The purpose was to assess compliance with the national radiotherapy guidelines regarding treatment planning for post-operative left sided breast cancer. All radiotherapy departments volunteered to take part in the audits. The audits were a joint project between the Norwegian Breast Cancer Group (NBCG) and the quality assurance group in radiotherapy (KVIST) at the Norwegian Radiation Protection Authority.

Material and methods: The audit topic was post-operative left sided breast cancer radiotherapy, focusing on indication and radiotherapy treatment planning. The audit standard was the national guidelines for breast cancer treatment (developed by NBCG). The auditors had a list of items for evaluating clinical practice against the audit criteria. A total of 180 treatment files were audited. The findings were grouped to analyse the degree of guideline compliance for indication, treatment technique, delineation of treatment volumes and organs at risk, and dose related parameters.

Results: The treatment was in accordance with the guidelines for indication, treatment technique and lung delineation in 98% of the cases, with minor deviations in 2%. For delineation of the clinical target volume (CTV) and the heart, minor deviations were found in 44% of the cases, major deviations in 6%. The dose distribution to the CTV was in accordance with the guidelines in 89% of cases, with minor deviations in 11%. A minimum dose to CTVbreast/breast wall of at least 95% of prescribed dose ($D98\% \geq 95\%$) was attained in 39%, whereas $D98\% \geq 90\%$ was attained in 93% of the cases. According to the auditors, the guideline principles were attained without or with minor deviance in 87% of the cases. Conclusion: For a large majority of audited cases, the radiotherapy was planned in accordance to the guideline principles. Delineation of target volumes and optimal dose distribution are challenging tasks, for which improvements are still desirable.

S10-O3

Inspection of Cardiology departments in Norway: Are they making it great in radiation protection?

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Purpose: Cardiology departments are one of the largest users of medical radiation, and interventional cardiology procedures are increasing, both in frequency and complexity. The procedures have great clinical benefits for patients, but they have the potential to induce radiation injuries. Staff involved in interventional cardiology receives the highest occupational doses in Norway, and skin burns of patients have been reported. Focus and awareness on radiation protection (RP) in cardiology is therefore crucial to reduce the associated radiation risks. To identify the level of implementation of RP for patients and staff, and compliance with the RP regulation, the Norwegian Radiation Protection Authority carried out inspections with all cardiology departments in Norway.

Method: The inspections were conducted (2013–14) as quality system reviews, based on document reviews, interviews, on-site inspections and observations of interventional procedures. Focus topics were organisation of RP, role and involvement of RP officer and medical physicists, education and training in RP, justification and optimisation, protection of staff and patients, personal dosimetry, local standard dose, monitoring and follow-up of patient doses and performance of quality control of X-ray equipment. Cardiology interventions are centralised in eight hospitals in Norway, seven public and one private.

Results: The inspections revealed that most of the hospitals had non-compliances according to the RP regulation. Most deviations were associated with education in RP and follow-up of patients who had received high radiation doses. Lack of systematic optimisation of procedures and estimation of the eye lens doses to evaluate the risk of exceeding the new proposed dose limit for cardiologists with high personal dosimetry readings (worn over the apron) was common. Other common non- compliances dealt with establishment of local diagnostic reference levels and their systematic use in optimization of cardiology practice. Notification of unintended incidences, especially patient doses much higher than intended, was not systematically reported in the hospitals quality system. The inspections revealed a need for increased awareness of RP in cardiology practice.

Conclusion: Level of compliance with some of the requirements given in the RP regulation was poor. Inspections turned out to increase the awareness of RP in cardiology and are identified as an effective tool for improving RP and safety.

S10-O4

Measurement of eye lens radiation doses to staff during percutaneous coronary interventional procedures

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Introduction

Interventional cardiology staff has a high exposure to radiation. In light of the recently lowered recommended occupational eye lens dose limit by ICRP, it is now even more justified to continuously monitor staff radiation dose. However, measuring eye lens doses is challenging because the dose at any measurement point differs from the dose to the eye lens, depending on variations in staff position relative to the patient, x-ray tube geometry (projection angle), radiation quality, patient size and the presence of radiation protection shields and/or personal lead glasses, as well as other factors

Aim

The main aim of this study was to investigate the relationship between the absorbed dose at different measurement positions near the eye and the absorbed dose to the lens during cardiac interventional x-ray procedures. Furthermore, we aimed to assess the effect of projection angle, patient size, lead shields and protective eyewear on this relationship.

Methods

Phantom measurements were carried out using two anthropomorphic phantoms representing patient and operator, respectively. The operator phantom was at the typical position of the interventionist. The head phantom was modified so that a dosimeter could be inserted at the anatomical eye lens position. Two direct dosimeters calibrated in terms of Hp(0,07) were used to measure the equivalent dose, one dosimeter placed in eye lens position and the other at one of three chosen measurement positions on the forehead (left temple, above the left eye and mid forehead above the nasal bridge). Several exposures were made with automatic exposure parameters, projection angles (lateral and craniocaudal) ranging from -90 to 90 degrees, two patient sizes (standard phantom and additional 5 cm PMMA), and for two types of commonly used lead glasses.

Results

The results indicated that for a wide range of projection angles, the lowest deviation from the dose to the eye lens was received for the dosimeter placed on the left temple, with an mean (SD) deviation of 2.7 (5.1) %. The other two measurement points, above left eyebrow and mid forehead, deviated with -15 (7.5) % and -30.5 (4.1) % respectively. The same tendencies were seen for a larger patient size.

The mean (SD) dose reduction ability of lead glasses was 7.1 (1.6) % and 6.8 (2.2) % for the two models, while a properly positioned ceiling-mounted lead shield completely blocked stray radiation to the eye lens.

Conclusion

Eye lens dosimetry is challenging in a clinical environment because dosimeters have to be positioned at a remote position from the eye lens. For cardiac interventional x-ray procedures, a dosimeter placed at the left temple provides the most accurate measurement of eye lens dose. Protective lead eyewear has limited dose reduction ability of less than 10 %, while a ceiling-mounted lead shield provides a very effective way to reduce the dose to the eye lens

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S10-P1

Frequency of Medical X-ray Examinations in Iceland in 2013

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Introduction: The frequency of medical X-ray examinations has been investigated with a collection of national data every 5 years for the past two decades. The most recent collection of data was done in 2014 for the year 2013. In this report the results of this collection will be presented.

Materials and methods: Medical X-ray and Nuclear Medicine (NM) examinations are performed at 38 different locations in Iceland: hospitals, health care facilities, private clinics and research facilities. These examinations are stored (saved) in 6 different RIS/PACS systems. Information was collected about examination types according to RES codes, imaging modalities, department or the location where the examinations were performed and patient identification information (sex, date of birth and national id). Along with information about X-ray and NM examinations, information about other medical imaging examinations were collected (Ultrasound (US) and Magnetic Resonance Imaging (MRI)).

Results and discussion: The number of all medical diagnostic imaging examinations in Iceland in 2013 is presented. This number has decreased by a small fraction from 2008 when this information was collected last. Conventional X-ray examinations have decreased considerably, while there is a steady increase in CT, Interventional, MRI and US examinations. The number of NM examinations has continued to decrease from 2008 as it did in the period from 2003. A collection of patient dose data from the major contributing X-ray and NM-departments is ongoing and should be concluded by the end of this year or early 2016.

S10-P2

Ra-223 planar whole body scan and SPECT of surgically removed bone

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Introduction: Radionuclide therapy with Ra-223 dichloride has been initiated for prostate cancer patients with symptomatic bone metastases. Typical prescribed activity is 50 kBq/kg bodyweight, 6 times at 4 weeks interval. Ra-223 is an alpha-emitter, but it also emits gamma radiation, that can be utilized for imaging purposes.

Methods and materials: The possibility of imaging Ra-223 treated patients and bone was investigated. We acquired a SPECT scan 27 days after the last treatment of ex-vivo bone, which was removed during hip surgery. The hip bone was stored in a plastic container in a formaldehyde solution. A planar whole body scan of a patient (in-vivo), who was administered 4.4 MBq Ra-223 one hour before scanning, was also acquired. Half-life of Ra-223 is 11.43 d. The decay chain of Ra-223 to stable Pb-207 involves 6 stages, all with shorter half-lives (ms to min) than Ra-223. Four stages are by alpha, two by beta emission. Total emitted energy in the decay is 28 MeV; X-ray and gamma lines (1% of total energy) in the interval 80-400 keV allow external detection. A dual head Philips Precedence SPECT-CT with MEGP collimators was used. A one hour planar whole body scan was acquired with 40 mm/min scan speed and 2.78 mm pixel size. SPECT acquisition was performed in step and shoot mode with a 128x128 matrix size, 4.66 mm pixels and 128 angles. Data was acquired 600 s per angle resulting in a total acquisition time of almost 11 hours. Reconstruction was performed with a resolution recovery OSEM method (Astonish) with 3 iterations and 8 subsets. Attenuation Correction was performed with a 140 kVp low dose CT.We compared a set-up with two 20% width energy windows. One at 269 keV corresponding to the most intense gamma-line (14% yield) and another one at 84 keV overlapping the two most intense X-ray emissions at 84 and 81 keV (25% and 15% yield).

Results: The lower energy window resulted in visually better images than the higher energy window in both cases. The SPECT of the bone revealed that spatial allocation of the counts is best in the 84 keV window. The 269 keV SPECT showed a significant amount of counts in areas without bone indicating a bigger influence of scatter.

Discussion and Conclusion: LEGP collimators might improve the quality for the 84 keV SPECT, as long as downscatter from higher energy photons does not become a problem. We have showed that it is possible to image patients treated with Ra-223 and that it is best done with the 84 keV window.

S10-P3

Whole body counting of radium-223 for monitoring of staff in radionuclide therapy

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Aim: Radionuclide therapy with Ra-223 dichloride has been initiated in many countries for prostate cancer patients with bone metastases (Xofigo, Bayer). Prescribed activity is 50 kBq/kg, 6 times at 4 weeks interval. To license use of Ra-223, Danish authorities requested proof of ability to document compliance with dose limits. We used our low-background whole body counter (WBC).

Materials and Methods: Half-life of Ra-223 is 11.43 d; it decays to stable Pb-207 through 6 stages, with half-lives ms to minutes. Four stages are by alfa, two by beta emission. Total emitted energy per decay is 28 MeV; X- and gamma lines at 80-400 keV (1% of total energy) allow external detection. ICRP dose coefficient for Ra-223 inhalation is 6.9 mSv/kBq, hence dose limit for workers (20 mSv) may be received by a single inhalation of 3 kBq, or by a weekly uptake of ~60 Bq. The WBC is underground and shielded with 15 cm of steel lined with Pb and Cd. The 4 detectors are 6*4” NaI-cylinders. A WB phantom with human levels of K-40 in water was configured from 1L and 2L bottles corresponding to different person weights. A sample of 40 kBq Ra-223 was moved around to simulate uniform distribution or focal lung uptake. Various detector configurations were tested using 20-445 keV window and T= 1800 s for background (BG, “cold” phantom) and test. We compared sensitivity S (cps/Bq) to BG to determine minimal detectable activity (MDA) and minimal quantifiable activity (MQA). Optimized settings have been used on 10 treatment days.

Results: With std configuration for WB measurement, S =0.026 cps/Bq and BG=30 cps in the 77 kg phantom. From 55 kg to 88 kg, MDA was 20-24 Bq and MQA (10%) 60-74 Bq. In a configuration optimized for lung uptake, S increased to 0.14 cps/Bq with BG=37 cps. Resultant MDA (T reduced to 1200 s) was improved to 5-6 Bq, and MQA (10%) to 15-18 Bq. In consideration of Tc-99m contaminations (Bq-amounts), the window was reduced to 210-445 keV, with an increase of MDA to 8-10 Bq and MQA (10%) to 26-31 Bq. To date, no internal contamination of staff with Ra-223 has been detected.

Conclusion: MDA and MQA limits with WBC are sufficient to document compliance with dose limits. Knowledge of true BG is essential, and must be individually determined on each treatment day for quoted figures to be obtained. If low-level “continuous” uptake can be excluded, standard (personal) BG may still be sufficient to detect any dose significant uptake in case of a contamination incident.

S10-P4

Developments in first choice from conventional X-rays to CT for selected studies

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In the past 15 years extensive development within the field of CT has taken place. As a result, for some indications the first choice of examination method has changed from the conventional x-ray examination to CT examination. One example is the practice changes within kidney and colon examination over the past 10 years.

We have compared the development in the numbers of examinations, performed in hospitals and clinics in Denmark, for the following examinations:

- Intravenous urography (IVU) vs. CT urography
- Colon fluoroscopy vs. CT colography

Patient doses for selected x-ray examinations are collected from hospitals and clinics by the National Institute of Radiation Protection with the aim of setting national diagnostic reference levels (DRLs). DRLs have existed for many years for IVU and colon fluoroscopy, while a DRL for CT colography has been set in Denmark primo 2015, and a DRL for CT urography Is still preliminary due to lack of sufficient data.

For both urography and colon examinations, a CT examination has become the first choice. By switching from the conventional X-ray examination to CT scanning, the dose to the patient is increased significantly. However, the CT scanning will provide more information, which may result in a faster diagnosis by avoiding the supplementary examinations. It is important that the CT examinations are justified and optimized.

We will continue to follow the CT doses for these and other examinations, as a lot of work within dose optimization is being carried out both at the producer and locally at the hospitals.

S11-O1

Swedish Radiation Safety Authority: Systematic monitoring and evaluation of work practices an important aspect of improving radiation safety for patients

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Introduction

Background

Inspections performed by the Swedish Radiation Safety Authority (SSM) have shown that radiographers do not follow some basic radiation protection guidelines to reduce the radiation dose to patients. In addition, hospital management insufficiently monitors compliance with these guidelines [1]. Through this review, SSM wants to illustrate these shortcomings.

Reviewed requirements:

- To ensure that the right patient is examined.
- To ask women of childbearing age about possible pregnancy.
- Use of lead shielding of gonads.
- Use of compression.

Section 20 of the Swedish Radiation Safety Authority's regulations (SSMFS 2008:35) on General Obligations in Medical and Dental Practices using Ionising Radiation requires that irradiation as far as possible is to be minimised by appropriate selection of equipment and appropriate design of quality controls, working methods and routines and education or training programmes.

Section 9 of the Swedish Radiation Safety Authority's regulations (SSMFS 2008:31) regarding X-ray diagnostics requires the working methods to indicate which radiation protection measures are to be applied for a specific examination. The extent of the examination and the dose should be adjusted so that the radiation dose is as low as reasonably possible, while at the same time ensuring that the required diagnostic information is obtained.

In 2012, 199 radiology departments in Sweden were requested by SSM to report on existing guidelines regarding identity checks, X-ray examinations of women of childbearing age, when to use lead shielding of gonads for male patients and when to use compression. Thirteen radiology departments lacked guidelines and were requested to establish them.

Over the course of 2013 and 2014, the Swedish Radiation Safety Authority conducted a follow-up to the survey of basic radiation protection guidelines. Heads of staff at 94 radiology departments were requested to evaluate their compliance with existing guidelines for reducing radiation doses to patients.

This report contains a summary of the use of basic radiation protection guidelines at Swedish radiology departments.

Objective

The purpose of this review is to improve radiation protection of patients.

Aim

The aim of this review is to increase the adherence to basic radiation protection guidelines for X-ray examinations.

Definitions

Licence holder:	The natural or legal person operating the practice involving radiation and licensed according to the Radiation Protection Act for such activities (SSMFS 2008:35). In this context, the licence holder can be a county council or a private healthcare company engaged in a radiology practice in one or more radiological departments.

Method

SSM's review covered 29 licence holders in both public and private healthcare that operate medical practices involving ionising radiation at a total of 94 radiology departments. The review was performed on two occasions; the first review was conducted during the period January to June 2013 and the second during the period from March to May 2014. Via a web form on SSM's website, the radiology departments were to report on their compliance to basic radiation protection guidelines.

They were to report on:

- The number of examinations in which the guidelines should be applied.
- The number of examinations where guidelines had been applied.
- The number of examinations for which the guidelines had not been applied, but they had documentation that stated reasons for deviation from guidelines.

Delimitations

Both public and private healthcare providers were included in the review. Only radiology departments that conducted more than 10,000 examinations per year were included in the review. The numbers of examinations were collected from data reported to SSM in 2012 in conjunction with the Swedish Radiation Safety Authority's survey of guidelines for basic radiation protection [2]. The radiology departments included in the study accounted for approximately 86 per cent of the conventional X-ray examinations and about 95 per cent of the computer tomography examinations that were carried out in Sweden in 2011.

Regulations

Operations involving ionising radiation must meet the requirements of the Radiation Protection Act (1988:220), the Radiation Protection Ordinance (1988:293) and the Swedish Radiation Safety Authority's regulations (SSMFS). The purpose of radiation protection legislation is to protect humans, animals and the environment against the harmful effects of ionising radiation now and in the future.

The basis for practically all national radiation protection legislation is the International Commission on Radiological Protection's (ICRP) recommendations. The starting points of radiation-safe healthcare are the three basic principles of radiation protection [3]:

- Justification
- Optimisation
- Dose constraints

Justification is the process of determining whether a planned activity involving radiation is beneficial overall, i.e. whether the benefits outweigh the harm. Optimisation is the process of determining the radiation dose to be as low as reasonably achievable while at the same time ensuring that the desired result of the exposure is achieved. Dose constraints are not to be exceeded, but do not exist for patients, only for occupational and public exposure. If a radiological examination is justified and optimised, it can always be carried out regardless of the radiation dose the patient has been exposed to during previous examinations. Emphasis is to be on the eligibility assessment and optimisation of the current examination instead of the radiation dose the patient previously received.

Requirements for medical exposures

Section 20 of the Swedish Radiation Safety Authority's regulations (SSMFS 2008:35) on General Obligations in Medical and Dental Practices using Ionising Radiation requires that any radiation should be taken into account. The probability and consequences of accidental or incorrect exposure are as far as possible to be minimised by appropriate selection of equipment and appropriate design of quality controls, working methods and routines and education or training programmes.

- A thorough identity check to ensure that the right patient will undergo an X-ray examination is of great importance to avoid accidental exposure. On several occasions, incorrect or accidental exposures have occurred in Sweden because of mistakes in conjunction with identity checks according to deviation reports submitted to SSM.
- To ask women of childbearing age if they are pregnant before X-ray examination of the lower abdominal region is an effective way to avoid exposing a foetus to accidental irradiation.
- The use of compression in X-ray examinations of the abdomen and lower back is an effective way to reduce radiation dose to the patient. Three centimetres compression will reduce the radiation dose to the patient by half.
- To use lead shielding of male gonads when performing X-ray examinations of the lower pelvis or hips is an effective and simple way to minimise the radiation dose to the gonads. Studies have shown a reduction in radiation dose by up to 96.2 per cent for the gonads by using gonad protection [4] [5].

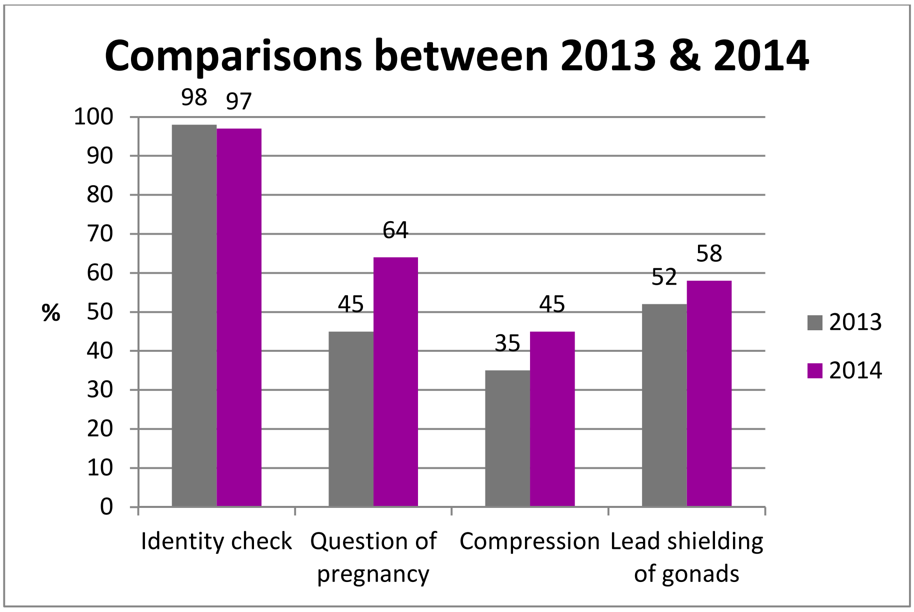
Section 9 of the Swedish Radiation Safety Authority's regulations (SSMFS 2008:31) regarding X-ray diagnostics requires the working methods to indicate which radiation protection measures are to be applied for a specific examination, such as the use of compression and lead shielding of gonads. The licence holder is required to have an established quality assurance programme that includes checking compliance to guidelines.

Summary of compliance review

Accounting for the periods 2013 and 2014

For the period in 2013, 12 out of the 29 licence holders had the ability to report on the use of basic radiation protection guidelines within the specified response time and for the requested period. About 33 per cent of the participating 94 radiology departments were able to follow up and evaluate the use of guidelines during the first period in 2013.

For the period 2014, 27 out of the 29 licence holders had the ability to report on the use of basic radiation protection guidelines within the specified response time and for the requested period. Four out of the 94 radiological departments were unable to follow up and evaluate their use of guidelines. (Detailed information regarding responses from the different licence holders is shown in Appendix: Figure 1-4.)



Ensuring that the right patient is examined

The results from 2013 show that of the radiological departments that had the ability to evaluate the compliance to guidelines regarding identity checks, their compliance was at just under 98 per cent. The results for 2014 showed that the 27 licence holders that had the ability to follow up and evaluate the compliance to guidelines regarding identity checks had average guideline compliance at just under 97 per cent.

Asking women of childbearing age if they are pregnant

The radiological departments that had the ability to evaluate the compliance to guidelines regarding asking women of childbearing age about possible pregnancy showed a compliance of 45 per cent in 2013. In 2014 the 27 reporting licence holders had an average compliance of 64 per cent in terms of guidelines regarding asking women of childbearing age about pregnancy. However, there were large variations in compliance between the different licence holders, ranging between 2 and 100 per cent compliance.

Use of compression

In 2013, of the radiological departments that had the ability to evaluate the use of compression in accordance with guidelines, their compliance was just under 35 per cent. In 2014 the compliance to guidelines regarding the use of compression averaged about 45 per cent. The compliance diverges between 4 per cent up to 100 per cent between the different licence holders.

Use of lead shielding of gonads

In 2013, of the radiological departments that had the ability to evaluate the compliance to guidelines regarding the use of lead shielding of gonads when examining male patients, their compliance was 52 per cent. In 2014, the average value in Sweden for the compliance to guidelines regarding the use of lead shielding of male gonads was 58 per cent. Guideline compliance ranged from 7 per cent to 100 per cent for the various licence holders.

Comparison between 2013 and 2014

When comparing the two periods 2013 and 2014, SSM has noted that the licence holders have increased their ability to produce required data for a certain period regarding the use of basic radiation protection guidelines. For the period in 2013, only 33 per cent of the participating radiological departments were able to produce required data. For the period in

2014, 93 per cent of the licence holders were able to produce required data. This should indicate that the licence holders now have the ability to evaluate their compliance to basic radiation protection guidelines.

Despite this improvement, the compliance to basic radiation protection guidelines is still low, around 50 per cent, indicating that there is great potential for improvement. The spread is wide and several licence holders, in both public and private healthcare, have 100 per cent compliance to guidelines.

<u>Identity checks</u>

For identity checks to ensure that the right patient is being examined, the average guideline compliance is good. This was also the outcome for the period in 2013.

<u>Pregnancy question</u>

Average guideline compliance before starting a radiological examination increased from 45 per cent to 64 per cent when it comes to asking women in childbearing age if they are pregnant. This gives improved compliance to guidelines by 40 per cent between 2013 and 2014.

<u>Use of compression</u>

The average use of compression in conjunction with radiological examinations of the abdomen and lumbar spine has increased from 35 per cent to 45 per cent, resulting in increased guideline compliance by 32 per cent.

<u>Use of lead shielding of gonads</u>

For the period in 2014, the average compliance to guidelines regarding lead shielding of male gonads was 58 per cent. This is a slightly higher percentage than the period in 2013, when average compliance was 52 per cent, an increase in compliance with routines by 11 per cent.

References

Radiation Protection Act, SFS 1988:220.

Radiation Protection Ordinance, SFS 1988:293.

The Swedish Radiation Safety Authority’s regulations (SSMFS 2008:31) regarding X-ray diagnostics.

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2. SSM2012-852 Enkät – Strålsäkerhetsmyndighetens undersökning om användning av strålskyddsåtgärder.
3. Annals of the ICRP, publication 103.
4. Grobe H, Sommer M, Koch A, Hietschold V, Henniger J, Abolmaali N. Dose reduction in computed tomography: the effect of eye and testicle shielding on radiation dose measured in patients with beryllium oxide-based optically stimulated luminescence dosimetry.
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Appendix

Chart 1 Identity checks

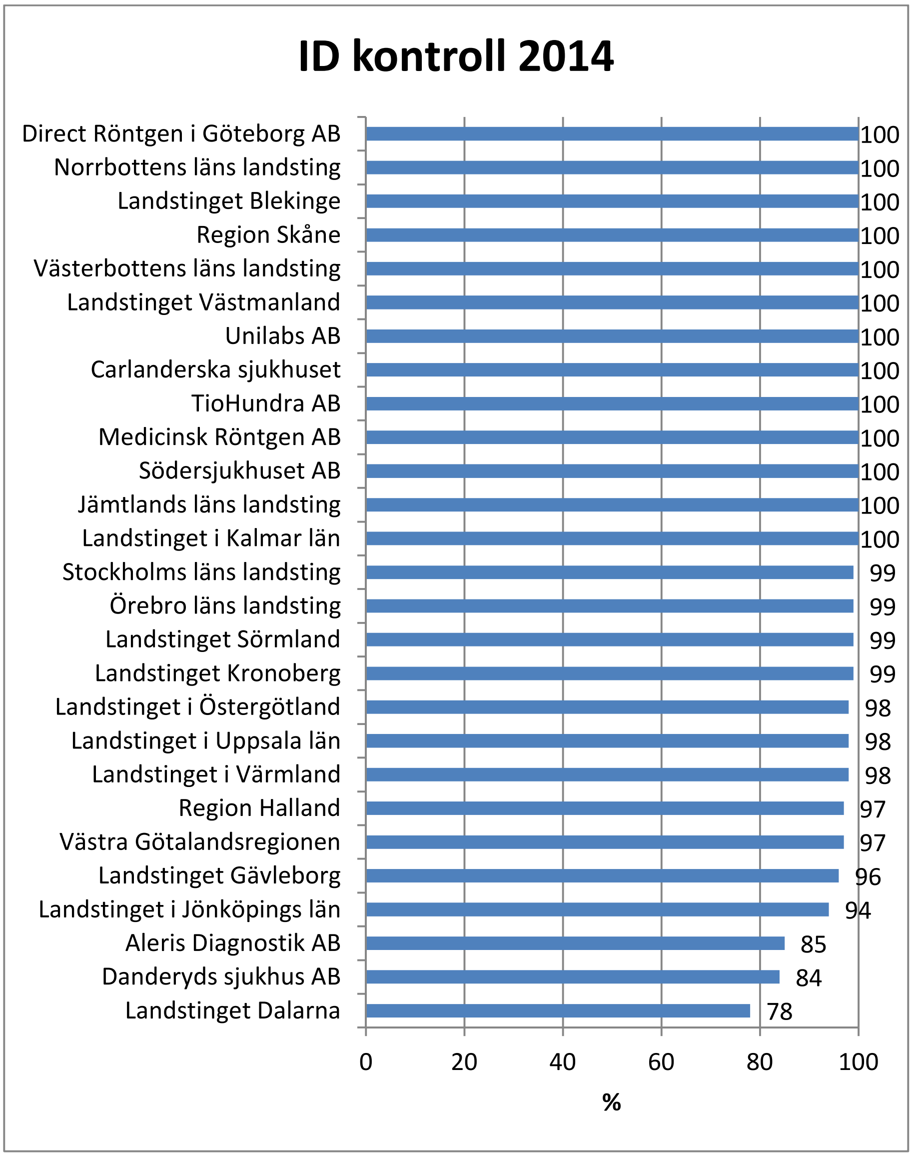


Chart 2 Pregnancy question

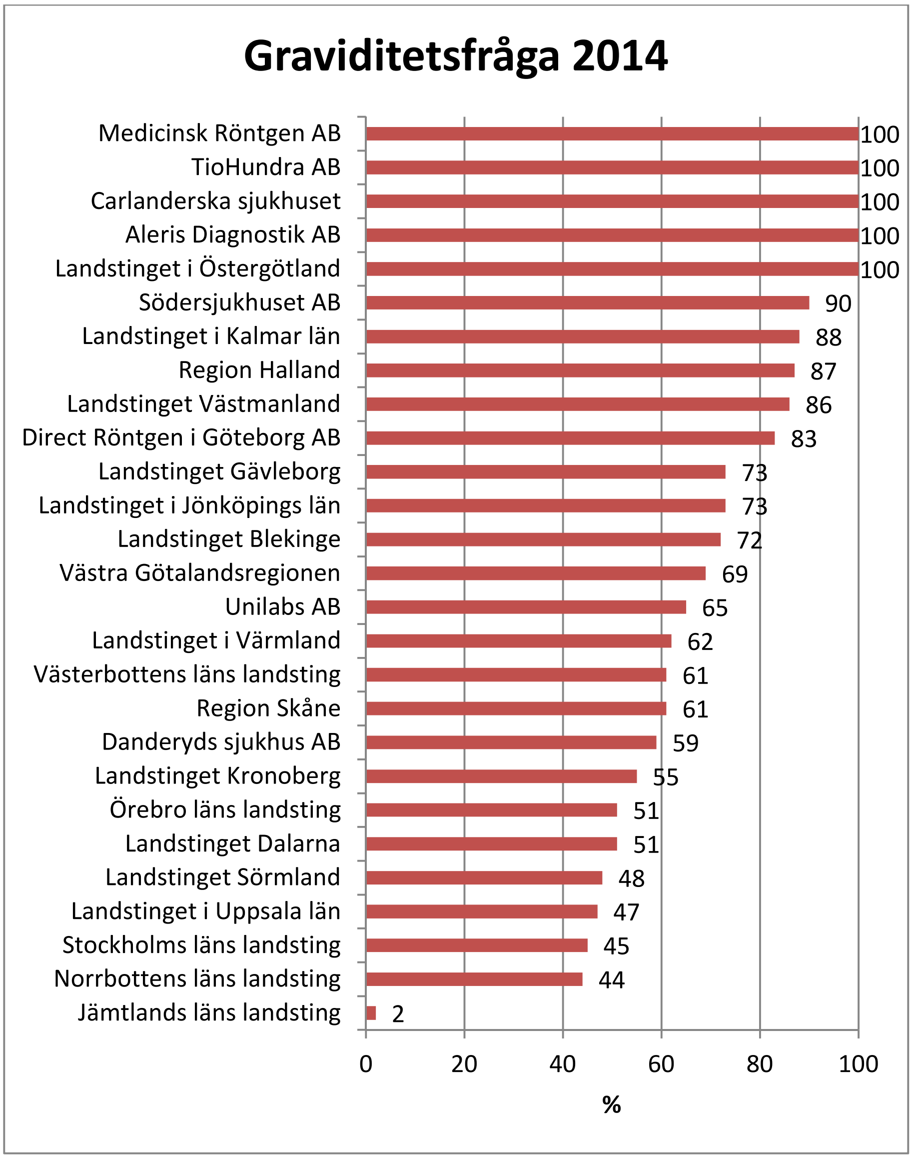
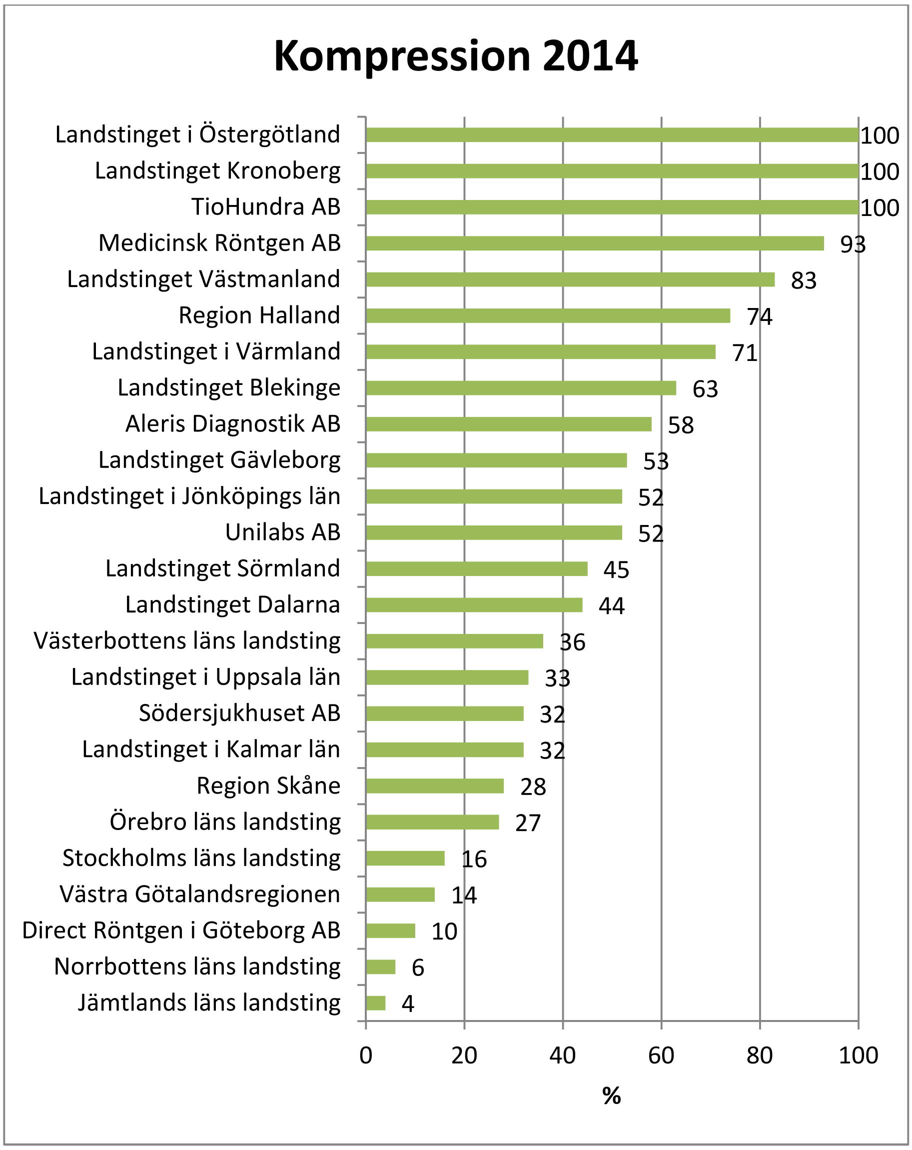
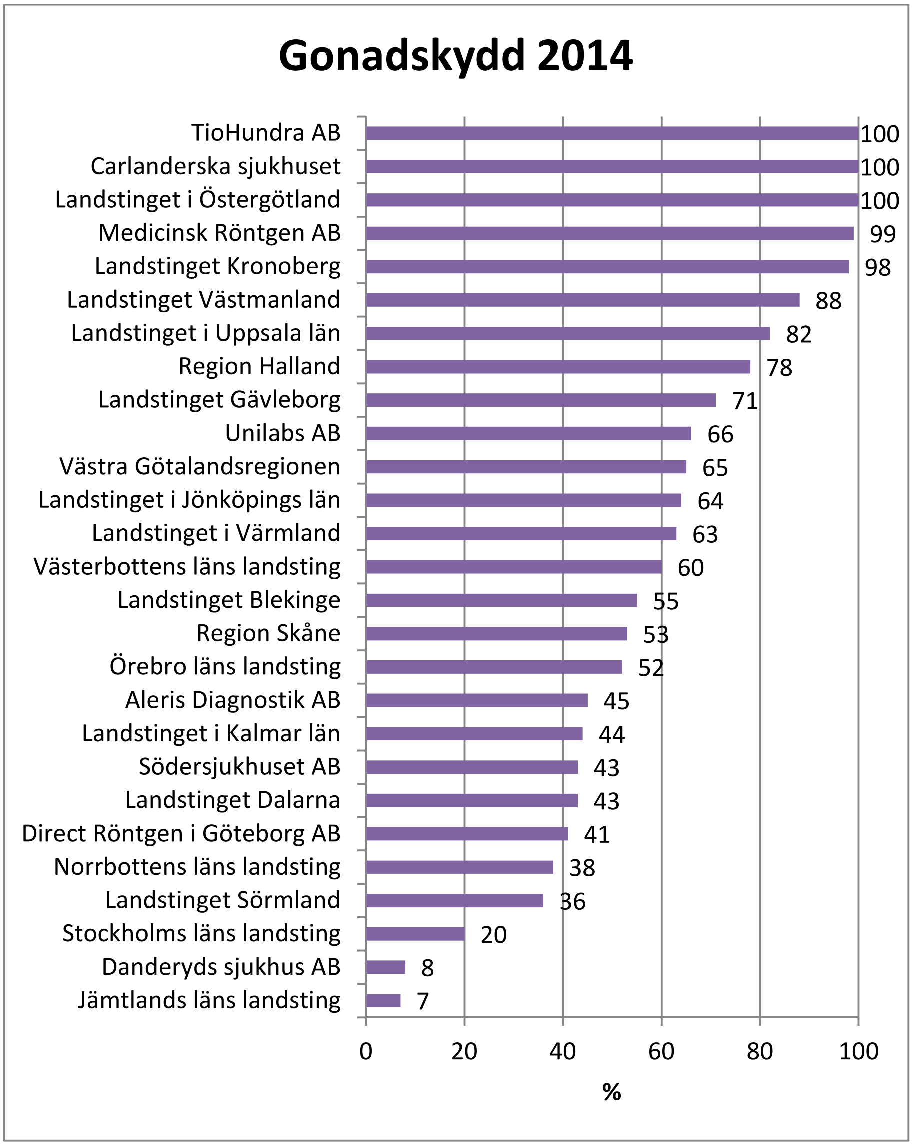


Chart 3 Use of compression



Note that two of the 27 licence holders that had the ability to report on their use of guidelines in 2014 are not included in the evaluation of compliance to guidelines regarding the use of compression. During the evaluation period, these two licence holders had no examinations requiring the use of compression.

Chart 4 Use of lead shielding of gonads



S11-O2

Inspections of x-ray equipment at Danish public hospitals

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During the last 3 years NIRP has been inspecting every single x-ray equipment at Danish public hospitals. This has been done by following one responsible medical physicist at a time. At the inspections, we have been looking at different aspects of radiation protection as well as controls of the equipment. This talk will summarise the general findings as well as some more peculiar observations.

S11-O3

Electronic inspection of industrial radiography companies in Norway

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In Norway, there are at any given time 70-80 companies with a license to perform industrial radiography. Given the companies' geographical spread and the limited staff of the Norwegian Radiation Protection Authority (NRPA), conducting on-site inspections with desired frequency is a formidable task. Therefore, the NRPA is currently testing "electronic inspections" as an addition to on-site inspections. The aim is to increase inspection frequency greatly, with only a modest increase in workload, hopefully raising awareness of radiation protection in the radiography community. The method relies on the radiography companies self-reporting on a number of subjects, put to them in a web-based questionnaire (Easyresearch by Questback). Questions pertain to the companies having written radiation protection procedures, operators having accredited radiation protection certificates, radiation sources being registered in the NRPA source database, etc. In accordance with the goal of keeping workload low, we do not ask to see the actual documents. For the same reason, the questions are predominantly multiple-choice, thus limiting the number of possible answers. This allows for automated analysis of the replies, as we can decide beforehand the answers that are not in compliance with regulations. Seeing as the questionnaire is just now closed for entries, only rudimentary results are available at the time of writing. However, the response rate alone qualifies the electronic inspections as a success. 67 out of 70 companies (96 %) completed the questionnaire. The high response rate can probably be attributed to the fact that we made it clear that the questionnaire had the same regulatory status as an on-site inspection, meaning the companies were obliged to answer under the radiation protection regulation. Further data analysis and follow-up of companies remain, but as a teaser we can reveal that the initial analysis revealed 37 cases of non-compliance with the Norwegian radiation protection regulation, reported from 21 companies. From an NRPA point of view, the method of electronic inspection is so far deemed a success. Putting in a moderate amount of work, we have received a large amount of data. Judging from response rate and informal feedback, the new inspection format also seems to be well received in the radiography community, many citing it as an incitement to review their own practices and procedures.

S11-O4

Inspections in non-medical use of radiation in Finland in 2010-2014

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This study analyzes inspections conducted in 2010 – 2014 by the Finnish Radiation and Nuclear Safety Authority (STUK) in the use of radiation in industry, research, education and servicing, repair and installation in Finland. The total number of inspections made was 1020. In 800 cases STUK issued orders to correct observed non-compliance. This means that only 22 % of the inspections were done without any orders issued. The total number of issued orders was 3345. The most common reason was lack of warning signs for radiation or insufficient additional protective devices in the radiation source. It was also noticed that local rules regarding the preparedness for abnormal events were missing in many places where radiation is used. Also quite many orders concerned the organization itself, such as unspecified duties of the radiation safety officer or insufficient supplementary training of the radiation safety officer. The data collected in 2010-2014 will be used in the future to target more inspections to areas were most of the non-compliance were observed. During the years 2010-2014 STUK has emphasized the preparedness for abnormal events which can now be observed as the number of organizations inspected lacking local rules for abnormal events incidents decreased from 100 in 2010 to 56 in 2014.

S11-O5

New procedures for disposal of ionisation chamber smoke detectors

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Recycling is a hot topic today. Many consumers are aware that waste containing electrical and electronic equipment (WEEE) should be disposed for recycling.

Hence recycling sites receive disposed ionisation chamber smoke detectors (ICSDs). Some of the recycling sites treat the ICSDs as WEEE. In the recycling industry shredding is commonly used, hence there is a risk of the ICSDs being shredded together with WEEE. This could lead to contamination of the shredded material, and in the worst case internal exposure to americium-241 for workers dealing with shredded material. To prevent this, NIRP, in cooperation with DPA-system (Dansk Producent Ansvar/Danish Producer Responsibility) and the Environmental Protection Agency, is presently introducing new procedures for the recycling industry in order to ensure full separation of ICSDs from WEEE. The focus is on guiding the recycling sites and the recycling industry in correct handling, separation from WEEE; especially optical smoke detectors, and disposal of ICSDs. In continuation of the new procedures NIRP will carry out inspections of the relevant sites at the recycling industry.

S11-P1

Regulatory Authority Records from the 2014-2015 Blood Irradiator Inspection Campaign

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Denmark hosts 17 Cs-137 based blood irradiation facilities with a maximum permitted inventory ranging up to 105 TBq. In addition to blood irradiation activities, the facilities are used in research projects. The use of these facilities is regulated through the National Board of Health order no. 985 of 11 July 2007 on the use of sealed radioactive sources. Blood irradiator facilities are mostly associated to hospital blood banks in close proximity to many workplaces and patient treatment areas. The operation of blood irradiators thus requires a high level of safety and security. In 2014/15, inspections at 7 blood irradiation facilities were carried out, resulting in several regulatory required actions and remarks regarding safety and security, including:

- Closure of facilities where practice is considered no longer justified.
- Removal of flammable materials from blood irradiation room.
- Reconstruction of blood irradiation room.
- Update of alarm and warning systems.
- Stressing of procedure for control of access to blood irradiation room.
- Enhanced registration and control of staff access to blood irradiation room.

S11-P2

Norway has phased out gamma based blood irradiators

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The 13 Norwegian gamma based blood irradiators have been phased out or replaced by almost risk-free blood irradiators based on X- ray technology. Acquisition and use of new gamma based blood irradiators is no longer considered justified by the NRPA. The Norwegian phase–out policy was carried out for security reasons and concerns related to possible consequences of a malevolent act. The phase - out process was initiated by the NRPA and strongly supported and further enhanced by the Ministry of Health and Care Services, the owner of the hospitals and the blood banks. Consensus among relevant national authorities and suitable regulations made the phase-out process possible and time efficient. The Breivik 22.7.2011 attack on governmental buildings and the following security report have been an implicit fundamental driving force in this security process.

S11-P3

Results from an All-inclusive IAEA-based Inspection Approach for Industrial Irradiation Facilities

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Denmark hosts three Co-60 based industrial irradiation facilities with a maximum permitted inventory ranging from 7.000 to 69.000 TBq. The use of these facilities is regulated through the National Board of Health order no. 985 of 11 July 2007 on the use of sealed radioactive sources. These facilities host the largest inventory of sealed sources in Denmark, and as such, undergo continuous efforts to maintain and improve safety and security. Annual inspections are conducted on each of the facilities. Since 2011, inspections have been carried out in accordance with IAEA recommendations (IAEA-TECDOC-1526), making use of the Inspection Record template: “Irradiator”. In the early years, focus for the inspections was centered on regulatory approval of contingency plans regarding accidents and incidents. In recent years, the focus of regulatory oversight has shifted to aspects of safety related to normal operations and foreseeable operational occurrences such as: ”shallow source pond water levels” or ”source rack does not descend”. In addition to the annual inspections, specific topical sessions are organized for the industrial irradiator facilities. Such sessions have included meetings between facility staff, regulators and local emergency response units regarding the special requirements for first response operations at industrial irradiator facilities. A separate session on safety and security during source change operations has also been conducted.

S11-P4

NORGIR

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NORGIR is an acronym for Nordic group in industrial radiation and research. The group has radiation experts from all of the five Nordic radiation protection (RP) authorities. Meetings and workshops are held annually or biannually and reports are given to annual meetings of Nordic RP directors. In this poster, some of the agenda items that have been discussed in the previous meetings are listed together with some recommendation to the RP directors.

S11-P5

Survey on needs for changes in the Finnish radiation legislation and on regulatory oversight – The perspectives of practitioners

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This study surveyed needs for changes in the Finnish radiation legislation, and evaluated the regulatory oversight. The total renewal of the Finnish radiation legislation will be done simultaneously with the transposition of the Council directive (2013/59/Euratom) into national legislation before 6th February, 2018. The survey was performed on behalf of the Finnish Radiation and Nuclear Safety Authority (STUK), and it was conducted in collaboration with the School of Management in University of Tampere. The online survey was open in February and March 2015. STUK informed on the survey nationwide on web pages and by sending e-mails and an e-news letter to radiation safety officers, dosimetry services, education and training organizations of radiation safety officers and other interest groups. The survey included also questions concerning the use of non-ionizing radiation (NIR), such as ultrasound and MRI, in healthcare. The data was analyzed by both quantitative and qualitative methods. The total number of answers to the survey was 182. The majority of those (101) represented the use of radiation in healthcare. The second largest group (46) was industry including the use of radiation in industry, research, education and training, product and security control, trade, import and export of radiation equipments and sources, production of radioactive isotopes, installation, manufacture and service of radiation equipment and sources. Most of the answerers were satisfied with the current Finnish radiation legislation. However, many proposals were given, such as improving the definition of an undertaking and regulating qualifications of both radiation protection officers and experts. About one third of the answerers in the use of radiation in industry and one fourth in health care did not find needs for changes in radiation legislation. Most of the answerers both in health care and in industry were as well satisfied with the STUK's regulatory oversight, figures being 65 % and 67 % correspondingly. However, a few proposals to enhance came up e.g. concerning co-operation between different authorities and needs to simplify practical guidance on the safety of radiation. In conclusion, the results provide valuable feedback from users of radiation and other professionals for the radiation legislation renewal in Finland.

S11-P6

Nordic Working Group on Medical Applications

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[3]*Icelandic Radiation Safety Authority*
[4]*Norwegian Radiation Protection Authority*
[5]*Swedish Radiation Safety Authority*

The ‘Nordic Working Group on X-ray Diagnostics’ was founded in 1978 as a cooperation between the radiation protection authorities in all of the five Nordic countries. In the last couple of years, the scope of the working group has been broadened to include also nuclear medicine and radiotherapy. As a consequence, the name of the working was changed in 2014 to ‘Nordic Working Group on Medical Applications’.

The working group meets annually where the focus is national reports on recent activities and discussions on subjects of common interest. Between meetings the group forms a valuable forum for discussions through email, and a few telephone conferences has also been held.

The activities of the group during the last four year in the form of common statements and workshops will be presented as well as the current focus areas.

At the moment, the main focus areas of the group are:

- National implementation of EU-BSS
- Level 2 justification (level of implementation in Nordic countries)
- Referral guidelines
- Regulatory management of proton therapy units
- Recommendations on shielding of thyroid and gonads
- Automatic dose monitoring in x-ray examinations

S11-P7

Focused inspections in Industrial Radiography

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SST/SIS is testing a new IAEA concept (IAEA IR Road Map) in the supervision of Industrial Radiography (IR) companies: Where there used to be a high focus on supervision of the operational part of practice - in the field - we seek the implementation of an IAEA concept to achieve a more complete picture of the conditions of the field - both for individual companies and the industry in Denmark. In addition to the lessons learned from field monitoring, the use of IAEA IR Road Map provide a better picture of Qualification and training, learning from incidents, individual monitoring, workplace monitoring & warning systems, client interface, equipment, internal controls & inspections and emergency preparedness & response. The results are presented in comparison with the IAEA's findings and an average of the Danish companies. The Danish companies can be compared to the responses from 150 companies in 31 countries. The study is currently in progress and results will be presented at the conference.

S12-O1

Indoor and outdoor radon levels in Iceland

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Abstract

We report on a nation-wide survey of indoor radon (Rn-222) in Icelandic homes. The annual mean radon concentration was measured on the ground floor or basement of 250 homes around the island with etch track detectors. Volunteers were sought so the measurement locations were not randomly assigned. Additionally, measurements were made in 32 kindergartens and 19 public swimming pools.

The results show that the radon concentration in Iceland is very low. The mean is 13 Bq/m^3, and the median 9 Bq/m^3. The distribution of the results is heavily biased towards the lower values with a number of the results at or below the minimum detectable activity, 95% of the results below 40 Bq/m^3 and the highest value is 79 Bq/m^3. No appreciable differences were found between the different regions of Iceland except that in the North of the country, slightly higher values were found. Measurements in kindergartens and swimming pools gave even lower values.

These results, which match expectations given what is known about the Icelandic bedrock and from previous spot measurements, imply a mean dose to the population from radon inhalation around 0,2 mSv/year[1]. This value is almost certainly an overestimate, since only ground floors and basements are included in the study.

We have ongoing continuous measurements of radon in outdoor air in Reykjavík by a liquid scintillation system, done in collaboration with the University of Iceland‘s Science Institute. The measurement system is called Autoradon and results from a few months of measuring show that outdoor air in Reykjavík has a radon concentration of about 1-5 Bq/m^3. Simultaneous, continuous measurements of indoor radon in a university building show little or no variation in daily or weekly radon concentration.

Introduction

The bedrock in Iceland is predominantly made of basalt with low levels of uranium. Therefore one should expect low concentrations of radon in air. Measurements of radon in Iceland have mostly been done in the context of geological/geophysical research, for example in an attempt to predict earthquakes. No large surveys have been done previously on indoor radon levels in Iceland but two earlier studies are noteworthy.

The Icelandic and Danish radiation protection institutes made joint measurements in 18 basements around Iceland in 1982 [2]. The results showed low radon levels, 11 Bq/m^3 in Reykjavik (average from 10 measurements), the highest value was 26 Bq/m^3, and many of the measurements where around or under the detection limit.

In 2003 a study was made on a method of radon measurements with liquid scintillator vials [3]. 51 houses in the capital area were measured during the summer. The sample locations were not standardized as the study focused on the measurement method rather than the indoor radon levels. This method measures the average radon level of the previous 12 hours. The results showed again low radon levels, with a mean of 4,7 Bq/m^3 and median 2,8 Bq/m^3.

Because of these results of earlier studies, and the composition of the Icelandic bedrock, radon has not been considered a general health risk in Iceland.

In 2011, IRSA joined in the European Commission's (EC) European Indoor Radon Map (EIRM) project coordinated by the EC's Joint Research Centre (JRC), so in 2012 - 2013 a nationwide survey on indoor radon concertation was conducted [1].

Methods

Sampling period

IRSA acquired 500 etch track detectors from Radosys in Hungary (PADC/CR-39 detecor chip). These detectors are usually exposed for 3 - 6 months sampling periods with a lower limit of detection (LLD) at about 15-30 Bq/m^3.The LLD is approximately inversely related to exposure time, but longer exposure time can cause increased uncertainty due to the plastic hardening.

At the request of IRSA, Radosys investigated the detector response curve for a 12 month exposure to a low level radon concentration. With the resulting correction factor, IRSA was assured that the uncertainty would be under 15% for a 12 month exposure at a reference concentration of 150 Bq/m^3, but due to the increased sample period a LLD of approximately 7 Bq/m^3 was achieved.

Sampling locations

Volunteers were sought passively through a website and actively by phone, where needed. Attempt were made to distribute sample points as broadly as possible while tracing the population density, but the sampling location were not random. In all, detectors were sent to 278 homes (of which 250 were retrieved) across the country. Measurements were also made at two kind of work places: kindergartens and public swimming pools. The kindergartens were selected due to being most often in a single-floor in building and with high occupancy. Swimming pools where of interest because of the large volume of water (in some places geothermal water) used. Detectors were sent to 31 kindergartens and 30 swimming pools (31 and 19 respectively were retrieved).A map of all the sampling locations is in Figure 1.

Volunteer instructions

Participants were instructed to place the detectors on the building's lowest floor and in an inhabited room (preferably a bedroom). In 2012 there were around 129.000 homes in Iceland [4], so the survey included 0,2% of all homes. The detectors were sent to volunteers over a 3 month period and collected about year later so the sampling time was around 9-13 months. The detectors were then sent back to the manufacturer for reading, in a vacuum sealed plastic package (to prevent a radon contamination during the travel from Iceland). Half of the measurements were in the capital area (49%) but most of the populated areas in Iceland were covered (Figure 1).

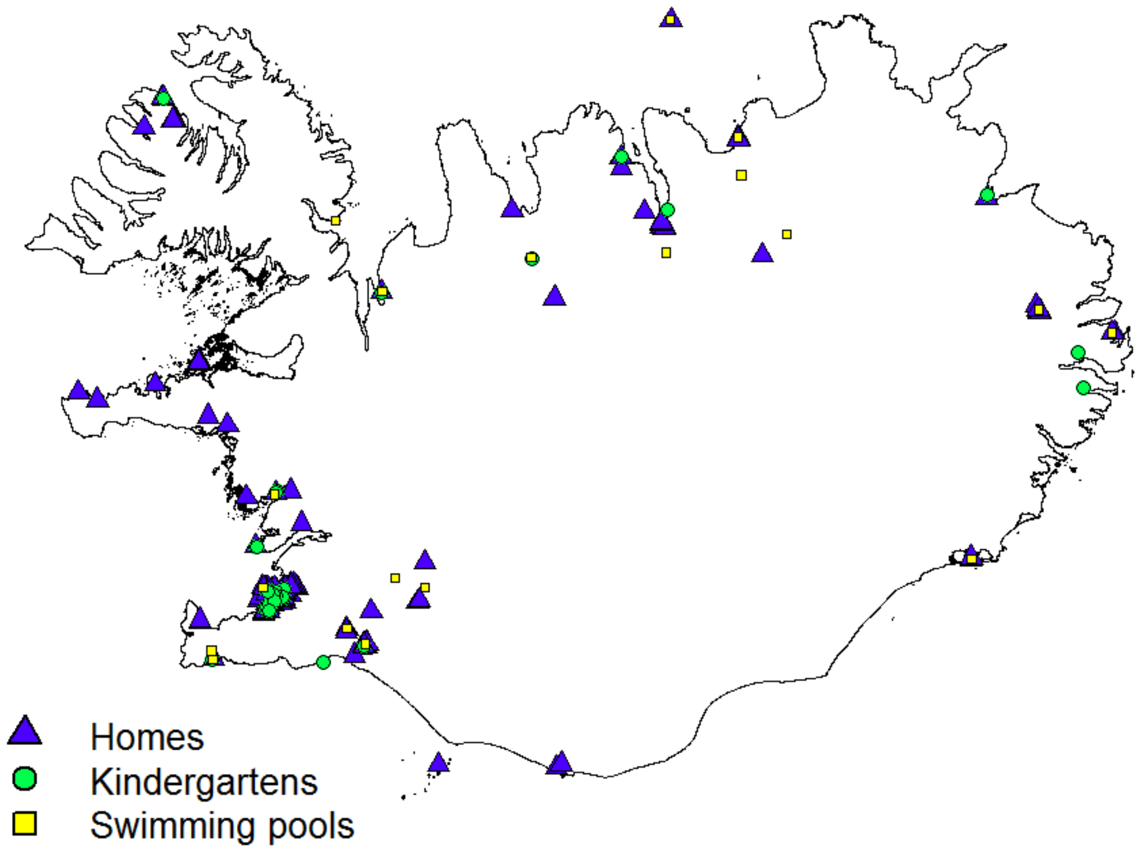

Figure 7. Distribution of etch track detectors around Iceland.

Continuous measurements

In collaboration with the Science Institute of University of Iceland, IRSA made continuous radon measurements in a basement in a university building (Raunvísindastofnun) and on outdoor air. The measurement system used is called Autoradon. It has 15 ml liquid scintillator fluid continuously pumped with the sampled air. The system is inside a 5 cm lead shield and the measurements are done with 214-Po counting [5]. The background is 6 counts/24 hour which is equivalent to a mean concentration of 0,4 Bq/m^3 over 24 hours. The LLD of the system is 0,9 Bq/m^3 for 24 hour measurements. Two system were running simultaneously, one measuring indoor radon level in the basement and the other measuring the nearby outdoor radon levels at 1 meter height. The building is nearby the coast. The Autoradon system automatically records a measurements once every hour so it could capture diurnal variation, but the radon concentration results were averaged over whole days (24 hours).

Results

Etch track detectors

As suspected, radon levels in Iceland are found to be very low by each measurement. From the etch track detectors in homes, the mean radon levels is 13 Bq/m^3 and the median 9 Bq/m^3. Only 5 % of the results are over 40 Bq/m^3 and the highest measurement was 79 Bq/m^3. Many of the etch track detector results are at or around the detection limit so the mean and the median reported here are likely overestimates. Distributions of measurements is shown in Figure 2.

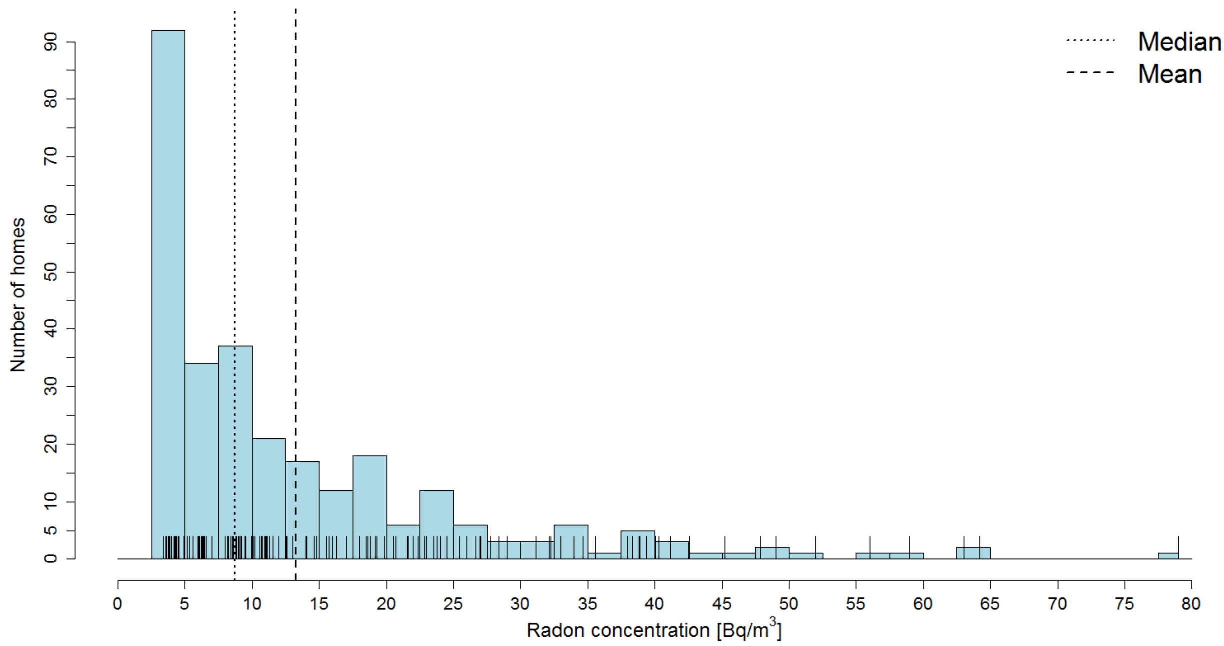

Figure 8. A histogram of radon concentration in 250 Icelandic homes. The short lines at the x-axis represent individual measurements.

The measurements in kindergartens and swimming pools gave even lower results, perhaps due to better air-conditioning. For kindergarten the mean radon levels is 11 Bq/m^3 and the median 6 Bq/m^3 and for public swimming pools the mean radon levels is 6 Bq/m^3 and the median 5 Bq/m^3. Comparison between these measurements can be seen in Figure 3.

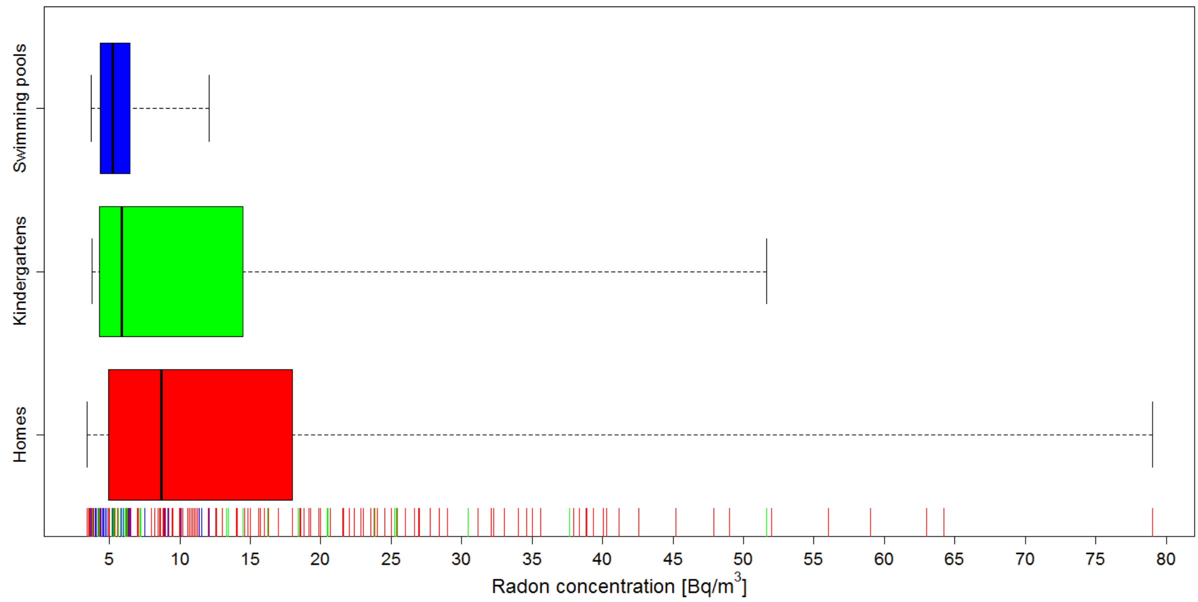

Figure 9. A box plot with whiskers of the radon concentration results divided into location type: homes (Red), kindergartens (green), swimming pools (blue). The short lines at the x-axis represent individual measurements.

Comparison between different parts of the country (Figure 4) shows similarly low levels. Slightly higher radon levels are seen in the northern part of Iceland but the variability between different parts of the country is much lower than variability of the measurements within each part.

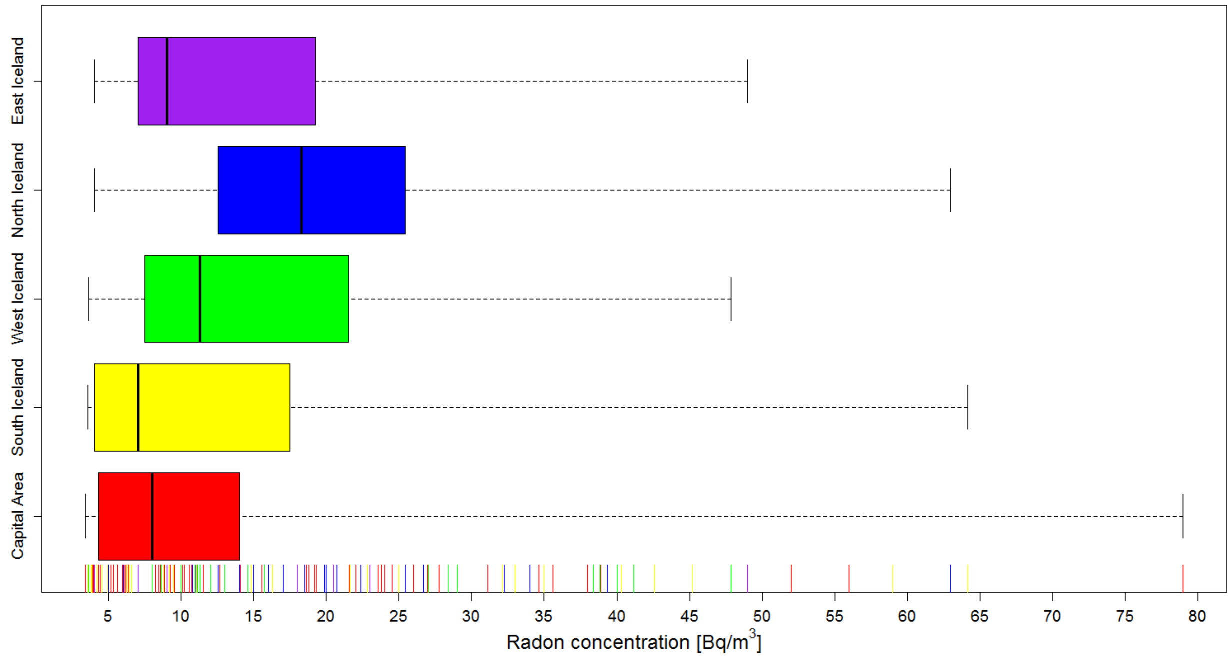

Figure 10. A box plot with whiskers showing the radon concentration, by region: Capital area (red), South (yellow), West (green), North (blue), East (purple). Horizontal lines in the boxes represent the median values. The short lines at the x-axis represen individual measurements.

<u>Autoradon measurements</u>

The radon levels measured by the Autoradon system are very low as expected. From early May to mid-July 2015 the average concentration is 1,6 Bq/m^3 for outdoor air and 3,6 Bq/m^3 for indoor air. Some of the outdoor measurements are close or below the LLD. The difference between the indoor and outdoor measurement is small (see Figure 5). This probably results from generally low levels in both indoor and outdoor air and from good air-conditioning of the University building.

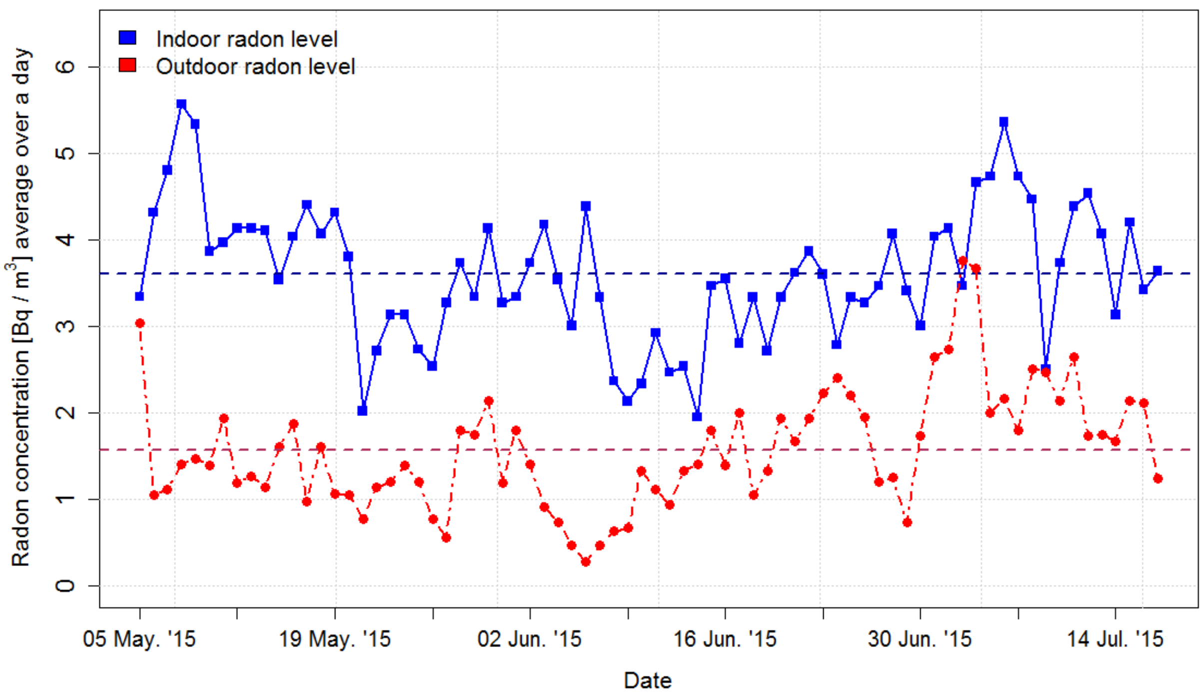

Figure 11. Results from the Autoradon system. Indoor air is from a basement lab at the Science institute. Daily average of radon concentration is plotted against the date. The dotted lines shows the average over the time period.

The Autoradon system that measures the indoor air was moved to a very small storage room under the stairway in the same building. It is used for storing cleaning supplies and poorly ventilated. In this room higher radon concentration was measured, up to 35 Bq/m^3. We did not see any noticeable variation within the day or week in the data from May to July and the data series is not long enough to show any seasonal variation.

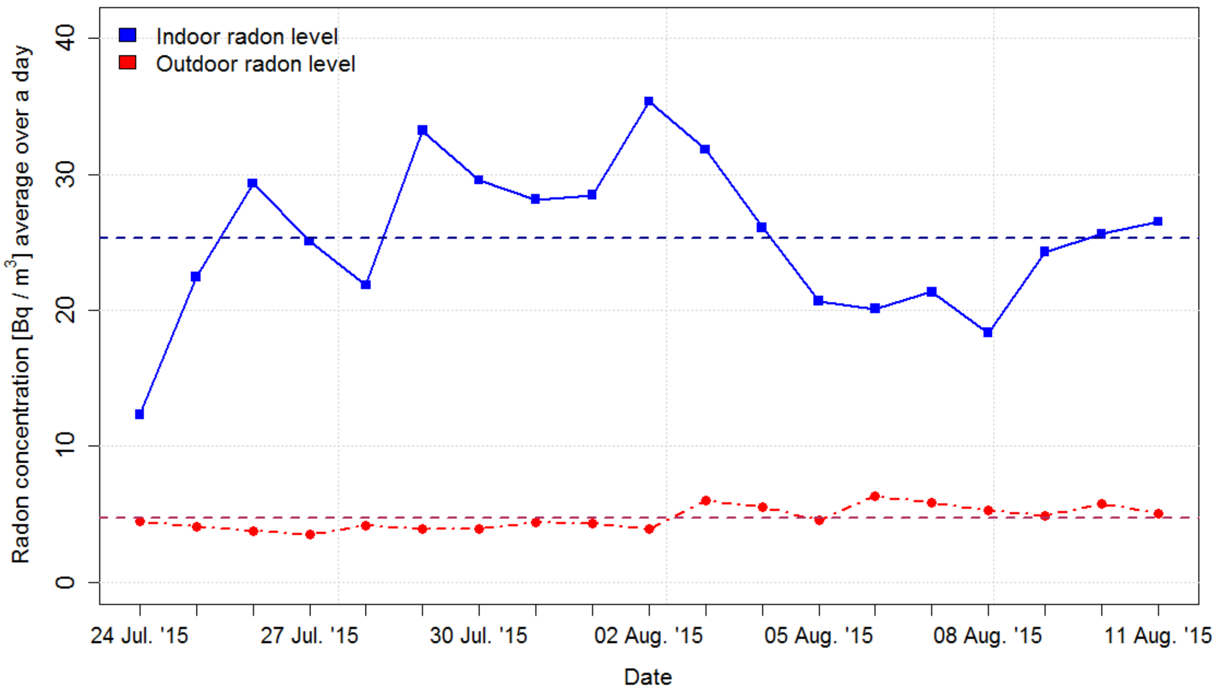

Figure 12. Results from the Autoradon system. Indoor air is from a small storage room (not ventilated) in the basement at the Science institute. Daily average of radon concentration is plotted against the date. The dotted lines shows the average over the time period

Conclusion

As expected, the radon concentrations are very low in Iceland, as both the etch track detectors and the Autoradon system showed. Indoor levels of radon in countries worldwide can range from 10 Bq/m^3 to more than 100 Bq/m^3 [6], so Iceland is among the countries that have the lowest radon concentrations. It should be noted that our measurements only consist of measurement of 1st floor and basement, also that many of the results were around the detection limit.

Based on the results from the etch detectors, the dose to the Icelandic population from radon inhalation has been estimated to be 0,2 mSv/year. This is most likely an overestimation since many of the results from the etch track detectors are around or below the lower limit of detection.

Nowhere did our measurements exceed reference levels of radon that our neighboring countries have (from 100-300 Bq/m^3), not even in a small non ventilated supply room in the basement of the University of Iceland.

Outdoor radon levels in Iceland are low as one would expect for an island with a bedrock with low radioactivity. The measuring place for the outdoor air is probably affected by its closeness to the sea. One might suspect a little higher radon levels measured farther away from the coast.

We can conclude that radon is not a health concern in Icelandic homes.

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S12-O2

NORM in Norwegian Mineral Industry

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Due to new waste and discharge regulations in Norway (from 2011) many companies using minerals in their production process are now bound to apply for discharge permits for radioactive substances. The discharge permits require both that discharges be quantified and that environmental surveys be conducted. This is a new and unknown issue for most mineral companies and there are little Norwegian comparable data available for mainland discharges or background specific activities in for example fjords. This paper will present a case study from a mineral company in western Norway, with discharges to a fjord and to air. Results from discharge analysis and the environmental survey will be presented and discussed. Also other aspects of the consequences of the process, such as doses to personnel, the public and the environment will be looked into.

S12-O3

TENORM in geothermal applications in Iceland

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Abstract

Sulphide scales in geothermal applications in Iceland are rare and mostly confined to a geothermal area in the Reykjanes peninsula in SW-Iceland. This paper reports the existence of technically enhanced levels of naturally occurring radioactive material, TENORM, in hard sulphide scales from two wells in this area. This is the first time that TENORM above the exemption limit of 1 Bq/g has been discovered in Iceland.

Introduction

An exemption/clearance limit for naturally occurring radionuclides in solid materials in secular equilibrium with their progeny is 1 Bq/g, according to Table A, part 2 of the EU Basic Safety Standard (EU BSS) [1]. This applies to the U-238 decay chain of radionuclides which includes Ra-226, Rn-222, Po-210 and Pb-210 as members.

This paper reports that hard sulphide scales in geothermal surface pipes in the Reykjanes peninsula contain TENORM from the U-238 chain, above this exemption limit. This was not expected since levels of natural radioactivity are very low in Iceland.

Natural radioactivity in Iceland

A very low level of natural radioactivity in Iceland is confirmed by various studies, including a recent nationwide survey of indoor radon which reports Radon Rn-222 concentrations with a mean of 13 Bq/m^3 and a median of 9 Bq/m^3. These values are among the lowest in the world [2]. Ambient gamma, terrestrial and cosmic radiation in Iceland is also low, around 50 nSv/h, verified for instance by measurements from four continuously monitoring radiation meters in the south-western, south-eastern, north-western and north-eastern corners of Iceland, available at the web-site of IRSA. The total radiation exposure in Iceland from natural sources has been estimated to be in the region of only 1 mSv/year [3].

Iceland has no oil and gas industry or other industries that typically produce TENORM as a by-product. The country has a number of geothermal plants that have been observed to produce TENORM in other countries, such as USA, associated with geothermal applications, i.e. in scales in pipes close to wells (boreholes). In Iceland, however, there is generally very little scaling in geothermal pipes and TENORM has not previously been detected.

Reykjanes geothermal field

The Reykjanes geothermal system is located at the tip of the Reykjanes peninsula, 50 km southwest of the capital Reykjavík (Figure 1). Thirty-four wells have been drilled in the area which is mainly composed of hyaloclastite rocks interbedded with lava flows of tholeiitic composition and pillow basalts at a greater depth [4], [5].

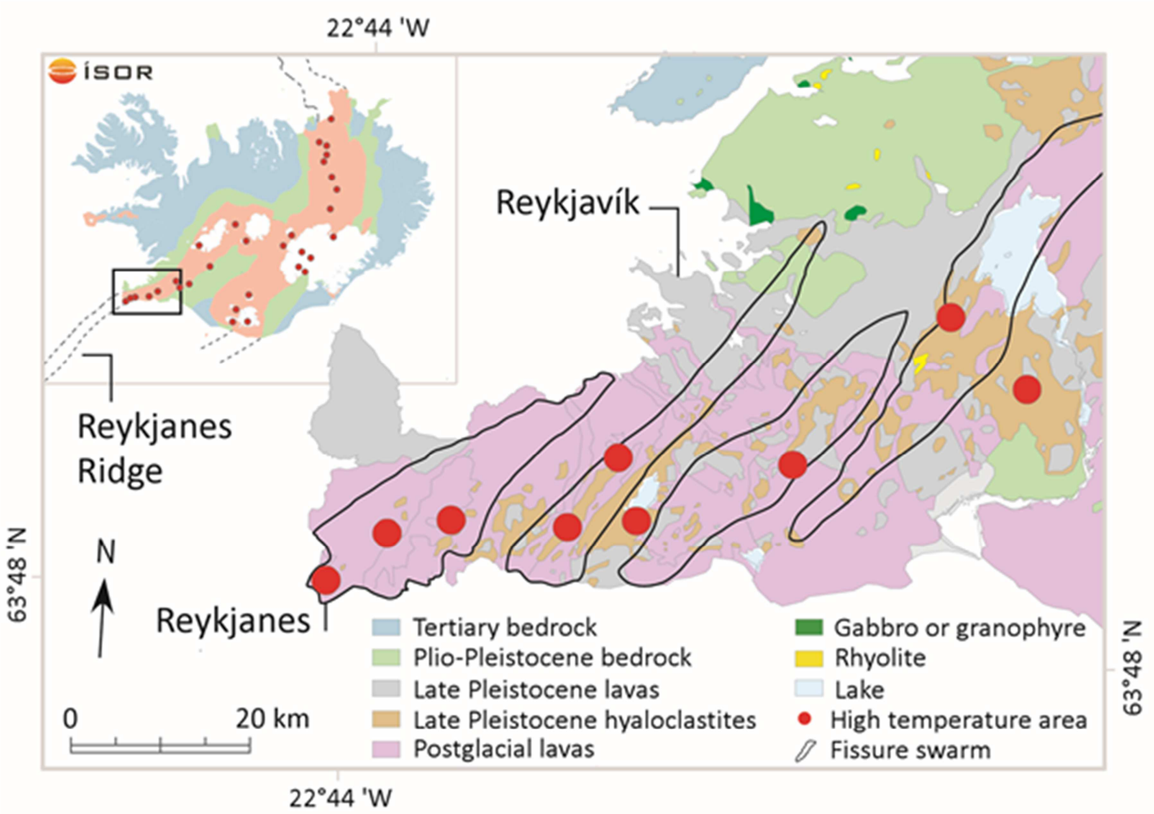

Figure 1. Geology map of Reykjanes Peninsula. Iceland is in the insert with the location of the Reykjanes Peninsula outlined. The high-temperature area, Reykjanes, is furthest to the west [6].

The Reykjanes reservoir fluid, with seawater salinity, has reacted with the basaltic host rock at elevated temperatures between 270-340°C. The concentrations of metals and trace elements in the reservoir fluid collected at 1350–1500 m depth are; Cu ~15 mg/kg, Zn 5-25 mg/kg, Fe 10-140 mg/kg, Mn ~2.5 mg/kg, Pb 100-300 µg/kg, Ag 30-100 µg/kg and Au 1-6 µg/kg [7]. The chemical properties of the reservoir fluid have long been studied for the purpose of producing salt and valuable minerals. Commercial attempts at doing that have however been unsuccessful while electrical power production has proved to be viable. The geothermal fluid has been utilized since 2006 by HS Orka in a 100 MW electricity power plant, Reykjanesvirkjun.

Figure 2. Scale precipitated in a pipe from well RN-22. The scale is formed in one year from 2006-2007. An orange has been placed for size comparison [6].

As the fluid ascends from a well it starts to boil due to pressure decrease resulting in precipitations of mainly sphalerite ((Zn,Fe)S) up the well from about -1200 m depth. At surface downstream of an orifice plate, a sudden decrease in pressure from ~37 bar to 22 bar causes rapid boiling (flashing) which results in abundant precipitation of sulphides as shown in Figure 2. In order to better understand the build-up of scales and the resulting decrease in fluid-flow in the surface pipeline, a detailed geochemical and mineralogy study has been done on the scales by Hardardottir [6], and her co-workers [8]. The scales are primarily composed of sphalerite, with some chalcopyrite ($CuFeS_2$), small amounts of galena (PbS) and traces of other sulphides (mainly Cu-sulphides like bornite (Cu_5FeS_4) and digenite (Cu_9S_5) [6].

TENORM in scales

The scales emit beta radiation which can be detected with hand held meters. Some alpha radiation can also be detected. Only very weak gamma radiation is present but it does however enable identification of Pb-210 and Po-210 nuclides through gamma-spectrometry in a laboratory as can be seen in Figure 3.

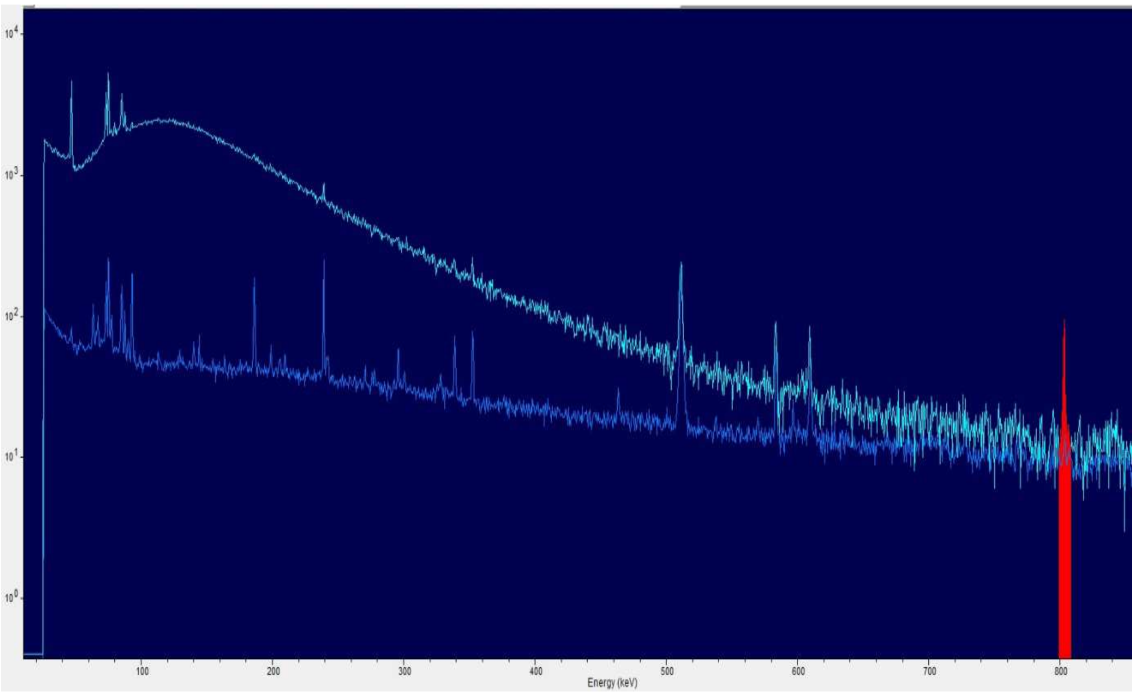

Figure 3. A gamma-spectrum emitted by the scaling. A peak for Po-210 is highlighted in red. A low voltage peak at 47 keV due to Pb-210 is also visible. A background spectrum without these peaks is shown for comparison.

Analyses of gamma-spectra as in Figure 3 show the presence of nuclides from the natural U-238 chain (Figure 4).

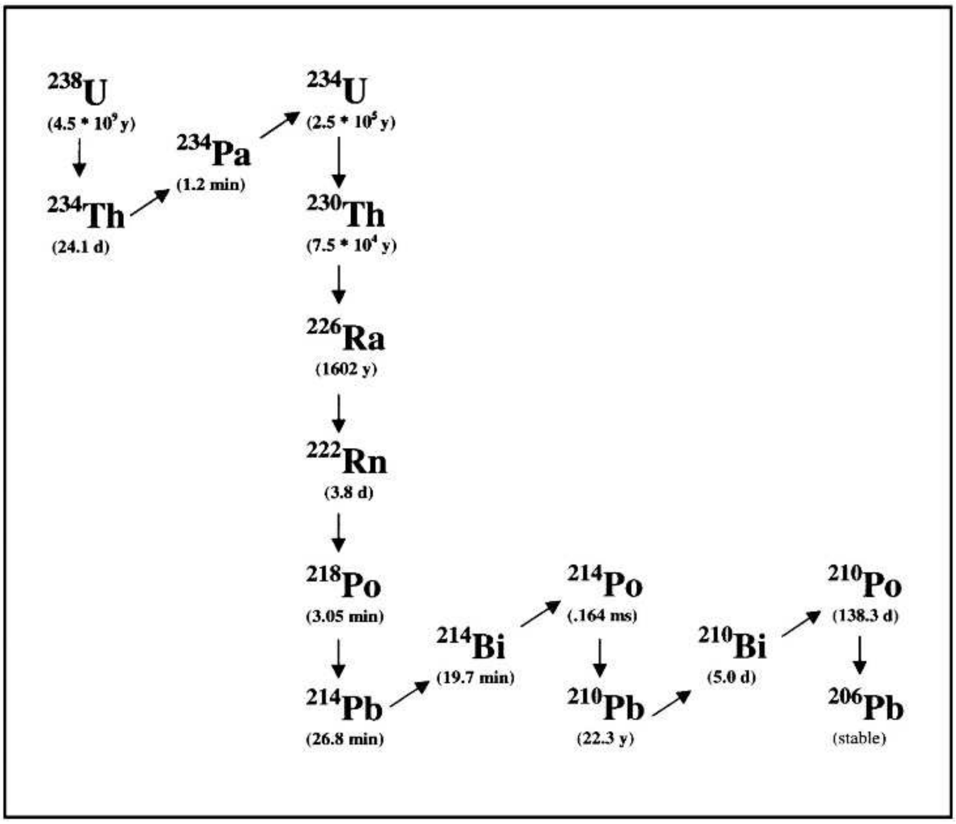

Figure 4. The U-238 decay chain. The radionuclides Pb-210 and Po-210 originate from radium (through radon). The half-life of each nuclide is in parentheses.

Samples from scales originating in surface pipes from two wells (boreholes) were obtained in June 2015 for accredited analysis at STUK Environmental Radiation

Surveillance and Emergency Preparedness, an accredited testing laboratory. A report was prepared, dated 31st July (see Appendix). These wells are currently the most productive wells of the power plant Reykjanesvirkjun in terms of electricity and in amount of scaling. The analysis showed the following results for these samples:

Table 1: Activity in scales from two wells when measured on 14th and 15th July 2015. The uncertainty of results (2 sigma) indicates that the results are, with a 95% confidence interval, within the given limit values.

Radio-nuclide	**Well RN-12 Bq/g**	**Well RN-11 Bq/g**
Ra-226	< 0.002	< 0.002
Pb-210	33.7 ± 4.1	51.5 ± 9.3
Po-210	123.0 ± 20.0	214.0 ± 30.0

Discussion

The existence of TENORM in Iceland has now been confirmed for the first time. The volume of geothermal scales produced each year through cleaning of the geothermal pipes is estimated by the operator in 2015 to be a few metric tons and less than one cubic meter. In 2007, it was estimated to be 4 tons [6]. Values for comparison from geothermal plants in other countries are few in number and therefore this paper presents the following information on the more common oil and gas facilities. A report from IAEA on radiation protection and management of TENORM in the oil and gas industry [9] states that a “production facility may generate quantities of scales and sludge ranging from less than 1 t/a to more than 10 t/a, depending on its size” (page 69). In the IAEA report (page 56) the following table is presented (see Table 2):

Table 2: Concentrations of naturally occurring radioactive material in oil, gas and by-products. This table is given in reference [9] as Table no. III.

Radio-nuclide	Crude oil Bq/g	Natural gas Bq/m^3	Produced water Bq/L	Hard scale Bq/g	Sludge Bq/g
U-238	0.000 000 1–0.01		0.0003–0.1	0.001–0.5	0.005–0.01
Ra-226	0.0001–0.04		0.002–1200	0.1–15 000	0.05–800
Po-210	0–0.01	0.002–0.08		0.02–1.5	0.004–160
Pb-210		0.005–0.02	0.05–190	0.02–75	0.1–1300
Rn-222		5–200 000			
Th-232	0.000 03–0.002		0.0003–0.001	0.001–0.002	0.002–0.01
Ra-228			0.3–180	0.05–2800	0.5–50
Ra-224			0.5–40		

The values for the Reykjanes scales seem to be higher than those in the oil and gas industry with regard to the activity of Po-210. These atypical values and the absence of Ra-226 make the processes that made these scales very interesting from a scientific point of view.

NORM and TENORM is included in the Icelandic Act on Radiation Protection and any potential radiation protection issues associated with TENORM in geothermal applications will be addressed as needed by the Icelandic Radiation Safety Authority.

The author expresses his gratitude to the recently deceased Seppo Klemola at STUK who first provided him with an estimate of a Po-210 activity above 10 Bq/g in a scaling sample from Reykjanes. This paper is dedicated to his memory.

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S12-P1

NKS: Developing Methods for Reliable and Efficient Radiological Characterization of NORM Contaminated Objects

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As a bi-product of the extraction process and pipe transport of oil and gas in the North Sea, NORM materials accumulate as scale deposits on the inside of piping. During maintenance operations and decommissioning of oil and gas production facilities, piping and other equipment with potential NORM contamination is routinely characterized, in order to ensure allocation to the proper waste stream. However, routine inspections by RPOs on North Sea platforms or radiation portal monitor alarms in scrapyards show that a review and update of the present monitoring program is warranted. Erroneous characterization of NORM contaminated components represents a potential health hazard and a financial liability risk. Consequently, consensus has developed between radiation protection authorities and industry to optimize existing procedures or develop new procedures for reliable characterization of potentially NORM contaminated equipment. Swedish, Norwegian and Danish radiation protection authorities have entered collaboration with DTU NUTECH and two major Danish oil and gas operators to define the project CONCORE (Characterisation of NORM Contaminated Objects: Reliable & Efficient). NKS committed 400.000 DKK in 2014 and 363.000 DKK in 2015 for the realization of this project. The characterization procedure to be developed during this project may potentially find use in other industries such as paper production or nuclear facilities, where non-standard methods for radiological characterization may be required.

S12-P2

The Swedish Radiation Safety Authority's Radioanalytical Laboratory: who are we and what do we do?

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The radio analytical laboratory (RAL) at the Swedish Radiation Safety Authority (SSM) consists of four scientists and engineers with a broad competence in the field of radio analytics and radio physics. RAL's instrument park consists of five HPGe detectors, two low-background LSCs, two alpha spectrometry systems with a total of 18 chambers, and a whole-body counting facility with a thyroid measuring station. In addition to the instrument park, there are two radiochemical laboratories for sample preparation, radiochemical separation, and source preparation. While none of RAL's methods has ISO 17025 accreditation, the ambition is that all work is carried out according to this standard. RAL provides support to several units within SSM that work with supervision. For example, the laboratory analyzes samples gathered from clearance inspections of buildings and lands, from evaluations of radioactive ashes and their need to be deposited, and also samples from general investigations of industries with the possible enrichment of NORM. RAL also analyzes samples for the national environmental monitoring program, a program that was created because of requirements set up by national and international organs, such as the EU. Furthermore, RAL performs intercomparison tests with the Swedish nuclear power plants' laboratories, both on environmental samples from their monitoring programs and on discharged water samples from the power plants. The average yearly throughput in the laboratory is 600 samples and about 35 people are measured with the whole-body counter every year. RAL is part of the national network of laboratories responding to nuclear and radiological emergencies. The laboratory has a coordinating role in many of the activities of this network and regularly arranges intercomparison exercises for the other laboratories. RAL is also an active member of the IAEA network ALMERA, which is a cooperative effort of IAEA to establish a network that can provide reliable analyses of environmental samples in the event of an accidental or intentional release of radioactivity. RAL's research activities comprise the areas of radiometric measurement development, radio analytical methods, marine radioecology, retrospective dosimetry, internal dosimetry, and in vivo measurements techniques. Additionally, the laboratory takes an active part in the supervision of PhD and Master Students. Our research partners are Linköping University, IAEA, NRPA, EURADOS, ANKA, and CNA.

S12-P3

Gross alpha and beta radioactivity levels measurement in mining ponds in Jos Metropolis-Plateau State, Nigeria

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Twenty (20) water samples from the mining ponds were collected from Jos in which all the samples were drawn in two litres plastic containers. Few drops of nitric acid were added for preservation and after the total dissolved solids had been estimated, the samples were evaporated then transferred into planchet for counting. The counting were done using MPC-2000 Dual Phosphor detector. The gross alpha activity from this mining pond ranged from (0.006±0.002 – 0.144 ± 0.003) Bq/L with a mean value of 0.0382±0.007 Bq/L. The gross beta activity concentration ranged from (0.006±0.330 – 11.319±0.519) Bq/L with a mean value of 1.72±0.356 Bq/L. When these values are compared with the International Standard, all the values obtained for gross alpha were within the acceptable limit while for the gross beta, nine of these values were above the acceptable limit.

The committed effective dose for gross alpha ranged from (0.002 – 0.030) mSv/yr for adults while the values for children ranged from 0.18 – 3.55) mSv/yr. In comparison with WHO Standard of 0.1 mSv/yr for the general public some the ponds may be highly radioactive to children than adults. This suggests that these sources of water are radioactive to human health and therefore care must be taken when using the water from these nine sources.

KEYWORDS: Gross Alpha, Gross Beta, Radioactivity, Effective Dose and Activity Concentration

INTRODUCTION

Nature has provided immense natural resources to humanity. Water is one such natural resource which is essential for human life and for health and the environment. The first health requirement for any developing country is the abundance of clean water supply. Water the most indispensable and precious natural resources are expected to be free from pollution. It has two parameters which are closely linked, quantity and quality. In nature, all water contains some impurities. Water quality is one of the highest priority environmental issues. The water resources of a nation determines the food production, industrial growth, public health and hence its economy (Marbanaing, 2011).

Natural radioactivity is always present in the environment. Water especially ground water, is not free of radioactive isotopes from naturally decaying series of ^{238}U, ^{232}Th and ^{40}K. It is natural to find radionuclides in drinking water. They get into water as it comes in contact with radioactive materials in the solids. The activity concentrations of natural radionuclides in ground water are connected to the activity concentrations of ^{238}U, ^{232}Th and their decay products in the ground water and bedrock (Vesterbacka, 2007). This is due to ground water reacting with the ground water and bedrock and releasing quantities of dissolved components that depend on the mineralogical and geochemical composition of the soil and rock, redox conditions and the residence time of ground water in the soil and bedrock (Fasae, 2013).

Clean and plentiful water provides the foundation for prosperous communities. We rely on clean water to survive, yet right now we are heading towards a water crisis. Dirty water is the world's biggest health risk and continues to threaten both quality of life and public health in the developing countries like Nigeria. When water from rain and melting snow runs off roofs and roads into our rivers, it picks up toxic chemicals, dirt, trash and disease-carrying organisms along the way which may increase its radioactive content. Many of our water resources also lack basic protections thereby making them vulnerable to pollution from factory farms, industrial plants to mention but a few. The aforementioned can lead to drinking contamination of water bodies which is most likely to come from naturally occurring radionuclides which are potential contributors of alpha, beta and gamma radioactivity in underground water supply system (Onoja, 2005). Radioactivity in drinking water is principally derived from leaching of radionuclides from rocks and soils. Dissolving from underground minerals and move along with the water as well as deposition of radionuclides from the atmosphere. On the other hand (Forte *et al*, 2006) established that, the geographical/geological formation of an area determines to some extent the radionuclides present in water. Wells and boreholes constricted in bedrocks within such areas could show some levels of natural radioactivity. Enhanced levels of Uranium, Thorium and their daughters' products might be present in waters that are rich in natural radioactivity. Uranium isotopes (^{238}U, ^{234}U and ^{235}U) have a non-negligible radiotoxicity (WHO, 1978; Malcome-Lawes, 1979). Furthermore, several radionuclides in the radioactive decay chain starting from ^{238}U, ^{235}U are highly radiotoxic.

According to Ahmed, 2004 the most radiotoxic and the most important among such radionuclides is radium, which is a known carcinogen and exists in several isotopic forms. The predominant radium isotopes in groundwater are ^{238}Ra, an alpha emitter with a half-life of 1600 years and beta emitters, with a half-life of 5.8 years (Lyenger, 1990; Marovic *et al*, 1996; Sidhu and Breithart, 1998). Considering the high radiotoxicity of alpha and beta emitters their presence in water and their associated health risks it is glaring that they require proper attention. It is known that even small amounts of radioactive substance may produce a damaging biological effect and that ingested and inhaled radiation can be a serious health risk (Rowland, 1993). When radium is taken into the body, its metabolic behavior is similar to that of calcium and an appreciable fraction is deposited in the bone while the remaining fraction being distributed almost uniformly in soft tissues (Wrenn, *et al*, 1985). An important aspect of radium protection is the prevention of its entry into the human body, the critical pathway being ingestion through the food chain or drinking water (Kahlos and Asikainen, 1980).

Materials and Method

Twenty (20) water samples were collected from the mining ponds in Jos metropolis of Plateau State- Nigeria. The samples were collected in 2 litres plastic containers with about 1% air space left for thermal expansion. To minimize contamination, the containers were first rinsed three times with samples water before use. The water samples were immediately acidified with 20ml ± 1ml of nitric acid per litre of sample collected to minimize absorption of radioactivity into the walls of the containers (ISO, 9697 & 9698; 1992a). The samples were then tightly covered and kept in the laboratory. For the purposes of analysis the samples were slowly evaporated without boiling down to a volume of 50ml at a furnace temperature of 60^0C. The residue was then transferred to a stainless steel planchet, dried and allowed to equilibrate with ambient temperature and weighed. The counting time was 3000s.

Results and Discussion

The results are as shown in Table 1. The gross alpha activity in the mining ponds water ranged from (0.006 ± 0.002)Bq/L for TMP09 to TMP02 where the activity concentration was (0.144 ± 0.003)Bq/L with a mean value of (0.0382 ± 0.0007)Bq/L. The gross beta activity concentration ranged from (0.355 ± 0.330)Bq/L from TMP11 to (11.319 ± 0.519)Bq/L for TMP09 with mean value (1.721 ± 0.356)Bq/L.

The alpha activity concentrations in all the ponds water samples examined were within the practical screening level of 0.1Bq/L except for samples collected from TMP06 and TMP09 which are slightly above this value. The gross beta activity values for eighteen (18) out of the twenty mining ponds analyzed were above the recommended WHO acceptable limit of 1.0 Bq/L. In other words, all of the twenty (20) mining ponds analyzed only TMP01 and TMP03 were within the screening level. These high levels of gross beta activity may suggest the presence of pollutants of anthropogenic origin as screening for beta activities in the environment is screening for artificial or anthropogenic radionuclides (Ezekiel, *et al*, 2013). Such pollutants in the tin mining areas could be the mines tailings (which are made up of mainly zircon and monazite) which are usually rich in Uranium and Thorium.

The committed effective dose gives a good approximation of the effective dose rate of our bodies and is a function of the quality of water consumed for year among other factors. It is assumed that on the average adults consume about two litres of water per day which is equivalent to 73L/yr, while children consume about 200L/yr (Fernandez *et al*, 1992, WHO 2004, Fasae, 2013). The committed quantities because of small effective half-lives are practically realized within one year after intake (Fasae, 2013). In this work the committed effective dose (CED) over one year was calculated using the formula given by Fasae, 2013 as:

$$CED = I\ A\ C \times 365 \tag{1}$$

Where I = the daily water consumption, A = the alpha activity concentration in Bq/L and C = is the dose conversion factor for ingestion and for an adult the value is 2.8×10^{-4}mSv/Bq while for children it is 1.5×10^{-3} mSv/Bq. For a given gross alpha and beta the values for the estimated effective dose for gross alpha ranged from (0.002 – 0.030)mSv/yr with a mean value of 0.009mSv/yr for an adult while the values for children ranged from (0.18 – 3.55) mSv/yr with a mean value of 0.624mSv/yr. When these values are compared with the WHO acceptable limit of 0.1mSv/yr for the general public, the mining ponds may be considered to be highly radioactive to children than the adults.

Conclusion

Gross alpha and beta activity concentrations in mining ponds and water of Jos metropolis, Plateau State were measured. The results obtained for gross beta were generally higher than the gross alpha activity concentrations. The estimated committed effective dose to children and adults were also calculated in which some values were above the WHO standard of 0.1mSv/yr for the general public. Hence long term accumulated effects should be guided particularly in the ponds where elevated activity concentrations were obtained during the survey.

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